NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

DRAFT CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE	PAGES
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2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

Transnuclear, Inc.
7135 Minstrel Way, Suite 300
Columbia, Maryland 21045

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Transnuclear, Inc., application dated August 7, 2006

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5. (a) Packaging

(1) Model No.: TN-40

(2) Description: For descriptive purposes, all dimensions are approximated nominal values. Actual dimensions with tolerances are as indicated on the Drawings.

The TN-40 is designed to transport up to 40 Pressurized Water Reactor (PWR) spent nuclear fuel assemblies discharged from the Prairie Island Nuclear Generating Plant (PINGP). These assemblies have been stored prior to shipment in the TN-40 package used as a dry storage cask at PINGP under SNM-2506. The TN-40 packaging consists of a basket assembly, a containment vessel, a package body which also functions as the gamma shield and neutron shield, and impact limiters. A transport frame, which is not part of the packaging, is used for tie-down purposes.

The containment vessel components consist of the inner shell and bottom inner plate, shell flange, lid outer plate, lid bolts, penetration cover plates and bolts (vent and drain), and the inner metallic seals of the lid seal and the vent and drain seals. The containment vessel prevents leakage of radioactive material from the cask cavity. It also maintains an inert atmosphere (helium) in the cask cavity. The overall containment vessel length is approximately 170.5 in. with a wall thickness of 1.5 in. The cylindrical cask cavity has a nominal diameter of 72.0 in. and a length of 163 in.

Double metallic seals are used for the lid closure. To preclude air in-leakage, the cask cavity is pressurized with helium above atmospheric pressure. The cask is accessed via draining and venting ports. Double metallic seals are utilized to seal these two lid penetrations. The over-pressure (OP) port provides access to the volumes between the double seals in the lid and cover plates for leak testing purposes. The OP port cover is not part of the containment boundary.

5.(a)(2) Description (Continued)

The carbon steel packaging body, which also functions as the gamma shielding, is around the inner shell and the bottom inner plate of the containment vessel. The 8.0 in. and 8.75 in. gamma shield completely surround the containment vessel shell and bottom plate, respectively. A 6.0 in. thick shield plate is also welded to the inside of the 4.5 in. thick lid outer plate.

Radial neutron shielding is provided by a borated polyester resin compound surrounding the gamma shield shell. The total radial thickness of the resin and aluminum is 4.50 in. The array of resin-filled containers is enclosed within a 0.50 in. thick outer steel shell. The aluminum container walls also provide a path for heat transfer from the gamma shield shell to the outer shell. A pressure relief valve is mounted on top of the resin enclosure to limit the possible internal pressure increase under hypothetical accident conditions.

The basket structure consists of an assembly of stainless steel cells joined by a fusion welding process and separated by aluminum and poison plates which form a sandwich panel. The panel consists of two aluminum plates separated by a poison plate. The aluminum plates provide the heat conduction paths from the fuel assemblies to the cask inner plate. The poison material provides the necessary criticality control. The opening of the cells is 8.05 in. x 8.05 in. which provides a minimum of 1/8 in. clearance around the fuel assemblies. The overall basket length (160.0 in.) is less than the cask cavity length to allow for thermal expansion and fuel assembly handling.

The impact limiters consist of balsa wood and redwood blocks encased in stainless steel plates. The impact limiters have an outside diameter of 144 in., and an inside diameter of 92 in. to accommodate the cask ends. The bottom limiter is notched to fit over the lower trunnions. The impact limiters are attached to each other using tie rods. The impact limiters are also attached to the outer shell of the cask with bolts. Each impact limiter is provided with fusible plugs that are designed to melt during a fire accident, thereby relieving excessive internal pressure. Each impact limiter has lifting lugs for handling, and support angles for holding the impact limiter in a vertical position during storage. An aluminum spacer is placed on the cask lid prior to mounting the top impact limiter to provide a smooth contact surface between the lid and the top impact limiter.

The nominal external dimensions, with impact limiters, are 261 in. long by 144 in. wide. The total weight of the package is 271,500 pounds (lbs.).

5.(a)(3) Drawings

The packagings are fabricated and assembled in accordance with the Transnuclear, Inc., Drawing Nos.:

10421-71-1, Rev. 4.
10421-71-2, Rev. 2, sheets 1 and 2.
10421-71-3, Rev. 1.
10421-71-4, Rev. 0.
10421-71-5, Rev. 0.
10421-71-6, Rev. 0.
10421-71-7, Rev. 1.

5.(a)(3) Drawings (Continued)

10421-71-8, Rev. 0.
10421-71-9, Rev. 0.
10421-71-10, Rev. 0.
10421-71-40, Rev. 1.
10421-71-41, Rev. 1.
10421-71-42, Rev. 0.
10421-71-43, Rev. 0.
10421-71-44, Rev. 0.

5.(b) Contents

(1) Type, form, and quantity of material

The characteristics of the contents of the TN-40 packaging are limited to the following.

I. Fuel shall be unconsolidated.

II. Fuel shall be limited to the following fuel types with specifications depicted in Table 1-1 of this certificate:

i. Exxon 14X14 Standard,
ii. Exxon 14x14 High Burnup,
iii. Exxon 14X14 TOPROD,
iii. Westinghouse (WE) 14X14 Standard, and
iv. Westinghouse 14X14 OFA.

III. Fuel shall only have been irradiated at the PINGP Unit 1, cycles 1 through 16 or Unit 2, cycles 1 through 15.

IV. The fuel assemblies from Unit 1, Region 4, i.e., assemblies identified as D-01 through D-40, are not authorized contents.

V. Fuel may include burnable poison rod assemblies (BPRAs) provided:

i. the BPRAs have cooled for a minimum of 25 years, and
ii. the maximum exposure of the BPRA(s) shall be 30,000 Megawatt-Days per Metric Ton of Uranium (MWd/MTU).

VI. Fuel may include thimble plug assemblies (TPAs) provided:

i. the minimum cooling time of the TPAs is 25 years,
ii. the maximum exposure of the TPA(s) shall not exceed 125,000 MWd/MTU, and

5.(b)(1) Contents – Type, form, and quantity of material (Continued)

iii. only TPAs that do not have water displacement rods extending into the active fuel may be loaded into the cask.

VII. The combined weight of a fuel assembly and any BPRA or TPA shall not exceed 1330 lbs.

VIII. The combined weight of all fuel assemblies, BPRAs, and TPAs in a single cask shall not exceed 52,000 lbs.

IX. The fuel shall not be a Damaged or Oxidized Fuel Assembly; a Damaged or Oxidized Fuel Assembly is:

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

X. The number of assemblies in the container shall not exceed 40.

XI. The assembly average burnup shall be greater than or equal to the burnup calculated according to the following equations:

$B = -1,259.8X^2 + 20,242X - 23,617$; for fuel assemblies with BPRA insertions during depletion
$B = -366.95X^2 + 14,770X - 17,200$; for fuel assemblies without BPRA insertions during depletion

Where:
B = Burnup (MWd/MTU),
X = Initial enrichment (weight percent (wt%) U-235)

XII. The minimum cooling time for the contents is 30 years. Various combinations of minimum assembly average enrichment and maximum assembly average burnup prior to transport shall be in accordance with Table 1-2 in this certificate.

XIII. The maximum decay heat per fuel assembly shall not be more than 0.475 kW and 19 kW per cask including the BPRAs and TPAs as bounded by Table 1-2 in this Certificate of Compliance.

XIV. The boron-10 (B-10) in the neutron poison plates in the cask must be enriched to at least 90% and uniformly distributed in the plates with a minimum areal density of 10 mg/cm^2.

XV. Integral Fuel Burnable Absorber is not an authorized content.

5.(b)(1) Contents – Type, form and quantity of material (Continued)

XVI. Fuel assemblies with the following irradiation history shall be authorized for transport:

i. The power density shall not exceed 14 MW/Assembly,
ii. The minimum moderator density shall be 0.705 g/cm^3,
iii. The maximum moderator temperature shall be 584 K (592°F),
iv. The maximum fuel temperature shall be 901 K (1,162°F), and
v. The maximum average soluble boron concentration shall be 600 parts per million.

XVII. The maximum length of the assembly axial blankets is 6.2 inches.

XVIII. The maximum cooling time of the spent fuel shall not exceed 200 years. If the cooling time of the spent fuel assemblies in the cask exceeds 200 years at the time of shipment, a reassessment of criticality safety of the cask shall be performed prior to transport.

Table 1-1 Fuel Assembly Specifications[1,2]

	Fuel Assembly Type				
Fuel Characteristics	Exxon 14x14 Standard	Exxon 14x14 High Burnup	Exxon 14x14 TOPROD	WE 14x14 Standard	WE 14x14 OFA
Max. Active Fuel Length (in.)	144	144	144	144	144
Max. Number of Fuel Rods per Assembly	179	179	179	179	179
Max. Fuel Rod Pitch (in.)	0.556	0.556	0.556	0.556	0.556
Min. Clad Thickness (in.)	0.0300	0.0310	0.0295	0.0243	0.0243
Min. Clad Outer Diameter (OD) (in.)	0.424	0.417	0.426	0.422	0.400
Clad Material	Zr-4	Zr-4	Zr-4	Zr-4	Zr-4
Max. Pellet OD (in.)	0.3565	0.3565	0.3505	0.3659	0.3444
Min. Guide/Instrument Tube OD (in.)	16@0.541 1@0.424	16@0.541 1@0.424	16@0.541 1@0.424	16@0.539 1@0.422	16@0.528 1@0.4015
Max. Guide/Instrument Tube Inner Diameter (in.)	16@0.507 1@0.374	16@0.507 1@0.374	16@0.507 1@0.374	16@0.505 1@0.3734	16@0.490 1@0.3499
Max. Assembly and BPRA Length (in.)	161.3	161.3	161.3	161.3	161.3
Max. Assembly Width (in.)	7.763	7.763	7.763	7.763	7.763

Table 1-1 Fuel Specifications (Continued)

	Fuel Assembly Type				
Fuel Characteristics	Exxon 14x14 Standard	Exxon 14x14 High Burnup	Exxon 14x14 TOPROD	WE 14x14 Standard	WE 14x14 OFA
Maximum MTU/Assembly	0.380	0.380	0.380	0.410	0.380
Maximum Initial Assembly Average Enrichment (wt% U-235)	3.85	3.85	3.85	3.85	3.85
Maximum Assembly Average Burnup (MWd/MTU)	45,000 (see Table 1-2[2])	45,000 (see Table 1-2)	45,000 (see Table 1-2)	45,000 (see Table 1-2)	45,000 (see Table 1-2)
Minimum Cooling Time (years)	30 (see Table 1-2)	30 (see Table 1-2)	30 (see Table 1-2)	30 (see Table 1-2)	30 (see Table 1-2)

Notes:
1. Pre-irradiated dimensions used in the design analyses and may be verified against as-built records.
2. Table 1-2 is located in this certificate.

Table 1-2 Required Minimum Cooling Time for Spent Fuel Assemblies[1,2,3,4]

	Minimum Assembly Average Initial Enrichment (wt.% U-235)								
Maximum Assembly Average Burnup (GWd/MTU)	2	2.25	2.35	2.75	3	3.25	3.4	3.6	3.85
17	30	30	30	30	30	30	30	30	30
18	30	30	30	30	30	30	30	30	30
19	30	30	30	30	30	30	30	30	30
20	30	30	30	30	30	30	30	30	30
21	30	30	30	30	30	30	30	30	30
22	30	30	30	30	30	30	30	30	30
23	30	30	30	30	30	30	30	30	30
24	30	30	30	30	30	30	30	30	30
25	30	30	30	30	30	30	30	30	30
26	30	30	30	30	30	30	30	30	30
27	30	30	30	30	30	30	30	30	30
28	30	30	30	30	30	30	30	30	30

Table 1-2 Required Minimum Cooling Time for Spent Fuel Assemblies (Continued)

	Minimum Assembly Average Enrichment (wt.% U-235)								
Maximum Assembly Average Burnup (GWd/MTU)	2	2.25	2.35	2.75	3	3.25	3.4	3.6	3.85
29			30	30	30	30	30	30	30
30			30	30	30	30	30	30	30
31			30	30	30	30	30	30	30
32			30	30	30	30	30	30	30
33			30	30	30	30	30	30	30
34			30	30	30	30	30	30	30
35			30	30	30	30	30	30	30
36			30	30	30	30	30	30	30
37			30	30	30	30	30	30	30
38			30	30	30	30	30	30	30
39			30	30	30	30	30	30	30
40			30	30	30	30	30	30	30
41			30	30	30	30	30	30	30
42			30	30	30	30	30	30	30
43					30	30	30	30	30
44						30	30	30	30
45						30	30	30	30

Notes:

1. For fuel characteristics that fall between the assembly average enrichment values in Table 1-2 of this certificate, use the next lower enrichment, and next higher burnup to determine minimum fuel cooling time.
2. Fuel assemblies that were located in the Rod Cluster Control Assembly control bank D position during Unit 1 cycle 1 and Unit 2 cycle 1 shall have a minimum cooling time of greater than 35 years.
3. The assembly average enrichment and the assembly average burnup are the enrichment and burnup averaged over the fuel assembly, including the initial enrichment of the axial blankets.
4. Fuel assemblies with a maximum average burnup and a minimum average enrichment for which no cooling time is specified in the table are not authorized contents.

5.(c) Criticality Safety Index: 0.0

6. In addition to the requirements of Subpart G of 10 CFR Part 71:

   (a) The package must be prepared for shipment and operated in accordance with the "Operating Procedures" in Chapter 7 of the application, as supplemented.

6. Additional Requirements to Subpart G of 10 CFR Part 71 (Continued)

(b) Each packaging must be acceptance tested and maintained in accordance with the "Acceptance Tests and Maintenance Program" in Chapter 8 of the application, as supplemented.

(c) Fuel must meet the content requirements of Condition 5(b) and have been placed into the package under SNM-2506 prior to May 2011. Any additional reuse of the packaging after unloading the original content is prohibited.

(d) This certificate applies to only the 29 TN-40 packages already fabricated and in use at the PINGP under its 10 CFR Part 72 license.

(e) As part of the preparation for transport, the 48 as-installed 1.375-in. diameter SA-320 Grade LA43 closure lid bolts shall be replaced by the SA-540 Grade B23 Class 1 bolts of the same configuration.

(f) As part of the preparation for transport, a 0.75-in. thick by 71.75-in. diameter aluminum spacer shall be installed between the cask lid and the payload.

(g) As part of the preparation for transport, the metallic seals used in the package and the vent and drain ports shall be replaced and tested to a maximum allowable leak rate of 1.0×10^{-4} ref-cm^3/sec (at a sensitivity of 5.0×10^{-5} ref-cm^3/sec or less) in compliance with ANSI N14.5.

(h) As part of the preparation for transport, the user shall perform a leak rate test of the entire containment boundary prior to shipment, with an acceptance criterion of 1.0×10^{-4} ref-cm^3/sec (at a sensitivity of 5.0×10^{-5} ref-cm^3/sec or less) in compliance with ANSI N14.5. This test is necessary to meet the intent of the containment acceptance tests.

(i) A temperature survey shall be performed on each loaded package and the results compared to calculated outer shell temperatures from SAR thermal model analysis in Section 3.4.7 of the FSAR, with appropriate adjustments for decay heat and ambient temperature. The temperature difference between calculated and measured values shall not exceed ±25°F.

(j) For casks previously loaded under 10 CFR Part 72 to comply with 10 CFR 71.85(a), a neutron and a gamma dose rate survey must be performed over the entire surface of the cask. Total dose rates from these surveys must meet the regulatory limits in 10 CFR 71.47. This comprehensive measurement requirement is necessary to meet the intent of the shielding acceptance and periodic tests.

(k) For casks that are configured for storage, the operating procedures prescribed in Section 7.4 of the FSAR must be used to convert the storage configuration to transportation configuration of the package.

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6. Additional Requirements to Subpart G of 10 CFR Part 71 (Continued)

   (l) Prior to shipment, the user shall evaluate all mitigation actions and corrective actions that have been performed within any Aging Management Plan under SNM-2506 for storage, and verify that important to safety structures and components conform to the design specified in Condition 5(a)(3).

7. Transport by air is not authorized.

8. Packagings must be marked with Package Identification Number USA/9313/B(U)F-96.

9. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

13. The personnel barrier shall be installed at all times while transporting a loaded overpack.

14. Expiration date: May 30, 2016.

REFERENCES

Transnuclear, Inc., application dated: August 7, 2006.

As supplemented: November 8, 2006; June 29, August 24, and September 11, 2007; May 8, and August 29, 2008; December 10, 2009; March 6, March 15, March 30, April 23, May 7, June 18, July 30, August 26, September 15, 2010, December 22, 2010, and May 2x, 2011.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Michael D. Waters, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: ____________________.