72-1029

TRANSNUCLEAR
AN AREVA COMPANY

November 10, 2004
ANUH-01-04-09

Mr. Jose Cuadrado
Spent Fuel Project Office, NMSS
U. S. Nuclear Regulatory Commission
11555 Rockville Pike M/S 0-13-D-13
Rockville, MD 20852

Subject: Submittal of Revision 4 of Application for Amendment No.1 of Advanced NUHOMS® Certificate of Compliance (CoC) No. 1029 for Dry Spent Fuel Storage Casks (TAC No. L23606)

Dear Mr. Cuadrado:

Transnuclear Inc. (TN) herewith submits Revision 4 of this application (Changed Pages only) to address staff comments discussed in a telecom on 11/09/04.

Should you or your staff require additional information to support review of this application, please do not hesitate to contact me at 510-744-6053 or Mr. Jayant Bondre at 510-744-6043.

Sincerely,

U. B. Chopra

U. B. Chopra
Licensing Manager

Docket 72-1029

Enclosures:
1. Justification for Revising Technical Specification Tables 2-5 and 2-6.
2. ~~Ten (10) copies of Application for Amendment No. 1 to Advanced NUHOMS® COC 1029 (Replacement Pages Only, Proprietary Version).~~
3. Three (3) copies of Application for Amendment No. 1 to Advanced NUHOMS® COC 1029 (Replacement Pages Only, Non-Proprietary Version).

CC: Mr. Ray Wharton
Spent Fuel Project Office, NMSS
U. S. Nuclear Regulatory Commission
11555 Rockville Pike M/S 0-13-D-13
Rockville, MD 20852

NMSS01

Justification for Deletion of Gamma and Neutron sources from the Technical Specification Tables 2-5 and 2-6:

Advanced NUHOMS SAR Section A.5.2 describes the neutron and gamma source specification used in the shielding analysis of the 24PT4 DSC. As discussed in this Section, the design basis source terms used in the evaluation result in dose rates on the surface of the AHSM and TC that are greater than the dose rates due to fuels with burnups, initial enrichment and cooling time combination allowed for storage per the Fuel Qualification Tables. This was evaluated by the staff and found acceptable as documented in the Safety Evaluation Report (SER) Section 5.2.3 for the 24PT4 DSC.

The Fuel Qualification Tables 2-9 through 2-16 are already a part of the Technical Specification for the 24PT4 DSC. All the fuel assemblies that will be loaded in the 24PT4 DSC are required to meet the burnup, initial enrichment and minimum cooling times given in these tables. Therefore, the gamma and neutron source parameters as specified in Technical Specification Tables 2-5 and 2-6 are not necessary and should be deleted.

Encl 3

ATTACHMENT B

SUGGESTED CHANGES TO THE TECHNICAL SPECIFICATIONS FOR THE ADVANCED NUHOMS® SYSTEM

(ADDITION OF 24PT4-DSC TO THE ADVANCED NUHOMS® SYSTEM)

OPERATING CONTROLS AND LIMITS

2.0 Functional and Operating Limits

2.1 Fuel To Be Stored In The 24PT1-DSC

The spent nuclear fuel to be stored in each 24PT1-DSC/AHSM at the ISFSI shall meet the following requirements:

a. Fuel shall be INTACT FUEL ASSEMBLIES or DAMAGED FUEL ASSEMBLIES. DAMAGED FUEL ASSEMBLIES shall be placed in screened confinement cans (*FAILED FUEL CANS*) inside the 24PT1-DSC guidesleeves. DAMAGED FUEL ASSEMBLIES shall be stored in outermost guidesleeves located at the 45, 135, 225 and 315 degree azimuth locations.

b. Fuel types shall be limited to the following:

   UO_2 Westinghouse 14x14 (WE 14x14) Assemblies (with or without IFBA fuel rods), as specified in Table 2-1.

   WE 14x14 Mixed Oxide (MOX) Assemblies, as specified in Table 2-1

   Fuel burnup and cooling time is to be consistent with the limitations specified in Table 2-4 for UO2 fuel.

   Control Components stored integral to WE 14x14 Assemblies in a 24PT1-DSC, shall be limited to Rod Cluster Control Assemblies (RCCAs), Thimble Plug Assemblies (TPAs), and Neutron Source Assemblies (NSAs). Location of control components within a 24PT1-DSC shall be selected based on criteria which does not change the radial center of gravity by more than 0.1 inches.

c. The maximum heat load for a single FUEL ASSEMBLY, including control components, is 0.583 kW for SC FUEL ASSEMBLIES and 0.294 kW for MOX FUEL ASSEMBLIES. The maximum heat load per 24PT1-DSC, including any integral Control Components, shall not exceed 14 kW when loaded with all SC FUEL ASSEMBLIES and 13.706 kW when loaded with MOX FUEL ASSEMBLIES.

d. Fuel can be stored in the 24PT1-DSC in any of the following configurations:

   1) A maximum of 24 INTACT WE 14x14 MOX or SC FUEL ASSEMBLIES; or
   2) Up to four WE 14x14 SC DAMAGED FUEL ASSEMBLIES, with the balance INTACT WE 14x14 SC FUEL ASSEMBLIES; or
   3) One MOX DAMAGED FUEL ASSEMBLY with the balance INTACT WE 14x14 SC FUEL ASSEMBLIES.

   A 24PT1-DSC containing less than 24 FUEL ASSEMBLIES may contain dummy FUEL ASSEMBLIES in FUEL ASSEMBLY slots. The dummy FUEL ASSEMBLIES are unirradiated, stainless steel encased structures that

Table 2-5 **PWR Fuel Specification of Intact Fuel to be stored in NUHOMS® 24PT4-DSC**

Fuel Design:	INTACT CE 16x16 PWR FUEL ASSEMBLY or equivalent reload fuel that is enveloped by the FUEL ASSEMBLY design characteristics as listed in Table 2-7 and the following requirements:
Fuel Damage:	Fuel with known or suspected cladding damage in excess of pinhole leaks or hairline cracks or an assembly with partial and/or missing rods is not authorized to be stored as "INTACT PWR FUEL."
Physical Parameters(1)	
Unirradiated Length (in)	176.8
Cross Section (in)	8.290
Assembly Weight (lbs)	1500(2)
No. of Assemblies per DSC	≤ 24 intact assemblies
Max. U Content (kg)	455.5
Fuel Cladding	Zircaloy-4 or ZIRLO™
RECONSTITUTED FUEL ASSEMBLIES	DAMAGED FUEL Rods replaced by either stainless rods (up to 8 rods per assembly) or Zircaloy clad uranium rods (any number of rods per assembly)
Nuclear and Radiological Parameters	
Maximum Initial ^{235}U Enrichment (wt %)	Per Table 2-8 and Figure 2-4
Fuel Burnup and Cooling Time	Per Table 2-9, Table 2-10, Table 2-11 and Table 2-12 For RECONSTITUTED FUEL with stainless steel replacement rods per Table 2-13, Table 2-14, Table 2-15 and Table 2-16
Decay Heat	Per Figure 2-1, Figure 2-2 or Figure 2-3.

Notes:

(1) Nominal values shown unless stated otherwise.

(2) Does not include weight of Poison Rodlets (25 lbs each).

Table 2-6 PWR Fuel Specifications of DAMAGED FUEL to be Stored in NUHOMS® 24PT4-DSC

Fuel Design:	DAMAGED CE 16x16 PWR FUEL ASSEMBLY or equivalent reload fuel that is enveloped by the FUEL ASSEMBLY design characteristics as listed in Table 2-7 and the following requirements:
Fuel Damage:	DAMAGED FUEL may include assemblies with known or suspected cladding defects greater than pinhole leaks or hairline cracks or an assembly with partial and/or missing rods (i.e., extra water holes). DAMAGED FUEL ASSEMBLIES shall be encapsulated in individual FAILED FUEL CANS and placed in Zones A and/or B as shown in Figure 2-4. FUEL DEBRIS and DAMAGED FUEL Rods that have been removed from a DAMAGED FUEL ASSEMBLY and placed in a ROD STORAGE BASKET are also considered as DAMAGED FUEL. Loose FUEL DEBRIS not contained in a ROD STORAGE BASKET, may also be placed in a FAILED FUEL CAN for storage, provided the size of the debris is larger than the FAILED FUEL CAN screen mesh opening. FUEL DEBRIS may be associated with any type of UO_2 fuel provided that the maximum uranium content and enrichment limits are met.
Physical Parameters(1)	
Unirradiated Length (in)	176.8
Cross Section (in)	8.290
Assembly Weight (lbs)	1500(2)
No. of Assemblies per DSC	≤ 12 DAMAGED ASSEMBLIES, balance INTACT
Max. U Content (kg)	455.5
Fuel Cladding	Zircaloy-4 or ZIRLO™
RECONSTITUTED FUEL ASSEMBLIES	DAMAGED FUEL Rods replaced by either stainless rods (up to 8 rods per assembly) or Zircaloy clad uranium rods (any number of rods per assembly)
Nuclear and Radiological Parameters	
Initial ^{235}U Enrichment (wt %)	Per Table 2-8 and Figure 2-4.
Fuel Burnup and Cooling Time	Per Table 2-9, Table 2-10, Table 2-11 and Table 2-12 For RECONSTITUTED FUEL with stainless steel replacement rods per Table 2-13, Table 2-14, Table 2-15 and Table 2-16
Decay Heat	Per Figure 2-1, or Figure 2-2 or Figure 2-3

Notes:

(1) Nominal values shown unless stated otherwise.

(2) Does not include weight of Poison Rodlets (25 lbs each).

Table A.2.1-1
PWR Fuel Specification of Intact Fuel to be Stored in NUHOMS® 24PT4-DSC

Fuel Design:	Intact CE 16x16 PWR fuel assembly or equivalent reload fuel that is enveloped by the fuel assembly design characteristics as listed in Table A.2.1-3 and the following requirements:
Fuel Damage:	Fuel with known or suspected cladding damage in excess of pinhole leaks or hairline cracks or an assembly with partial and/or missing rods is not authorized to be stored as "intact PWR Fuel."
Physical Parameters[1]	
Unirradiated Length (in)	176.8
Cross Section (in)	8.290
Assembly Weight (lbs)	1500[2]
Max. U Content (kg)	455.5
No. of Assemblies per DSC	≤ 24 intact assemblies
Fuel Cladding	Zircaloy-4 or ZIRLO™
Reconstituted Fuel Assemblies	Damaged fuel rods replaced by either stainless rods (up to 8 rods per assembly) or Zircaloy clad uranium rods (any number of rods per assembly).
Nuclear and Radiological Parameters	
Maximum Initial ^{235}U Enrichment (wt %)	Per Table A.2.1-4 and Figure A.2.1-4
Fuel Assembly Average Burnup and Cooling Time	Per Tables A.2.1-5, A.2.1-6, A.2.1-7, A.2.1-8. For Reconstituted Fuel with stainless steel replacement rods, per Tables A.2.1-9, A.2.1-10, A.2.1-11, A.2.1-12.
Decay Heat	Per Figure A.2.1-1, A.2.1-2 or A.2.1-3.

Notes:

(1) Nominal values shown unless stated otherwise.

(2) Does not include weight of Poison Rodlets (25 lbs each).

Table A.2.1-2
PWR Fuel Specifications of Damaged Fuel to be Stored in NUHOMS® 24PT4-DSC

Fuel Design:	Damaged CE 16x16 PWR fuel assembly or equivalent reload fuel that is enveloped by the fuel assembly design characteristics as listed in Table A.2.1-3 and the following requirements:
Fuel Damage:	Damaged fuel may include assemblies with known or suspected cladding defects greater than pinhole leaks or hairline cracks or an assembly with partial and/or missing rods (i.e., extra water holes). Damaged fuel assemblies shall be encapsulated in individual Failed Fuel Cans and placed in Zones A and/or B as shown in Figure A.2.1-4. Fuel debris and damaged fuel rods that have been removed from a damaged fuel assembly and placed in a Rod Storage Basket are also considered as damaged fuel. Loose fuel debris, not contained in a Rod Storage Basket may also be placed in a Failed Fuel Can for storage, provided the size of the debris is larger than the Failed Fuel Can screen mesh opening. Fuel debris may be associated with any type of UO_2 fuel provided that the maximum uranium content and initial enrichment limits are met.

Physical Parameters[1]	
Unirradiated Length (in)	176.8
Cross Section (in)	8.290
Assembly Weight (lbs)	1500[2]
Max. U Content (kg)	455.5
No. of Assemblies per DSC	≤ 12 damaged assemblies, balance intact.
Fuel Cladding	Zircaloy-4 or ZIRLO™
Reconstituted Fuel Assemblies	Damaged fuel rods replaced by either stainless rods (up to 8 rods per assembly) or Zircaloy clad uranium rods (any number of rods per assembly).
Nuclear and Radiological Parameters	
Initial ^{235}U Enrichment (wt %)	Per Table A.2.1-4 and Figure A.2.1-4.
Fuel Assembly Average Burnup and Cooling Time	Per Tables A.2.1-5, A.2.1-6, A.2.1-7, A.2.1-8. For Reconstituted Fuel with stainless steel replacement rods, per Tables A.2.1-9, A.2.1-10, A.2.1-11, A.2.1-12.
Decay Heat	Per Figure A.2.1-1, or A.2.1-2 or A.2.1-3.

Notes:

(1) Nominal values shown unless stated otherwise.

(2) Does not include weight of Poison Rodlets (25 lbs each).

All pressure boundary components are constructed of Type 316 stainless steel. Non-pressure boundary components welded to the pressure boundary components are also constructed of Type 316 stainless steel. The lead shield plugs are made of ASTM B29 lead.

The 24PT4-DSC cylindrical shell and bottom end assembly including the bottom shield plug assembly, outer bottom cover plate, and the grapple ring assembly, and the internal basket assembly, are shop-fabricated components. The top shield plug assembly and the outer top cover plate are shop-fabricated and tested for fit-up but installed at the plant after the spent fuel assemblies have been loaded into the 24PT4-DSC internal basket.

The 24PT4-DSC shell assembly is designed, fabricated, examined and tested in accordance with the requirements of Subsection NB of the ASME Code including Code Case N-595-1 [A3.2] for closure welds. The circumferential and longitudinal shell plate weld seams are full penetration butt welds. The butt weld joints are fully radiographed and inspected according to the requirements of NB-5000 of the ASME Boiler and Pressure Vessel Code.

The 24PT4-DSC top closure is compliant with Code Case N-595-1 and NRC's ISG-15 [A3.3]. The inner top cover plate of the top shield plug assembly is welded to the 24PT4-DSC shell to complete the pressure boundary as shown in Figure A.3.1-2. The outer top cover plate is sealed by a separate, redundant closure weld. All closure welds are multiple-layer welds and are examined by multi-level liquid penetrant methods to effectively eliminate leaks through welds.

The top end assembly of the 24PT4-DSC design incorporates a vent/siphon block, with two small-diameter penetrations into the 24PT4-DSC cavity for draining and filling operations. The vent port is terminated at the bottom of the shield plug assembly. The other port is attached to a siphon tube, which continues to the bottom of the 24PT4-DSC cavity. The ports include dog-leg type offsets to minimize radiation streaming. The vent and siphon ports terminate in normally closed quick-connect fittings.

During fabrication, leak tests of the 24PT4-DSC shell assembly are performed in accordance with ANSI N14.5-1997 [A3.4] *to demonstrate that the shell assembly, including the bottom closure, is leak tight.*

The stringent design and fabrication requirements described above ensure that the pressure retaining confinement function is maintained for the design life of the 24PT4-DSC. Pressure monitoring instrumentation is not used since penetration of the pressure boundary would be required. The penetration itself would then become a potential leakage path and, by its presence, compromise the leaktightness of the 24PT4-DSC design.

Transfer of the 24PT4-DSC from the TC into the AHSM is performed using a hydraulic ram that applies a load to the outer bottom cover plate, at the center of the 24PT4-DSC. During insertion of the 24PT4-DSC into the AHSM, the load is shared by the outer bottom cover plate and the inner bottom cover plate.

Frictional loads during 24PT4-DSC transfer are reduced by application of a dry film lubricant to the hardened nitronic surface of the AHSM support rails and the TC. The lubricant chosen for this application is a tightly adhering inorganic lubricant with an inorganic binder. The dry film lubricant provides a thin, clean, dry, layer of lubricating solids that is intended to reduce wear, and prevent galling in metals. It is applied as a thin sprayed coating, similar to paint, using a carefully controlled process. The lubricant is not affected by water and is designed to be highly resistant to aggressive chemicals. This product is designed for radiation service and has a low coefficient of sliding friction for stainless steel.

The internal basket assembly, shown in Figure A.3.1-3, provides structural support for and geometric separation of the SFAs. The basket assembly consists of 24 stainless steel guidesleeve assemblies, 28 carbon steel spacer discs, and four-support rod/spacer sleeve assemblies. The support rods and spacer sleeves are fabricated of precipitation hardened martensitic stainless steel.

The spacer disc details, shown in Figure A.3.1-4, identify the twenty-four cutouts for the SFAs and the four support rods. The spacer discs maintain cross-sectional spacing and support for the fuel assemblies and the guidesleeves when the 24PT4-DSC is in the horizontal position. When the 24PT4-DSC is in the vertical position, the spacer discs are held in place by the support rods and spacer sleeves; the rod assemblies maintain longitudinal separation between discs during all normal operating and postulated accident conditions. Fuel weight is transferred to the top or bottom cover plates by direct bearing.

Damaged fuel assemblies are stored in Failed Fuel Cans. The Failed Fuel Can is provided with a welded bottom closure and a removable top closure. Slots are provided in the Failed Fuel Can to allow independent removal of the can and the enclosed fuel assembly. Failed Fuel Cans are provided with screens at the bottom and top to contain fuel debris and allow fill/drainage of water from the Failed Fuel Can.

A.3.1.1.2 General Description of the AHSM

No change.

A.3.1.2 24PT4-DSC and AHSM Design Criteria

No change.

A.3.1.2.1 24PT4-DSC Design Criteria

A.3.1.2.1.1 Stress Criteria

No change.

The 24PT4-DSC is designed utilizing linear elastic and non-linear elastic-plastic analytical methods. ASME Code Service Level A and B allowables are used for normal and off-normal operating conditions, respectively. Service Level C and D allowables are used for accident conditions.

The 24PT4-DSC shell is designed by analysis to meet the criteria of the ASME Boiler and Pressure Vessel Code Section III, Division 1, Subsection NB, 1992 Edition through 1994 Addenda, supplemented by Code Case N-595-1 [A3.2], ISG-15 [A3.3] and ISG-18 [A3.18]. Stress criteria for pressure boundary components are summarized in Table 3.1-2. Stress criteria for (partial penetration) pressure boundary top closure welds are summarized in Table A.3.1-2.

Table A.3.6-1
24PT4-DSC On-Site Load Combinations

	Horizontal DW		Vertical DW		Internal Pressure[8]	External Pressure	Thermal Condition	Lifting Loads	Other Loads	Service Level
	DSC	Fuel	DSC	Fuel						
NON-OPERATIONAL LOAD CASES										
NO-1 Fabrication Leak Testing	—	—	—	—	—	*14.7 psi*	*70°F*	—	*310 kip axial*	*Test*
NO-2 Fabrication *Leak/Pressure* Testing	—	—	—	—	24 psi	—	70°F	—	310 kip axial	Test
NO-3 DSC Uprighting	X	—	—	—	—	—	70°F	X	—	B
NO-4 DSC Vertical Lift	—	—	X	—	—	—	70°F	X	—	B
FUEL LOADING LOAD CASES										
FL-1 DSC/Cask Filling	—	—	Cask	—	—	Hydrostatic	120°F Cask	—	—	A
FL-2 DSC/Cask Filling	—	—	Cask	—	Hydrostatic	Hydrostatic	120°F Cask	—	—	A
FL-3 DSC/Cask Transfer	—	—	Cask	—	Hydrostatic	Hydrostatic	120°F Cask	—	—	A
FL-4 Fuel Loading	—	—	Cask	X	Hydrostatic	Hydrostatic	120°F Cask	—	—	A
FL-5 Transfer to Decon	—	—	Cask	X	Hydrostatic	Hydrostatic	120°F Cask	—	—	A
FL-6 Inner Cover Plate Welding	—	—	Cask	X	Hydrostatic	Hydrostatic	120°F Cask	—	—	A
FL-7 Fuel Deck Seismic Loading	—	—	Cask	X	Hydrostatic	Hydrostatic	120°F Cask	—	Note 9	C
DRAINING & DRYING LOAD CASES										
DD-1 DSC Blowdown	—	—	Cask	X	Hydrostatic+20 psi	Hydrostatic	120°F Cask	—	—	B
DD-2 Vacuum Drying	—	—	Cask	X	0 psia	Hydrostatic+14.7 psia	120°F Cask	—	—	B
DD-3 Helium Backfill	—	—	Cask	X	12 psig	Hydrostatic	120°F Cask	—	—	B
DD-4 Final Helium Backfill	—	—	Cask	X	7.0 psig	Hydrostatic	120°F Cask	—	—	B
DD-5 Outer Cover Plate Welding	—	—	Cask	X	7.0 psig	Hydrostatic	120°F Cask[13]	—	—	B
TRANSFER TRAILER LOADING										
TL-1 Vertical Transfer to Trailer			Cask	X	≤ 20.0 psig	—	0°F Cask	—	—	A
TL-2 Vertical Transfer to Trailer			Cask	X	≤ 20.0 psig	—	120°F Cask	—	—	A
TL-3 Laydown	Cask	X			≤ 20.0 psig	—	0°F Cask[13]	—	—	A
TL-4 Laydown	Cask	X			≤ 20.0 psig	—	120°F Cask	—	—	A

	Horizontal DW		Vertical DW		Internal Pressure[8]	External Pressure	Thermal Condition	Handling Loads	Other Loads	Service Level
	DSC	Fuel	DSC	Fuel						
TRANSFER TO / FROM ISFSI										
TR-1 Axial Load - Cold	Cask	X	—	—	≤ 20.0 psig	—	0°F Cask[13]	1g Axial	—	A
TR-2 Transverse Load - Cold	Cask	X	—	—	≤ 20.0 psig	—	0°F Cask[13]	1g Transverse	—	A
TR-3 Vertical Load - Cold	Cask	X	—	—	≤ 20.0 psig	—	0°F Cask[13]	1g Vertical	—	A
TR-4 Oblique Load - Cold	Cask	X	—	—	≤ 20.0 psig	—	0°F Cask[13]	½ g Axial+½ g Trans+½ g Vert	—	A
TR-5 Axial Load - Hot	Cask	X	—	—	≤ 20.0 psig	—	104°F Cask	1g Axial	—	A
TR-6 Transverse Load - Hot	Cask	X	—	—	≤ 20.0 psig	—	104°F Cask	1g Transverse	—	A
TR-7 Vertical Load - Hot	Cask	X	—	—	≤ 20.0 psig	—	104°F Cask	1g Vertical	—	A
TR-8 Oblique Load - Hot	Cask	X	—	—	≤ 20.0 psig	—	104°F Cask	½ g Axial+½ g Trans+½ g Vert	—	A
TR-12 Top End Drop	This drop is not credible for the horizontal NUHOMS® system.									
TR-9 Bottom End Drop	This drop is not credible for the horizontal NUHOMS® system.									
TR-10 Side Drop	Note 1		—	—	≤ 26.0 psig	—	104°F Cask[2]		75G drop[1]	D
TR-11 Corner Drop	Note 1		—	—	≤ 26.0 psig	—	104°F Cask[2]		25g Drop[1][5]	D

See the following page for Notes.

A.3.7 References

[A3.1] Nuclear Regulatory Commission, Safety Evaluation Report of Safety Analysis Report for the Standardized NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel, December 1994, USNRC Docket Number 72-1004.

[A3.2] American Society of Mechanical Engineers, Boiler & Pressure Vessel Code, Section II, Section III, 1992 Edition with Addenda through 1994 with Code Cases N-595-1 and N-499-1.

[A3.3] NRC Spent Fuel Project Office, Interim Staff Guidance, ISG-15, Materials Evaluation.

[A3.4] *ANSI N14.5-1997, "American National Standard for Radioactive Materials, Leakage Tests on Packages for Shipment", February 1998.*

[A3.5] Holman, W.R., Langland, R. T., "Recommendations for Protecting Against Failure by Brittle Fracture in Ferritic Steel Shipping Containers up to Four Inches Thick," NUREG/CR-1815, September 1981.

[A3.6] Teitz, T. E., "Determination of the Mechanical Properties of a High Purity Lead and a 0.058% Copper-lead Alloy," WADC Technical Report 57-695, ASTIA Document No. 151165, Stanford Research Institute, Menlo Park, CA, April 1958.

[A3.7] American Society of Mechanical Engineers, *Boiler & Pressure Vessel Code,* Section III, Division 1, Code Case N-499-1, *Use of SA-533 Grade B, Class 1 Plate and SA-508 Class 3 Forgings and their Weldments for Limited Elevated Temperature Service,* Section III, Division 1; Approval Date: December 12, 1994, Reaffirmed October 2, 2000, Expires October 2, 2003.

[A3.8] Transnuclear, Inc., Final Safety Analysis Report for the Standardized NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel, Revision 8, June 2004, USNRC Docket Number 72-1004.

[A3.9] Swanson Analysis Systems Inc., ANSYS Engineering Analysis System User's Manual, Versions 5.3, 5.6.2, and 8.1, Swanson Analysis Systems, Inc., Pittsburgh, PA.

[A3.10] Levy, Chin, Simonen, Beyer, Gilbert and Johnson, Recommended Temperature Limits for Dry Storage of Spent Light Water Reactor Zircalloy-Clad Fuel Rods in Inert Gas, May 1987, Pacific Northwest Laboratory, PNL Document PNL-6189.

[A3.11] Johnson, A. B. and E. R. Gilbert, Technical Basis for Storage of Zircalloy-Clad Spent Fuel in Inert Gas, September 1983, Pacific Northwest Laboratory, PNL Document PNL-4835.

[A3.12] "Consolidated Safety Analysis Report for IF-300 Shipping Cask", NEDO-10084, Vectra Technologies, Inc., Revision 4, March, 1995.

Table A.4.4-10
24PT4-DSC Cavity Pressure Analysis Summary

Condition	$T_{He,ave}$ (°F)	n_{He} (g-mole)	n_{fill} (g-mole)	n_{fiss} (g-mole)	P (psig)	Thermal Criteria (psig)[1]	Pressures Used in Stress Analysis (psig) Table A.3.1-*4*
Normal	546.7	294.8	1.65	3.51	17.5	20	20
Off-Normal	550.1	294.8	16.5	35.13	22.3	23.4	26
Accident	713	294.8	164.59	351.3	80.7	*90*	*100*

(1) These criteria are used for thermal analyses only. The off-normal and accident thermal criteria have additional margin to account for the effect of the fission gases in the 24PT4-DSC cavity on the thermal results.

<u>Nomenclature used in table</u>

$T_{He,ave}$	Average helium temperature
n_{He}	Number of moles of helium backfill
n_{fill}	Number of moles of fuel rod fill gas released to 24PT4-DSC cavity
n_{fiss}	Number of moles of fission gas released to 24PT4-DSC cavity
P	24PT4-DSC cavity pressure

3. Although steam may be produced through boiling of the cavity water, its presence in the weld joint area during the shield plug assembly welding operations will be essentially blocked at the interface between the shield plug and the support ring. What little steam may be present is displaced by the argon shielding gas used in the GTAW process. This shielding gas is heavier than steam and is delivered at a sufficiently high rate (usually 30 – 50 ft^3/hr) to assure that the steam is excluded from the weld joint. Finally, if moisture somehow did enter the weld area, the resulting weld bead porosity would be readily detectable by the visual inspection of each pass performed by the welding operator and the dye penetrant (PT) examination performed on the surface of the root pass.

Therefore, the only potential concern associated with steam generation is shielding. An unexpectedly high loss of water within the 24PT4-DSC cavity during these loading operations could result in increased occupational exposure. The following analysis is presented to identify to the licensees the time for the water in the 24PT4-DSC cavity to boil so that corrective action can be planned and implemented as necessary to address ALARA concerns.

The *HEATING7* model, *as described in Section 4.7.3,* conservatively does not credit any heat transfer in the axial direction. Homogenized effective thermal properties of the 24PT4-DSC cavity are calculated based on the weight, volume and material of the components. Radiation heat transfer within the 24PT4-DSC cavity is neglected. All temperatures in the 24PT4-DSC are initially assumed to be at the maximum spent fuel pool temperature. The exterior of the cask is assumed to radiate and convect heat to the prevailing ambient conditions of the fuel building. The analyses are performed for a building temperature of 120°F and a fuel pool temperature of 140°F. The results are tabulated in Table A.4.7-3 and shown in Figure A.4.7-9 for canister decay heat loads ranging from 12 to 24 kW. Note that the starting time to reach boiling is based on a very conservative assumption that following the placement of the first fuel assembly in the DSC, the DSC heat load is 24 kW (i.e., all 24 fuel assemblies loaded simultaneously in the DSC).

A.4.7.4 Pressure During Loading of Cask

The maximum pressure during cask blowdown is 20 psig. This is discussed in Chapter A.3.

A.7 CONFINEMENT

A.7.1 Confinement Boundary

Sections of this Chapter have been identified as "No change" due to the addition of 24PT4-DSC to the Advanced NUHOMS® system. For these sections, the description or analysis presented in the corresponding sections of the FSAR for the Advanced NUHOMS® system with 24PT1-DSC is also applicable to the system with 24PT4-DSC.

The 24PT4-DSC is a high integrity austenitic stainless steel welded vessel that provides confinement of radioactive materials, encapsulates the fuel in a helium atmosphere and provides biological shielding during 24PT4-DSC closure, transfer and storage operations. The 24PT4-DSC is designed to maintain confinement of radioactive material within the limits of 10CFR 72.104(a), 10CFR 72.106(b) and 10CFR 20 under normal, off-normal, and credible accident conditions. Chapters A.3, A.4 and A.11 conclude that the design including the helium atmosphere within the 24PT4-DSC will adequately protect the spent fuel cladding against degradation that might otherwise lead to gross ruptures during storage.

The cylindrical shell, the inner cover plates of the top and bottom shield plug assemblies, the vent and siphon block, the vent and siphon cover plates, and associated welds form the pressure retaining confinement boundary for the spent fuel (see Figure A.7.1-1). The outer top cover plate and associated welds function as a redundant welded barrier for confining radioactive material within the 24PT4-DSC. The dimensions and material descriptions for the confinement boundary assemblies and the redundant welded barriers are discussed in Chapter A.1. The components important to safety are identified in Chapter A.2.

A.7.1.1 Confinement Vessel

The cylindrical shell to inner bottom cover plate weld is made during fabrication of the 24PT4-DSC and is fully compliant to ASME Section III [A7.1], Subsection NB. The vent and siphon block to shell weld is also made during fabrication. The inner top cover plate to shell closure weld is made after fuel loading. These top closure welds are fully compliant to ASME Code Case N-595-1. Both top plug penetrations (siphon and vent ports) are welded after drying operations are complete.

Stringent design and fabrication requirements ensure that the confinement function of the 24PT4-DSC is maintained. The shell and inner bottom cover plate are pressure tested in accordance with the ASME Code, Section III, Subarticle NB-6300. This pressure test is performed after installation of the inner bottom cover plate *and may be performed concurrently with the leak test, provided the requirements of NB-6300 are met.*

Following the pressure test, a leak test of the shell assembly, including the inner bottom cover plate, is performed in accordance with ANSI N14.5 [A7.2]. *These tests are typically performed at the fabricator. The acceptance criteria for the test is "leaktight" as defined in ANSI N14.5-1997* [A7.2].

The process involved in leak testing the 24PT4-DSC shell assembly involves temporarily sealing the shell top end. The gas filled envelope and evacuated envelope testing methodologies have *the required test sensitivity to demonstrate leaktight construction and are used for leak testing. A helium mass spectrometer is used to detect any leakage as defined in ANSI N14.5* [A7.2].

During final drying and sealing operations of the 24PT4-DSC, the top closure confinement welds are completed to confine radioactive materials within the cavity. The inner top cover plate is welded to the shell using automated welding equipment. Once the 24PT4-DSC has been vacuum dried, backfilled with helium and vent and siphon port penetrations welded, the outer top cover plate is lowered onto the 24PT4-DSC. The outer top cover plate is welded in place using automated welding equipment. The outer top cover plate and associated closure weld to the shell acts as a redundant barrier for confining radioactive material within the 24PT4-DSC throughout its service life.

A.7.1.2 Confinement Penetrations

All penetrations in the 24PT4-DSC confinement boundary are welded closed.

A.7.1.3 Seals and Welds

The austenitic stainless steel welds made during fabrication of the 24PT4-DSC that affect the confinement boundary include the weld applied to the inner bottom cover plate and the circumferential and longitudinal seam welds applied to the shell. These welds are examined (radiographic or ultrasonic and liquid penetrant) according to the requirements of Subsection NB of the ASME Code. The vent and siphon block-to-shell weld is also made during fabrication and is liquid penetrant examined in accordance with Subsection NB of the ASME Code.

The welds of the vent and siphon port covers, and the inner top cover plate to shell, completed during closure operations, and the vent and siphon block to shell weld define the confinement boundary at the top end of the 24PT4-DSC. These welds are applied using multiple-layer techniques with multi-level PT in accordance with Subsection NB of the ASME Code and Code Case N-595-1. This effectively eliminates any pinhole leaks which might occur in a single-pass weld, since the chance of pinholes being in alignment on successive weld passes is negligibly small. Figure A.7.1-1 provides a graphic representation of the confinement boundary welds.

A.7.1.4 Closure

The 24PT4-DSC is entirely closed by welding and thus, no closure devices are utilized for confinement.

A.7.1.5 Leak Testing Requirements

Per ISG-18 [A7.5] leak testing during closure operations is not required if the following criteria are met:

1. The canister is constructed from austenitic stainless steel. (See SAR drawing in Section A.1.5.2)

2. The canister closure welds meet the guidance of ISG-15 [A7.4] (or approved alternative), Section X.5.2.3 "Weld Design and Specifications.".

   - Multi-pass weld (See SAR drawing in Section A.1.5.2)
   - Stress reduction factor of 0.8 (Table A.3.1-2 specifies a conservative factor of 0.7 to all allowables.)

3. The canister maintains its confinement integrity during normal conditions, anticipated occurrences, credible accidents and natural phenomena, as required in 10CFR Part 72. (Chapters A.3 and A.11 evaluate various operating conditions and accident scenarios demonstrating confinement integrity).

4. Records documenting the fabrication and closure welding of canisters shall comply with the provisions of 10CFR Part 72.174, "Qualify Assurance Records" and ISG-15. Records storage should comply with ANSI N45.2.9, "Requirements for Collection, Storage, and Maintenance of Quality Assurance Records for Nuclear Power Plants." (Chapter A.13/13 discusses the QA program under which DSC fabrication is performed. This QA program complies with 10CFR 72 QA requirements.)

5. Activities related to inspection, evaluation, documentation of fabrication, and closure welding of canisters shall be performed in accordance with a NRC-approved quality assurance program as required in 10CFR Part 72, Subpart G, "Quality Assurance." (Chapter A.13/13 discusses the QA program under which DSC fabrication is performed. This QA program complies with 10CFR 72 QA requirements.)

The above criteria are met by the 24PT4-DSC design and fabrication, therefore no leak testing is required *during closure operation.* However, leak testing per ASME B&PV Code, Code Case N-595-1 will be performed.

A.7.4 Supplemental Data

A.7.4.1 Confinement Monitoring Capability

No change.

A.7.4.2 References

[A7.1] American Society of Mechanical Engineers, Boiler & Pressure Vessel Code, Section III, 1992 Edition with Addenda through 1994, including exceptions allowed by Code Case N-595-1.

[A7.2] *ANSI N14.5-1997, "American National Standard for Radioactive Materials, Leakage Tests on Packages for Shipment", February 1998.*

[A7.3] NRC Spent Fuel Project Office, Interim Staff Guidance, ISG-5, Revision 1, Confinement Evaluation.

[A7.4] NRC Spent Fuel Project Office, Interim Staff Guidance, ISG-15, Revision 0, Materials Evaluation.

[A7.5] NRC Spent Fuel Project Office, Interim Staff Guidance, ISG-18, The Design Qualification of Final Closure Welds on Austenitic Stainless Steel Canisters as Confinement Boundary for Spent Fuel Storage and Containment Boundary for Spent Fuel Transportation, dated May 2, 2003.

A.9 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

Sections of this Chapter have been identified as "No change" due to the addition of 24PT4-DSC to the Advanced NUHOMS® system. For these sections, the description or analysis presented in the corresponding sections of the FSAR for the Advanced NUHOMS® system with 24PT1-DSC is also applicable to the system with 24PT4-DSC.

A.9.1 Acceptance Criteria

No change.

A.9.1.1 Visual Inspection

No change.

A.9.1.2 Structural

No change.

A.9.1.3 *Leak Tests and* Hydrostatic Pressure Tests

24PT4-DSC leakage test is performed on the confinement system at the fabricator's facility. This test is usually performed using the helium mass spectrometer method. Alternative methods are acceptable, provided that the required sensitivity is achieved. Personnel performing the leakage tests, both at the fabricator and the loading site, are qualified in accordance with SNT-TC-1A [A9.1].

The 24PT4-DSC shell (longitudinal and circumferential welds and inner bottom cover assembly) is pressure tested in accordance with Subsection NB-6000 of the ASME Code.

A.9.1.4 Components

No change.

A.9.1.4.1 Valves, Rupture Discs, and Fluid Transport Devices

No change.

A.9.1.4.2 Gaskets

No change.

A.9.4 Supplemental Information

A.9.4.1 References

[A9.1] SNT-TC-1A, "American Society for Nondestructive Testing, Personnel Qualification and Certification in Nondestructive Testing," 1984.

[A9.2] ANSI N14.5-1997, "American National Standard for Radioactive Materials, Leakage Tests on Packages for Shipment", February 1998.

[A9.3] ASTM E94, "Guide for Radiographic Testing", 1993.

[A9.4] ASTM E142, "Method for Controlling Quality of Radiographic Testing", 1992.

[A9.5] ASTM E545, "Method for Determining Image Quality in Direct Thermal Neutron Radiographic Examination", 1991.

[A9.6] Hahn, G. J., Statistical Intervals for a Normal Population, Part I," Journal of Quality Technology, Vol. 2, No. 3, July 1970.

[A9.7] Hahn, G. J., Statistical Intervals for a Normal Population, Part II," Journal of Quality Technology, Vol. 2, No. 4, October 1970.

[A9.8] Owen, D. B., "A Survey of Properties and Applications of the Noncentral t-Distribution," Technometrics, Vol. 10, No. 3, August 1968.

[A9.9] "SCR607, Factors for One-Sided Tolerance Limits and for Variables Sampling Plans," U.S. Department of Energy, Sandia Corporation, March 1963.

[A9.10] The Merck Index, *9th Edition, Merck & Co., 1976.*

[A9.11] *Grant (ed.),* Hackh's Chemical Dictionary, *4th edition, McGraw-Hill, 1969.*

[A9.12] *Lipp, A., "Boron Carbide: Production, Properties, Application," Reprint from* Technische Rundschau, *Nos. 14, 28, 33 (1995) and 7 (1966).*

[A9.13] *Stoto, T. et al., "Swelling and Microcracking of Boron Carbide Subjected to Fast Neutron Irradiations,"* Journal of Applied Physics, *Vol. 68, No.7, October 1, 1990, pp. 3198-3206.*

[A9.14] ASTM C751, "Standard Specification for Nuclear-Grade Boron Carbide Pellets."

[A9.15] ASTM C750, "Standard Specification for Nuclear-Grade Boron Carbide Powder."