TRANSNUCLEAR
AN AREVA COMPANY

May 7, 2010
E-29342

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 8, 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) involving the basket structural analysis, the thermal analysis, and the shielding analysis. The changed SAR pages are provided herein as Enclosures 2 and 3, for the proprietary and non-proprietary SAR versions, respectively. The only difference between the replacement SAR pages for the proprietary and non-proprietary SAR versions is the SAR cover page. Enclosure 1 provides instructions for SAR page removal and insertion.

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 me at 410-910-6881.

Sincerely,

Jayant Bondre

Jayant Bondre, PhD
Vice President - Engineering

cc: Meraj Rahimi (NRC SFST) (8 copies of this cover letter and Enclosures 1 and 2, provided in a separate mailing)

Enclosures:

1. SAR Page Replacement Instructions
2. Changed Pages for the TN-40 Application Safety Analysis Report, Revision 8, Proprietary version
3. Changed Pages for the TN-40 Application Safety Analysis Report, Revision 8, Non-proprietary version

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

NMSS01

SAR Page Replacement Instructions

Proprietary Version	
Old Page	**Replacement Page(s)**
SAR Cover Page, Rev. 7	SAR Cover Page, Rev. 8
TOC-4	TOC-4
TOC-7	TOC-7
1-8	1-8
2-28	2-28
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2-30	2-30
2.10.5-11	2.10.5-11
2.10.5-11F	2.10.5-11f
None	2.10.5-11g
2.10.5-21	2.10.5-21
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Non-proprietary Version	
Old Page	**Replacement Page(s)**
SAR Cover Page, Rev. 7	SAR Cover Page, Rev. 8
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TOC-7	TOC-7
1-8	1-8
2-28	2-28
2-29	2-29
2-30	2-30
2.10.5-11	2.10.5-11
2.10.5-11F	2.10.5-11f
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SAR Page Replacement Instructions

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Enclosure 3 to TN E-29342

Changed Pages for the TN-40 Application Safety Analysis Report, Revision 8, Non-proprietary Version

NON-PROPRIETARY

AREVA

TRANSNUCLEAR, INC.

TN-40

TRANSPORTATION PACKAGING

SAFETY ANALYSIS REPORT

Revision 8

May 2010

7135 Minstrel Way, Suite 300 • Columbia, MD 21045

Chapter 2 - Appendices

LIST OF FIGURES

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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 or 0.475 kW per fuel assembly, including the BPRAs, and TPAs.

2.7.1 30 Foot Free Drop

In Appendix 2.10.8, the ADOC computer program is used to determine the impact limiter dimensions. The ADOC program is used to estimate the deformation of the impact limiters, the forces on the cask and the cask deceleration due to impact of the packaging on an unyielding surface. The full size impact limiter geometry and wood orientation are designed based on these results.

A one-third scale test impact limiter is fabricated to match the full size impact limiter geometry and wood properties requirements. Four drop orientations are performed to determine the deformations and decelerations of the impact limiters. The test results are used to establish the baseline g loads for the component structural evaluations.

The four drop tests on the one-third scale models of the TN-40 transport package impact limiters are documented in Appendix 2.10.9. For the slapdown drop case, the second impact (combined transverse g load and rotational g load) is a more severe impact to the components than the first impact. Therefore the reported g load for the slapdown is based on the second impact. The maximum g loads for the 90° end drop, 0° side drop, CG over corner drop, and 20° slapdown are as follows:

30 Foot Drop Orientation	***G* Load Measured by Testing** (See Table 2.10.9-1 of Appendix 2.10.9)
90° End Drop	54 g Axial
0° Side Drop	51 g Transverse
CG Over Corner Drop	34 g Axial
20° Slapdown (Second Impact)	58 g(1), 62 g(2)

(1) The g load measured at this location represents the maximum combined transverse and rotational g load for the basket structural analysis due to the slapdown drop case.
(2) The maximum combined g load at the top end of the cask body (at the outer surface of the cask lid).

The effect of low temperature (-20 °F) on the tested impact limiter is not available due to lost test data. However, based on the similar design, TN-68 impact limiter testing [30], chilling the impact limiter wood to -20 °F will increase the g load by 14%. This is based on the measured g load from the TN-68 testing that is approximately 75 g (-20 °F) and the maximum g load predicted by ADOC that is approximately 66 g (room temperature) (75/66 ≈ 14%). An increase of 15% in the g loads from testing is used to bound the low temperature effect on the wood properties.

A. Cask Body G Loads

Based on the above description, the following table summarizes the baseline g loads for the cask body structural evaluations. First, the g loads are multiplied by the factor due to low temperature effect and then these g loads are increased by additional factors for use as cask bounding baseline g loads.

Baseline G Loads for Cask Body Structural Analyses

30 Foot Drop Orientation	Bounding Test G Loads	(-20 °F) Low Temperature Factor	Bounding Baseline G Loads Used for Cask Body Structural Analyses
90° end drop	54 g axial	1.15 x 54 = 62	68 g (axial)
CG over corner drop	34 g axial	1.15 x 34 = 39	41 g (axial)
0° side drop	51 g transverse	1.15 x 51 = 59	75 g side drop analysis bounds both side drop and slapdown drop
20° slapdown (second impact)	62 g transverse	1.15 x 62 = 71	

B. Basket G Loads

To establish the baseline g loads, the g loads from the test at basket locations are multiplied by the appropriate dynamic load factors and factors due to low temperature effect (if appropriate).

Baseline G Loads for Basket Structural Analysis

Ambient Condition	*Drop Orientation*	*Basket Cross Section Location*	*Max G Load from Test (g)*	*DLF Factor(1)*	*Low Temperature Effect(2)*	*Baseline G Load (g)*
100 °F	*Side drop*	*Mid*	*51*	*1.08*	*-*	*55*
	Slapdown	*Top/Bot*	*58(3)*	*1.08*	*-*	*63*
	End drop	*Uniform*	*54*	*1.08*	*-*	*58*
-20 °F	*Side drop*	*Mid*	*51*	*1.08*	*1.15*	*63*
	Slapdown	*Top/Bot*	*58(3)*	*1.08*	*1.15*	*72*
	End drop	*Uniform*	*54*	*1.08*	*1.15*	*67*

(1) Dynamic load factor, see Appendix 2.10.6
(2) Wood property low temperature effect
(3) Test g load at basket location

C. Fuel Drop G Loads

For the fuel drop analyses, the side drop and slapdown orientation, baseline g loads are established by first multiplying the g loads from the test by the appropriate dynamic load factors and factors due to low temperature effect. These g loads are then increased by additional factors and are used as the baseline g loads. Dynamic analysis using testing time history is used in the end drop analysis. The baseline g loads for the fuel drop analyses are listed in the following table.

Baseline G Loads for Fuel Rod Structural Analysis

Drop Orientation	Bounding Test G Loads	Load Factor	Bounding Baseline G Load Used in the Fuel Rod Analysis
End drop	(1)	(1)	(1)
Side drop	51 g	51 x 1.10(2) x 1.15(3) = 65 g	75 g side drop analysis bounds both side drop and slapdown drop
Slapdown	58 g	58 x1.10(2) x 1.15(3) = 73 g	

(1) The acceleration time history curve from the TN40 1/3 scale end drop test (Figure 2.10.9-24) was used. Since the test model was 1/3 of the original size, all of the acceleration values are scaled by 1/3 and all of the times are scaled by a factor of 3. Furthermore, for the -20 °F temperature effect, the acceleration values were increased by 15% and the time values were decreased by 15%.
(2) Dynamic load factor, see Appendix 2.10.7
(3) Wood property low temperature effect

D. Cask Body Structural Analysis

The cask body stress evaluations are described in Appendix 2.10.1. The stress analyses were performed before the 1/3 scale impact limiter testing *was* done; therefore g loads used in Appendix 2.10.1 are based on the ADOC estimation.

Elastic analyses are used for all the cask body drop analyses in Appendix 2.10.1. Therefore, in order to calculate the stresses due to the bounding baseline g loads resulted from the testing, the load combinations as described in Table 2-17 are performed as follows:

- calculated load combination stresses based on the individual load stresses calculated in Appendix 2.10.1
- calculated the new load combination stresses by increasing the g values vs. g values used in the earlier calculations
- results of these new load combination stresses are listed in Tables 2-18 and 2-19

2.7.1.1 End Drop

The TN-40 cask body end drop stress analysis performed in Appendix 2.10.1 is based on 1 g. Linear elastic analysis is used for all the cask body structural analysis; the calculated stresses can be ratioed to match the bounding baseline g values. The stress results from the end drop were increased by the factor of baseline g values (68 g) to g values (1 g) used in the calculations.

These increased stress values are used in the end drop load combinations as indicated in Table 2-17 (combination numbers A1 to A4). In all cases, bolt pre-load effects and fabrication stresses are included. For the hot environment condition, 100 psig internal pressure, and impact load cases are combined. For the cold environment evaluation, 25 psig external pressure, and impact load cases are combined.

Table 2-18 lists the maximum nodal combined stress intensities ($P_L + P_B + Q + F$) for the bottom and lid end drop under hot environment conditions and cold environment conditions based on the baseline g values.

From Table 2-18, the maximum nodal stress intensity is 21.36 ksi and occurs at the inner shell due to cold load combination for drop on lid end. The membrane allowable is 45.5 ksi (P_m); therefore the minimum factor of safety is 2.13. Note that this stress intensity (21.36 ksi) corresponds to a nodal stress intensity ($P_L + P_B + Q + F$) value that is conservatively compared with the Code membrane (P_m) allowable.

2.7.1.2 C.G. Over Corner Drop

The TN-40 cask body CG over corner drop stress analysis performed in Appendix 2.10.1 is based on the 32 g axial and 14 g transverse. Linear elastic analysis is used for all the cask body structural analysis; the calculated stresses can be ratioed to match the

45° Orientation

Pressure for 1 g = p cos45°
= 1.109 × 0.7071 = 0.7842 psi
Pressure for 200 g = 222 × 0.7071 = 157 psi

60° Orientation

Horizontal pressure for 1 g = p sin60°
= 1.109 × 0.866 = 0.9604 psi
Horizontal pressure for 200 g = 222 × 0.866 = 192 psi

Vertical pressure for 1 g = p cos60°
= 1.109 × 0.5 = 0.5545 psi
Vertical pressure for 200 g = 222 × 0.5 = 111 psi

The accelerations applied in each run are as follows.

Orientation	Inertial Load (g)	a_x (g)	a_y (g)	a_z (g)
0°	200	200	0	0
30°	200	173.21	0	-100.00
45°	200	141.42	0	-141.42
60°	200	100.00	0	-173.21
90°	200	0	0	-200

The pressure load distributions for the 0°, 30°, 45°, 60°, and 90° analyses are shown in Figures 2.0.5-23 through 2.10.5-27.

2.10.5.3.3 Buckling Analysis and Results

A maximum load of 200 g was applied to each analysis. The automatic time stepping option AUTOTS was activated. This option lets the program decide the actual size of the load sub-step for a converged solution. The last load step with a converged solution is the buckling load of the model.

The ANSYS input, buckling loads, and factors of safety for 0°, 30°, 45°, 60°, and 90° side drops are summarized in Table 2.10.5-11. Displacement patterns, at the last converged sub-step (buckling load) for the five cases are shown in Figures 2.10.5-28 through 2.10.5-32. It may be seen that the displacements are not excessive at the last converged load step.

A linear elastic side drop analysis of the basket is performed using ANSYS. The maximum compressive force in the bottom most fuel compartment wall is calculated to be 42,530 lb. Since the basket is modeled as an 8 in. section, the unit load in the fuel compartment wall is 42,530/8 = 5,316 lb/in.

Conclusion

The maximum compressive load for *the* 75 g accident side drop load is 5,316 lb/in. The buckling load from the test specimen is 9,219 lb/in. Therefore, the factor of safety against buckling of the panel is 1.73 (9,219/5,316). Based on the test, it can be concluded that the basket can withstand up to 130 g (75 x 1.73) before reaching the buckling load.

2.10.5.6 Summary

Nonlinear analyses with bilinear material properties and small deflections were performed in ANSYS for the critical azimuth side drop orientations to determine the membrane and membrane plus bending stresses in all basket components. It was shown that the stresses at 75 g for a 30 foot drop are below allowable stress limits. For these analyses, steel tubes, including the intermediate aluminum plates, are connected together in the out-of-plane direction so that they will bend in unison under surface pressure or other lateral loading to simulate through-the-thickness support provided by the Boral® plates.

The same model was used to determine the critical buckling load for the basket, except large displacement and stress stiffening options were used. The buckling analyses are reported in Section 2.10.5.3. From these analyses a minimum *buckling load* of 88.54 *g* was determined. In addition, several sensitivity analyses were performed (Section 2.10.5.5) to investigate the effect of modeling assumptions and geometrical imperfections.

The first sensitivity analysis investigated the effect of connecting the steel tubes and the intermediate aluminum plates in the out-of-plane direction. The steel tubes were allowed to separate from each other to correctly simulate the buckling behavior. The results showed that even though the buckling loads when the basket is oriented in 0 and 90 degrees direction are lower than previously calculated, the minimum *buckling load* of *88.54 g* is still bounding.

The second sensitivity analysis investigated the effect of initial imperfections. An initial imperfection was applied to all steel tubes and the buckling load was calculated for the bounding basket orientation. The results showed that the initial imperfection does not have an effect on the calculated buckling load.

Furthermore, compression tests were performed to determine the basket panel strength during side impact (Section 2.10.5.5.3) *at room temperature as well as elevated temperature*. The results showed a safety factor of 1.73 with respect to the design load of 75 g.

ANSYS buckling analyses performed in Section 2.10.5.3 and Section 2.10.5.5 for an 8.0 inch sector assumes temperatures at the hottest section for the 100 °F ambient conditions. The minimum calculated buckling load of 88.5 g provides sufficient safety *factors for all loading conditions (basket baseline g loads are provided in Section 2.7.1 of Chapter 2) except for the slapdown impact when the ambient condition is -20 °F. For the -20 °F ambient conditions, the top and bottom portions of the basket are at lower temperatures than the temperatures used in the buckling analyses and lower temperature will increase the buckling load. The average temperatures in the basket periphery for each condition are provided in the table below. The temperature dependent material properties for SA-240 Gr. 304 and SB-209 6061-T651 at these temperatures are interpolated from data provided in Table 2.10.5-1. It is seen that the Young's modulus for SA-240 Gr. 304 and SB-209 6061-T651 increase by 2.2% and 4.4%, respectively, when the temperature decreases from 330 °F to 210 °F. Also the yield strength for SA-240 Gr. 304 and SB-209 6061-T651 increase by 15.7% and 31.0%, respectively, when the temperature decreases from 330 °F to 210 °F.*

The effect of basket temperature on the buckling load is evaluated using the results from limit load tests presented in Section 2.10.5.5.3. The limit load tests were performed at room temperature (70 °F) as well as elevated temperatures (365 to 529 °F). It is seen that because of the higher Young's modulus and yield strength at lower temperatures, the load at collapse for the tests performed at room temperature is much higher then the load at collapse for higher temperature. Using the test results from Section 2.10.5.5.3:

Average load at collapse at room temperature: 13,777 lb/in.

Average load at collapse at elevated temperature: 10,858 lb/in.

Average elevated temperature: 433 °F

Room temperature: 70 °F

The buckling load is 27% higher for the room temperature tests than at elevated temperature. Assuming a linear relationship the buckling load would increase by 9.4% for a 126 °F decrease in temperature. Therefore, the adjusted buckling load at 210 °F for the -20 °F ambient condition is 96.9 g (88.54 x 1.094).

Therefore, the safety factors for buckling load with its respective g load are:

Ambient Condition	Drop Orientation	Average Temperature in the Basket Periphery[1] (°F)	Average Temperature in the Basket Periphery used in the Analysis (°F)	Lowest Buckling G Load (g)	Baseline G Load (g)	Safety Factor
100 °F	Side drop	336	336	88.5	55	1.61
	Slapdown	312	336	88.5	63	1.40
-20 °F	Side drop	234	336	88.5	63	1.40
	Slapdown	210	210	96.9	72	1.35

(1) The average temperatures in the basket periphery are calculated from the ANSYS results files generated in the Section 3.4 NCT thermal analysis.

It is concluded from the above analyses and tests that the basket is structurally adequate when subjected to the loads resulting from a hypothetical accident condition drop of 30 feet.

Table 2.10.5-11
Fuel Basket Buckling Analysis Results

Basket Side Drop Orientation	Maximum Load used in Analyses			Last Converged Load (g)
	Maximum Acceleration (g)	Vertical Pressure (psi)	Horizontal Pressure (psi)	
0°	200	222	0	145.44
30°	200	192	111	88.54
45°	200	157	157	92.54
60°	200	111	192	92.54
90°	200	0	222	115.15

CHAPTER 3
THERMAL EVALUATION

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A personnel barrier prevents access to the outer surfaces of the cask body. The barrier, which consists of a stainless steel mesh attached to stainless steel tubing, will enclose the cask body between the impact limiters, and have an open area of approximately 80%.

Several thermal design criteria are established for the TN-40 to ensure the package meets all its functional and safety requirements. These are:

- Containment of radioactive material and gases is a major design requirement. Seal temperatures must be maintained within specified limits to satisfy the NCT leak tight containment requirement. A maximum temperature limit of 536 °F (280 °C) is set for the metallic seals (double metallic O-rings) in the containment vessel lid [11].
- To maintain the stability of the neutron shield resin, a maximum allowable NCT temperature of 300 °F (149 °C) is set for the neutron shield [15].
- In accordance with 10 CFR 71.43(g) [1] the maximum temperature of accessible package surfaces in the shade is limited to 185 °F (85 °C).
- Maximum fuel cladding temperature limits of 400 °C (752 °F) for NCT and 570 °C (1058 °F) for HAC are set for the fuel assemblies with an inert cover gas, as concluded in *r*eference [14].
- A temperature limit of 230 °F is set for wood to prevent excessive reduction in structural properties at elevated temperatures [17].

The NCT ambient temperature range is –20 to 100 °F (-29 to 38 °C) per 10 CFR 71.71(b) [1]. In general, all the thermal criteria are associated with maximum temperature limits and not minimum temperatures. All materials can be subjected to the minimum environment temperature of -40 °F (-40 °C) without adverse effects, as required by 10 CFR 71.71 (c)(2) [1].

The TN-40 thermal analysis is conservatively based on a maximum total heat load of 22 kW from 40 fuel assemblies and with a maximum of 0.55 kW per fuel assembly, even though the maximum total heat load is *19* kW. A peak power factor of 1.2 and an active length of 144 in. are considered for calculation of the decay heat profile of the fuel assemblies, as described in Section 3.4.1.3. A description of the detailed NCT analyses is provided in Section 3.4 and HAC analyses in Section 3.5. A summary of the NCT analysis results is provided in Table 3-1. The thermal evaluation concludes that for this thermal design heat load, all design criteria are satisfied.

3.4.1.5 Solar Heat Load

The total insolation for a 12-hour period in a day is 1475 Btu/ft^2 for curved surfaces and 738 Btu/ft^2 for flat surfaces not transported horizontally as per 10 CFR Part 71.71(c)(1) [1]. This insolation is averaged over a 24-hr period (daily averaged value) and applied as a constant steady state value to the external surfaces of the thermal models. A solar absorptivity of 0.30 is used for the painted outer surfaces of the packaging. Daily averaging of the solar heat load is justified based on the large thermal inertia of the TN-40 transport package.

3.4.2 Maximum Temperatures

Steady state thermal analyses are performed using the maximum decay heat load of 0.55 kW per assembly (22 kW total per cask), 100 °F ambient temperature and the maximum insolation. The temperature distribution within the cask body and basket is shown in Figure 3-10. The temperature distributions as calculated in the fuel assemblies and the neutron shield are shown in Figure 3-11. The temperature distributions within the impact limiter wood and basket rail*s* are shown in Figure 3-12. A summary of the calculated cask component temperatures is listed in Table 3-1.

3.4.3 Maximum Accessible Surface Temperature in the Shade

The analysis shows that without the personnel barrier, the maximum accessible cask surface temperature at 100 °F ambient in the shade is 208 °F and exceeds the limit of 185 °F. Therefore, a personnel barrier is required for transport operation at the maximum design basis transportation heat load of *19* kW per cask.

The accessible surfaces of the TN40 packaging consist of the personnel barrier and outermost vertical and radial surfaces of the impact limiters. The personnel barrier surrounds the cask body between impact limiters and has an open area of at least 80%.

The presence of the barrier has negligible effect on heat transfer between the cask surface and the environment. Convection is not affected because *the* distance between the barrier and *the* cask and the 80% open area of the barrier ensures *that* the airflow around the cask is not restricted. Radiant heat transfer to or from the cask surface is not significantly affected because the 80% opening of the barrier allows the cask to "see" the ambient environment and the distance between the cask and the barrier ensures the screen is very close to ambient temperature. Thus, the 20% of the barrier that the cask sees is also very close to the ambient temperature.

With the installation of the personnel barrier, the accessible packaging surfaces are limited to the impact limiter and the barrier outer surfaces. The NCT model with full insolation at 100 °F ambient temperature shows that the accessible surface temperature of the impact limiters does not exceed 115 °F. The maximum accessible surface temperature of the impact limiters and the maximum cask outer shell temperature at 100 °F ambient in the shade are 106 °F and 208 °F, respectively. *The accessible surface temperature of the packaging in the shade is calculated as follows.*

3.4.4 Minimum Temperatures

Under the minimum temperature condition of -40 °F (-40 °C) ambient, the resulting packaging component temperatures will approach -40 °F if no credit is taken for the decay heat load. Since the package materials, including containment structures and the seals, continue to function at this temperature, the minimum temperature condition has no adverse effect on the performance of the TN-40.

Temperature distributions at ambient temperatures of -40 °F and -20 °F with maximum decay heat and no insulation are determined. Table 3-2 lists the results of the analyses and the temperature distributions are shown in Figure 3-13.

3.4.5 Maximum Internal Pressure

The maximum NCT internal pressure is calculated in Chapter 4.

3.4.6 Maximum Thermal Stresses

The maximum NCT thermal stresses are calculated in Chapter 2.

3.4.7 Evaluation of Cask Performance for Normal Conditions of Transport

The thermal analysis of NCT demonstrates that the TN-40 cask design meets all applicable requirements, as documented in Table 3-1. The maximum temperatures calculated using conservative assumptions are well below specified limits. The maximum temperature of any containment structural component is less than 251 °F (122 °C). The maximum seal temperature (195 °F, 91 °C) during NCT is well below the 536 °F (280 °C) long-term limit specified for continued seal function. The maximum neutron shield temperature is below 300 °F (149 °C) and no degradation of the neutron shielding is expected. The predicted maximum fuel cladding temperature (495 °F, 257 °C) is well within *the* allowable fuel temperature limit of 752 °F (400 °C).

The maximum temperature differences across the gaps between various cask layers based on the results of the cask thermal analysis are as follows:

1 °F between the cask inner shell and the gamma shield shell
10 °F between the gamma shield shell and the neutron shield aluminum boxes
12 °F between the neutron shield aluminum boxes and the outer shell

The maximum total temperature difference across the cask shells resulting from the gaps is 23 °F.

Although a maximum heat load of 19 kW is allowed *during* transport of *the* TN-40 cask, the thermal performance is evaluated for *a* 22 kW heat load. This conservatism increases the margins of the maximum temperatures to the allowable limits significantly.

Thermal performance tests conducted on similar designs documented in [18] show that the thermal model considers adequately the insulating effect of the neutron shield and the gaps between multiple shells of the cask and bounds properly the uncertainties and imperfections expected in the fabrication of this type of cask.

Since most of the TN-40 casks to be transported are already loaded and since each cask is limited to *a* one-time shipment, the following measurements *taken* prior to shipment ensure the adequacy of the cask thermal performance and compliance with 10 CFR 71.85(a) in lieu of fabrication tests:

1) Perform a radiological survey over the cask outer surface as required in Chapter 7.0, *Section* 7.4 prior to transport. This survey assures compliance with 10 CFR 71.47 and 71.87 and will indicate the existence of any excessive defects, cracks, or void space through the cask shells.
2) Perform a thermal survey over the outer surface to determine the maximum cask outer shell, *cask lid, and cask bottom plate surface temperatures* prior to transport. *The measurements are to be performed along the approximate midsection of the outer shell at approximately 0°, 90°, 180°, and 270° orientations and along the centerlines of the cask lid and cask bottom plate. Comparison of these measured temperatures to the corresponding calculated temperatures using the analytical model of the cask described in this chapter indicates the sufficiency of the thermal performance prior to shipment. The model configuration will be adjusted so as to reflect the as-measured cask configuration and the cask heat load at the time of measurement. Note that the analysis described in this chapter is based on the cask in a horizontal orientation with impact limiters installed, 100 °F ambient with insolance, and a heat load of 22 kW.*
   An acceptance criterion of ±25 °F (temperature difference between calculated and measured values) is justifiable for the thermal survey, as discussed below.
3) Perform a leak test of the seals at *the* final destination *as required in Chapter 7.0, Section 7.2.* This test is an indicator that the containment properties of the cask *are* not affected by the thermal performance of the cask.

Based on the thermal evaluation of the cask, the margins of the fuel cladding and seal temperatures when compared to the allowable limits are:
752 - 495 = 257 °F for fuel cladding, and
536 -195 = 341 °F for seals.

If the gaps between various cask components assumed in the analysis are not equivalent to the gaps present in the as-fabricated cask, the measured temperatures will differ from the analytical values.

Considering the temperature differences across the gaps discussed previously, a 25 °F temperature difference between the measured and analytical values corresponds to cask gap sizes being more than double the size of the analytical gaps. Given the large critical component temperature margins, a 25 °F temperature change decreases the evaluated margin by less than 10% for containment critical components and their maximum temperatures remain well below the allowable limits.

3.5 Thermal Evaluation for Hypothetical Accident Conditions

The TN-40 cask is evaluated under the HAC sequence of 10 CFR 71.73 [1]. The top impact limiter protects the TN-40 cask lid containing the lid and port seals from the thermal accident environment. Analytical models are developed as discussed in Sections 3.5.2 and 3.5.3 to demonstrate that seal temperatures are below their material temperature limits during HAC.

3.5.1 Fire Accident Evaluation

The fire thermal evaluation is performed primarily to demonstrate the containment integrity of the TN-40. This is assured as long as the metallic seals in the lid remain below 536 °F and the cask cavity pressure is less than 100 psig. A full-length, 90 degree symmetric cask model as described in Section 3.4.1 is used for the evaluation. The model is modified to represent two crushed impact limiters as described in Section 3.5.3.

Reference [8] reports an average convective heat transfer coefficient of 4.5 Btu/hr-ft^2-°F for a railroad tank car fire test. The same parameter is utilized for the HAC fire accident evaluation.

3.5.2 Boundary Conditions for the HAC

The boundary conditions described in Section 3.4.1. are modified for the HAC fire. During the pre-fire and post-fire phases, convection and radiation from the external surface of the model replicate the NCT analysis (100 °F ambient). During the fire phase, a constant convective heat transfer coefficient of 4.5 Btu/hr-ft^2-°F is used. All gaps are removed during the fire and restored immediately after the fire. This assumption is conservative in that it ensures maximum heat transfer into the cask during the fire and minimum heat transfer from the cask during the post-fire cooling period. As required by 10 CFR 71.73 [1], a 30 minute 1,475 °F temperature fire with an emittance of 0.9 and a surface absorptivity of 0.8 is applied to the model. An emissivity of 0.9 and an absorptivity of unity are used for the cask external surfaces after the fire accident condition in order to bound the problem.

The sensitivity study that documents the effects of fire emissivity of 1.0 on thermal performance of the TN-40 cask is discussed in Appendix 3.7.3.

A detailed description of the model, including the method used to calculate the maximum fuel cladding temperature and the average cavity gas temperature, is provided in Section 3.4.1. The decay heat load used in this analysis corresponds to a conservative total heat load of 22 kW from 40 assemblies (0.55 kW/assy) with a peaking factor of 1.2 even though the design basis total heat load for transportation condition is *19* kW per cask.

3.5.3 Crushed Impact Limiter Models

In order to maximize the effect of the fire on cask components during and after the fire accident, the impact limiter finite element model developed in Section 3.4.1.2 is modified to reflect deformation due to a 30 foot drop. The maximum amount of crush experienced by the impact limiter in a given direction is assumed to occur everywhere on the limiter. Crushed impact limiter configurations based on side, corner and slap down drops are considered:

1. A crushed impact limiter corresponding to the side drop resulting in the shortest radial distance between the fire ambient and the cask surface. The maximum radial deformation of top and bottom impact limiters is 13.42 in. and 13.58 in., respectively. The impact limiters are thus modeled with a uniform

Table 3-1
NCT Component Temperatures in the TN-40 Package

Component	Normal Transport		
	Maximum (°F)	**Minimum* (°F)**	**Allowable Range(°F)**
Outer Shell	214	-40	**
Radial Neutron Shield	229	-40	-40 to 300
Inner Shell	251	-40	**
Basket Rail	257	-40	**
Basket *(Fuel Compartments)*	444	-40	**
Gamma Shield Shell	248	-40	**
Fuel Cladding	495	-40	752 max.
Impact Limiter Wood	224	-40	230
Cask Bottom Inner Plate	*234*	*-40*	**
Cask Lid	*192*	*-40*	**
*Vent and Drain Port Seal*******	*192*	*-40*	*-40 to 536*
Lid O-ring Seal****	195	-40	-40 to 536
Average Cavity Gas***	345	-40	N/A
Accessible Surface Temperature in Shade	134	-40	185 max

* Assuming no credit for decay heat and a daily average ambient temperature of -40 °F

** The components perform their intended safety function within the operating range.

*** A conservative value of 348 °F is used for calculating MNOP.

**** The elements between the cask inner shell and cask lid at radius (cylindrical x-coordinate) between 36.43" and 41.38" and height (cylindrical z-coordinate) between 164.55" and 171.55" represent the location of the lid seal in the model.

***** *The elements within the cask lid at radius (cylindrical x-coordinate) between 22.2" and 37.5" and height (cylindrical z-coordinate) between 171.49" and 173.70" represent the location of the vent and drain port seal in the model.*

Table 3-2
Temperature Distribution In The TN-40 Package
(Low Ambient Temperature, Max Decay Heat)

	Maximum Component Temperature (°F)	
Component	**-40 °F Ambient Temperature**	**-20 °F Ambient Temperature**
Outer Shell	88	106
Radial Neutron Shield	102	120
Gamma Shield Shell	123	140
Inner Shell	127	144
Basket Rails	133	151
Fuel Cladding	386	401
Cask Bottom Inner Plate	*109*	*126*
Cask Lid	*63*	*81*
*Vent and Drain Port Seal***	*63*	*81*
Lid O-ring Seal*	68	86
Basket (Fuel Compartments)	330	346

* The elements between the cask inner shell and cask lid at radius (cylindrical x-coordinate) between 36.43" and 41.38" and height (cylindrical z-coordinate) between 164.55" and 171.55" represent the location of the lid seal in the model.

** *The elements within the cask lid at radius (cylindrical x-coordinate) between 22.2" and 37.5" and height (cylindrical z-coordinate) between 171.49" and 173.70" represent the location of the vent and drain port seal in the model.*

Table 3-3
Maximum HAC Transient and Post-Fire Steady-State Maximum Temperatures During Fire Accident

Component	Maximum Transient Temperature (°F)	Maximum Post-Fire Steady-State Temperature (°F)*****	Allowable Range (°F)
Impact Limiter Outer Surface	1431 (end of fire)	*147*	**
Outer Shell Surface	1084 (end of fire)	*252*	**
Cask Bottom Inner Plate	*343 (One hour after fire)*	*260*	**
Cask Lid	*289 (4.2 hours after fire)*	*231*	**
*Vent and Drain Port Seal***	*284 (4.2 hours after fire)*	*230*	*536*
Lid O-ring Seal***	325 (one hour after fire)	*229*	536
Gamma Shield Shell	694 (end of fire)	*273*	**
Cask Rail / Shim	330 (one hour after fire)	*283*	**
Inner Shell	403 (end of fire)	*277*	**
Basket (Fuel Compartment)	480 (20 hours after fire)	474	**
Fuel Cladding	529 (26 hours after fire)	524	1058
Average Cavity Gas	387* (10.2 hours after fire)	*374*	**

* An average cavity gas temperature of 441 °F is considered for calculating of cavity gas pressure in Chapter 4 for conservatism.

** The components perform their intended safety function within the operating range.

*** The elements between the cask inner shell and cask lid at radius (cylindrical x-coordinate) between 36.43" and 41.38" and height (cylindrical z-coordinate) between 164.55" and 171.55" represent the location of the lid seal in the model.

**** *The elements within the cask lid at radius (cylindrical x-coordinate) between 22.2" and 37.5" and height (cylindrical z-coordinate) between 171.49" and 173.70" represent the location of the vent and drain port seal in the model.*

***** Thermal analysis results at 40 hours after end of fire conservatively used.

CHAPTER 5
SHIELDING EVALUATION

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approximately 50% indicating that the source terms are calculated using a very conservative representation of total cobalt content.

Using the same approach, the total cobalt content employed in the plenum, top and bottom end fitting regions is lower by a factor of 1.2, 1.4, and 1.9, respectively. This indicates that the dose rates calculated for the top and bottom ends of the package could potentially be under-estimated. However, this potential non-conservatism is offset by the use of a 30-year cooling time which is sufficient to reduce the cobalt source term in the end fittings. Further, any contribution from the active fuel region (conservative by a factor of 1.7) to the top and bottom end dose rates will serve to reduce this potential non-conservatism due to the use of a significantly higher cobalt source.

5.2.1 Axial Source Distribution

PWR plant operations data for over twenty 14 x 14 fuel assemblies with approximately 36 to 42 GWD/MTU burnup are averaged into a typical profile, shown as maximum profile in Figure 5-2. Also shown in Figure 5-2 is the axial profile from Reference [3] for 38-42 GWD/MTU burnup fuel. The third profile shown in Figure 5-2 is a bounding profile and used in this analysis. The bounding profile is also applicable for fuel with blankets at the ends of the active zone. The use of a bounding profile ensures that the most penalizing profile is utilized that includes and accounts for both blanketed and un-blanketed fuel. The bounding profile is derived from an evaluation that includes fuel assemblies with blankets. Enrichment used in the shielding analysis is minimum assembly average and accounts for blankets. Burnup used in shielding analysis is maximum assembly average and accounts for blankets.

Using a burnup profile accounting for the presence of blankets with low enriched or natural uranium increases dose rates on the side of the package.

The conservative axial profile containing 12 axial zones is utilized in the shielding evaluation. The top and bottom 17% of the assembly are divided into two zones each and the middle 66% are divided into 8 approximately equal zones. The peaking factors range from 0.700 at the bottom and top, to a maximum of 1.16 just below the middle. The gamma source is directly proportional to the burnup and the neutron source is proportional to the fourth power of the burnup. This data is presented in Table 5-12.

5.2.2 Gamma Source

The gamma source terms for the design basis spent fuel assembly is provided in Table 5-6. Table 5-6 presents the source terms for a Westinghouse 14 x 14 standard assembly with an initial enrichment of 2.35 wt%, maximum average burnup of 42,000 MWD/MTU, 24.4 year decay with a 13 year cooled TPA insert.

The gamma source spectra are presented in the 18-group structure consistent with the SCALE 27n-18γ cross section library. The conversion of the source spectra from the default ORIGEN-S energy grouping to the SCALE 27n-18γ energy grouping is performed directly through the ORIGEN-S code. The SAS2H/ORIGEN-S input file for this fuel assembly is provided in Section 5.6.

The gamma source for the fuel assembly hardware is primarily from the activation of cobalt. This activation contributes primarily to SCALE Energy Groups 36 and 37. The gamma source for the plenum region, the top fitting region and the bottom fitting region is provided in Table 5-6.

The spent fuel assemblies may contain irradiated fuel inserts (BPRA, TPA) which also provide a gamma source which is primarily from activated cobalt. The gamma source from a TPA corresponding to maximum host assembly burnup of 125,000 MWD/MTU and cooled for 13 years is shown in Table 5-7. This gamma source is added to the irradiated fuel gamma source in the plenum and top end fitting regions.

An axial burnup profile has been developed as discussed in Section 5.2.1 above. Table 5-12 provides design axial gamma peaking factors that were utilized in the MCNP shielding model.

- Numerous BECT combinations are utilized to determine the decay heat and radiation source terms. The dose rates from response functions are then estimated by adjusting the cooling times to ensure that the resulting dose rates are less than 9.8 mrem/hour. Cooling times are also adjusted such that the resulting decay heat values are below 525 watts per fuel assembly. The results of these calculations are shown in Table 5-9 where the minimum required cooling times are calculated as a function of burnup and enrichment.

The calculated dose rate and decay heat along with the cooling time are then utilized according to the steps above to determine the bounding radiological source term. The final design basis radiological source term was generated by adding the TPA source term to the fuel/hardware source term because the BPRA source term is already included. Note that the TPA source term was not explicitly included in the cooling time calculations. The cooling times calculated are reduced to a simplified look up table as a function of spent fuel parameters to summarize the loading parameters for the TN-40 transport package and are shown in Table 5-8.

Table 5-9 shows the results of the evaluation which define the spent fuel assembly cooling times to meet radiological and decay heat limits necessary for burnups ranging from 17 GWD to 45 GWD and enrichments between 2.0 wt% and 3.85 wt%. The TN-40 package containing fuel assemblies with parameters defined in this table will meet the dose rate and thermal criteria for transport. Table 5-10 shows the estimated dose rates and Table 5-11 shows the calculated decay heat corresponding to the cooling times shown in Table 5-9. All assemblies producing a decay heat of less than 21 kW per package or 525 watts per assembly are radiation (dose rate) limited. A fuel qualification table (FQT) for loading purposes based on this evaluation is provided in Table 1-2 (also shown in Table 5-8). The FQT is generated by conservatively rounding the cooling times shown in Table 5-9 up to the nearest value greater than 15 years.

The response function calculated dose rates shown in Table 5-10 are categorized into two sets: radiation limited and thermal limited. The dose rates for the "radiation limited" BE combinations are indicated by using a value of 10 mrem/hour while those for the thermal limited BE combinations are shown by using a value of less than 10 mrem/hour.

The FQT is developed such that any BECT combination shown in Table 5-9 that results in an estimated dose rate of 10 mrem/hour (rounded up) can be used in the shielding calculations. However, the bounding parameters are burnup of 42 GWD/MTU at the enrichment of 2.35 wt% U-235 and a cooling time of 24.4 years to calculate the design basis source terms.

In order to ensure that the burnup and enrichment combinations employed previously also remain bounding for a cooling time of 30 years, a simple calculation is performed as described below. Based on the fuel qualification shown in Table 5-8, it is sufficient to demonstrate that the source terms (and the resulting dose rates) calculated using an enrichment of 2.35 wt% U-235 and a burnup of 42 GWD/MTU (design basis) are greater than those calculated for the following two combinations:

1. *enrichment of 3.00 wt% U-235 and a burnup of 43 GWD/MTU (combination 1)*
2. *enrichment of 3.25 wt% U-235 and a burnup of 45 GWD/MTU (combination 2)*

All other combinations of burnup and enrichment are bounded by the above combinations since a uniform cooling time of 30 years is employed.

The neutron source terms for combination 1 are lower than the design basis by approximately a factor of 0.77 while the total primary gamma source terms are higher by approximately a factor of 1.05. The primary gamma dose rate estimated using the response function shown in Table 5-20 for combination 1 is slightly lower than that for the design basis.

The neutron source terms for combination 2 are lower than the design basis by approximately a factor of 0.81 while the total primary gamma source terms are higher by approximately a factor of 1.1. The primary gamma dose rate estimated using the response function shown in Table 5-20 for combination 2 is comparable (within 0.5%) to that for the design basis.

As discussed in Section 5.4, the neutron source term contribution to the total dose rate at the side is greater than 70%. Since the neutron source terms for the design basis is greater than that for these two configurations, it can be concluded that the design basis combination will result in bounding source terms for a cooling time of 30 years.

In summary, the source terms from the design basis combination are bounding for all fuel assemblies with acceptable burn-up enrichment combinations with cooling times greater than or equal to 30 years.

5.3 Model Specification

The Monte Carlo computer code MCNP [5] is used for calculating the gamma and neutron doses in this analysis. A more advanced version of the code MCNP [9] is also employed to determine the dose rates for the models that include tolerances as described in Section 5.4.

respectively. The results shown in Table 5-19 with the 30-year cooled fuel demonstrate that the dose rates (maximum 7.79 mrem/hr) are below the limits (10 mrem/hr) with sufficient margin for a 10 ft wide vehicle. *As shown in Section 5.2.5, the source term for the 30-year cooled fuel is determined using the bounding combination of enrichment and burnup. This ensures that the dose rates shown in Table 5-19 are design basis dose rates for 30-year cooled fuel.*

The dose rate analysis was performed using MCNP [5, 9]. Selected inputs for MCNP are included in Section 5.7.

5.5 Uncertainties and Conservatism in the Shielding Evaluation

The shielding evaluation described in Section 5.4 is based on a conservative representation of the geometry, material and source description. This section provides a description of the uncertainties associated with the shielding evaluation and demonstrates that the evaluation sufficiently covers these uncertainties.

Uncertainties can be due to 1) methodology – directly as a result of employing a computer program and interpretation of results and 2) modeling – use of geometrical, material and source representations. Since the calculated dose rates are based on results from source term and shielding calculations, this discussion extends to uncertainties from both the source term and shielding calculations.

The SAS2H/ORIGEN-S modules of the SCALE computer code package [1] were employed to determine the design basis source terms. The SCALE package and the 44 Group ENDF-B(V) cross section library has been extensively benchmarked to determine the uncertainties associated with the methodology. The SAS2H/ORIGEN-S prediction of the concentrations of the principal radioactive isotopes for both gamma and neutron source terms has been determined to be within 10% of measured data implying that the methodology uncertainty associated with SAS2H/ORIGEN-S is acceptably low. This indicates that the SAS2H code predictions are generally accurate and that the resulting source terms are appropriate, if not conservative. Further, comparison of surface dose rates from numerous measurements for various TN casks also confirm the applicability of the source term calculation methodology.

The calculational uncertainty in the models is more than compensated for by the use of conservative modeling parameters. The main source of conservatism is in the specification of fuel assembly hardware materials (including cobalt content) and irradiation history.

The fuel qualification methodology described in Section 5.2.5 also includes conservatisms in the employment of response functions. The selection of the bounding BECT parameters for calculating source terms among the various candidate BECT combinations (since they all result in approximately the same dose rate at 2 m) has an uncertainty that is less than 2% since the estimated dose rates are within 0.1 mrem/hr of each other. This is more than compensated for by the use of a scaling factor of 1.05 in the response function calculations.

The use of a Monte Carlo method for shielding calculations will result in a methodology uncertainty that is based on the standard deviation associated with the results. The dose rates from the calculations have associated standard deviations within 2%. In general, the MCNP methodology uncertainty is within 5% and is directly related to the acceptance criteria of the results. The modeling uncertainty is accounted for by employing the minimum geometrical tolerances for major components important to shielding. Additional conservatism in the use of the minimum guaranteed hydrogen content in the neutron resin ensures that the uncertainty of the resulting dose rates is minimized.

In general, the uncertainty associated with the MCNP methodology is more than compensated for with the conservatisms in the shielding models.

In summary, the methodology uncertainty associated with the source term and shielding calculations is within 10% and is directly related to the acceptance criteria (code benchmarking) of these methods – SAS2H/ORIGEN-S and MCNP. The source term and shielding analyses have been performed to ensure that the uncertainty associated with modeling (geometry, materials and source specification) is bounded by the conservatisms discussed above.

In addition, the dose rates shown in Table 5-2, Table 5-18 and Table 5-19 include the effect of geometrical and material tolerances. Most importantly, an analysis of the expected inventory in Chapter 6 indicates that the spent fuel parameters (burnup and enrichment) employed in the source term calculations are conservative (such a fuel assembly does not exist). Finally, the dose rates calculated with a cooling time of 30 years (required by the criticality analysis) ensure that the calculated dose rates are well below the applicable regulatory limits (Table 5-19).

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

Normal Conditions of Transport	Package Contact Dose Rate mSv/hour (mrem/hour)			Closed Vehicle Surface mSv/hour[1] (mrem/hour)	2 Meters from Closed Vehicle Surface mSv/hour (mrem/hour)		
Radiation	**Top**	**Side**	**Bottom**		**Top**	**Side[2]**	**Bottom**
Gamma	0.069 (6.9)	0.35 (35)	0.059 (5.9)	0.23 (23)	-	0.031 (3.1)	-
Neutron	0.0004 (0.04)	0.38 (38)	0.0008 (0.08)	0.26 (26)	-	0.037 (3.7)	-
Total	0.069 (6.9)	0.73 (73)	0.060 (6.0)	0.49 (49)	<0.069 (<6.9)	0.068 (6.8)	<0.060 (<6.0)
Limit	2 *(200)*	10 (1000)	2 *(200)*	2 (200)	0.1 (10)	0.1 (10)	0.1 (10)

[1] Closed 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.

[2] Side dose rates are calculated with 30-year cooled fuel and 2 m from the edge of the impact limiters. The maximum dose rate due to 24.4-year cooled fuel at 2 m from one edge of the impact limiter is 0.0906 mSv/hour (9.06 mrem/hour). The maximum dose rate due to 30-year cooled fuel at 2 m from the edge of the 120" vehicle is 0.0779 mSv/hour (7.79 mrem/hour).

Hypothetical Accident Conditions[3]	1 Meter from Package Surface mSv/hour (mrem/hour)		
Radiation	**Top**	**Side[4]**	**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)

[3] The neutron shield and the impact limiters are removed.

[4] 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.

Table 5-4
Westinghouse 14 X 14 STD Fuel Assembly Hardware Characteristics

Item	Material	Average Mass (kg/assembly)
Fuel Zone		
Cladding	Zircaloy	83.4
Spacers	Inconel	5.37
Guide & Instrument Tubes	Stainless Steel[1]	7.74
Fuel-Gas Plenum Zone		
Cladding	Zircaloy	4.13
Springs	Stainless Steel	5.68
Guide & Instrument Tubes	Stainless Steel[1]	0.38
Spacer	Inconel	0.68
Top End Fitting Zone		
Top Nozzle	Stainless Steel	6.30
Hold Down Springs	Inconel	0.51
Bottom End Fitting Zone		
Bottom Nozzle	Stainless Steel	7.89
	Total	122.0

(1) The zircaloy guide and instrument tubes are modeled as stainless steel to include the effect of BPRAs in the source term calculations.

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. Check if the surface dose rates and the surface contamination 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 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.

8.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

8.1 Acceptance Tests

The following reviews, inspections, and tests shall be performed on the TN-40 packaging prior to initial transport. Many of these tests will be performed at the fabricator's facility prior to delivery of the cask to the utility for use. Tests will be performed in accordance with written procedures approved by Transnuclear, Inc. For the TN-40 casks that have been fabricated, loaded and used for storage under 10CFR72 requirements, use of acceptance tests performed during their fabrication are also acceptable.

8.1.1 Visual Inspection

Visual inspections are performed at the fabricator's facility prior to initial use to ensure that the packaging conforms to the drawings and specifications. The visual inspections include:

- cleanliness inspections,
- visual weld inspections as required by ASME Code [1],
- inspection of sealing surface finish, and
- dimensional inspections for conformance with the drawings included in Chapter 1 and *referenced* in the Certificate of Compliance.

The visual inspection includes verifying that all specified coatings are applied and the packaging is clean and free of cracks, pinholes, uncontrolled voids or other defects that could significantly reduce its effectiveness. To the maximum extent practical, weld inspection is performed in accordance with the applicable ASME code sections [1]. Dimensions and tolerances shown on the drawings provided in Chapter 1 are confirmed by measurements. The sealing surfaces on the flange, lid and covers are inspected to ensure that there are no gouges, cracks or scratches that could result in an unacceptable leakage.

Prior to shipping, the packaging will be inspected to ensure that it is in good physical condition. This inspection shall include verification that all accessible cask surfaces are free of grease, oil or other contaminants, and that all cask components are in an acceptable condition for use.

8.1.2 Structural and Pressure Tests

The structural analyses performed on the packaging are presented in Chapter 2. To ensure that the packaging can perform its design function, the structural materials are chemically and physically tested to confirm that the required properties are met. To the maximum extent practical, welding is performed using qualified processes and qualified personnel, according to the ASME Boiler and the Pressure Vessel Code [1]. Base materials and welds are examined in accordance with the ASME Boiler and Pressure Vessel code requirements. NDE requirements for welds are specified on the drawings

The leak test port (the overpressure port in the storage configuration) is closed by the overpressure transport cover with a single metallic seal. This flange and seal are not part of the containment boundary. The quick connect couplings in the vent and drain ports are not part of the containment boundary.

There are no valves or rupture discs on the TN-40 packaging containment.

8.2.4 Shielding

There are no periodic tests or inspections required for the TN-40 shielding. Radiation surveys will be performed of the package exterior to ensure that the limits specified in 10 CFR 71.47 are met prior to shipment.

8.2.5 Thermal

There are no periodic tests or inspections required for the TN-40 heat transfer components. However, a thermal survey of the cask exterior *as described in Section 3.4.7* will be performed prior to transport to ensure the thermal performance of the packaging.