TRANSNUCLEAR
AN AREVA COMPANY

August 12, 2010
E-29722

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852

Subject: ANUH-01.0150, Standardized Advanced NUHOMS® Updated Final Safety Analysis Report (UFSAR), Revision 4, and 10 CFR 72.48 Summary Report for the Period 8/16/08 to 8/12/10, Docket 72-1029

Reference: Letter from Robert Grubb (TN) to NRC Document Control Desk, "ANUH-01.0150, Standardized Advanced NUHOMS® Updated Final Safety Analysis Report (UFSAR), Revision 3, and 10 CFR 72.48 Summary Report for the Period 3/21/07 to 8/15/08, Docket 72-1029," August 15, 2008 (E-26868)

Pursuant to 10 CFR 72.248, Transnuclear, Inc., (TN) has updated ANUH-01.0150, the Standardized Advanced NUHOMS® UFSAR, last updated by the referenced submittal, and herewith submits UFSAR Revision 4 updates for docketing. This update incorporates changes implemented by TN pursuant to 10 CFR 72.48, through the date of this letter. Changes are identified with revision bars and italicized text.

I certify that this submittal accurately presents changes made since the referenced submittal.

This submittal includes proprietary information which may not be used for any purpose other than to support NRC staff review of the submittal. In accordance with 10 CFR 2.390, I am providing an affidavit (Enclosure 1) specifically requesting that you withhold this proprietary information from public disclosure. This submittal also includes security-related information. Accordingly, two versions of replacement pages, Proprietary and Public, are provided, as Enclosures 2 and 3, respectively.

Also, pursuant to 10 CFR 72.48(d)(2), TN hereby submits the 72.48 summary report for the Standardized Advanced NUHOMS® System for the period 8/16/08 to 8/12/10, as Enclosure 4. The most recent summary report was provided by the referenced submittal.

Should you have any questions regarding this submittal, please do not hesitate to contact Mr. Don Shaw at 410-910-6878 or me at 410-910-6881.

Sincerely,

Jayant Bondre, PhD
Vice President - Engineering

cc: Eric Benner (NRC SFST) (5 copies of this cover letter and Enclosures 1, 2 and 4, provided in a separate mailing)

NMSS01
NMSS

7135 Minstrel Way, Suite 300, Columbia, MD 21045
Phone: 410-910-6900 • Fax: 410-910-6902

Enclosures:

1. Affidavit Pursuant to 10 CFR 2.390
2. Replacement Pages for ANUH-01.0150, Standardized Advanced NUHOMS® UFSAR, Revision 4 (Proprietary Version)
3. Replacement Pages for ANUH-01.0150, Standardized Advanced NUHOMS® UFSAR, Revision 4 (Public Version)
4. 72.48 Summary Report for the Standardized Advanced NUHOMS® System for the Period 8/16/08 to 8/12/10

AFFIDAVIT PURSUANT
TO 10 CFR 2.390

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

I, Jayant Bondre, depose and say that I am a Vice President of Transnuclear, Inc., duly authorized to execute this affidavit, and have reviewed or caused to have reviewed the information which is identified as proprietary and referenced in the paragraph immediately below. I am submitting this affidavit in conformance with the provisions of 10 CFR 2.390 of the Commission's regulations for withholding this information.

The information for which proprietary treatment is sought is contained in Enclosure 2 as listed below:

- Portions of UFSAR Drawing ANUH-01-4001, Revision 5
- Portions of UFSAR Drawing NUH-03-4011, Revision 6
- Portions of UFSAR Drawing NUH-05-4010, Revision 5

These documents have been appropriately designated as proprietary.

I have personal knowledge of the criteria and procedures utilized by Transnuclear, Inc. in designating information as a trade secret, privileged or as confidential commercial or financial information.

Pursuant to the provisions of paragraph (b) (4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure, included in the above referenced document, should be withheld.

1) The information sought to be withheld from public disclosure involves certain safety analysis report drawings related to the design of the Standardized Advanced NUHOMS® System which are owned and have been held in confidence by Transnuclear, Inc.

2) The information is of a type customarily held in confidence by Transnuclear, Inc. and not customarily disclosed to the public. Transnuclear, Inc. has a rational basis for determining the types of information customarily held in confidence by it.

3) Public disclosure of the information is likely to cause substantial harm to the competitive position of Transnuclear, Inc. because the information consists of descriptions of the design and analysis of dry spent fuel storage systems, the application of which provide a competitive economic advantage. The availability of such information to competitors would enable them to modify their product to better compete with Transnuclear, Inc., take marketing or other actions to improve their product's position or impair the position of Transnuclear, Inc.'s product, and avoid developing similar data and analyses in support of their processes, methods or apparatus.

Further the deponent sayeth not.

Jayant Bondre
Vice President, Transnuclear, Inc.

Subscribed and sworn to me before this 12th day of August, 2010.

Notary Public

My Commission Expires

LORA M. DOUGHERTY
COMMISSION EXPIRES
OCT 17 2010
HOWARD COUNTY, MD
NOTARY PUBLIC

Enclosure 3 to TN E-29722

Replacement Pages for ANUH-01.0150, the Standardized Advanced NUHOMS® UFSAR, Revision 4 (Public Version)

ANUH-01.0150
Revision *4*

UPDATED FINAL SAFETY ANALYSIS REPORT

FOR THE

STANDARDIZED ADVANCED NUHOMS®

HORIZONTAL MODULAR STORAGE SYSTEM

FOR IRRADIATED NUCLEAR FUEL

NON-PROPRIETARY REPORT

By

Transnuclear, Inc.
Columbia, MD

August *2010*

REVISION LOG SHEET

FSAR Revision	Date	Record of Changes/FCNs	Changed Pages
0	3/19/03	None	All
1	3/21/05	FCNs 721029-39, 40, 62, 65, 81, 89, 92, 124, 126, 165, 169 & 175	See List of Effective Pages
2	8/17/06	FCNs 721029-182, 185, 103 R-1, 162 R-1, 166, 173 R-1, 176 R-1,177 and 204	See List of Effective Pages
3	8/15/08	FCNs 721029-202, 205, 206, 208, 215, 220, 222 R1, 232, 239, 246, 257, 272	See List of Effective Pages
4	*8/12/10*	*FCNs 721029-275, 280 R-1, 285, 294, 303, 311, 312 R-1, 316*	*See List of Effective Pages*

Revision 2 of this UFSAR incorporates changes implemented due to the approval of CoC 1029 Amendment 1, effective May 16, 2005. It also incorporates modifications implemented per 10 CFR 72.48 from March 21, 2005 through August 15, 2006.

Revision 3 of this UFSAR incorporates modifications implemented per 10 CFR 72.48 from August 16, 2006 through August 15, 2008. This revision also includes a full list of effective pages.

Revision 4 of this UFSAR incorporates modifications implemented per 10 CFR 72.48 from August 16, 2008 through August 12, 2010. This revision also includes a full list of effective pages.

List Of Effective Pages (Non-Proprietary Version)

Page or description	Rev.	Date
Title Page	*4*	August *2010*
i	*4*	08/*10*
ii	3	08/08
iii	2	08/06
iv	2	08/06
v	*4*	08/*10*
vi	2	08/06
vii	2	08/06
viii	2	08/06
ix	2	08/06
x	2	08/06
xi	2	08/06
xii	2	08/06
xiii	2	08/06
xiv	2	08/06
xv	2	08/06
xvi	2	08/06
xvii	2	08/06
xviii	2	08/06
xix	2	08/06
xx	2	08/06
xxi	2	08/06
xxii	2	08/06
xxiii	2	08/06
xxiv	2	08/06
xxv	2	08/06
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LOEP-2	*4*	08/*10*
LOEP-3	*4*	08/*10*
LOEP-4	*4*	08/*10*
LOEP-5	*4*	08/*10*
LOEP-6	*4*	08/*10*
LOEP-7	*4*	08/*10*
LOEP-8	*4*	08/*10*
LOEP-9	*4*	08/*10*
LOEP-10	*4*	08/*10*
LOEP-11	*4*	08/*10*
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1.1-2	0	02/03
1.1-3	0	02/03
1.1-4	0	02/03
1.2-1	0	02/03
1.2-2	0	02/03
1.2-3	0	02/03
1.2-4	0	02/03
1.2-5	0	02/03
1.2-6	0	02/03
1.2-7	2	08/06
1.2-8	0	02/03

Page or description	Rev.	Date
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1.2-10	0	02/03
1.2-11	0	02/03
1.2-12	0	02/03
1.2-13	2	08/06
1.3-1	0	02/03
1.4-1	3	08/08
1.4-2	0	02/03
1.4-3	0	02/03
1.4-4	0	02/03
1.5-1	2	08/06
DWG: (sh. 1 of 3) NUH-05-4010NP	*5*[1]	*08/10/10*
DWG: (sh. 2 of 3) NUH-05-4010NP	*5*[1]	Not shown
DWG: (sh. 3 of 3) NUH-05-4010NP	*5*[1]	Not shown
DWG: (sh. 1 of 5) NUH-03-4011NP	*6*[1]	*08/11/10*
DWG: (sh. 2 of 5) NUH-03-4011NP	*6*[1]	Not shown
DWG: (sh. 3 of 5) NUH-03-4011NP	*6*[1]	Not shown
DWG: (sh. 4 of 5) NUH-03-4011NP	*6*[1]	Not shown
DWG: (sh. 5 of 5) NUH-03-4011NP	*6*[1]	Not shown
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2.1-2	0	02/03
2.1-3	0	02/03
2.1-4	0	02/03
2.1-5	0	02/03
2.1-6	0	02/03
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2.2-2	0	02/03
2.2-3	0	02/03
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2.2-6	0	02/03
2.2-7	0	02/03
2.2-8	0	02/03
2.2-9	0	02/03
2.3-1	0	02/03
2.3-2	2	08/06
2.3-3	0	02/03
2.3-4	0	02/03

[1] Because SAR drawings were revised throughout the licensing period, their revision level may be higher than the overall UFSAR revision level.

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2.3-5	0	02/03
2.3-6	1	03/05
2.4-1	0	02/03
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2.5-3	0	02/03
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2.6-2	2	08/06
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3.1-2	0	02/03
3.1-3	3	08/08
3.1-4	0	02/03
3.1-5	0	02/03
3.1-6	2	08/06
3.1-7	0	02/03
3.1-8	0	02/03
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3.1-12	1	03/05
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3.1-14	0	02/03
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3.1-36	0	02/03
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3.3-3	0	02/03

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3.4-3	2	08/06
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3.6-36	0	02/03
3.6-37	0	02/03
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3.6-42	*4*	08/*10*
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4.4-3	0	02/03

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4.7-8	0	02/03
4.8-1	0	02/03
4.8-2	2	08/06
4.8-3	0	02/03
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6.5-7	0	02/03
6.5-8	0	02/03
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8.1-6	1	03/05
8.1-7	2	08/06
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8.1-9	3	08/08
8.1-10	1	03/05

Page or description	Rev.	Date
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8.2-6	0	02/03
8.2-7	0	02/03
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[1] Because SAR drawings were revised throughout the licensing period, their revision level may be higher than the overall UFSAR revision level.

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INTERPRETATION OF WELD SYMBOLS PER ANSI / AWS 2.4

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SAFETY ANALYSIS REPORT
GENERAL LICENSE
NUHOMS®24PT1-DSC
MAIN ASSEMBLY

DRAWING NO. NUH-05-4010NP | SCALE NONE | SHEET 1 OF 3

DRAWING NO. NUH-05-4010NP SHEET 1 OF 3

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5	REVISED PER FCN 721029-232 AND -257; UPDATED THE TITLE BLOCK	8/6/08
4	INCORPORATED FCN 721029-103	8/11/06

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SAFETY ANALYSIS REPORT
GENERAL LICENSE NUHOMS®
ADVANCED HORIZONTAL STORAGE MODULE
MAIN ASSEMBLY

DRAWING NO.	SCALE	SHEET
NUH-03-4011NP	NONE	1 OF 5

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

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PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

Table 3.6-21
Computed Forces and Capacities of Ties and Keys

#	Tie/Key	Force (kips)	Capacity (kips)	Ratio
1	Tie Beam at Top (3 Beams) (8 #8 Rebars at each Beam)	892.2	1017	0.88
1A	Tie Beam at Top (3 Beams) (5 #9 Rebars at each Beam)	892.2	972	0.92
2	Bottom Tie Rod (4 Rods)	249.2	*303.6*	0.*82*
3	Key between Top Shield Block and Base Unit (Transverse)	491.3	955	0.51
4	Key between Top Shield Block and Base Unit (Longitudinal)	393.1	1,910	0.21
5	Vertical Tie Rods between Top Shield Block and Base Unit (8 Rods)	424	536.8	0.79
6	Vertical Shear Key between Modules	1,368	1605	0.85
7	Horizontal Shear Key between Modules	456.7	462	0.99

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SAFETY ANALYSIS REPORT
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MAIN ASSEMBLY

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Table A.4.4-9
Fuel Assembly Characteristics for Pressure Analysis

Parameter	CE 16x16 Zircaloy Clad
Number of fuel rods	236
Maximum rod fill pressure (psig)	380 ± 15
Maximum rod void volume (in^3)	1.53
Quantity of fission gas per assy (g-moles)	43.57

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	548.0	313.9	1.65	3.51	17.5	20	20
Off-Normal	550.1	313.9	16.5	35.13	22.3	23.4	26
Accident	713	313.9	164.59	351.3	80.8	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.

Nomenclature used in table

$T_{He,ave}$	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 *2 sets of spent fuel building* T_{sfb} *and fuel pool* T_{pool} *temperatures:*
a) $T_{pool} = 100$ *°F,* $T_{sfb} = 90$ *°F;*
b) $T_{pool} = 140$ *°F,* $T_{sfb} = 120$ *°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).

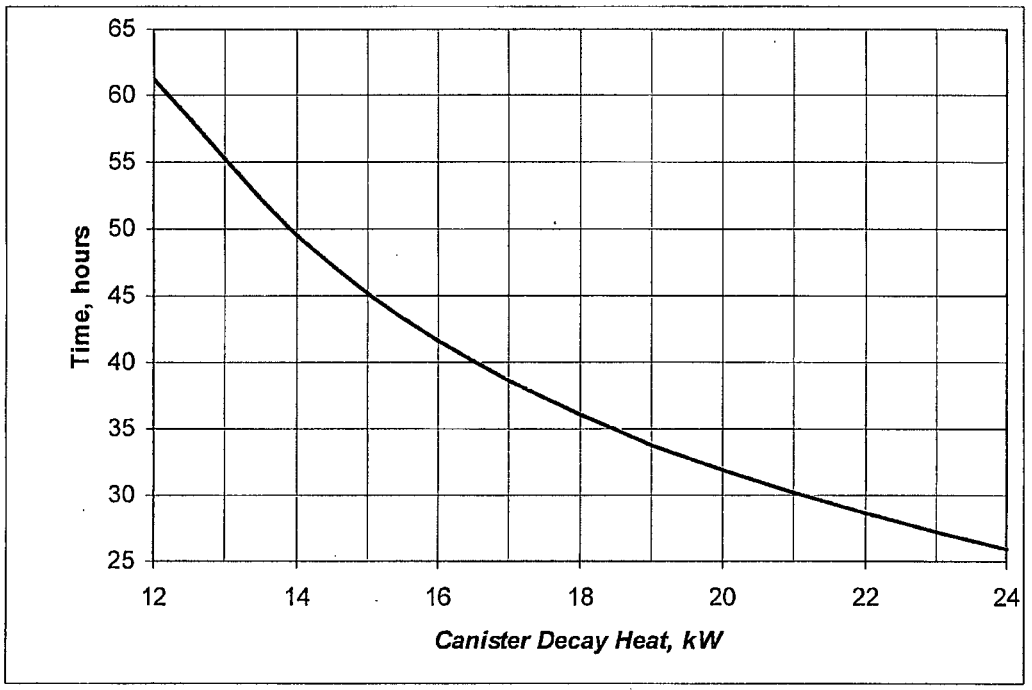
A.4.7.4 Pressure During Loading of Cask

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

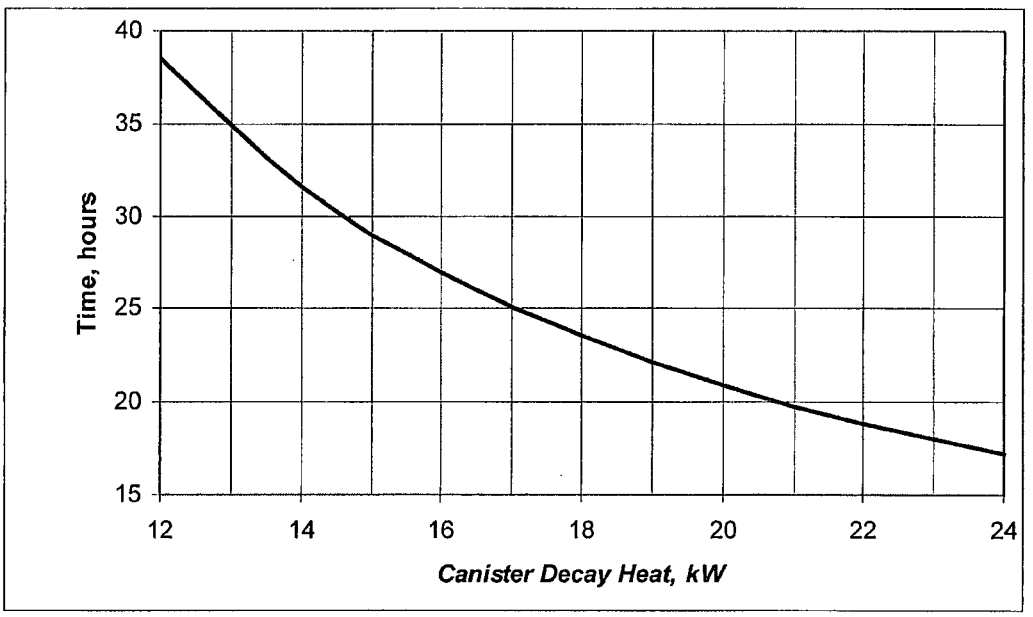
Table A.4.7-3
Summary of Water Heatup Calculation

Decay Heat, kW	Time, hrs* T_{pool}=100 °F, T_{sfb}=90 °F	Time, hrs* T_{pool}=140 °F, T_{sfb}=120 °F
12	61.2	38.6
14	49.5	31.6
16	41.8	27.0
18	36.1	23.6
20	31.9	21.0
22	28.7	18.9
24	26.0	17.2

* Time is to be conservatively measured from the time that the first assembly is loaded in the DSC to account for heatup of water in the DSC while still in the spent fuel pool.



a) T_{pool}=100°F, T_{sfb}=90°F



b) T_{pool}=140°F, T_{sfb}=100°F

Figure A.4.7-9
Time to Reach Boiling Conditions inside 24PT4-DSC Cavity

Table A.4.9-2
Thermal Properties Used in Calculation of Fuel Effective Conductivities

Temperature (°F)	Conductivity (BTU/min -in-°F)
Zircalloy [A4.35]	
200	0.0109
300	0.0115
400	0.0121
500	0.0126
600	0.0131
800	0.0142
Fuel Pellet, Irradiated UO_2 *[A4.39], [A4.40]*[(1)]	
80	*3.178e-3*
260	*2.945e-3*
440	*2.712e-3*
620	*2.472e-3*
800	*2.279e-3*
980	*2.111e-3*
1160	*1.966e-3*
1340	*1.846e-3*
Type 304/304L Stainless Steel (See Section A.4.2)	
Helium (See Section A.4.2)	
Air (See Section A.4.2)	

(1) The Computed Effective Transverse Thermal Conductivity of CE 16x16 Fuel Assembly in Table A.4.9-3 is based on un-irradiated UO_2 conductivity from [A4-35]. The effect of using un-irradiated UO_2 conductivity on the maximum fuel cladding temperature is approximately 1 °F and considered as insignificant.

[A4.31] FLUENT™, Version 6.1, FLUENT, Inc., Lebanon, NH, 2003.

[A4.32] ICEPAK™, Version 4.1, FLUENT, Inc., Lebanon, NH, 2003.

[A4.33] "Characteristics of Spent Fuel, High Level Waste, And Other Radioactive Wastes Which May Require Long-Term Isolation," DOE/RW-0184, Volume 3 of 6, dated December 1987.

[A4.34] "Domestic Light Water Reactor Fuel Design Evolution, Volume III," Nuclear Assurance Corporation, September 1981, DOE/ET/47912-3.

[A4.35] NUREG/CR-0497, A Handbook of Materials Properties for Use in the Analysis of Light Water Reactor Fuel Rod Behavior, MATPRO - Version 11 (Revision 2), EG&G Idaho, Inc., TREE-1280, September 1981.

[A4.36] SAND90-2406, Sanders, T. L., et al., A Method for Determining the Spent-Fuel Contribution to Transport Cask Containment Requirements, TTC-1019, UC-820, November 1992.

[A4.37] "Spent Nuclear Fuel Effective Thermal Conductivity Report", prepared TRW Environmental Safety Systems, Inc. for DOE Civilian Radioactive Waste Management System (CRWMS), Report BBA000000-01717-5705-00010, Rev. 0, July 1996.

[A4.38] "SONGS Unit 2/3 Fuel Assembly Materials and Masses" SCE No. N-1020-162. Transnuclear, Inc. No. SCE-23.0100-11.

[A4.39] K. Minato, et. al., "Thermal Conductivities of Irradiated UO_2 and (U, Gd)O_2", Journal of Nuclear Materials, 300 (2002) 57-64.

[A4.40] Ronchi, et. al., "Effect of Burn-up on the Thermal Conductivity of Uranium Dioxide up to 100,000 MWdt", Journal of Nuclear Materials, 327 (2004) 58-76.

72.48 Summary Report for the Standardized Advanced NUHOMS® System for the Period 8/16/08 to 8/12/10

Enclosure 4 Part 1 - DESIGN CHANGES

Licensing Review (LR) 721029-275, Rev. 0* – *(incorporated into UFSAR Revision 4)

Change Description

The proposed activity involved the addition of ASTM A193, CL2, GR B8M2 and ASTM A453, GR 660 as material specifications for the stud and ASTM A453 Grade 660 or ASTM A194 Grade 8M for the nut of the advance horizontal storage module (AHSM) module-to-module bottom tie assembly.

The AHSM base stud and associated nut, as part of the module-to-module bottom tie assembly, is used as a tension element to prevent sliding separation between modules during a seismic event.

Evaluation Summary

A calculation was performed which demonstrated the proposed materials for the Base stud, with F_y = 75 ksi and F_t = 95ksi, and for the nut with 75 ksi proof stress, will perform their design functions with adequate safety margins, as follows.

Allowable tensile stress in the tie rod = 67.5 ksi (0.9Fy)

Tensile capacity of the tie rod = 75.9 kips

The stress limit is controlled by the tie rod tensile stress with a total capacity in 4 ties = 75.9 x 4 = 303.6 kips

Total tensile force in the bottom tie (from Table 3.6-21 of the UFSAR) = 249.2 kips

Stress ratio in the bottom tie = 249.2/303.6 = 0.82

This stress ratio is well below the allowable stress ratio of 1, and therefore, the structural capacity of the bottom tie is not impacted.

The eight 72.48 evaluation criteria were met.

72.48 Summary Report for the Standardized Advanced NUHOMS® System for the Period 8/16/08 to 8/12/10

Licensing Review (LR) 721029-311, Rev. 0* – *(incorporated into UFSAR Revision 4)

Change Description

The proposed activity involved allowing the use of ASTM A193 Grade B8R for the bolts and ASTM A194 Grade 8R for the nuts of advanced horizontal storage module (AHSM) 1 1/2" embedments. The current requirement was for materials per ASTM A453 Grade 660 or ASTM A193 Grade B8M2 Class 2 for externally threaded fasteners and ASTM A453 Grade 660 or ASTM A194 Grade 8M for nuts.

The stud and associated nuts, as part of the base-to-walls and base-to-door connections, are used to attach the end and rear walls and door to the base, and prevent sliding separation between walls and a base in a seismic event. The 1 1/2" embedments are also used to secure the transfer cask to the module during DSC transfer operations.

Evaluation Summary

The change does not introduce incompatible material with the concrete or a material that will corrode at a higher rate than the currently specified material. As evaluated in a calculation, the use of the proposed materials will have no affect on the stability of the AHSM. The proposed materials for the 1 1/2" embedment will perform their design functions, as discussed below.

The load on the base-to-wall connection using the1 1/2" embedments is 61.6 kips. The yield strength specified for Grade B8R in ASTM A193 is 55 ksi. The calculation evaluates the embedment strength at 219 degrees F with a capacity of 0.9 x $F_{Y(219\ deg\ F)}$ = 44.8 ksi

1.41 in^2 x 44.8 = 63.2 kips, which is > 61.6 kips; therefore this is acceptable.

Ultimate steel capacity = 141 kips, which is < 197 kips (pullout capacity); therefore this is acceptable.

The calculation indicates that the load on the door-to-base connection using the 1 1/2" embedments is 68.0 kips. The yield strength specified for Grade B8R in ASTM A193 is 55 ksi. The calculation evaluates the embedment strength at 100 degrees F with a capacity of 0.9 x $F_{y(100\ deg\ F)}$ = 49.5 ksi

1.41 in^2 x 49.5 = 69.8 kips, which is > 68.0 kips; therefore this is acceptable.

Ultimate steel capacity = 141 kips, which is < 149 kips (pullout capacity); therefore this is acceptable.

ASTM A 193/A 193M - 03 specifies that bolts, studs and stud bolts shall be furnished with nuts which shall conform to ASTM A 194/A 194M. The proposed materials for the nut and coupling nut used in the 1 1/2" embedments meet the requirements of ASTM A 194/A 194M. Strength, hardness, and chemical composition properties of materials for the components of the embedment are compatible.

The eight 72.48 evaluation criteria were met.

72.48 Summary Report for the Standardized Advanced NUHOMS® System for the Period 8/16/08 to 8/12/10

Licensing Review (LR) 721029-312, Rev. 1 – (incorporated into UFSAR Revision 4)

Change Description

Based on NRC Information Notice 2009-23, "Nuclear Fuel Thermal Conductivity Degradation," this proposed activity evaluated the impact of irradiated UO_2 pellet conductivity and thermal performance on the Standardized Advanced NUHOMS® System.

Evaluation Summary

The effective fuel conductivity used in the UFSAR for the 24PT1 DSC design is based on test data, which includes the effect of irradiation. Therefore, the performance of the 24PT1 DSC is unaffected by the proposed activity and no further evaluation was required for the 24PT1 DSC.

Calculations of effective fuel conductivities in the 24PT4 DSC design were based on un-irradiated UO_2 conductivity. Using the irradiated UO_2 conductivity will reduce the fuel effective conductivity and potentially increases the maximum fuel cladding temperature in the 24PT4. The evaluation considered therefore the design of the 24PT4 DSC.

The impact of the irradiated UO_2 conductivity on the thermal performance of the 24PT4 was determined. Based on the evaluation, the impact of the irradiated UO_2 conductivity on the maximum component temperatures, including the fuel cladding temperature, is approximately 1 °F. This amount of impact is considered insignificant. The internal pressure of the DSC is proportional to the average cavity gas temperature. Based on Table A4.4-10 of the UFSAR, an increase of 1°F for the cavity gas temperature increases the internal pressure by only 0.1% as shown in the following table. Therefore, the effect of irradiated UO_2 on the DSC internal pressure is also insignificant.

Condition	$T_{gas, avg}$ Table A4.4-10 of the UFSAR	Absolute Temperature Increase Ratio	Internal Pressure Increase
Normal	546.7	(547.7+460)/(546.7+460) = 1.0010	0.10%
Off-Normal	550.1	(551.1+460)/(550.1+460) = 1.0010	0.10%
Accident	713	(714+460)/(713+460) = 1.0009	0.09%

The irradiated UO_2 conductivity does not affect the 24PT4 DSC structure directly. The calculation showed that the effect of irradiated UO_2 conductivity on the maximum component temperatures and hence on the temperature gradients is approximately 1 °F and is insignificant. Therefore, the impact of the irradiated UO_2 conductivity on the thermal stresses of the 24PT4 DSC remain insignificant. Thus the stresses in the DSC shell assembly remain bounded by those reported in the UFSAR.

There is no adverse impact on the confinement capabilities of the cask as there are no new leak paths introduced and the effect of irradiated UO_2 on the cavity internal pressures is insignificant, as discussed above.

There is no adverse impact on shielding, as the irradiated UO_2 conductivity does not change the

source term limits of the fuel assemblies nor impact the shielding properties of the materials.

There is no adverse impact on criticality. There will be no dispersal or reconfiguration of pellet material. The structure, the geometry, and the neutron absorbing capability of the 24PT4 DSC materials remain unaffected by the irradiated UO_2 conductivity.

There is no adverse impact on operations. The temperature limits for operations, such as vacuum drying, remain unaffected. The increase of the maximum fuel cladding temperature for these operations is insignificant. Therefore, there are no required changes to the procedures for loading, unloading, or any other cask handling operations.

These results demonstrated that considering the effects of irradiated UO_2 conductivity does not adversely affect the system design functions.

The eight 72.48 evaluation criteria were met.

72.48 Summary Report for the Standardized Advanced NUHOMS® System for the Period 8/16/08 to 8/12/10

Enclosure 4 Part 2 - NONCONFORMANCES

LR 721029-227 Rev. 1 - (no associated UFSAR change)

Change Description

The proposed activity evaluated a non-conformance situation in which 155 kips of hydraulic ram force was applied to a dry shielded canister (DSC) outer bottom cover plate (OBCP) during insertion of the DSC into an advanced horizontal storage module (AHSM). The design basis load is 80 kips.

Evaluation Summary

The affect of applying a force of 155 kips to the OBCP of the DSC was evaluated. A stress evaluation was performed to assess the effect of the 155 kip ram push overload on the DSC. The 80 kips ram push load is an off-normal load condition that is evaluated to meet Service Level B stress limits. Therefore the 155 kip overload was also evaluated against the same Service Level B stress limits.

Revision 1 of this evaluation addressed the 155 ram push load based on elastic and plastic analyses using the design basis ANSYS models of the DSC used for the evaluation of the load cases under consideration (LD-3 and LD-4 in UFSAR Table A.3.6-1). Both the elastic analysis and plastic analysis methodologies have been used for the design basis evaluations of the 24PT4 DSC documented in the UFSAR. The load combination consists of the 155 kip ram push load plus deadweight, combined with and without internal pressure (for the above load cases the analysis without internal pressure are controlling). In addition because of the large magnitude of the applied load, which causes additional bending deformation of the outer cover in the region inside the grapple ring, a bounding case considering load transfer through the lead to the inner cover was considered for the elastic analysis.

The results of the evaluation showed that primary membrane (Pm) and primary plus secondary stresses (Pm + Pb + Q) in the pressure boundary components at the bottom end of the DSC (canister shell and Inner Bottom Cover Plate as shown in UFSAR Figure A.1.2-1) are within the ASME Service Level B limits. The allowable stresses are based on ASME Code values at temperature, considering the actual heat load of the DSC. For primary membrane plus bending the DSC shell meets ASME stress limits based on ASME Code allowables; however, for the IBCP the acceptability of the primary membrane plus bending stresses are based on plastic analysis per NB-3228 at temperatures developed considering the actual heat load. The non-pressure boundary components (the OBCP, inner grapple ring, and outer grapple ring and grapple ring support) are also within the Level B stress limits when evaluated based on CMTR properties and refined temperature data for these components. Thus, in addition to the elastic stress analysis, limit analysis in accordance with NB-3228.1 was performed. NB-3228.1 states that the limits on primary membrane and membrane plus bending stress intensity need not be satisfied at a specific location if it can be shown by limit analysis that the specified loadings do not exceed two-thirds of the lower bound collapse load. The limit analyses are elastic-perfectly plastic analyses using a load of 232.5 kips (155 kips = % of 232.5 kips) and a conservative yield value of 1.5 times the Code value of S_m for Type 316 at 500 °F (S_m = 18,000 psi).

Based on the results of this evaluation, the primary membrane and primary plus secondary stresses in the DSC bottom end components due to the 155 kip ram push load are within ASME

Code limits. To address those components not meeting the primary membrane plus bending stress limits, the DSC was evaluated in accordance with the plastic analysis provisions of the ASME Code (NB3228). The limit analysis provisions of NB-3228.1 were applied to demonstrate code compliance of the 24PT4 DSC for the increased ram push load.

There is no impact on the mechanical, thermal, shielding and criticality analyses.

The eight 72.48 evaluation criteria were met.

LR 721029-295 Rev. 0* - *(no associated UFSAR change)

Change Description

The proposed activity involved an evaluation of the impact of leaving a poly bristle in a single fuel assembly before it was placed into a certain model dry shielded canister (DSC) for long term storage. The bristle was not recoverable from the fuel assembly during pre-dry-storage inspections.

Evaluation Summary

The effect of introducing a small amount of foreign material into the DSC cavity along with the fuel assembly was evaluated as follows.

The material was identified as a polystyrene bristle approximately 6 inches in length with a mass of approximately 0.01 grams. To maximize the impact of any contribution from the foreign material (bristle) it was conservatively assumed that the internal DSC atmosphere was at 14.7 psia (0 psig) and complete vaporization of the organic bristle occurred. The basic chemical composition of poly (styrene:acrylonitrile) is a mixture of carbon, hydrogen, nitrogen and possibly some chlorine atoms, so assuming that the total mass converts to hydrogen gas will maximize the volume increase.

The density of hydrogen = 0.0056 lbs/ft^3.

The weight of the bristle = 0.01 grams = 2.21 x10^{-5} lbs, resulting in an increase in volume of (2.21.x 10^{-5}/ 0.0056) = 0.004 ft^3. (This is conservative as the carbon portion of the polystyrene is unlikely to turn to gas and if it did would result in a lower volume.)

There are two structural concerns with regards to the introduction of a small quantity of foreign material into the DSC cavity:

1. Impact on the pressure boundary and the stored fuel assemblies, and
2. Impact of the foreign material on the internal DSC environment, including the inert atmosphere and pressure.

Impact on pressure boundary and stored fuel assemblies:

Given that the polystyrene bristle is placed within a DSC three scenarios are possible:

1. The bristle did not melt/vaporize during vacuum drying (VD) operations and remains

present at the start of HSM storage. This material is then present in an inert dry atmosphere (helium).

2. The bristle melted and vaporized during VD operations and all that remains is residue.

3. The bristle decomposed/melted during VD operations, but did not vaporize. It thus remains as a solid reconfigured piece of material.

The third option is the most likely scenario, given that poly (styrene:acrylonitrile) is a thermoplastic material that melts at 100 to 300°C into a highly viscous liquid and then solidifies upon cooling. The temperature in the DSC is likely to exceed the melting temperature during the VD process.

In the unlikely event that the bristle vaporized and the decomposition products were not removed during VD then residual bristle gas components can only result in corrosion of the DSC components and/or the spent fuel assemblies (SFA's) if the internal environment were corrosive. The lack of an oxidizing agent in the DSC and the inert helium gas fill of 99.75% of the free volume will preclude any corrosive attack on the DSC components.

Regarding the impact of the bristle upon the SFA's, to quantify any impact of the bristle on corrosion of the fuel cladding, corrosion rates for zirconium were researched. An extremely conservative case involves liquid hydrochloric acid on zirconium, and gives a rate of < 0.001 inch/year. The nominal cladding thickness for the SFA's involved is 0.025" and full thickness corrosion would require more than 25 years immersed in liquid hydrochloric acid, not the dry helium atmosphere provided in the 24PT4 DSC.

Both the stainless steel pressure boundary and the zirconium cladding are resistant to corrosion and it would take many years, even assuming unrealistic environments, to reduce thickness to a level of concern. Therefore, corrosion from a very small amount of hydrogen or chlorine in a dry helium (inert gas) atmosphere is not a concern for the pressure boundary or the fuel cladding.

Impact on the internal DSC environment, including the inert atmosphere and pressure:

The impact of the bristle on DSC pressurization is assessed as follows:

The DSC free volume of the model involved is 247 ft^3. It was conservatively assumed that the helium atmosphere is at 14.7 psia (0 psig) to maximize the impact of any contribution from off-gassing of the bristle. As shown above, a conservative estimate for the additional gas volume due to vaporization of the bristle is 0.004 ft^3. This results in a ratio of the increase in internal pressure ((247 + 0.004)/ 247) = 1.000 which cannot be detected. Thus there is no impact on the design basis pressure.

The eight 72.48 evaluation criteria were met.

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LR 721029-318 Rev. 0* - *(no associated UFSAR change)

Change Description

The proposed activity involved a non-conformance associated with use of ASTM A193 Class 2 Grade B8M2 fasteners which did not meet the yield strength specified in the standard and used in the design basis. The reported yield strength is 73 ksi and that required by ASTM A193 is 75 ksi. The subject fasteners are advanced horizontal storage module (AHSM) shield door bolts and the studs used at the bottom side-to-side connection between AHSMs.

Evaluation Summary

Shield Door Bolts

The licensing design basis calculation evaluates the shield door connection bolt. Incorporating the decrease in yield strength from 75 ksi to 73 ksi showed values within the capacity limits of the bolts, as follows.

Tensile Load on Door Connection per bolt = 68.0 kips
Tensile Capacity of the Bolt at 73 ksi yield strength = 92.6 kips

92.6 kips > 68.0 Kips, yielding a stress ratio of 0.73.

Side-to-Side Connections

The licensing design basis calculation evaluates the side-to-side connection fasteners. Incorporating the decrease in yield strength from 75 ksi to 73 ksi shows values within the capacity limits of the fasteners, as follows.

Tensile force load per tie = 228.3/4 = 57.1 kips
Tensile force per one tie due to torsion in the module = 25.0 kips
The total tensile force in the bottom tie $= (57.1^2 + 25.0^2)^{1/2} = 62.3$ kips

Tensile Capacity at 73 ksi yield strength = 73.87 kips

62.3 kips < 73.87 kips, yielding a stress ratio of 0.84.

The eight 72.48 evaluation criteria were met.

LR 721029-327 Rev. 0* - *(no associated UFSAR change)

Change Description

The proposed activity involved allowing the storage of certain model DSC's containing small rust flakes which originate from the reactor top flange during fuel movements. These small flakes are basically iron oxide formed during the refueling outage which become lodged in the peripheral fuel assemblies in the reactor core and are difficult to remove. The intention is to allow the inclusion of small rust flakes (total <1 gram) in any one fully loaded certain model DSC and an AHSM.

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Evaluation Summary

The effect of introducing a small amount of inert foreign material in the form of hematite iron oxide particles into the DSC cavity along with the fuel assemblies is evaluated as follows.

The debris has been identified as flakes of hematite iron oxide (Fe_2O_3) that formed at low temperatures by oxidation of the reactor vessel flange (alloy steel). This occurs during a refueling outage when the flange is exposed to oxygen and borated water. The particles are less than .25 grams each for an addition of <1 gram to the DSC contents assuming that the spent fuel assemblies (SFA's) have attached rust particles. Additionally, the flakes have been contaminated with minute quantities of fine particles from the degraded Boraflex™ polymer found in the spent fuel pool.

There are two structural concerns with regards to the introduction of a small quantity of foreign material into the DSC cavity:

1. Impact on the pressure boundary and the stored fuel assemblies, and
2. Impact of the foreign material on the internal DSC environment, including the inert atmosphere and pressure.

The rust flakes of hematite iron oxide are totally inert and will not break down into their constituent atoms under the radiation field found in a DSC. As such the small pieces of debris will not constitute a danger to the SFA cladding or the DSC stainless steel shell once the interior of the canister is drained, dried and inerted with helium. The presence of small flakes of inert iron oxide would have no affect on any of the fuel assembly components or the DSC materials.

The small rust particles (total <1 gram) are inert and will have no affect on DSC pressurization. The amount of polymer contamination is too small to generate any measureable changes on the DSC pressure even if the polymer is not removed during vacuum drying.

There is no adverse impact on the mechanical design function. The inert rust particles are very small in size (0.375" x 0.5" x 0.010" thick) and of insignificant weight (< 1 gram) so that no problems are anticipated in successfully vacuum drying the contents of the DSC.

There is no adverse impact on the thermal design function. The limiting source term, and therefore the decay heat load limit for each fuel assembly and the total DSC remain unchanged.

There is no adverse impact on the shielding design function. The "as left" configuration is unchanged from the assumed configuration in the shielding calculations; therefore, there is no change in the design basis source term used in the shielding evaluation.

There is no adverse impact on the criticality design function. The fuel cladding is not affected by this minute amount of inert material.

There is no impact on the confinement capabilities of the DSC's as there are no new leak paths introduced.

The eight 72.48 evaluation criteria were met.

72.48 Summary Report for the Standardized Advanced NUHOMS® System for the Period 8/16/08 to 8/12/10

LR 721029-332 Rev. 0* - *(no associated UFSAR change)

Change Description

The proposed activity involved a non-conformance associated with accepting an overpressure condition of 50 psig for a dry shielding canister (DSC) which exceeded the analyzed blowdown pressure of 20 psig.

Evaluation Summary

The thermal, criticality, shielding and confinement design functions are not affected by the overpressure condition. The design basis for blowdown pressure was exceeded and a structural evaluation was performed as discussed below.

An evaluation was performed which showed that stresses in bounding areas of the components and welds increased over that analyzed previously but stayed within the ASME code allowable values, as follows.

The DSC was previously evaluated for a blowdown pressure of 20 psig. The analysis was performed in accordance with ASME section III Division 1 NB-3000 and an FEA model was used. Using the same model as in the previous analysis, the DSC was evaluated for structural adequacy for a blowdown pressure of 50 psig.

A review of the results shows that the stresses, from the higher blowdown pressure, meet the allowable stress limits contained in the previous analysis. In addition, the results show that component and weld stresses are below the ASME code limits as defined in ASME section III Division 1 NB-3226 for a hydrostatic test pressure.

The eight 72.48 evaluation criteria were met.