NRC FORM 618 (8-2000) 10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9068	7	71-9068	USA/9068/B( )F	1	OF	3

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)*

U.S. Department of Energy
Washington, DC 20585.

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

U.S. Department of Energy application dated November 7, 1991, as supplemented

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.: BCL-2

(2) Description

A steel encased, lead shielded shipping package. The packaging is provided with a recessed, plug-type lid and gasketed, bolted closure; lifting and tie-down devices; and a drain line penetration. Containment for the contents is provided by an inner can assembly or by material in special form. The packaging has dimensions, weight, and shielding as follows:

Exterior height, in.	18.2
Exterior diameter, in.	15.5
Cavity height, in.	5.25
Lead shielding, in.	4.5
Loaded weight, lb.	1,360 (incl. 110-lb. skid)

(3) Drawings

The packaging is constructed in accordance with Battelle Memorial Institute Drawing No. BCL2-01, Sheets 1 and 2, Rev. D.

The inner can assembly is constructed in accordance with Battelle Memorial Institute Drawing No. BCL2-47, Rev. B.

5. (b) Contents

(1) Type and form of material

Byproduct material, source material, and special nuclear material in solid metal or oxide form, which is packaged within the inner can assembly specified in Item 5(a)(3), or which meets the requirements of special form radioactive material.

(2) Maximum quantity of material per package

Not to exceed 200 watts decay heat, and

(i) Fissile material not to exceed 50 grams U-235 equivalent mass.

(ii) Fissile material not to exceed 2,000 grams U-235 equivalent mass.

(c) Transport Index for Criticality Control (Criticality Safety Index)

Minimum transport index to be shown on label for nuclear criticality control:

For contents described in 5(b)(1) and limited in 5(b)(2)(i): 0.4

For contents described in 5(b)(1) and limited in 5(b)(2)(ii): 100

6. Plutonium in excess of 20 curies per package must be in the form of metal, metal alloy or reactor fuel elements.

7. The U-235 equivalent mass must be determined by the following method:

U-235 equivalent mass equals U-235 mass plus 1.75 times U-233 mass plus 1.60 times Pu mass.

8. At the time of delivery of the loaded package to a carrier for transport, the package contents must be (1) dry (contents of inner can assembly must not decompose up to a temperature of 750°F) and the fissile material unmoderated (H to X atomic ratio less than 2), and (2) so limited that the dose rate will not exceed 10 millirem per hour at one meter from the external surface of the package.

9. The maximum gross weight of the cavity contents must not exceed 20 pounds (inner can assembly, radioactive material, etc.)

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

    (a) Each package shall be maintained in accordance with Section 8.0 of the application, as supplemented.

    (b) The package shall be prepared for shipment and operated in accordance with Section 7.0 of the application, as supplemented.

11. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR §71.12.

12. Expiration date: May 31, 2007.

REFERENCES

U.S. Department of Energy application dated November 7, 1991.

Supplements dated: April 10, 1992; January 27, 1997; and May 29, 2002.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: June 3, 2002

NRC FORM 618
(3-96)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIALS PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. PACKAGE IDENTIFICATION NUMBER	d. PAGE NUMBER	e. TOTAL NUMBER PAGES
9069	11	USA/9069/B( )F	1	3

2. PREAMBLE

a. This certificate is issued to certify that the 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)*

Department of Energy
Washington, DC 20585

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION:

Westinghouse Electric Corporation
application dated October 30, 1981,
as supplemented.

c. DOCKET NUMBER 71-9069

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.: MO-1

(2) Description

Steel overpack consisting of a 12 gauge outer shell (45" x 47" x 206") and a 10 gauge inner shell (max. 37" x 37" x 186"). The volume between the shells is filled with a shock-and-thermal-insulating material consisting of rigid polyurethane foam. The upper and lower sections of the overpack are secured by 12 ratchet binders and 12 high strength 5/8" latch pins. The fuel assemblies are held in place within the overpack by a strongback and adjustable clamping assembly (shock mounted). Neutron absorber plates are located between the fuel assemblies. The package is equipped with lifting, tie-down and pressure relief devices. Gross weight of the package is 8,600 pounds.

(3) Drawings

The packaging is constructed in accordance with Westinghouse Electric Corporation Drawing No. 1581F50, Sheets 1 and 2, Rev. 1. Fuel rod container is constructed in accordance with Westinghouse Electric Corporation Drawing No. C5650D55, Rev. 1.

5.(b) Contents

(1) Type and form of material

Uranium dioxide as stainless steel or aluminum clad unirradiated rods of the following specifications:

	SST Clad	AL Clad
Pellet diameter (max), in	0.446	0.406
Rod diameter (nom), in	0.476	0.475
Fuel length (max), in	70.0	61.0
^{235}U enrichment (max), w/o	4.02	2.5

(2) Maximum quantity of material per package

Two inner containers as described in 5(a)(3) containing not more than a total of 70 kilograms U-235.

(c) Transport Index for Criticality Control

Minimum transport index to be shown on label for nuclear criticality control: 1.6

6. Two (2) neutron absorber plates consisting of 0.19" thick, full length, stainless steel containing 1.3 percent minimum boron or 0.19" thick OFHC copper must be installed between the active area of the fuel assemblies.

7. Fuel rods must be closely packed in the fuel rod container on no more than an equivalent metal-to-metal square lattice. Partially loaded fuel rod containers must be fitted with a minimum of three, equally spaced blocks, of which the noncombustible portion of the blocks and the method by which they are secured must assure that the rods are maintained on no more than an equivalent metal-to-metal square lattice within the fuel rod container.

8. Each fuel assembly must be unsheathed or must be enclosed in an unsealed, polyethylene sheath which will not extend beyond the ends of the fuel assembly. The ends of the sheath must not be folded or taped in any manner that would prevent the flow of liquids into or out of the sheathed fuel assembly. Alternatively, the fuel assembly may be enclosed in an elongated plastic bag or sheath along its full length. At the bottom end of the fuel assembly, the bag will be cut off or folded back to assure that the entire cross section of the lower end of the assembly is unobstructed. When the folding is used, the portion of the sheath that is folded back will be cinched with tape near its end to hold it in place, and the length will be such that when the assembly is loaded in the packaging, the folded sheath will be clamped in place in at least two grid locations. The top end of the bag may be gathered together and taped closed. However, the top end then will be slit on all four sides. The slits will run perpendicular to the axis of the assembly and will extend the inner distance between the top nozzle pads and spring clamps (approximately 60 percent of the length of each side). The slits will be made in a plane near that formed by the top of the pads and clamps.

9. 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 Chapter 6.0 of the application.

   (b) Each packaging must meet the acceptance tests and maintenance program of Chapter 7.0 of the application.

10. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR §71.12.

11. Expiration date: December 31, 2002.

REFERENCES

Westinghouse Electric Corporation application dated October 30, 1981.

Westinghouse supplements dated January 24, 1992 and December 31, 1996.

Department of Energy supplements dated: April 2 and June 14, 1984; December 24, 1996; and November 7 and December 10, 1997.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Cass R. Chappell

Cass R. Chappell, Chief
Package Certification Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: December 16, 1997

NRC FORM 618 (8-2000) 10 CFR 71	CERTIFICATE OF COMPLIANCE FOR RADIOACTIVE MATERIAL PACKAGES			U.S. NUCLEAR REGULATORY COMMISSION
1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE / PAGES
9070	15	71-9070	USA/9070/B(U)	1 OF 3

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)*

Packaging Technology, Inc.
4507-D Pacific Highway East
Tacoma, WA 98424-2633

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

VECTRA Technologies, Inc. application dated July 21, 1994, as supplemented.

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.: N-55

(2) Description

A low carbon steel overpack filled with rigid polyurethane foam. The containment vessel is a 55-gallon steel drum. The overpack is a right circular cylinder 48 inches high by 32 inches diameter with a 34-1/2-inch high by 24-inch diameter cavity. The 18 or 20-gauge galvanized steel shell is filled with 3-pound per cubic foot rigid polyurethane foam. The inner shell is molded fiberglass. Closure of the upper and lower (lid and body) sections of the overpack is provided by four toggle clamps, and a neoprene gasket at the stepped joint between the two sections. Four lugs are provided for lifting. The steel drum is minimum 18-gauge steel with a minimum 14-gauge lid and a gasket. Closure of the drum is by way of a 12-gauge locking ring with dropped forged lugs and a 5/8-inch diameter bolt and lock nut. The package gross weight is approximately 750 pounds.

(3) Drawing

The packaging is constructed in accordance with Nuclear Packaging, Incorporated Drawing No. X-60-200D, Rev. C, or X-60-200D-SP, Rev. J.

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9070	15	71-9070	USA/9070/B(U)	2	OF	3

(b) Contents

(1) Type and form of material

(a) Radioactive material in the form of dewatered, solid or solidified materials meeting the requirements of low specific activity material, contained in steel drums.

(b) Radioactive material meeting the requirements of special form radioactive material, contained in steel drums.

(c) Radioactive material in the form of solid metal pieces or activated solid metal components, contained in steel drums.

(2) Maximum quantity of material per package

Greater than Type A quantities of radioactive material. Fissile material contents not to exceed the generally licensed mass limits as specified in 10 CFR §§71.18 and 71.22. Plutonium in excess of 20 curies per package must be in the form of metal, metal alloy or reactor fuel elements, or must meet the requirements of special form radioactive material. Internal decay heat not to exceed 3 watts.

6. The maximum weight of contents, including drum, not to exceed 550 pounds.

7. The steel drum must be in accordance with Appendix 1.3.2 of the supplement dated October 20, 1994.

8. The drum must be securely positioned in the overpack.

9. Contents must be securely positioned so that protrusions will not puncture the drum under normal or accident conditions.

10. The lifting lugs must be rendered inoperable for tie-down during transport.

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

(a) The package must meet the Acceptance Tests and Maintenance Program of Chapter 8.0 of the application; and

(b) The package shall be prepared for shipment and operated in accordance with the Operating Procedures in Chapter 7.0 of the application.

(c) Authorization by this certificate only applies to the N-55 package S/N PT-001, fabricated by Packaging Technology on January 21, 1999.

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9070	15	71-9070	USA/9070/B(U)	3	OF	3

12. The packaging authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.12.

13. Expiration date: January 31, 2005.

REFERENCES

VECTRA Technologies, Incorporated, application dated July 21, 1994.

Supplements dated: August 22 and October 20, 1994; and February 6, 1998.

Transnuclear, Inc., supplement dated February 5, 1998, and December 3, 1999.

Packaging Technology, Incorporated, letter dated April 11, 2000.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: November 6, 2000

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9081	13	71-9081	USA/9081/B()	1	OF	3

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)*

   Duratek
   140 Stoneridge Drive
   Columbia, SC 29210

   b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

   Chem-Nuclear Systems, Inc., application dated November 24, 1987, as supplemented.

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.: CNS 1-13C

   (2) Description

   A steel encased lead shielded shipping cask. The packaging is a steel double-walled, lead-filled circular cylinder. A steel, plug-type, lead-filled lid is attached with twelve, 1-1/4" bolts; and a silicone gasket. Outer steel sheets are separated from the cask walls with small diameter wires. The lead shielding is 5" in the sides, 6" in the base and 5-3/4" in the lid. Two bolted-on steel lugs are for lifting only. The lid has a steel U-bar for lifting. The cavity drain line is closed with a plug. The cask is 39" in diameter and 68-1/2" long. The cavity is 28-1/2" in diameter and 54" long. The package weight is about 26,000 pounds.

   (3) Drawings

   The packaging is constructed in accordance with Chem-Nuclear Systems, Inc., Drawing Nos. C-110-E-0005, Sheets 1, 2, and 3, Rev. 7; and C-112-B-0006, Rev. A.

(b) Contents

Type, form, and maximum quantity of material per package

(i) Greater than Type A quantity of byproduct material as solid metal. Decay heat not to exceed 600 watts; or

(ii) Decay heat not to exceed 5 watts, and:

Process solids, either dewatered, solid, or solidified in a secondary sealed container meeting the requirements for low specific activity material; or Solid reactor components in secondary containers, as required, that meet the requirements for low specific activity material.

6. (a) For any package containing water and/or organic substances which could radiolytically generate combustible gases, determination must be made by tests and measurements or by analysis of a representative package such that the following criteria are met over a period of time that is twice the expected shipment time:

(i) The hydrogen generated must be limited to a molar quantity that would be no more than 5% by volume (or equivalent limits for other inflammable gases) of the secondary container gas void if present at STP (i.e., no more than 0.063 g-moles/ft^3 at 14.7 psia and 70°F); or

(ii) The secondary container and cask cavity must be inerted with a diluent to assure that oxygen must be limited to 5% by volume in those portions of the package which could have hydrogen greater than 5%.

For any package delivered to a carrier for transport, the secondary container must be prepared for shipment in the same manner in which determination for gas generation is made. Shipment period begins when the package is prepared (sealed) and must be completed within twice the expected shipment time.

(b) For any package containing materials with radioactivity concentration not exceeding that for low specific activity material, and shipped within 10 days of preparation, or within 10 days after venting of drums or other secondary containers, the determination in (a) above need not be made, and the time restriction in (a) above does not apply.

7. Shoring must be provided to minimize movement of contents during accident conditions of transport.

8. Maximum gross weight of the contents, secondary container, and shoring is limited to 5,000 pounds.

9. The lid closure to the cask shall be secured by twelve, SA-354, Type BD, 1-1/4"-7 UNC x 2-1/4" long bolts torqued to 320 ft-lbs ± 10% (lubricated) or 420 ft-lbs ± 10% (dry).

10. The cask shall be delivered to a carrier dry and the cavity drain line shall be sealed with appropriate sealant applied to threads of pipe plug.

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9081	13	71-9081	USA/9081/B()	3	OF	3

11. Prior to each shipment, the leak test described in Section 8.2 of the application must be performed. No package is to be delivered to a carrier for transport with a detectable leak using the method of Section 8.2.

12. Radiation measurements shall be made to determine that the dose rate does not exceed 30 mrem/hr at one meter from the surface of a dry loaded cask.

13. Prior to each shipment, the lift lugs must be removed from the packaging.

14. The contents described in 5(b)(ii) shall be transported on a motor vehicle, railroad car, aircraft, inland water craft, or hold or deck of a seagoing vessel assigned for sole use of the licensee.

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

    (a) The package shall be prepared for shipment and operated accordance with the Operating Procedures in Chapter 7 of the application.

    (b) The package shall be maintained in accordance with the Maintenance Program in Chapter 8 of the application.

16. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR §71.12.

17. Expiration date: January 31, 2003.

REFERENCES:

Chem-Nuclear Systems, Inc. application dated November 24, 1987.

Supplements dated: November 24, 1992, October 31, 1997, July 28, 1999, January 5, 2000, and April 23, 2001.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

E. William Brach

E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: July 25, 2001

NRC FORM 618
(3-96)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIALS PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. PACKAGE IDENTIFICATION NUMBER	d. PAGE NUMBER	e. TOTAL NUMBER PAGES
9098	9	USA/9098/B( )	1	3

2. PREAMBLE

a. This certificate is issued to certify that the 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)*

Department of Energy
Washington, DC 20585

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION:

Department of Energy application dated March 31, 1998, as supplemented.

c. DOCKET NUMBER 71-9098

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 Nos.: CI-20WC-2 and CI-20WC-2A

(2) Description

Steel encased, wooden outer protective jackets with a uranium shielded cask and inner steel containment vessel. The protective jackets are constructed of disks and rings of plywood, which are glued together and reinforced with steel rods. The protective jackets are contained within an 18-gauge steel drum. The shielded casks have depleted uranium shields encapsulated in steel with a gasketed and bolted flange closure with six, 3/8"-16 UNC-2A x 3/4" long bolts. The inner containment vessel is a 2.73" OD x 5.56" long 416 stainless steel, gasketed and threaded container. The gross weight of the packages is about 400 pounds.

Model No.	CI-20WC-2	CI-20WC-2A
Protective jackets overall dims, in	24-1/4x22x28-3/4	24-1/4x18x26-1/4
U(D) thickness, in	2	1.8
Cavity dims, in	3.1x6H	3.1x6H

Page 2 - Certificate No. 9098 - Revision No. 9 - Docket No. 71-9098

(3) Drawings

The packagings are constructed in accordance with Cintichem Inc. Drawing Nos.:

Model No. CI-20WC-2
101259, Rev. D and 100964, Rev. H

Model No. CI-20WC-2A
101354, Rev. G and 101326, Rev. F

Inner Containment Vessel
101401, Rev. C

(b) Contents

(1) Type and form of material

(i) Mo-99/Tc-99 in normal form as solids or liquids.

(ii) I-131 in normal form as liquids.

(2) Maximum quantity of material per package

(i) For contents described in 5(b)(1)(i):
1,000 curies

(ii) For contents described in 5(b)(1)(ii):
200 curies

6. Contents must be contained within the inner containment vessel specified in 5(a)(3).

7. 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 (PO-05, PO-06 and PO-08) of the application.

b. The package must be maintained in accordance with the maintenance procedures (PO-06) of the application.

c. The inner containment vessel neoprene O-ring seal must be replaced prior to each shipment.

d. Prior to each shipment, the loaded inner containment vessel must show no leakage when tested to a sensitivity of at least 1×10^{-5} std-cm^3/sec.

e. The inner containment vessel must be leak tested within 12 months prior to use in accordance with the leak test procedure (PO-07) of the application. The inner containment vessel must show no leakage greater than 1×10^{-7} std-cm^3/sec.

Page 3 - Certificate No. 9098 - Revision No. 8 - Docket No. 71-9098

8. Structural parts of the packaging which could be used as tie-down devices must be securely covered or locked during transport in such a manner as to prevent their use for that purpose.

9. The packages authorized by this certificate are hereby approved for use under the general license provisions of 10 CFR §71.12.

10. Expiration date: May 31, 2004.

REFERENCES

Department of Energy application dated March 31, 1998.

Supplements dated: November 4, 1998, and April 19, 1999.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

E. William Brach

E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: 6/7/99

NRC FORM 618
(3-96)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIALS PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. PACKAGE IDENTIFICATION NUMBER	d. PAGE NUMBER	e. TOTAL NUMBER PAGES
9099	9	USA/9099/B(U)F-85	1	2

2. PREAMBLE

a. This certificate is issued to certify that the 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)*

U.S. Department of Energy
Washington, DC 20585

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION:

ATR Fresh Fuel Shipping Container
Safety Analysis Report, INEL-94/0275,
January 27, 1999, as supplemented

c. DOCKET NUMBER 71-9099

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.: ATR

(2) Description

The inner container is a right parallelepiped, 69-7/16 inches x 26-13/16 inches x 6-15/16 inches, constructed of 3/4-inch plywood, covered with 16-gauge steel. The top and bottom are lined with high density polyethylene foam and with a 0.020-inch cadmium plate. Wood spacers covered with sponge rubber and with a 0.020-inch thick cadmium plate provide separation for four fuel assemblies. Positive closure is provided by a continuous hinge, and two wire sealed hinge pins provide access.

The inner container is enclosed within an overpack, 73-15/16 inches x 31-3/4 inches x 11-3/16 inches, constructed of 1-inch plywood, framed by steel angle members and covered with 18-gauge steel. Aluminum, honeycomb impact limiters are fixed to the ends of the overpack. Positive closure of the overpack is provided by four hinge pins which are secured in place using 1/16-inch diameter cotter pins. The package weight is approximately 853 pounds.

(3) Drawings

The packaging is fabricated in accordance with EG&G Idaho, Inc., Drawing No. 445721, Sheets 1, 2, and 3; and EG&G Idaho, Inc., Drawing No. 445722, Sheets 1 and 2.

(b) Contents

(1) Type and form of material

Unirradiated ATR fuel elements. Each element contains 19 formed fuel plates, clad in Aluminum 6061. Each element contains a maximum of 1,100 grams of U-235 in uranium that is enriched to a maximum of 94 wt% in the U-235 isotope.

(2) Maximum quantity of material per package

Up to four (4) unirradiated ATR fuel elements. Total U-235 content not to exceed 4,400 grams per package.

Page 2 - Certificate No. 9099 - Revision No. 9 - Docket No. 71-9099

(c) Transport Index for Criticality Control

Minimum transport index to be shown on label for nuclear criticality control: 4.2

6. The contents must be maintained within its compartment and the active fuel length must be completely within the region of the cadmium covered spacers. Wood spacers may be used to accomplish this.

7. 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.

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

8. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.12.

9. Expiration date: January 31, 2004.

REFERENCES

ATR Fresh Fuel Shipping Container Safety Analysis Report, INEL-94/0275, January 27, 1999.

Supplements dated: February 18, 1999 and April 27, 2000.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: June 15, 2000

NRC FORM 618
(3-96)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIALS PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. PACKAGE IDENTIFICATION NUMBER	d. PAGE NUMBER	e. TOTAL NUMBER PAGES
9102	8	USA/9102/B( )	1	2

2. PREAMBLE

a. This certificate is issued to certify that the 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)*

Neutron Products, Inc.
22301 Mt. Ephraim Road
Dickerson, MD 20842

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION:

Neutron Products, Inc., application dated August 31, 1977, as supplemented.

c. DOCKET NUMBER 71-9102

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.: NPI-20WC-6

(2) Description

A steel encased lead shielded cask contained within a wooder overpack. The cask is 24 inches in diameter with a 3//8-inch thick steel spherical shell and a cavity formed by an 8-1/4-inch ID by 3/8-inch thick steel tube. Positive closure of the shielded cask is accomplished by bolted end covers at each end of the cavity. The overpack is a 48-inch diameter, 12 gauge steel body with a wooden shell 38-1/4 inches in height made of 3/4-inch thick plywood sheets glued together and reinforced by 16 steel tie rods and 32 lug screws. Positive clsoure of the overpack lid is accomplished by 3 equally spaced bracket assemblies with attached chains and held together with a 3/8-inch by 4-inch welded ring. The maximum packge gross weight is 6,000 pounds.

(3) Drawings

The Model No. NPI-20WC- packaging is constructed in accordance with Neutron Products, Inc. Drawing No. 240010, Rev. C. The overpack is constructed in accordance with Neutron Products Inc. Drawing Nos. 240160, Sheet 1, Rev. None and 240160, Sheet 2, Rev. A.

(b) Contents

(1) Type and form of material

Cobalt 60, as sealed sources which meet the requirements of special form radioactive material.

(2) Maximum quantity of material per package

The maximum activity must not exceed 9,500 curies. The maximum internal decay heat must not exceed 150 thermal watts.

Page 2 - Certificate No. 9102 - Revision No. 8 - Docket No. 71-9102

6. The contents must be secured in the drum assembly (Item 11) so as to restrict movement in any direction to less than 0.25 inch by lead, steel or tungsten full diameter plugs and spacers.

7. The gross weight of the packaging must not exceed 6,000 pounds and the inner shielded cask shall be snug-fitting within the wooden overpack.

8. 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 the supplement dated September 21, 1993.

   (b) The package must meet the Acceptance Test and Maintenance program in the supplement dated September 21, 1993.

9. The packaging authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR §71.12.

10. Expiration date: October 31, 2003.

REFERENCES

Neutron Products, Inc., application dated August 31, 1977.

Supplements dated: February 6, 1978; July 31, 1985; August 2 and September 7, 1988; September 21, 1993; and September 23, 1998.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Cass R. Chappell

Cass R. Chappell, Chief
Package Certification Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: October 16, 1998

NRC FORM 618
(3-96)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIALS PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. PACKAGE IDENTIFICATION NUMBER	d. PAGE NUMBER	e. TOTAL NUMBER PAGES
9107	6	USA/9107/B(U)	1	2

2. PREAMBLE

a. This certificate is issued to certify that the 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)*

Amersham Corporation
40 North Avenue
Burlington, MA 01803

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION:

Technical Operations, Inc. application dated December 30, 1982, as supplemented.

c. DOCKET NUMBER 71-9107

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.: 771

(2) Description

The Model No. 771 shipping container is designed for use as a source changer, storage container and Type B Shipping Container for radiographic sources. The capacity of the container is 110 curies of cobalt 60. The container will accept certain Tech/Ops wire mounted radiographic sources which have been deemed to meet the requirements of special form. The Model No. 771 Source Changer measures 23 inches long, 24 inches wide and 20 inches high. The radioactive source assembly is housed in a Zircalloy or Titanium "S" tube. The "S" tube is surrounded by depleted uranium metal as shielding material. The depleted uranium shield assembly is encased in a steel housing. The void space between the depleted uranium shield assembly and the inner container is filled with a rigid polyurethane foam. The gross weight of the container is 690 pounds.

5. (3) Drawings

The packaging is constructed in accordance with the Technical Operations, Inc. Drawing No. 77190, Sheets 1 through 6, Rev. 0.

(b) Contents

(I) Type and form of material

Cobalt 60 as sealed sources that meet the requirements of special form radioactive material.

(2) Maximum quantity of material per package

110 curies

Page 2 - Certificate No. 9107 - Revision No. 6 - Docket No. 71-9107

6. Source assemblies for use in this packaging are limited to those assemblies as identified in Section 1-3 of Technical Operations, Inc. application dated December 30, 1982.

7. Nameplates shall be fabricated of materials capable of resisting the fire test of 10 CFR Part 71 and maintaining their legibility.

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

   (1) The package must be prepared for shipment and operated in accordance with the Operating Procedures in the supplement dated April 29, 1998; and,

   (2) Each package must be maintained and acceptance tested in accordance with the Acceptance Tests and Maintenance Program in the supplement dated April 29, 1998.

9. The packaging authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR §71.12.

10. Expiration date: June 30, 2003.

REFERENCES

Technical Operations, Inc., application dated December 30, 1982.

Supplements dated February 16, April 13, and April 28, 1993; and April 29, 1998.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Cass R. Chappell

Cass R. Chappell, Chief
Package Certification Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: June 18, 1998

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE	PAGES
9132	14	71-9132	USA/9132/B(M)F	1 OF	5

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)*
U.S. Department of Energy
Washington, DC 20585

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION
Nuclear Packaging, Inc. application
dated April 22, 1985, as supplemented

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.: T-3

(2) Description

A stainless steel and lead shielded irradiated fuel shipping package (cask). The cask is a right circular cylinder with upper and lower steel encased rigid polyurethane foam (20 lb/ft3) impact limiters. The overall dimensions are 213.2 inches in length and 2 inches in diameter. The cask without the impact limiters measures 177.2 inches in length and 26.44 inches in diameter.

The outer cask shell is comprised of a 1-inch thick stainless steel shell overlayed with a 10 gauge stainless steel cover. Between these two materials is a 0.08-inch diameter wire wrap, providing an air gap for additional thermal protection.

The inner shell (containment vessel) is a standard seamless stainless steel Schedule 40 pipe having an outside diameter of 8.625 inches with a nominal wall thickness of 0.322 inch. The annular space between the inner and outer shells is filled with lead having a thickness of approximately 8 inches.

Both the inner and outer shells are welded at each end to heavy steel closure plates with conical surfaces to assist in positioning and sealing. The containment vessel measures 147 inches in length by 7.981 inches in diameter.

The containment vessel is sealed at the bottom end with a 11.83-inch thick stainless steel plug with two Viton O-ring seals. The top end of the containment vessel is sealed with a 11.625-inch thick stainless steel plug with two Viton O-ring seals. The bottom plug is retained by a closure plate secured by eight, ½"-13UNC x 2-1/4-inch ASTM A320, Grade L7 socket head cap screws. The top plug is secured in place utilizing 16, ½"-13UNC x 1-3/4-inch ASTM A320, Grade L7 hex flange screws.

5.(a) (2)(continued)

No drain or vents penetrate directly into the containment vessel. A drain/vent line opens directly into the area between the two O-ring seals at each end of the cask (end plugs). During shipment, the lines are sealed with Viton O-ring sealed threaded fasteners.

The cask is provided with six trunions, four spaced 90 degrees apart at the top end and two spaced at 180 degrees apart at the bottom end of the cask. The cask is tied down at the forward and aft ends by means of a cradle and yoke assembly. The gross weight of the cask and contents is 38,200 pounds.

(3) Drawings

The packaging is constructed in accordance with Energy Research and Development Administration (ERDA) Drawing No. H-4-66230, Sheets 1, 3, 5, and 6, Revision No. 0, and Sheets 2 and 4, Revision No. 1. For payloads in spent fuel containers, the applicable drawings are DOE Drawing Nos. H-3-47474, Sheets 1 and 2, Revision No. 0, and H-4-66535, Revision No. 0, and Los Alamos Drawing No. 54Y-110854, Sheets 1 and 2, Revision No. B.

5.(b) Contents

Type, form, and maximum quantity of material per package

Irradiated, (a) mixed oxide (MOX) fuel pins and assemblies; (b) reactor fuel comprised of U-235 and/or Pu-239 oxides, carbides, nitrides, or metallic alloys; and (c) structural components. The minimum cooling time of each assembly and rod must be 90 days, and the cask may contain 1,400 thermal watts. Prior to irradiation, the fuel and structural components must have the following specifications:

	Type	Fuel Description*	Array Description	Maximum Fissile Package Loading	Pin Dimensions
(1)	217-Pin DFA assembly	31% PuO_2 - 69% UO_2 (natural U)	Hexagonal array w/pins at 0.26" center-to-center	11.2 kg	0.23" dia 36" active fuel length
(2)	217-Pin MOX fuel pins	50% max PuO_6 + $^{235}UO_2$ - remainder natural UO_2	Circular array groups of pins in seven compartments in 5" Schedule 5 Pipe	27.5 kg	0.23"-0.29" dia. 36" active fuel length

	Type	Fuel Description*	Array Description	Maximum Fissile Package Loading	Pin Dimensions
(3)	109-Pin MOX fuel pins	35% PuO_2 -65% UO_2 (86% U-235)	Circular array individual pins contained in 0.44" dia. tubes	26.2 kg	0.23"-0.29" dia. 36" active fuel length
(4)	55-Pin MOX fuel pins	35% PuO_2 -65% UO_2 (86% U-235)	Circular array individual pins contained in 0.625" dia. tubes	13.2 kg	0.23"-0.29" dia. 36" active fuel length
(5)	37-Pin MOX fuel pins	35% PuO_2 -65% UO_2 (86% U-235)	Circular array individual pins contained in 0.75" dia. tubes	8.9 kg	0.23"-0.29" dia. 36" active fuel length
(6)	42-Pin MOX	35% PuO_2 -65% UO_2 (86% U-235)	Circular array individual pins contained in 0.625" dia. tubes	10.1 kg	0.23"-0.29" dia. 36" active fuel length
(7)	40-Pin MOX fuel pins	35% PuO_2 -65% UO_2 (86% U-235)	Circular array individual pins contained in 0.625" dia. tubes	9.6 kg	0.23"-0.29" dia. 36" active fuel length
(8)	19-Pin MOX fuel pins	35% PuO_2 -65% UO_2 (86% U-235)	Circular array individual pins contained in 0.88" dia. tubes	4.6 kg	0.23"-0.29" dia. 36" active fuel length
(9)	PU compounds fuel pins (spent fuel containers)	50% PUX max-UX X=C,N, or 0 (94% U-235)	Unrestricted array individual pins contained in SS 5-inch Schedule 40 pipe	8.0 kg	Container cavity 5.047" dia. by 38.9" length
(10)	LAMPRE fuel pins (spent fuel container)	97.5% Pu max-X alloy X=Fe, Co or Cs	Circular array individual pins contained in 0.625" or 0.75" dia. steel tubes	8.0 kg	0.425" dia. 38" active fuel length

	Type	Fuel Description*	Array Description	Maximum Fissile Package Loading	Pin Dimensions
(11)	Structural components (incl. control assemblies)	Dosimetry foils	–	1.0 kg	–
(12)	24 max. Pins. U-Pu carbide fuel pins	85-94%(Pu-U)C -6 to 15% (Pu-U_2)C_3. Max 23% Pu, uranium is not enriched	Circular array; individual pins contained in 0.625-in. dia. tubes within 5-in. Schedule 40 pipe	3.0 kg	0.37" outer dia. 36" active fuel length
(13)	18 max. Pins. Sodium bonded (fuel-to-clad)	10% Zr-20% Pu max. Remainder U (U enriched to 40% max. (U-235)	Circular array; individual pins contained in 0.625-in. diam. tubes within 5-in. Schedule 40 pipe	1.9 kg	0.30" outer dia. 36" active fuel length

*All plutonium in the fuel types (1) thru (8) contains at least 10% Pu-240; fuel type (9) has no limit for PU-240; type (10) contains at least 6% PU-240.

5.(c) Transport Index for Criticality Control

Minimum transport index to be shown on label for nuclear criticality control: 100

6. Content 5.(b)(1) shown in AEC Drawing No. H-4-21500, Rev. 9, and ERDA Drawing No. H-4-66230, Sheet 5, Rev. 0.

Contents 5.(b)(2), (3), (4), and (5) must be contained within inner container Ident 69 described by ERDA Drawing Nos. H-4-66160, Sheet 1, Rev. 0, and H-4-66230, Sheets 5 and 6, Rev. 0.

Contents 5.(b)(6), (7), (8), (12) and (13) must be contained within inner container Ident 1578 described by ERDA Drawing Nos. H-4-66160, Sheet 2, Rev. 0, and H-4-66230, Sheets 5 and 6, Rev. 0.

Contents 5.(b)(9) and (10) shown in DOE Drawing No. H-3-47474, Sheets 1 and 2, Revision No. 0, and Los Alamos Drawing No. 54Y-110854, Sheets 1 and 2, Revision No. B must be contained within the Ident 69 Liner shown in ERDA Drawing No. H-4-66230, Sheets 5 and 6, Revision No. 0, and DOE Drawing No. H-4-66535, Revision No. 0.

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9132	14	71-9132	USA/9132/B(M)F	5	OF	5

7. The cask must be shipped dry (no water coolant in cask cavity). Shipment of sodium wetted fuel rods (external) is authorized for up to 200 g of sodium provided the additional requirements of Section 7.4 of the application are adhered to.

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

   (a) Each package must meet the Acceptance Tests and Maintenance Program of Chapter 8 of the application, as supplemented. The leak test to satisfy ANSI N 14.5 and Regulatory Guide 7.4 in Section 8.1.3 of the application must be a test having sufficient sensitivity to detect a leak rate (air at standard temperature and pressure leaking to 10^{-2} atm) of 10^{-7} atm cc/sec. The results of these tests must be documented and retained for the life of the cask.

   (b) Each package shall be operated and prepared for shipment in accordance with the Operating Procedures of Chapter 7 of the application, as supplemented.

9. Any repair to the trunnions because of out-of-roundness or weld failure must be authorized by NRC prior to returning the package to service.

10. The containment closure bolts (as specified by Note 9, Drawing No. H-4-66230, Sheet 1, Revision No. 0) must be torqued to 70 ± 10 ft-lb.

11. The cask authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR §71.12.

12. Expiration Date: April 1, 2006.

REFERENCES

Nuclear Packaging, Inc., application dated April 22, 1985.

Supplements dated: October 8 and 31, 1985; February 4, 1986; March 21, 1986; May 24, 1988; September 11, 1990; March 22, 1991; February 21, 1996; and February 22, 2001.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: March 14, 2001

NRC FORM 618
(3-96)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIALS PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. PACKAGE IDENTIFICATION NUMBER	d. PAGE NUMBER	e. TOTAL NUMBER PAGES
9148	5	USA/9148/B(U)	1	2

2. PREAMBLE

a. This certificate is issued to certify that the 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)*

Amersham Corporation
40 North Avenue
Burlington, MA 01803

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION:

Technical Operations, Inc. application dated March 24, 1981, as supplemented.

c. DOCKET NUMBER 71-9148

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.: 770

(2) Description

A steel encased uranium shielded source changer for radiographic sources in special form. The source changer measures 23 inches long, 24 inches wide, and 20 inches high. The radioactive source assembly is housed in Zircalloy or titanium "S" tube. The "S" tube is surrounded by depleted uranium metal shield. The depleted uranium shield assembly is encased in two steel containers. The void space between the depleted uranium shield assembly and the inner container is filled with a rigid polyurethane foam. The gross weight of the container is 813 pounds.

(3) Drawing

The packaging is constructed in accordance with Technical Operations, Inc. Drawing No. 77090 - Sheets 1 through 6, Rev. 3.

(b) Contents

(1) Type and form of material

Cobalt 60 as sealed sources that meet requirements of special form radioactive material.

(2) Maximum quantity of material per package

550 curies

Page 2 - Certificate No. 9148 - Revision No. 5 - Docket No. 71-9148

6. The source must be secured in the shielded position of the packaging by the shipping plug, source assembly, and locking device. The shipping plug, source assembly used must be fabricated of materials capable of resisting a 1475°F fire environment for one-half hour and maintaining their positioning function. The ball stop of the source assembly must engage the locking device. The flexible cable of the source assembly and shipping plug must be of sufficient length and diameter to provide positive positioning of the source in the shielded position.

7. Name plates must be fabricated of materials capable of resisting the fire test of 10 CFR Part 71 and maintaining their legibility.

8. The lifting eye bolts (2) must be removed prior to shipment and the holes covered to prevent their use as a tie-down device during transport.

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

   (a) The package shall be prepared for shipment and operated in accordance with the operating procedures in the application; and

   (b) The package shall be maintained in accordance with the maintenance program in in the application.

10. The packaging authorized by this certificate is hereby approved for use under the general license provision of 10 CFR §71.12.

11. Expiration date: March 31, 2002.

REFERENCES

Technical Operations, Inc. application dated March 24, 1981.

Supplements dated: January 18, and May 10, 1982; February 25, and April 16, 1992; September 20, 1996.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Cass R. Chappell

Cass R. Chappell, Chief
Package Certification Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: March 19, 1997

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9150	6	71-9150	USA/9150/B(U)-85	1	OF	3

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)*
U.S. Department of Energy
Washington, D.C. 20585

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION
PAT-2 (Plutonium Air-Transportable Model 2) Safety Analysis Report, SAND81-0001, printed July 1981, as supplemented.

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.: PAT-2

(2) Description

A superalloy primary containment vessel (TB-2) surrounded by a protective overpack (AQ-2). The contents which may be in canisters are contained within a capsule (C-1) within the TB-2.

The AQ-2 overpack is a right circular cylinder, approximately 356 mm (14 inches) high and 381 mm (15 inches) in diameter with protruding handles attached to the cylinder outer walls. The outer shell is a double walled stainless steel structure with rounded end caps, riveted on the bottom and bolted at the top. An inner grain oriented maple wood protective case house the TB-2; it is surrounded by a titanium load spreader which is further surrounded by a grain oriented redwood protective case.

The TB-2 containment vessel consists of (2) iron-base superalloy sections, bolted together with (20) bolts, forming an 88 mm (3.46 inch) diameter sphere. A copper gasket held between knife-edge sealing beads on the mating hemispherical surfaces of the TB-2 provides a seal.

The C-1 capsule is a stainless steel cylinder with a nominal 44 mm (1.80 inch) diameter and a nominal 70 mm (2.76 inch) length; it has a screw top lid which is sealed with teflon tape.

Brass or aluminum canisters may be used in the C-1 capsule to hold various radioactive contents. The canisters may have quartz or glass liners.

The package gross weight is approximately 73 pounds (33 kg).

5.(a) (3) Drawing and Specifications

The packaging is constructed in accordance with specifications and drawings, as listed by document number, issue, and title in the List of Data LD-T67000-000, page 1, issue D and page 2, issue D (Chapter 9 of Safety Analysis Report, SAND81-0001, printed July 1981).

(b) Contents

(1) Type an form of material

Plutonium, uranium, or mixtures of plutonium-uranium in various isotopic compositions in solid form as:

(i) oxide powder, sintered oxide pellets, and metal;

(ii) plutonium sulfate tetrahydrate, $Pu(SO_4)_2\ 4H_2O$, and plutonium nitrate dihydrate, $Pu(NO_3)_4\ 2H_2O$.

(2) Maximum quantity of material per package

(i) For the contents described in 5(b)(1)(i):

Not to exceed 15 grams fissile material, 120 grams mass, 2 watts decay heat, or 0.5 gram water.

(ii) For the contents described in 5(b)(1)(ii):

Not to exceed 3 grams or 0.5 gram water in addition to the water of hydration.

6. Up to 9 grams of polyvinylcholride (PVC), 18 grams of quartz (SiO_2) or glass, 50 grams of brass, and 16 grams of aluminum may be used within the C-1 capsule for packaging of contents. Up to 0.3 gram of polytetra-fluoroethylene (PTFE) tape may be used to seal the C-1 capsule.

7. The C-1 capsule need not be leak tested when the activity of plutonium contents does not exceed 20 Ci per package.

8. A maximum of 2.0 grams of aluminum foil may be used to shim the C-1 within the TB-2 to avoid relative movement between the two.

9. Prior to first use, each package must meet the criteria for the acceptance tests specified in section 8.1 of Chapter 8 of the Safety Analysis Report (SAND81-0001, printed July 1981).

10. Prior to each shipment, the package must meet the criteria for inspections and tests specified in section 8.2 of Chapter 8 of the Safety Analysis Report (SAND81-0001, printed July 1981).

11. Periodic testing and maintenance of the package must be in accordance with section 8.3 of Chapter 8 of the Safety Analysis Report (SAND81-0001, printed July 1981).

12. Operating procedures must be in accordance with Chapter 7 of the Safety Analysis Report (SAND81-0001, printed July 1981).

13. Through special arrangement with the carrier, the shipper shall ensure observance of the following operational controls for each shipment of plutonium by air:

    (a) The package(s) must be stowed aboard aircraft on the main deck in the aft-most location that is possible for cargo of its size and weight. No other type of cargo may be stowed aft of the package(s).

    (b) As an alternative to (a), packages must be stowed in the aft-most lower cargo compartment. No other type of cargo may be stowed aft of the package(s).

    (c) Package(s) must be secured and restrained to prevent shifting under normal transport.

    (d) Cargo which bears the "EXPLOSIVE A" label may not be transported aboard an aircraft carrying a PAT-2 package(s).

14. The package authorized for use by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.12.

15. The package authorized by this certificate is hereby approved for transportation of plutonium by air.

16. Expiration date: July 31, 2006.

<u>REFERENCES</u>

PAT-2 (Plutonium Air-Transportable Model 2) Safety Analysis Report, SANDIA Report No. SAND81-0001, July 1981.

DOE application dated April 19, 1983. Supplements dated August 3, 1983; July 15, 1986; July 16, 1991; May 29, 1996; and May 24, 2001.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: August 2, 2001

NRC FORM 618
(3-96)
10 CFR 71

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIALS PACKAGES

U.S. NUCLEAR REGULATORY COMMISSION

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. PACKAGE IDENTIFICATION NUMBER	d. PAGE NUMBER	e. TOTAL NUMBER PAGES
9152	13	USA/9152/B( )F	1	4

2. PREAMBLE

a. This certificate is issued to certify that the 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)*

U.S. Department of Energy
Washington, DC 20585

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION:

U.S. Department of Energy application dated February 26, 1988, as supplemented.

c. DOCKET NUMBER 71-9152

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.: CNS 1-13C II

(2) Description

A shipping cask for radioactive waste. The packaging consists of a double-walled steel circular cylinder separated by 16-gauge wires, 39-1/8" in diameter and 68-1/2" high with a central steel lined cavity 26-1/2" in diameter and 45-1/6" high, approximately 5" of lead surrounds the central cavity. Closure is accomplished by a steel, plug type, lead filled cover secured by twelve (12), 1-1/4" bolts and seal provided by a flat silicone rubber gasket and a silicone rubber O-ring with a sealed 3/8" test port between the gaskets. Approximately 6" of lead are in the base and cover. The cask is equipped with a cavity drain line sealed with a 3/8" cap screw and gasket, a steel lifting hook for the cover, and top and bottom impact limiters filled with 16.5 lb/ft^3 rigid polyurethane foam clad in steel. The impact limiters are attached to the cask by six (6), 1" ratchet binders. The overall dimensions with impact limiters is 60" in diameter and 99-5/8" high. The package gross weight is approximately 27,000 lbs.

(3) Drawing

The packaging is constructed in accordance with Chem-Nuclear Systems, Inc., Drawing No. E-1-436-111, Sheets 1 and 2, Rev. D.

Page 2 - Certificate No. 9152 - Revision No. 13 - Docket No. 71-9152

5. (b) Contents

(1) Type and form of material

(i) Greater than Type A quantity of nonfissile radioactive material as solidified or dewatered process solids (resins) within a sealed secondary container; or

(ii) Greater than Type A quantity of irradiated solid reactor components within a sealed secondary container.

(iii) Greater than Type A quantity of irradiated fuel (dewatered) within secondary containers described in Chem-Nuclear Systems, Inc. application dated July 16, 1985.

(2) Maximum quantity of material per package

For the contents described in 5(b)(1)(i), (ii), and (iii):

Not to exceed a decay heat generation of 800 watts and 3,000 pounds including weight of the contents and secondary container; and

For the contents described in 5(b)(1)(i):

Residual water in the secondary container not to exceed the activity stated in Table 4.5.2-1 of the application.

For the contents described in 5(b)(1)(iii):

The maximum U-235 enrichment of the uranium oxide fuel material must not exceed 3 w/o. The average burnup of the fuel material must not exceed 3,165 MWD/MTU and must be cooled for at least 6.0 years. Fissile contents not to exceed 400 grams U-235 prior to irradiation.

(c) Transport Index for Criticality Control

Minimum transport index to be shown on label for nuclear criticality control:

For contents described in 5(b)(1)(iii): 100

6. As needed, appropriate shoring must be used in the cask cavity to limit movement of the secondary container during accident condition of transport.

7. The cask cover must be secured by 12, SA-354, Type BD, 1-1/4"-7UNC x 2-1/4" long bolts torqued to 270 ft-lbs ± 10% (lubricated) or 360 ft-lbs ± 10% (dry).

8. Prior to each shipment, the leak test described in Appendix 8B of the application must be performed. No package is to be delivered to a carrier for transport with a detectable leak using the method of Appendix 8B.

9. (a) For any package containing water and/or organic substances which could radiolytically generate combustible gases, determination must be made by tests and measurements or by analysis of a representative package such that the following criteria are met over a period of time that is twice the expected shipment time:

(i) The hydrogen generated must be limited a to a molar quantity that would be no more than 5% by volume (or equivalent limits for other inflammable gases) of the secondary container gas void if present at STP (i.e., no more than 0.063 g-moles/ft^3 at 14.7 psia and 70°F); or

(ii) The secondary container and cask cavity must be inerted with a diluent to assure that oxygen must be limited to 5% by volume in those portions of the package which could have hydrogen greater than 5%.

For any package delivered to a carrier for transport, the secondary container must be prepared for shipment in the same manner in which determination for gas generation is made. Shipment period begins when the package is prepared (sealed) and must be completed within twice the expected shipment time.

(b) For any package containing materials with radioactivity concentration not exceeding that for low specific activity material, and shipped within 10 days after venting of drums or other secondary containers, the determination in (a) above need not be made, and the time restriction in (a) above does not apply.

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

(i) Each package must meet the acceptance tests and be maintained in accordance with the Maintenance Program of Section 8 of the application.

The leak tests described in Appendixes 8-A and 8-B of the application may be performed in accordance with EG&G Idaho, Inc. letter dated December 20, 1982 which was submitted with the Department of Energy consolidated application dated February 26, 1988. Maintenance and repair records shall be furnished to the packaging owner.

(ii) The O-ring must be replaced quarterly with new seals. The flat lid gasket must be replaced annually. The test port and drain line seals must be replaced before each loaded shipment.

Page 4 - Certificate No. 9152 - Revision No. 13 - Docket No. 71-9152

11. The package authorized by this certificate in hereby approved for use under the general license provisions of 10 CFR §71.12.

12. Expiration date: May 31, 2004.

REFERENCES

Department of Energy consolidated application dated: February 26, 1988.

Department of Energy supplements dated: May 12, 1989; April 11, 1994 and March 24, 1999.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Cass R. Chappell

Cass R. Chappell, Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: May 21, 1999

NRC FORM 618
(3-96)
10 CFR 71

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIALS PACKAGES

U.S. NUCLEAR REGULATORY COMMISSION

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. PACKAGE IDENTIFICATION NUMBER	d. PAGE NUMBER	e. TOTAL NUMBER PAGES
9157	8	USA/9157/B(U)-85	1	2

2. PREAMBLE

a. This certificate is issued to certify that the 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)*

Industrial Nuclear Company
14320 Wicks Blvd.
San Leandro, CA 94577

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION:

Industrial Nuclear Company Application dated June 8, 1999, as supplemented.

c. DOCKET NUMBER 71-9157

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.: IR-100

(2) Description

The Model No. IR-100 package is approximately 8.87 inches long, 4.5 inches wide, and 8.5 inches high. The radioactive material contents consist of iridium-192 in source assemblies that meet the requirements for special form material. The source assemblies are positioned within a zircalloy or titanium "S" tube within the IR-100. The "S" tube is surrounded by a shield assembly made of depleted uranium. The uranium shield assembly is encased in a stainless steel housing. The space between the uranium shield assembly and the stainless steel casing is filled with a rigid polyurethane foam. The maximum weight of the IR-100 exposure device is 53 pounds and the maximum shield weight is 32.5 pounds.

(3) Drawings

The packaging is constructed in accordance with Industrial Nuclear Company Drawing Nos.: IR 100-1A, Rev. 3 and IR 100-1B, Rev.2.

(b) Contents

(1) Type and form of material

Iridium 192 as sealed sources that meet the requirements of special form radioactive material.

(2) Maximum quantity of material per package

120 (output) curies

Output curies are determined in accordance with American National Standard N432-1980, "Radiological Safety for the Design and Construction of Apparatus for Gamma Radiography."

6. The source must be secured in the shielded position of the packaging by the shipping plug, source assembly lock, and lock cap. The shipping plug, source assembly lock, and lock cap used must be fabricated of materials capable of resisting a 1475°F fire environment for one-half hour and maintaining their positioning function. The ball stop of the source assembly lock must engage the locking device. The flexible cable of the source assembly and shipping plug must be of sufficient length and diameter to provide positive positioning of the source in the shielded position.

7. The name plate on the exposure device must be fabricated of materials capable of resisting the fire test of 10 CFR Part 71 and maintaining its legibility.

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

   (a) The package must meet the Acceptance Tests and Maintenance Program of Section 8 of the application; and

   (b), Each package shall be operated and prepared for shipment in accordance with the operating procedures in accordance with Section 7 of the application.

9. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR §71.12.

10. Expiration date: September 30, 2004.

REFERENCES

Industrial Nuclear Company application dated June 8, 1999.

Supplements dated: June 9, August 6 and September 9, 1999.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

E. William Brach

E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: 9/16/99

NRC FORM 618
(3-96)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIALS PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. PACKAGE IDENTIFICATION NUMBER	d. PAGE NUMBER	e. TOTAL NUMBER PAGES
9165	4	USA/9165/B(U)	1	2

2. PREAMBLE

a. This certificate is issued to certify that the 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)*

AEA Technology/QSA Inc.
40 North Avenue
Burlington, MA 01803

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION:

Amersham Corporation application dated August 4, 1995, as supplemented.

c. DOCKET NUMBER 71-9165

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.: 855

(2) Description

A steel encased, uranium shielded source changer. Primary components consist of an outer carbon steel shell, rigid polyurethane potting material, uranium shield, eight Titanium "J" tubes, source stop, top and bottom support plates and a gasketed lid which is secured with eight, 3/8"-16 UNC x 5/8" long hex head bolts. The contents are secured and positioned within the "J" tubes by means of a source cable locking device. The package has an outside diameter of approximately 11.25 inches and outside height of approximately 14.75 inches which includes the lid eyebolt. The maximum total weight of the package is approximately 195 pounds.

(3) Drawing

The packaging is constructed in accordance with Amersham Corporation Drawing No. R85590, Rev. B, Sheet No. 1 to Sheet No. 5.

(b) Contents

(1) Type and form of material

Iridium-192 sources which meet the requirements of special form radioactive material.

(2) Maximum quantity of material per package

1,000 curies (output) with no more than 240 curies in a single source.

Output curies are determined in accordance with American National Standard N432-1980, "Radiological Safety for The Design and Construction of Apparatus for Gamma Radiography."

Page 2 - Certificate No. 9165 - Revision No. 4 - Docket No. 71-9165

6. The cover bolts shall be provided with tamperproof seal in accordance with 10 CFR §71.43(b).

7. The two (2), 1/4-inch diameter vent holes in the side of the packaging shall be provided with tight fitting rubber or plastic plugs to preclude the entry of rain water into the packaging.

8. The name plate shall be fabricated of material capable of resisting the fire test of 10 CFR Part 71 and maintaining its legibility.

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

   (a) Each packaging must meet the Acceptance Tests and Maintenance Program in Section 8 of the application, as supplemented.

   (b) The package shall be prepared for shipment and operated in accordance with the Operating Procedures in Section 7 of the application, as supplemented.

10. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR §71.12.

11. Expiration date: December 31, 2003.

REFERENCES

Amersham Corporation application dated August 4, 1995.

Supplements dated: September 21, September 28, and November 29, 1995; November 24, 1998.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Cass R. Chappell

Cass R. Chappell, Chief
Package Certification Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: December 16, 1998

NRC FORM 618 (8-2000) 10 CFR 71	CERTIFICATE OF COMPLIANCE FOR RADIOACTIVE MATERIAL PACKAGES		U.S. NUCLEAR REGULATORY COMMISSION	
a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE / PAGES
9168	12	71-9168	USA/9168/B(U)	1 OF 3

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)*

Duratek
140 Stoneridge Drive
Columbia, SC 29210

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Chem-Nuclear Systems, Inc. application dated February 26, 1990, as supplemented.

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.: CNS 8-120B

(2) Description

The packaging is a carbon steel encased, lead shielded 74-inch OD by 88-inch high cask for radioactive waste materials. The cask is a right circular cylinder with a 62-inch ID by 75-inch high cavity. The walls of the cask contain a lead thickness of 3.35 inches encased in 0.75-inch thick inner steel shell and 1-1/2-inch thick outer steel shell. The exposed sides of the package are provided with a thermal barrier consisting of a 5/32-inch diameter wire wrap on 12-inch centers and covered with a 3/16-inch thick steel jacket. The bottom weldment is made of two, 3-1/4-inch thick carbon steel plates. The primary lid is sealed with a double silicone O-ring and 20 equally spaced 2-inch diameter bolts. The centered secondary lid is sealed with a double silicone O-ring and twelve equally spaced 2-inch diameter bolts, and covers a 29-inch opening in the primary lid. The optional drain line is sealed with a 3/4-inch diameter cap screw and a silicone O-ring. The lid sealing surfaces are stainless steel and the space between the double O-ring seals is provided with a test port for leak testing.

The top and bottom of the cask are provided with steel encased, rigid polyurethane foam impact limiters. The impact limiters are secured to each other about the cask with eight 1-inch diameter ratchet binders. The impact limiters are 102 inches in diameter and the overall height of the package with the impact limiters attached is 132 inches.

The package is provided with four tie-down and two removable lifting devices. Each lid is provided with three lifting lugs. The gross weight of the packaging and contents is approximately 74,000 pounds.

(a) Packaging (Continued)

(3) Drawings

The packaging is constructed in accordance with Chem-Nuclear Systems, Inc. Drawing No. C-110-E-0007, Sheets 1, 2, and 3, Revision No. 10.

(b) Contents

(1) Type and form of material

(i) Byproduct material in the form of dewatered resins, solids, or solidified waste contained within secondary containers; or

(ii) Radioactive material in the form of activated reactor components.

(2) Maximum quantity of material per package

Type B quantity of radioactive material, not to exceed 2,000 times a Type A quantity, 100 thermal watts, and 14,680 pounds including weight of the contents, secondary containers, and shoring. The contents may include fissile materials provided the mass limits of 10 CFR 71.53 are not exceeded.

6. Except for close fitting contents, wood shoring must be placed between the secondary containers, or activated components, and the cask cavity to prevent movement during accident conditions of transport.

7. The cask primary lid must be secured by twenty and the secondary lid by twelve, 2"-8UNC-2A x 4-3/4" or twelve, 2"-8UNC-2A x 4" long hex cap screws with a flat washer torqued to 500 ft-lbs ± 50 ft-lbs (lubricated).

8. Prior to each shipment, the package must be leak tested in accordance with Section 8.2.2.2 of the application. For contents that meet the definition of low specific activity material or surface contaminated objects in 10 CFR 71.4, and also meet the exemption standard for low specific activity material and surface contaminated objects in 10 CFR 71.10(b)(2), the pre-shipment leak test is not required.

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

(i) Each package must meet the acceptance tests and be maintained in accordance with the Acceptance Tests and Maintenance Program of Section 8.0 of the application,

(ii) The seals must be replaced with new seals if inspection shows any defects or every 12 months, whichever occurs first. The tests ports and optional drain line must be appropriately plugged and sealed prior to transport, and

(iii) The package must be prepared for shipment and operated in accordance with the operating procedures of Section 7.0 of the application.

10. (a) For any package containing water or organic substances which could radiolytically generate combustible gases, determination must be made by tests and measurements or by analysis of a representative package such that the following criteria are met over a period of time that is twice the expected shipment time:

(i) The hydrogen generated must be limited to a molar quantity that would be no more than 5% by volume (or equivalent limits for other inflammable gases) of the secondary container gas void if present at STP (i.e., no more than 0.063 g-moles/ft^3 at 14.7 psia and 70°F); or

(ii) The secondary container and cask cavity must be inerted with a diluent to assure that oxygen must be limited to 5% by volume in those portions of the package which could have hydrogen greater than 5%.

For any package delivered to a carrier for transport, the secondary container must be prepared for shipment in the same manner in which determination for gas generation is made. Shipment period begins when the package is prepared (sealed) and must be completed within twice the expected shipment time.

(b) For any package containing materials with a radioactivity concentration not exceeding that for low specific activity material, and shipped within 10 days of preparation, or within 10 days after venting of drums or other secondary containers, the determination in (a) above need not be made, and the time restriction in (a) above does not apply.

11. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.12.

12. Expiration date: June 30, 2005.

REFERENCES

Chem-Nuclear Systems, Inc., application dated February 26, 1990.

Supplements dated: February 22, 1994; February 23, 1995; September 1, 1998; May 25 and June 1, 1999; and May 26, August 23 and 30, December 8, 2000, January 30, 2001, and April 23, 2001.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

E. William Brach

E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: June 25, 2001

NRC FORM 618
(3-96)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIALS PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. PACKAGE IDENTIFICATION NUMBER	d. PAGE NUMBER	e. TOTAL NUMBER PAGES
9183	13	USA/9183/B( )F	1	4

2. PREAMBLE

a. This certificate is issued to certify that the 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)*

NAC International, Inc.
655 Engineering Drive
Suite 200
Norcross, Georgia 30092

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION:

NAC International, Inc. application dated May 26, 1989, as supplemented

c. DOCKET NUMBER 71-9183

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.: NAC-1

(2) Description

A steel and lead shielded shipping cask. The cask is a right circular cylinder with upper and lower steel encased balsa impact limiters. The overall dimensions are 214 inches in length and 50 inches in diameter. The gross weight of the cask is approximately 49,000 pounds. The inner cavity is 178 inches long and 13.5 inches in diameter. The thickness of the inner shell is 5/16 inch, and the thickness of the outer shell is 1-1/4 inches. The two stainless steel shells are welded to a 2-inch thick stainless steel shield disc at the bottom. The annulus between the inner and outer shells is filled with lead (lead thickness: 6-5/8 inches maximum, 5 inches minimum).

The stainless steel lid is a frustum of a cone 7.5 inches thick. The lid is secured to the cavity flange by six, ASTM-A320, Grade L43, 1-1/4-inch diameter bolts. The seal is provided by two polytetrafluoroethylene O-rings. Four trunnions, two located on either side of the upper or lower impact limiter, are provided. Other cask features include two drain valves located in the bottom shield disc, vent valve, head closure gasket leak check valve, and rupture disc - pressure relief valve system located in the cavity flange. For transport, the cask may be enclosed in an expanded metal cage or closed shipping container.

Page 2 - Certificate No. 9183 - Revision No. 13 - Docket No. 71-9183

5.(a) (Continued)

(3) Drawings

The Model No. NAC-1 shipping cask is constructed in accordance with Nuclear Fuel Services, Inc., Drawing No. E 10080, Sheets 1 through 4, Rev. 22.

(b) Contents

(1) Type and form of material

(i) Clad, irradiated, metallic natural uranium fuel rods.

(ii) Solid, non-fissile, irradiated hardware.

(2) Maximum quantity of material per package

The cavity content must not exceed a thermal decay heat load of 750 watts and a weight of 3,700 lbs., including weight of component spacers (or fuel basket) used in the cask cavity to limit movement of contents during shipment. Fuel rods are additionally limited as follows:

(i) 21 intact rods or 6 encapsulated (defective) rods. Each defective rod will be encapsulated in either a 2.75-inch I.D. failed fuel rod can, as shown on Nuclear Assurance Corporation Drawing No. 340-108-D2, Rev. 9, or a 4.00-inch I.D. failed fuel rod can, as shown on Nuclear Assurance Corporation Drawing No. 340-108-D1, Rev. 9. Defective rods encapsulated in the 2.75-inch I.D. failed fuel rod cans will be shipped in a six rod capacity liner, as shown on Nuclear Assurance Corporation Drawing No. 491-001, Rev. 0., and defective rods encapsulated in the 4.00-inch I.D. failed fuel rod cans will be shipped in a three rod capacity liner, as shown on Nuclear Assurance Corporation Drawing No. 347-211-F19, Rev. 5.

(ii) 1,600 MWD/MTU average burn-up.

(iii) Minimum 365-day cooling time after irradiation.

(c) Transport Index for Criticality Control

Minimum transport index to be shown on label for nuclear criticality control: 0.4

Page 3 - Certificate No. 9183 - Revision No. 13 - Docket No. 71-9183

6. The cask cavity must be dry (no free water) when delivered to a carrier for transport.

7. As needed, appropriate component spacers (fuel basket and axial spacers for shipment of fuel rods) must be used in the cask cavity to limit movement of contents during accident conditions of transport.

8. The cask may be shipped in a closed shipping container provided that the closed container, the cask tie-down and support system, and the transport vehicle (trailer) meet the applicable requirements of the Department of Transportation. Tie-down devices which are a structural part of the package must comply with 10 CFR §71.45.

9. When the cask is shipped in a closed shipping container, the center of gravity of the combined cask, closed shipping container, and trailer must not exceed 75 inches.

10. When the cask is shipped in a closed shipping container, the internal heat load must not exceed 750 watts.

11. In lieu of the requirements of 10 CFR §71.87(e), the licensee must perform periodic maintenance and testing of O-rings, drain and vent ball valves, relief valves, and rupture discs of the cask as indicated in the table given below. During inactive periods, the maintenance and testing frequency may be disregarded provided that the package is brought into full compliance prior to the next use of the package.

Cask Component	Period	Test/Action
Ball Valve	Each Shipment	Hydro test to 30 psig*
Ball Valve	Annually	Replace seats and seals
O-rings	Each Shipment	Test to 30 psig*
O-rings	Annually	Test to 100 psig*
Inner Containment Vessel	Annually	Test to 100 psig*
Cavity Relief Valve	Annually	Test at set point
Cavity Rupture Disc	Annually	Replace
Neutron Shield Tank Rupture Disc	Annually	Replace
Impact Limiters	Annually	Test to 5 psig*

*There must be no visual (pressure gauge) indications of pressure drop for the component under test during a 10-minute test period. Otherwise, corrective action must be taken and the test repeated until such time as the component meets the specified tests. (Test to pressures equal to or greater than those indicated.)

Page 4 - Certificate No. 9183 - Revision No. 13 - Docket No. 71-9183

12. The package shall be prepared for shipment and operated in accordance with the operating procedures in Chapter 7 of the application, as supplemented.

13. Each package must be maintained in accordance with the maintenance program in Chapter 8 of the application.

14. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR §71.12.

15. Expiration date: September 30, 2004.

REFERENCES

NAC International, Inc. application dated May 26, 1989.

Supplements dated January 29 and March 20, 1990; August 4, 1994; and August 31, 1999.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Susan F. Shankman for

E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: September 24, 1999

NRC FORM 618
(3-96)
10 CFR 71

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIALS PACKAGES

U.S. NUCLEAR REGULATORY COMMISSION

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. PACKAGE IDENTIFICATION NUMBER	d. PAGE NUMBER	e. TOTAL NUMBER PAGES
9184	5	USA/9184/B(U)	1	2

2. PREAMBLE

a. This certificate is issued to certify that the 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)*

Packaging Technology, Inc.
4507 D Pacific Highway East
Tacoma, WA 98424-2633

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION:

Nuclear Packaging, Inc. consolidated application dated March 31, 1989, as supplemented.

c. DOCKET NUMBER 71-9184

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.: PAS-1

(2) Description

The packaging consists of a primary containment vessel (20.5" OD x 23.4" OH) enclosed inside a secondary containment vessel and radiation shield (32.5" OD x 39.0" OH). The 15 milliliter water sample is contained within a undefined sample cask. Additionally, four iodine collection cartridges and four offgas vials are maintained inside the foam shoring above the sample cask. Loose vermiculite surrounds the perimeter of the sample cask to absorb the water sample should leakage occur. Completely surrounding the secondary containment vessel and radiation shield is a foam filled steel encased overpack (48.0" OD x 66.0" OH) which provides impact and thermal protection.

The primary containment vessel, which is constructed of 304 stainless steel varying in thickness from 3/4" to 1.25", is provided with double Viton O-ring seals and a sealed test port between the seals for leak testing. The assembly is secured with eight, 3/8"-16 UNC x 8" long screws.

The secondary containment vessel and radiation shield provides 0.75" thick steel and 5.1" thick lead shielding in the radial direction, 2.0" thick steel and 5.1" thick lead shielding on the bottom, and 3.5" thick steel and 4.8" thick lead shielding on the top. The lid is secured with eight, 1.0"-8 UNC x 3.0 long bolts. The lid is sealed with two Viton O-rings with a sealed test port between the seals for leak testing.

The overpack provides about 7.25" thick foam on the sides and about 13" on the top and bottom. The two halves of the overpack are held together by eight, 3/4"-10 UNC x 1.5" long bolts. A Neoprene gasket prevents rain water from entering the overpack.

The weight of the package including a maximum sample cask weight of 1,375 pounds, is about 12,800 pounds.

Page 2 - Certificate No. 9184 - Revision No. 5 - Docket No. 71-9184

5.(a)(3) Drawings

The package is constructed in accordance with Nuclear Packaging, Inc. Drawing No. X-20-218D, Sheets 1 and 2, Rev. C.

(b) Contents

(1) Type and form of material

(i) Radioactive material in form of liquid or gaseous samples in sample casks, cartridges and vials.

(ii) Byproduct and activation materials as solids and process solids or resins, either dewatered, solid, or solidified in secondary containers.

(2) Maximum quantity of material per package

50 Ci of mixed fission and activation products, 15 milliliters of liquid, one sample cask or secondary container and four cartridges and four vials.

6. In addition to the requirements of Subpart G of 10 CFR Part 71, each package prior to first use must meet the acceptance tests and criteria specified in Section 8.1, must be maintained in accordance with Section 8.2, and must be prepared for shipment in accordance with Chapter 7.0 of the application, and the supplement dated July 8, 1994.

7. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR §71.12.

8. Expiration date: July 31, 2004.

REFERENCES

Nuclear Packaging, Inc., consolidated application dated March 31, 1989.

Supplement dated: April 7, 1989.

VECTRA Technologies, Inc., supplements dated: July 8, 1994 and January 30, 1998.

Transnuclear, Inc., supplement dated January 30, 1998.

Packaging Technology, Inc., Supplement dated: April 30,1999.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: 7/21/99

NRC FORM 618
(3-96)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIALS PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. PACKAGE IDENTIFICATION NUMBER	d. PAGE NUMBER	e. TOTAL NUMBER PAGES
9185	5	USA/9185/B(U)-85	1	2

2. PREAMBLE

a. This certificate is issued to certify that the 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)*

Industrial Nuclear Company
14320 Wicks Blvd.
San Leandro, CA 94577

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION:

Industrial Nuclear Company application dated July 1, 1999, as supplemented.

c. DOCKET NUMBER 71-9185

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.: OP-100

(2) Description

The Model No. OP-100 package consists of either an IR-50 source changer, or an IR-100 exposure device, which is positioned within a 10 gallon drum. The drum is made of 20 gauge steel, and is closed with a 12 gauge closure ring and a 5/8 inch diameter steel bolt. Plywood members are used to position and support either the IR-50 or IR-100 within the steel drum.

The IR-50 source changer and the IR-100 exposure device are approximately 8.87 inches long, 4.5 inches wide, and 8.5 inches high. The radioactive material contents consist of iridium-192 in source assemblies that meet the requirements for special form material. The source assemblies are positioned within a zircalloy or titanium "S" tube within the IR-50 or IR-100. The "S" tube is surrounded by a shield assembly made of depleted uranium. The uranium shield assembly is encased in a stainless steel housing. The space between the uranium shield assembly and the stainless steel casing is filled with a rigid polyurethane foam. The maximum weight of the IR-50 source changer is 53 pounds, the maximum weight of the IR-100 exposure device is 50 pounds, and the maximum gross weight of the Model No. OP-100 package is 75 pounds.

(3) Drawings

The packaging is constructed in accordance with Industrial Nuclear Company Drawing Nos.: OP 100-1, Rev. 3, IR 50-1A, Rev. 2, IR 50-1B, Rev. 1, IR 100-1A, Rev. 3, and IR 100-1B, Rev. 2.

(b) Contents

(1) Type and form of material

Iridium-192 as sealed sources that meet the requirements of special form radioactive material.

Page 2 - Certificate No. 9185 - Revision No. 5 - Docket No. 71-9185

(b) Contents (continued)

(2) Maximum quantity of material per package

120 (output) curies

Output curies are determined in accordance with American National Standard N432-1980, "Radiological Safety for the Design and Construction of Apparatus for Gamma Radiography."

6. The source shall be secured in the shielded position of the packaging by the source assembly lock, lock cap, and the shipping plug (IR-100 only). The source assembly lock, lock cap, and the shipping plug (IR-100 only), must be fabricated of materials capable of resisting a 1475°F fire environment for one-half hour and maintaining their positioning function. The ball stop of the source assembly must engage the source assembly lock. The flexible cable of the source assembly and shipping plug must be of sufficient length and diameter to provide positive positioning of the source in the shielded position.

7. The name plate on the overpack must be fabricated of materials capable of resisting a 1475°F fire environment for one-half hour and maintain its legibility. The two vent holes in the side of the overpack must be covered with tape or rubber (plastic) plugs to prevent entry of rain water.

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

(a) The package shall be prepared for shipment in accordance with the Operating Procedures of Chapter 7 of the application and

(b) Each package must meet the Acceptance Tests and Maintenance Program of Chapter 8 of the application.

9. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR §71.12.

10. Expiration date: November 30, 2003.

REFERENCES

Industrial Nuclear Company application dated July 1, 1999.

Supplements dated: September 14 and December 29, 1999.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: 2/26/00

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9186	13	71-9186	USA/9186/B(U)F	1	OF	5

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)*

U.S. Department of Energy
Division of Naval Reactors
Washington, DC 20858

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Safety Analysis for Shipping S8G Power Units in the S-6213 Container, Rev. 7, dated June 16, 1975, as supplemented; and Safety Analysis for Shipment of S6W Shipboard Power Units in the Model 2 S-6213 PUSC, as supplemented.

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 Nos: Model 1, S-6213 Power Unit Shipping Container

Model 2, S-6213 Power Unit Shipping Container

(2) Description

A power unit shipping container (PUSC) for shipment of a power unit complete with control rods and control rod drive mechanisms installed.

The Model 1 S-6213 PUSC consists of a carbon steel cylindrical shell approximately 9-1/4 feet in outside diameter by 39-1/2 feet long, including hemispherical steel end impact limiters, with 10-3/4-foot outside diameter central flanges joining the barrel and cover halves. The Model 2 S-6213 PUSC is of the same design as the Model 1, except that the primary container material is HY-80 steel. A power unit is supported in the PUSC by a centrally located thick circular steel plate (PU head) which is clamped between the central mating flanges of the PUSC and fastened by 94, 2-inch diameter high strength studs. The upper and lower extremities of the power unit cantilever into the barrel and cover halves without additional support except for the longest control rod drive mechanisms (S8G Power Unit Type B only). A lower support adapter is installed in the barrel end of the container during shipment of the S6W prototype power unit and the S6W shipboard power unit. A shipping/lifting ring, a flange adapter, and a lower support adapter are installed in the container during shipment of the S9G shipboard power unit.

The PUSC is shipped in the horizontal position on a support frame which is secured to a specially built flatbed rail car. The PUSC, including frame and contents, weighs approximately 490,000 pounds for shipments of Type A and B, S8G power units.

5.(a) Packaging (Continued)

(2) Description (Continued)

The weight of the PUSC, including frame and contents is approximately 438,900 pounds for shipment of the S6W prototype power unit, 429,900 pounds for shipment of the S6W shipboard power unit, and 329,000 pounds for shipment of the S9G shipboard power unit.

(3) Drawings

The Model 1 and Model 2 S-6213 PUSC are constructed in accordance with the Drawings included in the applications (see references, below).

5.(b) Contents

(1) Type and form of material

(i) Unirradiated Naval Reactors Type A or B S8G power unit as described in Chapter 5 of the application and containing uranium enriched in the U-235 isotope.

(ii) Unirradiated S6W advanced fleet reactor prototype power unit or unirradiated S6W advanced fleet reactor shipboard power unit as described in Chapter 6 of "S6W Prototype Power Unit in S-6213 Power Unit Shipping Container Safety Analysis Report" WAPD-REO(c)1219, Revision 1, and containing uranium enriched in the U-235 isotope.

(iii) Unirradiated S6W high performance fleet core shipboard power unit, as described in addendum to Chapter 6 of "S6W Shipboard Power Unit in S-6213 Power Unit Shipping Container Safety Analysis Report For Packaging," WAPD-REO(c)-1457 and WAPD-REO(c)-1566, and containing uranium enriched in the U-235 isotope.

(iv) Unirradiated S9G shipboard power unit, as described in Chapter 6 of "S9G Shipboard Power Unit in S-6213 Power Unit Shipping Container Safety Analysis Report For Packaging," Revision 2, and containing uranium enriched in the U-235 isotope.

(2) Maximum quantity of material per package

For the Model 1 S-6213 PUSC:

One Type A S8G Power Unit, or
One Type B S8G Power Unit, or
One S6W Advanced Fleet Reactor Prototype Power Unit, or
One S6W Advanced Fleet Reactor Shipboard Power Unit, or
One S6W High Performance Fleet Core Shipboard Power Unit, or
One S9G Shipboard Power Unit.

5.(b) Contents (Continued)

For the Model 2 S-6213 PUSC:

One S6W Advanced Fleet Reactor Shipboard Power Unit, or
One S6W High Performance Fleet Core Shipboard Power Unit, or
One S9G Shipboard Power Unit.

5.(c) Transport Index for Criticality Control

Minimum transport index to be shown on label for nuclear criticality control: 100

6. The Model 1 S-6213 PUSC shall be designated as B( )F. Use of Model 1 S-6213 PUSC packaging fabricated after August 31, 1986, is not authorized.

7. All control rods shall be restrained in the power unit fuel cells by the control rod holddown latches.

8. For the Model 1 S-6213 PUSC, in addition to the requirements of Subpart G of 10 CFR Part 71, a determination shall be made, for each shipment, of the "g" forces that the package or packaging has been subjected to during transport.

   (a) A nondestructive examination of the entire length of both inner and outer surfaces of the four tie-down support bracket-to-container wall butt welds shall be conducted:

      (1) if the packaging (with or without contents) has been subjected to "g" forces in excess of 2 g's in any direction through the center of gravity of the package since the last inspection, and

      (2) following the fourth shipment*, and

      (3) after every second shipment* following the fourth shipment

---

* This requirement shall not be construed to require an inspection if previous shipment had been inspected in accordance with (8(a)(1)) above.

(b) The nondestructive examination in accordance with a written procedure may be by either:

(1) The liquid penetrant method in accordance with:

(i) Article 6, Section V, ASME Code, or

(ii) MIL-STD-271E, "Nondestructive Testing Requirements for Metals," Section 5, October 31, 1973, or

(iii) NAVSHIPS 250-1500-1, "Welding Standard," Section 12.5

(2) or the magnetic particle method in accordance with:

(i) Article 7, Section V, ASME Code (Yoke Technique; Dry Particle Method; direct or rectified current), or

(ii) MIL-STD-271E, Section 4; specifically 4.3.1 (General) and 5.6.1 (coatings), 4.3.3 (Dry Powder), 4.3.3.3.6 (Continuous), and 4.3.3.3 (Procedure) as excepted by using direct or rectified current, 4.3.3.3.3 (Yoke Technique), 4.3.2.5 (sensitivity and cleaning), and 4.3.1.3 (smoothness), or

(iii) NAVSHIPS 250-1500-1, Section 12.4, 12.4.1 (General), 12.4.3 (Dry powder), 12.4.3.3.2.1 (Yoke Technique) using direct or rectified current.

(c) If any indications, as defined in accordance with either:

(1) Paragraph UA-93(a), Appendix VIII, Division 1, Section VIII, ASME Code (with 7(b)(2)(i), above), or

(2) Paragraphs UA-72 and UA-73, Appendix VI, Division 1, Section VIII, ASME Code (with 7(b)(2)(i), above), or

(3) Class 1 acceptance criteria of NAVSEA 0900-LP-003-8000, "Surface Inspection Acceptance Standards for Metal," with Change 2, July 1, 1974 (with 7(b)(1)(ii) or 7(b)(2)(ii), above), or

(4) NAVSHIPS 250-1500-1, Section 10.3.2 (with 7(b)(1)(iii) or 7(b)(2)(iii), above), as noted,

are detected, the packaging shall be repaired and reinspected prior to use and shall be inspected prior to each shipment thereafter. Any defects shall be reported in accordance with 10 CFR §71.95.

9. Expiration date: May 31, 2007

REFERENCES

For the Model 1 S-6213 PUSC:

U.S. Naval Reactors application dated July 24, 1975.

Supplements dated: June 3, 1977; July 24, 1978; Naval Reactors letter G#C89-2838, dated May 22, 1989; Naval Reactors letter G#C90-03664, dated September 5, 1990; Naval Reactors letter G#92-03563, dated June 17, 1992; and Naval Reactors letter G#C92-03714, dated October 2, 1992; Naval Reactors letter G#97-03425, dated February 7, 1997; Naval Reactors letter G#C97-03614, dated September 29, 1997; and Naval Reactors letter G#01-03619, dated December 11, 2001.

For the Model 2 S-6213 PUSC:

U.S. Naval Reactors application G#C91-11165, dated December 19, 1991.

Supplements dated: Naval Reactors letter G#92-03563, dated June 17, 1992; and Naval Reactors letter G#C92-03714, dated October 2, 1992; Naval Reactors letter G#97-03425, dated February 7, 1997; Naval Reactors letter G#C97-03614, dated September 29, 1997; and Naval Reactors letter G#01-03619, dated December 11, 2001.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

E. William Brach

E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date March 18, 2002

NRC FORM 618
(3-96)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIALS PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. PACKAGE IDENTIFICATION NUMBER	d. PAGE NUMBER	e. TOTAL NUMBER PAGES
9187	4	USA/9187/B(U)	1	2

2. PREAMBLE

a. This certificate is issued to certify that the 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)*

AEA Technology/QSA Inc.
40 North Avenue
Burlington, MA 01803

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION:

Tech/Ops application dated December 27, 1983, as supplemented.

c. DOCKET NUMBER 71-9187

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.: 865

(2) Description

A steel encased, uranium shielded radiographic exposure device 5" OD x 12.25" long. The device is provided with 0.88" OD x 9.25" long handle and two 1.38" x 5.5" long triangular shaped legs. Primary components consist of an outer steel shell, internal bracing, depleted uranium shield, and a source tube. The contents are securely positioned in the source tube by a source holder assembly and actuator and locking assembly. Tamper-indicating seals are provided on the packaging and a 0.12-inch thick steel outer cover is bolted over the source actuator and locking assembly for additional protection during transport. The total weight of the package is approximately 59 pounds.

(3) Drawings

The packaging is constructed in accordance with the following Tech/Ops Drawing Nos.: 86590, Sheets 1 through 5, Rev. 1; 86500-10, Rev. 0; and Amersham Corporation Drawing No. R86591, Rev. A.

(b) Contents

(1) Type and form of material

Iridium-192 as sealed source must meet the requirements of special form radioactive material.

(2) Maximum quantity of material per package

240 curies (output)

Output curies are determined in accordance with American National Standard N432-1980, "Radiological Safety for the Design and Construction of Apparatus for Gamma Radiography."

Page 2 - Certificate No. 9187 - Revision No. 4 - Docket No. 71-9187

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

(a) Each packaging must meet the Acceptance Tests and Maintenance Program in Section 8, of the October 29, 1993, supplement.

(b) The package shall be prepared for shipment and operated in accordance with the Operating Procedures in Section 7, of the November 24, 1998, supplement.

7. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR §71.12.

8. Expiration date: December 31, 2003.

REFERENCES

Tech/Ops application dated December 27, 1983.

Amersham Corporation supplements dated: March 15, 1984, November 8, 1988, and August 16, and October 29, 1993, and November 20, 1995.

AEA Technology Supplement dated November 24, 1998.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Cass R. Chappell

Cass R. Chappell, Chief
Package Certification Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: December 7, 1998

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9196	19	71-9196	USA/9196/AF-85	1	OF	3

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)*
Duratek
140 Stoneridge Drive
Columbia, South Carolina 29210

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION
Chem-Nuclear Systems, LLC application dated February 17, 1999, as supplemented.

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.: UX-30

(2) Description

Overpack for 30-inch enriched uranium hexafluoride (UF_6) cylinders. The overpack is a right circular cylinder constructed of two stainless steel shells with the volume between the shells filled with 6-inch thick foam (7.8 - 9.8 PCF). A stepped and gasketed horizontal joint permits the top half of the overpack to be removed from the base. The package "halves" are secured with ten indexed, cross-locking "ball lock" pins. The overpack is 43.5" in diameter by 96" long. The maximum gross weight of the package is 8270 lbs.

Two types of 30 inch uranium hexafluoride cylinders may be carried in the UX-30 overpack. These are (1) an ANSI N14.1 Standard 30B cylinder, or (2) a CBC Watertight™ Model 195 cylinder.

The CBC Watertight™ Model 195 cylinder is essentially a 30B cylinder equipped with a Valve Protective Cover (VPC) that bolts over and protects the cylinder valve during transport. The VPC is a special design feature that provides additional assurance against the inleakage of water to the containment system and is an enclosure that retains any leakage.

(3) Drawings

The Model No. UX-30 packaging is fabricated in accordance with Chem-Nuclear Systems LLC, Drawing No. C-110-B-57922-0002, Sheets 1 through 3, Rev. 2. The CBC Watertight™ Model 195 Cylinder is fabricated in accordance with Columbiana Boiler Company drawings: 71800-C, Rev. 2, SK-71800-B-3, Rev. 1, SK-71800-B-4, Rev. 1, SK-71800-5, Rev. 0 and SK-71800-B-7, Rev. 0.

(b) Contents

(1) Type and form of material

UF_6 enriched in the U-235 isotope.

(2) Maximum quantity of material per package

(i) ANSI standard N14.1 30B cylinder or CBC Watertight™ Model 195 cylinder: 5,020 pounds UF_6 enriched to not more than 5 w/o in the U-235 isotope. The maximum H/U atomic ratio for the UF_6 is 0.088.

c) Transport Index for Criticality Control (Criticality Safety Index)

Minimum criticality safety index for the UX-30 overpack containing a standard ANSI N14.1 30B cylinder	5.0
Minimum criticality safety index to be shown on the label for the UX-30 overpack containing a CBC Watertight™ Model 195 Cylinder	0.0

6. The ANSI standard 30B, 30-inch diameter UF_6 cylinder, must be fabricated, inspected, tested and maintained in accordance with a) American National Standard N14.1-1995 or an earlier version of ANSI N14.1 in effect at the time of fabrication or b) American National Standard N14.1-1995 or an earlier version of ANSI N14.1 in effect at the time of fabrication and ISO 7195:1993(F). Cylinders must be fabricated in accordance with Section VIII, Division I, of the ASME (American Society of Mechanical Engineers) Boiler and Pressure Vessel Code and be ASME Code stamped.

7. The CBC Watertight™ Model 195 cylinder (new or retrofitted cylinders) must be fabricated, inspected, tested, and maintained in accordance with CBC Watertight™ Model 195 cylinder specification no. CBC-WT-M195, Revision 2 dated April 5, 2002.

8. When the optional 4 lid lifting clips are used instead of the top lugs, the top lid (cover) must be lifted with a spreader bar (saddle).

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

(a) Prior to each shipment, the overpack gaskets must be inspected. These gaskets must be replaced if inspection shows any defects or every 12 months, whichever occurs first.

(b) Each packaging must meet the Acceptance Tests and Maintenance Program of Chapter 8 of the application, as supplemented.

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

(d) Prior to each shipment, the stainless steel components of the packaging must be visually inspected. Packagings in which stainless steel components show pitting, corrosion, cracking, or pinholes are not authorized for transport.

10. The 30-inch diameter UF_6 cylinder valve stem and plug may be tinned with ASTM B32, alloy 50A or Sn50 solder material, or a mixture of alloy 50A or Sn50 with alloy 40A or Sn40A material, provided the mixture has a minimum tin content of 45 percent.

11. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.12.

12. Expiration date: February 28, 2006.

<u>REFERENCES</u>

Chem-Nuclear Systems, LLC application dated February 17, 1999.

United States Enrichment Corporation supplement dated April 14, 1997. Chem-Nuclear Systems, LLC supplements dated May 10, 1999, April 14, June 22, October 31, and December 4, 2000, April 23, October 11, and October 19, 2001 and April 16, 2002.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Charles L. Miller for
E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: September 26, 2002

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9200	10	71-9200	USA/9200/B(M)F	1	OF	5

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)*

Department of Energy
Washington, D.C. 20585

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Nuclear Packaging, Inc., application dated April 6, 1991 as supplemented.

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.: 125-B

(2) Description

A stainless steel and lead shielded shipping cask. The contents are shipped dewatered. The cask is a right circular cylinder, 65.5-inch outer diameter by 207.5-inch length. The cavity dimensions are 51.25-inch diameter by 192.5-inch length. A 1.0-inch thick stainless steel inner shell, 3.88-inch thick lead annulus and 2.0-inch thick stainless steel outer shell, and 7.50-inch thick welded stainless steel bottom plate make up the cask body. A ten gauge stainless steel thermal shield surrounds the cask outer shell with standoff provided by a wire wrap on a 3.3-inch pitch spacing. The outer lid is 7.50-inch thick stainless steel equipped with a 300 psig rupture disc. The seal is provided by 2 Neoprene O-rings secured by 32, 1-1/2-6 UNC closure bolts. A test port is provided between the O-rings. The lid is also provided with a vent port. Protrusions from the outer cask external cylindrical surface include 2 lifting and 4 tie-down trunnions, 1 shear block for fitting to the shipping skid, and 16 impact limiter attachment lugs (8 at each end of the cask). The impact limiters are 120 inches in diameter by 75 inches long fabricated from 1/4-inch thick stainless steel and filled with closed-cell polyurethane foam. Each impact limiter is secured to the cask by 8, 1-1/4-7 UNC bolts necked down to 1 inch. Plastic pipe plugs are provided in each impact limiter. The overall dimensions of the cask with upper and lower impact limiters are 120-inch outer diameter by 279.5-inch length.

5.(a)(2) Description (continued)

A separate inner vessel (fuel/canister basket) is positioned within the cask cavity. The inner vessel consists of 7, 14.5-inch ID by 0.38-inch wall pipes with a welded bottom plate and top end fixture plate which provides a 151-inch long cavity for the canisters. The pipe assembly is positioned within a 50.25-inch OD by 1.0-inch thick steel shell with a 2.0-inch thick welded bottom plate. The space between the pipes and steel shell contain stainless steel structural members and solid neutron moderator and absorber. The top of each tube is shielded by a 10-inch thick stainless steel plug. The inner lid is 5.0-inch thick stainless steel equipped with 2, 300 psig rupture discs in series. The lid has 2 Neoprene O-rings and is secured to the inner vessel by 24, 3/4-10 UNC closure bolts. A test port is provided between the O-rings. The lid is also provided with a vent port.

A fuel, filter, or knockout canister is positioned within the inner vessel with canister impact limiters and a top 10.0-inch thick stainless steel shield plug. Each canister is 14.0-inch OD by 150.0-inch long by 0.25-inch wall and contains Boral sheets or B_4C rods. Canister containment is not required with closure provided by welded or bolted plate with 2 or 4 fittings.

The weight of the cask (100,500 pounds), impact limiters (11,700 pounds each), inner vessel (37,000 pounds), canisters (1,046 to 1,440 pounds each), and canister contents (1,500 to 1,894 pounds each) is approximately 181,500 pounds.

(3) Drawings

(i) The packaging is constructed in accordance with Nuclear Packaging Inc., Drawing No. X-101-100, Sheets 1 through 7, Rev. T.

(ii) The canisters are constructed in accordance with Babcock and Wilcox Company Drawing Nos.: 1161299D, Rev. 1; 1161300D, Rev. B1; and 1161301D, Rev. 1.

(b) Contents

(1) Type and form of material

(i) Byproduct and special nuclear material in the form of irradiated fuel particles, partial fuel rods, partial assemblies, and core debris. The maximum pre-irradiation U-235 enrichment must not exceed 2.98 weight percent. The average burnup of the fuel material must not exceed 3,165 MWD/MTU and be cooled for at least 6.0 years.

5.(b)(1) (continued)

(ii) Irradiated core structural components, contaminated defueling equipment, and filter-aid materials.

Except for close fitting contents, dunnage must be provided in the shipping cask cavity sufficient to prevent significant movement of the contents and secondary containers relative to the outer packaging under accident conditions.

(iii) Byproduct and special nuclear material in the form of internal contamination inside the inner vessel. Internal contamination shall not exceed the limits for surface contaminated objects as defined in 10 CFR §71.4.

(2) Maximum quantity of material per package

Seven fuel, knockout, or filter canisters or any combination thereof within the inner vessel. The radioactive decay heat load must not exceed 100 watts in each canister. The gross weight of each canister must not exceed 2,940 pounds.

(c) Transport Index for Criticality Control

Minimum transport index to be shown on label for nuclear criticality control: 100

6. The cask cavity and inner vessel must be dry when delivered to a carrier for transport, except for free water which may be present following drip drying of the canisters for a minimum of 2 minutes after removal from the storage pool. The canisters must be loaded and dewatered in accordance with Section 7.1.1 of the application which includes approximately 2 atm of argon, nitrogen, or helium cover gas. The cask cavity and inner vessel must be filled with argon, nitrogen, or helium at 1.0 atm pressure.

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

(a) Prior to each shipment, the inner and outer lid seals must be inspected. The seals must be replaced with new seals if inspection shows any defects or every 12 months, whichever occurs first; and

(b) Each package must meet the Acceptance Tests and Maintenance Program of Section 8.0 of the application.

(c) The package must be prepared for shipment and operated in accordance with Section 7.0 of the application.

8. For any canister containing water and/or organic substances which could radiolytically generate combustible gases, a determination must be made by tests and measurements or by analysis of a representative canister that the following criteria are met over a period of time that is twice the expected shipment time:

8. (continued)

The hydrogen generated must be limited to a molar quantity that would be no more than 5% by volume (or equivalent limits for other inflammable gases) of the canister gas void if present at STP (i.e., no more than 0.063 g-moles/ft^3 at 14.7 psia and 70°F); or that oxygen is limited to 5% by volume in those portions of the canister which could have hydrogen greater than 5%.

For any package delivered to a carrier for transport, the canister must be prepared for shipment in the same manner in which determination for gas generation is made. Shipment period begins when the canister is closed and must be completed within twice the expected shipment time.

9. Bolt torque:

The outer cask lid must be secured by 32, ASTM A320, Grade L43 (Cadmium plated), 1-1/2-6 UNC-2A x 5.5 long bolts torqued to 780-945 ft-lbs (lubricated).

The inner vessel lid must be secured by 24, ASTM A320, Grade L43 (Cadmium plated), 3/4-10 UNC-2A x 2.25 long bolts torqued to 130-158 ft-lbs (lubricated).

The upper and lower overpack limiters must each be secured by 8, ASTM A320, Grade L43 (Cadmium plated), 1-1/4-7 UNC-2A x 41.75 long bolts torqued to 225-270 ft-lbs (lubricated).

10. Except for the contents specified in 5.(b)(1)(iii), prior to each shipment, the shipper must confirm that the cask and inner vessel are properly sealed by tests as specified in Appendix 7.4 or Section 8.2.2 of the application. The test is satisfied if no leakage is detected using a test with a minimum sensitivity of 1×10^{-3} atm-cm^3/s.

11. The neoprene O-ring seals used in the containment vessel closure must be fabricated from neoprene material specified as Cascade Gaskets compound number CG 100-111-60.

12. The shipper may use a tarpaulin to cover the cask during time of transport.

13. The package authorized by the certificate is hereby approved for use under the general provisions of 10 CFR §71.12.

14. Expiration date: March 31, 2006.

REFERENCES

Nuclear Packaging, Inc. application dated April 6, 1991.

Supplements dated: April 9 and 15, 1991.
Department of Energy supplements dated: February 21, 1996; and February 1, 2001.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

M. Wayne Hodges for

E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date 30 March, 2001

NRC FORM 618 (8-2000) 10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9202	8	71-9202	USA/9202/B(U)F	1	OF	5

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)*

Department of Energy
Washington, DC 20585

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Transnuclear, Inc. application
dated January 19, 1989, as supplemented.

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-BRP

(2) Description

The TN-BRP is a right circular cylindrical cask designed for shipment of up to 85 BWR spent fuel assemblies. The total empty weight of the package is approximately 179,600 pounds. The payload capacity is approximately 43,170 pounds. The overall dimensions of the package, with impact limiters, are 244.5 inches long by 131 inches diameter. The cask body is 190.5 inches long by 83.25 inches in diameter. The cask has a cylindrical payload cavity which is 171 inches long and 64 inches in diameter. The volume of the cavity is approximately 185 cubic feet.

The containment vessel consists of a 9.62-inch thick forged steel (ASME SA-350; Grade LF3) cylindrical shell, with bottom plate and lid. The bottom plate and lid are made from 9.75-inch thick steel (ASME SA-350, Grade LF3). The 74.75-inch diameter lid is bolted to the cask with forty-eight, 1-5/8-inch diameter steel (ASME SA 540 Grade B24, Class 1) bolts. The cask is sealed with a viton O-ring mounted in a groove machined in the underside of the lid. The containment vessel is penetrated by access and vent ports in the lid, and two gas sampling ports and a research instrumentation port in the cask body.

The spent fuel assemblies are housed in a specially designed 44 compartment fuel basket. Each compartment can accommodate two BRP fuel assemblies stacked end-to-end. Peripheral inserts fabricated from an aluminum alloy are positioned between the fuel basket and cask cavity wall. Each fuel cell has a top and bottom end cap to confine damaged fuel.

5.(a) Packaging (continued)

(2) Description (continued)

The cask is provided with steel encased balsa-red wood impact limiters. The limiters have an outer diameter of 131 inches, an inner diameter of 91 inches, and a thickness of 20 to 26 inches. Each impact limiter is attached to the cask by four equally spaced 2.25-inch diameter bolts. The impact limiters are also connected to each other with fourteen 1.50-inch diameter tie rods.

The cask has four lifting lugs welded to the lid, and four lifting/ tiedown trunnions bolted to the cask body.

(3) Drawings

(i) The packaging is constructed in accordance with the following Transnuclear, Inc. Drawings:

3024-150-1, Rev. 5	Longitudinal Section
3024-150-2, Rev. 5	Transverse Section
3024-150-3, Rev. 2	Shell and Bottom
3024-150-4, Rev. 2	Lid
3024-150-5, Rev. 3	Trunnion
3024-150-6, Rev. 4	Front Impact Limiter
3024-150-7, Rev. 3	Rear Impact Limiter
3024-150-11, Rev. 3	Packaging Penetrations
3024-150-12, Rev. 3	Lid Bolt
3024-150-13, Rev. 7	Parts List
3024-150-14, Rev. 2	Trunnion Shoulder Bolt
3024-150-16, Rev. 1	Impact Limiters Spacers
3024-150-19, Rev. 3	Impact Limiter Tierods & Tierod Brackets
3024-150-26, Rev. 0	Front Impact Limiter & Tierod Bracket Assembly
3024-150-27, Rev. 0	Rear Impact Limiter & Tierod Bracket Assembly
3024-150-31, Rev. 0	Impact Limiter Attachment Bolt
3024-150-32, Rev. 0	Disc Spring at Impact Limiter

(ii) The fuel assembly basket is constructed in accordance with the following Transnuclear, Inc. Drawings:

3024-150-8, Rev. 1	Basket General Arrangement
3024-150-9, Rev. 1	Basket Typical Cross Section
3024-150-10, Rev. 1	Basket Plane View
3024-150-15, Rev. 0	Type A and B Spacers
3024-150-17, Rev. 2	Packaging Peripheral Inserts
3046-70-1, Rev. 1	Top Cap
3046-70-2, Rev. 4	Bottom Cap

5.(b) Contents

(1) Type and form of material

(i) Irradiated BWR uranium oxide fuel assemblies, damaged or intact, as described in the application and including the following specifications:

Assembly Type	Array	Pellet dia. (in.)	Clad Thickness	Rod OD (in.)	Pitch (in)	Mass (U) Kg
B	11x11	0.275/0.373	0.034/0.031	0.344/0.449	0.577	132.9
C	11x11	0.275/0.373	0.034/0.031	0.344/0.449	0.577	121.8
C thinclad	11x11	0.282/0.399	0.025/0.031	0.344/0.449	0.577	133.1
D54/D55	7x7	0.620	0.040	0.700	0.921	139.4
D52/D53	7x7	0.607/0.620	0.040	0.700	0.921	142.8
D51	8x8	0.500	0.035	0.570	0.807	118.4
D50	8x8	0.488/0.500	0.035	0.570	0.807	122.7
E	9x9	0.471	0.040	0.5625	0.707	141.2
F	9x9	0.471	0.040	0.5625	0.707	141.2
F-Pu	9x9	0.471	0.040	0.5625	0.707	141.2
Reload E-G	9x9	0.471	0.040	0.5625	0.707	141.2
Reload E-G/F	9x9	0.471	0.040	0.5625	0.707	141.2
Reload E-G/Pu	9x9	0.471	0.040	0.5625	0.707	141.2
Modified E-G	9x9	0.471	0.040	0.5625	0.707	141.2
EP	9x9	0.471	0.040	0.5625	0.707	123.0

The BWR fuel assemblies have a maximum burnup of 25,000 MWD/MTU.
The minimum cooling time for any assembly is fourteen years.

(2) Maximum quantity of material per package

(i) Eighty-five BWR assemblies.

(ii) Maximum decay heat per package not to exceed 6.39 kilowatts.
Maximum 103 watts per BWR assembly.

5.(b) Contents (continued)

(2) Maximum quantity of material per package (continued)

(iii) Above fuel assemblies to be positioned in the fuel baskets as shown in the drawings referenced in 5(a)(3)(ii), and as described in Chapter 7 of the application.

(c) Transport Index

Minimum transport index for nuclear criticality control: 0

Minimum transport index to be shown on label: 9

6. The surface temperature of the package must remain at or above -10°F during transport.

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

a. The packaging must be prepared for shipment and operated in accordance with the Operating Procedures of Chapter 7 of the application.

b. The packaging must meet the Acceptance Tests and Maintenance Program of Chapter 8 of the application.

c. The packaging must be loaded in accordance with Section 7.1.2 and Chapter 1 of the application.

8. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR §71.12.

9. Expiration Date: June 30, 2004.

10. This certificate authorizes a one-time shipment from the DOE West Valley Demonstration Project in West Valley, New York to the Idaho National Engineering & Environmental Laboratory. This certificate expires upon completion of the shipment, or by the above expiration date, whichever occurs first.

NRC FORM 618 (8-2000) 10 CFR 71	CERTIFICATE OF COMPLIANCE FOR RADIOACTIVE MATERIAL PACKAGES			U.S. NUCLEAR REGULATORY COMMISSION
a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE / PAGES
9202	8	71-9202	USA/9202/B(U)F	5 OF 5

REFERENCES

Transnuclear Inc. application dated January 19, 1989.

Supplements dated: March 22, 1989; December 19, 1990; March 4 and October 3, 1991; April 21 and November 7, 1994; April 27, 1999; April 27, October 12, October 27, November 14 and November 15, 2000; January 25, January 26, March 8 and October 11, 2001.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date 10/27/01

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9203	13	71-9203	USA/9203/AF	1	OF	4

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)*

Framatome ANP, Inc.
P.O. Box 11646
Lynchburg, VA 24506-1646

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Framatome Cogema Fuels application dated May 31, 1996, as supplemented.

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.: DHTF

(2) Description

The packaging consists of a 14-gauge stainless steel containment vessel, 9.5 inches by 9.5 inches by 17.5 inches high, with a bolted and gasketed top flange closure and stainless steel welded bottom plate. The containment vessel is centered and supported in a steel drum by industrial cane fiberboard of 16.5 ± 2 lbs/ft^3 density.

Closure of the containment vessel is maintained by a 3/8-inch thick carbon steel lid and 1/8-inch thick silicone rubber gasket secured with eight, 3/8-16NC by 1-1/2 long hex bolts and nuts. The 16-gauge steel outer drum is approximately 34 inches high and 22.5 inches in diameter. The drum closure is a 16-gauge lid with a 12-gauge bolt locking ring with drop forged lugs, one of which is threaded, having a 5/8-inch diameter bolt and lock nut.

The gross weight of the packaging and contents is 490 pounds.

(3) Drawings

The packaging is constructed and assembled in accordance with Framatome Cogema Fuels Drawing Nos. 1249874E, Rev. 5; 1259100C, Rev. 0; 1259101C, Rev. 0; and 1215600D, Rev. 6.

5.(b) Contents

(1) Type and form of material

Dry uranium oxide solid pellets, annular pellets, or scrap, packaged either on trays or bagged, as shown in Framatome Cogema Fuels 1215600D, Rev. 6.

(i) Solid pellets on stainless steel trays. The minimum pellet diameter is 0.315 inch and the maximum pellet diameter is 0.4075 inch.

(ii) Bagged solid pellets or scrap, or any combination. The maximum pellet diameter is 0.4075 inch.

(iii) Bagged solid pellets or scrap, or any combination. The maximum pellet diameter is 0.375 inch.

(iv) Bagged annular pellets. The minimum pellet diameter is 0.291 inch and the maximum pellet diameter is 0.304 inch, with an annulus from 0.045 to 0.065 inch in diameter.

(2) Maximum quantity of material per package

The maximum weight of contents and all packaging materials within the inner container is 275 lbs. The maximum quantity of polyethylene is 149 grams per pellet box.

(i) For the contents described in Item 5(b)(1)(i), enrichment and fissile quantities are limited as follows:

Max. Enrichment (wt % U-235)	Max. UO_2 mass (kg)	Max. U-235 mass (kg)	Max. Number Pellet Boxes
5.0	112	4.83	4

(ii) For the contents described in Item 5(b)(1)(ii), enrichment and fissile quantities are limited as follows:

Max. Enrichment (wt % U-235)	Max. UO_2 mass (kg)	Max. U-235 mass (kg)	Max. Number Pellet Boxes
5.0	84	3.62	3

(iii) For the contents described in Item 5(b)(1)(iii), enrichment and fissile quantities are limited as follows:

Max. Enrichment (wt % U-235)	Max. UO_2 mass (kg)	Max. U-235 mass (kg)	Max. Number Pellet Boxes
3.85	112	3.72	4

5.(b) (2) Maximum quantity of material per package (Continued)

(iv) For the contents described in Item 5(b)(1)(iv), enrichment and fissile quantities are limited as follows:

Max. Enrichment (wt % U-235)	Max. UO_2 mass (kg)	Max. U-235 mass (kg)	Max. Number Pellet Boxes
5.0	84	3.55	3
3.75	112	3.55	4

(c) Transport Index for Criticality Control

Minimum transport index to be shown on label for nuclear criticality control: 1.2

6. Each package must have a stainless steel plate (spacer) positioned between pellet boxes, as shown on Framatome Cogema Fuels Drawing No. 1249874E, Rev. 4.

7. For packages containing fewer than four loaded pellet boxes, solid aluminum spacer blocks, as shown on Framatome Cogema Fuels Drawing No. 1259100C, Rev. 0, must be substituted for all missing boxes.

8. For contents described in Item 5(b)(1)(i) and limited in Item 5(b)(2)(i), stainless steel trays must be positioned between each layer of pellets, and on the top and bottom of the pellet stack. Additional trays must be inserted in partially filled pellet boxes to provide a snug fit.

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

(a) Prior to each shipment the containment vessel gasket must be inspected. The gasket must be replaced if the inspection shows any defects or signs of degradation.

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

(c) Each packaging must meet the Acceptance Tests and Maintenance Program of Chapter 8 of the application, as supplemented October 29, 1999.

10. The eight, 3/8-inch containment vessel bolts must be torqued to 35 ft-lbs ± 10% and the 5/8-inch closure ring bolt and lock nut must be torqued to 70 ft-lbs ± 10%. Immediately following each loading of a package, the closure ring must be inspected to assure it is fully seated (engaged).

11. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.12.

12. Expiration date: February 28, 2006.

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9203	13	71-9203	USA/9203/AF	4	OF	4

REFERENCES

Framatome Cogema Fuels application dated May 31, 1996.

Supplements dated: August 15, and September 9 and 10, 1996; September 26 and October 9, 1997; March 5, April 28, and May 8, 1998; October 29, 1999; November 13 and December 20, 2000; and February 6 and 9, 2001.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

E. William Brach

E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: March 18, 2002

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE	PAGES
9204	8	71-9204	USA/9204/B(U)-85	1 OF	5

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)*

Duratek
140 Stoneridge Drive
Columbia, SC 29210

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Chem-Nuclear Systems, LLC, application dated March 22, 2000, as supplemented.

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.: CNS 10-160B

(2) Description

A cylindrical carbon steel and lead shielded shipping cask, designed to transport radioactive waste material. The cask is transported in the upright position and is equipped with steel encased, rigid polyurethane foam impact limiters on the top and bottom. The package has approximate dimensions, shielding, and weight as follows:

Cask height	88 inches
Cask outer diameter	78-1/2 inches
Cask cavity height	77 inches
Cask cavity diameter	68 inches
Overall package height, with impact limiters	130 inches
Overall package diameter, with impact limiters	102 inches
Lead shielding thickness	1-7/8 inches
Gross weight (packaging and contents)	72,000 lbs
Maximum total weight of contents, shoring, secondary containers, and optional shield insert	14,500 lbs

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

The cask body consists of a 1-1/8-inch thick carbon steel (ASME SA516 or SA537) inner shell, a 1-7/8-inch thick lead gamma shield, and a 2-inch thick carbon steel outer shell (ASME SA516). The inner and outer shells are welded to a 5-1/2-inch thick carbon steel bottom plate. The cask cavity has an optional 11-gage stainless steel liner. A 12-gage stainless steel thermal shield surrounds the cask outer shell in the region between the impact limiters. The impact limiters are secured to each other around the cask by eight ratchet binders.

The cask lid is a 5-1/2-inch thick carbon steel plate, and has a 31-inch diameter opening equipped with a secondary lid. The primary lid is sealed with a double silicone O-ring and 24 equally spaced 1-3/4-inch diameter bolts. The secondary lid is 46 inches in diameter, is centered within the primary lid, and is sealed to the primary lid by a double silicone O-ring and 12 equally spaced 1-3/4-inch diameter bolts. The space between the double O-ring seals is provided with a test port for leak testing the primary and secondary lid seals.

The optional cask drain and vent ports are sealed with a plug and an O-ring seal.

The package is equipped with four tie-down lugs welded to the cask outer shell. Two lifting lugs and two redundant lifting lugs are removed during transport. The lid is equipped with three lifting lugs which are covered by the top impact limiter and rain cover during transport.

An optional carbon steel shield insert may be used within the cask cavity.

(3) Drawings

The packaging is constructed and assembled in accordance with Chem-Nuclear Systems Drawing No. C-110-D-29003-010, Sheets 1 through 5, Rev. 12.

An optional shield insert is constructed in accordance with Chem-Nuclear Systems Drawing No. C-119-B-0018, Rev. 1.

(b) Contents

(1) Type and form of material

(i) Byproduct, source, and special nuclear material in the form of solids, dewatered resins or process solids, or solidified waste, contained within secondary containers. Explosives, corrosives, non-radioactive pyrophorics, and compressed gases are prohibited. Pyrophoric radionuclides may be present only in residual amounts less than 1 weight percent. The total amount of potentially volatile organic compounds present in the headspace of a secondary container is restricted to 500 parts per million; or

(ii) Radioactive material in the form of activated reactor components.

5.(b) (2) Maximum quantity of material per package

Type B quantity of radioactive material, not to exceed 3,000 times a Type A quantity. Decay heat not to exceed 100 watts. Total weight of contents, shoring, secondary containers, and optional shield insert not to exceed 14,500 pounds. Contents may include fissile material contaminants provided the mass limits of 10 CFR 71.53 are not exceeded. Plutonium content not to exceed 0.74 TBq (20 curies).

6. Except for close fitting contents, shoring must be placed between the secondary containers or activated components and the cask cavity to prevent movement during accident conditions of transport.

7. The cask primary lid must be secured by 24, and the secondary lid by 12, 1-3/4"-8UNC x 5-3/8" long hex cap screws with a flat washer, torqued to 300 ft-lbs ± 30 ft-lbs (lubricated). The optional drain and vent port plugs must be torqued to 20 ± 2 ft-lbs.

8. Lift lugs must be removed from the cask body prior to transport.

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

(a) Each packaging must meet the Acceptance Tests and Maintenance Program of Chapter 8 of the application; and

(b) The package must be prepared for shipment and operated in accordance with the Operating Procedures of Chapter 7 of the application; and

(c) The primary lid, secondary lid, and the optional vent and drain seals must be replaced with new seals if inspection shows any defects or every 12 months, whichever occurs first.

10. The package must be leak tested as follows:

(a) Prior to each shipment, the package must be leak-tested in accordance with Section 8.2.2.2 of the application. For contents that meet the definition of low specific activity material or surface contaminated objects in 10 CFR 71.4, and also meet the exemption standard for low specific activity material and surface contaminated objects in 10 CFR 71.10(b)(2), the pre-shipment leak-test is not required.

(b) The packaging containment system must be leak tested in accordance with Section 8.1.3 of the application prior to first use of any packaging, after the third use, within the twelve month period prior to each use, and after seal replacement.

11. (a) For any package containing water and/or organic substances which could radiolytically generate combustible gases, a determination must be made by tests and measurements or by analysis of a representative package that the following criteria are met over a period of time that is twice the expected shipment time:

(1) The hydrogen generated must be limited to a molar quantity that would be no more than 5% by volume (or equivalent limits for other inflammable gases) of the secondary container gas void if present at STP (i.e., no more than 0.063 g-moles/ft^3 at 14.7 psia and 70°F); or

(2) The secondary container and cask cavity must be inerted with a diluent to assure that oxygen is limited to 5% by volume in those portions of the package which could have hydrogen greater than 5%.

For any package delivered to a carrier for transport, the secondary container must be prepared for shipment in the same manner in which determination for gas generation is made. Shipment period begins when the package is prepared (sealed) and must be completed within twice the expected shipment time.

(b) For any package containing materials with a radioactivity concentration not exceeding that for low specific activity material, and shipped within 10 days of preparation, or within 10 days after venting of drums or other secondary containers, the determination in (a) above need not be made, and the time restriction in (a) above does not apply.

(c) For any package containing RH-TRU the following additional conditions apply:

(1) Waste content codes and classification, physical form, chemical properties, chemical compatibility, gas distribution, and pressure buildup, container and contents configuration, isotopic characterization and fissile content, must be determined and limited in accordance with Appendix 4.10.2 of the application; and

(2) Each waste container must not exceed the decay heat limits in Section 10 of the applicable site specific appendix to Appendix 4.10.2, or must satisfy the requirements of Attachment B, "Methodology for Determination of Decay Heats and Hydrogen Gas Generation Rates for Remote Handled Transuranic Content Codes," for each site specific appendix to Appendix 4.10.2; and

(3) One or more filter vents must be installed in the 55-gallon drum payload container and any sealed secondary containers overpacked in the payload container. Filter vents must meet the minimum specifications in Section 8, "Payload Container and Contents Configuration" of the applicable site specific appendix to Appendix 4.10.2; and

(4) The payload container authorized for shipment of RH-TRU in the Model No. CNS 10-160B is the 55-gallon drum. Up to ten 55-gallon drums of RH-TRU waste may be packaged in the cask.

NRC FORM 618 (8-2000) 10 CFR 71	CERTIFICATE OF COMPLIANCE FOR RADIOACTIVE MATERIAL PACKAGES			U.S. NUCLEAR REGULATORY COMMISSION
a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE / PAGES
9204	8	71-9204	USA/9204/B(U)-85	5 OF 5

12. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.12.

13. Expiration date: October 31, 2005.

REFERENCES

Chem-Nuclear Systems, LLC, application dated March 22, 2000.

Supplements dated May 10 and November 7, 2000; and January 5 and April 13, 2001.

Duratek supplements dated April 23 and July 24, 2001, and June 14, 2002.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

M. Wayne Hodges for

E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: October 15, 2002

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9206	8	71-9206	USA/9206/B(U)F	1	OF	4

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)*

   Department of Energy
   Washington, DC 20585

   b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

   Transnuclear, Inc. application
   dated September 1, 1989, as supplemented.

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-REG

(2) Description

The TN-REG package is a cylindrical steel cask designed for shipment of up to 40 PWR spent fuel assemblies. The package, with impact limiters attached, is approximately 234 inches long and 131 inches in diameter. The total empty weight of the package is approximately 181,000 pounds. The maximum weight of the contents, including the fuel basket assemblies and end caps, is approximately 52,360 pounds. The cask is transported in a horizontal orientation on a specially designed shipping frame.

The containment vessel consists of a 9.25-inch thick forged steel (ASME SA-350; Grade LF3) cylindrical shell and lid. The lid is approximately 82.25 inches in diameter and has a maximum thickness of 8.5 inches. The lid is bolted to the cask with forty-eight 1-5/8 inch steel (ASME SA-540, Grade B24, Class 1) bolts. The cask is sealed with a Viton O-ring mounted in a groove machined in the underside of the lid. A second metallic O-ring is provided to leak test the Viton O-ring. The containment vessel is penetrated by access and vent ports in the lid, and two gas sampling ports and a research instrumentation port in the cask body.

The spent fuel assemblies are positioned within a 40 compartment fuel basket. Each compartment can accommodate a single PWR assembly. Peripheral inserts fabricated from an aluminum alloy are positioned between the fuel basket and cask cavity wall. Each fuel cell has a top and bottom end cap to confine damaged fuel.

5.(a)(2) Continued

The cask is equipped with impact limiters made of balsa and redwood encased in carbon steel shells. The impact limiters have an outer diameter of 131 inches, an inner diameter of 91 inches, and a thickness ranging from 20 to 26 inches. Each impact limiter is attached to the cask by four 2.25-inch diameter bolts. The impact limiters are also connected to each other with fourteen 1.5-inch diameter tie rods.

The cask has four lifting lugs welded to the lid, and four lifting/tie down trunnions bolted to the cask body.

(3) Drawings

(i) The packaging is constructed in accordance with the following Transnuclear, Inc. Drawings:

3024-150-6, Rev. 4	Front Impact Limiter
3024-150-7, Rev. 3	Rear Impact Limiter
3024-150-11, Rev. 3	Packaging Penetrations
3024-150-12, Rev. 3	Lid Bolt
3024-150-19, Rev. 3	Impact Limiter Tierods and Tieroad Brackets
3024-150-21, Rev. 4	Longitudinal Section
3024-150-22, Rev. 3	Transverse Sections
3024-150-23, Rev. 1	Shell and Bottom
3024-150-24, Rev. 1	Lid
3024-150-25, Rev. 1	Trunnion
3024-150-26, Rev. 0	Front Impact Limiter and Tierod Bracket Assembly
3024-150-27, Rev. 0	Rear Impact Limiter and Tierod Bracket Assembly
3024-150-31, Rev. 0	Impact Limiter Attachment Bolt
3024-150-32, Rev. 0	Disc Spring at Impact Limiter
3024-150-33, Rev. 3	Parts List
3024-150-36, Rev. 1	Impact Limiter Front Spacer

(ii) The fuel basket assembly is constructed in accordance with the following Transnuclear, Inc. Drawings:

3024-150-28, Rev. 0	Basket-General Arrangement
3024-150-29, Rev. 1	Basket-Typical Cross Section
3024-150-30, Rev. 0	Basket-Plan View
3024-150-37, Rev. 2	Peripheral Inserts
3046-70-3, Rev. 1	Top Cap
3046-70-4, Rev. 4	Bottom Cap

(iii) The poison rod assemblies are constructed in accordance with the following Transnuclear, Inc. Drawing:

3024-150-34, Rev. 0 Fuel Assemblies-B_4C Poison Rod Assembly

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9206	8	71-9206	USA/9206/B(U)F	3	OF	4

5.(b) Contents

(1) Type and form of material

(i) Irradiated PWR uranium oxide fuel assemblies, damaged or intact, as described in the application and including the following specifications:

Fuel form	UO_2 pellets
Nominal pellet diameter	0.367 inch
Cladding material	Zircaloy
Cladding thickness	0.024 inch
Maximum fuel rod length	162 inches
Maximum active fuel rod length	144 inches
Assembly array	14 x 14
Maximum initial fuel pin pressure at 70°F	l atm
Maximum initial U^{235} enrichment	3.5% w/o
Initial uranium loading	382.18 kg

The PWR fuel assemblies have a maximum burnup of 15,000 MWD/MTU. The minimum cooling time for any assembly is 17 years. Thirty-eight of the forty fuel assemblies contain either a burnable poison assembly or a control rod assembly.

(2) Maximum quantity of material per package

(i) Maximum of forty PWR fuel assemblies.
(ii) Maximum decay heat per package not to exceed 4.16 kilowatts. Maximum 135 watts per PWR assembly.
(iii) Above fuel assemblies to be positioned in the fuel baskets as shown in the drawings referenced in 5(a)(3)(ii), and as described in Chapter 7 of the application.

(3) Transport Index

Minimum transport index for nuclear criticality control: 0

Minimum transport index to be shown on label: 10

6. The surface temperature of the package must remain at or above -10°F during transport. |

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

(a) Each packaging must be prepared for shipment and operated in accordance with the operating procedures in Chapter 7 of the application. After loading, the cask must be vacuum dried and backfilled with nitrogen at one atmosphere as described in Chapter 7 of the application.

7. Continued

    (b) Each packaging must meet the Acceptance Tests and Maintenance Program of Chapter 8 of the application.

    (c) The packaging must be loaded in accordance with Section 7.1.2 and Chapter 1 of the application.

8. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR §71.12.

9. Expiration Date: May 31, 2005.

10. This certificate authorizes a one-time shipment from the DOE West Valley Demonstration Project in West Valley, New York to the Idaho National Engineering & Environmental Laboratory. This certificate expires upon completion of the shipment, or by the above expiration date, whichever occurs first.

REFERENCES

Transnuclear, Inc. application dated September 1, 1989.

Supplements dated: March 7 and October 22, 1990; January 7 and February 11, 1991; November 7, 1994; March 2 and 15, 1995; February 8, 1999; March 30, April 27, October 12, October 27, November 14 and November 15, 2000; January 25, January 26, March 8 and October 11, 2001.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date 10/27/01

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)*

   ATG Nuclear Services, LLC
   1550 Bear Creek Road
   Kingston, TN 37763

   b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

   Allied Technology Group, Inc., application dated May 31, 2002.

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.: 10-142

(2) Description

Steel encased, lead shielded cask for solid radioactive material. The overall dimensions of the cask and impact limiters are 112-inch diameter by 130-inch height. The cask consists of two concentric carbon steel cylindrical shells surrounding a 3-1/2-inch thick lead shield. The ½-inch thick inner shell has a 66-inch ID, and the 1-inch thick outer shell has a 76-inch OD. The base consists of two, 3-inch thick welded steel plates of 66- and 74-inch diameters. The base is welded to the steel cylindrical shells. A stepped welded lid, secured by 16, 1-1/2-6 UNC-2A bolts or studs and nuts, is comprised of two, 3-inch thick steel plates containing an opening for a secondary lid of similar construction with one additional 1-inch thick upper plate. Within the primary lid there is a 16-inch or 29-inch centered secondary lid. The 16-inch secondary lid is secured by 8, 7/8-inch bolts or studs and nuts, and the 29-inch secondary lid is secured by 16, 1-1/4-inch bolts or studs and nuts. The lids are sealed with a solid silicone flat gasket. The containment cavity is 66 inches in diameter by 72 inches high. A plugged drain port is located at the cask bottom and the lid is provided with a plugged test port. Toroidal impact limiters are located at the top and bottom of the cask. The impact limiters are 10-gauge steel sheets filled with rigid polyurethane and are equipped with plastic plugs. As an option, interior and exterior surfaces of the cask body and interior surfaces of the upper lid may be covered with 12-gauge 304 stainless steel cladding and seal welded.

All exposed side walls are covered with a stainless steel thermal barrier. Four skewed lugs, welded to the outer shell are used for tie-down. The package gross weight is approximately 68,000 pounds.

5.(a) Packaging (Continued)

(3) Drawings

The packaging is constructed and assembled in accordance with ATG Nuclear Services, Inc., Drawing No. X-103-110-SNP, Sheets 1 through 5, Rev. E.

(b) Contents

(1) Type and form of material

(i) Dewatered, solid, or solidified waste which may be in secondary containers;

(ii) Activated components which may be in secondary containers;

(iii) Dewatered, solid or solidified material, meeting the requirements for low specific activity material, which may be in secondary containers; or

(iv) Dewatered or solidified ion exchange resin from light water reactors, in secondary containers.

(2) Maximum quantity of material per package

Decay heat not to exceed 400 watts. Fissile materials not to exceed the limits of 10 CFR 71.53. Maximum weight of contents, including dunnage and secondary containers, not to exceed 10,000 pounds.

For the contents specified in 5(b)(1)(i) and 5(b)(1)(ii):

Not to exceed a Type A quantity of transuranic materials.

6.(a) For any package containing water and/or organic substances which could radiolytically generate combustible gases, determination must be made by tests and measurements or by analysis of a representative package such that the following criteria are met over a period of time that is twice the expected shipment time:

(1) The hydrogen generated must be limited to a molar quantity that would be not more than 5% by volume (or equivalent limits for other inflammable gases) of the secondary container gas void if present at STP (i.e., no more than 0.063 g-moles/ft^3 at 14.7 psia and 70°F); or

(2) The secondary container and cask cavity must be inerted with a diluent to assure that oxygen must be limited to 5% by volume in those portions of the package which could have hydrogen greater than 5%.

For any package to be delivered to a carrier for transport, the secondary container must be prepared for shipment in the same manner in which determination for gas generation is made. Shipment period begins when the package is prepared (sealed) and must be completed within twice the expected shipment time.

(b) For any package containing materials with radioactivity concentration not exceeding that for low specific activity material, and shipped within 10 days of preparation, or within 10 days after venting of drums or other secondary containers, the determination in (a) above need not be made, and the time restriction in (a) above does not apply.

7. Except for close fitting contents, dunnage must be provided in the shipping cask cavity sufficient to prevent significant movement of the contents or secondary containers relative to the outer packaging under normal condition.

8. Bolt/Stud and Nut Torque:

The primary cask lid bolts or studs and nuts must be torqued to 300 ± 25 ft-lbs (lubricated).

The secondary cask lid bolts or studs and nuts must be torqued to 200 ± 10 ft-lbs (lubricated).

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

   (a) Prior to each shipment, the packaging seals must be inspected. The seals must be replaced with new seals if inspection shows any defects or every 12 months, whichever occurs first. Cavity drain and test ports must be sealed with appropriate sealant applied to the pipe plug threads.

   (b) The package must be prepared for shipment and operated in accordance with the Operating Procedures in Section 7.0 of the application.

   (c) Each package must meet the Acceptance Tests and Maintenance Program in Section 8.0 of the application.

   (d) For contents that meet the definition of low specific activity material or surface contaminated objects in 10 CFR 71.4, and also meet the exemption standard for low specific activity material and surface contaminated objects in 10 CFR 71.10(b)(2), the pre-shipment leak test is not required.

10. Use of intumescent coating fire shield is not authorized.

1. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.12.

12. Expiration date: August 31, 2007.

REFERENCES

Allied Technology Group, Inc., application dated May 31, 2002.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

E. William Brach

E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: August 6, 2002

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIALS PACKAGES

U.S. NUCLEAR REGULATORY COMMISSION

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. PACKAGE IDENTIFICATION NUMBER	d. PAGE NUMBER	e. TOTAL NUMBER PAGES
9210	5	USA/9210/B( )	1	3

2. PREAMBLE

a. This certificate is issued to certify that the 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)*

ATG Nuclear Services, LLC
669 Emory Valley Road
Oak Ridge, TN 37830

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION:

Scientific Ecology Group, Inc., application dated October 26, 1993, as supplemented.

c. DOCKET NUMBER 71-9210

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.: 10-135B

(2) Description

Steel encased, lead shielded cask for solid radioactive material. The overall dimensions of the cask are 112-inch diameter by 130-inch height. The cask consists of two concentric carbon steel cylindrical shells surrounding a 3-1/2-inch thick lead shield. The ½-inch thick inner shell has a 66-inch ID, and the 1-inch thick outer shell has a 76-inch OD; the base consists of two, 3-inch thick welded steel plates of 66- and 74-inch diameters. The base is welded to the steel cylindrical shells by a combination of fillet and full penetration groove welds. The top of the cask is provided with a primary lid and a secondary lid. The primary lid is of a stepped construction which is made of two, 3-inch thick steel plates of 76-inch diameter and 66-inch diameter joined together to form an integral 6-inch thick lid. The primary lid is secured to the cask body through 16, 1-1/2 - 6 UNC high strength bolts. The secondary lid which covers the 29-inch diameter hole at the center of the primary lid is also of stepped construction consisting of two, 3-inch thick plates. The secondary lid is secured to the primary lid through 16, 1-1/4 - 7 UNC high strength bolts. High temperature silicone gaskets are provided at the cask-primary lid and the primary lid-secondary lid interfaces. The latter is also provided with an additional Neoprene seal.

Two impact limiters are located at the top and bottom of the cask. The impact limiters are 10-gauge stainless steel shells filled with rigid polyurethane. The inner surfaces of the cask and the lid are clad with 12-gauge 304-stainless steel. The portion of the cask body that is not covered by the impact limiters is covered with a 10-gauge 304-stainless steel thermal shield. There is a 1/4-inch gap between the shell and the thermal shield which is maintained using 1/4-inch spacers.

The package gross weight is limited to 68,000 pounds.

(3) Drawings

The packaging is constructed in accordance with Scientific Ecology Group, Inc., Drawing No. STD-02-106, Sheets 1 and 2, Rev. 1.

Page 2 - Certificate No. 9210 - Revision No. 5 - Docket No. 71-9210

(b) Contents

(1) Type and form of material:

(i) Dewatered, solid, or solidified waste in secondary containers;

(ii) Activated solid components in secondary containers; or

(iii) Dewatered or solidified ion exchange resins from light water reactors, in secondary containers.

(2) Maximum quantity of material per package:

Greater than Type A quantities of radioactive materials which may contain fissile quantities limited to the amounts as exempted under 10 CFR §71.53. Not to exceed a Type A quantity of transuranic materials except for the contents specified in 5(b)(1)(iii) and materials of low specific activity. Internal decay heat not to exceed 400 watts and the maximum weight of contents including secondary containers not to exceed 10,000 pounds.

6. (a) For any package containing water and/or organic substances which could radiolytically generate combustible gases, determination must be made by tests and measurements or by analysis of a representative package such that the following criteria are met over a period of time that is twice the expected shipment time:

(1) The hydrogen generated must be limited to a molar quantity that would be no more than 5% by volume (or equivalent limits for other inflammable gases) of the secondary container gas void if present at STP (i.e., no more than 0.063 g-moles/ft^3 at 14.7 psia and 70°F); or

(2) The secondary container and cask cavity must be inerted with a diluent to assure that oxygen must be limited to 5% by volume in those portions of the package which could have hydrogen greater than 5%.

For any package delivered to a carrier for transport, the secondary container must be prepared for shipment in the same manner in which determination for gas generation is made. Shipment period begins when the package is prepared (sealed) and must be completed within twice the expected shipment time.

(b) For any package containing materials with radioactivity concentration not exceeding that for low specific activity material, and shipped within 10 days of preparation, or within 10 days after venting of drums or other secondary containers, the determination in (a) above need not be made, and the time restriction in (a) does not apply.

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

(a) The package must meet the Acceptance Test and Maintenance Program of Section 8.0 of the application, as supplemented.

(b) The package shall be prepared for shipment and operated in accordance with the Operating Procedures of Section 7.0 of the application, as supplemented.

8. The containment vessel must be leak tested to 1.3 x 10^{-6} atm-cm^3/sec (at the standard conditions of ANSI N14.5):

   (a) Prior to the first use of each package;

   (b) After the package's third use;

   (c) Within twelve months of the last leak test; and

   (d) Whenever gaskets are replaced.

9. Prior to each shipment, the containment system shall be leak tested to 5.0 x 10^{-3} atm-cm^3/sec (at the standard conditions of ANSI N14.5) to verify that it has been properly assembled. For contents that meet the definition of low specific activity material or surface contaminated objects in 10 CFR §71.4, and also meet the exemption standard for low specific activity material and surface contaminated objects in 10 CFR §71.10(b)(2), the pre-shipment leak test is not required.

10. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR §71.12.

11. Expiration date: January 31, 2005

REFERENCES

Scientific Ecology Group, Inc., application dated October 26, 1993.

Supplements dated: April 5 and October 31, 1994.

Molten Metal Technology, Inc., supplement dated February 24, 1998.

ATG Nuclear Services, LLC, supplements dated: December 1, 1998; August 9 and 11, 1999; and November 30, 1999.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

E. William Brach

E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: February 4, 2000

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9212	2	71-9212	USA/9212/B(M)F-85	1	OF	4

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)*
    Department of Energy
    Washington, DC 20585

    b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION
    Westinghouse Electric Corporation application dated December 20, 1996, as supplemented.

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: RH-TRU 72-B

(2) Description

A stainless steel, lead-shielded cask designed to provide double containment for shipment of transuranic waste materials. The packaging consists of a cylindrical stainless steel and lead cask body, a separate inner stainless steel vessel, and foam-filled impact limiters at each end of the cask body.

The cask body (outer cask) consists of a 1 1/2-inch thick, 41 5/8-inch outer diameter stainless steel outer shell, and a 1-inch thick, 32 3/8-inch inside diameter stainless steel inner shell, with 1 7/8 inches of lead shielding between the two shells. The cask bottom is 5-inch thick stainless steel plate. The cask is closed by a 6-inch thick stainless steel lid, and 18, 1 1/4-inch diameter bolts. The main closure lid has a double bore-type O-ring seal. The containment seal is the inner butyl O-ring seal, which is leak testable. The cask lid has a single vent/sampling port that is sealed with leak testable butyl O-ring seals.

The separate inner vessel consists of a 3/8-inch thick, 32-inch outside diameter stainless steel shell, and a 1 1/2-inch thick stainless steel bottom plate. The inner vessel is closed by a 6 1/2-inch thick stainless steel lid, and eight, 7/8-inch diameter bolts. The inner vessel closure lid has three bore-type O-ring seals. The containment seal is the middle butyl O-ring seal, which is leak testable. The inner vessel lid has a helium backfill port and a combination vent/sampling port that are sealed with leak-testable butyl O-ring seals.

A polyurethane foam-filled stainless steel impact limiter is attached to each end of the cask body using six, 1 1/4-inch diameter bolts. The radioactive contents are packaged within a stainless or carbon steel waste canister that is placed in the inner vessel.

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

The approximate dimensions and weights of the package are as follows:

Overall package length	187 3/4 inches
Impact limiter diameter	76 inches
Cask length	141 3/4 inches
Cask outer diameter (OD)	41 5/8 inches
Inner vessel length	130 inches
Inner vessel OD	32 inches
Cask lead shield thickness	1 7/8 inches
Maximum package weight (including contents)	45,000 pounds
Maximum weight of contents (including waste canister)	8,000 pounds

(3) Drawings

The packaging is constructed and assembled in accordance with Packaging Technology Drawing No. X-106-500-SNP, Sheets 1-9, Rev. 3.

The fixed lid waste canister is constructed and assembled in accordance with Packaging Technology Drawing No. X-106-501-SNP, Rev. 3. The removable lid waste canister is constructed and assembled in accordance with Packaging Technology Drawing No. X-106-502-SNP, Rev. 1.

(b) Contents

(1) Type and form of material

Byproduct, source, and special nuclear material in the form of dewatered, solid or solidified materials and waste, within the stainless or carbon steel waste canister described in Item 5(a)(3). Explosives, corrosives (pH less than 2 or greater than 12.5), and compressed gases are prohibited. Within a waste canister radioactive and non-radioactive pyrophorics must not exceed 1 weight percent. Flammable volatile organics are limited along with hydrogen to ensure the absence of flammable gas mixtures in RH-TRU waste payloads as described in Section 5.0 of Appendix 1.3.7, Rev. 3, June 2002, of the application.

(2) Maximum quantity of material per package.

Not to exceed 8,000 pounds, including the weight of the waste canister.

Fissile material not to exceed 325 grams Pu-239 equivalent for RH-TRU waste containers containing materials in which the form or distribution of the fissile radionuclides are not restricted as described in Section 3.1, "Nuclear Criticality" of Appendix 1.3.7, Rev. 3, June 2002, of the application. Pu-239 equivalent is determined in accordance with Section 3.0 of Appendix 1.3.7, Rev. 3, June 2002, of the application. Low enriched uranium is authorized for waste containers containing material that is primarily uranium (in terms of heavy metal component) and the waste matrix is distributed within the canister in such a manner that the maximum enrichment does not exceed 0.96% uranium (U-235) fissile equivalent mass in any location of the waste material.

Maximum decay heat per package not to exceed 50 watts for organic wastes and 300 watts for inorganic waste, and not to exceed the limits in Section 5.2, "Decay Heat" of Appendix 1.3.7, Rev. 3, June 2002, of the application.

(c) Transport Index for Criticality Control (Criticality Safety Index)

Minimum transport index to be shown on label for nuclear criticality control: 0.0

6. Waste content codes and classification, physical form, chemical properties, chemical compatibility, gas generation, fissile content, decay heat, isotopic inventory, weight, and radiation dose rate must be determined and limited in accordance with Appendix 1.3.7, Rev. 3, June 2002, of the application "Remote-Handled Transuranic Waste Authorized Methods for Payload Control (RH-TRAMPAC)."

7. Each waste canister must not exceed the decay heat limits in Section 5.2 of Appendix 1.3.7, Rev. 3, June 2002, of the application, or must be tested for gas generation in accordance with Appendix 1.3.7, Rev. 3, June 2002, of the application, Section 5.0, "Gas Generation Requirements."

8. A RH-TRU waste canister may be comprised of inner containers with different content codes provided that the hydrogen gas generation rate limit or decay heat limit for all of the inner containers within the payload is assumed to be the same as the content code with the lowest hydrogen gas generation rate limit or decay heat limit.

9. The waste canister and any sealed secondary containers greater than 4 liters in size overpacked in the waste canister must be vented in accordance with the minimum specifications in Appendix 1.3.5 of the application "Specification for Filter Vents."

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

(a) Each package must be prepared for shipment and operated in accordance with the procedures described in Chapter 7.0, "Operating Procedures," of the application, as supplemented.

(b) Each packaging must be tested and maintained in accordance with the procedures described in Chapter 8.0, "Acceptance Tests and Maintenance Program," of the application, as supplemented.

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9212	2	71-9212	USA/9212/B(M)F-85	4	OF	4

11. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.12.

12. Expiration date: February 28, 2005.

REFERENCES

Westinghouse Electric Corporation, application dated December 20, 1996.

Supplements dated: March 26 and August 23, 1999, November 14, 2000, January 25 and August 29, 2001, and June 14 and November 27, 2002.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

John D. Monninger, Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date December 27, 2002

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIALS PACKAGES

U.S. NUCLEAR REGULATORY COMMISSION

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. PACKAGE IDENTIFICATION NUMBER	d. PAGE NUMBER	e. TOTAL NUMBER PAGES
9215	5	USA/9215/B(U)	1	3

2. PREAMBLE

a. This certificate is issued to certify that the 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)*

Neutron Products, Inc.
22301 Mt. Ephraim Road
P.O. Box 68
Dickerson, MD 20842

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION:

Neutron Products, Inc. application dated September 14, 1992, as supplemented.

c. DOCKET NUMBER 71-9215

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.: NPI-20WC-6 MkII

(2) Description

A steel encased, lead shielded cask contained within a DOT Specification 20WC-6 wooden overpack. The cask is 24 inches in diameter with a 3/8-inch thick steel spherical shell and a cavity formed by an 8-1/4-inch ID by 3/16-inch thick steel tube. Positive closure of the shielded cask is accomplished by bolted end covers at each end of the cavity. The maximum package gross weight is 6,000 pounds.

(3) Drawings

The Model No. NPI-20WC-6 MkII packaging is constructed in accordance with Neutron Products, Inc. Drawing Nos. 240116, Rev. D, and 240122, Sheet 1 of 2, Rev. G, Sheet 2 of 2, Rev. -.

(b) Contents

(1) Type and form of material

Cobalt-60 as sealed sources which meet the requirements of special form radioactive material.

(b) Contents (Continued)

(2) Maximum quantity of material per package

(i) For sources contained within drum assembly shown as Item 5 on Neutron Products, Inc. Drawing No. 240122, Sheet 1 of 2, Rev. G:

Maximum activity not to exceed 15,000 curies, maximum decay heat not to exceed 240 watts.

(ii) For sources contained within drum assembly shown as Item 4 on Neutron Products, Inc. Drawing No. 240122, Sheet 2 of 2, Rev. -:

Maximum activity not to exceed 9,500 curies, maximum decay heat not to exceed 150 watts.

(iii) For sources contained within drum assembly shown as Item 2 on Neutron Products, Inc. Drawing No. 240122, Sheet 2 of 2, Rev. -:

Maximum activity not to exceed 6,300 curies, maximum decay heat not to exceed 100 watts.

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

(a) The package must meet the Acceptance Tests and Maintenance Program of Section 8.0 of the application.

(b) The package shall be prepared for shipment and operated in accordance with the Operating Procedures of Section 7.0 of the application.

7. The contents must be secured in the drum assembly so as to restrict movement in any direction to less than 0.25 inch, by lead, steel, or tungsten full diameter plugs and spacers.

8. The gross weight of the package must not exceed 6,000 pounds, and the inner shield cask shall be snug-fitting with the wooden overpack.

9. The packaging authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR §71.12.

10. Expiration date: October 31, 2002.

REFERENCES

Neutron Products, Incorporated application dated September 14, 1992.

Supplements dated: October 29, 1992; November 17, 1993; and September 8, 1997.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Cass R. Chappell

Cass R. Chappell, Chief
Package Certification Section
Spent Fuel Project Office
Office of Nuclear Material
Safety and Safeguards

Date: 10/30/97

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9216	7	71-9216	USA/9216/B()F	1	OF	4

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)*

Duratek
140 Stoneridge Drive
Columbia, SC 29210

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Chem-Nuclear Systems, Inc. application dated November 24, 1987, as supplemented.

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.: CNS 1-13G

(2) Description

Steel encased lead shielded shipping cask. A double-walled steel cylinder protective jacket encloses the cask during transport. It is bolted to a steel pallet. The cask is closed by a lead-filled flanged plug fitted with a silicone rubber gasket and bolted closure. The cavity is equipped with a drain line and the physical description is as follows:

Cask height, in	67.19
Cask diameter, in	38.5
Cavity height, in	54.0
Cavity diameter, in	26.5
Lead shielding, in	5.0
Protective jacket height, in	81.8
Protective jacket width, in	68.0
Packaging weight, lb	25,500

(3) Drawings

The packaging is constructed in accordance with Chem-Nuclear Systems, Inc. Drawing Nos.: C-110-B-06402-001, Rev. A; C-110-B-06402-002, Rev. 2; C-110-B-06402-003, Rev. 4; and C-110-B-06402-004, Rev. A.

5. (b) Contents

(1) Type, form and maximum quantity of material per package

Plutonium in excess of 20 curies per package must be in the form of metal, metal alloy or reactor fuel elements; and

(i) Byproduct material and special nuclear material as solid metal or oxides. Decay heat not to exceed 600 watts. The radioactive material shall be in the form of fuel rods, or plates, fuel assemblies, or meeting the requirements of special form radioactive material.

500 gm U-235 equivalent mass; or

(ii) Neutron sources meeting the requirements of special form radioactive material.

500 gm U-235 equivalent mass. Decay heat not to exceed 50 watts; or

(iii) Irradiated PuO_2 and UO_2 fuel rods clad in Zircalloy or stainless steel. Decay heat not to exceed 600 watts. All fuel rods shall be contained within a closed 5-inch Schedule 40 pipe with a maximum useable length of 39-5/8 inches.

1,200 gm fissile material with no more than 300 gm fissile material per 5-inch Schedule 40 pipe.

(iv) Process solids, either dewatered, solid, or solidified in a secondary sealed container meeting the requirements for low specific activity radioactive material. Fissile materials must meet the exemption standards in 10 CFR §71.53.

(v) Solid nonfissile irradiated metal hardware, reactor control rods (blades), reactor start-up sources, and segmented boron carbide tubes (tube contents not to exceed a Type A quantity).

(vi) Radioactive (Hot Cell) waste materials immobilized with cement grout and contained in a 55-gallon (or extended 55-gallon drum) DOT Specification 17H or 17C steel drum, lid and closure. The waste material must be packaged in accordance with the Procedural Outline of the Immobilization of Cell Waste Using Cement Grout, Attachment D of the application. The cement grout must be at least 50 volume percent (estimated) of the drum contents and relatively uniformly distributed throughout the drum. At least 3/4" thick layer of grout must cover all radioactive waste contents. Decay heat not to exceed 100 watts, and fissile material not to exceed 500 grams U-235 equivalent mass.

5. (c) Transport Index for Criticality Control

Minimum transport index to be shown on label for nuclear criticality control:

For contents described and limited in 5(b)(1)(i), 5(b)(1)(ii), 5(b)(1)(iii), and 5(b)(1)(vi): 62.5

6. The U-235 equivalent mass is determined by U-235 mass plus 1.66 times U-233 mass plus 1.66 times Pu mass.

7. (a) For any package containing water and/or organic substances which could radiolytically generate combustible gases, determination must be made by tests and measurements or by analysis of a representative package such that the following criteria are met over a period of time that is twice the expected shipment time:

   (i) The hydrogen generated must be limited to a molar quantity that would be no more than 5% by volume (or equivalent limits for other inflammable gases) of the secondary container gas void if present at STP (i.e., no more than 0.063 g-moles/ft^3 at 14.7 psia and 70°F); or

   (ii) The secondary container and cask cavity must be inerted with a diluent to assure that oxygen must be limited to 5% by volume in those portions of the package which could have hydrogen greater than 5%.

   For any package delivered to a carrier for transport, the secondary container must be prepared for shipment in the same manner in which determination for gas generation is made. Shipment period begins when the package is prepared (sealed) and must be completed within twice the expected shipment time.

   (b) For any package containing materials with radioactivity concentration not exceeding that for low specific activity material, and shipped within 10 days of preparation, or within 10 days after venting of drums or other secondary containers, the determination in (a) above need not be made, and the time restriction in (a) above does not apply.

8. For packaging of neutron sources, the cavity drain line must be closed with a plug with a melting temperature of 200°F and the cask cavity must be dry before delivery of the package to a carrier.

9. For packaging of other than neutron sources, the cask must be delivered to a carrier dry and the cavity drain line must be closed with a plug which will maintain its seal at temperatures up to at least 620°F.

10. For the shipment of irradiated metal hardware, the use of the auxiliary shielded inner container and shoring plug shown in Chem-Nuclear Systems, Inc. Drawing Nos. 8651-E-02, Rev. A and 8651-C-01, Rev. B is authorized. The inner container must be provided with vent and drain lines.

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9216	7	71-9216	USA/9216/B()F	4	OF	4

11. Shoring must be provided to minimize movement of contents during accident conditions of transport.

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

    (a) The package shall be prepared for shipment and operated in accordance with Chem-Nuclear Systems, Inc. Operating Procedures, Section 7.0.

    (b) Prior to each shipment the silicone rubber lid gasket(s) must be inspected. This gasket(s) must be replaced if inspection shows any defects or every twelve (12) months, whichever occurs first. Cavity drain line must be sealed with appropriate sealant applied to threads of pipe plug.

    (c) Prior to each shipment the baseplate to cask shell weld must be visually inspected in accordance with Chem-Nuclear Systems, Inc. Operating Procedures, Section 7.0.

    (d) The packaging must meet Chem-Nuclear Systems, Inc. Acceptance Tests and Maintenance Program, Section 8.0.

13. For packaging of neutron sources, 50 times measured neutron dose rate at one meter from the surface of a cask must be less than 1,000 mrem/hr.

14. The contents described in 5(b)(1)(iv) must be transported on a motor vehicle, railroad car, aircraft, inland water crafts, or hold or deck of a seagoing vessel assigned for sole use of the licensee.

15. The package authorized by this certificate is hereby approved for use under the general license provision of 10 CFR §71.12.

16. Expiration date: December 31, 2002.

REFERENCES

Chem-Nuclear Systems, Inc. application dated November 24, 1987.

Supplement dated: November 24, 1992, October 31, 1997, March 31, 1999, and April 23, 2001.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date July 10, 2001

NRC FORM 618 (8-2000) 10 CFR 71	CERTIFICATE OF COMPLIANCE FOR RADIOACTIVE MATERIAL PACKAGES			U.S. NUCLEAR REGULATORY COMMISSION	
1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE	PAGES
9217	12	71-9217	USA/9217/AF	1 OF	4

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)*

Framatome ANP Richland, Inc.
2101 Horn Rapids Road
Richland, WA 99352-0130

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Siemens Power Corporation application dated January 26, 2000, as supplemented.

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.: ANF-250

(2) Description

A uranium oxide powder/pellet shipping container. The packaging consists of a 16-gauge steel inner vessel, approximately 11-1/2 inches ID by 57 inches long, with a bolted and gasketed top flange closure and steel welded bottom plate. The inner vessel is centered and supported in a 22-1/2-inch ID by 68-3/8-inch long, 16-gauge steel drum by twelve 1/4-inch diameter spring steel rods welded to the inner vessel at the top and the bottom of the vessel. A 3/8-inch thick steel flange and a 16-gauge inner band position and support the top of the inner vessel within the outer container. The annulus between the inner vessel and outer container is filled with vermiculite.

The inner vessel is closed by six ½-inch square shank studs with hex head nuts at each end. The outer container is closed with a 12-gauge locking ring with drop forged lugs and a 5/8-inch diameter bolt and lock nut. A product container insert is positioned within the inner vessel.

The maximum gross weight of the packaging and contents is 616 pounds.

(3) Drawings

(i) The ANF-250 shipping container is constructed in accordance with Siemens Power Corporation Drawing No. EMF-306,175, Rev. 16.

(ii) The pellet shipping suit case is constructed in accordance with Siemens Power Corporation Drawing No. EMF-304,306, Rev. 8.

(iii) The powder and pellet product container inserts are constructed in accordance with Siemens Power Corporation Drawing No. EMF-306,176, Rev. 6, Sheets 1 and 2.

5.(b) Contents

(1) Type and form of material

(i) Dry uranium oxide powder enriched to a maximum 5.0 w/o in the U-235 isotope.

(ii) Dry uranium oxide pellets enriched to a maximum 5.0 w/o in the U-235 isotope.

(iii) Uranium oxide pellets enriched to a maximum of 1 w/o in the U-235 isotope.

(iv) Uranium oxide powder enriched to a maximum of 1 w/o in the U-235 isotope.

(2) Maximum quantity of material per package

Not to exceed 310 pounds and:

(i) For the contents described in 5(b)(1)(i):

The contents not to exceed the following:

Maximum Enrichment (wt% U-235)	Maximum Uranium Mass (kg U)	Maximum U-235 Mass (kg U-235)
3.4	62.4	2.12
3.8	41.0	1.56
4.6	31.2	1.44
5.0	27.7	1.38

Not to exceed a maximum mass of 1149 g H, considering all sources of hydrogenous material within the inner vessel. The contents must be contained in product container described in 5(a)(3)(ii).

(ii) For the contents described in 5(b)(1)(ii):

The total contents not to exceed 120 kg U, with the U-235 content not to exceed 6 kg. Not to exceed a maximum mass of 1149 g H, including a maximum mass of 600 g polyethylene, considering all sources of hydrogenous material within the inner vessel. The contents must be contained in product container described in 5(a)(3)(iii).

(iii) For the contents described in 5(b)(1)(iii):

The total contents not to exceed 120 kg U, with the U-235 content not to exceed 1.2 kg. The contents must be contained in product container described in 5(a)(3)(iii).

(iv) For the contents described in 5(b)(1)(iv):

The total contents not to exceed 120 kg U, with the U-235 content not to exceed 1.2 kg. The contents must be contained in product container described in 5(a)(3)(ii).

5.(c) Transport Index for Criticality Control

Minimum transport index to be shown on label for nuclear criticality control:

For contents described in 5(b)(1)(i) and limited in 5(b)(2)(i):	1.8
For contents described in 5(b)(1)(ii) and limited in 5(b)(2)(ii):	0.6
For contents described in 5(b)(1)(iii) and 5(b)(1)(iv), and limited in 5(b)(2)(iii) and 5(b)(2)(iv):	0.4

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.

b. The packaging must meet the Acceptance Tests and Maintenance Program in Chapter 8 of the application.

7. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.12.

8. Expiration date: June 30, 2005.

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9217	12	71-9217	USA/9217/AF	4	OF	4

REFERENCES

Siemens Power Corporation application dated January 26, 2000.

Supplements dated: January 31, June 6, June 15 and September 29, 2000; and February 6, and August 21, 2001.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Charles L. Miller for

E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: August 30, 2001

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9218	14	71-9218	USA/9218/B(U)F-85	1	OF	5

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)*

Department of Energy
Washington, DC 20585

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Westinghouse Electric Corporation application dated August 11, 1999, as supplemented

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.: TRUPACT-II

(2) Description

A stainless steel and polyurethane foam insulated shipping container designed to provide double containment for shipment of contact-handled transuranic waste. The packaging consists of an unvented, 1/4-inch thick stainless steel inner containment vessel (ICV), positioned within an outer containment assembly (OCA) consisting of an unvented 1/4-inch thick stainless steel outer containment vessel (OCV), a 10-inch thick layer of polyurethane foam and a 1/4 to 3/8-inch thick outer stainless steel shell. The package is a right circular cylinder with outside dimensions of approximately 94 inches diameter and 122 inches height. The package weighs not more than 19,250 pounds when loaded with the maximum allowable contents of 7,265 pounds.

The OCA has a domed lid which is secured to the OCA body with a locking ring. The OCV containment seal is provided by a butyl rubber O-ring (bore seal). The OCV is equipped with a seal test port and a vent port.

The ICV is a right circular cylinder with domed ends. The outside dimensions of the ICV are approximately 73 inches diameter and 98 inches height. The ICV lid is secured to the ICV body with a locking ring. The ICV containment seal is provided by a butyl rubber O-ring (bore seal). The ICV is equipped with a seal test port and vent port. Aluminum spacers are placed in the top and bottom domed ends of the ICV during shipping. The cavity available for the contents is a cylinder of approximately 73 inches diameter and 75 inches height.

5.(a)(3) Drawings

The packaging is constructed in accordance with Packaging Technology, Inc., Drawing No. 2077-500 SNP, Sheets 1 through 11, Rev. T. The contents are positioned within the packaging in accordance with TRUPACT-II Authorized Methods for Payload Control (TRAMPAC), Rev. 19a, Appendix 2.1, "Specifications for Authorized Payload Containers and Payload Assembly Configurations." The standard pipe overpack is constructed and assembled in accordance with U.S. Department of Energy, Drawing No. 163-001, Sheets 1 through 3, Rev. 2. The S100 pipe overpack is constructed and assembled in accordance with U.S. Department of Energy, Drawing No. 163-002, Sheets 1 through 2, Rev. 1. The S200 pipe overpack is constructed and assembled in accordance with U.S. Department of Energy, Drawing No. 163-003, Sheets 1 through 2, Rev. 0.

(b) Contents

(1) Type and form of material

Dewatered, solid or solidified transuranic and tritium-contaminated materials and wastes. Materials must be packaged in one of the following payload containers: a 55-gallon drum, a 100-gallon drum, a standard waste box (SWB), a standard pipe overpack, an S100 pipe overpack, an S200 pipe overpack, or ten-drum overpack (TDOP). The payload containers are described in TRAMPAC, Rev. 19a, Appendix 2.1, "Specifications for Authorized Payload Containers and Payload Assembly Configurations." Materials must be restricted to prohibit explosives, corrosives, nonradioactive pyrophorics and pressurized containers. Within a payload container, radioactive pyrophorics must not exceed 1 percent by weight, and free liquids must not exceed 1 percent by volume. Flammable organics and methane are limited along with hydrogen to ensure the absence of flammable gas mixtures in TRU waste payloads as described in Chapter 5.0 of TRAMPAC, Rev. 19a. For payloads of content code LA 154, the absence of flammable gas mixtures is ensured as described in Appendix 1.2 of the TRAMPAC, Rev. 19a.

(2) Maximum quantity of material per package

Contents not to exceed 7,265 pounds including shoring and secondary containers. The maximum gross weight for a payload container not to exceed the following:

(i) 1,000 pounds per 55-gallon drum,
(ii) 328 pounds per 6-inch standard pipe overpack,
(iii) 547 pounds per 12-inch standard pipe overpack,
(iv) 650 pounds per S100 pipe overpack,
(v) 547 pounds per S200 pipe overpack,
(vi) 1,000 pounds per 100-gallon drum,
(vii) 4,000 pounds per SWB, and
(viii) 6,700 pounds per TDOP.

5.(b)(2) Maximum quantity of material per package (continued)

Maximum number of payload containers per package and authorized packaging configurations are as follows:

(i) 14 55-gallon drums,
(ii) 14 standard pipe overpacks,
(iii) 14 S100 pipe overpacks,
(iv) 14 S200 pipe overpacks,
(v) 6 100-gallon drums,
(vi) 2 SWBs,
(vii) 2 SWBs, each SWB containing 1 bin,
(viii) 2 SWBs, each SWB containing up to 4 55-gallon drums,
(ix) 1 TDOP,
(x) 1 TDOP, containing up to 10 55-gallon drums,
(xi) 1 TDOP, containing up to 6 85-gallon drums each overpacking one 55-gallon drum,
(xii) 1 TDOP, containing 1 SWB,
(xiii) 1 TDOP, containing 1 bin within an SWB, or
(xiv) 1 TDOP, containing up to 4 55-gallon drums within an SWB.

Fissile material not to exceed the following:

Payload Container Type	Pu-239 Equivalent Per Payload Container	Pu-239 Equivalent Per Package
55-gallon drum	200 grams	325 grams
Standard pipe overpack	200 grams	2,800 grams
S100 pipe overpack	200 grams	2,800 grams
S200 pipe overpack	200 grams	2,800 grams
100-gallon drum	200 grams	325 grams
SWB	325 grams	325 grams
TDOP	325 grams	325 grams

Pu-239 equivalent must be determined in accordance with TRAMPAC, Rev. 19a, Section 3.1, "Nuclear Criticality."

The S100 pipe overpack and the S200 pipe overpack payloads shall meet the curie limits specified in TRAMPAC, Rev. 19a, Appendices 2.3 and 2.4, respectively.

Maximum decay heat per package not to exceed 40 watts. Decay heat per payload container not to exceed the values given in TRAMPAC, Rev. 19a, Table 5.5-1, "List of Approved Alpha-numeric Shipping Categories, Maximum Allowable Hydrogen Gas Generation Rates, and Maximum Allowable Wattages," or calculated for approved shipping categories in accordance with the methodology specified in Appendix 5.5 of TRAMPAC, Rev. 19a. For content code LA 154 payloads, decay heat per payload container not to exceed the values specified in Appendix 1.2 of TRAMPAC, Rev. 19a.

5. (c) Transport Index for Criticality Control

Minimum transport index to be shown on label for nuclear criticality control: 0.0

6. Physical form, chemical properties, chemical compatibility, configuration of waste containers and contents, isotopic inventory, fissile content, decay heat, weight, center of gravity, and radiation dose rate must be determined and limited in accordance with TRAMPAC, Rev. 19a.

7. Each payload container must be assigned to a shipping category in accordance with TRAMPAC, Rev. 19a, Section 5.1, "Payload Shipping Category." For a payload assembly made up of payload containers with the same or equivalent shipping categories, each payload container and payload assembly must not exceed the allowable wattage in accordance with TRAMPAC, Rev. 19a, Appendix 5.5, "Derivation of Payload Shipping Category Decay Heat Limits" or must be tested for gas generation in accordance with TRAMPAC, Rev. 19a, Appendix 5.7, "Unified Flammable Gas Test Procedure." For a payload made up of payload containers with different (nonequivalent) shipping categories, the flammability index of each payload container must not exceed 50,000 in accordance with TRAMPAC, Rev. 19a, Appendix 6.3, "Mixing of Shipping Categories and Determination of the Flammability Index." Each content code LA 154 payload container must be assigned to a shipping category of "LA 154A," "LA 154B," "LA 154C," or "LA 154D", in accordance with Appendix 1.2 of TRAMPAC, Rev. 19a. Content code LA 154 payload containers may only be assembled with other payload containers belonging to content code LA 154 or dunnage in accordance with Appendix 1.2 of TRAMPAC, Rev. 19a. For a payload of content code LA 154 containers with different shipping categories, the flammability index of each payload container must not exceed 50,000 in accordance with Appendix 1.2 of TRAMPAC, Rev. 19a.

8. Payload containers within a package shall be selected in accordance with TRAMPAC, Rev. 19a, Section 6.0, "Payload Assembly Requirements." Payload containers of content code LA 154 shall be assembled in accordance with Appendix 1.2 of TRAMPAC, Rev. 19a.

9. Each payload container must be equipped with filtered vents meeting the minimum requirements of TRAMPAC, Rev. 19a, Section 2.5, "Specification for Filter Vents." Drums which were not equipped with filtered vents during storage must be aspirated in accordance with TRAMPAC, Rev. 19a, Section 5.3, "Venting and Aspiration."

10. The shipping period for any mode of transport is not to exceed 60 days. For content code LA 154 shipments, the shipping period as defined in Appendix 1.2 or TRAMPAC, Rev. 19a is not to exceed 5 days.

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

(a) Each package must be prepared for shipment and operated in accordance with the procedures described in Chapter 7.0, "Operating Procedures," of the application, as supplemented. For content code LA 154 payloads, each package must be prepared for shipment and operated in accordance with the procedures described in Chapter 7.0 of the application, as modified by Appendix 1.2 of TRAMPAC, Rev. 19a.

(b) Each package must be tested and maintained in accordance with the procedures described in Chapter 8.0, "Acceptance Tests and Maintenance Program," of the application, as supplemented.

(c) Prior to each shipment, the lid and vent port seals on the inner and outer containment vessels must be leak tested in accordance with Appendix 7.4.2 of the application, "Assembly Verification Leak Test."

(d) All free standing water must be removed from the inner containment vessel cavity and the outer containment vessel cavity before shipment.

12. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.12.

13. Expiration date: June 30, 2004.

REFERENCES

Westinghouse Electric Corporation application dated August 11, 1999.

upplements dated: July 23 and October 7, 1999, April 14 and November 30, 2000, May 15, 2001 and arch 15, 2002.

TRUPACT II Authorized Methods for Payload Control (TRAMPAC), Rev. 19a, March 2002.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Charles L. Miller for

E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: July 5, 2002

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE	PAGES
9221	4	71-9221	USA/9221/B( )F	1 OF	3

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)*

U.S. Department of Energy
Division of Naval Reactors
Washington, DC 20585

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Safety Analysis for Radioactive Material Shipping Cask NRBK-41 dated November 2, 1995, as supplemented

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.: NRBK-41

(2) Description

Top loading cylindrical lead shielded 304L stainless steel clad casks for the shipment of irradiated test specimens. The cask has an outside diameter of 27.16 inches and is 40 inches high. The outer shell is 1/2-inch thick stainless steel. The cask cavity is 5 inches in diameter by 16 inches deep and is provided with a bottom drain. The cavity shell is 1/4-inch thick stainless steel and is shielded by 10 inches of lead. The cask is closed by a lead-filled flanged plug fitted with an elastometer O-ring gasket and bolted closure. The cask has a seal-welded, 1/4-inch thick, stainless steel outer thermal shield which provides a 1/16-inch air gap between the outer surface of the cask outer shell and the inside surface of the thermal shield. A one-inch thick stainless steel plate is welded to the bottom of cask. A second one-inch thick stainless steel plate with a 1/8-inch deep, 25.5-inch diameter recess is welded to the first plate to provide a thermal shield for the bottom surface of the cask. The cask is bolted to a 48-inch square, all welded, "I" beam skid. Gross weight of the package is approximately 9,000 pounds.

(3) Drawings

The packaging is constructed in accordance with Battelle Memorial Institute Drawing No. 41-0001, Sheet 1, Rev. D, and Sheet 2, Rev. E, and Westinghouse Electric Corporation Drawing No. 1755E01, Rev. D.

5. (b) Contents

(1) Type and form of material

Byproduct and special nuclear material in solid form, contained within either the MIN-41 or the HIP-41 product containers. The MIN-41 container is constructed in accordance with Westinghouse Electric Corporation, Drawing No. 2D77456 Rev. F. The HIP-41 product container is constructed in accordance with Westinghouse Electric Corporation Drawing No. 5D06622, Rev. B.

(2) Maximum quantity of material per package

The fissile contents of the package must be limited to a maximum of 350 equivalent grams of U-235. The number of equivalent grams of U-235 is determined by the equation: l.0 x grams U-235 + l.4 x grams U-233 + l.6 x grams plutonium. The maximum decay heat load per package must not exceed 240 Btu/hr.

Plutonium in excess of twenty (20) curies per package must be in the form of metal, metal alloy or reactor fuel elements.

5. (c) Transport Index for Criticality Control

Minimum transport index to be shown on label for nuclear criticality control: 0.0

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

(a) The package must be operated in accordance with the Operating Procedures in Section 7.0 of the application, as supplemented.

(b) The package must be maintained in accordance with the Maintenance Procedures in Section 8.2 of the application, as supplemented.

7. The NRBK-41 shipping container may be covered with a wrapping of polyvinyl chloride (PVC) during shipment provided the shipment is made in a closed vehicle. The applicable requirements of 10 CFR §71.87 must be satisfied prior to wrapping the shipping container.

8. Expiration date: September 30, 2006.

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9221	4	71-9221	USA/9221/B( )F	3	OF	3

REFERENCES

Safety Analysis for Radioactive Material Shipping Cask No. NRBK-41 dated November 2, 1995.

Supplements: Naval Reactors letters S#96-11965 dated August 28, 1996, and S#01-10827 dated March 16, 2001.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Susan M Frant for

E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: April 10, 2001

NRC FORM 618 (8-2000) 10 CFR 71	CERTIFICATE OF COMPLIANCE FOR RADIOACTIVE MATERIAL PACKAGES			U.S. NUCLEAR REGULATORY COMMISSION
1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE / PAGES
9225	33	71-9225	USA/9225/B(U)F-85	1 OF 17

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)*

NAC International, Inc.
655 Engineering Drive
Suite 200
Norcross, GA 30092

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Nuclear Assurance Corporation application dated January 14, 2000, as supplemented.

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.: NAC-LWT

(2) Description

The LWT is a steel-encased, lead-shielded shipping cask. The cask is designed to transport one PWR assembly, two BWR assemblies, up to 15 metallic fuel rods, up to 42 MTR and DIDO fuel assemblies and plates, up to 25 individual PWR rods, up to 25 individual high burnup PWR or BWR rods, up to 140 TRIGA fuel elements, or up to 560 TRIGA fuel cluster rods. The overall dimensions of the package, with impact limiters, are 232 inches long by 65 inches in diameter. The cask body is approximately 200 inches in length and 44 inches in diameter. The cask cavity is 178 inches long and 13.4 inches in diameter. The volume of the cavity is approximately 14.5 cubic feet.

The cask body consists of a 0.75-inch-thick stainless steel inner shell, a 5.75-inch-thick lead gamma shield, a 1.2-inch-thick stainless steel outer shell, and a neutron shield tank. The inner and outer shells are welded to a 4-inch-thick stainless steel bottom end forging. The cask bottom consists of a 3-inch-thick, 20.75-inch-diameter lead disk enclosed by a 3.5-inch-thick stainless steel plate and bottom end forging. The cask lid is 11.3-inch-thick stainless steel stepped design, secured to a 14.25-inch-thick ring forging with twelve 1-inch diameter bolts. The cask seal is a metallic O-ring. A second teflon O-ring and a test port are provided to leak test the seal. Other penetrations in the cask cavity include the fill and drain ports, which are sealed with port covers and O-rings.

The neutron shield tank consists of a 0.24-inch-thick stainless steel shell with 0.50-inch-thick end plates. The neutron shield region is 164-inches long and 5-inches thick. The neutron shield tank contains an ethylene glycol/water solution that is 1% boron by weight.

5.(a)(2) Description (continued)

The cask is equipped with aluminum honeycomb impact limiters. The top impact limiter has an outside diameter of 65.25 inches and a maximum thickness of 27.8 inches. The bottom impact limiter has an outside diameter of 60.25 inches and maximum thickness of 28.3 inches. Both impact limiters extend 12 inches along the side of the cask body.

The maximum weight of the package is 52,000 pounds and the maximum weight of the contents and basket is 4,000 pounds.

(3) Drawings

(i) The packaging is constructed in accordance with the following Nuclear Assurance Corporation Drawings:

LWT 315-40-01, Rev. 4	Cask Assembly
LWT 315-40-02, Rev. 14	Body Assembly
LWT 315-40-03, Rev. 16 (Sheets 1-6)*	Transport Cask Body
LWT 315-40-04, Rev. 10	Cask Lid Assembly
LWT 315-40-05, Rev. 9	Upper Impact Limiter
LWT 315-40-06, Rev. 9	Lower Impact Limiter
LWT 315-40-08, Rev. 14 (Sheets 1-4)	Cask Parts Detail

* Packaging Unit Nos. 1, 2, 3, 4, and 5 are constructed in accordance with Drawing No. LWT 315-40-03, Rev. 6 (Sheets 1-6).

(ii) The fuel assembly baskets are constructed in accordance with the following Nuclear Assurance Corporation and NAC International Drawings:

LWT 315-40-09, Rev. 2	PWR Basket Spacer
LWT 315-40-10, Rev. 4	PWR Basket
LWT 315-40-11, Rev. 2	BWR Basket Assembly
LWT 315-40-12, Rev. 3	Metal Fuel Basket Assembly
LWT 315-40-045, Rev. 4	42 MTR Element Base Module
LWT 315-40-046, Rev. 4	42 MTR Element Intermediate Module
LWT 315-40-047, Rev. 4	42 MTR Element Top Module
LWT 315-40-048, Rev. 1	42 MTR Element Cask Assembly
LWT 315-40-049, Rev. 4	28 MTR Element Base Module
LWT 315-40-050, Rev. 4	28 MTR Element Intermediate Module
LWT 315-40-051, Rev. 4	28 MTR Element Top Module
LWT 315-40-052, Rev. 1	28 MTR Element Cask Assembly
LWT 315-40-070, Rev. 3	7 Cell Basket TRIGA Base Module
LWT 315-40-071, Rev. 3	7 Cell Basket TRIGA Intermediate Module
LWT 315-40-072, Rev. 3	7 Cell Basket TRIGA Top Module
LWT 315-40-079, Rev. 1	TRIGA Fuel Cask Assembly
LWT 315-40-080, Rev. 2	7 Cell Poison Basket TRIGA Base Module

5.(a)(3)(ii) Drawings (continued)

LWT 315-40-081, Rev. 2	7 Cell Poison Basket TRIGA Intermediate Module
LWT 315-40-082, Rev. 2	7 Cell Poison Basket TRIGA Top Module
LWT 315-40-083, Rev. 0	Spacer, LWT Cask Assembly TRIGA Fuel
LWT 315-40-084, Rev. 2	LWT Transport Cask Assy 140 TRIGA Elements
LWT 315-40-090, Rev. 2	35 MTR Element Base Module
LWT 315-40-091, Rev. 2	35 MTR Element Intermediate Module
LWT 315-40-092, Rev. 2	35 MTR Element Top Module
LWT 315-40-094, Rev. 2	35 MTR Element Cask Assembly
LWT 315-40-096, Rev. 2	Fuel Rod Insert, TRIGA Fuel
LWT 315-40-098, Rev. 1	Can Assembly, LWT Pin Shipment
LWT 315-40-099, Rev. 3 (Sheets 1-3)	Can Weldment, PWR/BWR Transport Canister
LWT 315-40-100, Rev. 1 (Sheets 1-2)	Lids, PWR/BWR Transport Canister
LWT 315-40-101, Rev. 0	4 x 4 Insert, PWR/BWR Transport Canister
LWT 315-40-102, Rev. 1	5 x 5 Insert, PWR/BWR Transport Canister
LWT 315-40-103, Rev. 0	Pin Spacer, PWR Transport Canister
LWT 315-40-104, Rev. 0	LWT Cask Assembly, PWR Transport Canister
LWT 315-40-105, Rev. 3 (Sheets 1-2)	PWR Insert, PWR/BWR Transport Canister
LWT 315-40-106, Rev. 1 (Sheets 1-3)	MTR Plate Canister, LWT Cask
LWT 315-40-108, Rev. 1 (Sheets 1-3)	7 Cell Basket, Top Module, DIDO Fuel
LWT 315-40-109, Rev. 1 (Sheets 1-3)	7 Cell Basket, Intermediate Module, DIDO Fuel
LWT 315-40-110, Rev.1 (Sheets 1-3)	7 Cell Basket, Bottom Module, DIDO Fuel
LWT 315-40-111, Rev. 0	LWT Transport Cask Assy DIDO Fuel
LWT 315-40-113, Rev. 0	Spacer, Top Module DIDO Fuel

5.(b) Contents

(1) Type and form of material

(i) Irradiated PWR fuel assemblies. The maximum fuel assembly weight is 1650 pounds, the maximum average burnup is 35,000 MWD/MTU, the minimum cool time is 2 years, and the maximum initial fuel pin pressure at 70°F is 565 psig. The fuel assemblies consist of uranium dioxide pellets within zircaloy or ZIRLO cladding, with the specifications listed below, and with fuel rod pitch, rod diameter, clad thickness, and pellet diameter as described in Table 1.2-5, of the application, as supplemented.

Fuel Type	No. Fuel Rods	Max. Initial Uranium Enrichment (w/o U-235)	Max. Initial Uranium Mass (MTU)	Max. Active Fuel Length (in.)
B&W 15x15	208	3.5	0.4750	144.0
B&W 17x17	264	3.5	0.4658	143.0
CE 14x14	176	3.7	0.4037	137.0
CE 16x16	236	3.7	0.4417	150.0
WE 14x14 Std	179	3.7	0.4144	145.2
WE 14x14 OFA	179	3.7	0.3612	144.0
WE 15x15	204	3.5	0.4646	144.0
WE 17x17 Std	264	3.5	0.4671	144.0
WE 17x17 OFA	264	3.5	0.4282	144.0
Ex/ANF 14x14 WE	179	3.7	0.3741	144.0
Ex/ANF 14x14 CE	176	3.7	0.3814	134.0
Ex/ANF 15x15 WE	204	3.7	0.4410	144.0
Ex/ANF 17x17 WE	264	3.5	0.4123	144.0

5.(b)(1) Type and form of material (continued)

(ii) Irradiated BWR fuel assemblies. The maximum fuel assembly weight is 750 pounds, the maximum average burnup is 30,000 MWD/MTU, the minimum cool time is 2 years, and the maximum initial fuel pin pressure at 70°F is 565 psig. The fuel assemblies consist of uranium dioxide pellets within zircaloy or ZIRLO cladding, with the specifications listed below, and with fuel rod pitch, rod diameter, clad thickness, and pellet diameter as described in Table 1.2-6, of the application, as supplemented.

Fuel Type	No. Fuel Rods	No. Water Rods	Max. Initial Uranium Enrichment (w/o U-235)	Max. Initial Uranium Mass (MTU)	Max. Active Fuel Length (in.)
GE 7x7	49	0	4.0	0.1923	146
GE 8x8-1	63	1	4.0	0.1880	146
GE 8x8-2	62	2	4.0	0.1847	150 (1)
GE 8x8-4	60	4	4.0	0.1787	150 (1,2)
GE 9x9	74	2	4.0	0.1854	150 (1,3,4)
	79	2	4.0	0.1979	150 (1,4)
Ex/ANF 7x7	49	0	4.0	0.1960	144
Ex/ANF 8x8-1	63	1	4.0	0.1764	145.2
Ex/ANF 8x8-2	62	2	4.0	0.1793	150
Ex/ANF 9x9	79	2	4.0	0.1779	150
	74	2	4.0	0.1666	150 (3)

(1) Six-inch natural uranium blankets on top and bottom.
(2) One large water hole - 3.2 cm ID, 0.1 cm thickness.
(3) Two large water holes occupying seven fuel rod locations - 2.5 cm ID, 0.07 cm thickness.
(4) Shortened active fuel length in some rods.

(iii) Irradiated PWR rods, consisting of uranium dioxide pellets within zircaloy or ZIRLO cladding. The maximum uranium enrichment is 5 weight percent U-235, the maximum active fuel length is 150 inches, and the maximum pellet diameter is 0.3765 inches. The maximum burnup is 60,000 MWD/MTU and the minimum cool time is 150 days. Up to two rods may have a maximum burnup of 65,000 MWD/MTU.

5.(b)(1) Type and form of material (continued)

(iv) Irradiated MTR fuel elements composed of U-Al, U_3O_8-Al, or U_3Si_2-Al positioned within the MTR fuel basket specified in 5.(a)(3)(ii). Loose fuel plates must meet the requirements of the MTR fuel element content tables and must be loaded into an MTR plate canister prior to shipment. The fuel elements are composed of aluminum clad plates, with initial uranium enrichment up to 94.0 weight percent U-235. The maximum burnup and the minimum cool time shall be consistent with the decay heat limits in Item 5.(b)(2)(iv) and shall be determined using the operating procedures in Section 7.1.5 of the application.

NISTR MTR fuel elements specifications are listed in Item 5.(b)(1)(iv)(a), generic MTR fuel elements are listed in Item 5.(b)(1)(iv)(b), and expanded fuel specifications applicable to LEU MTR fuel (up to 25.0 wt %^{235}U) are listed in Item 5.(b)(1)(iv)(c).

(a) NISTR MTR Fuel Content Description

Parameter	Plate	Plate (cut in half)
Enrichment, wt % ^{235}U	≤94	
Number of fuel plates	≤17	≤34
^{235}U content per plate	≤22	≤11
Plate thickness (cm)	≥0.115	
Clad Thickness (cm)	≥0.02	
Active fuel width (cm)	≤6.6	
Active fuel height (cm)	≥54 cm	27 to 30
Maximum ^{235}U content per element (g)	≤380	

(b) Generic MTR Fuel Content Description

Parameter	Limiting Values[2]					
Enrichment, wt. % ^{235}U	≤94					
Number of fuel plates	≤23	≤19	≤23[1]	≤17	≤19	≤23
^{235}U content per plate	≤18	≤20	≤20[1]	≤21	≤21	≤16.5
Plate thickness (cm)	≥0.115	≥0.115	≥0.123[1]	≥0.115	≥.200	≥0.115
Clad Thickness (cm)	≥0.02					
Active fuel width (cm)	≤6.6	≤6.6	≤6.6	≤6.6	≤6.6	≤7.3
Active fuel height (cm)	≥56					
^{235}U content per element (g)	≤380[2]					

Notes:

1. HEU (>90 wt% ^{235}U enriched) MTR fuel having 23 plates with up to 20 g of ^{235}U per plate, with a minimum plate thickness of 0.123 cm, must have at least 2.0 cm of non-fuel material at the ends of each element. This fuel may also be loaded up to 460 g ^{235}U per plate.

2. At enrichments ≤25 wt% ^{235}U, MTR fuel elements with extended fuel characteristics may be loaded with the specifications defined in 5.(b)(1)(iv)(c)

(c) Expanded LEU MTR Fuel Content Description

Parameter	Base	≤7.0 cm Active Fuel Width			≤7.1 cm Active Fuel Width		≤7.15 cm Active Fuel Width		
Enrichment, wt. % ^{235}U	≤25	≤25			≤25		≤25		
Number of fuel plates	≤23	≤23			≤17	≤23	≤22	≤23	≤23
^{235}U content per plate	≤22	≤22	≤22	≤21.5	≤22		≤22	≤21.5	≤22
Plate thickness (cm)	≥0.115	≥0.119	≥0.115	≥0.115	≥0.115	≥0.200	≥0.119		
Clad Thickness (cm)	≥0.02								
Active fuel width (cm)	≤6.6	≤7.0			≤7.1		≤7.15		
Active fuel height (cm)	≥56	≥56	≥63	≥56	≥56		≥56	≥56	≥61
^{235}U content per element (g)	≤420	≤470			≤470		≤470		

5.(b)(1) Type and form of material (continued)

(v) Metallic fuel rods containing natural enrichment uranium pellets with aluminum cladding 0.080-inches thick. The fuel pellet diameter is 1.36 inches and the maximum fuel rod length is 120.5 inches. The maximum weight of uranium per rod is 54.5 kg with a maximum average burnup of 1,600 MWD/MTU and a minimum cooling time of one year.

(vi) Irradiated TRIGA fuel elements with a 0.225" diameter zirconium rod in the center and meeting the following specifications:

	TRIGA HEU (Notes 1& 2)	TRIGA LEU (Notes 1& 2)	TRIGA LEU (Notes 1& 2)
Fuel Form	Clad U-ZrH rod	Clad U-ZrH rod	Clad U-ZrH rod
Maximum Element Weight, lbs	13.2	13.2	6.4
Maximum Element Length, in	45	45	28.4
Element Cladding	Stainless Steel	Stainless Steel	Aluminum
Clad Thickness, in	0.02	0.02	0.03
Active Fuel Length, in	15	15	14-15 (Note 4)
Element Diameter, in	1.478 max.	1.478 max.	1.47 max.
Fuel Diameter, in	1.435 max.	1.435 max.	1.41 max.
Maximum Initial U Content/Element, kilograms	0.196	0.845	0.205
Maximum Initial ^{235}U Mass, grams	137	169	41
Maximum Initial ^{235}U Enrichment, weight percent	70	20	20
Zirconium Mass, grams	2060	1886 - 2300	2300
Hydrogen to Zirconium Ratio, max.	1.6	1.7	1.0
Maximum Average Burnup, MWD/MTU	460,000 (80% ^{235}U)	151,100 (80% ^{235}U)	151,100 (80% ^{235}U)
Minimum Cooling Time	90 days (Note 3)	90 days (Note 3)	90 days (Note 3)

Notes:
1. Mixed TRIGA LEU and HEU contents authorized.
2. TRIGA Standard, instrumented and fuel follower control rod type elements authorized.
3. Maximum decay heat of any element is 7.5 watts.

4. Aluminum clad fuel with 14 inch active fuel is solid and has no central hole with a zirconium rod.

5.(b)(1) Type and form of material (continued)

(vii) Irradiated TRIGA fuel cluster rods with a maximum average burnup of 600,000 MWD/MTU (80% ^{235}U) and a minimum cooling time of 160 days meeting the following specifications prior to irradiation:

	TRIGA Fuel Cluster Rods
Fuel Form	Clad U-ZrH rod
Maximum Rod Weight, lbs	1.5
Maximum Rod Length, in	31
Rod Cladding	Incoloy 800
Minimum Clad Thickness, in	0.015
Maximum Active Fuel Length, in	22.5
Maximum Fuel Pellet Diameter, in	0.53
Maximum U Content/Rod, grams	48.6
Maximum ^{235}U Mass, grams	45.4
Maximum ^{235}U Enrichment, weight percent	93.3
Maximum Zirconium Mass, grams	421
Hydrogen to Zirconium Ratio, max.	1.6

(viii) Irradiated high burnup PWR rods, consisting of uranium dioxide pellets within zircaloy or ZIRLO cladding. The maximum uranium enrichment is 5 weight percent U-235, the maximum active fuel length is 150 inches, and the maximum pellet diameter is 0.3765 inches. The maximum burnup is 80,000 MWD/MTU and the minimum cool time is 150 days.

5.(b)(1) Type and form of Material (continued)

(ix) Irradiated high burnup BWR rods, consisting of uranium dioxide pellets within zircaloy or ZIRLO cladding. The maximum uranium enrichment is 5 weight percent U-235, the maximum active fuel length is 150 inches, and the maximum pellet diameter is 0.490 inches. The maximum burnup is 80,000 MWD/MTU and the minimum cool time is between 150 - 270 days, as specified in the table below:

BWR Fuel Type Array Size	Burnup, b (GWD/MTU)	Minimum Cool Time (days)
7 x 7	$b \leq 60$ $60 < b \leq 70$ $70 < b \leq 80$	210 240 270
8 x 8[1]	$b \leq 80$	150

Note 1: Includes rods from all larger BWR assembly arrays (e.g., 9 x 9, 10 x 10)

(x) Irradiated DIDO fuel elements composed of U-Al, U_3O_8-Al, or U_3Si_2-Al positioned within the DIDO fuel basket specified in 5.(a)(3)(ii). The fuel elements are composed of four concentric tubes of varying diameters. The fuel elements have an initial enrichment up to 94.0 weight percent U-235. The fuel elements shall have the specifications listed below:

Parameter	LEU[1]	MEU[1]	HEU[1]
Maximum ^{235}U content per Element	≤ 190 g	≤ 190 g	≤ 190 g
Maximum Uranium content per Element	≤ 1000 g	≤ 475.0 g	≤ 211.1g
Minimum Fuel Tube Thickness	0.130 cm	0.130 cm	0.130 cm
Minimum Clad Thickness	0.0325 cm	0.0325 cm	0.0325 cm
Maximum Outer Diameter	9.535 cm	9.535 cm	9.535 cm
Minimum Nominal Inner Diameter	6.08 cm	6.08 cm	6.08 cm
Minimum Initial Enrichment	19 wt% ^{235}U	40 wt% ^{235}U	90 wt% ^{235}U

1 The maximum burnup and minimum cool time shall be consistent with the decay heat limits in Item 5.(b)(2)(ix) and shall be determined using the operating procedures in Section 7.1.4 of the application.

5.(b)(2) Maximum quantity of material per package

Not to exceed 4,000 pounds, including contents and fuel assembly basket.

(i) For the contents described in Item 5.(b)(1)(i): one PWR assembly positioned within the PWR fuel assembly basket. Maximum decay heat not to exceed 2.5 kilowatts per PWR assembly.

(ii) For the contents described in Item 5.(b)(1)(ii): two BWR assemblies positioned with the BWR fuel assembly basket. Maximum decay heat not to exceed 1.1 kilowatts per BWR assembly.

(iii) For PWR rods as described in Item 5.(b)(1)(iii): up to 25 intact individual rods in a Type 304 stainless steel spacer canister with a wall thickness of at least 0.12 inches positioned within the PWR or BWR basket. Maximum decay heat not to exceed 1.41 kilowatts per package.

(iv) For MTR fuel elements as described in Items 5.(b)(1)(iv):

Up to 42 fuel elements positioned within the MTR fuel assembly basket (7 fuel elements per basket module). Each of the MTR basket cell openings may contain a loose plate canister. The contents of each loose plate canister are limited to the number of fuel plates, dimensions, and masses that are equivalent to an intact MTR fuel element, as specified in Items 5.(b)(1)(iv).

(a) The maximum decay heat is not to exceed 1.26 kilowatts per package, with each MTR fuel assembly basket module not to exceed 210 watts.

(b) HEU, MEU, and LEU MTR fuel elements with decay heat not exceeding 30 watts per element may be loaded in any basket position.

(c) Mixed HEU, MEU, and LEU MTR contents, with decay heat limits as specified above, are authorized.

(d) MTR fuel elements with corrosion and/or mechanically damaged cladding are authorized, provided the total surface area of through-clad corrosion and/or mechanical damage does not exceed 2,775 cm^2 per package.

(e) For HEU-MTR fuel elements only, the center fuel element in any basket module is not to exceed 120 watts. The two exterior fuel elements vertically in-line with the center assembly for transport are not to exceed 70 watts.

(v) For the contents described in Item 5.(b)(1)(v): up to 15 intact metallic fuel rods positioned within the appropriate basket. Maximum decay heat not to exceed 0.036 kilowatts per rod. Total weight of all rods not to exceed 1,805 pounds.

5.(b)(2) Maximum quantity of material per package (continued)

(vi) For failed metallic fuel rods of the type described in Item 5.(b)(1)(v):

(a) Up to six canisters containing one defective metallic fuel rod per canister. The canisters are 2.75-inch I.D. failed fuel rod canisters as shown on Nuclear Assurance Corporation Drawing No. 340-108-D2, Rev. 10, and are placed in a six-hole liner as shown on Nuclear Assurance Corporation Drawing No. 315-040-43, Rev. 1. The maximum decay heat load for a defective metallic fuel rod is limited to 5 watts; or

(b) Up to three canisters containing either up to three defective metallic fuel rods per canister or up to 10 failed fuel filters per canister. The canisters are 4.00-inch I.D. failed fuel rod canisters as shown on Nuclear Assurance Corporation Drawing No. 340-108-D1, Rev. 10, and are placed in a three-hole basket as shown on Nuclear Assurance Corporation Drawing No. 315-40-12, Rev. 3. The weight of the filters is limited to 125 pounds per canister. For canisters containing fuel rods, the maximum decay heat load is 15 watts per canister; and for canisters containing filters, the maximum decay heat load is 5 watts per canister. The plutonium content of the filters shall not exceed 20 curies per package.

(vii) For TRIGA fuel elements as described in Item 5.(b)(1)(vi):

Maximum decay heat not to exceed 7.5 watts per TRIGA fuel element (or equivalent for failed fuel) and 1050 watts per package. TRIGA fuel elements must be positioned in either the non-poisoned TRIGA fuel basket or in the poisoned TRIGA fuel basket. Fuel may not be loaded in the center cell of the non-poisoned TRIGA fuel basket.

(a) Up to 120 fuel elements in the non-poisoned TRIGA fuel basket, and up to 140 fuel elements in the poisoned TRIGA fuel basket (4 fuel elements per basket cell).

(b) Up to 12 screened canisters in the non-poisoned TRIGA fuel basket, and up to 14 screened canisters in the poisoned TRIGA fuel basket. The screened canisters are in accordance with NAC International Drawing Nos. 315-40-074, Rev. 1, 315-40-075, Rev. 1, and 315-40-076, Rev. 1. Up to four intact TRIGA fuel elements per screened canister.

5.(b)(2) Maximum quantity of material per package (continued)

(c) Up to 12 sealed canisters in the non-poisoned TRIGA fuel basket, and up to 14 sealed canisters in the poisoned TRIGA fuel basket. The sealed canisters are in accordance with NAC International Drawing Nos. 315-40-086, Rev. 0, 315-40-087, Rev. 3, and 315-40-088, Rev. 2. Up to a maximum equivalent of two fuel elements in the form of intact fuel, failed fuel or fuel debris per sealed canister. If the total failed fuel plutonium content of a package is greater than 20 Ci, all failed fuel containing plutonium must be enclosed in a sealed canister which is then leak tested to 3.2×10^{-7} std cm^3/sec (He) prior to shipment.

(d) Mixed intact and failed fuel contents are authorized. Base and top fuel basket modules may contain intact fuel elements, screened canisters, or sealed canisters. Intermediate fuel basket modules may contain only intact TRIGA fuel elements.

(viii) For TRIGA fuel cluster rods as described in Item 5.(b)(1)(vii):

Maximum decay heat not to exceed 1.875 watts per TRIGA fuel cluster rod (or equivalent for failed fuel) and 1050 watts per package. TRIGA fuel cluster rods must be positioned in either the non-poisoned TRIGA fuel basket or in the poisoned TRIGA fuel basket. Fuel may not be loaded in the center cell of the non-poisoned TRIGA fuel basket.

(a) Up to 480 rods in the non-poisoned TRIGA fuel basket, and up to 560 rods in the poisoned TRIGA fuel basket. TRIGA fuel cluster rods must be positioned within the fuel rod inserts as shown on NAC International Drawing No. 315-40-096, Rev. 2.

(b) Up to 12 sealed canisters in the non-poisoned TRIGA fuel basket, and up to 14 sealed canisters in the poisoned TRIGA fuel basket. The sealed canisters are in accordance with NAC International Drawing Nos. 315-40-086, Rev. 0, 315-40-087, Rev. 3, and 315-40-088, Rev. 2. Up to a maximum equivalent of six TRIGA fuel cluster rods in the form of intact fuel, failed fuel or fuel debris per sealed canister. If the total failed fuel plutonium content of a package is greater than 20 Ci, all failed fuel containing plutonium must be enclosed in a sealed canister which is then leak tested to 3.2×10^{-7} std cm^3/sec (He) prior to shipment.

(c) Mixed intact and failed fuel contents are authorized. Base and top fuel basket modules may contain intact fuel rods or sealed canisters. Intermediate fuel basket modules may contain only intact fuel rods.

(ix) For high burnup PWR rods as described in Item 5.(b)(1)(viii): up to 25 intact individual rods in the appropriate insert, placed within a sealed or free-flow canister, and positioned within the standard PWR basket. Maximum decay heat not to exceed 2.3 kilowatts per package.

5.(b)(2) Maximum quantity of material per package (continued)

(x) For high burnup BWR rods as described in Item 5.(b)(1)(ix): up to 25 intact individual rods in the appropriate insert, placed within a sealed or free-flow canister, and positioned within the standard PWR basket. Maximum decay heat not to exceed 2.1 kilowatts per package.

(xi) For DIDO fuel as described in Item 5.(b)(1)(x)

Up to 42 DIDO fuel elements with a maximum decay heat not to exceed 25 watts per DIDO fuel element provided retention spacer is present for top basket. Maximum decay heat is 1.05 kilowatts per package. If retention spacer is not present, then maximum decay heat not to exceed 18 watts per DIDO fuel element and a total of 756 watts per package.

5.(c) Transport Index for Criticality Control

Minimum transport index to be shown on label for nuclear criticality control:

(1)	For TRIGA fuel elements, TRIGA fuel cluster rods, metallic fuel rods, MTR fuel assemblies, up to 25 PWR fuel rods, and up to 25 high burnup PWR or BWR rods:	0.0
(2)	For PWR fuel assemblies:	100
(3)	For BWR fuel assemblies:	5.0
(4)	For DIDO fuel assemblies:	12.5

6. Known or suspected failed fuel assemblies (rods) or elements, and fuel with cladding defects greater than pin holes and hairline cracks are not authorized, except as described in Items 5.(b)(2)(iv)(d), 5.(b)(2)(vi), 5.(b)(2)(vii)(c), and 5.(b)(2)(viii)(b).

7. The cask must be dry (no free water) when delivered to a carrier for transport.

8. Bolt torque: The cask lids bolts must be torqued to 260 ft-lbs. The bolts used to secure the vent and drain port covers must be torqued to 100 inch-lbs.

9. Prior to each shipment, the package must be leak tested to 1×10^{-3} std cm^3/sec, except that replaced seals must be leak tested to 5.5×10^{-7} std cm^3/sec (He). Prior to first use, after third use, and at least once within the 12-month period prior to each subsequent use, the package must be leak tested to 5.5×10^{-7} std cm^3/sec (He).

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

    (a) The metallic O-ring seal must be replaced prior to each shipment; and

    (b) Each package must meet the Acceptance Tests and Maintenance Program of Chapter 8 of the application, as supplemented; and

    (c) The package shall be prepared for shipment and operated in accordance with the Operating Procedures of Chapter 7 of the application, as supplemented. If the cask is loaded under water or water is introduced into the cask cavity, the cask must be vacuum dried as described in Chapter 7 of the application. The cask cavity must be backfilled with 1.0 atm of helium when shipping PWR or BWR assemblies.

11. When shipping PWR, BWR, MTR, DIDO assemblies, TRIGA fuel elements, TRIGA fuel cluster rods, individual PWR rods, or high burnup PWR or BWR rods, the neutron shield tank must be filled with a mixture of water and ethylene glycol which will not freeze or precipitate in a temperature range from -40 °F to 250 °F. The water and ethylene glycol mixture must contain at least 1% boron by weight.

12. A personnel barrier must be used when shipping PWR or BWR assemblies. Shipments of MTR, DIDO fuel assemblies, TRIGA fuel elements, TRIGA fuel cluster rods, individual PWR rods, or high burnup PWR or BWR rods must use the ISO container or a personnel barrier.

13. Packages used to ship metallic fuel rods may be shipped in a closed shipping container provided that the closed container, the cask tie-down and support system and transport vehicle (trailer) meet the applicable requirements of the Department of Transportation. When the cask is shipped in a closed shipping container, the center of gravity of the combined cask, closed shipping container and trailer must not exceed 75 inches.

14. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.12.

15. Expiration Date: February 28, 2005.

REFERENCES

NAC International, Inc., application dated January 14, 2000.

Supplements dated: February 11 and 18; April 10 and 21; May 1, 22 and 26; June 5, 12 and 20; August 23 and 31, October 2, 6, and 16, November 14, and December 19 and 27, 2000. March 1, and 15; April 27, July 3 and 20; August 22, 2001; and September 12 and 13, 2001. February 28, April 12 and September 9, 2002.

FOR THE U. S. NUCLEAR REGULATORY COMMISSION

John D. Monninger, Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: November 14, 2002

NRC FORM 618
(3-96)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIALS PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. PACKAGE IDENTIFICATION NUMBER	d. PAGE NUMBER	e. TOTAL NUMBER PAGES
9226	0	USA/9226/B(U)F-85	1	7

2. PREAMBLE

a. This certificate is issued to certify that the 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)*

General Atomics
3550 General Atomics Court
San Diego, California 92121-1194

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION:

General Atomics application dated
August 31, 1994, as supplemented

c. DOCKET NUMBER

71-9226

4. CONDITIONS

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

5. **5.a Packaging**

(1) Model No.: GA-4

(2) Description

The GA-4 Legal Weight Truck Spent Fuel Shipping Cask consists of the packaging (cask and impact limiters) and the radioactive contents. The packaging is designed to transport up to four intact pressurized-water reactor (PWR) irradiated spent fuel assemblies as authorized contents. The packaging includes the cask assembly and two impact limiters, each of which is attached to the cask with eight bolts. The overall dimensions of the packaging are approximately 90 inches in diameter and 234 inches long.

The containment system includes the cask body (cask body wall, flange, and bottom plate); cask closure; closure bolts; gas sample valve body; drain valve; and primary O-ring seals for the closure, gas sample valve, and drain valve.

Cask Assembly

The cask assembly includes the cask, the closure, and the closure bolts. Fuel spacers are also provided when shipping specified short fuel assemblies to limit the movement of the fuel. The cask is constructed of stainless steel, depleted uranium, and a hydrogenous neutron shield. The cask external dimensions are approximately 188 inches long and 40 inches in diameter. A fixed fuel support structure divides the cask cavity into four spent fuel compartments, each approximately 8.8 inches square and 167 inches long. The closure is recessed into the cask body and is attached to the cask flange with 12 1-inch diameter bolts. The closure is approximately 26 inches square, 11 inches thick, and weighs about 1510 lbs.

The cask has two ports allowing access to the cask cavity. The closure lid has an integral half-inch diameter port (hereafter referred to as the gas sample valve) for gas sampling, venting, pressurizing, vacuum drying, leakage testing, or inerting. A 1-inch diameter port in the bottom plate allows draining, leakage testing, or filling the cavity with water. A separate drain valve opens and closes the port. The primary seals for the gas sample valve and drain valve are recessed from the outside cask surface as protection from punctures. The gas sample valve and the drain valve also have covers to protect them during transport.

Cask

The cask includes the containment (flange, cask body, bottom plate and drain valve seals); the cavity liner and fuel support structure; the impact limiter support structure; the trunnions and redundant lift sockets; the depleted uranium gamma shield; and the neutron shield and its outer shell. The cask body is square, with rounded corners and a transition to a round outer shell for the neutron shield. The cask has approximately a 1.5 inch thick stainless steel body wall, 2.6 inch thick depleted uranium shield (reduced at the corners), and 0.4 inch thick stainless steel fuel cavity liner.

The cruciform fuel support structure consists of stainless steel panels with boron-carbide (B_4C) pellets for criticality control. A continuous series of holes in each panel, at right angles with the fuel support structure axis, provides cavities for the B_4C pellets. The fuel support structure is welded to the cavity liner and is approximately 18 inches square by 166 inches long and weighs about 750 lbs.

The flange connects the cask body wall and fuel cavity liner at the top of the cask, and the bottom plate connects them at the bottom. The gamma shield is made up of five rings, which are assembled with zero axial tolerance clearance within the depleted uranium cavity, to minimize gaps. The impact limiter support structure is a slightly tapered 0.4 inch thick shell on each end of the cask. The shell mates with the impact limiter's cavity and is connected to the cask body by 36 ribs.

The neutron shield is located between the cask body and the outer shell. The neutron shield design maintains continuous shielding immediately adjacent to the cask body under normal conditions of transport. The details of the design are proprietary. The design, in conjunction with the operating procedures, ensures the availability of the neutron shield to perform its function under normal conditions of transport.

Two lifting and tie-down trunnions are located about 34 inches from the top of the cask body, and another pair is located about the same distance from the bottom. The trunnion outside diameter is 10 inches, increasing to 11.5 inches at the cask interface. Two redundant lift sockets are located about 26 inches from the top of the cask body and are flush with the outer skin.

Materials

All major cask components are stainless steel, except the neutron shield, the depleted uranium gamma shield, and the B_4C pellets contained in the fuel support structure. All O-ring seals are fabricated of ethylene propylene.

Impact Limiters

The impact limiters are fabricated of aluminum honeycomb, completely enclosed by an all-welded austenitic stainless steel skin. Each of the two identical impact limiters is attached to the cask with eight bolts. Each impact limiter weighs approximately 2,000 lbs.

(3) Drawings

The package shall be constructed and assembled in accordance with the following GA Drawing Number:

Drawing No. 031348,
sheets 1 through 19, Revision D
GA-4 Spent Fuel Shipping Cask Packaging Assembly

5.b Contents of Packaging

(1) Type and Form of Material:

(a) Intact fuel assemblies. Fuel with known or suspected cladding defects greater than hairline cracks or pinhole leaks is not authorized for shipment.

(b) The fuel authorized for shipment in the GA-4 package is irradiated 14x14 and 15x15 PWR fuel assemblies with uranium oxide fuel pellets. Before irradiation, the maximum enrichment of any assembly to be transported is 3.15 percent by weight of uranium-235 (^{235}U). The total initial uranium content is not to exceed 407 Kg per assembly for 14x14 arrays and 469 Kg per assembly for 15x15 arrays.

(c) Fuel assemblies are authorized to be transported with or without control rods or other non-fuel assembly hardware (NFAH). Spacers shall be used for the specific fuel types, as shown on sheet 17 of the Drawings.

(d) The maximum burnup for each fuel assembly is 35,000 MWd/MTU with a minimum cooling time of 10 years and a minimum enrichment of 3.0 percent by weight of ^{235}U or 45,000 MWd/MTU with a minimum cooling time of 15 years (no minimum enrichment).

(e) The maximum assembly decay heat of an individual assembly is 0.617 kW. The maximum total allowable cask heat load is 2.468 kW (including control components and other NFAH when present).

(f) The PWR fuel assembly types authorized for transport are listed in Table 1. All parameters are design nominal values.

(2) Maximum Quantity of Material per Package

(a) For material described in 5.b(1): four (4) PWR fuel assemblies.

(b) For material described in 5.b(1): the maximum assembly weight (including control components or other NFAH when present) is 1,662 lbs. The maximum weight of the cask contents (including control components or other NFAH when present) is 6,648 lbs., and the maximum gross weight of the package is 55,000 lbs.

Table 1 - PWR Fuel Assembly Characteristics

Fuel Type Mfr.-Array (Versions)	Design Initial U (kg/assy.)	No. of Fuel Rods	Fuel Rod Pitch (in.)	Pellet Diameter (in.)	Zr Clad Thickness (in.)	Active Fuel Length (in.)
W-15x15 (Std/ZC)	469	204	0.563	0.3659	0.0242	144
W-15x15 (OFA)	463	204	0.563	0.3659	0.0242	144
BW-15x15 (Mk.B,BZ,BGD)	464	208	0.568	0.3686	0.0265	142
Exx/A-15x15 (WE)	432	204	0.563	0.3565	0.030	144
CE-15x15 (Palisades)	413	204	0.550	0.358	0.026	144
CE-14x14 (Ft.Calhoun)	376	176	0.580	0.3765	0.028	128
W-14x14 (Model C)	397	176	0.580	0.3805	0.026	137
CE-14x14 (Std/Gen.)	386	176	0.580	0.3765	0.028	137
Exx/A-14x14 (CE)	381	176	0.580	0.370	0.031	137
W-14x14 (OFA)	358	179	0.556	0.3444	0.0243	144
W-14x14 (Std/ZCA,/ZCB)	407	179	0.556	0.3674	0.0225	145.5
Exx/A-14x14 (WE)	379	179	0.556	0.3505	0.030	142

5.c Transport Index for Criticality Control

Minimum transport index to be shown on the label for nuclear criticality control: 100

6. **Fuel assemblies with missing fuel pins shall not be shipped unless dummy fuel pins that displace an equal amount of water have been installed in the fuel assembly.**

7. **For operating controls and procedures, in addition to the requirements of Subpart G of 10 CFR Part 71:**

a. Each package shall be both prepared for shipment and operated in accordance with detailed written operating procedures. Procedures for both preparation and operation shall be developed using the specifications contained within the application. At a minimum, those procedures shall require the following provisions:

(1) Identification of the fuel to be loaded and independent verification that the fuel meets the specifications of Condition 5.b of the CoC.

(2) That before shipment the licensee shall:
(a) Perform a measured radiation survey to assure compliance with 49 CFR 173.441 and 10 CFR 71.47 and assure that the neutron and gamma measurement instruments are calibrated for the energy spectrums being emitted from the package.
(b) Verify that measured dose rates meet the following correlation to demonstrate compliance with the design bases calculated hypothetical accident dose rates:
3.4 x (peak neutron dose rate at any point on cask surface at its midlength) +
1.0 x (gamma dose rate at that location) $\leq$ 1000 mR/hr.
(c) Verify that the surface removable contamination levels meet the requirements of 49 CFR 173.443 and 10 CFR 71.87.
(d) Inspect all containment seals and closure sealing surfaces for damage. Leak test all containment seals with a gas pressure rise test after final closure of the package. The leak test shall have a test sensitivity of at least 1×10^{-3} standard cubic centimeters per second of air (std-cm^3/sec) and there shall be no detectable pressure rise. A higher sensitivity acceptance and maintenance test may be required as discussed in Condition 7.b(5), below.

(3) Before leak testing, the following closure bolt and valve torque specifications:
(a) The cask lid bolts shall be torqued to 235 ± 15 ft-lbs.
(b) The gas sample valve and drain valve shall be torqued to 20 ± 2 ft-lbs.

(4) During wet loading operations and prior to leak testing, the removal of water and residual moisture from the containment vessel in accordance with the following specifications:
(a) Cask evacuation to a pressure of 0.2 psia (10 mm Hg) or less for a minimum of 1 hour.
(b) Verifying that the cask pressure rise is less than 0.1 psi in 10 minutes.

(5) Before shipment, independent verification of the material condition of the neutron shield as described in SAR Section 7.1.1.4 or 7.1.2.4.

b. All fabrication acceptance tests and maintenance shall be performed in accordance with detailed written procedures. Procedures for fabrication, acceptance testing, and maintenance shall be developed using the specifications contained within the application and shall include the following provisions:

(1) All containment boundary welds, except the final fabrication weld joint connecting the cask body wall to the bottom plate, shall be radiographed and liquid-penetrant examined in accordance with ASME Code Section III, Division 1, Subsection NB. Examination of the final fabrication weld joint connecting the cask body wall to the bottom plate may be ultrasonic and progressive liquid penetrant examined in lieu of radiographic and liquid penetrant examination.

(2) The upper lifting trunnions and redundant lifting sockets shall be load tested, in the cask axial direction, to 300 percent of their maximum working load (79,500 lbs. minimum) per trunnion and per lifting socket, in accordance with the requirements of ANSI N14.6. The upper and lower lifting trunnions shall be load tested, in the cask transverse direction, to 150 percent of their maximum working load (20,625 lbs. minimum) per trunnion, in accordance with the requirements of ANSI N14.6.

(3) The cask containment boundary shall be pressure tested to 150% of the design pressure per 10 CFR 71.85(b). The minimum test pressure shall be 120 psig.

(4) All containment seals shall be replaced within the 12-month period prior to each shipment.

(5) A fabrication leakage test shall be performed on all containment components including the O-ring seals prior to first use. Additionally, all containment seals shall be leak tested after the third use of each package and within the 12-month period prior to each shipment. Any replaced or repaired containment system component shall be leak tested. The leakage tests shall verify that the containment boundary leakage rate does not exceed the design leakage rate of 1×10^{-7} std-cm^3/sec. The leak tests shall have a test sensitivity of at least 5×10^{-8} std-cm^3/sec.

(6) The depleted uranium shield shall be gamma scanned with 100 percent inspection coverage during fabrication to ensure that there are no shielding discontinuities. The neutron shield supplier shall certify that the shield material meets the minimum specified requirements (proprietary) used in the applicant's shielding analysis.

(7) Qualification and verification tests to demonstrate the crush strength of each aluminum honeycomb type and lot to be utilized in the impact limiters shall be performed.

(8) The boron carbide pellets, fuel support structure and fuel cavity dimensions, and ^{235}U content in the depleted uranium shall be fabricated and verified to be within the specifications of Table 2 to ensure criticality safety.

Table 2

Specified Parameter	Minimum	Maximum
B_4C boron enrichment	96 wt% ^{10}B	N/A
Diameter of each B_4C pellet	0.426 in	0.430 in
Height of each B_4C pellet stack	7.986 in	8.046 in
Mass of ^{10}B in each B_4C pellet stack	31.5 g	N/A
Mass of each B_4C pellet stack	43.0 g	45.0 g
Diameter of each fuel support structure hole	0.432 in	0.44 in
Fuel support structure nominal hole pitch	N/A	0.55 in
Fuel support structure hole depth minus B_4C pellet-stack height (at room temperature)	0.009 in	0.129 in
Thickness of each fuel support structure panel	0.600 in	0.620 in
Fuel cavity width	N/A	9.135 in
^{235}U content in depleted uranium shielding material	N/A	0.2 wt%

8. This package is approved for exclusive-use transport by rail, truck or marine.

9. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.12.

10. Expiration Date: October 31, 2003.

REFERENCES

General Atomics Safety Analysis Report for the GA-4 Legal Weight Truck Spent Fuel Shipping Cask, Revision G (Proprietary) and Revision H (Non-Proprietary), transmitted by letter dated August 5, 1998.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

William F. Kane

William F. Kane, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: October 27, 1998

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)*

General Electric Company
Vallecitos Nuclear Center
6705 Vallecitos Road
Sunol, CA 94586

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

General Electric Company application dated December 12, 2000, as supplemented.

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.: 2000

(2) Description

A steel encased lead shielded shipping cask. The cask is within a double-walled overpack with toroidal shell impact limiters at each end. The overall dimensions are approximately 131.5 inches in height and 72.0 inches in diameter. The cask is transported in the upright or horizontal position. The gross weight of the package is approximately 33,550 lbs.

The cask is constructed of two concentric 1-inch thick 304 stainless steel cylindrical shells (ASTM A 240) joined at the bottom end to a 6-inch thick 304 stainless steel forging (ASTM A 182). The annulus between the two shells is filled with lead approximately 4 inches thick. The cask is approximately 71.0 inches in height and has an outer diameter of 38.5 inches. The cask cavity is approximately 26.5 inches in diameter and 54.0 inches deep.

The cask lid is 304 stainless steel and lead, has a stepped design, and is fully recessed into the cask top flange. The lid is secured to the cask body by 15, 1.25-inch diameter socket head screws. The cask is sealed by elastomeric O-rings bonded to a thin aluminum disc-shaped ring. The cask is equipped with a seal test port on the side of the cask body, a vent port in the cask lid, and a drain port near the bottom of the cask.

The cask is positioned within an overpack constructed from two 0.5-inch thick concentric 304 stainless steel cylindrical shells (ASTM A 240). The shells are separated radially by eight equally spaced tubes and horizontally by two tube sections. A 304 stainless steel toroidal shell impact limiter is attached to each end of the overpack. The overpack opens just above the lower impact limiter for access to the cask. The top of the overpack is joined to the base by 15, 1-3/8-inch diameter shoulder screws.

5(a) (2) Description (Continued)

Gussets on the top and bottom impact limiters provide tie-down points for the package. The cask body is equipped with attachment plates for lifting devices. The cask lifting devices are detached during transport.

(3) Drawings

(i) The packaging is constructed and assembled in accordance with General Electric Company Drawing Nos. 129D4946, Rev. 10; 105E9520, Rev. 4; and 105E9521, Rev. 5.

(ii) Packaging Serial No. 2001 is constructed and assembled in accordance with General Electric Company Drawing Nos. 129D4946, Rev. 10; 101E8718, Rev. 12; and 101E8719, Rev. 12.

(iii) The HFIR fuel basket and liner are constructed and assembled in accordance with General Electric Company Drawing No. 105E9523, Rev. 3.

(iv) The multifunctional rack is constructed and assembled in accordance with General Electric Company Drawing No. 105E9555, Rev. 2.

(v) The barrel rack is constructed and assembled in accordance with General Electric Company Drawing No. 166D8066, Rev. 2.

(vi) The material basket is constructed in accordance with General Electric Company Drawing No. 183C8356, Rev. 2. The material basket may be used with the multifunctional rack and the barrel rack.

(vii) The TSR fuel basket is constructed and assembled in accordance with General Electric Company Drawing No. 105E9560, Rev. 2.

(viii) The MTR fuel basket is constructed and assembled in accordance with General Electric Company Drawing No. 105E9557, Rev. 9.

(b) Contents

(1) Type and form of material

(i) Irradiated fuel rods, which may be cut or segmented.

(ii) Byproduct, source, or special nuclear material in solid form.

(iii) Irradiated High Flux Isotope Reactor (HFIR) fuel assembly, positioned within the HFIR fuel basket and liner as specified in 5(a)(3). The HFIR fuel assembly is fabricated in accordance with Oak Ridge National Laboratory Drawing Nos. M-11524-OH-101-D, Rev. 0, and M-11524-OH-102-D, Rev. 0.

5.(b) (1) Type and form of material (Continued)

(iv) Irradiated Tower Shielding Reactor (TSR) fuel elements, positioned within the TSR fuel basket specified in 5(a)(3).

(v) Irradiated MTR-type fuel assemblies, positioned within the MTR fuel basket specified in 5(a)(3). The fuel assemblies may be sectioned only in the non-fuel bearing region of the assembly. The fuel assemblies are composed of aluminum clad plates, and are limited as follows:

Fuel material	U_3O_8	UAl_x	U_{METAL}
Max. uranium enrichment (w/o U-235)	94.0	94.0	95.0
Max. active fuel thickness (in)	0.023	0.020	0.020
Min. clad thickness (in)	0.014	0.015	0.015
Max. U-235 per fuel assembly (g)	355	290	110
Max. U-235 mass per fuel basket cell (g)	710	580	220
Max. burnup (GWd/MTU)	568	568	568
Min. cool time (days)	120	120	120

Fuel material	U_3Si_2	UAl_x
Max. uranium enrichment (w/o U-235)	20.0	20.0
Max. active fuel thickness (in)	0.020	0.100
Min. clad thickness (in)	0.015	0.010
Max. U-235 per fuel assembly (g)	347	150
Max. U-235 mass per fuel basket cell (g)	694	300
Max. burnup (GWd/MTU)	122	122
Min. cool time (days)	120	120

Note: The enrichments, masses, and dimensions shall be based on values prior to irradiation.

5.(b) (1) Type and form of material (Continued)

(vi) Irradiated TRIGA fuel elements, positioned with the MTR fuel basket specified in 5(a)(3). The fuel material consists of $UZrH_x$ in cylindrical elements, with aluminum, stainless steel, or inconel cladding. The H to Zr ratio in the fuel ranges from approximately 1.0 to 1.7. Some fuel elements contain graphite reflectors in each end of the fuel element. The fuel elements are limited as follows:

Approximate rod diameter (in)	1-1/2	½	1-1/2	1-1/2	½
Graphite reflectors	With or without reflectors	With or without reflectors	With reflectors	With reflectors	Without reflectors
Uranium concentration in fuel (w/o U)	8 - 45	10 - 45	8.5 min.	8.5 min.	10 min.
Max. rod length (in)	30	30	30	30	30
Max. active fuel length (in)	15	22	15	15	22
Min. clad thickness (in)	0.02	0.016	0.02	0.02	0.016
Max. uranium enrichment (w/o U-235)	20.0	20.0	70.0	94.0	94.0
Max. active fuel diameter (in)	1.435	0.51	1.435	1.435	0.51
Max. U-235 per rod (g)	165	44 (max. 15 rods per basket cell) 33 (max. 20 rods per basket cell)	140	220	44 (max. 15 rods per basket cell) 33 (max. 20 rods per basket cell)
Max. U-235 mass per fuel basket cell (g)	560	660	560	660	660
Max. burnup (GWd/MTU)	427	427	427	568	568
Min. cool time (days)	120	120	120	120	120

Note: The enrichments, masses, and dimensions shall be based on values prior to irradiation.

5.(b) (2) Maximum quantity of material per package

Not to exceed 5,450 lbs, including fuel baskets, carrier racks, shoring, secondary containers, and shielding liner.

(i) For the contents described in 5(b)(1)(i):

600 watts decay heat; and

Fissile contents not to exceed 1175 grams U-235 equivalent mass with initial enrichment not to exceed 5 weight percent in the fissile isotope; minimum pellet diameter of 0.3 inch, maximum burnup of 45 GWd/MTU, and minimum cooling time of 120 days; or

Fissile contents not to exceed 1750 grams U-235 equivalent mass with initial enrichment not to exceed 5 weight percent in the fissile isotope; minimum pellet diameter of 0.35 inch, maximum burnup of 38 GWd/MTU, and minimum cooling time of 120 days. Fuel rods must be contained in closed, 5-inch schedule 40 pipe, with a maximum of 437.5 grams U-235 equivalent per pipe; or

Fissile contents not to exceed 242 grams U-235 equivalent mass with initial enrichment not to exceed 5 weight percent in the fissile isotope; minimum pellet diameter of 0.3 inch, maximum burnup of 52 GWd/MTU, and minimum cooling time of 180 days.

(ii) For the contents described in 5(b)(1)(ii):

2000 watts decay heat. Fissile contents not to exceed 500 grams U-235 equivalent mass. Carrier racks specified in 5(a)(3)(iv) or 5(a)(3)(v) must be used for contents exceeding 600 watts decay heat per package.

(iii) For the contents described in 5(b)(1)(iii):

One HFIR fuel assembly. The fuel assembly is composed of one inner fuel element, with up to 2628 grams U-235, and one outer fuel element, with up to 6872 grams U-235. The maximum uranium enrichment is 93.2 weight percent U-235. The maximum burnup per assembly is 2300 MWd, the minimum cool time is two years. Decay heat not to exceed 600 watts per package.

(iv) For the contents described in 5(b)(1)(iv):

A maximum of 4393 grams U-235 per package. The maximum uranium enrichment is 94.0 weight percent U-235. Decay heat not to exceed 35 watts per package. The TSR fuel elements must be positioned and limited within the TSR fuel basket as follows:

Lower fuel basket section - Up to 4 upper or lower fuel elements, or a combination of upper and lower fuel elements, for a total U-235 mass of 1412 grams.

5.(b) (2) Maximum quantity of material per package (Continued)

Middle fuel basket section - Up to 4 fuel cover (lune) plates, for a total U-235 mass of 304 grams.

Upper fuel basket section - Up to 6 annular fuel elements plus one cylindrical fuel element, for a total U-235 mass of 2677 grams.

(v) For the contents described in 5(b)(1)(v):

Weight of contents, including fuel elements, spacers, shoring, and hardware, not to exceed 42.8 lbs per fuel basket cell.

Decay heat not to exceed any of the following: 1500 watts per package, 120 watts per cell, 35 watts per cell in the upper half of the fuel basket, 85 watts per cell in the lower half of the fuel basket, 765 watts in the lower half of the fuel basket (i.e., the lower half of all 21 cells combined).

Failed fuel elements are permitted provided the damage is limited to cladding defects due to corrosion, nicks, and scratches. Failed fuel elements must be structurally and geometrically intact.

(vi) For the contents described in 5(b)(1)(vi):

Weight of contents, including fuel elements, spacers, shoring, and hardware, not to exceed 42.8 lbs per fuel basket cell.

For stainless steel and inconel clad fuel, decay heat not to exceed any of the following: 1500 watts per package, 120 watts per cell, 35 watts per cell in the upper half of the fuel basket, 85 watts per cell in the lower half of the fuel basket, 765 watts in the lower half of the fuel basket (i.e., the lower half of all 21 cells combined).

For aluminum clad fuel, decay heat not to exceed either of the following: 630 watts per package, 30 watts per cell.

(c) Transport Index for Criticality Control (Criticality Safety Index)

Minimum transport index to be shown on label for nuclear criticality control:

For the contents described in 5(b)(1)(i), 5(b)(1)(ii), and 5(b)(1)(iii); and limited in 5(b)(2)(i), 5(b)(2)(ii), and 5(b)(2)(iii):	100
For the contents described in 5(b)(1)(iv), 5(b)(1)(v), and 5(b)(1)(vi); and limited in 5(b)(2)(iv), 5(b)(2)(v), and 5(b)(2)(vi):	0.0

6. Plutonium in excess of twenty curies per package must be in the form of metal, metal alloy or reactor fuel elements.

7. The U-235 equivalent mass is determined by U-235 mass plus 1.66 times U-233 mass plus 1.66 times Pu mass.

8. Bolt torque:

   The cask lid bolts must be torqued to 690 ft-lbs (lubricated).

   The bolts used to secure the top of the overpack to the overpack base must be torqued to 100 ft-lbs (dry).

9. (a) For any package containing organic or inorganic substances which could radiolytically generate combustible gases, determination must be made by tests and measurements or by analysis of a representative package such that the following criteria are met over a period of time that is twice the expected shipment time:

   (i) The hydrogen generated must be limited to a molar quantity that would be no more than 5% by volume (or equivalent limits for other inflammable gases) of the secondary container gas void if present at STP (i.e., no more than 0.063 g-moles/ft^3 at 14.7 psia and 70°F); or

   (ii) The secondary container and cask cavity must be inerted with a diluent to assure that oxygen must be limited to 5% by volume in those portions of the package which could have hydrogen greater than 5%.

   For any package delivered to a carrier for transport, the secondary container must be prepared for shipment in the same manner in which determination for gas generation is made. Shipment period begins when the package is prepared (sealed) and must be completed within twice the expected shipment time.

   (b) For any package containing materials with a radioactivity concentration not exceeding that for low specific activity material, and shipped within 10 days of preparation, or within 10 days after venting of drums or other secondary containers, the determination in (a) above need not be made, and the time restriction in (a) above does not apply.

10. Prior to each shipment (except for contents meeting the requirements of special form radioactive material), the package must be leak tested to 1×10^{-3} std cm^3/sec. Prior to first use, after the third use, and at least once within the 12-month period prior to each subsequent use, the package must be leak tested to 1×10^{-7} std cm^3/sec.

11. The cask must be vacuum dried prior to shipment if contents are loaded under water, or if water is introduced into the cask cavity. During shipments for which vacuum drying is performed, the cask cavity must be filled with helium.

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

    (a) Prior to each shipment the cask seal must be inspected. The seal must be replaced with a new seal if inspection shows any defects or every 12 months, whichever occurs first; and

    (b) Each package must meet the Acceptance Tests and Maintenance Program of Chapter 8 of the application, except that inspections in Section 8.2 of the application must be performed at least once within the 12-month period prior to each use; and

    (c) The package must be prepared for shipment and operated in accordance with the Operating Procedures of Chapter 7 of the application.

13. Appropriate carrier racks or shoring must be provided to minimize movement of contents during accident conditions of transport. A lead liner, as shown in General Electric Company Drawing No. 129D4922, Rev. 2, which was included in the March 29, 1989, supplement, may be used inside the cask.

14. Each batch of ethylene propylene seals must be tested in accordance with Section 8.1.4.2 of the application.

15. Fissile mass limits for reactor fuel are based on fissile mass prior to irradiation.

16. For the contents described in 5(b)(1)(v) and 5(b)(1)(vi), the package may be transported horizontally. For horizontal transport, the package must be secured to the truck bed with the top end of the package (closure end) facing the front (cab) of the truck.

17. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR §71.12.

18. Expiration date: March 31, 2006.

REFERENCES

General Electric Company application dated December 12, 2000.

Supplements dated: December 20, 2000; March 16 and 27, 2001; and March 22, 2002.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: April 22, 2002

NRC FORM 618
(3-96)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIALS PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. PACKAGE IDENTIFICATION NUMBER	d. PAGE NUMBER	e. TOTAL NUMBER PAGES
9233	5	USA/9233/B(U)	1	3

2. PREAMBLE

a. This certificate is issued to certify that the 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.
Four Skyline Drive
Hawthorne, NY 10532-2120

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION:

Transnuclear, Inc. application
dated November 22, 1988, as supplemented.

c. DOCKET NUMBER 71-9233

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-RAM

(2) Description

The package is a steel encased lead shielded cask with wood impact limiters attached at both ends. The cask is a right circular cylinder. The overall dimensions of the packaging are approximately 178 inches long and 92 inches diameter with the impact limiters installed. The cask body is approximately 129 inches long with an outer diameter of 51 inches. The cask cavity has a length of approximately 111 inches and an inside diameter of 35 inches. The cask body is made of a 0.75-inch stainless steel inner shell, a 5.88-inch thick lead annulus, a 1.5-inch thick stainless steel outer shell, a 0.5-inch thick inner bottom plate and a 2.5-inch thick outside bottom plate. The lead shielding is 6 inches thick in the bottom end of the cask. The outer shell of the cask body is covered with a stainless steel thermal shield. The closure lid consists of a 2.5-inch thick outer stainless steel plate and a 0.5-inch thick inner stainless steel plate separated by 6 inches of lead shielding. The lid is secured by sixteen 1.5-inch diameter closure bolts. Two concentric silicone O-rings are installed in grooves on the underside of the lid. The cask is equipped with a sealed leak test port between the O-rings, a vent port in the closure lid and a sealed drain port in the bottom of the cask.

Each impact limiter is attached to the cask by eight 1.75-inch diameter bolts. The cask is equipped with 6 trunnions, four at the top and two at the bottom.

The gross weight of the package is approximately 80,000 pounds, including maximum contents of 9,500 pounds.

Page 2 - Certificate No. 9233 - Revision No. 5 - Docket No. 71-9233

5.(a) Packaging (continued)

(3) Drawings

The packaging is constructed in accordance with Transnuclear, Inc. Drawing Nos. 990-701, Rev. 6; 990-702, Rev. 6; 990-703, Rev. 6; 990-704, Rev. 3; 990-705, Rev. 4; 990-706, Rev. 3; 990-707, Rev. 3; 990-708, Rev. 5; and 990-709, Rev. 1.

(b) Contents

(1) Type and Form of Material

Dry irradiated and contaminated non-fuel-bearing solid materials contained within a secondary container.

(2) Maximum quantity of material per package

Greater than Type A quantities of radioactive material which may include fissile material provided that the fissile material does not exceed the generally licensed mass limits specified in 10 CFR 71.18, 71.20 and 71.22. The contents may not exceed 2,000 times an A_2 quantity. The decay heat of the contents may not exceed 300 watts. The maximum gross weight of the contents, secondary container and shoring is limited to 9,500 pounds.

6. As appropriate, shoring must be used in the secondary container sufficient to prevent significant movement of the contents under accident conditions.

7. Both the inner cask cavity and the secondary container must be free of water when the package is delivered to a carrier for transport.

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

(a) Prior to each shipment, the lid seals must be inspected. The seals must be replaced with new seals if inspection shows any defects or every 12 months, whichever occurs first;

(b) The package shall be prepared for shipment and operated in accordance with the Operating Procedures of Section 7.0 of the application; and

(c) The package must meet the Acceptance Tests and Maintenance Program of Section 8.0 of the application.

9. The package authorized by the certificate is hereby approved for use under the general license provisions of 10 CFR §71.12.

10. Expiration date: January 31, 2005

REFERENCES

Transnuclear, Inc. application dated November 22, 1988.

Supplements dated: January 13, May 18, June 5, July 21, July 28, and August 11, 1989; January 4, 1990; December 18, 1997; August 20, 1998; and December 7, 1999.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: February 4, 2000

NRC FORM 618 (8-2000) 10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9234	16	71-9234	USA/9234/B(U)F	1	OF	3

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)*

Columbiana Hi Tech Front End, LLC
200 Railroad Street
P.O. Box 68
Columbiana, OH 44408

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Nuclear Containers, Inc. application dated January 11. 1993, as supplemented

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.: NCI-21PF-1

(2) Description

Overpack for 30-inch enriched uranium hexafluoride (UF_6) cylinders. The valve end of the cylinder may be equipped with a valve protection device. The overpack is a right circular cylinder constructed of two stainless steel shells with the volume between the shells filled with fire resistant, phenolic-foam per USAEC Specification SP-9, Rev. 1, and Supplement K/TL-729. The volume between the 1/4-inch thick end closure plates of the two shells is filled with oak wood blocks which are cross-laminations of 3 layers of boards glued and nailed together. A stepped and gasketed horizontal joint permits the top half of the overpack to be removed from the base. The package "halves" are secured with ten, 1-inch stainless steel toggle closures. The overpack is 43-5/8 inches O.D. by 92 inches long. The maximum gross weight of the package, including the valve protection device, is 8875 pounds.

(3) Drawing

The Model No. NCI-21PF-1 packaging is fabricated in accordance with Nuclear Containers, Inc. Drawing No. DED-206-B, Sheets 1 through 11, Rev. 5. The valve protection device and the valve protection device gauge are fabricated and assembled in accordance with United States Enrichment Corporation Drawing Nos. VPD-0001, Rev. 1, VPD-0002, Rev. 2, and VPD-0003, Rev. 1.

5.(b) Contents

(1) Type and form of material

Uranium hexafluoride contained within a Model 30B cylinder.

(2) Maximum quantity of material per package

5,020 pounds uranium hexafluoride. Uranium enriched to not more than 5 w/o in the U-235 isotope. The total quantity of radioactive material within a package may not exceed a Type A quantity.

(c) Transport Index for Criticality Control

Minimum transport index to be shown on label for nuclear criticality control: 5.0

6. The Model 30B cylinders must be fabricated, inspected, tested, and maintained in accordance with American National Standard N14.1 (1990 Edition). Cylinders must be fabricated in accordance with Section VIII, Division I, of the ASME (American Society of Mechanical Engineers) Boiler and Pressure Vessel Code and be ASME code stamped.

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

(a) Each packaging must meet the Acceptance Tests and Maintenance Program of Chapter 8 of the application.

(b) The package shall be prepared for shipment and operated in accordance with the Operating Procedures of Chapter 7 of the application.

(c) The torque on the overpack closures must be 110 ± 10 foot-pounds. Within the 12-month period prior to shipment, the torque must be checked in accordance with the procedure described in the supplement dated November 19, 1996.

8. Packagings manufactured by Nuclear Containers, Incorporated, during the period November 30, 1991, to October 1, 1994, and having NCI serial Nos. 487 through 619, but excluding 487A and 488A, are authorized for use.

9. Model No. NCI-21PF-1 packages must be equipped with the valve protection device described in 5(a)(3). The valve protection device must be installed in accordance with the procedures specified in the supplement dated November 30, 2000.

10. Prior to each shipment, the stainless steel components of the packaging must be visually inspected. Packagings in which stainless steel components show pitting, corrosion, cracking, or pinholes are not authorized for transport.

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9234	16	71-9234	USA/9234/B(U)F	3	OF	3

11. The Model 30B cylinder valve stem and plug may be tinned with ASTM B32, alloy 50A or Sn50 solder material, or a mixture of alloy 50A or Sn50 with alloy 40A or Sn40A material, provided the mixture has a minimum tin content of 45 percent.

12. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR §71.12.

13. Expiration date: December 31, 2003.

REFERENCES

Nuclear Containers, Inc. application dated January 11, 1993.

Supplements dated: September 10, 1993; July 21, 1994; November 19, 1996; February 26, April 21, May 15, July 9, and August 11, 1997; September 9, 1998; July 13 and November 30, 2000; and April 11, 2002.

United States Enrichment Corporation supplement dated: April 14, 1997.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: July 17, 2002

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9235	6	71-9235	USA/9235/B(U)F-85	1	OF	12

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)*

NAC International
3930 East Jones Bridge
Norcross, Georgia 30092

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

NAC International, Inc. application dated December 30, 1996, as supplemented

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.: NAC-STC

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

A steel, lead and polymer (NS4FR) shielded shipping cask for (a) directly loaded irradiated PWR fuel assemblies, (b) intact, damaged and/or the fuel debris of Yankee Class or Connecticut Yankee irradiated PWR fuel assemblies in a canister, and (c) non-fissile, solid radioactive materials (referred to hereafter as Greater Than Class C (GTCC) as defined in 10 CFR Part 61) waste in a canister. The cask body is a right circular cylinder with an impact limiter at each end. The package has approximate dimensions as follows:

Cavity diameter	71 inches
Cavity length	165 inches
Cask body outer diameter	87 inches
Neutron shield outer diameter	99 inches
Lead shield thickness	3.7 inches
Neutron shield thickness	5.5 inches
Impact limiter diameter	124 inches
Package length:	
without impact limiters	193 inches
with impact limiters	257 inches

The maximum gross weight of the package is about 260,000 lbs.

The cask body is made of two concentric stainless steel shells. The inner shell is 1.5 inches thick and has an inside diameter of 71 inches. The outer shell is 2.65 inches thick and has an outside diameter of 86.7 inches. The annulus between the inner and outer shells is filled with lead.

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

The inner and outer shells are welded to steel forgings at the top and bottom ends of the cask. The bottom end of the cask consists of two stainless steel circular plates which are welded to the bottom end forging. The inner bottom plate is 6.2 inches thick and the outer bottom plate is 5.45 inches thick. The space between the two bottom plates is filled with a 2-inch thick disk of a synthetic polymer (NS4FR) neutron shielding material.

The cask is closed by two steel lids which are bolted to the upper end forging. The inner lid (containment boundary) is 9 inches thick and is made of Type 304 stainless steel. The outer lid is 5.25 inches thick and is made of SA-705 Type 630, H1150 or 17-4PH stainless steel. The inner lid is fastened by 42, 1-1/2-inch diameter bolts and the outer lid is fastened by 36, 1-inch diameter bolts. The inner lid is sealed by two O-ring seals. The outer lid is equipped with a single O-ring seal. The inner lid is fitted with a vent and drain port which are sealed by O-rings and cover plates. The containment system seals may be metallic or Viton. Viton seals are used only for directly-loaded fuel that is to be shipped without long-term interim storage.

The cask body is surrounded by a 1/4-inch thick jacket shell constructed of 24 stainless steel plates. The jacket shell is 99 inches in diameter and is supported by 24 longitudinal stainless steel fins which are connected to the outer shell of the cask body. Copper plates are bonded to the fins. The space between the fins is filled with NS4FR shielding material.

Four lifting trunnions are welded to the top end forging. The package is shipped in a horizontal orientation and is supported by a cradle under the top forging and by two trunnion sockets located near the bottom end of the cask.

The package is equipped at each end with an impact limiter made of redwood and balsa. Two impact limiter designs consisting of a combination of redwood and balsa wood, encased in Type 304 stainless steel are provided to limit the g-loads acting on the cask during an accident. The predominately balsa wood impact limiter is designed for use with all the proposed contents. The predominately redwood impact limiters may only be used with directly loaded fuel or the Yankee-MPC configuration.

The contents are transported either directly loaded (uncanistered) into a stainless steel fuel basket or within a stainless steel transportable storage canister (TSC). The TSC, including its welded shield and structural lids, represents the separate inner container for the purposes of meeting 10 CFR 71.63.

The directly loaded fuel basket within the cask cavity can accommodate up to 26 PWR fuel assemblies. The fuel assemblies are positioned within square sleeves made of stainless steel. Boral or TalBor sheets are encased outside the walls of the sleeves. The sleeves are laterally supported by 31, ½-inch thick, 71-inch diameter stainless steel disks. The basket also has 20 heat transfer disks made of Type 6061-T651 aluminum alloy. The support disks and heat transfer disks are connected by six, 1-5/8-inch diameter by 161-inch long threaded rods made of Type 17-4 PH stainless steel.

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

The TSC shell, bottom plate, and welded shield and structural lids are fabricated from stainless steel. The bottom is a 1-inch thick steel plate for the Yankee-MPC and 1.75-inch thick steel plate for the CY-MPC. The shell is constructed of 5/8-inch thick rolled steel plate and is 70 inches in diameter. The shield lid is a 5-inch thick steel plate and contains drain and fill penetrations for the canister. The structural lid is a 3-inch thick steel plate. The canister contains a stainless steel fuel basket that can accommodate up to 36 intact Yankee Class fuel assemblies and Reconfigured Fuel Assemblies (RFAs), or up to 26 intact Connecticut Yankee fuel assemblies with RFAs, with a maximum weight limit of 35,100 lbs. Alternatively, a stainless steel GTCC waste basket is used for up to 24 containers of waste.

One TSC fuel basket configuration can store up to 36 intact Yankee Class fuel assemblies or up to 36 RFAs within square sleeves made of stainless steel. Boral sheets are encased outside the walls of the sleeves. The sleeves are laterally supported by 22 ½-inch thick, 69-inch diameter stainless steel disks, which are spaced about 4 inches apart. The support disks are retained by split spacers on eight 1.125-inch diameter stainless steel tie rods. The basket also has 14 heat transfer disks made of Type 6061-T651 aluminum alloy.

The second fuel basket is designed to store up to 26 Connecticut Yankee Zirc-clad assemblies enriched to 3.93 wt. percent, stainless steel clad assemblies enriched up to 4.03 wt. percent, RFAs, or damaged fuel in CY-MPC damaged fuel cans (DFCs). Zirc-clad fuel enriched to between 3.93 and 4.61 wt. percent, such as Westinghouse Vantage 5H fuel, must be stored in the 24-assembly basket. Assemblies approved for transport in the 26-assembly configuration may also be shipped in the 24-assembly configuration. The construction of the two basket configurations is identical except that two fuel loading positions of the 26-assembly basket are blocked to form the 24-assembly basket.

RFAs can accommodate up to 64 Yankee Class fuel rods or up to 100 Connecticut Yankee fuel rods, as intact or damaged fuel or fuel debris, in an 8x8 or 10x10 array of stainless steel tubes, respectively. Intact and damaged Yankee Class or Connecticut Yankee fuel rods, as well as fuel debris, are held in the fuel tubes. The RFAs have the same external dimensions as a standard intact Yankee Class, or Connecticut Yankee fuel assembly.

The TSC GTCC basket positions up to 24 Yankee Class or Connecticut Yankee waste containers within square stainless steel sleeves. The Yankee Class basket is supported laterally by eight 1-inch thick, 69-inch diameter stainless steel disks. The Yankee Class basket sleeves are supported full-length by 2.5-inch thick stainless steel support walls. The support disks are welded into position at the support walls. The Connecticut Yankee GTCC basket consists of GTCC waste containers supported full length by a 1.75-inch thick shell, which is laterally supported by twelve welded 1.25-inch thick outer ribs. The GTCC waste containers accommodate radiation activated and surface contaminated steel, cutting debris (dross) or filter media, and have the same external dimensions of Yankee Class or Connecticut Yankee fuel assemblies.

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

The Yankee Class TSC is axially positioned in the cask cavity by two aluminum honeycomb spacers. The spacers, which are enclosed in a Type 6061-T651 aluminum alloy shell, position the canister within the cask during normal conditions of transport. The bottom spacer is 14- inches high and 70-inches in diameter, and the top spacer is 28-inches high and also 70-inches in diameter.

The Connecticut Yankee TSC is axially positioned in the cask cavity by one stainless steel spacer located in the bottom of the cask cavity.

5.(a)(3) Drawings

(i) The cask is constructed and assembled in accordance with the following Nuclear Assurance Corporation (now NAC International) Drawing Nos.:

423-800, sheets 1-2, Rev. 6
423-802, sheets 1-6, Rev. 15
423-803, sheets 1-2, Rev. 4
423-804, sheets 1-3, Rev. 3
423-805, sheets 1-2, Rev. 4
423-806, Rev. 4
423-807, sheets 1-2, Rev. 1
423-811, sheets 1-2, Rev. 5
423-812, Rev. 2
423-900, Rev. 5
423-209, Rev. 0
423-210, Rev. 0
423-901, Rev. 2

(ii) For the directly loaded configuration, the basket is constructed and assembled in accordance with the following Nuclear Assurance Corporation (now NAC International) Drawing Nos.:

423-870, Rev. 2
423-871, Rev. 1
423-872, Rev. 6
423-873, Rev. 1
423-874, Rev. 2
423-875, sheets 1-2, Rev. 4

(iii) For the Yankee Class TSC configuration, the canister, and the fuel and GTCC waste baskets are constructed and assembled in accordance with the following NAC International Drawing Nos.:

455-800, sheets 1-2, Rev. 2
455-801, sheets 1-2, Rev. 3
455-820, Rev. 1
455-870, Rev. 4
455-871, sheets 1-2, Rev. 6
455-872, sheets 1-2, Rev. 9
455-873, Rev. 3
455-881, sheets 1-3, Rev. 6
455-887, sheets 1-3, Rev. 4
455-888, sheets 1-2, Rev. 6
455-891, sheets 1-2, Rev. 1
455-892, sheets 1-2, Rev. 2
455-893, Rev. 3
455-894, Rev. 2
455-895, sheets 1-2, Rev. 4

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

(iv) For the Yankee Class TSC configuration, RFAs are constructed and assembled in accordance with the following Yankee Atomic Electric Company Drawing Nos.:

YR-00-060, Rev. 1
YR-00-061, Rev. 1
YR-00-062, Rev. 1
YR-00-063, Rev. 1
YR-00-064, Rev. 1
YR-00-065, Rev. 1
YR-00-066, Rev. 1

(v) The Balsa Impact Limiters are constructed and assembled in accordance with the following NAC International Drawing Nos.:

423-257, Rev. 2
423-258, Rev. 2
423-843, Rev. 2
423-859, Rev. 0

(vi) For the Connecticut Yankee TSC configuration, the canister and the fuel and GTCC waste baskets are constructed and assembled in accordance with the following NAC International Drawing Nos.:

414-801, sheets, 1-2 Rev. 1
414-820, Rev. 0
414-870, Rev. 2
414-871, sheets 1-2, Rev. 2
414-872, sheets 1-2, Rev. 2
414-873, Rev. 0
414-881, sheets 1-2, Rev. 3
414-882, sheets 1-2, Rev. 3
414-887, sheets 1-4, Rev. 3
414-888, sheets 1-2, Rev. 3
414-889, sheets 1-2, Rev. 3
414-891, Rev. 3
414-892, sheets 1-3, Rev. 3
414-893, sheets 1-2, Rev. 2
414-894, Rev. 0
414-895, sheets 1-2, Rev. 3

vii) For the Connecticut Yankee TSC configuration, DFCs and RFAs are constructed and assembled in accordance with the following NAC International Drawing Nos.:

414-901, Rev. 0
414-902, sheets 1-2, Rev. 1
414-903, sheets 1-2, Rev. 1
414-904, sheets 1-3, Rev. 0

5.(b) Contents

(1) Type and form of material

(i) Irradiated PWR fuel assemblies with uranium oxide pellets. Each fuel assembly may have a maximum burnup of 45 GWD/MTU. The minimum fuel cool time is defined in the Fuel Cool Time Table, below. The maximum heat load per assembly is 850 watts. Prior to irradiation, the fuel assemblies must be within the following dimensions and specifications:

Assembly Type	14x14	15x15	16x16	17x17	17x17 (OFA)	Framatome-Cogema 17x17
Cladding Material	Zirc-4	Zirc-4	Zirc-4	Zirc-4	Zirc-4	Zirconium Alloy
Maximum Initial Uranium Content (kg/assembly)	407	469	402.5	464	426	464
Maximum Initial Enrichment (wt% ^{235}U)	4.2	4.2	4.2	4.2	4.2	4.5
Minimum Initial Enrichment (wt% ^{235}U)	1.7	1.7	1.7	1.7	1.7	1.7
Assembly Cross-Section (inches)	7.76 to 8.11	8.20 to 8.54	8.10 to 8.14	8.43 to 8.54	8.43	8.425 to 8.518
Number of Fuel Rods per Assembly	176 to 179	204 to 216	236	264	264	264(1)
Fuel Rod OD (inch)	0.422 to 0.440	0.418 to 0.430	0.382	0.374 to 0.379	0.360	0.3714 to 0.3740
Minimum Cladding Thickness (inch)	0.023	0.024	0.025	0.023	0.023	0.0204
Pellet Diameter (inch)	0.344 to 0.377	0.358 to 0.390	0.325	0.3225 to 0.3232	0.3088	0.3224 to 0.3230
Maximum Active Fuel Length (inches)	146	144	137	144	144	144.25

Note (1) - Fuel rod positions may also be occupied by solid poison shim rods or solid zirconium alloy or stainless steel fill rods.

5.(b)(1)(i) Contents - Type and Form of Material - Irradiated PWR fuel assemblies (Continued)

FUEL COOL TIME TABLE
Minimum Fuel Cool Time in Years

	Fuel Assembly Burnup (BU)															
Uranium Enrichment (wt% U-235)	BU ≤ 30 GWD/MTU				30 < BU ≤ 35 GWD/MTU				35 < BU ≤ 40 GWD/MTU				40 < BU ≤ 45 GWD/MTU			
Fuel Type	14x14	15x15	16x16	17x17	14x14	15x15	16x16	17x17	14x14	15x15	16x16	17x17	14x14	15x15	16x16	17x17
1.7≤E<1.9	8	7	6	7	10	10	7	9	--	--	--	--	--	--	--	--
1.9≤E<2.1	7	7	5	7	9	9	7	8	12	13	9	11	--	--	--	--
2.1≤E<2.3	7	7	5	6	9	8	6	8	11	11	8	10	--	--	--	--
2.3≤E<2.5	6	6	5	6	8	8	6	7	10	10	8	9	14	15	12	14
2.5≤E<2.7	6	6	5	6	8	7	6	7	10	9	7	9	13	14	10	12
2.7≤E<2.9	6	6	5	5	7	7	5	6	9	9	7	8	12	12	9	11
2.9≤E<3.1	6	5	5	5	7	7	5	6	9	8	6	8	11	11	8	10
3.1≤E<3.3	5	5	5	5	7	6	5	6	8	8	6	7	10	10	8	9
3.3≤E<3.5	5	5	5	5	6	6	5	6	8	7	6	7	10	10	7	9
3.5≤E<3.7	5	5	5	5	6	6	5	6	7	7	6	7	9	9	7	9
3.7≤E<3.9	5	5	5	5	6	6	5	6	7	7	6	7	9	9	7	9
3.9≤E<4.1	5	5	5	5	6	6	5	6	7	7	6	7	8	9	7	9
4.1≤E≤4.2	5	5	5	5	5	6	5	6	6	7	6	7	8	8	7	9
4.2<E<4.3	--	--	--	5[(1)]	--	--	--	6[(1)]	--	--	--	7[(1)]	--	--	--	9[(1)]
4.3≤E≤4.5	--	--	--	5[(1)]	--	--	--	6[(1)]	--	--	--	7[(1)]	--	--	--	8[(1)]

Note (1): Framatome-Cogema 17x17 fuel only.

5.(b)(1) Contents - Type and Form of Material (Continued)

(ii) Irradiated intact Yankee Class PWR fuel assemblies or RFAs within the TSC. The maximum initial fuel pin pressure is 315 psig. The fuel assemblies consist of uranium oxide pellets with the specifications, based on design nominal or operating history record values, listed below:

Assembly Manufacturer/Type	UN 16x16	CE [(1)] 16x16	West. 18x18	Exxon [(2)] 16x16	Yankee RFA
Cladding Material	Zircaloy	Zircaloy	SS	Zircaloy	Zirc/SS
Maximum Number of Rods per Assembly	237	231	305	231	64
Maximum Initial Uranium Content (kg/assembly)	246	240	287	240	70
Maximum Initial Enrichment (wt% ^{235}U)	4.0	3.9	4.94	4.0	4.94
Minimum Initial Enrichment (wt% ^{235}U)	4.0	3.7	4.94	3.5	3.5
Maximum Assembly Weight (lbs)	850	850	900	850	850
Maximum Burnup (Mwd/MTU)	32,000	36,000	32,000	36,000	36,000
Maximum Decay Heat per Assembly (kW)	0.28	0.347	0.28	0.34	0.11
Minimum Cool Time (yrs)	11.0	8.1	19.0	9.0	8.0
Maximum Active Fuel Length (in)	91	91	92	91	92

Notes:

(1) Combustion Engineering (CE) fuel with a maximum burnup of 32,000 Mwd/MTU, a minimum enrichment of 3.5 wt percent ^{235}U, a minimum cool time of 8.0 years, and a maximum decay heat per assembly of 0.304 kW is authorized.

(2) Exxon assemblies with stainless steel in-core hardware shall be cooled a minimum of 16.0 years with a maximum decay heat per assembly of 0.269 kW.

5.(b)(1) Contents - Type and Form of Material (Continued)

(iii) Solid, irradiated, and contaminated hardware and solid, particulate debris (dross) or filter media placed in a GTCC waste container, provided the quantity of fissile material does not exceed a Type A quantity and does not exceed the mass limits of 10 CFR 71.53.

(iv) Irradiated intact Connecticut Yankee Class PWR fuel assemblies, RFAs, or DFCs within the TSC. The maximum initial fuel pin pressure is 475 psig. The fuel assemblies consist of uranium oxide pellets with the specifications, based on design nominal or operating history record values, listed below:

Assembly Manufacturer/Type	West. 15x15	B&W, & other (GA, NUMEC) 15x15	West. Vantage 5H	CY-MPC[1] RFA	CY-MPC[2] DFC
Cladding Material	SS	Zircaloy	Zircaloy	Zirc/SS	Zirc/SS
Maximum Number of Assemblies	26	26	24	4	4
Maximum Initial Uranium Content (kg/assembly)	433.7	397.1	390	212	433.7
Maximum Initial Enrichment (wt% ^{235}U)	4.03	3.93	4.61	4.61[3]	4.61[3]
Minimum Initial Enrichment (wt% ^{235}U)	3.0	2.95	2.95	2.95	2.95
Maximum Assembly Weight (lbs)	1,500	1,380	1,230	1,500	1,500
Maximum Burnup (Mwd/MTU)	38,000	43,000	43,000	43,000	43,000
Maximum Decay Heat per Assembly (kW)	0.654	0.654	0.654	0.321	0.654
Minimum Cool Time (yrs)	10.0	10.0	10.0	10.0	10.0
Maximum Active Fuel Length (in)	121.8	121.35	120.6	121.8	121.8

Notes:
1. Reconfigured Fuel Assemblies (RFA) must be loaded in one of the 4 oversize fuel loading positions.
2. Damaged Fuel Cans (DFC) must be loaded in one of the 4 oversize fuel loading positions.
3. Enrichment of the fuel within each DFC or RFA is limited to that of the basked configuration in which it is loaded.

5.(b) Contents (Continued)

(2) Maximum quantity of material per package

(i) For the contents described in Item 5.(b)(1)(i): 26 PWR fuel assemblies with a maximum total weight of 39,650 lbs. and a maximum decay heat not to exceed 22.1 kW per package.

(ii) For the contents described in Item 5.(b)(1)(ii): Up to 36 intact fuel assemblies to the maximum content weight limit of 30,600 lbs. with a maximum decay heat of 12.5 kW per package. Intact fuel assemblies shall not contain empty fuel rod positions and any missing rods shall be replaced by a solid Zircaloy or stainless steel rod that displaces an equal amount of water as the original fuel rod. Mixing of intact fuel assembly types is authorized.

(iii) For intact fuel rods, damaged fuel rods and fuel debris of the type described in Item 5.(b)(1)(ii): up to 36 RFAs, each with a maximum equivalent of 64 full length Yankee Class fuel rods and within fuel tubes. Mixing of directly loaded intact assemblies and damaged fuel (within RFAs) is authorized. The total weight of damaged fuel within RFAs or mixed damaged RFA and intact assemblies shall not exceed 30,600 lbs. with a maximum decay heat of 12.5 kW per package.

(iv) For the contents described in Item 5.(b)(1)(iii): for Connecticut Yankee GTCC waste up to 24 containers of GTCC waste. The total cobalt-60 activity shall not exceed 196,000 curies. The total weight of the waste containers shall not exceed 18,743 lbs. with a maximum decay heat of 5.0 kW. For all others, up to 24 containers of GTCC waste. The total cobalt-60 activity shall not exceed 125,000 curies. The total weight of the waste and containers shall not exceed 12,340 lbs. with a maximum decay heat of 2.9 kW.

(v) For the contents described in Item 5.(b)(1)(iv): up to 26 Connecticut Yankee fuel assemblies, RFAs or damaged fuel in CY-MPC DFCs for stainless steel clad assemblies enriched up to 4.03 wt. percent and Zirc-clad assemblies enriched up to 3.93 wt. percent. Westinghouse Vantage 5H fuel and other Zirc-clad assemblies enriched up to 4.61 wt. percent must be installed in the 24-assembly basket, which may also hold other Connecticut Yankee fuel types. The construction of the two basket configurations is identical except that two fuel loading positions of the 26 assembly basket are blocked to form the 24 assembly basket. The total weight of damaged fuel within RFAs or mixed damaged RFAs and intact assemblies shall not exceed 35,100 lbs. with a maximum decay heat of 0.654 kW per assembly for a canister of 26 assemblies. A maximum decay heat of 0.321 kW per assembly for Connecticut Yankee RFAs and of 0.654 kW per canister for the Connecticut Yankee DFCs is authorized.

5.(c) Transport Index for Criticality Control (Criticality Safety Index)

Minimum transport index to be shown on label for nuclear criticality control: 0.0

6. Known or suspected damaged fuel assemblies or rods (fuel with cladding defects greater than pin holes and hairline cracks) are not authorized, except as described in Item 5.(b)(2)(iii).

7. For contents placed in a GTCC waste container and described in Item 5.(b)(1)(iii): and which contain organic substances which could radiolytically generate combustible gases, a determination must be made by tests and measurements or by analysis that the following criteria are met over a period of time that is twice the expected shipment time:

   The hydrogen generated must be limited to a molar quantity that would be no more than 4% by volume (or equivalent limits for other inflammable gases) of the TSC gas void if present at STP (i.e., no more than 0.063 g-moles/ft^3 at 14.7 psia and 70°F). For determinations performed by analysis, the amount of hydrogen generated since the time that the TSC was sealed shall be considered.

8. For damaged fuel rods and fuel debris of the quantity described in Item 5.(b)(2)(iii) and 5.(b)(2)(v): if the total damaged fuel plutonium content of a package is greater than 20 Ci, all damaged fuel shall be enclosed in a TSC which has been leak tested at the time of closure. For the Yankee Class TSC the leak test shall have a test sensitivity of at least 4.0 X 10^{-8} cm^3/sec (helium) and shown to have a leak rate no greater than 8.0 X 10^{-8} cm^3/sec (helium). For the Connecticut Class TSC the leak test shall have a test sensitivity of at least 1.0 X 10^{-7} cm^3/sec (helium) and shown to have a leak rate no greater than 2.0 X 10^{-7} cm^3/sec (helium).

9. 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.

   (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, except that the thermal testing of the package (including the thermal acceptance test and periodic thermal tests) must be performed as described in NAC-STC Safety Analysis Report, Revision STC-02E, dated August 2002.

   (c) For packaging Serial Numbers STC-1 and STC-2, only one of these two packagings must be subjected to the thermal acceptance test as described in Section 8.1.6 of the NAC-STC Safety Analysis Report, Revision STC-02E.

10. Prior to transport by rail, the Association of American Railroads must have evaluated and approved the railcar and the system used to support and secure the package during transport.

11. Prior to marine or barge transport, the National Cargo Bureau, Inc., must have evaluated and approved the system used to support and secure the package to the barge or vessel, and must have certified that package stowage is in accordance with the regulations of the Commandant, United States Coast Guard.

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9235	6	71-9235	USA/9235/B(U)F-85	12	OF	12

12. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.12.

13. Expiration date: March 31, 2004.

REFERENCES

NAC International, Inc., application dated December 30, 1996.

NAC International, Inc. supplements dated April 30, May 7, July 28 and 31, 1997; August 7, December 5, 12, 19, and 30, 1998; January 15, February 12, 23, and 27, March 1 and 22, 1999; October 5, June 7, August 1, and November 8, 2000; December 6, 14, and 28, 2001; February 21, March 22, May 31, June 13, July 8 and 18, and August 23, September 23, October 23, and November 22, 2002.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

John D. Monninger, Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: December 19, 2002

NRC FORM 618 (8-2000) 10 CFR 71 U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9239	12	71-9239	USA/9239/AF	1	OF	4

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)*

Westinghouse Electric Company
LLC (WELCO)
P.O. Box 355
Pittsburgh, PA 15230

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Westinghouse Electric Corporation application dated February 14, 2002, as supplemented.

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 Nos.: MCC-3, MCC-4, and MCC-5

(2) Description

The MCC packages are shipping containers for unirradiated uranium oxide fuel assemblies. The packagings consist of a steel fuel element cradle assembly equipped with a strongback and an adjustable fuel element clamping assembly. The cradle assembly is shock mounted to a 13-gauge carbon steel outer container by shear mounts. The MCC-3 container is closed with thirty ½-inch T-bolts. The MCC-4 and MCC-5 containers are closed with fifty ½-inch T-bolts.

The MCC-3 and MCC-4 containers are permanently equipped with vertical Gd_2O_3 neutron absorber plates that are mounted on the center wall of the strongback. Additional horizontal Gd_2O_3 neutron absorber plates, mounted on the underside of the strongback, are required for the contents as specified.

The MCC-5 container is permanently equipped with both the vertical and horizontal Gd_2O_3 neutron absorber plates. Additional vee-shaped, guided Gd_2O_3 neutron absorber plates are required for the contents as specified.

Approximate dimensions of the MCC-3 packaging are 44-1/2 inches O.D. by 194-1/2 inches long. The gross weight of the packaging and contents is 7,544 pounds. The maximum weight of the contents is 3,300 pounds.

Approximate dimensions of the MCC-4 packaging are 44-1/2 inches O.D. by 226 inches long. The gross weight of the packaging and contents is 10,533 pounds. The maximum weight of the contents is 3,870 pounds.

5. (a) Packaging (continued)

Approximate dimensions of the MCC-5 packaging are 44-1/2 inches O.D. by 226 inches long. The gross weight of the packaging and contents is 10,533 pounds. The maximum weight of the contents is 3,700 pounds.

(3) Drawings

The MCC-3 packaging is constructed in accordance with Westinghouse Electric Corporation Drawing No. MCCL301, Sheets 1, 2, 3, and 4, Rev. 6.

The MCC-4 packaging is constructed in accordance with Westinghouse Electric Corporation Drawing No. MCCL401, Sheets 1, 2, 3, 4, and 5, Rev. 9.

The MCC-5 packaging is constructed in accordance with Westinghouse Electric Corporation Drawing No. MCCL501, Sheets 1 through 10, Rev. 6.

(b) Contents

(1) Type and form of material

Unirradiated PWR uranium dioxide fuel assemblies with a maximum uranium-235 enrichment of 5.0 weight percent.

The fuel assemblies shall meet the specifications given in Westinghouse Drawing No. 6481E15, Rev. 3, and in the following tables of Appendix 1-4 of the application, as supplemented:

Table 1-4.1, Rev. 10	Fuel Assembly Parameters 14x14 Type Fuel Assemblies
Table 1-4.2, Rev. 10	Fuel Assembly Parameters 15x15 Type Fuel Assemblies
Table 1-4.3, Rev. 10	Fuel Assembly Parameters 16x16 Type Fuel Assemblies*
Table 1-4.4, Rev. 10	Fuel Assembly Parameters 17x17 Type Fuel Assemblies*
Table 1-4.5, Rev. 10	Fuel Assembly Parameters VVER-1000 Type Fuel Assembly**

* 16x16 CE fuel assemblies and the 17x17 W-STD/XL fuel assemblies may be shipped only in the Model No. MCC-4 package.

** VVER-1000 fuel assemblies may be shipped only in the Model No. MCC-5 package.

5. (b) Contents (continued)

(2) Maximum quantity of material per package

Two (2) fuel assemblies

(c) Transport Index for Criticality Control (Criticality Safety Index)

Minimum transport index to be shown on label for nuclear criticality control: 0.4

6. For shipments of 14x14, 15x15, 16x16, and 17x17 fuel assemblies with U-235 enrichments of over 4.65 wt% and up to 5.0 wt%, horizontal Gd_2O_3 neutron absorber plates shall be positioned underneath each assembly. The horizontal absorber plates shall be placed horizontally on the underside of the strongback, as specified in the respective drawings in Condition 5(a)(3) for the MCC-3 and MCC-4 models.

7. For shipments of VVER-1000 fuel assemblies with U-235 enrichments of over 4.80 wt% and up to 5.0 wt%, a guided Gd_2O_3 neutron absorber plate shall be positioned underneath each assembly. The guided absorber plates shall be placed horizontally on the topside of the strongback, as specified in the drawings in Condition 5(a)(3) for the MCC-5 model.

8. Each fuel assembly must be unsheathed or must be enclosed in an unsealed plastic sheath which may not extend beyond the ends of the fuel assembly. The ends of the sheath may not be folded or taped in any manner that would prevent flow of liquids into or out of the sheathed fuel assembly.

9. The dimensions, minimum Gd_2O_3 loading and coating specifications, and acceptance testing of the neutron absorber plates shall be in accordance with the "Gd_2O_3 Neutron Absorber Plates Specifications," Appendix 1-6, Rev. 10, of the application, as supplemented. The minimum Gd_2O_3 coating areal density on the vertical and horizontal neutron absorber plates shall be 0.054 g-Gd_2O_3/cm^2. The minimum Gd_2O_3 coating areal density on guided neutron absorber plates shall be 0.027 g-Gd_2O_3/cm^2.

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

(a) Each package shall be prepared for shipment and operated in accordance with the "Routine Shipping Container Utilization Summary Operating Procedures," in Chapter 7 of the application, as supplemented; and

(b) Each package shall be tested and maintained in accordance with the "Acceptance Tests, Maintenance Program, and Recertification Program," in Chapter 8 of the application, as supplemented, and as specified in the respective drawings in Condition 5(a)(3) for the MCC-3, MCC-4, and MCC-5 models.

11. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR §71.12.

12. Expiration date: March 31, 2007.

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9239	12	71-9239	USA/9239/AF	4	OF	4

REFERENCES

Westinghouse Electric Corporation application dated February 14, 2002.

Supplements dated: March 6, 2002.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: March 14, 2002

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9246	3	71-9246	USA/9246/AF	1	OF	2

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)*

National Institute of Standards and
Technology
Gaithersburg, MD 20899

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

National Institute of Standards and Technology application dated February 7, 1992, as supplemented.

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.: ST

(2) Description

A closed steel pipe for the transport of an unirradiated research reactor fuel element. The pipe is a 5-1/2-inch OD carbon steel pipe, approximately 71 inches in length, with a closed bottom end and flanged top end. The top end is closed by a cover plate, which is 1/4-inch thick, and 6-1/2 inches in diameter, and a gasket. The cover plate is secured to the pipe flange by 8 cap screws. A wooden nozzle support and top support position the fuel assembly within the pipe. The package weighs approximately 75 pounds, including the fuel element.

(3) Drawing

The packaging is constructed and assembled in accordance with National Institute of Standards and Technology Drawing No. D-04-048, Sheet 1, Rev. 3, and Sheet 2, Rev. 3.

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9246	3	71-9246	USA/9246/AF	2	OF	2

5. (b) Contents

(1) Type and form of material

Unirradiated NBSR fuel element composed of enriched uranium and aluminum.

(2) Maximum quantity of material per package

One fuel element containing not more than 360 grams U-235. The total quantity of radioactive material within a package may not exceed a Type A quantity.

(c) Transport Index for Criticality Control

Maximum transport index to be shown on label for nuclear criticality control: 50.0

6. In addition to the requirements of Subpart G of 10 CFR Part 71, the package shall be prepared for shipment, operated, and maintained in accordance with the loading, unloading, and quality assurance procedures in the application. Prior to each shipment, the shipper shall make the determinations specified in the NIST "ST" Series Shipping Container Shipper's Checklist in the application.

7. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.12.

8. Expiration date: November 30, 2006.

REFERENCES

National Institute of Standards and Technology application dated February 7, 1992.

Supplements dated: February 14, 1992; August 7, 1996; and August 17, 2001.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: November 15, 2001

NRC FORM 618 (8-2000) 10 CFR 71	CERTIFICATE OF COMPLIANCE FOR RADIOACTIVE MATERIAL PACKAGES			U.S. NUCLEAR REGULATORY COMMISSION
a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE / PAGES
9248	17	71-9248	USA/9248/AF	1 OF 6

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)*

Framatome ANP, Inc.
2101 Horn Rapids Road
Richland, WA 99352-0130

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Siemens Power Corporation application dated November 25, 1998, as supplemented.

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 Nos.: SP-1, SP-2, and SP-3

(2) Description

Fuel assembly and fuel rod shipping containers. The packages consist of a right rectangular metal inner container and a wooden outer container, with cushioning material between the inner and outer containers.

The metal inner container is approximately 11-1/2 inches by 18 inches by 179-1/2 inches long and is positioned within a wooden outer container approximately 30 inches by 31 inches by 207 inches long. The SP-1 and SP-2 packagings differ in the length of the metal inner container and end piece. The SP-3 packagings have a reduced spacing between the fuel assembly channels and the outer surface of the metal inner container. Cushioning is provided between the inner and outer containers by phenolic impregnated honeycomb and ethafoam, or equivalent. Closure of the metal inner container and the wooden outer container is accomplished by bolts. A pressure relief (breather) valve is provided on the inner container, and is set for 0.5 psi differential. The maximum weight of the packaging and contents is 2,800 pounds.

(3) Drawings

The packagings are fabricated and assembled in accordance with the following Siemens Nuclear Power Corporation/Advanced Nuclear Fuels Corporation Drawing Nos.:

EMF-304,416, Rev. 13.
EMF-306,272, Rev. 9.
EMF-308,257, Rev. 5.
EMF-309,141, Rev. 1.
EMF-309,818, Rev. 0.

5.(a) (4) Product Containers

(i) Five-inch, Schedule 40, stainless steel pipe fitted with screw type or flange closure. The product container shall be vented if it contains materials which decompose at less than 1475 °F.

(ii) Rod shipping container as shown on Siemens Power Corporation Drawing No. EMF-309,141, Rev. 1.

(b) Contents

(1) Type and form of material

(i) UO_2 fuel assemblies in a 7 x 7, an 8 x 8, or a 9 x 9 square array with a maximum fuel cross-section area of 25 square inches, maximum fuel length of 174 inches and maximum average enrichment of 3.3 w/o U-235. Minimum zircaloy clad thickness is 0.025 inches; maximum pellet diameter is 0.555 inches. Any number of water rods in any arrangement is permitted.

(ii) UO_2 fuel assemblies in a 7 x 7, an 8 x 8, or a 9 x 9 square array with a maximum fuel length of 174 inches, and a maximum average enrichment between 3.3 to 4.0 w/o U-235. The maximum pellet diameter is 0.555 inch, and the minimum clad thickness is 0.025 inch. Any number of water rods in any arrangement is permitted, including part length rods. Each assembly contains at least 4 rods with nominal 2 weight percent Gd_2O_3, which are in non-perimeter locations and are symmetric about the diagonal.

(iii) UO_2 fuel assemblies with a maximum U-235 enrichment of 5.0 percent by weight, and a maximum average U-235 enrichment of 4.0 percent by weight. Each fuel assembly is made up of fuel rods in a 10 x 10 square array, with a maximum fuel cross section of 5.022 inches square, a nominal pitch of 0.511 inch, and a maximum fuel length of 174 inches. The maximum pellet diameter is 0.3356 inch, the minimum clad thickness is 0.0225 inch, and the maximum U-235 enrichment in any edge rod is 4.0 percent by weight. Each assembly contains at least 6 rods with nominal 2 weight percent Gd_2O_3, which are symmetric about the diagonal, and each assembly contains at least 4 water rods in the 4 central rod positions.

(iv) UO_2 fuel rods with a maximum U-235 enrichment of 5.0 percent by weight, and a minimum Gd_2O_3 content of 1.0 percent by weight. The rods may be clad with zircaloy, steel or aluminum. The rods have a maximum fuel pellet diameter of 0.5 inch, and a maximum fuel length of 169 inches.

5.(b) (1) Type and form of material (Continued)

(v) UO_2 fuel assemblies composed of fuel rods in a 10 x 10 square array, with a maximum fuel cross section of 5.0 inches square, and a maximum fuel length of 174 inches. The maximum U-235 enrichment is 5.0 weight percent, the maximum U-235 enrichment for all edge rods is 4.0 weight percent, and the maximum average enrichment, excluding perimeter rods and rods containing gadolinia (Gd_2O_3), is 4.0 weight percent U-235. The maximum pellet diameter is 0.35 inch, and the minimum clad thickness is 0.018 inch. Each assembly must have a water channel in the central 3 x 3 rod positions. Any number of additional water rods in any arrangement is permitted, including part length rods. Each assembly must include at least twelve rods with a minimum nominal content of 2.0 weight percent gadolinia (Gd_2O_3), in a pattern symmetric about one of the assembly diagonals. At least eight of the twelve gadolinia rods must be located in rows 2 and 9, and in columns 2 and 9 of the assembly.

(vi) UO_2 fuel assemblies composed of fuel rods in a 10 x 10 square array, with a maximum fuel cross section of 5.0 inches square, and a maximum fuel length of 174 inches. The maximum U-235 enrichment is 5.0 weight percent. The maximum pellet diameter is 0.35 inch, and the minimum clad thickness is 0.018 inch. Each assembly must have a water channel in the central 3 x 3 rod positions. Any number of additional water rods in any arrangement is permitted, including part length rods. Each assembly must include at least eight rods with a minimum nominal gadolinia (Gd_2O_3) content of 2.0 weight percent in all axial regions with enriched pellets. Additional gadolinia rod specifications are included in supplement dated April 30, 1996.

(vii) UO_2 fuel assemblies composed of fuel rods in a 9 x 9 square array, with a maximum fuel cross section of 5.0 inches square, and a maximum fuel length of 174 inches. The maximum U-235 enrichment is 5.0 weight percent. The maximum pellet diameter is 0.40 inch, and the minimum clad thickness is 0.015 inch. Each assembly must have a water channel in the central 3 x 3 rod positions. Any number of additional water rods in any arrangement is permitted, including part length rods. Each assembly must include at least eight rods with a minimum nominal gadolinia (Gd_2O_3) content of 2.0 weight percent in all axial regions with enriched pellets. Additional gadolinia rod specifications are included in supplement dated April 30, 1996.

(viii) UO_2 fuel assemblies composed of fuel rods in a 9 x 9 square array, with a maximum fuel cross-section of 25 square inches, a maximum fuel length of 174 inches, and a maximum average uranium enrichment of 4.0 weight percent U-235. The nominal pellet diameter is 0.370 inch. At least the center 3 x 3 rod locations must be a water channel. Each assembly must include at least eight rods with a minimum nominal gadolinia (Gd_2O_3) content of 2.0 weight percent in all axial regions with enriched pellets. The eight gadolinia rod locations are shown in Figure 1 of the supplement dated July 27, 1999.

5.(b) (1) Type and form of material (Continued)

(ix) UO_2 fuel assemblies composed of fuel rods in a 10 x 10 square array, with a maximum fuel cross section of 5.0 inches square, and a maximum fuel length of 174 inches. The maximum U-235 enrichment is 5.0 weight percent, the maximum U-235 enrichment for all edge rods is 4.75 weight percent, the maximum U-235 enrichment for the four (4) corner edge rods is 3.05 weight percent, and the maximum U-235 enrichment for the eight (8) edge rods immediately adjacent to the four corner edge rods is 3.55 weight percent. The pellet diameter is between 0.30 and 0.3957 inch. Each assembly must have a water channel in a central 3 x 3 position. Any number of additional water rods in any arrangement is permitted, including part length rods. Each assembly must include at least ten rods with a minimum nominal content of 2.0 weight percent gadolinia (Gd_2O_3) in all axial regions with the enriched pellets, and in a pattern symmetric about one of the assembly diagonals. At least ten gadolinia rods must be located in rows 2 and 9, and in columns 2 and 9 of the assembly and cannot be immediately adjacent to another one of the ten gadolinia rods; however, diagonally adjacent is permitted. An additional upper tie plate (UTP) shipping shim may be added between the UTP and the fueled region. This UTP shim may consist of a maximum of 345 g plastic or plastic composite.

(2) Maximum quantity of material per package

Total weight of contents (fuel assemblies, or fuel rods and rod shipping containers) not to exceed 1265 pounds. Total quantity of radioactive material within a package may not exceed a Type A quantity.

(i) For the contents described in 5(b)(1)(i), 5(b)(1)(ii), 5(b)(1)(iii), 5(b)(1)(v), 5(b)(1)(vi), 5(b)(1)(vii), 5(b)(1)(viii), and 5(b)(1)(ix):

Two full length fuel assemblies. Two short fuel assemblies may be substituted for each full length fuel assembly provided the two short assemblies are shipped end-to-end and the total fuel length does not exceed 174 inches.

(ii) For the contents described in 5(b)(1)(iv):

Two product containers specified in 5.(a)(4). Each product container may contain any number of loose fuel rods.

5.(c) Transport Index for Criticality Control (Criticality Safety Index)

Minimum transport index to be shown on label for nuclear criticality control:

(1) For contents described in 5(b)(1)(i), 5(b)(1)(ii), 5(b)(1)(iii), 5(b)(1)(iv), and 5(b)(1)(viii), and limited in 5(b)(2)(i) and 5(b)(2)(ii): 0.4

(2) For contents described in 5(b)(1)(v), 5(b)(1)(vi), 5(b)(1)(vii), 5(b)(1)(ix), and limited in 5(b)(2)(i): 1.0

6. Each fuel assembly must be unsheathed or must be enclosed in an unsealed, polyethylene sheath which may not extend beyond the ends of the fuel assembly. The ends of the sheath may not be folded or taped in any manner that would prevent the flow of liquids into or out of the sheathed fuel assembly.

7. Polyethylene shipping shims may be inserted between rods within fuel assemblies as follows:

(a) For contents described in 5(b)(1)(i) and 5(b)(1)(ii), up to a maximum of 0.20 gram H_2O hydrogen equivalent per cubic centimeter averaged over the assembly.

(b) For contents described in 5(b)(1)(v), up to a maximum of 0.25 gram H_2O hydrogen equivalent per cubic centimeter averaged over the assembly.

(c) For contents described in 5(b)(1)(viii), up to a maximum volume fraction of 0.13 averaged over the void volume of the assembly.

(d) For contents described in 5(b)(1)(iii), 5(b)(1)(vi), and 5(b)(1)(vii), polyethylene shipping shims are not permitted.

(e) For contents described in 5(b)(1)(ix), up to a maximum volume fraction of 0.14 averaged over the void volume of the assembly.

8. Only contents described in 5(b)(1)(viii) and 5(b)(1)(ix) are authorized for transport in Model No. SP-3 packages.

9. Maximum average enrichment means the highest average enrichment through any cross sectional plane of the assembly.

10. 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 dated November 25, 1998.

    (b) Each packaging must be acceptance tested and maintained in accordance with the Acceptance Tests and Maintenance Program in Chapter 8 of the application dated November 25, 1998.

11. The package authorized by this certificate is hereby authorized for use under the general license provisions of 10 CFR §71.12.

12. Expiration date: February 28, 2004.

REFERENCES

Siemens Power Corporation application dated November 25, 1998.

Supplements dated: December 2 and 15, 1998; February 23, April 12, and July 27, 1999; September 29 and November 17, 2000; February 6 and 9, March 21, and October 3, 2001.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date January 3, 2002

NRC FORM 618
(3-96)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIALS PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. PACKAGE IDENTIFICATION NUMBER	d. PAGE NUMBER	e. TOTAL NUMBER PAGES
9250	7	USA/9250/B(U)F-85	1	5

2. PREAMBLE

a. This certificate is issued to certify that the 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)*

Babcock and Wilcox Company
P. O. Box 785
Lynchburg, VA 24505

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION:

Babcock and Wilcox Company
application dated December 17, 1997.

c. DOCKET NUMBER 71-9250

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.: NNFD 5X22

(2) Description

A shipping container for unirradiated uranium of any enrichment. The outer packaging is a 16-gauge steel drum, approximately 22-1/2 inches in diameter and 34-3/4 inches high, with a heavy-duty clamp ring and forged lugs. The inner vessel (containment vessel) is a Schedule 40S stainless steel pipe with a welded bottom cap and a top weldneck flange. The inner vessel lid is a blind flange which is bolted to the weldneck flange with eight hex-head bolts. The closure includes double silicone O-ring seals and a leak-test port. The dimensions of the inner vessel are approximately 5 inches ID by 22 inches high. The inner vessel is centered within the outer drum by fiberboard and supported by plywood disks. The maximum weight of the package, including contents, is 300 pounds.

(3) Drawings

The packaging is constructed in accordance with Babcock & Wilcox Company Drawing Nos. 1220276E, Rev. 2, and 1220277E, Rev. 5.

(b) Contents

(1) Type and form of material

(i) Unirradiated uranium as solid compounds or alloys which do not decompose at temperatures up to 250 °F, and uranium oxides as powder or pellets. The uranium may be of any U-235 or U-233 enrichment. Carbide compounds are not authorized.

Page 2 - Certificate No. 9250 - Revision No. 7 - Docket No. 71-9250

(b) Contents (Continued)

(1) Type and form of material (Continued)

(ii) Unirradiated solid uranyl nitrate in the form of uranyl nitrate dihydrate crystals, which may have small amounts of uranyl trihydrate crystals interspersed. The uranyl nitrate crystals shall have a uranium content that is from 52.5 to 56.0 weight percent. The uranyl nitrate shall be packaged in Teflon primary containers that will not melt at temperatures up to 94 °C. The uranium may be of any U-235 enrichment.

(iii) Unirradiated uranium as solid metal. The uranium may be of any U-235 enrichment.

(iv) Unirradiated liquid uranyl nitrate solution in sealed glass containers or screw top plastic vials, each within one or more additional plastic vials with taped lids, and within a sealed product can or polyethylene bottle containing a sufficient amount of vermiculite to absorb twice the liquid contents present. The uranium may be of any U-235 enrichment. U-233 greater than a Type A quantity is not permitted.

(2) Maximum quantity of material per package and transport index for criticality control

The weight of the contents, including secondary containers, inserts, and other materials in the inner vessel, shall not exceed 50 pounds, and:

(i) For the material described in Items 5(b)(1)(i) and 5(b)(1)(ii), above, with a maximum H/U of 3, considering all sources of moderation in the inner vessel:

Fissile Material	Maximum Fissile Material per Package (kg)	Minimum Transport Index to be Shown on Label for Nuclear Criticality Control
U-235	9.0	2.0
U-235	1.6	0.5

Page 3 - Certificate No. 9250 - Revision No. 7 - Docket No. 71-9250

5.(b) Contents (continued)

(2) Maximum quantity of material per package and transport index for criticality control (continued)

(ii) For the material described in Items 5(b)(1)(i) and 5(b)(1)(ii), above, with a maximum H/U of 20, considering all sources of moderation in the inner vessel:

Fissile Material	Maximum Fissile Material per Package (kg)	Minimum Transport Index to be Shown on Label for Nuclear Criticality Control
U-233	0.5	1.8
U-235	4.0	2.0

(iii) For the material described in Item 5(b)(1)(iii), above, with a maximum H/U of 3, considering all sources of moderation in the inner vessel:

Fissile Material	Maximum Fissile Material per Package (kg)	Minimum Transport Index to be Shown on Label for Nuclear Criticality Control
U-235	9.0	2.5
U-235	1.6	0.5

(iv) For the material described in Item 5(b)(1)(iii), above, with a maximum H/U of 3, considering all sources of moderation in the inner vessel, and with a solid aluminum disk insert positioned in the inner vessel, as shown on Babcock & Wilcox Company Drawing No. 1220277 E, Rev. 5 (Part No. 6).

Fissile Material	Maximum Fissile Material per Package (kg)	Minimum Transport Index to be Shown on Label for Nuclear Criticality Control
U-235	9.0	2.0

Page 4 - Certificate No. 9250 - Revision No. 7 - Docket No. 71-9250

5.(b) Contents (continued)

(2) Maximum quantity of material per package and transport index for criticality control (continued)

(v) For the material described in Item 5(b)(1)(iii), above, with a maximum H/U of 20, considering all sources of moderation in the inner vessel:

Fissile Material	Maximum Fissile Material per Package (kg)	Minimum Transport Index to be Shown on Label for Nuclear Criticality Control
U-235	4.0	2.0
U-233	0.5	1.8

(vi) For the material described in Item 5(b)(1)(iv), above:

Fissile material shall not exceed 400 grams U-235. The quantity of uranyl nitrate shall not exceed 1000 mL of solution.

Minimum transport index to be shown on label for nuclear criticality control: 0.4

6. The vent holes on the outer steel drum shall be capped or taped closed during transport and storage to preclude entry of rain water into the packaging.

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

(a) Each package shall be operated and prepared for shipment in accordance with Chapter 7 of the application, as supplemented.

(b) Each package shall be acceptance tested and maintained in accordance with Chapter 8 of the application.

8. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR §71.12.

9. Expiration date: January 31, 2003.

Page 5 - Certificate No. 9250 - Revision No. 7 - Docket No. 71-9250

REFERENCES

Babcock and Wilcox Company application dated December 17, 1997.

Supplement dated: March 25, 1998

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Cass R. Chappell

Cass R. Chappell, Chief
Package Certification Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: May 14, 1998

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE	PAGES
9251	11	71-9251	USA/9251/AF	1 OF	3

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)*

Framatome ANP, Inc.
P.O. Box 11646
Lynchburg, VA 24506-1646

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

B&W Fuel Company application
dated May 26, 1992, as supplemented.

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

(l) Model No.: BW-2901

(2) Description

A shipping container for low-enriched uranium oxide powder and pellets, composed of an inner container, surrounded by insulating material, and an outer drum. The inner cross sectional dimensions of the inner container are a maximum 11.15-inch square by 29.5-inch long. The inner container is constructed of minimum 14-gauge steel, with bolted and gasketed top flange closure and welded bottom sheet. The inner container is centered and supported in an 18-gauge steel drum with 16-gauge head and DOT Specification 17H or an equivalent DOT UN1A2/Y1.5/100 closure by asbestos or ceramic sheet, plywood, hardboard, and insulating material. The drum has approximate inner cross sectional dimensions of 22.5-inch by 34-inch height. The uranium oxide is packaged in boxes, and wood boards position the boxes within the inner container. Three borated aluminum plates (approximately 25 inches by 9.25 inches by 0.375 inch) are positioned within the inner container. The maximum gross weight of the package is 660 pounds.

(3) Drawings

The packaging is constructed in accordance with B&W Fuel Company Drawing Nos. 1215597D, Rev. 5, 1215598B, Rev. 1, 1215599E, Rev. 4, and 1283759D, Rev. 0.

(b) Contents

(1) Type and form of material

(i) Sintered uranium oxide pellets enriched to a maximum 5.05 weight percent U-235. The minimum pellet diameter is 0.315 inch, and the maximum pellet diameter is 0.375 inch.

(ii) Uranium dioxide as powder, pellets, or any combination thereof, enriched to a maximum 5.05 weight percent U-235.

(2) Maximum quantity of material per package

370 pounds, with the U-235 content not to exceed 7.47 kg. The maximum weight of the uranium oxide, pellet boxes, and all packaging materials within the inner container is 427 pounds. Uranium oxide must be packaged in accordance with B&W Fuel Company Drawing Nos. 1215597D, Rev. 5, and 1283759, Rev. 0. The maximum mass of polyethylene within the inner container shall not exceed 1000 grams per package. Maximum quantity of radioactive material within a package may not exceed a Type A quantity.

(c) Transport Index for Criticality Control (Criticality Safety Index)

Minimum transport index to be shown on
label for nuclear criticality control: 0.7

6. Each package must be shipped with borated aluminum plates positioned within the inner container, on the top of, between, and on the bottom of the rows of pellet boxes. The three borated plates must have dimensions and boron concentration, and must be positioned in accordance with B&W Fuel Company Drawing No. 1215597D, Rev. 5.

7. For packages with fewer than six pellet boxes, solid aluminum or wood pellet box spacers must be substituted for pellet boxes. The pellet boxes, pellet box spacers, borated plates, and wood boards must provide a snug axial and cross sectional fit in the inner container.

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

(a) Each packaging must be maintained and acceptance tested in accordance with Chapter 8 of the application; and

(b) The package must be prepared for shipment and operated in accordance with the Operating Procedures of Chapter 7 of the application.

(c) Prior to each shipment the insert (containment vessel) gasket shall be inspected. This gasket shall be replaced if inspection shows any defects or every twelve (12) months, whichever occurs first.

9. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR §71.12.

10. Expiration date: October 31, 2007.

REFERENCES

B&W Fuel Company application dated May 26, 1992.

Supplements dated: August 3 and October 30, 1992; April 30, 1993; May 24 and September 22, 1995; February 29, April 22, and July 1, 1996; July 30, 1997; March 26, 1999; November 13, 2000; February 9, 2001; and August 16, 2002.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Jack Guttmann

E. William Brach, Director for
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: October 9, 2002

NRC FORM 618 (8-2000) 10 CFR 71	CERTIFICATE OF COMPLIANCE FOR RADIOACTIVE MATERIAL PACKAGES			U.S. NUCLEAR REGULATORY COMMISSION
1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE / PAGES
9252	4	71-9252	USA/9252/AF	1 OF 3

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)*

Framatome ANP, Inc.
P.O. Box 11646
Lynchburg, VA 24506-1646

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

B&W Fuel Company application dated March 9, 1993, as supplemented.

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.: 51032-2

(2) Description

A steel shipping container for fuel bundles, consisting of a strongback and fuel bundle clamping assembly, shock mounted to a steel outer container. Nine separator blocks, which are 6" x 8" x 8-1/2" long and have a 3/8" thick wall and a rectangular gusset plate welded inside, are bolted between fuel bundles. The outer container is composed of an 11 gauge steel shell approximately 43" diameter by 216" long. The maximum weight of the package, including contents, is 7,500 pounds.

(3) Drawings

The packaging is constructed and assembled in accordance with the following B&W Fuel Company Drawing Nos.: 1215926 C, Rev. 1; 1215929 D, Rev. 2; 1215930 D, Rev. 2; 1215931 D, Rev. 2; 1215932 D, Rev. 2; 1215933 D, Rev. 2; 1215934 C, Rev. 1; 1215935 D, Rev. 2; 1216010 D, Rev. 1.

(b) Contents

(1) Type and form of material

Unirradiated fuel assemblies, composed of uranium dioxide fuel pellets clad in zircaloy tubes. Uranium is enriched to a maximum of 5.05 w/o in the U-235 isotope. The fuel assemblies may contain inserted control rod assemblies. The fuel assemblies have the following specifications:

Type	15x15	15x15	17x17	17x17	15x15
Rods Per Assembly	208	204	264	264	204
Nominal Rod Pitch (in.)	0.568	0.563	0.501	0.496	0.5625
Maximum Pellet Diameter (in.)	0.3707	0.3671	0.3252	0.3232	0.3672
Maximum Pellet Density (%TD)	97.5	97.5	97.5	97.5	97.5
Nominal Clad OD (in.)	0.430	0.422	0.379	0.374	0.422
Nominal Clad ID (in.)	0.377	0.370	0.332	0.326	0.368
Assembly Cross Section (in.)*	8.520	8.445	8.517	8.432	8.438
Active Fuel Length (in.)	144	144	144	144	120
Maximum U-235 Loading (kg)	25.20	24.24	24.62	24.32	20.20

* Assembly cross section is the product of the nominal rod pitch and the number of rods per edge.

(2) Maximum quantity of material per package

Two fuel assemblies. Total weight of fuel assemblies, including control rod assemblies, not to exceed 3400 pounds. Maximum quantity of radioactive material within a package may not exceed a Type A quantity.

5. (c) Transport Index for Criticality Control

Minimum transport index to be shown on label for nuclear criticality control: 0.4

6. Each fuel assembly must be unsheathed or must be enclosed in an unsealed polyethylene sheath which will not extend beyond the ends of the fuel assemblies. The ends of the sheaths must not be folded or taped in any manner that would prevent the flow of liquids into or out of the sheathed fuel assemblies.

7. Hydrogenous shims are not permitted within the fuel assemblies.

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

   (a) The package shall be prepared for shipment and operated in accordance with Chapter 7.0 of the application.

   (b) Each packaging shall be maintained in accordance with Section 8.2 of the application.

   (c) Each packaging shall meet the acceptance tests in Section 8.1 of the application.

The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR §71.12.

10. Expiration date: September 30, 2003.

REFERENCES

B&W Fuel Company application dated March 9, 1993.

Supplements dated: May 10, and July 7, 1993; April 13, 1994; August 6, 1998; November 13, 2000; and February 9, 2001.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: February 14, 2001

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9253	7	71-9253	USA/9253/B(U)F-85	1	OF	7

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)*

U.S. Department of Energy
Washington, DC 20585

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Safety Analysis Report for the TN-FSV Package, dated March 31, 1993, as supplemented; Safety Analysis Report Addendum for the Oak Ridge Container in the TN-FSV Packaging, dated June 15, 2001, as supplemented.

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-FSV

(2) Description

A steel and lead shielded shipping cask for irradiated nuclear fuel. The cask has two shipping configurations: Configuration 1 for shipping irradiated Fort St. Vrain high temperature gas cooled reactor (HTGR) fuel elements, and Configuration 2 for shipping irradiated fuel parts and intact irradiated Peach Bottom Unit 1 fuel elements within a secondary containment vessel. The cask is a right circular cylinder, with a balsa and redwood impact limiter at each end. The package has approximate dimensions and weights as follows:

Cavity diameter	18 inches
Cavity length	199 inches
Cask body outer diameter	31 inches
Lead shield thickness	3.44 inches
Package overall outer diameter, including impact limiters	78 inches
Package overall length, including impact limiters	247 inches
Packaging weight (Configuration 1)	42,000 pounds
Gross package weight, including contents (Configurations 1 and 2)	47,000 pounds

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

The cask body is made of two concentric shells of Type 304 stainless steel, welded to a bottom plate and a top closure flange. The inner shell has an ID of 18 inches and is 1.12 inches thick. The outer shell has an OD of approximately 30 inches and is 1.5 inches thick. The annular space between the inner and outer shells is filled with lead. The bottom plate is 5.5-inch thick Type 304 stainless steel. The closure lid is 2.5-inch thick Type 304 stainless steel, and is fully recessed into the cask top flange. The lid is fastened to the cask body by 12, 1-inch diameter closure bolts. The lid is sealed with double O-ring seals with a leak test port. A vent port and drain port are sealed with single O-rings and cover plates. Configuration 1 uses silicone O-ring seals and Configuration 2 uses butyl O-ring seals. The cask body is covered with a stainless steel thermal shield composed of 0.25-inch thick stainless steel plate over a wire wrap. The impact limiters are constructed of balsa and redwood encased in stainless steel shells.

The cask has two lifting sockets bolted to the cask top flange. Two rear trunnions are provided for cask tie-down.

For Configuration 1:

Irradiated hexagonal HTGR fuel elements are shipped in Configuration 1. The fuel elements are stacked in a carbon steel fuel storage container, which has an OD of approximately 17.6 inches and an overall length of 195 inches. The fuel storage container has a 0.5-inch thick shell, a 2.0-inch thick bottom plate, and a 1.5-inch thick lid. The lid accommodates a removable depleted uranium plug.

For Configuration 2:

Irradiated fuel parts and intact Peach Bottom Unit 1 fuel elements are shipped in Configuration 2. Canisters, containing either fuel parts or a single intact Peach Bottom fuel element, are loaded into a separate, secondary containment vessel, the Oak Ridge Container. The Oak Ridge Container is composed of a right circular cylindrical vessel and a basket assembly. The stainless steel vessel has a 10-gage (0.135-inch) wall thickness, an overall length of approximately 198 inches, and an outside diameter of approximately 20 inches at the lid end. The lid is approximately 7 inches thick and is closed by 12, 1/2-inch diameter bolts and two butyl O-ring seals. There is a single penetration through the lid which is closed by a bolted port cover and two butyl O-ring seals. The basket is composed of a series of discs, tie rods, and support tubes, with five fuel compartment tubes arranged in a star-like configuration. The basket incorporates fixed borated aluminum neutron poison plates. Flux trap spacers are positioned axially between stacked fuel parts canisters, and the canisters and spacers are positioned within a stainless steel sleeve that forms the fuel compartment. Canisters containing fuel parts (called Oak Ridge Canisters) and canisters containing intact Peach Bottom fuel elements may be shipped together.

5. (a) (3) Drawings

The TN-FSV packaging is constructed and assembled in accordance with the following Transnuclear, Inc. Drawing Nos.:

1090-SAR-1, Rev. 3	1090-SAR-6, Rev. 3
1090-SAR-2, Rev. 3	1090-SAR-7, Rev. 3
1090-SAR-3, Rev. 3	1090-SAR-8, Rev. 3
1090-SAR-4, Rev. 3	1090-SAR-9, Rev. 3
1090-SAR-5, Rev. 4	1090-SAR-10, Rev. 2

The Oak Ridge Container and internals are constructed and assembled in accordance with the following Transnuclear, Inc. Drawing Nos.:

3044-70-1, Rev. 5	3044-70-6, Rev. 2
3044-70-2, Rev. 3	3044-70-7, Rev. 2
3044-70-3, Rev. 2	3044-70-8, Rev. 1
3044-70-4, Rev. 2	3044-70-9, Rev. 0
3044-70-5, Rev. 2	

The Oak Ridge Canister is constructed and assembled in accordance with the following Lockheed Martin Energy Systems, Inc. Drawing No.:

X3E020566A175, Rev. 0

(b) Contents

(1) Type and form of material

(i) <u>For Configuration 1:</u>

Irradiated HTGR fuel elements within a fuel storage container. Each fuel element consists of a graphite block containing fuel rods. The fuel is composed of thorium/uranium carbide and thorium carbide fuel particles within the fuel rods. The graphite block is hexagonal in cross section and is approximately 14.2 inches across the flats and 31.2 inches long. Each fuel element contains a maximum of 1.4 kg of uranium enriched to a maximum of 93.5 weight percent U-235 and approximately 11.3 kg of thorium. The maximum burnup is approximately 70,000 MWd/MTIHM, and the minimum cool time is 1600 days.

5(b) (1) Type and form of material (Continued)

(ii) <u>For Configuration 2</u>:

Irradiated, intact Peach Bottom Unit 1, Core 2, fuel elements within aluminum canisters with steel liners. Each fuel element consists of stacked graphite annular rings, or compacts, with an inner diameter of approximately 1.75 inches and an outer diameter of approximately 2.75 inches. The fuel is composed of coated thorium/uranium carbide particles within the graphite. The active fuel length is approximately 90 inches. The fuel element may include associated hardware such as top plug, reflector apparatus, grappling hook, etc. Each fuel element contains a maximum of 0.25 kg of uranium enriched to a maximum of 93.15 weight percent U-235 and approximately 1.5 kg of thorium prior to irradiation. The maximum burnup is approximately 73,000 MWd/MTIHM and the minimum cool time is 27 years.

(iii) <u>For Configuration 2</u>:

Irradiated fuel parts within Oak Ridge Canisters, as described in Item No. 5(a)(3), above. The minimum fuel cool time is 15 years. The maximum fissile mass prior to irradiation per Oak Ridge Canister is limited as shown below:

Canister Group	Maximum mass U-235 per canister (grams)	Maximum mass Pu-239 + Pu-241 per canister (grams)
1	475	0
2	865	191
3	200	415
4	275	160
5	910	0

5.(b) (2) Maximum quantity of material per package

Total weight of contents and packaging material within the TN-FSV cavity not to exceed 5,000 pounds. For Configuration 1 this includes fuel elements, fuel storage container, and depleted uranium shield plug. For Configuration 2 this includes fuel materials, Oak Ridge Container, basket, Oak Ridge Canisters, Peach Bottom fuel canisters, flux trap spacers, and other packaging materials.

(i) For the contents described in Item 5(b)(1)(i):

Six fuel elements, with decay heat not to exceed 60 watts per fuel element.

(ii) For the contents described in Item 5(b)(1)(ii) and 5(b)(1)(iii):

Total weight of fuel materials, canisters, and flux trap spacers within the Oak Ridge Container not to exceed 1,789 pounds. Decay heat not to exceed 120 watts per package. The maximum decay heat per Oak Ridge Canister is 35 watts, except that the maximum decay heat per Oak Ridge Canister in the position next to the lid is 7 watts. The maximum decay heat in any cross sectional region corresponding to the axial length of an Oak Ridge Canister is 55 watts, except that the maximum decay heat in the cross sectional region next to the lid is 35 watts.

Canisters containing intact Peach Bottom fuel elements and Oak Ridge Canisters containing irradiated fuel parts must be loaded into the Oak Ridge Container fuel compartments as follows:

Loading Pattern	One Fuel Compartment	Other Four Fuel Compartments
1	Four Group 2 Canisters	Four Group 1 Canisters
2	Four Group 5 Canisters	Four Group 1 Canisters
3	One Peach Bottom Element and One Group 4 Canister	One Peach Bottom Element and One Group 4 Canister
4	Two Group 3 Canisters and Two Group 4 Canisters	One Peach Bottom Element and One Group 4 Canister

Flux trap spacers, as shown in Transnuclear, Inc. Drawing No. 3044-70-3, must be positioned axially between any two Oak Ridge Canisters.

5. (c) Transport Index for Criticality Control (Criticality Safety Index)

Minimum transport index to be shown on label for nuclear criticality control: 100

6. The package must be leak tested as follows:

(a) For Configuration 1:

(1) In the 12-month period prior to shipment and after seal replacement, each containment seal must be tested to show a leak rate no greater than 1×10^{-3} ref-cm^3/sec. The leak test must have a sensitivity of at least 5×10^{-4} ref-cm^3/sec.

(2) Prior to each shipment, the package seals (main seal and vent seal) must be leak tested in accordance with Section 7.1.2 of the Safety Analysis Report. The acceptance criterion is a leak rate no greater than 1×10^{-3} ref-cm^3/sec. The test must have a sensitivity of at least 1×10^{-3} ref-cm^3/sec. The drain seal must also be tested if the drain port cover has been removed since the seal was last leak tested.

(b) For Configuration 2:

(1) In the 12-month period prior to shipment and after seal replacement, each containment seal of the outer cask and the Oak Ridge Container must be tested to show a leak rate no greater than 1×10^{-7} ref-cm^3/sec. The leak test must have a sensitivity of at least 5×10^{-8} ref-cm^3/sec.

(2) Prior to each shipment, the Oak Ridge Container containment seals (main seal and vent seal) and the outer cask containment seals (main seal and vent seal) must be leak tested in accordance with Section 7.1.2 of the Addendum. The seals must show no leakage greater than 1×10^{-7} ref-cm^3/sec or no leakage when tested to a sensitivity of at least 1×10^{-3} ref-cm^3/sec. The drain seal of the outer cask must also be tested if the drain port cover has been removed since the seal was last leak tested.

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9253	7	71-9253	USA/9253/B(U)F-85	7	OF	7

7. 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 of Chapter 7 of the Safety Analysis Report for Configuration 1, and Chapter 7 of the Addendum for Configuration 2.

(b) Each packaging must meet the acceptance tests and must be maintained in accordance with the Acceptance Tests and Maintenance Program of Chapter 8 of the Safety Analysis Report. In addition, for Configuration 2, each packaging must meet the acceptance tests and must be maintained in accordance with the Acceptance Tests and Maintenance Program of Chapter 8 of the Addendum.

(c) Prior to each shipment for Configuration 1 and Configuration 2, the cask main closure seal and vent seal must be inspected. The drain seal must be inspected if the drain port cover has been removed during preparation for shipment. All seals must be replaced within the 12-month period prior to shipment, or earlier if inspection shows any defect. In addition, prior to each shipment for Configuration 2, the Oak Ridge Container main closure seal and vent seal must be inspected. All seals must be replaced within the 12-month period prior to shipment, or earlier if inspection shows any defect.

The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR §71.12.

9. Expiration date: May 31, 2004.

REFERENCES

Public Service Company of Colorado application dated March 31, 1993; as supplemented February 24, June 2, and June 14, 1994; and September 11 and December 7, 1995.

U.S. Department of Energy supplements dated: March 24, 1997; March 24, 1999; June 15, September 18, and October 2, 2001.

Transnuclear, Inc. supplements dated September 19, 2001; and March 1, May 17, and June 14 and 21, 2002.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

John D. Monninger, Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: November 14, 2002

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE	PAGES
9255	8	71-9255	USA/9255/B(U)F-85	1 OF	

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.
Four Skyline Drive
Hawthorne, NY 10532

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION
Transnuclear West Inc., consolidated application dated December 13, 2000.

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.: NUHOMS® MP187 Multi-Purpose Cask

(2) Description:

The NUHOMS® MP187 Multi-Purpose Cask (package) consists of an outer cask, into which one of the four different dry shielded canisters (DSC) is placed. During shipment, energy-absorbing impact limiters are utilized for additional package protection.

Cask

The purpose of the cask is to provide containment and shielding of the radioactive materials contained within the DSC during shipment. The cask is constructed of stainless steel and lead with a neutron shield of cementitious material. The inside cavity of the cask is a nominal 68 inches in diameter and 187 inches long. The bottom access closure is approximately 5 inches thick and 17 inches in diameter, secured by 12 1-inch diameter bolts. The top closure is approximately 6.5 inches thick and is secured by 36 2-inch diameter bolts. Both closures are sealed by redundant O-rings.

Containment is provided by a stainless steel closure lid bolted to the stainless steel cask. The containment system of the NUHOMS® MP187 transportation cask consists of (a) the inner shell, (b) the bottom end closure plate, (c) the top closure plate, (d) the top closure inner O-ring seal, (e) the ram closure plate, (f) the ram closure inner O-ring seal, (g) the vent port screw, (h) the vent port O-ring seal, (i) the drain port screw, and (j) the drain port O-ring seal. No credit is given to the DSC as a containment boundary.

Shielding is provided by 4 inches of stainless steel, 4 inches of lead, and approximately 4.3 inches of neutron shielding. The overall length of the cask is approximately 200 inches; the outer diameter is approximately 93 inches. The maximum gross weight of the package, with impact limiters, is approximately 282,000 lbs. The total length of the package with the impact limiters attached is approximately 308 inches. Four removable trunnions (two upper and two lower) are provided for handling and lifting.

Dry Shielded Canisters (DSCs)

The purpose of the DSC, which is placed within the transport cask, is to permit the transfer of spent fuel assemblies, into or out of a storage module, a dry transfer facility, or a pool as a unit. The DSC also provides additional axial biological shielding during handling and transport. The DSC consists of a stainless steel shell and a basket assembly. The approximately 5/8-inch thick shell has an outside diameter of about 67 inches and an external length of about 186 inches. The DSC basket assembly provides criticality control and contains a storage position for each fuel assembly. The basket is composed of circular spacer discs machined from thick carbon steel plates. Axial support for the DSC basket is provided by four high strength steel support rod assemblies. Carbon steel components of each DSC basket assembly are electrolytically coated with a thin layer of nickel to inhibit corrosion.

On the bottom of each DSC is a grapple ring, which is used to transfer a DSC horizontally from the cask into and out of dry storage modules. Because of the nature of the fuel that is to be transported, four different types of DSCs are designed for the package. Variations in the DSC configurations are summarized below:

- Fuel-Only Dry Shielded Canister (FO-DSC)

  The FO-DSC has a cavity length of approximately 167 inches and has solid carbon steel shield plugs at each end. The FO-DSC is designed to contain up to 24 intact Babcock and Wilcox (B&W) pressurized water reactor (PWR) spent fuel assemblies. The FO-DSC basket assembly consists of 24 guide sleeve assemblies with integral borated neutron absorbing plates, 26 spacer discs, and 4 support rod assemblies.

- Fuel/Control Components Dry Shielded Canister (FC-DSC)

  The FC-DSC has an internal cavity length of approximately 173 inches to accommodate fuel with the B&W control components installed. To obtain the increased cavity length, the shield plugs are fabricated from a composite of lead and steel. The FC basket is similar to the FO-DSC except that the support rod assemblies and guide sleeves are approximately 6-inches longer. The FC-DSC is also designed to contain up to 24 intact B&W PWR spent fuel assemblies with control components.

- Failed Fuel Dry Shielded Canister (FF-DSC)

  The FF-DSC has an internal cavity length of approximately 173 inches to accommodate 13 damaged B&W PWR spent fuel assemblies. Because the cladding has been locally degraded, individual (screened) fuel cans are provided to confine any gross loose material, maintain the geometry for criticality control, and facilitate loading and unloading operations. The FF-DSC is similar to FC-DSC in most respects with the exception of the basket assembly. The FF-DSC basket may be fabricated from austenitic stainless steel.

• 24PT1 Dry Shielded Canister (24PT1-DSC)

The 24PT1-DSC has an internal cavity length of approximately 167 inches with a solid carbon steel shield plug at each end. The 24PT1-DSC will accommodate 22 to 24 Westinghouse (WE) 14 x14 PWR spent fuel assemblies, including control components. Control components authorized that are integral to WE 14x14 fuel assemblies include rod cluster control assemblies, thimble plug assemblies, and neutron source assemblies only. Fuel assemblies may be damaged or intact as described in 5.b(2)(a). The 24PT1-DSC basket assembly consists of 24 guide sleeve assemblies with integral borated neutron absorbing plates, 26 spacer discs, and 4 support rod assemblies. Up to four screened individual failed fuel cans are provided for storage of damaged fuel within the guide sleeve assemblies. These failed fuel cans are similar in configuration to the FF-DSC failed fuel cans.

Impact Limiters

The impact limiter shells are fabricated from stainless steel. Within that shell are closed-cell polyurethane foam and aluminum honeycomb material. The impact limiter is attached to the cask by carbon steel bolts. Each impact limiter is bolted to the cask body through the neutron shield top and bottom support rings. The weight of each impact limiter is approximately 15,800 lbs.

(3) Drawings

The package shall be constructed and assembled in accordance with the following Transnuclear West Drawing Numbers:

NUH-05-4000NP, Revision 8,
Sheets 1 through 2
MP187 Multi-Purpose Cask
General Arrangement

NUH-05-4001, Revision 14,
Sheets 1 through 6
MP187 Multi-Purpose Cask
Main Assembly

NUH-05-4002, Revision 4
Sheets 1 and 2
MP187 Multi-Purpose Cask
Impact Limiters

NH-05-4003, Revision 9,
Sheets 1 and 2
NUHOMS® MP187 Multi-Purpose Cask
On-Site Transfer Arrangement

NUH-05-4004, Revision 15,
Sheets 1 through 5
NUHOMS® FO-DSC & FC-DSC
PWR Fuel Main Assembly

NUH-05-4005, Revision 14,
Sheets 1 through 5
NUHOMS® FF-DSC
PWR Fuel Main Assembly

NUH-05-4006NP, Revision 6,
Sheets 1 and 2
NUHOMS® MP187 Multi-Purpose
Transportation Skid/Personnel Barrier

NUH-05-4010, Revision 2,
Sheets 1 through 6
NUHOMS® - 24PT1-DSC
Main Assembly

5.b Contents of Packaging

(1) Type and Form of Material:

(a) Intact fuel assemblies - Assemblies containing fuel rods with no known or suspected cladding defects greater than hairline cracks or pinhole leaks are authorized when contained in the FO-DSC, FC-DSC, or 24PT1-DSC.

(b) Damaged fuel assemblies - Assemblies containing fuel rods with known or suspected cladding defects greater than hairline cracks or pinhole leaks or with cracked, bulging, or discolored cladding are authorized when contained in a failed fuel can in the FF-DSC or the 24PT1-DSC. Spent fuel, with plutonium in excess of 20 curies per package, in the form of debris, particles, loose pellets, and fragmented rods or assemblies are not authorized. Damaged fuel assemblies may be shipped with or without control components.

(c) (i) The fuel authorized for shipment in the NUHOMS®-MP187 FO, FC, or FF DSC is B&W 15x15 uranium oxide PWR fuel assemblies with a maximum initial pellet enrichment of 3.43% by weight of U235, and a total uranium content not to exceed 466 Kg per assembly.

(ii) The fuel authorized for shipment in the NUHOMS®-MP187 24PT1-DSC is WE 14x14 stainless steel clad (SC) or zircaloy clad mixed oxide (MOX) PWR fuel assemblies as described in Table 2.

(d) Intact B&W 15x15 fuel assemblies without control components shall be shipped only in the FO-DSC. Intact B&W 15x15 fuel assemblies with control components shall be shipped only in the FC-DSC.

(e) Intact WE 14x14 fuel assemblies with or without control components shall be shipped only in the 24PT1-DSC. Control components authorized are integral to WE 14x14 fuel assemblies include rod cluster control assemblies, thimble plug assemblies, and neutron source assemblies only.

(f) (i) The maximum burn-up and minimum cooling times for the individual B&W 15x15 assemblies shall meet the requirements of Table 1. In addition, the fuel shall have been decayed for a time sufficient to meet the thermal criteria of 5.b(1)(g) and (h). The maximum total allowable cask heat load is 13.5 kW.

(ii) The maximum enrichment, burn-up and minimum cooling times for the individual WE 14x14 fuel assemblies shall meet the requirements of Table 2. In addition, the fuel shall have been decayed for a time sufficient to meet the thermal criteria of 5.b.(1)(g) and (h). The maximum total allowable cask heat load for the 24 PT1-DSC is per Table 2.

(g) (i) The maximum assembly decay heat (including control components when present) of B&W 15x15 individual fuel assembly is 0.764 kW, referred to as Type I, or 0.563 kW, referred to as Type II.

(ii) The maximum assembly decay heat (including control components when present) of WE 14x14 individual fuel assembly is per Table 2.

5.b Contents of Packaging:

(1) Type and Form of Material Continued:

(h) (i) Control components for B&W 15x15 fuel assemblies stored in the FO, FC and FF-DSCs shall be cooled for at least 8 years.

(ii) Control components for WE 14x14 fuel assemblies stored in the 24PT1-DSC shall be cooled for at least 10 years.

(2) Maximum quantity of material per package:

(a) (i) For material described in 5.b(1) to be stored in the FO, FC or FF-DSCs: 24 PWR intact fuel assemblies or 13 damaged fuel assemblies, with no more than 15 damaged fuel rods per assembly. Where a DSC is to be loaded with fewer fuel assemblies than the DSC capacity, dummy fuel assemblies with the same nominal weight as a standard fuel assembly shall be installed in the unoccupied spaces.

(ii) For material described in 5.b(1) to be stored in the 24PT1-DSC: 22 to 24 PWR fuel assemblies of which up to four may be damaged WE 14x14 SC fuel assemblies with the balance intact WE 14x14 SC or MOX fuel assemblies. No more than one damaged WE 14x14 MOX fuel assembly can be stored per 24PT1-DSC with the balance intact WE 14x14 SC fuel assemblies. The damaged fuel assemblies shall have no more than 14 damaged fuel rods per assembly and shall be stored in the four outer corner fuel assembly locations along the 45°, 135°, 225°, 315° azimuth of the 24PT1-DSC. A DSC may include two empty slots if they are located on symmetrically opposite locations with respect to the 0° - 180° and 90°-270° DSC axes. Any additional empty fuel slots shall be loaded with dummy fuel assemblies that displace the same or greater amount of volume and with the same nominal weight as a standard fuel assembly. Fuel spacers shall be located at the bottom and top of each fuel assembly to center the fuel assemblies within the DSC. Failed fuel cans require only bottom spacers since a top spacer is integral to each failed fuel can.

(b) For material described in 5.b(1): the approximate maximum payload (including control components when present) is 81,100 lbs.

Table 1- FO, FC and FF-DSC Fuel Assembly Burn-up vs. Cooling Time

Maximum Burn-up (MWD/MTIHM)*	Minimum Enrichment in the Active Fuel Region (w/o U-235)	Minimum Required Type I Cooling Time (years)	Minimum Required Type II Cooling Time (years)	Maximum Burn-up (MWD/MTIHM)*	Minimum Enrichment in the Active Fuel Region (w/o U-235)	Minimum Required Type I Cooling Time (years)	Minimum Required Type II Cooling Time (years)
<23,200	n/a	5	5	33,000	2.90	7	10
23,200	2.38	5	5	34,000	2.95	7	11
24,000	2.43	5	6	35,000	2.67	7	14
25,000	2.49	5	6	35,000	2.99	7	11
26,000	2.55	5	7	36,000	3.03	8	13
27,000	2.61	5	7	37,000	3.00	8	14
28,000	2.66	5	8	37,000	3.07	8	14
29,000	2.00	6	10	38,000	3.11	9	15
29,000	2.71	5	8	39,000	3.15	9	16
30,000	2.76	5	8	40,000	3.19	9	17
31,000	2.81	6	9				
32,000	2.86	6	10	* Megawatt Days per Metric Ton of Initial Heavy Metal			

Table 2 - 24PT1-DSC Fuel Assembly Burnup vs. Cooling Time

Fuel Type	Maximum Enrichment (Weight %)	Minimum Enrichment (Weight %)	Maximum Burnup (MWD/ MTU)	Minimum Cooling Time / Max Heat Load Per Cask / Max Assembly Heat Load (Incl. Control Components[1])
WE 14x14 Stainless Steel Clad (SC) (May include Integral Fuel Burnable Absorber, boron coated fuel pellets)	4.05 ^{235}U	3.76 ^{235}U	45,000	38 years/14 kW/ 0.583 kW
		3.36 ^{235}U	40,000	
		3.12 ^{235}U	35,000	
WE 14x14 MOX	0.71 ^{235}U 2.84 fissile Pu (64 rods) 3.10 fissile Pu (92 rods) 3.31 fissile Pu (24 rods)	2.78 fissile Pu (64 rods) 3.05 fissile Pu (92 rods) 3.25 fissile Pu (24 rods)	25,000	30 years/13.706 kW/ 0.294 kW

Notes:

1 Control component cooling time must be a minimum of 10 years.

5.c Transport Index for Criticality Control

Minimum transport index to be shown on the label for nuclear criticality control: "0"

6. Type I fuel assemblies shall be loaded only into the four innermost cells of a DSC, while Type II assemblies may be loaded into any cell when using the FO-DSC or the FC-DSC. The FF-DSC has no Type I or II placement restrictions. The 24PT1-DSC has restrictions on the location of damaged fuel assemblies per Section 5.b.(2).

7. For operating controls and procedures, in addition to the requirements of Subpart G of 10 CFR Part 71:

   a Each package shall be both prepared for shipment and operated in accordance with the Operating Procedures in Chapter 7 of the application, as supplemented.

   b. All fabrication acceptance tests and maintenance shall be performed in accordance with the Acceptance Tests and Maintenance Program in Chapter 8 , as supplemented. In addition, this shall include:

      (1) With the exception of the weld between the inner shell and top forging, all longitudinal and circumferential inner shell welds, which form the containment boundary of the cask, shall be radiographically inspected (RT) with acceptance standards in accordance with the ASME Code, Section III, Division 1, NB-5320. The weld between the inner shell and top forging shall be verified by RT or ultrasonically inspected (UT). The substitution of UT for the examination of the completed weld may be made provided the examination is performed using detailed written procedures, proven by actual demonstration to the satisfaction of the inspector as capable of detecting and locating defects described in ASME Code, Section III, Division 1 Subsection NB

      (2) Verification of the DSC outer top cover plate weld by either volumetric or multilayer PT examination. If PT is used, at a minimum, it must include the root, each successive 1/4 inch weld thickness, and the final layer. The inspection of the weld must be performed by qualified personnel and shall meet the acceptance requirements of ASME B&PVC Section III, NB-5350. The inspection process, including findings (indications) shall be made a permanent part of the licensee's records by video, photographic, or other means providing an equivalent retrievable record of weld integrity.

      (3) The minimum lead thickness in the main cask body, away from the trunnions and the top and bottom forgings, shall be 3.90 inches.

      (4) The neutron shield shall have a minimum thickness of 4.31 inches.

8. This package is approved for exclusive use rail, truck or marine transport.

9. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.12.

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9255	8	71-9255	USA/9255/B(U)F-85	8	OF	8

10. Expiration Date: September 10, 2003.

REFERENCES

Transnuclear West Inc., consolidated Safety Analysis Report for the NUHOMS® MP187 Multi-Purpose Cask, dated December 13, 2000.

Transnuclear West Inc., letters dated January 30, 2001, August 24, 2001, September 21, 2001, and October 4, 2001.

Transnuclear, Inc., letters dated October 3, 2001, November 29, 2001, April 16, 2002, June 10, 2002, and July 23, 2002.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

M. Wayne Hodges /for

E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: October 18 ,2002

NRC FORM 618
(3-96)
10 CFR 71

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIALS PACKAGES

U.S. NUCLEAR REGULATORY COMMISSION

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. PACKAGE IDENTIFICATION NUMBER	d. PAGE NUMBER	e. TOTAL NUMBER PAGES
9258	0	USA/9258/B(U)-85	1	3

2. PREAMBLE

a. This certificate is issued to certify that the 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)*

MDS Nordion
447 March Road
Kanata, Ontario, Canada, K2K 1X8

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION:

MDS Nordion application dated June 30, 1998, as supplemented.

c. DOCKET NUMBER 71-9258

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.: F-294

(2) Description

A steel encased, lead shielded shipping cask for special form sources. The package consists of a cylindrical cask body with cooling fins, a closure plug, a cylindrical external fireshield, a top crush shield, a permanent skid, and a removable shipping skid. The special form sources are positioned by a source carrier within the cask cavity.

The cask body is constructed of a 1/2-inch thick inner stainless steel shell, and a 1/2-inch thick outer stainless steel shell. The annulus between the inner and outer shells is filled with lead, approximately 11 1/4 inches thick. The cask is closed by a 2 1/2 inch thick stainless steel closure lid and 16 one-inch diameter bolts. A lead radiation protection plug is fitted to the cask closure plate. Stainless steel fins are welded onto the exterior of the cask to dissipate heat. The cask is surrounded by a cylindrical fireshield which is constructed of ceramic fiber thermal insulation encased in mild steel shells. A composite assembly consisting of a finned crush shield that acts as an impact limiter and a fireshield is bolted to the top end of the cask. The cask is equipped with a fixed skid and a shipping skid composed of steel beams. The fixed skid includes a sheet of thermal insulation enclosed in steel.

Page 2 - Certificate No. 9258 - Revision No. 0 - Docket No. 71-9258

5(a)(2) cont. The approximate dimensions and weights of the package are as follows:

Cask body outer diameter (excluding cooling fins)	36 inches
Cask body height	52 1/4 inches
Cask cavity inside diameter	11 1/2 inches
Cask cavity inside height	19 3/4 inches
Lead shield thickness	11 1/4 inches
Fire shield outer diameter	47 inches
Overall package dimensions (including shipping skid)	
width	78 inches
length	78 inches
height	80 1/2 inches
Maximum contents weight	20 pounds
Maximum package weight (including contents)	21,000 pounds

(3) Drawings

The packaging is constructed in accordance with MDS Nordion drawing Nos.:

F629401-001, Sheets 1-5, Rev. D, and
F631301-001, Rev. B.

(b) Contents

(1) Type and form of material

Cobalt-60 as sealed sources which meet the requirements of special form radioactive material.

(2) Maximum quantity of material per package

360,000 Curies

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

(a) The package must meet the Acceptance Tests and Maintenance Program of Chapter 8.0 of the application.

(b) The package shall be prepared for shipment and operated in accordance with the Operating Procedures in Chapter 7.0 of the application.

Page 3 - Certificate No. 9258 - Revision No. 0 - Docket No. 71-9258

7. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR §71.12.

8. Expiration date: December 31, 2003.

REFERENCES

MDS Nordion application dated June 30, 1998.

Supplement dated: December 11, 1998.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Cass R. Chappell

Cass R. Chappell, Chief
Package Certification Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: 1/6/99

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIALS PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. PACKAGE IDENTIFICATION NUMBER	d. PAGE NUMBER	e. TOTAL NUMBER PAGES
9261	1	USA/9261/B(U)F-85	1	7

2. PREAMBLE

a. This certificate is issued to certify that the 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)*

Holtec International
Holtec Center
555 Lincoln Drive West
Marlton, NJ 08053

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION:

Holtec International Report No. HI-951251, *Safety Analysis Report for the Holtec International Storage, Transport, And Repository Cask System (HI-STAR 100 Cask System),* Revision 9, dated April 20, 2000.

c. DOCKET NUMBER 71-9261

4. CONDITIONS

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

5.

5 a. Packaging

(1) Model No.: HI-STAR 100 System

(2) Description

The HI-STAR 100 System is a canister system comprising a Multi-Purpose Canister (MPC) inside of an overpack designed for both storage and transportation (with impact limiters) of irradiated nuclear fuel. The HI-STAR 100 System consists of interchangeable MPCs which house the spent nuclear fuel and an overpack which provides the containment boundary, helium retention boundary, gamma and neutron radiation shielding, and heat rejection capability. The outer diameter of the overpack of the HI-STAR 100 is approximately 203 1/8 inches without impact limiters and approximately 305 7/8 inches with impact limiters. Maximum gross weight for transportation (including overpack, MPC, fuel, and impact limiters) is approximately 282,000 pounds. Specific tolerances are called out in drawings listed below.

Multi-Purpose Canister

There are three Multi-Purpose Canister (MPC) models, designated the MPC-24, MPC-68, and MPC-68F. All MPCs are designed to have identical exterior dimensions. A single overpack design is provided which is capable of containing each type of MPC. The two digits after the MPC designate the number of reactor fuel assemblies for which the respective MPCs are designed. The MPC-24 is designed to contain up to 24 Pressurized Water Reactor (PWR) fuel assemblies and the MPC-68 and MPC-68F are designed to contain up to 68 Boiling Water Reactor (BWR) fuel assemblies. Any MPC-68 loaded with material classified as fuel debris is designated as MPC-68F.

The HI-STAR 100 MPC is a welded cylindrical structure with flat ends. Each MPC is an assembly consisting of a honeycombed fuel basket, baseplate, canister shell, lid, and closure ring. The outer diameter and cylindrical height of each MPC is fixed. However, the number of spent fuel storage locations in each of the MPCs depends on the fuel assembly characteristics. For the HI-STAR 100 System transporting fuel debris in a MPC-68F, the MPC provides the second inner container, in accordance with 10 CFR 71.63. The MPC pressure boundary is a strength-welded enclosure constructed entirely of a stainless steel alloy.

5. a. (2) Description (continued)

Overpack

The HI-STAR 100 overpack is a multi-layer steel cylinder with a welded baseplate and bolted lid (closure plate). The inner shell of the overpack forms an internal cylindrical cavity for housing the MPC. The outer surface of the overpack inner shell is buttressed with intermediate steel shells for radiation shielding. The overpack closure plate incorporates a dual O-ring design to ensure its containment function. The containment system consists of the overpack inner shell, bottom plate, top flange, top closure plate, top closure inner O-ring seal, vent port plug and seal, and drain port plug and seal.

Impact Limiters

The HI-STAR 100 overpack is fitted with two impact limiters fabricated of aluminum honeycomb completely enclosed by an all-welded austenitic stainless steel skin. The two impact limiters are attached to the overpack with 20 and 16 bolts at the top and bottom, respectively.

(3) Drawings

The package shall be constructed and assembled in accordance with the following drawings or figures in Holtec International Report No. HI-951251, *Safety Analysis Report for the Holtec International Storage, Transport, And Repository Cask System (HI-STAR 100 Cask System)*, Revision 9:

(a) HI-STAR 100 MPC-24	Drawing C1395, Sheets 1-4, Rev. 1 Drawing C1396, Sheets 1-4, 6, Rev. 1; and Sheet 5, Rev. 0 Drawing BM-C1478, Sheets 1& 2, Rev. 1
(b) HI-STAR 100 MPC-68 and MPC-68F	Drawing C1401, Sheets 1-4, Rev. 1 Drawing C1402, Sheets 1-4, 6, Rev. 1; and Sheet 5, Rev. 0 Drawing BM-C1479, Sheets 1& 2, Rev. 1
(c) HI-STAR 100 Overpack	Drawing C1397, Sheet 1, Rev. 2; and Sheets 2-7, Rev. 1 Drawing C1398, Sheets 1-3, Rev. 1 Drawing C1399, Sheets 1-2, Rev. 1; and Sheet 3, Rev. 2 Drawing BM-C1476, Sheet 1, Rev. 1; and Sheet 2, Rev. 2
(d) HI-STAR 100 Impact Limiters	Drawing C1765, Sheets 1-6, Rev. 1; and Sheet 7, Rev. 0
(e) HI-STAR 100 Assembly for Transport	Drawing C1782, Rev. 1

5. b. Contents

(1) Type and Form, and Quantity of Material

(a) Fuel assemblies meeting the specifications and quantities provided in Appendix A to this Certificate of Compliance and meeting the requirements provided in Conditions 5.b(1)(b) through 5.b(1)(g) below are authorized for transportation.

(b) The following definitions apply:

Damaged Fuel Assemblies are fuel assemblies with known or suspected cladding defects, as determined by review of records, greater than pinhole leaks or hairline cracks, missing fuel rods that are not replaced with dummy fuel rods, or those that cannot be handled by normal means. Fuel assemblies which cannot be handled by normal means due to fuel cladding damage are considered fuel debris.

Damaged Fuel Containers (DFCs) are specially designed fuel containers for damaged fuel assemblies or fuel debris which permit gaseous and liquid media to escape while minimizing dispersal of gross particulates. The DFC designs authorized for use in the HI-STAR 100 are shown in Figures 1.2.10 and 1.2.11 of Holtec International Report No. HI-951251, Rev. 9.

Fuel Debris is ruptured fuel rods, severed rods, loose fuel pellets, and fuel assemblies with known or suspected defects which cannot be handled by normal means due to fuel cladding damage.

Incore Grid Spacers are fuel assembly grid spacers located within the active fuel region (i.e., not including top and bottom spacers).

Intact Fuel Assemblies are fuel assemblies without known or suspected cladding defects greater than pinhole leaks or hairline cracks and which can be handled by normal means. Partial fuel assemblies, that is fuel assemblies from which fuel rods are missing, shall not be classified as intact fuel assemblies unless dummy fuel rods are used to displace an amount of water greater than or equal to that displaced by the original fuel rod(s).

Minimum Enrichment is the minimum assembly average enrichment. Natural uranium blankets are not considered in determining minimum enrichment.

Planar-Average Initial Enrichment is the average of the distributed fuel rod initial enrichments within a given axial plane of the assembly lattice.

(c) For MPCs partially loaded with stainless steel clad fuel assemblies, all remaining fuel assemblies in the MPC shall meet the more restrictive of the two limits for the stainless steel clad fuel assemblies or the applicable Zircaloy clad fuel assemblies.

(d) For MPCs partially loaded with damaged fuel assemblies or fuel debris, all remaining Zircaloy clad intact fuel assemblies in the MPC shall meet the more restrictive of the two limits for the damaged fuel assemblies or the intact fuel assemblies.

Page 4 - Certificate No. 9261 - Revision No. 1 - Docket No. 71-9261

5.. b. (1) Type and Form, and Quantity of Material (continued)

(e) For MPC-68s partially loaded with array/class 6x6A, 6x6B, 6x6C, or 8x8A fuel assemblies, all remaining Zircaloy clad intact fuel assemblies in the MPC shall meet the more restrictive of the two limits for the 6x6A, 6x6B, 6x6C, and 8x8A fuel assemblies or the applicable Zircaloy clad fuel assemblies.

(f) PWR control rods, burnable poison rod assemblies, thimble plugs, and other non-fuel hardware are not authorized for transportation.

(g) BWR stainless-steel channels and control blades are not authorized for transportation.

c. Transport Index for Criticality Control

The minimum transport index to be shown on the label for nuclear criticality control: 0

6. For operating controls and procedures, in addition to the requirements of Subpart G of 10 CFR Part 71:

a. Each package shall be both prepared for shipment and operated in accordance with detailed written operating procedures. Procedures for both preparation and operation shall be developed. At a minimum, those procedures shall include the following provisions:

(1) Identification of the fuel to be loaded and independent verification that the fuel meets the specifications of Condition 5.b above.

(2) Before each shipment, the licensee or shipper shall verify and document that each of the requirements of 10 CFR 71.87 has been satisfied.

(3) The package must satisfy the following leak testing requirements:

(a) All overpack containment boundary seals shall be leak tested to show a leak rate of not greater than 4.3×10^{-6} atm cm^3/sec (helium). The leak test shall have a minimum sensitivity of 2.15×10^{-6} atm cm^3/sec (helium) and shall be performed:

(i) before the first shipment;
(ii) within the 12-month period prior to each successive shipment;
(iii) after detensioning one or more overpack lid bolts or the vent port plug; and
(iv) after each seal replacement.

(b) Before each shipment, all containment boundary seals shall be leak tested using a test with a minimum sensitivity of 1×10^{-3} atm cm^3/sec. If leakage is detected on a seal, then the seal must be replaced and leak tested per Condition 6.a(3)(a) above.

(c) Each containment boundary seal must be replaced after each use of the seal.

(4) The rupture discs on the neutron shield vessel shall be replaced every 5 years.

6. a. (continued)

(5) All MPCs shall be leak tested at the time of closure to show a leak rate of no greater than 5×10^{-6} atm cm^3/sec (helium).

(6) Water and residual moisture shall be removed from the MPC in accordance with the following specifications:

(a) The MPC shall be evacuated to a pressure of less than or equal to 3 torr.
(b) The MPC cavity shall hold a stable pressure of less than or equal to 3 torr for at least 30 minutes.

(7) Following vacuum-drying, the MPC shall be backfilled with 99.995% minimum purity helium: $\geq$ 1 atm and $\leq$ 28.3 psig for the MPC-24, and $\geq$ 1 atm and $\leq$ 28.5 psig for the MPC-68 and MPC-68F.

(8) Water and residual moisture shall be removed from the HI-STAR 100 overpack in accordance with the following specifications:

(a) The MPC shall be evacuated to a pressure of less than or equal to 3 torr.

(b) The overpack cavity shall hold a stable pressure of less than or equal to 3 torr for at least 30 minutes.

(9) Following vacuum drying, the overpack shall be backfilled with helium to $\geq$ 10 psig and $\leq$ 14 psig.

(10) The following fasteners shall be tightened to the torque values specified below:

Fastener	Torque (ft-lbs)
Overpack Closure Plate Bolts	2895 $\pm$ 90
Overpack Vent and Drain Port Plugs	45 +5/-0
Top Impact Limiter Attachment Bolts	256 +10/-0
Bottom Impact Limiter Attachment Bolts	1500 +45/-0
Tie-down Bolts	250 +20/-0
Transport Frame Bolts	250 +20/-0

(11) Verify that the appropriate fuel spacers, as necessary, are used to position the fuel in the MPC cavity.

b. All acceptance tests and maintenance shall be performed in accordance with detailed written procedures. Procedures for fabrication, acceptance testing, and maintenance shall be developed and shall include the following provisions:

(1) The overpack lifting trunnions shall be tested at 300% of the maximum design lifting load.

(2) The MPC shall be pressure tested to 125% of the design pressure. The minimum test pressure shall be 125 psig.

6. b. (continued)

(3) The overpack shall be pressure tested to 150% of the Maximum Normal Operating Pressure (MNOP). The minimum test pressure shall be 150 psig.

(4) The MPC lid-to-shell (LTS) weld shall be verified by either volumetric examination using the ultrasonic (UT) method or multi-layer liquid penetrant (PT) examination. The root and final weld layers shall be PT examined in either case. If PT alone is used, additional intermediate PT examination(s) shall be conducted after each approximately 3/8 inch of the weld is completed. The inspection of the weld must be performed by qualified personnel and shall meet the acceptance requirements of ASME B&PV Section III, NB-5350. The inspection process, including findings (indications) shall be made a permanent part of the licensee's records by video, photographic, or other means providing an equivalent retrievable record of weld integrity.

(5) The radial neutron shield shall have a minimum thickness of 4.3 inches and the impact limiter neutron shields shall have a minimum thickness of 2.5 inches. Before first use, the neutron shielding integrity shall be confirmed through a combination of fabrication process control and radiation measurements with either loaded contents or a check source. Measurements shall be performed over the entire exterior surface of the radial neutron shield and each impact limiter using, at a maximum, a 6 x 6 inch test grid.

(6) Periodic verification of the neutron shield integrity shall be performed within 5 years of each shipment. The periodic verification shall be performed by radiation measurements with either loaded contents or a check source. Measurements shall be performed at a minimum of 12 locations on the radial neutron shield and at a minimum of 4 locations on each impact limiter

(7) The first fabricated HI-STAR 100 overpack shall be tested to confirm its heat transfer capability. The test shall be conducted after the radial channels, enclosure shell panels, and neutron shield material have been installed and all inside and outside surfaces are painted per the Design Drawings specified in Section 1.4 of the SAR, Rev. 9. A test cover plate shall be used to seal the overpack cavity. Testing shall be performed in accordance with written and approved procedures. The test must demonstrate that the overpack is fabricated adequately to meet the design heat transfer capability.

(8) For each package, a periodic thermal performance test shall be performed every 5 years or prior to next use, if the package has not been used for transport for greater than 5 years, to demonstrate that the thermal capabilities of the cask remain within its design basis.

(9) The neutron absorber's minimum acceptable ^{10}B loading is 0.0267 g/cm^2 for the MPC-24 and 0.0372 g/cm^2 for the MPC-68, and 0.01 g/cm^2 for the MPC-68F. The ^{10}B loading shall be verified by chemistry or neutron attenuation techniques.

(10) The minimum flux trap size for the MPC-24 is 1.09 inches.

(11) The minimum fuel cell pitch for the MPC-68 and MPC-68F is 6.43 inches.

(12) The package containment verification leak test shall be per ANSI 14.5.

Page 7 - Certificate No. 9261 - Revision No. 1 - Docket No. 71-9261

7. The maximum gross weight of the package as presented for shipment shall not exceed 282,000 pounds.

8. The package shall be located on the transport vehicle such that the bottom surface of the bottom impact limiter is at least 6 feet (along the axis of the overpack) from the edge of the vehicle.

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

10. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.12.

11. Expiration Date: March 31, 2004

Attachment: Appendix A

REFERENCES:

Holtec International Report No. HI-951251, *Safety Analysis Report for the Holtec International Storage, Transport, And Repository Cask System (HI-STAR 100 Cask System)*, Revision 9, dated April 20, 2000.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

M. Wayne Hodges
for

E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: May 11, 2000

APPENDIX A

CERTIFICATE OF COMPLIANCE NO. 9261, REVISION 1

MODEL NO. HI-STAR 100 SYSTEM