TRANSNUCLEAR
AN AREVA COMPANY

June 18, 2010
E-29449

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

Subject: Transnuclear, Inc. (TN) Application for the TN-40 Transportation Packaging for Spent Fuel, Revision 9, Docket No. 71-9313, TAC No. L24106

Based on recent discussions with the NRC Staff, changes have been made to the TN-40 Transportation Application Safety Analysis Report (SAR) in the areas of thermal, shielding, criticality, and operations. The changed SAR pages are provided herein as Enclosures 2 and 3, for the proprietary and non-proprietary SAR versions, respectively. Enclosure 1 provides instructions for SAR page removal and insertion. Additionally, a revision to Transnuclear, Inc. Calculation 10421-051 is included.

This submittal includes proprietary information which may not be used for any purpose other than to support your staff's review of the application. In accordance with 10 CFR 2.390, I am providing an affidavit (Enclosure 4) specifically requesting that you withhold this proprietary information from public disclosure.

Should the NRC staff require additional information to support review of this application, please do not hesitate to contact Mr. Donis Shaw at 410-910-6878 or Jayant Bondre at 410-910-6881.

Sincerely,

Robert Grubb
Chief Operating Officer

cc: Meraj Rahimi (NRC SFST) (8 copies, provided in a separate mailing)

Enclosures:

1. TN-40 Revision 9 SAR Page Replacement Instructions
2. Changed Pages for the TN-40 Application Safety Analysis Report, Revision 9, Proprietary version
3. Changed Pages for the TN-40 Application Safety Analysis Report, Revision 9, Non-proprietary version
4. Affidavit Pursuant to 10 CFR 2.390
5. Transnuclear, Inc. Calculation 10421-051, Revision 1 (proprietary version)
6. Transnuclear, Inc. Calculation 10421-051, Revision 1 (non-proprietary version)

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NMSS

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Enclosure 1 to TN E-29449

TN-40 Revision 9 SAR Page Replacement Instructions

Proprietary Version	
Old page	**Replacement Page(s) (Revision 9)**
Cover Page	Cover Page
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5-1	5-1
5-2	5-2
5-9B	5-9b
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6-33E	6-33e
6-33F	6-33f
6-33G	None (remove page 6-33G)
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TN-40 Revision 9 SAR Page Replacement Instructions

Proprietary Version	
Old page	**Replacement Page(s) (Revision 9)**
6-71A	6-71a
6-71B	6-71b
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6-71I	6-71i
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6-84	6-84
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6-95D	6-95d
6-95E	6-95e
6-113	6-113
6-114	6-114
6-115	6-115
6-119	None (remove page 6-119)
7-5	7-5
7-13	7-13
Non-proprietary Version	
Old page	**Replacement Page(s) (Revision 9)**
Cover Page	Cover Page
1-8	1-8
5-1	5-1
5-2	5-2
5-9B	5-9b
5-9C	5-9c
5-51	5-51
6-ii	6-ii
6-6	6-6
6-18	None (remove page 6-18)
7-5	7-5
7-13	7-13

Enclosure 3 to TN E-29449

Changed Pages for the TN-40 Application Safety Analysis Report, Revision 9, Non-proprietary Version

NON-PROPRIETARY

AREVA

TRANSNUCLEAR, INC.

TN-40

TRANSPORTATION PACKAGING

SAFETY ANALYSIS REPORT

Revision 9
June 2010

7135 Minstrel Way, Suite 300 • Columbia, MD 21045

m. The fuel assemblies shall not be Unit 1 Region 4 fuel assemblies (i.e., assemblies identified as D-01 through D-40).

n. The maximum uranium loading per fuel assembly is 0.410 MTU.

o. The fuel shall not be a DAMAGED FUEL ASSEMBLY.

A DAMAGED FUEL ASSEMBLY is a spent nuclear fuel assembly that:

- is a partial fuel assembly, that is, a fuel assembly from which fuel pins are missing unless dummy fuel pins are used to displace an amount of water equal to that displaced by the original pins; or
- has known or is suspected to have structural defects or gross cladding failures (other than pinhole leaks) sufficiently severe to adversely affect fuel handling and transfer capability.

p. The characteristics of the specific fuel types authorized for shipment in the TN-40 Cask are provided in the table below. The table shows the pre-irradiated nominal design dimensions and specifications for the fuel.

Fuel Designations	**Exxon Standard (14x14)**	**Exxon Toprod (14x14)**	**Exxon High Burnup (14 x 14)**	**Westinghouse Standard (14x14)**	**Westinghouse OFA (14x14)**
Rod Pitch (in.)	0.556	0.556	0.556	0.556	0.556
Pellet OD (in.)	0.3565	0.3505	0.3565	0.3659	0.3444
Clad OD (in.)	0.424	0.426	0.417	0.422	0.400
Clad Thickness (in.)	0.0300	0.0295	0.0310	0.0243	0.0243
Number of Fueled Rods	179	179	179	179	179
Clad Material	Zr-4	Zr-4	Zr-4	Zr-4	Zr-4
Number of Guide Tubes	16	16	16	16	16
Number of Instrument Tubes	1	1	1	1	1
Active Fuel Length (in.)	144	144	144	144	144
Maximum Length (Assembly+BPRA) (in.)	161.3	161.3	161.3	161.3	161.3
Maximim Width (in.)	7.763	7.763	7.763	7.763	7.763

q. The maximum heat load is 19.0 kW per cask *and* 0.475 kW per fuel assembly, including the BPRAs, and TPAs.

5.0 SHIELDING EVALUATION

5.1 Discussion And Results

Shielding for the TN-40 package is provided mainly by the cask body. The cask body is made up of the containment vessel, the gamma shielding and the lid. For the neutron shielding, a borated polyester resin compound surrounds the gamma shield shell radially. Additional shielding is provided by the steel outer shell surrounding the resin layer and by the steel and aluminum structure of the fuel basket.

For transport, wood filled impact limiters are installed on either end of the cask and provide additional shielding for the ends and some radial shielding for the areas at either end of the radial neutron shield. Figure 5-1 shows the configuration of the package shielding. Table 5-1 lists the compositions of the shielding materials.

The fuel assemblies acceptable for transport in the TN-40 are listed in Section 1.2.3. Using the SAS2H/ORIGEN-S modules of SCALE [1], source terms are calculated. The bounding design basis fuel for dose rate has an initial enrichment of 2.35 wt% and a total maximum bundle-average burnup of 42,000 MWD/MTU with a 24.4 year decay time. Note that the criticality evaluation documented in Chapter 6 requires a minimum cooling time of 30 years. The evaluated decay heat is 21 kW/cask as opposed to the decay heat of 19 kW/cask (corresponding to a cooling time of 30 years).

The Westinghouse 14x14 standard fuel assembly contains the maximum heavy metal weight (Section 1.2.3) which results in bounding neutron and gamma source terms and is therefore identified as the most conservative fuel assembly. Section 5.2 describes the source specification and Section 5.4 describes the shielding analysis performed for the TN-40 cask. The shielding analysis models are described in Section 5.3.

Normal Conditions of Transport (NCT) are modeled with the neutron shielding and impact limiters intact. This shielding calculation is performed using the Monte Carlo computer code MCNP [5, 9]. Dose rates on the side, top and bottom of the TN-40 package are calculated for the various sources described in Section 5.2 and summed to give a total gamma and neutron dose rate.

Hypothetical Accident Conditions (HAC) assume that the neutron shield and the impact limiters are removed. This evaluation bounds the accident conditions since it is shown in Chapter 2 and Chapter 3 that the neutron shielding may be lost but the impact limiters remain on the cask during HAC. Shielding calculations for the HAC are also performed using MCNP.

The expected maximum dose rates (for NCT and HAC) from the TN-40 package are provided in Table 5-2. These dose rates are *the design basis dose rates for the TN-40 package* with a *minimum* cooling time of *30* years. Although this dose rate evaluation is performed using design basis fuel, *fuel qualification* evaluations were performed to determine that 15 year minimum cooled fuel is also acceptable for certain burnup and enrichment combinations. These evaluations were performed to determine the fuel assembly parameters of burnup, percent initial enrichment and cooling time that would result in decay heat and radiological sources that would meet the decay heat requirements (Chapter 3), source terms for containment (Chapter 4) and radiological sources that provide dose rates less than the current design basis fuel mentioned above and thus would be acceptable for transport *(from a shielding standpoint)* in the TN-40 package. Section 5.2 describes these evaluations in more detail.

The shielding calculations considered effects of tolerances. Since dose rates along the side of the transportation package are controlling, the cumulative effect of tolerances (+.05/-.01 in. on 1.50 in. thick inner shell and +/-.12 in. on 8.00 in. thick gamma shell) of steel thicknesses and tolerances (+/-.12 in. on 4.50 in.) of resin on the side of the cask is considered. Only tolerances in thicknesses of the neutron shielding, cask inner and gamma shells affect the dose rates along the side of the cask. Note, dose rates presented in Table 5-2, Table 5-18, and Table 5-19 include the effect of the described geometrical tolerances.

The effect of material tolerances is considered only for the neutron shielding resin. Only hydrogen concentration is considered significant enough to affect the dose rates. The weight percent of hydrogen considered in the design basis shielding analysis represents the minimum guaranteed composition following the resin qualification. The average measured hydrogen weight percent in the TN-40 casks is 5.21 while that employed in the calculations is 5.05. The boron content has an effect on the secondary gamma dose rate. However, a concentration of greater than 0.75% ensures that this concentration is saturated and is sufficient to reduce the contribution of the secondary gamma component. Therefore, a material tolerance calculation with boron is not performed.

The effect of the tolerances on dose rates at various distances from the ends and at radial distances from the side greater than 2 meters is not significant.

The following items are also considered when using dose rates in Table 5-2, Table 5-18, and Table 5-19.

- Design basis Westinghouse 14x14 Standard fuel assemblies with the bounding neutron and gamma source terms are utilized in the shielding evaluation.
- The fuel qualification methodology calls for conservatively adjusting the enrichment/burnup and cooling time of the loaded fuel assemblies (Table 5-8).
- Calculated dose rates are generally higher than measured dose rates, demonstrating the conservatisms in the shielding analysis methodology.
- The burnup-enrichment parameters of the design basis fuel assembly employed in the shielding evaluation are conservative compared to the burnup-enrichment distribution for the actual inventory of fuel assemblies as shown in Figure 6-17.

For the dose calculation around the TN-40, the source is divided into four separate regions: fuel, plenum, top end fitting, and bottom end fitting. The model is utilized in two separate computer runs consisting of contributions from the following sources:

- Primary gamma radiation from the active fuel and from activated hardware within the top end fitting, plenum region and bottom end fitting (axial and radial directions).
- Neutron radiation from the active fuel region and secondary gamma radiation from neutron interactions.

The sources in the active fuel region (gamma and neutron) are modeled as uniform radially but vary axially. The sources in the structural hardware regions (plenum, top end fitting, and bottom end fitting) are modeled as uniform both radially and axially. The results from the individual runs are summed to provide the total gamma, neutron and total dose for the package.

The statistical uncertainties are generally less than 5% for the majority of tallies except for local tally bins and the accident results. For the accident the neutron end dose rates have the highest relative error around 10%. The statistical uncertainties associated with the neutron dose rates on the top and bottom impact limiter surface are high, but since they contribute less than 1% (less than 0.1 mrem/hr) to the total dose this is acceptable.

The terminology for the dose locations is as follows. On the side of the cask results are reported on the surface of the cask ("contact"), at vertical planes extending up from a 10 feet wide vehicle ("vertical planes"), at the diameter of the impact limiters to represent the top and bottom of the package ("top/bottom"), 1 meter from the steel cask body (1 meter accident) and 2 meters from the vertical planes.

The results indicate peaking near the top and bottom of the cask and streaming in the upper trunnion/above the neutron shield regions. These results are expected due to the reduced shielding in these areas. It was determined that the normal conditions peak external surface dose rate of *60* mrem/hr occurs just above the neutron shield. This is approximately a factor of 1.8 times higher than the average on the cask surface. The localized peaking at the top of the cask is due to the absence of the neutron shield at the top. Neutron streaming was observed through the trunnion itself. However, the total dose rates just outside the trunnion were nearly the same as those averaged around the entire circumference of the cask.

Table 5-2 presents the maximum calculated dose at contact, at the vehicle's outer edge (assumed 10 ft wide vehicle), and at 2 m from the vehicle's outer edge. The calculated total dose rates at the various locations around the package are presented in Table 5-2, Table 5-18, and Table 5-19.

For the HAC, Table 5-2 also presents the maximum calculated doses at 1 m from the cask body.

The dose rates for an individual at the end of the rail car are presented in Table 5-17. These results are presented as a function of the length of the rail car.

On average, the dose rates are dominated by the neutron source term. The results indicate that typically the total dose rates are comprised of 25% to 30% primary gamma, 15% (n,γ) and 55% to 60% neutron. However, the primary gamma source produces the majority of the dose rate at the ends of the package; the average contribution from primary gamma is in the range of 80% to 85%. This is a direct result of the neutron shielding from the wood in the impact limiters. As expected, the accident dose rates are produced mostly from the neutron (94%) source due to the loss of the neutron shielding material and the impact limiter.

Typical average (beyond 130 cm above and below the active fuel midplane) contact dose rates on the side of the cask are approximately *36* mrem/hr (~55% neutron). At 2 meters from the side of the *vehicle surface* the dose rate is approximately *7.8* mrem/hr which is comprised of 4.*3* mrem/hr neutron, 0.*8* mrem/hr (n,γ) and *2.7* mrem/hr gamma. On the ends, the total contact dose rates are less than 7 mrem/hr with less than a 0.1 mrem/hr contribution from neutrons. All these dose rates are at the lower tolerance limits of the shielding materials thickness on side of the cask and are based on source terms calculated with a cooling time of *30* years. Addressed tolerances are specified at the end of Section 5.1.

Axial distribution of the total dose rate at various radial distances from the side of the transportation package is presented in Table 5-18 and Table 5–19. Note that the neutron shield extends from -187 cm to +205 cm axial range in the MCNP calculational model. The table shows that there is a dose rate increase from +190 cm to +230 cm axial coordinate range when considering dose rates at radial distances not exceeding the radius of impact limiters. The dose rate at that location is larger than at the middle of the cask because of less steel shielding due to the "flat area" near the trunnions and the absence of neutron shielding. Further, the MCNP calculational model for the top and plenum regions includes the gamma sources from BPRAs and TPAs. The bottom trunnions and the cask's side at axial coordinates less than -187 cm are encompassed by the bottom impact limiter and there are no BPRA/TPA sources in the bottom region (the ratio of top/bottom gamma source strength is roughly a factor of 1.2). Therefore, the increase in the dose rates near the bottom trunnions is lower.

Because of the lack of shielding (see Figure 5-3) and radiological source concentration near the top trunnions, dose rates were examined at axial coordinates around the trunnions. Specifically, dose rates near the top trunnions, on the flat part around the top trunnions as pointed out with the "P1" callout on a sketch of Figure 5-4, are evaluated. The contact dose rate at this point is *46.8* mrem/hr (*32.2* mrem/hr neutron, *3.9* mrem/hr (n,g) and 1*0.7* mrem/hr gamma). At 2 meters radial distance measured from the side of impact limiters, the total dose rate is *6.75* mrem/hr.

The dose rates shown in Table 5-18 and Table 5-19 correspond to 24.4-year cooled and 30-year cooled fuel, respectively. Two sets of 2 m dose rates are shown in these tables, based on a vehicle width of 10 ft and the edge of the impact limiters (12 ft wide),

Table 5-2
Summary of TN-40 Dose Rates
(Exclusive Use)

Normal Conditions of Transport	Package Contact Dose Rate mSv/hour (mrem/hour)			Vehicle Surface(1) mSv/hour (mrem/hour)	2 Meters from Vehicle Surface mSv/hour (mrem/hour)		
Radiation	**Top**	**Side**	**Bottom**		**Top**	**Side**	**Bottom**
Gamma	0.069 (6.9)	0.30 (30)	0.059 (5.9)	0.19 (19)	-	0.035 (3.5)	-
Neutron	0.0004 (0.04)	0.30 (30)	0.0008 (0.08)	0.21 (21)	-	0.043 (4.3)	-
Total	0.069 (6.9)	0.60 (60)	0.060 (6.0)	0.40 (40)	<0.069 (<6.9)	0.078 (7.8)	<0.060 (<6.0)
Limit	2 (200)	10 (1000)	2 (200)	2 (200)	0.1 (10)	0.1 (10)	0.1 (10)

(1) Vehicle surface is bounded axially by the external surfaces of the impact limiters and radially by the vertical planes extending from a 10 ft wide vehicle. The bounding radial dose rates are shown for all surfaces.

Hypothetical Accident Conditions(2)	1 Meter from Package Surface mSv/hour (mrem/hour)		
Radiation	**Top**	**Side(3)**	**Bottom**
Gamma	0.43 (43)	0.32 (32)	0.28 (28)
Neutron	0.68 (68)	5.34 (534)	1.45 (145)
Total	1.11 (111)	5.66 (566)	1.73 (173)
Limit	10 (1000)	10 (1000)	10 (1000)

(2) The neutron shield and the impact limiters are removed.

(3) Does not account for tolerances on side of the cask described at the end of Section 5.1. The effect of tolerances is less than 10%. It is not significant to the extent that dose rates would exceed regulatory limits.

LIST OF TABLES

LIST OF FIGURES

14. Oak Ridge National Laboratory, "Validation of the SCALE System for PWR Spent Fuel Isotopic Composition Analysis," ORNL/TM-12667, Published March 1995.

15. CAL-UDC-NU-000011 Rev A, , "Three Mile Island Unit 1 Radiochemical Assay Comparisons to SAS2H Calculations", Office of Civilian Radioactive Waste Management, U.S. Department of Energy, April 2002.

16. U.S. Nuclear Regulatory Commission, "Isotopic Analysis of High-Burnup PWR Spent Fuel Samples from the Takahama-3 Reactor," NUREG/CR-6798, Published January 2003, ORNL/TM-2001/259.

17. Radulescu G, Mueller D. E. and J. C. Wagner, "Sensitivity and Uncertainty Analysis of Commercial Reactor Criticals for Burnup Credit," Oak Ridge National Laboratory, January 2008, ORNL/TM-2006-87, NUREG/CR-6951.

18. Criticality Model, CAL-DS0-NU-0000003 REV 00A, Bechtel SAIC Company, Las Vegas, Nevada, 2004.

19. S. M. Bowman, W.C. Jordan, J. F. Mincey, C.V. Parks, and L. M. Petrie, "Experience with the SCALE Criticality Safety Cross-Section Libraries," Oak Ridge National Laboratory, NUREG/CR-6686, Published October 2000, ORNL/TM-1999/322.

20. C. V. Parks, M. D. DeHart, and J. C. Wagner, "Review and Prioritization of Technical Issues Related to Burnup Credit for LWR Fuel," NUREG/CR-6665, ORNL/TM-1999/303, Oak Ridge National Laboratory, February 2000.

7.1.3.7 Leak test the lid, vent and drain port cover seals by measuring the leakage of helium into the volume between the concentric metallic seals of the lid, vent and drain ports. The maximum acceptable cask seal leak rate is 1×10^{-4} ref cm^3/sec. The leak test shall be performed in accordance with ANSI N14.5 [1] using a method having adequate sensitivity to measure the maximum acceptable leak rate. Use of a helium spectrometer is the preferred method for this test, although the gas pressure rise test could also be used because of the small volume of the cavity between inner and outer metallic seals. After the leak test, replace the overpressure port cover.

7.1.3.8 If the cask does not pass the leak test, determine and correct the source of the leak. Repeat the leak test.

7.1.3.9 If the cask still does not pass the leak test, evaluate the test method or return the cask to the pool and replace the lid seals.

7.1.3.10 Re-engage the lift beam to the upper (top) trunnions of the cask.

7.1.3.11 Move the transport vehicle with transport frame in place into the loading position and prepare the upending/downending frame.

7.1.3.12 Lift the cask off the decontamination pad, and place the rear trunnions on the rear trunnion supports of the upending/downending frame.

7.1.3.13 Rotate the cask from the vertical to the horizontal position.

7.1.3.14 Using a spreader bar and lifting straps, lift the cask from the upending/downending frame and lower it onto the transport frame.

7.1.3.15 *Perform a neutron and gamma dose rate survey over the entire surface of the cask to demonstrate the adequacy of the shielding design and to check if the surface dose rates are within the regulatory limits. Check surface contamination levels to verify that levels are within the regulatory limits. Perform an external temperature survey as described in Section 3.4.7 for monitoring thermal performance.*

7.1.3.16 Install the tie-down straps.

7.1.3.17 Prior to installing the impact limiters, inspect them visually for damage. The impact limiters may not be used without repair if any wood has been exposed. Damage due to handling other than small dings and scratches must be evaluated for their effect on the performance during the hypothetical drop and puncture accidents.

7.1.3.18 Install the top impact limiter spacer on the front end (lid end) of the cask then remove the spacer lifting eye bolts.

7.1.3.19 Install the front (top) and the rear (bottom) impact limiters onto the cask. Lubricate the attachment bolts with Loctite N-5000 or an equivalent and torque to 60 - 80 ft-lb.

test does not replace the seal leakage test specified above in step 7.4.1.17.

7.4.1.20 Re-engage the lift beam to the upper (top) trunnions of the cask.

7.4.1.21 Move the transport vehicle with transport frame in place into the loading position and prepare the upending/downending frame.

7.4.1.22 Lift the cask, and place the rear trunnions on the rear trunnion supports of the upending/downending frame.

7.4.1.23 Rotate the cask from the vertical to the horizontal position.

7.4.1.24 Using a spreader bar and lifting straps, lift the cask from the upending/downending frame and lower it onto the transport frame.

7.4.1.25 Perform a neutron and gamma dose rate survey over the entire surface of the cask to demonstrate the adequacy of the shielding design *and to* check if the surface dose rates are within the regulatory limits. *Check surface contamination levels to verify that levels are within the regulatory limits.* Perform an external temperature survey as described in Section 3.4.7 for monitoring thermal performance.

7.4.1.26 Install the tie-down straps.

7.4.1.27 Prior to installing the impact limiters, inspect them visually for damage. The impact limiters may not be used without repair if any wood has been exposed. Damage due to handling other than small dings and scratches must be evaluated for their effect on the performance during the hypothetical drop and puncture accidents.

7.4.1.28 Install the top impact limiter spacer on the front end (lid end) of the cask and then remove the spacer lifting eye bolts.

7.4.1.29 Install the front (top) and the rear (bottom) impact limiters onto the cask. Lubricate the attachment bolts with Loctite N-5000 or an equivalent and torque to 60 - 80 ft-lb in the final pass.

7.4.1.30 Install thirteen impact limiter attachment tie-rods between the front and the rear impact limiters.

7.4.1.31 Render the impact limiter lifting lugs inoperable by covering the lifting holes or installing a bolt inside the holes to prevent their inadvertent use.

7.4.1.32 Install *the* security seal on one tie-rod and lock sleeve.

7.4.1.33 Install the personnel barrier.

7.4.1.34 Check the temperature on all accessible surfaces to make sure that it is <185°F.

7.4.1.35 Perform a final radiation and contamination survey to satisfy the shield test requirements and to assure compliance with 10 CFR 71.47 and 71.87.

7.4.1.36 Apply appropriate DOT labels and placards in accordance with 49 CFR 172. Prepare the final shipping documentation.

7.4.1.37 Release the loaded cask for shipment.

AFFIDAVIT PURSUANT
TO 10 CFR 2.390

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

I, Robert Grubb, depose and say that I am Chief Operating Officer 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 and as listed below:

- Certain Portions of SAR Chapter 6
- Transnuclear, Inc. Calculation 10421-051, Revision 1

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 is certain portions of the TN-40 safety analysis report discussion of the criticality analysis, 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 transportation 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.

Robert Grubb
Chief Operating Officer, Transnuclear, Inc.

Subscribed and sworn to me before this 18th day of June, 2010.

Arleastrice M Taylor
Notary Public

My Commission Expires 10 / 14 / 2012

ARLEASTRICE M. TAYLOR
NOTARY PUBLIC
HOWARD COUNTY, MD