NUREG-0383
Volume 2
Revision 26

Directory of Certificates of Compliance for Radioactive Materials Packages

Certificates of Compliance

U.S. Nuclear Regulatory Commission
Office of Nuclear Material Safety and Safeguards
Washington, DC 20555-0001

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NUREG-0383
Volume 2
Revision 26

Directory of Certificates of Compliance for Radioactive Materials Packages

Certificates of Compliance

Manuscript Completed: November 2006
Date Published: December 2006

Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety and Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

ABSTRACT

The purpose of this directory is to make available a convenient source of information on packaging approved by the U.S. Nuclear Regulatory Commission. To assist in identifying packaging, an index by Model Number and corresponding Certificate of Compliance Number is included at the front of Volumes 1 and 2. An alphabetical listing by user name is included in the back of Volume 3 of approved Quality Assurance programs. The reports include a listing of all users of each package design and approved Quality Assurance programs prior to the publication date of the directory as of October 31, 2006. Volume 1 is for internal use only.

U.S. NUCLEAR REGULATORY COMMISSION
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10/31/2006

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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
0361	8	71-0361	USA/0361/B(U)F-96	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. Nuclear Regulatory Commission
   Washington, D.C. 20555

   b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION
   NUREG-0361; Safety Analysis Report for the Plutonium Air Transportable Package Model No. PAT-1, 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-1

(2) Description

A stainless steel containment vessel (designated TB-1) surrounded by a stainless steel and redwood overpack (designated AQ-1). The contents are sealed within a stainless steel product can (designated PC-1) inside the containment vessel.

The AQ-1 overpack is a right circular cylinder, approximately 42-1/2 inches long by 24-1/2 inches outside diameter. The walls of the overpack consist of approximately 8 inches of grain oriented redwood encased within double stainless steel drums. The ends of the drums are doubly closed. A copper heat conducting element and an aluminum load distributer are encased within the redwood.

The TB-1 containment vessel is approximately 8-1/2 inches outside length by 6-3/4 inches outside diameter. The minimum wall thickness of the vessel is approximately 1/2 inch. The interior cavity of the vessel is a right circular cylinder, 4-1/4 inches diameter, with hemispherical ends. The vessel is closed by 12, 1/2-inch diameter bolts and doubly sealed with a copper gasket and knife edges and an elastomer O-ring.

The weight of the package is approximately 500 pounds. The weight of the TB-1 containment vessel, when loaded with 4.4 pounds of contents is approximately 41.7 pounds.

5. (a) (3) Drawings and Specifications

The Model No. PAT-1 packaging is fabricated in accordance with the drawings and specifications in Section 9.0 of the Safety Analysis Report, NUREG-0361 as supplemented by Issue B of Drawing Nos. 1004, 1009, 1013, 1016, 1017, 1018, 1019, 1020 and 1022.

(b) Contents

(1) Type and form of material

Plutonium oxide and its daughter products, in any solid form. The plutonium oxide may be mixed with uranium oxide and its daughter products, in any solid form.

(2) Maximum quantity of material per package and additional permissible contents

(i) Maximum 2.0 kg total radioactive material, plus: maximum 16 grams of water and 10 grams of polyethylene or polyvinylchloride bagging material. The maximum decay heat load of the contents may not exceed 25 watts.

(ii) Maximum 200 grams total radioactive material, plus: maximum one gram of water, maximum 200 grams of metal canning material (in addition to the PC-1 product can, Drawing No. 1024), maximum 64 grams of aluminum foil or honeycomb (in addition to the top spacer, Drawing No. 1015), maximum 175 grams of glass and maximum 35 grams polyethylene or polyvinylchloride bagging material. The maximum decay heat load of the contents may not exceed 25 watts.

(c) Criticality Safety Index

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

6. The PC-1 product can (Drawing No. 1024) and the top spacer (Drawing No. 1015) need not be used when the contents include 20 curies or less of plutonium.

7. Prior to first use, each packaging shall meet the acceptance tests and standards specified in Subsection 8.1 and Section 9.0 of the Safety Analysis Report.

8. Prior to each shipment, the package shall meet the tests and criteria specified in Subsection 8.2 of the Safety Analysis Report.

9. The package shall be prepared for shipment and operated in accordance with the procedures specified in Section 7.0 of the Safety Analysis Report.

The systems and components of each packaging shall meet the periodic tests and criteria specified in Subsection 8.3 of the Safety Analysis Report.

11. Repair and maintenance of the packaging shall be in accordance with Sections 8.0 and 9.0 of the Safety Analysis Report.

12. The packaging shall be designed, procured, fabricated, accepted, operated, maintained, and repaired in accordance with a quality assurance plan approved by the Nuclear Regulatory Commission for this purpose.

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 cargo may be stowed aft of the package(s).

    (b) The package(s) must be securely cradled and tied-down to the main deck of the aircraft. The tie-down system must be capable of providing package restraint against the following inertia forces acting separately relative to the deck of the aircraft: Upward, 2g; Forward, 9g; Sideward, 1.5g; Downward, 4.5g.

    (c) Cargo which bears one of the following hazardous material labels may not be transported aboard an aircraft carrying a package(s):

Explosive A	Non-Flammable Gas
Explosive B	Flammable Liquid
Explosive C	Flammable Solid
Spontaneously Combustible	Flammable Gas
Dangerous When Wet	Oxidizer
Organic Peroxide	Corrosive

This restriction does not apply to hazardous material cargo labeled solely as:

Radioactive I	Poison
Radioactive II	Poison Gas
Radioactive III	Irritant
Magnetized Materials	Etiologic Agent

14. Packagings may be marked with Package Identification Number USA/0361/B(U)F-85 until October 1, 2005, and must be marked with Package Identification Number USA/0361/B(U)F-96 after October 1, 2005.

15. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.12, until October 1, 2004, and under the provisions of 10 CFR 71.17 thereafter.

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

17. Expiration date: March 31, 2009.

REFERENCES

Safety Analysis Report for the Plutonium Air Transportable Package Model Number PAT-1, NUREG-0361, June 1978.

Sandia Laboratories application dated February 20, I980.
Supplements dated: July 27, 1990 and July 20, 1993.

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: March 24, 2004

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
4888	12	71-4888	USA/4888/B( )	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 the Air Force
Air Force Technical Application Center/CC
1030 S. HWY A1A
Patrick AFB, FL 32925-3002

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Teledyne Energy Systems applications dated April 26, 1985 and August 19, 1986, 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.: Sentinel-25A, LCG-25A; Sentinel-25B, LCG-25B
Sentinel-25C, LCG-25C; Sentinel-25C3, -25D, -25E, -25F

(2) Description

The packages are thermoelectric generators. The major components include the main housing, tungsten shield, housing flange, and electrical connectors. The approximate dimensions and weights for the various Model Nos. are as follows:

Model No.	Dimensions (inches)	Weight (lbs.)
Sentinel-25A, LCG-25A	25 OD x 25	3000
Sentinel-25B, LCG-25B	25 OD x 25	3300
Sentinel-25C, LCG-25C	24 OD x 32	2000
Sentinel-25C3	24 OD x 32	1300
Sentinel-25D	25 OD x 27	3300
Sentinel-25E	25 OD x 34	4200
Sentinel-25F	25 OD x 32	1400

5. (a) Packaging (continued)

(3) Drawings

The packagings are constructed in accordance with the following Drawing Nos.:

Model No.	Drawing Nos.
All Models	Isotopes, Inc. Drawing Nos.: 001-20000, Rev. E 001-20001, Rev. F 001-20002, Rev. F 001-20003, Sht. 1, Rev. B 001-80003
Sentinel-25A, LCG-25A	Martin Company Drawing Nos.: N0013100, Rev. A N0013108, Rev. D 001-40000, Rev. A Isotopes, Inc. Drawing Nos.: 001-10000, Rev. B 001-70024, Rev. C 001-70025, Sht. 1, Rev. D 001-70033, Shts. 1 & 2, Rev. A 001-70036 001-80005
Sentinel-25B, LCG-25B	Martin Company Drawing Nos.: N0013200, Rev. C 001-40012 Isotopes, Inc. Drawing Nos.: 001-70024, Rev. C 001-70025, Sht. 1, Rev. D 001-70033, Shts. 1 & 2, Rev. A 001-70036 001-80005
Sentinel-25C, LCG-25C	Martin Company Drawing Nos.: 001-40004, Rev. A 001-70010 001-70012, Rev. B 001-80004 Isotopes, Inc. Drawing Nos.: 001C10000, Sht. 1 Rev. D, & Sht. 3 001-70009, Rev. D

Model No. (continued)	Drawing Nos. (continued)
Sentinel-25C3	Isotopes, Inc. Drawing Nos.: 001C10000 Shts. 1 & 2, Rev. D 001-70009, Rev. D 001-70057, Rev. D 001-70060, Rev. C 001-40019, Rev. B
Sentinel-25D	Martin Company Drawing No. 001-80004 Isotopes, Inc. Drawing Nos.: 001D10000 Shts. 1 & 2, Rev. C 001-70036 001-70033 Shts. 1 & 2, Rev. A 001-70025 Sht. 1, Rev. D 001-70024, Rev. C 001-40015, Rev. C 001-40006, Rev. B
Sentinel-25E	Isotopes, Inc. Drawing Nos.: 001E10000, Shts. 1 & 2, Rev. E, & Sht. 3 001-70039, Rev. C 001-70025, Sht. 1, Rev. D & Sht. 2 001-70024, Rev. C 001-40017, Shts. 1 & 2, Rev. D 001-40006, Rev. B
Sentinel-25F	Isotopes, Inc. Drawing Nos.: 001F10000, Shts. 1 & 2, Rev. H* 001-70070, Rev. C 001-70060, Rev. C 001-70009, Rev. D 001-40025, Rev. A

*As modified by Figure 1 of the April 26, 1985, application.

(b) Contents

(1) Type and form of material

(i) Strontium 90 titanate doubly encapsulated in a Hastelloy or Uniloy fuel capsule which meet the requirements of special form radioactive material; or

(ii) Model No. Sentinel-25F may have, strontium fluoride doubly encapsulated in Hastelloy or Uniloy fuel capsule, with a Hastelloy C-276 liner which meets the requirements of special form radioactive material.

(2) The maximum quantity of material per package

125,000 curies

6. A barrier (permitting the free circulation of air) must be provided with sufficient separation distance to ensure that the requirement of 10 CFR 71.43 (g) will be met.

7. Eye-bolts shall be removed or covered during transportation to prevent their use as tie-down devices of packages.

8. In addition to the requirements of Subpart G of 10 CFR Part 71, each package shall be operated, prepared for shipment and maintained in accordance with the following Operating Procedures and Maintenance Programs:

Model No.	Operating Procedures	Maintenance Program
Sentinel-25A, LCG-25A	Appendix E of TES-3206, as revised	Appendix F of TES-3206, as revised
Sentinel-25B, LCG-25B	Appendix E of TES-3209, as revised	Appendix F of TES-3209, as revised
Sentinel-25C, LCG-25C	Appendix E of TES-3210, as revised	Appendix F of TES-3210, as revised
Sentinel-25C3	Appendix E of TES-3211, as revised	Appendix F of TES-3211, as revised
Sentinel-25D	Appendix E of TES-3212, as revised	Appendix F of TES-3212, as revised
Sentinel-25E	Appendix E of TES-3213, as revised	Appendix F of TES-3213, as revised
Sentinel-25F	Chapter VIII of TES-3202, as revised	Chapter IX of TES-3202, as revised

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: January 31, 2007

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

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REFERENCES

Teledyne Energy Systems applications dated April 26, 1985; and August 19, 1986.

Teledyne supplements dated: November 3, 1986; September 17 and December 2, 1991.

Department of the Air Force supplement dated: November 12, 1993; December 11, 1996; January 15, 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

e: January 29, 2002

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

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2. PREAMBLE

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

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

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

   a. ISSUED TO *(Name and Address)*

   Global Nuclear Fuel - Americas, L.L.C.
   P.O. Box 780
   Wilmington, NC 28402

   b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

   General Electric Company application dated September 10, 1997, 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.

   Packaging

   (1) Model No.: RA-3

   (2) Description

   A fuel assembly and fuel rod shipping container. Packagings are right rectangular boxes consisting of an outer container of wooden construction and a metal inner container separated by cushioning material.

   The metal inner container is approximately 11 inches by 18 inches by 178 inches long and is positioned within a wooden outer container approximately 30 inches by 30 inches by 207 inches long. Cushioning is provided between the inner and outer containers by phenolic impregnated honeycomb and ethafoam. Closure is accomplished by bolts. A pressure relief (breather) valve is provided on the inner container, and is set for 0.5 psi differential. The total weight of the packaging and contents is 2,800 pounds.

   (3) Drawings

   The packaging is constructed in accordance with the following General Electric Company Drawing Nos.:

   769E229, Revision 9
   769E231, Revision 8

5.(a) Packaging (Continued)

(4) *Product Container*

The fuel rod product container is constructed in accordance with General Electric Company Drawing No.:

0028B98, Revision 0

5.(b) Contents

(1) Type and form of material

(i) Unirradiated UO_2 fuel assemblies. Each fuel assembly is made up of either 60 or 62 rods in an 8 x 8 square array with maximum fuel cross-sectional area of 25 square inches and a maximum fuel length of 150 inches. The maximum U-235 enrichment is 5.0 percent by weight, and the maximum average enrichment is 5.0 percent by weight. The maximum pellet diameter, minimum clad thickness, water rod specifications, and poison rod specifications are in accordance with Section 6.1, Appendix 8-H, of the supplements dated June 27 and November 1, 1995.

(ii) Unirradiated UO_2 fuel assemblies. Each fuel assembly is made up of 74 full and partial length rods in a 9 x 9 square array with maximum fuel cross-sectional area of 25 square inches and a maximum fuel length of 150 inches. The maximum U-235 enrichment is 5.0 percent by weight, and the maximum average enrichment is 4.6 percent by weight. The maximum pellet diameter, minimum clad thickness, water rod specifications, and poison rod specifications are in accordance with Section 6.1, Appendix 8-I, of the supplements dated June 27 and November 1, 1995.

(iii) Unirradiated UO_2 fuel assemblies. Each fuel assembly is made up of 92 full and partial length rods in a 10 x 10 square array with maximum fuel cross-sectional area of 25 square inches and a maximum fuel length of 150 inches. The maximum U-235 enrichment is 5.5 percent by weight, and the maximum average enrichment is 5.0 percent by weight. The maximum pellet diameter, minimum clad thickness, water rod specifications, and poison rod specifications are in accordance with Section 6.1, Appendix 8-J, of the supplements dated June 27 and November 1, 1995.

(iv) Unirradiated UO_2 fuel assemblies. Each fuel assembly is made up of 92 full and partial length rods in a 10 x 10 square array with maximum fuel cross-sectional area of 25 square inches and a maximum fuel length of 150 inches. The maximum U-235 enrichment is 5.0 percent by weight, and the maximum average enrichment is 4.7 percent by weight. The maximum pellet diameter, minimum clad thickness, water rod specifications, and poison rod specifications are in accordance with Section 5.1 and Table 5.1 contained in Appendix 8-J(a) of the supplement dated May 10, 2005.

5.(b) Contents (Continued)

(v) Unirradiated UO_2 fuel rods, which are contained within the product container specified in 5(a)(4). The maximum U-235 enrichment is 5.0 percent by weight. The fuel rods are clad with zircaloy, incaloy, inconel, or stainless steel. The minimum pellet diameter is 0.340 inch, and the maximum pellet diameter is 0.515 inch.

(vi) Unirradiated UO_2 fuel rods, which may be loose or may be strapped together. The maximum U-235 enrichment is 5.0 percent by weight. The fuel rods are clad with zircaloy, incaloy, inconel, or stainless steel. The minimum pellet diameter is 0.340 inch, and the maximum pellet diameter is 0.515 inch.

(2) Maximum quantity of material per package

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

Two (2) fuel assemblies. Total quantity of radioactive material within a package may not exceed a Type A quantity.

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

Two (2) fuel bundles. A fuel bundle is defined as any number of fuel rods contained within the product container specified in 5(a)(4).

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

Two (2) fuel bundles. A fuel bundle is defined as a maximum of 14 fuel rods positioned within one side (channel) of the inner container.

(c) Criticality Safety Index

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):	0.4
For the contents described in 5(b)(1)(iv), and limited in 5(b)(2)(i):	0.8
For the contents described in 5(b)(1)(v), and limited in 5(b)(2)(ii):	6.3
For the contents described in 5(b)(1)(vi), and limited in 5(b)(2)(iii):	2.9

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 holders with a maximum effective thickness of 0.151 inches (0.3835 cm) may be placed surrounding the fuel assembly up to a maximum of 0.13 grams H_2O hydrogen equivalent per cubic centimeter averaged over the assembly. The effective holder thickness is the linear average of the maximum and minimum thickness.

8. Polyethylene shipping shims may be inserted between rods within the fuel assemblies up to a maximum of 0.10 grams H_2O hydrogen equivalent per cubic centimeter averaged over the assembly. The shipping shims may be used with or without the polyethylene holders.

9. For shipment of fuel rods described in 5(b)(1)(v) and 5(b)(1)(vi), each fuel rod may be contained within a polyethylene sheath with a maximum thickness of 0.01 inch. Dunnage is permitted within the product container, and within the inner container, provided that the dunnage does not have a hydrogen density greater than that of water.

10. Maximum average enrichment means the highest enrichment averaged over any axial zone of the assembly.

11. In addition to the requirements of Subpart G of 10 CFR Part 71, each packaging must meet the Acceptance Tests and Maintenance Program of Chapter 6 of the application, and the package must be prepared for shipment and operated in accordance with the Operating Procedures of Chapter 6 of the application.

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

13. Expiration date: March 31, 2008.

REFERENCES

General Electric Company application dated September 10, 1997.

Supplements dated: November 20, 1997; June 5 and 25, July 1 and 21, and August 14, 1998; October 14, 1999; December 19, 2002; January 21, and December 3, 2004; and April 18 and May 10, 2005.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Robert Lewis, Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: 07 May 2005

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
5059	13	71-5059	USA/5059/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)*
Nuclear Fuel Services, Inc.
P.O. Box 337, MS 123
Erwin, TN 37650

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION
Nuclear Fuel Services, Inc., application dated January 20, 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.: NFS Uranyl Nitrate Tank Trailer

(2) Description

A stainless steel bulk liquid cargo tank, which is permanently mounted to a semitrailer, designed for the transport of fissile uranyl nitrate solutions. The tank is cylindrical with torispherical heads. It is approximately 437 inches in length and 50-1/2 inches in diameter, with a shell and head wall thickness of 3/16 inch. The tank is covered with about 4 inches of fiberglass insulation with an outer jacket of 22-gage or 14-gage stainless steel. The nominal tank capacity is 3700 gallons with a design ullage of 3 percent. The tank is equipped with access ports on top of the tank .

(3) Drawings

The tank trailer is constructed in accordance with the following drawings:

Independent Metal Products Drawing No. UNF 2232, Sheet 1 of 3, Rev. A; and Sheet 3 of 3, Rev. A; and Nuclear Fuel Services, Inc., Drawing No. 000-M0337-D, Rev. -.

5. (b) Contents

(1) Type and form of material

Uranyl nitrate in dilute acid solution. The solution must meet the following limits:

Maximum uranium enrichment	5 weight percent U-235 (0.05 grams U-235/gram U)
Uranium concentration	70 - 350 grams U/liter solution
Maximum U-235 concentration	5 gram U-235/liter solution
Maximum solution freezing point	0°C (32°F)
Maximum solution density	13 pounds/gallon
Free HNO_3 concentration	0.1 M - 0.8 M

The uranium may be prepared from either non-recycled or recycled uranium. The uranium must meet the following limits:

Constituent	**Maximum concentration**
Tc-99 concentration	5 micrograms Tc-99/gram U
U-232 concentration	0.002 microgram U-232/gram U
U-234 concentration	2,000 micrograms U-234/gram U
U-236 concentration	25,000 micrograms U-236/gram U
Alpha activity (Np and Pu)	40 bequerel/gram U
Gamma activity (fission products)	440,000 MeV-bequerel/kilogram U

(2) Maximum quantity of material per package

Not more than 3,589 gallons and not more than 45,600 pounds net weight of uranyl nitrate acid solution. Total 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: 100

6. The solution must be at a temperature of 68°F or above at the time of packaging.

7. Transport of the package may only be initiated if the minimum predicted temperature along the transport route for the anticipated transport period is greater than 32°F. In the event freezing weather is encountered, the carrier must comply with the administrative procedures and controls as specified in Section 3.2.2 of the application.

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

   (a) Each package must be maintained in accordance with Chapter 8 of the application.

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

   (c) Each package shall comply with U.S. Department of Transportation requirements for the use, qualification, and maintenance of specification MC 311 cargo tanks, including the provisions of Subpart E of 49 CFR Part 180.

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: May 31, 2007.

REFERENCES

Nuclear Fuel Services, Inc., application dated January 20, 2000.

Supplements dated: October 13 and November 21, 2000; and October 19 and December 13, 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: June 11, 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
5086	12	71-5086	USA/5086/B(U)F	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)*

BWXT, Nuclear Products Division
P.O. Box 785
Lynchburg, VA 24505-0785

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

BWXT, Nuclear Products Division application dated December 23, 2003.

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.: UNC-2600

(2) Description

The inner container is an 11-gauge steel box with inside dimensions of 2-5/8" high x 7" wide x 96" long. The inner container is supported in a 22-1/2" ID by 102-1/2" long, 14-gauge steel drum by an insertable cage formed by nine 21-1/2" diameter by 3/8" thick steel plates, spaced approximately 12" apart, with a channel formed through the center of the plates by angle irons. The outer container closure is made with a 14-gauge drum lid with 12-gauge bolt locking ring with drop forged lugs, one of which is threaded, having a 5/8" diameter bolt.

(3) Drawings

The packaging is constructed in accordance with Thomas Gutman Consultant Drawing No. B-2600-2, Sheets 1 through 6, Rev. 3.

(b) Contents

(1) Type and form of material

Unirradiated, uranium-zirconium, fuel elements. The uranium may be enriched to any degree in the U-235 isotope.

(2) Maximum quantity of material per package

Up to 8.9 kilograms of U-235 per package. The ratio of the weight of U-235 to the weight of U-235 plus zirconium shall not exceed 0.074. The net weight of the contents shall not exceed 265 pounds.

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

Minimum transport index to be shown on label for nuclear criticality control: 1.4

3. 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 7 of the application.

(b) The package must be acceptance tested and maintained in accordance with Chapter 8 of the application.

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

8. Expiration date: February 28, 2009.

REFERENCES

BWXT, Nuclear Products Division dated December 23, 2003.

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: February 05, 2004

NRC FORM 618
(6-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. PACKAGE IDENTIFICATION NUMBER	d. PAGE	e. TOTAL NUMBER PAGES
5149	11	USA/5149/B( )F	1	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)*

BWX Technologies, Inc.
P.O. Box 785
Lynchburg, VA 24505

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Babcock & Wilcox Company application dated September 20, 1979, as supplemented.

c. DOCKET NUMBER 71-5149

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.: 814A

(2) Description

Steel container as described in Babcock & Wilcox Company's application dated September 20, 1979.

(b) Contents

(1) Type and form of material

Unirradiated fuel cluster.

(2) Maximum quantity of material per package

One fuel cluster containing U-235 with inserted poison fixture as specified in Babcock & Wilcox Company's application dated September 20, 1979.

(c) Criticality Safety Index

Minimum transport index to be shown on label for nuclear criticality control: 25.0

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

7. Use of packaging fabricated after August 31, 1986, is not authorized.

PRINTED ON RECYCLED PAPER

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 the Operating Procedures in the application supplement dated May 18, 1990.

   (b) The package shall be maintained in accordance with the Maintenance Program in the application supplement dated May 18, 1990.

9. Expiration date: October 1, 2008.

REFERENCE

Babcock & Wilcox application dated September 20, 1979.

Babcock & Wilcox supplements dated: May 18, 1990, and April 27, 1995.

BWX Technologies, Inc., supplements dated: June 1, 2000, and May 18, 2005.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Robert J. Lewis, Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: 06 Sept 2005

UNITED STATES NUCLEAR REGULATORY COMMISSION

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
5580	8	71-5580	USA/5580/B( )F	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)*

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

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Safety Analysis Report for S5W Power Unit shipping container dated August 9, 1968, as amended.

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.: S5W Power Unit

(2) Description

The S5W Power Unit shipping container (PUSC) is a container and support assembly designed to ship and store new naval reactor power units. The PUSC is comprised essentially of three major assemblies: (1) the outer frame, (2) the inner frame, and (3) the shipping container. During shipment, the shipping container is bolted to the inner frame in a horizontal position. Two trunnions welded to the middle section of the shipping container support the lower end of the container and also provide the means whereby the container can be rotated from the horizontal (shipping) attitude to the vertical (loading-unloading) attitude in the inner frame. The inner frame and shipping container are supported by the outer frame and pedestal through 80 elastic shock mounts, each of which is secured to both the inner frame and outer frame.

Approximate dimensions of the three major assemblies of the PUSC are: shipping container: 95 inches diameter by 234 inches; Inner Frame: 109 inches width by 52 inches height by 269 inches length; Outer Frame: 121 inches width by 56 inches height by 236 inches length. Maximum weight of the loaded PUSC is approximately 127,900 lbs.

(3) Drawings

The packaging is constructed in accordance with Westinghouse Electric Corporation Drawing Nos. 936F963, Rev. 3 and 936F964, Rev. 2.

NRC FORM 618 (8-2000) 10 CFR 71	CERTIFICATE OF COMPLIANCE FOR RADIOACTIVE MATERIAL PACKAGES		U.S. NUCLEAR REGULATORY COMMISSION	
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5580	8	71-5580	USA/5580/B( )F	2 OF 2

5.(b) Contents

(1) Type and form of material

Unirradiated fuel in the form of S3G Core 3 power units with control rods installed and secured in place by holddown mechanisms.

(2) Maximum quantity of material per package

One fuel assembly.

5.(c) Transport Index for Criticality Control

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

6. Expiration date: December 31, 2007

REFERENCE

Safety Analysis Report for S5W Power Unit Shipping Container, WAPD-OP(R)SA-820 dated August 9, 1968; Addendum to WAPD-OP(R)SA-820 dated September 28, 1987.

Naval Reactors Supplements dated: March 2, 1992 (G#92-03388) and June 11, 1997 (G#97-03513); and Naval Reactors letter (G#02-0820) dated June 13, 2002.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

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

Date July 29, 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
5607	12	71-5607	USA/5607/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

T-2 Shipping Package, Safety Analysis Report Consolidated Application dated 8/22/03, 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-2

(2) Description

Packaging for irradiated reactor fuel and components consisting of a lead encased in steel cask, removable containment vessel insert and shipping case.

The cask is a double-walled steel circular cylinder with thickened shielding in the center portion. The central cavity is 6.065 inches in diameter by 100 inches long. The lead shielding is 8.0 inches thick along a 45-inch center section reduced to 4.2 inches at each 36-inch end section. The containment vessel is positioned within the cask. Cask closure is accomplished by a gasketed and bolted steel plug. The cask is enclosed in the shipping case which is 36 inches in diameter by 133 inches long welded to a 4-foot by 6-foot steel pallet. The maximum weight of the packaging is approximately 20,600 pounds.

(3) Drawings

(i) The shipping case is constructed in accordance with DuPont Drawing Nos.: W716539, Rev. 1; 180191, Rev. 1; 180192, Rev. 0; 180193, Rev. 2; 180194, Rev. 0; 180197, Rev. 0; W716538, Rev. 0; 180195, Rev. 0; 180196, Rev. 0; and 180089, Rev. 0.

(ii) The cask is constructed in accordance with General Electric Drawing Nos.: 919D755, Rev. 1; 135C5202, Rev. 0; 153F966, Rev. 1; and 106D3721, Rev. 1; or it is constructed in accordance with DuPont Drawing Nos.: W239534, Rev. 2; 147214, Rev. 1; 147215, Rev. 2; and 147216, Rev. 1.

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

(iii) The ANL insert is constructed in accordance with Argonne National Laboratory Drawing Nos.: W0147-0227-DD, Rev 7; W0147-0228-DD, Rev. 6; W0147-0229-DC, Rev. 6; W0147-0231-DD, Rev. 3; W0147-0234-DC, Rev. 4; and W0147-0312-DE, Rev. 2.

(b) Contents

(1) Type and form of material

(i) Irradiated clad fuel in the form of solid metal, oxides, nitrides, and carbides of uranium, plutonium, or mixed uranium-plutonium contained within the ANL insert. The clad fuel may contain small quantities of Na or NaK. The minimum cooling time must be no less than 150 days.

(ii) Irradiated clad fuel pins of uranium dioxide enriched to up to 3.0 w/o in U-235 contained within the ANL insert. Average exposure of fuel not to exceed 18 megawatt days per kilogram. The clad fuel may contain small quantities of Na or NaK. The minimum cooling time must be no less than 90 days.

(iii) Irradiated reactor components held within the container shown in Drawing No. W0147-0234-DC, Rev. 4.

(2) Maximum quantity of material per package.

Internal decay heat not to exceed 208 watts, and:

(i) For the material described in 5(b)(1)(i), fissile material not to exceed 1.71 kg.

(ii) For the material described in 5(b)(1)(ii), fissile material (U-235) not to exceed 300 grams.

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

For the contents described in 5(b)(1)(i) and 5(b)(1)(ii), and limited in 5(b)(2)(i) and 5(b)(2)(ii):

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

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
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6. The contents must be shipped dry. When loaded underwater, the package must be dried using Consumer Power Company's procedure, "T-2 Cask Liner Assembly Drying Procedure," Proc. No. EE&T-C12, Rev. 1, 11/12/81.

7. The ANL Insert must be leak tested prior to first use and annually thereafter in accordance with the procedures specified in Argonne National Laboratories Document No. W0195-0054-ES-00.

8. Prior to each shipment, the package must be leak tested in accordance with procedures specified in HFEF Operating Instruction 6202.

9. In addition to the requirements of Subpart G of 10 CFR Part 71 and the other conditions of this certificate:

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

   (b) The package must be maintained in accordance with the Maintenance Program of Chapter 8 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, until October 1, 2004, and under provisions of 10 CFR 71.17 thereafter.

11. Expiration date: October 1, 2008.

REFERENCES

T-2 Shipping Package, Safety Analysis Report, Consolidated Application dated August 22, 2003.

Department of Energy supplement dated February 13, 2004.

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: March 03, 2004

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
5740	7	71-5740	USA/5740/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)*
U.S. Department of Energy
Washington, D.C. 20585

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION
Safety Analysis Report for Packaging (SARP) of the Oak Ridge National Laboratory TRU Californium Shipping Container, August 7, 1981, Rev. of Report No. ORNL-5409/R1, 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.

Packaging

(1) Model No.: ORNL TRU Californium Shipping Container

(2) Description

A 304L stainless steel encased concrete shipping cask. The outer shell consists of two, 1/2-inch thick, 66-inch diameter hemispherical heads joined by a 6-inch cylindrical section. The cylindrical cavity has a 1-inch thick stainless steel wall and is 3 inches in diameter x 6 inches long. Shielding consists of 30 inches of Blackburn Limonite concrete having a density of approximately 175 lb/ft3. Upper and lower level ball valves located at the end of concrete filled plugs define, isolate, and seal the cavity. Both of these plugs have 0-ring seals, are bolted in place and are protected with a gasketed cover plate. Fusible plugs are located in the cover plates and the shell.

The top ball valve and plug may be replaced by other plugs for multiple source shipments. Sources are contained in special form inner containers.

The cask is mounted onto a 1-inch thick steel base plate by eight steel 2-1/2 inch NPS Schedule 40 pipe struts. The cask is transported on a special trailer. The package gross weight is 23,500 pounds.

5. (a)(3) Drawings

The package and special trailer are constructed in accordance with Oak Ridge National Laboratory (ORNL) Drawing Nos.:
M-11230-EN-001-D Rev. 4
M-11230-EN-002-D Rev. 0
M-11230-EN-003-D Rev. 0
M-11230-EN-004-D Rev. 2
M-11230-EN-005-D Rev. 0
M-11230-EN-006-D Rev. 0
M-11230-EN-007-D Rev. 0
M-11230-EN-008-D Rev. 1
M-11230-EN-012-E Rev. 4
M-11230-EN-014-E Rev. 3
M-11230-EN-017-D Rev. 3
M-11230-EN-018-E Rev. 0
(Appendix A, August 7, 1981 revision of ORNL-5409/R1, as supplemented.)

(2) Contents

(1) Type and form of material

The contents consist of isotopes of Americium (Am), Curium (Cm), Berkelium (Bk), Californium (Cf), Einsteinium (Es) and Fermium (Fm) as a solid (metal, oxide, oxysulfate, or dry salt), contained in capsule(s) that meet the requirements of special form radioactive material.

(2) Maximum quantity of material per package:

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

Three (3) grams and the maximum internal heat not to exceed 5 watts.

6. The contents described in 5(b)(1) must be shipped in a seal welded special form inner container as described in section 5.2.1 of the application.

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

(1) Each packaging must be maintained in accordance with the supplement dated May 10, 1991; and

(ii) The package must be prepared for shipment and operated in accordance with the supplement dated May 10, 1991.

A minimum of two lifting ribs shall be used to lift the package.

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

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE	PAGES
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10. Revision No. 6 of this certificate may be used until July 31, 2007.

11. Expiration date: October 1, 2008. This certificate is not renewable.

REFERENCES

Safety Analysis Report for Packaging (SARP) of the Oak Ridge National Laboratory TRU Californium Shipping Container, August 7, 1981, revision of Report No. ORNL-5409/R1.

Supplements dated: April 4, 1986; March 26, April 23, and May 10, 1991; June 4, 1992; May 13, 1996; May 24, 2001; and April 12, 2006.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

for L. Christopher Regan, Acting Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: July 20, 2006

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
5757	8	71-5757	USA/5757/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 Report for Neutron Source Shipping container dated February 14, 1968, 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.: Neutron Source Shipping and Installation Container

(2) Description

The Neutron Source Shipping and Installation container consists of two structures, one nested within the other, having an overall envelope of 5 feet, 5 inches diameter by 9 feet, 5-5/8 inches length. The outer structure, the shipping container, is a ring of polyethylene 11-1/2 inches thick with an OD of 5 feet 4 inches and length of approximately 5 feet 2 inches. The polyethylene is canned in a ½-inch thick carbon steel shell. The inner structure, the replacement and installation container, fits into the cavity of the outer structure. This assembly consists of a 6-1/2 inch OD, 79-5/8 inches long stainless central tube, which is plugged at both ends by machined stainless steel forging. Three cavities are machined in the bottom end plug to contain the neutron source assemblies. A jacket of lead, 6 inches thick, encircles the central tube and this innermost layer of shielding to attenuate the gamma radiation. A wall of polyethylene, 8-1/2 inches thick, surrounds the lead shield and is canned with a ½-inch thick carbon steel plate. Gross weight is approximately 19,000 pounds.

(3) Drawings

The packaging is constructed in accordance with Westinghouse Electric Corporation Drawing Nos. 905D318, Rev. C; 905D315, Rev. F; and 905D285, Rev. A.

(b) Contents

(1) Type and form of material

(i) Radium-Beryllium special form radioactive material neutron source. These sources may be either new or irradiated and have surface contamination as a result of previous use.

(ii) Plutonium 238-Beryllium special form radioactive material neutron source. These sources may be either new or irradiated and have surface contamination as a result of previous use.

(2) Maximum quantity of material per package

(i) One, two, or three neutron sources as described in 5(b)(1)(i) and limited to a total emission rate of 1.9×10^8 n/sec. These sources are limited to a combined surface contamination of not more than an A_2 quantity of radioactive material.

(ii) One, two, or three neutron sources as described in 5(b)(1)(ii) and limited to a total emission rate of 2.5×10^9 n/sec. These sources are limited to a combined surface contamination of not more than an A_2 quantity of radioactive material.

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

Minimum transport index to be shown on label for nuclear criticality control: 11.2

6. The different types of sources shall not be intermixed within the same container for shipment.

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, 2008.

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
5757	8	71-5757	USA/5757/B( )F	3	OF	3

REFERENCES

Safety Analysis Report for Neutron Source Shipping Container, WAPD-OP(R)S-2473 dated February 14, 1968.

Supplements: Bettis Atomic Power Laboratory letter WAPD-OP(R)C-474 dated December 22, 1975. Naval Reactors letter G#92-03738, dated October 15, 1992; G#C97-03621 dated October 17, 1997; and G#02-4094 dated November 20, 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

: June 09, 2003

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
5796	15	71-5796	USA/5796/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)*

Advanced Medical Systems Inc.
121 North Eagle Street
Geneva, OH 44041

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Advanced Medical Systems, Inc. application dated June 21, 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 Nos.: 181375 and 181361

(2) Description

Overpacks that provide impact and thermal protection for teletherapy head assemblies or source exchange assemblies. The cubical overpacks covered with 16 gauge steel panels. Reinforcing steel straps and angles are welded together and spaced to limit the openings between them to less than 6 inches. Skid runners are provided to facilitate fork lift usage. Dimensions of the Model No. 181375 are 43.5"L x 39.75"W x 41"H with a maximum gross weight of 3,750 pounds. Dimensions of the Model No. 181361 are 39"L x 34.25"W x 44.5"H with a maximum gross weight of 4,000 pounds.

(3) Drawing

(I) The Model No. 181375 packaging is constructed in accordance with Advanced Medical Systems, Inc. Drawing Nos.: E590G; D16423A; D16423B; D16424D; D16479; D16568; C16580E; B46411; A46686A; E63790F; D181368G; D181369E (2 pages); D181375N; D184705; D184713; D200016G; D200043; D200073F; D200074C; D200075C; D200079C; C200742-1 THRU 5; B200743-1,5; and B200745-1 THRU 4.

(ii) The Model No. 181361 packaging is constructed in accordance with Advanced Medical Systems, Inc. Drawing Nos.: D-T60-478-B; C50104-B; D55100-A; C55103-B; C55105-B; D13706A-D (2 pages); D-181356-F; D-181357-F; D-181361-E, B181390-B; and D-200017-A.

5. (b) Contents

(1) Type and form of material

(I) Cobalt 60 sealed sources that meet the requirements of special form radioactive material; or

(ii) Cesium 137 in the form of cesium chloride encapsulated in sealed sources that meet the requirements of special form radioactive material.

(2) Maximum quantity of material per package

(I) 13,680 curies of cobalt 60 with a radioactive decay heat load not to exceed 200 watts; or

(ii) 2,200 curies of cesium 137 with a radioactive decay heat load not to exceed 17 watts.

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

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

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

7. Use of packaging fabricated after August 31, 1986, is not authorized.

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

9. Expiration date: August 31, 2007.

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

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REFERENCES

Advanced Medical Systems, Inc. application dated June 21, 2002.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Charles L. Miller for

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

Date: August 15, 2002

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
5797	15	71-5797	USA/5797/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)*
U.S. Department of Energy
Washington, D.C. 20585

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION
U.S. Department of Energy application dated May 30, 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.: Inner HFIR Unirradiated Fuel Element Shipping Container, and Outer HFIR Unirradiated Fuel Element Shipping Container

(2) Description

Packaging for unirradiated fissile radioactive material as fuel elements for the High Flux Isotope Reactor (HFIR). The containers are right circular cylinders with an 11-gauge carbon steel shell. The lid is attached to the container with sixteen 3/8-16x1-inch steel bolts. The steel shell is filled with stacked fir plywood rings. The plywood rings form a central cavity which is lined with 1-inch thick polyethylene foam.

The packaging for the inner HFIR fuel element has overall dimension of 25 inches OD bye 45 inches high, a 10-7/8-inch diameter by 30-1/4-inch deep cavity, and a 660 pound gross weight.

The packaging for the outer HFIR fuel element has overall dimensions of 31.5 inches OD x 45.75 inches high, a 17-3/8-inch diameter by 31-1/8-inch deep cavity, and a 1,050 pound gross weight.

(3) Drawings

(i) The packaging for the inner HFIR fuel is constructed in accordance with Martin Marietta Energy Systems, Inc., Drawing Nos. M-20978-EL-003E, Rev. E, and M-20978-EL-008E, Rev. C

5. (a) (3) Drawings (continued)

(ii) The packaging for the outer HFIR fuel is constructed in accordance with Martin Marietta Energy Systems, Inc., Drawing Nos. M-20978-EL-002E, Rev. D, and M-20978-EL-008E, Rev. C

(b) Contents

(1) Type and form of material

Uranium as U_3O_8-Al cermet, enriched up to 95% in the U-235 isotope, and clad in aluminum, 10-mils thick, and:

(i) For the packaging described in 5(a)(3)(i), the contents are described in ORNL/TM-9220, "Specifications for High Flux Isotope Reactor Fuel Elements HFIR-FE-3," and in the following Oak Ridge National Laboratory Drawing Nos.: E-42118, Rev. Q; E-42112, Rev. H; D-42113, Rev. G; E-42114, Rev. H; and E-42117, Rev. H.

(ii) For the packaging described in 5(a)(3)(ii) the contents are described in ORNL/TM-9220, "Specifications for High Flux Isotope Reactor Fuel Elements HFIR-FE-3," and in the following Oak Ridge National Laboratory Drawing Nos.: E-42126, Rev. M; E-42120, Rev. H; D-42121, Rev. H; D-42122, Rev H; and E-42125, Rev. J.

(2) Maximum quantity of material per package

(i) For the contents described in 5(b)(1)(i) not more than 2.63 kg of U-235.

(ii) For the contents described in 5(b)(1)(ii) not more than 6.88 kg of U-235.

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

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

6. The lid lifting attachments must be blocked as shown on Martin Marietta Energy Systems, Inc., Drawing No. M-20978-EL-009E, Rev. 2, to prevent inadvertent use of the attachments during transport.

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

   (a) Each package shall be maintained in accordance with the Maintenance Program in Chapter 8 of the application;

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

   (c) The fuel element shall meet the fabrication inspection requirements of ORNL/TM-9220, "Specifications for High Flux Isotope Reactor Fuel Elements HFIR-FE-3."

8. Use of packaging fabricated after December 31, 1976, is not authorized.

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: September 30, 2007.

REFERENCES

U.S. Department of Energy Application dated May 30, 1991.

Supplements dated: February 26, 1992; April 2, 1993; and September 23, 1996; September 2, 1998; February 24, 2000; and February 4, 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: ~~September 20,~~ 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
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2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*
Duratek
140 Stoneridge Drive
Columbia, SC 29210

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION
Chem-Nuclear Systems, Inc., application dated February 25, 1994.

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 3-55

(2) Description

The package is a steel-encased, lead-shielded cask with crushable impact limiters. The basic cask is a steel cylinder 133-3/4 inches long by 50-1/2-inches in diameter with maximum cavity dimensions of 36 inches in diameter by 116 inches long reduced to 111 inches by the shield ring attached to the lid cover. Shielding is provided by 6 inches of chemical lead in the sides and closure base plate and 5-1/4 inches in the closed end.

The outside steel encasement is made up of two, ½-inch plates on the sides and three plates totaling 2-5/8 inches on the end. The containment vessel is a 1/4-inch thick cylinder with a ½-inch end plate. The shells are welded together with the lead shielding poured to fill the annular and end spaces.

The removable, flanged and recessed base plate weldment consists of 3/8-inch and 1-1/4-inch outside plates and a 5/8-inch inside plate. The space between the plates is lead-filled.

The base plate is secured to the cask body by means of twelve, 1-1/2-inch high strength bolts and nuts and sealed with two silicone O-rings.

The cavity is penetrated by a vent line at the closed end and a drain line through the base plate. The vent line is sealed by a gasketed and shielded plug. The drain line is sealed with a 25 psig relief valve.

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

Cask appendages include two, 8-inch lifting trunnions and two, 4-inch removable tilting trunnions on the cask side.

Removable impact limiters are provided at the cask ends and at the two, 8-inch trunnions. The former consist of a series of 6-inch diameter closed end tubes. Each impact limiter has tubes approximately 6 inches long around the end periphery. The closure end impact limiter has 12 tubes, six about 6 inches long and six about 2 inches long, around the sides. The closed end impact limiter has six tubes about 6-inches long around the sides. A gusseted tube acts as the trunnion impact limiter.

The cask is secured horizontally to a skid which is mounted to the transport vehicle for shipment. An optional sunshade is provided.

The gross weight of the package, excluding the skid and sunshade is approximately 70,000 pounds. The skid weighs about 4,200 pounds.

(3) Drawings

The packaging is constructed in accordance with Chem-Nuclear Systems, Inc., Drawing Nos.: MOD 100, Rev. 14; C-111-D-0001, Rev. 0; and C-111-E-0002, Rev. 2; and ATCOR Drawing Nos.: MOD 139-1, Rev. K; MOD 140, Rev. C; MOD 124, Rev. 5; 0999-D-07, Rev. 8; and 0999-C-08, Rev. 9. An optional sunshade is constructed in accordance with Chem-Nuclear Systems, Inc., Drawing No. C-110-D-5001, Rev. 1.

(b) Contents

(1) Type and form of material

Depleted Antimony-Beryllium (Sb-Be) neutron sources and irradiated metal components packaged in secondary containers.

(2) Maximum quantity of material per package

Package internal decay heat load not to exceed 250 watts. The source strength of depleted neutron sources not to exceed 2.3 curies of Antimony-124.

6. (a) Both the inner cask cavity and the secondary container must be free of water when the package is delivered to a carrier for transport.

(b) Except for close fitting items, shoring must be placed between contents, secondary container and cask cavity to minimize secondary impacts due to accident sequence.

(c) The maximum gross weight of the contents, secondary container and shoring is limited to 9,220 pounds.

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
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7. Prior to each shipment, the silicone O-ring seals (base plate and vent plug) must be inspected, the seals must be replaced with new seals if inspection shows any defects or every six (6) months, whichever occurs first.

8. Prior to delivery of the package to a carrier for transport, the package containment cavity shall be leak tested. The sensitivity of the test shall be at least 1 x 10^{-1} atm-cm^3/sec (STP). In addition, the packaging containment cavity shall be leak tested at least once every twelve (12) months. The sensitivity of the test shall be at least 1 x 10^{-3} atm-cm^3/sec (STP).

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

10. Each packaging must meet the Acceptance Tests and Maintenance Program of Section 8.0 of the application.

11. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17. Fabrication of additional packagings after December 31, 1983, is not authorized.

12. Expiration date: October 1, 2008. This package is not renewable.

REFERENCES

Chem-Nuclear Systems, Inc. application dated February 25, 1994.

Supplements dated: February 16, 1999, December 5, 2000, January 23, February 2, March 2, April 23, 2001, October 3, 2002, and January 14, 2004.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

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

Date: March 11, 2005

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
5830	10	71-5830	USA/5830/B( )	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)*

Department of the Navy
Naval Sea Systems Command
Detachment
Radiological Affairs Support Office
PO Drawer 0260
NWS Yorktown, VA 23691-0260

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Minnesota Mining and Manufacturing Co.
Application dated June 28, 1968, 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.: SNAP-21

(2) Description

A thermoelectric generator 16 inches in diameter by 30 inches long packaged in a right circular metal protective enclosure 52 inches in diameter by 68 inches high. Main components of the generator consist of an outer Berylco-165 housing with flange; U-8 Mo shielding; thermal insulation; thermoelectric modules; and the heat source. Total weight of the package is 1,900 pounds.

(3) Drawings

The SNAP-21 is constructed in accordance with Minnesota Mining and Manufacturing Company Drawing No. B-SK-37-4014 and Drawings included in 3M Report No. MMM-3691-33.

(b) Contents

(1) Type and form of material

Strontium 90 titanate pellets doubly encapsulated by a thin inner liner and a 0.2-inch thick Hastelloy C primary containment capsule which meets the requirements of special form radioactive material.

(2) Maximum quantity of material per package 33,000 curies.

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

   (a) The package must be prepared for shipment, operated and maintained in accordance with Minnesota Mining and Manufacturing Company Report No. MMM 3691-42, "SNAP-21 Program, Phase II, Deep Sea Radioisotope-Fueled Thermoelectric Generator Power Supply System, Shipping and Handling Manual." For disposal shipments, temperature recorders and accelerometers are not required to be operational.

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

8. Revision No. 9 of this certificate may be used until December 1, 2006.

9. Expiration date: October 1, 2008. This certificate is not renewable.

REFERENCE

Minnesota Mining and Manufacturing Company application dated June 28, 1968.

Department of Navy supplements dated: June 8 and October 10, 1990, September 20, 1995, April 16, 1998, April 27, 2000, January 5, 2004, and October 18, 2005.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Robert Nelson, Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: 11/3/05

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
5862	9	71-5862	USA/5862/B( )	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)*

Department of the Air Force
HQ ATAC/SEG
1030 S. Highway A1A
Patrick AFB, FL 32925-3002

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Teledyne Energy Systems application dated June 26, 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.: Sentinel-100F

(2) Description

The package, a thermoelectric generator, is 45.5 inches in height with a base diameter of 24.5 inches (excluding mounting pads), and weighs approximately 2,600 pounds. The components include a Tungsten biological shield (10.705" X 13.837" OD) which is within the aluminum (6061) outer protective housing. Four 6061-T6 mounting pads at the base of the aluminum housing provide the shipping pallet attachment points.

(3) Drawings

The packaging is constructed in accordance with the following Isotopes, Inc. Drawing Nos.:

010F10000	Sheets 1-3 (Rev. C), Generator Assembly Sentinel 100F
010-20000	Sheets 1-2 (Rev. B), Fuel Capsule Assembly
010-70003	(Rev. A) Shield Body
010-70004	Shield Plug
001-90064	Sheets 1-2 (Rev. A), Shipping Crate Sentinel RTG
001-90039	Sheets 1-2 (Rev. J), Sheet 3 (Rev. H), and Sheet 4, Pallet Assembly

5. (b) Contents

(1) Type and form of material

Strontium-90 titanate doubly encapsulated in a stainless steel liner and Hastelloy or Uniloy HC capsule which meets the requirements of special form radioactive material.

(2) Maximum quantity of material per package

370,000 curies.

6. Fabrication of additional packagings is not authorized.

7. 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 supplement dated August 30, 1985.

(b) The package must be maintained in accordance with the Maintenance Program in the supplement dated August 30, 1985.

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

9. Expiration date: October 1, 2008. This certificate is not renewable.

REFERENCES

Teledyne Energy Systems application dated June 26, 1985.

Teledyne supplements dated: August 30, 1985; and July 26, 1990.

Department of the Air Force supplements dated: November 12, 1993; August 15, 1995; August 25, 2000; and August 30, 2005.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Robert J. Lewis, Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: 26 Sept 2005

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE	PAGES
5926	18	71-5926	USA/5926/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)*
General Electric Company
P.O. Box 460, Vallecitos Road
Pleasanton, CA 94566

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION
General Electric Company application dated January 18, 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.: GE-100

(2) Description

A steel encased lead shielded shipping cask. The cask is double-walled steel circular cylinder, 20-1/4-inch diameter by 26-7/8 inch high with a central cavity approximately 7-5/8-inch diameter by 10 inches high. Approximately 5-7/8 inches of lead surround the central cavity. The cask is equipped with a cavity drain line and lifting device. Closure is accomplished by a gasketed and bolted steel lead filled plug. For additional shielding lead, tungsten or uranium liners may be inserted in the cask cavity. The maximum weight of the packaging is 4,800 pounds.

(3) Drawings

The packaging is constructed in accordance with General Electric Company Drawing Nos. 129D4727, Rev. 5; 129D4729, Rev. 5; 129D4730, Rev. 4; and 129D4731, Rev. 1.

5. (b) Contents

(1) Type and form of material

(i) Byproduct and irradiated special nuclear material in the form of fuel rods, or plates, fuel assemblies, or meeting the requirements of special form radioactive material; or

(ii) Solid nonfissile irradiated metal hardware and reactor control rods (blades).

(2) Maximum quantity of material per package

Radioactive decay heat not to exceed 400 watts and 500 grams U-235 equivalent mass fissile material. (U-235 equivalent mass equals U-235 mass plus 1.66 times U-233 mass plus 1.66 times Pu mass).

Plutonium in excess of twenty (20) curies per package must be in the form of metal, metal alloy or reactor elements.

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

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

Minimum transport index to be shown on label for nuclear criticality control: 5.6

6. Shoring shall be provided to minimize movement of contents during accident conditions of transport.

7. At the time of delivery of the loaded package to a carrier for transport, the package contents shall be dry and the fissile material unmoderated (H to X atomic ratio less than 2).

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

(a) The package must be maintained in accordance with the maintenance procedures submitted with GE application dated January 18, 1993.

(b) The package must be prepared for shipment and operated in accordance with the operating procedures submitted with GE application dated January 18, 1993.

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: May 31, 2008.

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
5926	18	71-5926	USA/5926/B( )F	3	OF	3

REFERENCES

General Electric Company application dated January 18, 1993.

Supplements dated: March 3, 1993; November 19, 1997; and March 14, 2003.

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: May 30, 2003

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
5939	32	71-5939	USA/5939/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)*

General Electric Company
P.O. Box 460, Vallecitos Road
Sunol, CA 94566

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

General Electric Company application dated November 19, 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.: 1500

(2) Description

A steel encased lead shielded shipping cask. The cask is double-walled steel circular cylinder, approximately 30 1/4-inch diameter by 48 1/2 inches high with a central cavity approximately 7-inch diameter by 25 inches high. The diameter is reduced from 30 1/4 inches to 17 1/2 inches by cone construction at the top 7 inches of the cask. Approximately 11 inches of lead surround the central cavity. The cask is equipped with a cavity drain line and lifting device. Closure is accomplished by a gasketed and bolted steel lead-filled plug. A protective jacket consisting of an upright circular cylinder with open bottom and a protruding box section diametrically across the top and vertically down the sides attaches to a square pallet. Dimensions of the protective jacket are approximately 60 7/8 inches high by 50 inches wide across the box section. The outer cylindrical diameter is 36 1/2 inches and the pallet is 59 1/2 inches square. The maximum weight of the packaging is approximately 15,500 pounds.

(3) Drawings

(i) The packaging is constructed in accordance with General Electric Company Drawing Nos. 129D4748, Rev. 7; 129D4749, Rev. 5; and 129D4750, Rev. 9.

(ii) An optional canister insert is constructed in accordance with the following Chem-Nuclear Systems, Inc., Drawing Nos., supplement dated March 1, 1993:

C-110-D-48019-001, Rev. D; and C-110-A-48019-002, Rev. C.

5.(b) Contents

(1) Type and form of material

(i) Byproduct material and special nuclear material meeting the requirements of special form radioactive material and antimony pins encased in stainless steel, or

(ii) Byproduct material as $^{90}SrF_2$ or $^{137}CsCl$ capsules meeting Condition No. 6, below, or

(iii) Solid nonfissile irradiated metal hardware and reactor control rods (blades), or

(iv) Stainless steel encapsulated solid metal Co-60 sources, or

(v) Byproduct material as $^{137}CsCl$ capsules meeting Condition No. 7, below.

(2) Maximum quantity of material per package

Not to exceed a decay heat generation of 3,120 watts and

(i) Item 5(b)(1)(i) above:

500 grams U-235 equivalent mass. (U-235 equivalent mass equals U-235 mass plus 1.66 times Pu mass). Plutonium in excess of 20 curies per package must be in the form of metal, metal alloy or reactor fuel elements.

(ii) Item 5(b)(1)(ii) above:

458,000 curies.

(iii) Item 5(b)(1)(iv) above:

200,000 curies.

(iv) Item 5(b)(1)(v) above:

157,000 curies.

(c) Criticality Safety Index 5.7

6. For the contents described in 5(b)(1)(ii): The $^{90}SrF_2$ capsules must be in accordance with Vitro Drawing Nos. H-2-66759, Rev. 0; and H-2-66758, Rev. 0. The $^{137}CsCl$ capsules must be in accordance with Vitro Drawing Nos. H-2-66760, Rev. 0; and H-2-66761, Rev. 0. After fabrication, the $^{90}SrF_2$ and $^{137}CsCl$ capsules must be leak tested using a method having sufficient sensitivity to detect a leak rate of 10^{-8} atm cc/sec. Any capsule with a detectable leak may not be delivered to a carrier for transport.

7. For the contents described in 5(b)(1)(v): The $^{137}CsCl$ capsules must be contained in the canister insert described in item 5(a)(3)(ii), above. The $^{137}CsCl$ capsules must be constructed and tested in accordance with Section 1.2.3 of the Chem-Nuclear Systems, Incorporated supplement dated March 1, 1993. The canister insert must be operated, tested, and maintained in accordance with Chapters 7 and 8 of the Chem-Nuclear Systems, Inc., supplement dated March 1, 1993. The shipment period must be completed within 30 days following the placement of the canister lid on the canister insert.

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

   (a) Except for packaging Serial Number 1506, the package must be prepared for shipment, operated, and maintained in accordance with the "Shipping Package Assembly/Disassembly" sections of the application, supplement dated September 27, 2001.

   (b) The silicone rubber lid gaskets must be replaced within the 12-month period preceding each shipment. Prior to each shipment the silicone rubber lid gaskets must be inspected. The silicone rubber gaskets must be replaced if inspection shows any defects. Cavity drain line must be sealed with appropriate sealant applied to threads of pipe plug.

   (c) Packaging Serial Number 1506 must be prepared for shipment, operated, and maintained in accordance with Neutron Products, Inc., supplement dated October 10, 2002.

   (d) Packaging Serial Number 1506 must be bubble tested within the 12-month period preceding each shipment, and after each third use. The bubble test must be performed in accordance with Neutron Products, Inc., supplement dated October 10, 2002.

9. Except for packaging Serial Number 1506, the package may only be dry loaded and unloaded; loading or unloading under water is not authorized.

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

11. Expiration date: October 1, 2008. This package is not renewable.

REFERENCES

General Electric Company application dated November 19, 1992.

General Electric Company supplements dated December 12, 1997, August 13, 1998, and August 27 and September 27, 2001; and September 24, 2003.

Chem-Nuclear Systems, Inc., supplement dated March 1, 1993.

Neutron Products, Inc., supplements dated February 1 and October 10, 2002.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Meraj Rahimi

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

Date March 11, 2005

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
5957	28	71-5957	USA/5957/B( )F	1 OF 9

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

Department of Energy application dated April 18, 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. BMI-1

(2) Description

A steel-encased lead shielded shipping cask. The basic cask body is a cylinder 33.37 inches in diameter by 73.37 inches high formed by two concentric stainless steel shells whose annular region is filled with lead. The outer ½-inch thick shell has a 0.12-inch thick plate spot welded to it, providing a 0.06-inch thick air gap insulator. The inner shell is 15.5 inches inside diameter by 54 inches inside length. The cask lid is a stainless steel weldment having 7.75 inches of lead shielding. The cask lid is secured to the cask by twelve steel studs which are welded to the cask body. The cask is provided with a drain line with needle valve and plug, pressure gauge, and a pressure relief valve. The total cask weight, including maximum contents of 1,800 lbs, is 23,660 lbs.

(3) Drawings

The cask is constructed in accordance with the following Battelle Memorial Institute (BMI) Drawing Nos.: 43-6704-0001, Rev. B; and 41-4409-0003, Rev. B.

5.(a) Packaging (continued)

(4) Product Containers

The various authorized product containers are constructed in accordance with the following Drawing Nos.:

(i) Inner can assembly as shown in BMI Drawing No. 00-000-421, Rev. C.

(ii) Basket Assembly as shown in BMI Drawing Nos. BCL-000-500, Rev. A; BCL-000-501, Rev. A; and 0048, Rev. A.

(iii) Fermi Fuel Element copper casting assembly as shown in BMI Drawing No. K5928-5 0049D, Rev. to May 12, 1966.

(iv) Basket Assembly as shown in BMI Drawing No. 1020, Rev. B (or with alternate spacer shown in GI Drawing No. 334D2193) or GA Drawing No. 9590001, Rev. A. Failed fuel assemblies must be seal welded in aluminum or stainless steel tubes with wall and end cap thicknesses of at least 0.015 inch.

(v) Basket Assembly defined by BMI Drawing No. BCL-000-500, Rev. A, as modified by BMI Drawing Nos. 00-000-236, Rev. C, and BCL-000-502, Rev. B.

(vi) Basket Assembly and storage can defined by BMI Drawing No. 00-000-391, Rev. C, and Atomic International Drawing No. AIHL, S8DR 0019-01, Rev. A, respectively.

(vii) Inner can assembly as shown in Union Carbide Corporation Drawing No. 101501, Rev. A.

(viii) Basket Assembly as shown in University of Missouri Research Reactor (MURR) Drawing No. 2234, Sheets 1 through 5, Revision 0.

(ix) HFBR assembly basket and spacer plate as shown in Brookhaven National Laboratory Drawing Nos.: BNL 93-001, Sheets 1, 2, and 3, Rev. 2, and BNL 93-002, Sheet 1, Rev. 2.

(x) Basket assembly as shown in General Electric Company Drawing No. 183C8253, Rev. 1.

5.(b) Contents

(1) Type and form of material

(i) Intact irradiated MTR- or BRR-type fuel assemblies containing not more than 200 grams U-235 per assembly prior to irradiation. Uranium may be enriched to a maximum 93.5 w/o in the U-235 isotope. Active fuel length shall be approximately 25 inches.

(ii) Intact irradiated Enrico Fermi Core. A fuel assembly containing not more than 4.77 kgs U-235 prior to irradiation. Uranium may be enriched to 25.6 w/o in the U-235 isotope.

(iii) Greater than Type A quantity of radioactive material which may include uranium enriched in the U-235 isotope, U-233, plutonium, as metal, oxides, or compounds which are thermally stable up to 600°F. Plutonium in excess of twenty (20) curies per package must be in the form of metal, metal alloy, or reactor elements.

(iv) Greater than Type A quantity of byproduct material meeting the requirements of special form radioactive material.

(v) Greater than Type A quantity of byproduct material in normal form as metal, oxides, or compounds which are thermally stable up to 600°F.

(vi) Irradiated Triga-Type fuel assemblies described in Section 6.6 of the application (pp. 6-23 through 6-27).

(vii) Irradiated S8DR fuel elements 0.56-inch OD by 18.7 inches long by 0.010-inch wall thickness of Hastelloy-N. The fuel material is UZrH fully enriched in U-235.

(viii) Intact irradiated CP-5 fuel assemblies containing not more than 176 grams U-235 per assembly prior to irradiation. Uranium may be enriched to a maximum 93 w/o in the U-235 isotope. Active fuel length shall be 28.5 inches.

(ix) Solid nonfissile irradiated hardware which may contain encapsulated fission monitors.

(x) Irradiated uranium oxide waste enriched in the U-235 isotope up to a nominal 93 w/o which is thermally stable up to 800°F.

(xi) Irradiated uranium enriched in the U-235 isotope meeting the requirements of special form radioactive material.

(xii) Intact irradiated MURR fuel assemblies containing not more than 775 grams of U-235 per assembly prior to irradiation. Uranium may be enriched to a maximum 93.5 w/o in the U-235 isotope. Active fuel length shall be 24 inches.

5.(b) Contents (continued)

(1) Type and form of material (continued)

(xiii) Intact irradiated MITR-II fuel assemblies containing not more than a nominal 510 grams of U-235 per assembly prior to irradiation. Uranium may be enriched to a maximum 93.5 w/o in the U-235 isotope. Active fuel length shall be approximately 24 inches.

(xiv) Intact irradiated High Flux Beam Reactor (HFBR) fuel assemblies containing not more than a nominal 351 grams of U-235 per assembly prior to irradiation. Uranium may be enriched to a maximum of 93.5 w/o in the U-235 isotope. Active fuel length shall be nominal 24 inches.

(xv) Intact irradiated MTR-type fuel assemblies containing not more than 240 grams U-235 per assembly prior to irradiation. Uranium may be enriched to a maximum 93.5 w/o in the U-235 isotope. Active fuel length shall be approximately 25 inches.

(xvi) Irradiated MTR-type fuel sections containing not more than 176 grams U-235 per fuel section prior to irradiation. Uranium may be enriched to a maximum 93.5 w/o in the U-235 isotope. Active fuel length per fuel section shall be approximately 11 inches. The fuel assembly shall be sectioned only in the non-fuel bearing regions of the assembly.

(xvii) Intact irradiated MTR-type fuel assemblies containing not more than 282.7 grams U-235 per assembly prior to irradiation. Uranium may be enriched to a maximum 20 w/o in the U-235 isotope. Active fuel length shall be approximately 25 inches.

(2) Maximum quantity of material per package

The minimum cooling time of each fuel assembly and rod is 90 days, maximum decay heat generation per package not to exceed 1.5 kW, and the external dose rate not to exceed 10 mrem/hr 3 feet from the external surface of the cask and:

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

Twenty-four (24) fuel assemblies as contained in product containers specified in 5(a)(4)(ii) or 12 fuel assemblies as contained in product containers specified in 5(a)(4)(v).

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

One (1) fuel assembly as contained in product container specified in 5(a)(4)(iii).

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

480 grams U-233 or 480 grams Pu-239 or 800 grams U-235 as contained in product container specified in 5(a)(4)(i).

5.(b) Contents (continued)

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

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

Gamma sources securely confined in the cask cavity to preclude secondary impacts during accident conditions of transport. Thermal heat generation rate is limited to 200 watts.

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

Contained in product containers specified in 5(a)(4)(i) and limited to 200 thermal watts.

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

Thirty-eight (38) fuel assemblies as contained in product containers specified in 5(a)(4)(iv). Fuel assemblies with an initial enrichment (U-235 in U) of greater than 70 w/o U-235 are limited to 19 assemblies per product container. Shipments of less than 19 assemblies with a U-235 enrichment greater than 70 w/o may be combined with assemblies of 70 w/o U-235 or less provided: $x/38 + y/19 \leq 1$; x = no. assy's ≤ 70 w/o U-235, y = no. assy's > 70 w/o U-235.

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

Twenty-four (24) fuel elements per can and six sealed cans per basket as described in 5(a)(4)(vi). Each of the six cans may contain up to 818 g U-235 and 158 g hydrogen. The cask is limited to 4.908 kg U-235.

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

Twelve (12) fuel assemblies.

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

Thermal heat generation rate is limited to 200 watts.

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

Twenty-four (24) containers each limited to 352 grams U-235 as contained in product containers specified in 5(a)(4)(vii). The decay heat per container is limited to 20 watts. The containers must be leak tested in accordance with Union Carbide Corporation letter dated November 17, 1980.

5.(b) Contents (continued)

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

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

Twenty-four (24) capsules each limited to 100 grams U-235.

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

Eight (8) fuel assemblies as contained in the product container specified in 5(a)(4)(viii). The maximum burnup is 150 MWD/Assembly and the minimum cooling time of each fuel assembly is 150 days. The maximum radiation source term is 400,000 curies.

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

Eight (8) fuel assemblies, contained in the product container specified in 5(a)(4)(viii). The maximum decay heat per package is 200 watts.

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

Twenty (20) fuel assemblies contained in two baskets separated by a spacer plate as specified in 5(a)(4)(ix). Each shipment must contain twenty fuel assemblies. The maximum burnup is approximately 130 MWD/assembly, and the minimum cooling time is 470 days.

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

Twelve (12) fuel assemblies contained in product container specified in 5(a)(4)(v).

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

Forty (40) fuel sections contained in the product container specified in 5(a)(4)(x). When a shipment contains less than the maximum number of fuel sections (40), empty fuel section basket spaces must be provided with an aluminum or steel spacer in the form of an open-ended pipe with a minimum outer diameter of 2.5 inches and a minimum wall thickness of 0.125 inches. The spacer must be of sufficient length to replace the absent fuel sections.

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

Eight (8) fuel assemblies contained in the peripheral locations of the basket specified in 5(a)(4)(v). The maximum burnup is 14%, the maximum decay heat is 15 watts per fuel assembly, and the minimum cool time is 120 days. Four aluminum inserts, as shown in Lockheed Martin Drawing No. 507584, Rev. 1, must be positioned in each of the four center basket locations.

5.(c) Criticality Safety Index

(1) For the contents described in 5(b)(1)(iii) and 5(b)(1)(xv), and limited in 5(b)(2)(iii) and 5(b)(2)(xv): 0.4

(2) For the contents described in 5(b)(1)(i), 5(b)(1)(ii), 5(b)(1)(vi), 5(b)(1)(vii), 5(b)(1)(viii), 5(b)(1)(x), 5(b)(1)(xi), 5(b)(1)(xii), 5(b)(1)(xiii), 5(b)(1)(xiv), 5(b)(1)(xvi), and 5(b)(1)(xvii), and limited in 5(b)(2)(i), 5(b)(2)(ii), 5(b)(2)(vi), 5(b)(2)(vii), 5(b)(2)(viii), 5(b)(2)(x), 5(b)(2)(xi), 5(b)(2)(xii), 5(b)(2)(xiii), 5(b)(2)(xiv), 5(b)(2)(xvi), and 5(b)(2)(xvii): 100

6. For Item 5.(b)(1)(iii), mixtures of fissile material are authorized, provided the following equation is satisfied:

$$\frac{X}{480} + \frac{Y}{480} + \frac{Z}{800} \leq 1, \text{ where:}$$

X = Grams U-233 to be shipped
Y = Grams Pu-239 to be shipped
Z = Grams U-235 to be shipped

7. Except for the contents described in 5(b)(1)(ii), 5(b)(1)(iv) and 5(b)(1)(xii); and limited in 5(b)(2)(ii), 5(b)(2)(iv) and 5(b)(2)(xii), the cask must be shipped dry.

8. If the cask contents of 5(b)(1)(ii), 5(b)(1)(iv) or 5(b)(1)(xii) are shipped wet, the licensee must confirm that the pressure relief valve is operable (set pressure ≤ 75 psig). When needed, sufficient antifreeze in the cask must be used to prevent damage of any component of the package by freezing.

9. Loading and unloading operations of the contents described in 5(b)(1)(iii) and limited in 5(b)(2)(iii) must preclude contact of water with the contents.

10. When the contents of 5(b)(1)(vi) are loaded wet, the optional 0.5-inch diameter drain hole must be present in the primary basket lower plate to assure proper draining of the basket.

11. The presence and effectiveness of the Boral poison plate in the Basket Assemblies as shown in BMI Drawing Nos. BCL-000-500, Rev, A; 0048, Rev. A; and 00-000-236, Rev. C, must be verified by neutron measurements prior to first use and records maintained of such verification. Verification of the presence of the Boral must be made in each subsequent use.

12. Contents 5(b)(1)(i) and 5(b)(1)(x) may be mixed provided the sum of the product containers and fuel assemblies does not exceed 24.

13. Axial movement of fuel assemblies must be limited so that the active fuel region will remain correctly positioned with respect to the poisoned section of the basket. Removable spacers may be used in each section of the basket to limit axial movement of the assemblies.

14. Contents must be securely confined in the cask cavity to minimize movement.

15. Prior to each use, adequacy of containment vessel must be demonstrated by performance of the leak test described in Section 7.1.1.1 of the application.

16. Gaskets and seals (cask and fuel canister) must be replaced at least every 12 months or earlier if visible degradation occurs.

17. For contents described in 5(b)(1)(iii) and limited in 5(b)(2)(iii), the mass of fissile material contained in reactor fuel must be based on the mass prior to irradiation.

18. *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 of Chapter 7 of the application. Additionally, for the contents described in 5(b)(1)(xvii), the package must be prepared for shipment in accordance with the procedures specified in the supplement dated January 29, 1999.

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

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

20. Revision No. 27 of this certificate may be used until March 31, 2007.

21. Expiration date: October 1, 2008.

REFERENCES

Department of Energy application dated: April 18, 1995

Department of Energy supplements dated: November 20, 1995, September 4, 1998, January 29 and April 20, 1999, December 13, 2000, and February 16, 2006.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Robert A. Nelson, Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: March 9, 2006

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
5979	11	71-5979	USA/5979/B( )	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)*

Alpha-Omega Services, Inc.
9156 Rose Street
Bellflower, CA 90706

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Alpha-Omega Services, Inc. application dated June 1980, 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.: 5979

(2) Description

A shipping container for teletherapy cobalt sources. Configuration of the outer container is box-like measuring approximately 38" x 50" x 40". The box is lined with 4.5" of plywood with a 0.125" outer steel shell welded to an exterior angle framework. Transverse strips across the bottom facilitate use of a fork-lift and lifting lugs are provided at the four top corners. The inner shield vessel is essentially a 24" diameter, lead-filled, barrel-shaped configuration. Three different cylindrical plug inserts and bolted end caps provide flexibility to accommodate several sizes and shapes of sources. Gross weight is approximately 5,000 lbs.

(3) Drawings

The packaging is constructed in accordance with Alpha-Omega Services, Inc., Drawing Nos.: 0090, Rev. A; 0091, Rev. A; 0092, Rev. 1; and 0093, Rev. 0.

(b) Contents

(1) Type and form of material

Cobalt 60 or cesium 137 as sealed sources which meet the requirements of special radioactive material.

(2) Maximum quantity of material per package

13,000 curies Co-60 or 3,000 Cs-137, with decay heat load not to exceed 200 watts.

6. Lifting eyes shall be covered or blocked to prevent use as tie-down attachments.

7. The shield vessel closures shall be equipped with gaskets.

8. Bolts used to secure the shield vessel closure caps shall be secured against loosening by vibration during transport.

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

   a) Each package must meet the Maintenance Inspection Program of the supplement dated January 27, 2005; and

   b) The package must be prepared for shipment in accordance with the Operating Procedures of the supplement dated January 27, 2005.

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

11. Expiration date: October 1, 2008

REFERENCES

Alpha-Omega Services, Inc., application dated June 1980.

Supplements dated: April 12, 1983; May 22 and August 20, 1990; January 30, and November 16, 1995; July 5, 2000; October 26, and November 15, 2004, and January 27, 2005.

UNITED STATES NUCLEAR REGULATORY COMMISSION

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: March 9, 2005

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
5984	7	71-5984	USA/5984/B( )	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)*

J. L. Shepherd and Associates
1010 Arroyo Avenue
San Fernando, CA 91340-8095

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

J. L. Shepherd and Associates applications dated September 12, 1974; and April 26, 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.: 5984

(2) Description

A protective overpack which provides impact resistance, and thermal resistance for its contents which are contained within a single snug-fitting shielded inner container. The overpack consists of a vented-steel jacketed, laminated plywood outer container. Dimensions of the overpack are approximately 28" in diameter by 43" high and the plywood thickness is approximately 4" on the sides and 6" on the top and bottom. The total weight including weight of the contents is approximately 1,780 pounds.

(3) Drawings

The overpack is constructed in accordance with J. L. Shepherd and Associates Drawing Nos. A-0068-2C-1 dated March 8, 1969; and A-0068-2C dated April 26, 1995.

The inner shielded containers are constructed in accordance with J. L. Shepherd and Associates Drawing Nos. A-0068-1B, Rev. 2, or A-0068-1B-B, dated April 26, 1995, or A-0068-1B-A, dated April 26, 1995. The special form source capsule is constructed in accordance with J. L. Shepherd and Associates Drawing No. A-0068-10 dated January 30, 1969.

(b) Contents

(1) Type and form of material

Cesium 137 as cesium chloride sources doubly encapsulated in stainless steel tubes which meet the requirements of special form radioactive material.

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5 (b) Contents continued

(2) Maximum quantity of material per package

12,000 curies.

6. Use of packaging fabricated after August 31, 1986, is not authorized.

7. 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 "Inspection Operation, Handling and Maintenance Procedures" in the J. L. Shepherd and Associates submittal dated May 1, 1995.

b. The package must meet the "Acceptance Tests" and "Checkout and Maintenance Procedures" in the J. L. Shepherd and Associates submittal dated February 20, 1990.

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

Expiration date: August 31, 2007.

REFERENCES

J. L. Shepherd and Associates' applications dated September 12, 1974; and April 26, 2001.

Supplements dated: January 20, 1975; February 20, 1990; February 6, and May 1, 1995;April 11, 1996; and June 8, 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 12, 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
6003	20	71-6003	USA/6003/B( )F	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)*
U.S. Department of Energy
Division of Naval Reactors
Washington, DC 20585

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION
Safety Analysis Report for M-130 shipping container dated December 30, 1968, 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.: M-130

(2) Description

The Model No. M-130 shipping container is an upright cylinder 84 inches in diameter by 158 inches overall height. The container walls consist of a finned 1-inch thick outer shell fabricated from either carbon steel, carbon steel with stainless steel clad, or solid stainless steel, 10 inches of lead shielding, and a 1-inch thick inner pressure vessel fabricated from carbon steel clad with stainless steel. The top of the container is covered with a shielded closure head which is bolted to the container and seals the pressure vessel. An access opening with a bolted shield plug is provided in the closure head for loading and unloading spent fuel.

The pressure vessel has an inside diameter of 55 inches. The central region contains a secondary heat exchanger (not used during shipment) surrounded by 1/2-inch thick carbon steel backup cylinder 29 inches in diameter. The annulus which remains between the backup cylinder and the pressure vessel provides a space 13-inches wide and 130-inches high for spent fuel. The spent fuel is contained in the annulus by module holders designed for the particular core to be shipped.

The container has external penetrations to the pressure vessel for steam and water relief lines and a fill and drain line (which are capped during shipment) and a pressure sensing line which remains open to a pressure gage during shipment. The container also has penetrations which do not open to the pressure vessel for secondary heat exchanger lines (which are capped during shipment) and a temperature sensing line.

The container is supported on its transport vehicle by an "A" frame structure. Gross weight of the loaded container without its support structure is approximately 228,000 pounds.

5. (a) Packaging (continued)

(3) Drawings

The packaging is constructed in accordance with General Electric Drawing Nos. 247E209, Sheet 1, Rev. R; Sheet 2, Rev. K; Sheet 3, Rev. T; Sheet 4, Rev. U; Sheet 5 of 5, Rev. F, and 247E228, Rev. F.

(b) Contents

(1) Type and form of material

Irradiated fuel assemblies, activated corrosion products and structural parts containing up to 40 gallons of residual contaminated water. The fuel assemblies and structural parts are of the following types:

(i) Deleted.

(ii) Deleted.

(iii) Deleted.

(iv) D1G fuel modules of core types 1 or 2.

(v) D1G removable fuel assemblies of core types 1 or 2.

(vi) Deleted.

(vii) Deleted.

(viii) S3G-3/3A fuel module with or without control rods. The core age must be at least 4000 logging-corrected full-power hours.

(ix) Deleted.

(x) S3G-3/3A irradiated thermocouples and thermocouple cases.

(xi) S8G full size fuel cell with or without control rod.

(xii) S8G partial size fuel cell with or without control rod.

(xiii) Deleted.

(xiv) Deleted.

(xv) D2W fuel cells with control rods.

(xvi) NR-1 fuel modules with or without control rods.

5. (b) Contents (continued)

(xvii) Deleted.

(xviii) A1W-3 recoverable irradiated fuel modules. Fuel modules that use control rod shall have control rods inserted.

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

(i) Deleted.

(ii) Deleted.

(iii) 6 fuel assemblies as described in 5(b)(1)(iv) and 4 fuel assemblies as described in 5(b)(1)(v).

(iv) Deleted.

(v) 10 fuel assemblies as described in 5(b)(1)(viii).

(vi) 9 fuel assemblies as described in 5(b)(1)(viii).

(vii) 9 fuel assemblies as described in 5(b)(1)(viii) and 1 structure as described in 5(b)(1)(x).

(viii) 4 fuel cells as described in 5(b)(1)(xi) or 2 fuel cells as described in 5(b)(1)(xi) and 2 fuel cells as described in 5(b)(1)(xii).

(ix) Deleted.

(x) Deleted.

(xi) 4 fuel cells as described in 5(b)(1)(xv) plus 2 corner fuel cells or 1 RFA fuel cell.

(xii) 4 fuel modules as described in 5(b)(1)(xvi).

(xiii) Deleted.

(xiv) For contents described in 5(b)(1)(xviii), 6 fuel modules or 8 fuel modules, as described in supplement dated March 30, 1992.

(3) Shipments shall be further limited by thermal requirements as follows:

(i) Shipment of contents specified in 5(b)(1)(iv) and 5(b)(1)(v) and limited in 5(b)(2)(iii) shall be made no earlier than 75 days after shutdown and shall have a decay heat load not to exceed 33,500 Btu/hr per shipment.

(ii) Deleted.

5. (b) Contents (continued)

(iii) Shipment of contents specified in 5(b)(1)(viii), and 5(b)(1)(x) and limited in 5(b)(2)(v), 5(b)(2)(vi), and 5(b)(2)(vii) shall be made at a time after shutdown, as determined from Bettis Atomic Power Laboratory report WAPD-OP(PP)S-4401 dated June 29, 1979, and shall have a decay heat load not to exceed 28,620 Btu/hr for the shipboard core and 30,000 Btu/hr for the prototype core.

(iv) Deleted.

(v) Shipment of contents specified in 5(b)(1)(xi) or 5(b)(1)(xii), as limited by 5(b)(2)(vii), shall have a fully loaded container heat load not to exceed 15,400 Btu/hr per shipment.

(vi) Deleted.

(vii) Deleted.

(viii) Shipment of contents specified in 5(b)(1)(xv) and limited in 5(b)(2)(xi) shall have a heat load not to exceed 19,100 Btu/hr and shall be made no earlier than 420 days after shutdown.

(ix) Shipment of contents specified in 5(b)(1)(xvi) and limited in 5(b)(2)(xii) shall have a heat load not to exceed 6,000 Btu/hr and shall be made no earlier than 50 days after shutdown.

(x) Deleted.

(xi) Shipment of contents specified in 5(b)(1)(xviii) and limited in 5(b)(2)(xiv) shall have a heat load not to exceed 43,800 BTU/hr and shall be made no earlier than 400 days or 175 days for A1W-3E and A1W-3J fuel, after shutdown.

(c) Transport Index for Criticality Control

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

Except for the contents described in 5(b)(1)(iv) (Core 2), 5(b)(1)(v) (Core 2) and 5(b)(1)(viii) and limited in 5(b)(2)(iii) and 5(b)(2)(v)	100
For the contents described in 5(b)(1)(viii) and limited in 5(b)(2)(v)	25
For the contents described in 5(b)(1)(iv) (Core 2) and 5(b)(1)(v) (Core 2) and limited in 5(b)(2)(iii)	0

6. Deleted.

7. For shipments involving the contents specified in 5(b)(1)(viii) or 5(b)(1)(x), the thermocouples and thermocouple cases if included or the vacant module holder shall be located in the mid-position of either cage and module holder assembly.

8. Shipments shall be made in the dry condition, except for residual water as limited in 5(b)(1).

9. Container number three (M-130-3) has been modified by adding two 4-inch thick by 8-inch wide steel plates welded between fins 25 and 50 and between fins 110 and 135 at approximately 14.75 inches from the bottom of the container. The cooling fins in this localized area are removed to permit attachment of the plate directly to the outer shell of the container.

10. Container number four (M-130-4) has been modified by adding a 2-inch thick by 4-inch wide steel plate welded between fins 32 and 49 at approximately 18.4 inches from the bottom of the container. The cooling fins in this localized area are removed to permit attachment of the plate directly to the outer shell of the container.

11. Containers M-130-3, M-130-4, M-130-6, and M-130-7 may be used for the contents specified in 5(b)(1)(viii) and 5(b)(1)(x) only. Containers M-130-10 and M-130-15 may be used for the contents specified in 5(b)(1)(viii), 5(b)(1)(x), and 5(b)(1)(xviii) only.

12. Container M-130-11 may be used for the contents specified in 5(b)(1)(xvi) only.

13. Deleted.

14. Expiration date: September 30, 2007.

REFERENCES

Safety analysis report for M-130 shipping container, MAO-E8-703 dated December 30, 1968.

Supplements: Naval Reactors (NR) letters A#2256 dated February 24, and G#1931 dated March 3, 1969; General Electric Company (GE) letter ONP-74520-526 dated April 3, 1972; NR letter G#3207 dated April 27, 1972; GE letter ONP-74520-528 dated April 28, 1972; NR letter G#3250 dated June 6, 1972; GE letters ONP-74570-635 dated October 25, ONP-74570-654 dated December 4, and ONP-14570-666 dated December 12, 1972; ONP-74570-682 dated January 12, ONP-74570-698 dated January 31, ONP-74570-687 dated February 6, ONP-74390-65 dated March 26, and DLGN-85570-854 dated September 24, 1973; and DLGN-85570-901 dated January 10, 1974; NR letter G#4061 dated January 29, 1974; GE letters DLGN-85570-924 dated February 15, DLGN-85570-923 dated March 6, and DLGN-85570-969 dated May 24, 1974; NR letter G#4991 dated November 25, 1975; GE letters ONP-74340-JTT-73 dated December 17, 1975; CGN-85570-1145 dated September 9, CGN-85570-1146 dated September 10, and CGN-85570-1148 dated September 14, 1976; Bettis Atomic Power Laboratory letters WAPD-R(K)-1378 dated August 30, 1976, and WAPD-OP(PP)S-4401 dated June 29, 1979; NR letters G#6197 dated July 13, 1979, G#7022 dated July 14, WAPD-LP-(CES)SE-170 dated July 1981; and WAPD-LD-(CES)SE-181 dated September 1981; WAPD-LP(CES)SE-96 dated February 1982, G#7136 dated March 17, 1982; G#7160 dated May 18, 1982; G#7582 dated September 7, 1983; G#C87-5692 dated September 2, and G#C87-5689 dated September 23, 1987; G#C87-8008 dated January 19, G#C88-5931 dated May 12, and G#C88-5961 dated July 25, 1988; G#C89-2825 dated March 29, and G#C89-2863 dated August 11, 1989;

REFERENCES (continued)

G#C92-03392 dated March 30, and G#92-03729 dated October 20, 1992; G#C93-10935 dated October 8, 1993; G#96-03344 dated March 6, and G#96-03610 dated December 9, 1996; G#97-03543 dated July 10, G#C97-03685 dated December 19, 1997; and G#02-0754 dated 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

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
6078	31	71-6078	USA/6078/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
   P.O. Box 355
   Pittsburgh, PA 15230-0355

   b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

   Westinghouse Electric Company, LLC, application dated September 27, 2005.

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.: 927A1 and 927C1

(2) Description

A steel fuel bundle shipping container consisting of a strongback and fuel bundle clamping assembly, shock mounted to a steel outer container. The fuel bundles are separated by 3/16" thick, high carbon steel segmented separator blocks permanently attached to the strongback. The segmented separator blocks are 6" x 8" and are installed (welded) in segments to form a continuous block for the entire active length of the fuel assembly. The Model No. 927A1 package is approximately 43" in diameter by 189" long with an approximate gross weight of 6,700 lbs. The Model No. 927C1 package is approximately 43" in diameter by 216" long with an approximate gross weight of 7,300 lbs.

(3) Drawings

The Model Nos. 927A1 and 927C1 containers are constructed in accordance with Combustion Engineering, Inc. Drawing No. L-6078-01, Sheets 1 through 4, Rev. 5.

(b) Contents

(1) Type and form of material

(i) Model No. 927A1: unirradiated fuel bundles consisting of 0.38" diameter uranium dioxide fuel pellets clad in 0.028" thick zircaloy tubes in a 14 x 14 square array with a 0.58" pitch. Each fuel bundle consists of a maximum of 176 fuel rods with a maximum 5.0 w/o enrichment in the U-235 isotope, and contains not more than 19.6 kg U-235.

5.(b) (1) Contents (Continued)

(ii) Model No. 927A1: unirradiated fuel bundles consisting of 0.381" diameter uranium dioxide fuel pellets clad in 0.026" thick zircaloy tubes in a 14 x 14 square array with a 0.58" pitch. Each fuel bundle consists of a maximum of 176 fuel rods with a maximum 5.0 w/o enrichment in the U-235 isotope, and contains not more than 20.5 kg U-235. The fuel assembly may contain rods that have annular pellet zones at the top and bottom whose lengths shall not exceed 12". The annular pellets have the same parameters as the solid pellets, except that the inner diameter shall not exceed 0.1905".

(iii) Model No. 927A1: unirradiated fuel bundles consisting of 0.325" or 0.3255" diameter uranium dioxide fuel pellets clad in 0.025" thick zircaloy tubes in a 16 x 16 square array with a 0.506" pitch. Each fuel bundle consists of a maximum of 236 fuel rods with a maximum 5.0 w/o enrichment in the U-235 isotope, and contains not more than 20.76 kg U-235. The fuel assembly may contain rods that have annular pellet zones at the top and bottom whose lengths shall not exceed 12". The annular pellets have the same parameters as the solid pellets, except that the inner diameter shall not exceed 0.1625".

(iv) Model No. 927A1: unirradiated fuel bundles consisting of 0.31" diameter uranium dioxide fuel pellets clad in 0.024" thick zircaloy tubes in a 16 x 16 square array with a 0.472" pitch. Each fuel bundle consists of a maximum of 231 fuel rods with a maximum 5.0 w/o enrichment in the U-235 isotope, and contains not more than 11.68 kg U-235.

(v) Model No. 927C1: unirradiated fuel bundles consisting of 0.325" or 0.3255" diameter uranium dioxide pellets clad in 0.025" thick zircaloy tubes in a 16 x 16 square array with a 0.506" pitch. Each fuel bundle consists of a maximum of 236 fuel rods with a maximum 5.0 w/o enrichment in the U-235 isotope, and contains not more than 22.77 kg U-235. The fuel assembly may contain rods that have annular pellet zones at the top and bottom whose lengths shall not exceed 12". The annular pellets have the same parameters as the solid pellets, except that the inner diameter shall not exceed 0.1625".

(vi) Model No. 927C1: unirradiated fuel bundles consisting of 0.324" diameter uranium dioxide fuel pellets clad in 0.0235" thick zircaloy tubes in a 17 x 17 square array with a 0.501" pitch. Each fuel bundle consists of 264 fuel rods with a maximum 3.6 w/o enrichment in the U-235 isotope, and contains not more than 16.43 kg U-235.

(vii) Model No. 927C1: unirradiated fuel bundles consisting of 0.3225" diameter uranium dioxide pellets clad in 0.0225" thick zircaloy tubes in a 16 x 16 square array with a 0.506" pitch. Each fuel bundle consists of a maximum of 236 fuel rods with a maximum 5.0 w/o enrichment in the U-235 isotope, and contains not more than 22.0 kg U-235. The fuel assembly may contain rods that have annular pellet zones at the top and bottom whose lengths shall not exceed 12". The annular pellets have the same parameters as the solid pellets, except that the inner diameter shall not exceed 0.155".

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

Model No. 927A1: Two fuel bundles weighing not more than 1400 lbs. each.

Model No. 927C1: Two fuel bundles weighing not more than 1506 lbs. each.

(c) Criticality Safety Index: 15.7

6. Each fuel assembly shall be unsheathed or shall be enclosed in an unsealed, polyethylene sheath which will not extend beyond the ends of the fuel assembly. The ends of the sheath shall not be folded or taped in any manner that would prevent flow of liquids into or out of the sheathed fuel assembly.

7. 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 of Chapter 7 of the application.

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

8. Fabrication of additional packagings is not authorized.

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

10. Expiration date: October 1, 2008. This certificate is not renewable.

11. Revision No. 30 of this certificate may be used until October 1, 2006.

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
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<u>REFERENCES</u>

Westinghouse Electric Company, LLC, consolidated application dated September 27, 2005.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Robert A. Nelson, Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: 10/24/05

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
6206	31	71-6206	USA/6206/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 April 23, 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.: Model B

(2) Description

A fuel assembly shipping container consisting of a steel strongback clamping assembly, shock mounted to a steel outer container. Two 3/16-inch thick, 8-5/8-inch high and full length stainless steel plates containing 1.5% minimum boron are positioned between adjacent fuel assemblies. The outer container is approximately 40 inches in diameter by 200 inches long. Gross weight of the loaded container not to exceed 7,600 pounds.

(3) Drawings

The container is constructed in accordance with Framatome Cogema Fuels Drawing Nos. 1273422, Rev. 0; 1273423, Rev. 0; 1273424, Rev. 0; 1273425, Rev. 0; 1273426, Rev. 0; and 1273427, Rev. 0.

(b) Contents

(1) Type and form of material

Unirradiated, sintered UO_2 pellets in fuel rods. The maximum inner diameter and the minimum outer diameter of the fuel rod cladding, guide tubes and instrument tubes are in accordance with Table 3 of B&W Fuel Company supplement dated October 27, 1995; and the minimum guide tube outer diameter and minimum wall thickness are in accordance with Framatome Cogema Fuels supplement dated February 7, 1996. The locations of the guide tubes and instrument tubes are in accordance with Figures 2 through 5 of B&W Fuel Company supplement dated October 27, 1995. The rods are assembled into fuel assemblies. The fuel assemblies may contain Special Absorber Rods as described in Tables 6.4.2 and 6.4.3 of Framatome ANP's supplement dated October 15, 2002.

5. (b)(1) Contents (continued)

Fuel assemblies as described above have the following specifications:

Assembly Type	15x15	15x15	15x15	17x17	17x17	15x15
No. fuel rods	208	208	208	264	264	204
No. non-fuel tubes	17	17	17	25	25	21
Fuel rod pitch, in.	0.568	0.568	0.568	0.496	0.502	0.563
Maximum fuel pellet OD, in.	0.3707	0.3742	0.3622	0.3232	0.3252	0.3671
Tube material	zirconium alloy	zirconium alloy	zirconium alloy	zirconium alloy	zirconium alloy	zirconium alloy
Maximum active fuel length, in.	144	144	144	145.825	144	144
Maximum enrichment w/o U-235	5.05	5.05	4.98	5.05	5.05	5.05
Maximum U-235 Loading (kg)	25.1978	25.6758	23.7220	24.3108	24.6126	24.2355

(2) Maximum quantity of material per package

Two fuel assemblies. Total quantity of radioactive material within a package may not exceed a Type A quantity.

(c) Criticality Safety Index 6.3

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 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.

7. There must be a bow clamp to restrain each spacer grid and end fitting. The ratio of assembly weight to the number of clamp bows must not exceed 168 pounds per clamp.

8. The weight of the contents (fuel assemblies, control rods, spacers, etc.) must not exceed 3,360 pounds.

9. Fabrication of additional packagings is not authorized.

10. In addition to the requirements of Subpart G of 10 CFR Part 71, the package shall be operated and maintained 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 71.17.

12. Expiration date: October 1, 2008. This package is not renewable.

REFERENCES

B&W Fuel Company application dated April 23, 1990.

Supplements dated: July 23, 1990; May 4, August 18, August 25, and October 14, 1992; September 24, 1993; and April 8, May 2, and November 23, 1994; February 26, March 17, April 7, July 31, October 27, and December 1, 1995.

Framatome Cogema Fuels supplements dated February 7, 1996; January 20, March 19 and 26, and [illegible] 17, 1998; and August 29, September 8, and November 13, 2000; February 9, 2001; October 15, 2002; May 2 and July 25, 2003; and April 28, 2005.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Robert J. Lewis, Chief
Spent Fuel Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: 02 Sept 2005

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
6347	10	71-6347	USA/6347/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)*

General Atomics
P.O. Box 85608
3550 General Atomics Court
San Diego, CA 92186

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

General Atomics Company Application dated February 19, 1982, 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.: FSV-3

(2) Description

Inner container is a 18.5" ID x 34" high, 18-gage steel drum. Inner container is centered and supported in a 22.5" ID x 38.25" high, 16-gage steel drum. Void spaces between the inner and outer container and within the inner container are filled with vermiculite. Total weight, including contents, is 500 pounds.

(3) Drawing

The packaging is constructed in accordance with General Atomics Company Drawing No. FFE-613, Issue D.

(b) Contents

(1) Type and form of material

Unirradiated fuel element consisting of a graphite body, hexagonal in transverse cross-section, approximately 14.2" across the flats and 31.2" high. Dispersed in columns within the fuel element body there is a maximum 1.41 kg U-235 plus U-238 and Th-232. The U-235: U-238: Th-232 atomic ratio is about 1:0.07:8.3. The atomic ratio of carbon to the U-235 is in the range of 1800 to 1.

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

One fuel element containing not more than 1.41 kg U-235 and weighing not more than 320 pounds. Total 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: 1.3

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

(i) The package must be operated and prepared for shipment in accordance with the operating procedures of Chapter 6 of the application.

(ii) Each packaging must meet the Acceptance Tests and Maintenance Program of Chapter 7 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: September 30, 2007.

REFERENCE

General Atomics Company application dated February 19, 1982.

Supplements dated: March 9, 1982; February 24, 1992; February 28, 1997; and April 30, 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: September 20, 2002

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

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2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

BWX Technologies, Inc.
P.O. Box 785
Lynchburg, VA 24505

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Babcock & Wilcox Company application dated March 29, 2001.

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-10

(2) Description

The packaging consists of a containment vessel 5-9/16 inches OD by 22-3/8 inches high, constructed from a 5-inch schedule 40 steel pipe with a screw-type cap and a welded bottom plate. The containment vessel is centered and supported in a 55-gallon DOT specification 17C or 6C steel drum by industrial cane fiberboard.

The nominal gross weight of the packaging and contents is 350 pounds.

(3) Drawing

The packaging is constructed in accordance with Babcock and Wilcox Fuel Company Drawing No. 1198767E.

(b) Contents

(1) Type and form of material

Uranium metal, alloys or compounds. Uranium may be enriched to any degree in the U-235 isotope.

(2) Maximum quantity of material per package

Contents shall not exceed 100 pounds, and the U-235 content shall not exceed 350 grams. Maximum quantity of radioactive material within the package may not exceed a Type A quantity.

NRC FORM 618 (8-2000) 10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
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5. (c) *Criticality Safety Index:* 2.1

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

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

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

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

8. Expiration date: October 1, 2008. This certificate is not renewable.

REFERENCES

Babcock & Wilcox application dated March 29, 2001.

BWXT supplement dated March 29, 2006.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Robert A. Nelson, Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

UNITED STATES NUCLEAR REGULATORY COMMISSION

Date: Apr. 25, 2006

NRC FORM 618 (8-2000) 10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

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6386	17	71-6386	USA/6386/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)*

U.S. Department of Energy
Division of Naval Reactors
Washington, DC 20585

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Safety Analysis Report for 235R001 Shipping Container dated August 11, 1970, 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.: 235R001

(2) Description

The 235R001 shipping container structure is horizontal, having an oblong cross section and is fabricated from 0.104-inch thick carbon sheet steel. The container is 313 inches long and has a maximum weight of 4,640 pounds empty. The oblong cross section dimensions are approximately 35.5 inches high by 33.0 inches wide. The container was originally designed to ship unirradiated fuel modules of the AIG/A4W type. Subsequently, the container has been adapted to ship standard size or partial S8G fuel modules by use of a special frame assembly and cradle clamps, S3G-3 refueling modules using cell support assemblies, rodded or unrodded DIG fuel modules, and rodded or unrodded D2W fuel cells. The loaded container maximum weight is 12,200 pounds.

(3) Drawings

The packaging is constructed in accordance with Container Research Corporation Drawing Nos. 235R001, Rev. C, 235R004, Rev. C, and 235R005, Rev. 0, and Westinghouse Electric Corporation Drawing Nos. 973D425, Rev. 1, 903E693, Rev. 3, Sheet 1, 2 and 3 of 3, and 947J076, Rev. 0.

5.(b) Contents

(1) Type and form of material

Unirradiated fuel assemblies of the following types:

(i) A1G reactor cell without upper mechanism and with control rod, leadscrew and shipping fixture installed on rodded type modules.

(ii) Standard size S8G reactor cluster with regular or substitute support adapters and regular control rods. If only one cell is shipped per container, a dummy load shall be installed for balance.

(iii) Partial size S8G reactor cluster with regular or substitute support adapters and regular control rods. If only one cell is shipped per container, a dummy load shall be installed for balance.

(iv) S3G-3 refueling cells, with a maximum of one 0-1 reactor cell assembly per container.

(v) D1G fuel module, rodded.

(vi) D1G removable fuel assembly (RFA), unrodded.

(vii) A1G fuel cluster, fueled end only of full A1G reactor cell, rodded. Shipping poison rods are installed and are constructed in accordance with Westinghouse Electric Corporation Drawing Nos. 928E011, Rev. E or 1588E41, Sheet 1, Rev. J, and Sheet 2, Rev. C.

(viii) D2W side or central fuel cells with control rod and control rod holddown device.

(ix) D2W corner fuel cells, without shear blocks, unrodded.

(x) D2W side or central fuel cell and shear block with control rod inserted in rodded fuel cell.

(xi) D2W corner fuel cell, with shear block, unrodded.

(2) Maximum quantity of material per package

(i) One fuel assembly as described in 5(b)(1)(i), 5(b)(1)(x), or 5(b)(1)(xi).

(ii) Two fuel assemblies as described in 5(b)(1)(ii), 5(b)(1)(iii), 5(b)(1)(iv), 5(b)(1)(v), 5(b)(1)(vii), 5(b)(1)(viii), 5(b)(1)(ix).

(iii) Four fuel assemblies as described in 5(b)(1)(vi).

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5.(c) Criticality Safety Index

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

(1) For the contents described in 5(b)(1)(vii), 5(b)(1)(viii), 5(b)(1)(ix), and limited in 5(b)(2)(ii).: 50.0

(2) For contents described in 5(b)(1)(i), 5(b)(1)(ii), 5(b)(1)(iii), 5(b)(1)(iv), 5(b)(1)(v), 5(b)(1)(vi), 5(b)(1)(x), and 5(b)(1)(xi) and limited in 5(b)(2)(i), 5(b)(2)(ii) and 5(b)(2)(iii): 25.0

6. Expiration date: April 30, 2010.

REFERENCES

Safety Analysis Report for 235R001 Shipping Container, WAPD-OP(R)RD-357, dated August 11, 1970.

Supplements: Knolls Atomic Power Laboratory letter AIG 25-159, dated October 2, 1970. Bettis Atomic Power Laboratory letters WAPD-0P(R)RD-444, dated October 9, 1970; WAPD-OP(R)RD-476, dated October 26, 1970; and WAPD-OP(R)RD-488, dated October 30, 1970. Knolls Atomic Power Laboratory letters AIG 25-181, dated April 9, 1971; and AIG 25-191, dated May 11, 1971. Bettis Atomic Power Laboratory letters WAPD-OP(R)C-94, dated May 16, 1972; WAPD-OP(R)C-199, dated December 13, 1972; and WAPD-OP(R)C-229, dated March 6, 1973. Naval Reactors letters G#5078, dated January 26, 1976; G#5776, dated September 8, 1977; G#5905, dated January 23, 1978; G#5923, dated February 22, 1978; G#6095, dated August 17, 1978; G#6208, dated March 8, 1979; G#6373, dated September 4, 1979; G#68l3, dated October 17, 1980; G#C85-0467, dated July 17, 1985; G#C88-8112, dated October 18, 1988; G#90-03655, dated August 10, 1990; G#92-03560, dated June 15, 1992; G#96-03371, dated March 15, 1996, G#C97-03444 dated April 8, 1997; G#C99-03514, dated June 1, 1999; G#C99-03688, dated December 30, 1999; G#C02-0750, dated April 8, 2002; G#C03-00273, dated January 24, 2003; and G#C03-01695, dated July 14, 2003.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

James R. Hall
for

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

Date: March 25, 2005

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE	PAGES
6400	27	71-6400	USA/6400/B( )F	1 OF	9

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 15230-0355

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Westinghouse Electric Corporation application dated August 7, 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.: 6400

(2) Description

A protective overpack which provides impact and thermal protection for its contents. The inner shell (cavity) is approximately 76" x 76" x 172" constructed of 3/16" thick and 10-gauge mild steel. Closure of the cavity is by a 1/4" thick aluminum plate with silicone rubber gasket which is bolted to the main inner shell. The cavity is centered and supported in an outer 3/16" thick steel jacket by approximately 32" of polyurethane foam insulation at the end and 10" on the sides. A removable section or cap consisting of approximately 34" of polyurethane foam insulation encased in steel with a silicone rubber gasket is bolted to the main outer steel jacket. The overall dimensions of the package are approximately 8' x 8' x 20'. Vent holes are provided on the sides and ends of the container. Set into each corner of the outer container are standard I.S.O. steel castings. The total weight including weight of the contents is 45,000 pounds.

(3) Drawings

Packaging is constructed in accordance with one of the following sets of drawings: (1) Protective Packaging, Inc, Drawing Nos. 32106, Sheet 1, Rev. F and 32106, Sheet 2, Rev. 0; or (2) Westinghouse Electric Corporation Drawing No. 2020D08, Sheet 1 and 2, Rev. 0; or (3) Babcock and Wilcox Company Drawing No. 11-D-2130, Rev. 0; or (4) Protective Packaging, Inc., Drawing Nos. 32106-1, Sheet 1, Rev. F and 32106, Sheet 2, Rev. 0, as modified by Nuclear Packaging Inc. Drawing No. E.G.-60-01D, Sheets 1 and 2, Rev. 0; or (5) Protective Packaging, Inc. Drawing No. 32395, Sheets 1 through 9, Rev. B, as modified by Sandia Laboratories letter dated May 8, 1980; or (6) Lawrence Livermore National Laboratory Drawing Nos. AAA81-108683-00, Rev. 0 and AAA81-110194-00, Rev. 0.

5.(b) Contents

(1) Large, decontaminated equipment waste of such size as not to fit into a 55-gallon drum (with legs or other readily removable appendages removed). Not to exceed 200 grams plutonium within the package.

Equipment waste surfaces containing more than 0.5 Ci must be decontaminated to a smearable level of no more than 150,000 dpm/100 cm^2 prior to fixation or until successive decontamination cleaning operations do not reduce the smearable contamination levels by more than ten percent. After fixation, equipment waste surfaces must have a smearable level of contamination of no greater than 10,000 dpm/100 cm^2. Outer surfaces must have a smearable level of contamination of no greater than 20 dpm/100 cm^2. Prior to fixing of contamination, large equipment waste must be inspected to insure that: (a) all sharp or protruding objects have been removed, blunted or protected with packaging material, and (b) pipe caps, gasketed blind flanges, covers, etc., have been installed wherever possible. Following such inspection, the inner surfaces containing more than 0.5 Ci must be fixed with "strip" or "clear" coating. The inner surface(s) may alternatively be fixed with a polyurethane foam.

The large equipment waste must be enclosed in a tight-fitting, 1-inch thick plywood box constructed in accordance with Westinghouse Electric Corporation's Drawing No. 1620E43, Sheets 1, 2, 3, and 4, Rev. 3; a tight fitting 3/16" thick corrugated steel box constructed in accordance with Rockwell Hanford Operations' Drawing No. H-2-91888, Sheet 1, Rev. 0 (modified or unmodified); or enclosed in a tight fitting box constructed in accordance with General Electric Company Drawing Nos. 908E614, Rev. 1, and 908E619, Rev. 2 or 908E648, Rev. 0 or 908E649, Rev. 0; or enclosed in a tight fitting box constructed in accordance with Babcock and Wilcox Company Drawing No. LRC-70019 H, Rev. 2. The space between the equipment and the box must be filled with foam (1" minimum foam thickness) and between equipment (1/2" minimum foam thickness). Alternatively, gloveboxes contaminated and fixed as described above may be broken down as follows:

Glovebox windows are removed and separately packaged in 12-mil thick PVC bags and sealed. The inner bag is tape sealed and the outer bag is heat sealed.

Glovebox panels are cut to dimensions to fit inside the 3/16" thick corrugated steel burial crates constructed in accordance with Rockwell Hanford Operations' Drawing No. H-2-91888, Sheet 1, Rev. 0 (modified or unmodified). All sharp or protruding objects are removed, blunted, or protected with packaging material. The glovebox panels are bundled such that internal box surfaces are facing inward. Cut glovebox panels from not more than one glovebox are banded with metal strap banding such that two metal strap bands in each direction are placed around the length and width of the glovebox sections. The glovebox window and cut panel packages are enclosed and foamed in place within the box.

Blocking or dunnage is placed within the box to ensure a one inch foam barrier on the sides and bottom of the box. Likewise, dunnage is provided between the banded glovebox sections to maintain a 1/2" thick foam barrier between banded packages.

5.(b) Contents (continued)

(2) Decontaminated hard waste items, such as equipment, metal cans, tools, etc., must be double bagged within 12-mil thick PVC with each bag heat sealed. The total fissile quantity of all the sealed packages in one container must not exceed 200 grams.

Hard waste surfaces must be decontaminated to a smearable level of no more than 150,000 dpm/100 cm^2 prior to fixation or until successive decontamination cleaning operations do not reduce the smearable contamination levels by more than 10 percent. After fixation, hard waste surfaces must have a smearable level of contamination of no greater than 10,000 dpm/100 cm^2. Prior to fixing of contamination, hard waste must be inspected to insure that sharp or protruding objects have been removed, blunted, or protected with packaging material. Following such inspection, the outer surfaces must be fixed with "strip" or "clear" coating. Hard waste items such as furnace shells, muffles, or other items with large cavities not accessible for decontamination must be filled with foam within the cavities. Surfaces that are not easily accessible, e.g., interiors of small diameter tubing and piping which were in contact with process materials, must have been swabbed or immersed in cleaning solution to insure removal of residual material. Open ends of the tubing and piping must be sealed using mechanical fittings.

Alternately, large heavy walled process glassware must be painted inside and outside to fix contamination and double bagged in 12-mil thick PVC with each bag heat sealed. The glassware must be secured in a box constructed in accordance with General Electric Company Drawing No. 272E81-4, Rev. 0. The box must be filled with foam and total activity limited to less than two (2) Ci in a box.

Alternately, stainless steel transfer tubes and HEPA filters must be double bagged in 12-mil thick PVC with each bag heat sealed. The tubes/filters must be secured in a box constructed in accordance with General Electric Company Drawing No. 272E81-28, Rev. 0. The box must be filled with foam and total activity limited to less than 0.5 Ci in a box.

Alternately, round steel ducting must be capped and secured in a box constructed in accordance with General Electric Company Drawing No. 272E81-29, Rev. 0; 272E81-30, Rev. 0; or 272E81-31, Rev. 0. Outer surfaces ducting will have a smearable level of contamination no greater than 20 d/m/100 cm^2. The box must be filled with foam and total activity limited to less than 0.5 Ci in a box.

5.(b) Contents (continued)

Sealed packages and boxes of hard waste must be enclosed in a tight-fitting, 1-inch thick plywood box constructed in accordance with Westinghouse Electric Corporation's Drawing No. 1620E43, Sheets 1, 2, 3, and 4, Rev. 3; a tight-fitting 3/16" thick corrugated steel box constructed in accordance with Rockwell Hanford Operations' Drawing No. H-2-91888, Sheet 1, Rev. 0 (modified or unmodified); enclosed in a tight fitting box constructed in accordance with General Electric Company Drawing Nos. 908E614, Rev. 1 and 908E619, Rev. 2 or 908E648, Rev. 0 or 908E649, Rev. 0; or enclosed in a tight fitting box constructed in accordance with Babcock and Wilcox Company Drawing No. LRC-70019 H, Rev. 2. The space between the packages and the box must be filled with foam to a minimum thickness of 1 inch. Void spaces between the sealed packages must be filled with foam (1/2" minimum foam thickness).

(3) Glove box absolute (HEPA) filters must be double bagged within 12-mil thick PVC, with each bag heat sealed and packaged within DOT Specification 17H or 17C steel drums (maximum size of 55 gallons). Each drum must be lined with a sealed plastic liner and equipped with a standard drum closure. Each drum must not exceed a fissile quantity of 60 grams. Sealed drums must be enclosed in a tight-fitting 1-inch thick plywood box constructed in accordance with Westinghouse Electric Corporation's Drawing No. 1620E43, Sheets 1, 2, 3, and 4, Rev. 3; a tight-fitting 3/16" thick corrugated steel box constructed in accordance with Rockwell Hanford Operations' Drawing No. H-2-91888, Sheet 1, Rev. 0 (modified or unmodified); enclosed in a tight fitting box constructed in accordance with General Electric Company Drawing Nos. 908E614, Rev. 1 and 908E619, Rev. 2, or 908E648, Rev. 0, or 908E649, Rev 0; or enclosed in a tight fitting box constructed in accordance with Babcock and Wilcox Company Drawing No. LRC-70019 H, Rev. 2. The space between the drums and the box must be filled with foam to a minimum thickness of 1 inch. Void spaces between drums must be filled with foam (1/2" minimum foam thickness).

(4) Soft waste items such as sheeting, gloves, paper, prefilter media, polyethylene bottles, shoe covers, etc., must be double bagged in 12-mil thick PVC, with each bag heat sealed (bag size must not exceed 22" x 16" x 10") and packaged within DOT Specification 17H or 17C steel drums (maximum size of 55 gallons). Each drum must be lined with a sealed plastic liner and equipped with a standard drum closure. Each drum must not exceed a fissile quantity of 60 grams. Sealed drums must be enclosed in a tight-fitting 1-inch thick plywood box constructed in accordance with Westinghouse Electric Corporation's Drawing No. 1620E43, Sheets 1, 2, 3, and 4, Rev. 3; a tight-fitting 3/16" thick corrugated steel box constructed in accordance with Rockwell Hanford Operations' Drawing No. H-2-91888, Sheet 1, Rev. 0 (modified or unmodified); or enclosed in a tight fitting box constructed in accordance with Babcock and Wilcox Company Drawing No. LRC-70019 H, Rev. 2. The space between the drums and the box must be filled with foam to a minimum thickness of 1 inch. Void spaces between drums must be filled with foam (1/2" minimum foam thickness).

5.(b) Contents (continued)

(5) Liquid waste (decontamination solutions only) must be solidified in concrete in a 30-gallon drum which must be sealed in a plastic bag and centered and supported in a DOT Specification 17H or 17C 55-gallon steel drum by absorbent material. The 55-gallon drum must be lined with a sealed plastic liner and equipped with a standard drum closure. Each drum must not exceed a fissile quantity of 60 grams.

Alternatively, liquid waste is solidified in concrete in maximum size one (1) gallon packages which are double bagged and heat sealed in 12-mil thick PVC and placed with a DOT Specification 17H or 17C steel drum (maximum size of 55 gallons). The drum is lined with a sealed plastic liner and equipped with a standard drum closure. Each 55-gallon drum must not exceed a fissile quantity of 60 grams. For drums smaller than 55 gallons, the total fissile quantity of all the sealed packages (drums) in one container must not exceed 200 grams. Sealed drums must be enclosed in a tight-fitting 1-inch thick plywood box constructed in accordance with Westinghouse Electric Corporation's Drawing No. 1620E43, Sheets 1, 2, 3, and 4, Rev. 3; or a tight-fitting 3/16" thick corrugated steel box constructed in accordance with Rockwell Hanford Operations' Drawing No. H-2-91888, Sheet 1, Rev. 0 (modified or unmodified); enclosed in a tight-fitting box constructed in accordance with General Electric Company Drawing Nos. 908E614, Rev. 1 and 908E619, Rev. 2 or 908E648, Rev. 0 or 908E649, Rev. 0; or enclosed in a tight fitting box constructed in accordance with Babcock and Wilcox Company Drawing No. LRC-70019 H, Rev. 2. The space between the drums and the box must be filled with foam to a minimum thickness of 1 inch. Void spaces between drums must be filled with foam (1/2" minimum foam thickness).

(6) Uranium 233 oxide and thorium oxide in the form of intact LWBR-type fuel rods with the following limitations:

(i) Rods must be packaged within the Model No. 6400 packaging as described in Section 1 of WAPD-LP(FE)-220, Rev. 3 (February 1983);

(ii) The fuel content must not exceed 50 kg U-233 per shipment;

(iii) All rod storage containers must be filled to capacity (at least 70% of cross-sectional area) with rods or aluminum shim stock;

(iv) Each rod storage container must contain not more than one sub-container of 5/9 or 12 w/o BMU seed rods;

(v) Each rod storage container must weigh not more than 2,000 pounds;

(vi) The fuel rod heat generation must not exceed 30 watts; and

(vii) Operating Procedures and Acceptance Tests and Maintenance Program must be modified to meet the requirement of Item 11 of this approval.

5.(b) Contents (continued)

(7) Liquid analytical residues from the dissolution of spent reactor fuel rods, solidified in cement (see table, p. 3 of application*). The cement is contained in 1.5-gal steel can closed with a slip cover lid. The two primary cans are packed in a secondary steel can sealed with a press fit lid (see Figure 2 of application*). The secondary containment package contents are placed within a radiation shield (lid secured with six (6), 1/2"-13UNC bolts with welds in accordance with application*) centered in a DOT Specification 17-C 55-gal steel drum (see Figure 1 of application*). The drums are sealed with styrene-butadiene rubber gasket contained with a standard drum closer. Total weight of the drum will be less than 1,450 lb, and each drum will not exceed a fissile quantity of 12 g and 435 Ci of fission products.

Six (6), 55-gal sealed drum assemblies will be enclosed in a tight-fitting 3/16-in thick corrugated steel box constructed in accordance with Rockwell-Hanford Operations' Drawing No. H-2-91888, Sheet 1, Rev. 0 (modified or unmodified). The space between the drums and the box must be filled with foam to a minimum thickness of 1 inch. Void spaces between drums must be fitted with foam to a minimum thickness of 1/2 inch. Two (2) corrugated steel box assemblies may be transported in the packaging.

* U.S. Department of Energy letter dated April 15, 1983.

(8) Uranium 233 oxide and thorium oxide in the form of intact LWBR-type fuel rods with the following limitations:

(i) Rods must be packaged as shown in Figure 4, Application dated July 8, 1983, and contained within the Model No. NNFD-SA-2 packaging (Certificate of Compliance No. 5910);

(ii) The fuel content must not exceed 2.0 kg U-233 per shipment;

(iii) Each loaded LWBR Rod Transport Box must weigh not more than 99 pounds;

(iv) The fuel rod heat generation rate must not exceed 2 watts; and

(v) Operating Procedures and Acceptance Tests and Maintenance Program must be modified to meet the requirement of Item 11 of this approval.

5.(b) Contents (continued)

(9) Maximum of four (4) Cf-252 sources with the following limitations:

(i) Each source must be doubly encapsulated with the inner capsule meeting the requirements for special form radioactive material;

(ii) The total Cf-252 content must not exceed 6.1 mg;

(iii) The sources must be packaged in a shielded container as described in Chapter 1 of WAPD-LP(CE)POB-591 (January 1984); and

(iv) The decay heat generation from the source material must not exceed one watt.

(10) Compressed krypton-85 gas in mixture with other non-radioactive gases that are chemically compatible with the 3AA2015 cylinder. No fissile material (Requirement of 5.(c) does not apply). Shipment of krypton-85 gas is subject to the following limitations:

(i) Radioactivity not to exceed 2,700 curies. Maximum internal decay heat not to exceed 15 watts. Maximum volume of krypton-85 and other non-radioactive gases shall not exceed 1480 liters at STP (1 atm, 25°C);

(ii) The maximum initial fill pressure shall not exceed 500 psig at 25°C;

(iii) The DOT Specification 3AA2015 gas cylinder shall be certified for an operating load of 2,015 psig, at least once every 5 years by testing to 3,360 psig;

(iv) A minimum of 24 hours after loading with krypton-85 gas the krypton packaging primary containment shall have a leak rate of less than 0.0014 microcuries per second. The leak test shall be performed with the containment vessel within the lead shield container prior to placement within its thermal overpack;

(v) Content of the package shall be verified by mass spec analysis;

(vi) Acceptance, maintenance and use of the krypton package shall be in accordance with the procedures and requirements of Chapter 7 and 8 of Westinghouse Idaho Nuclear Company, Inc. Report No. WIN-236, Revision 1, March 1988. The retaining ring shall be tightened around the gas cylinder to a 40 to 50 inch-pound torque;

(vii) The position and securement of the krypton package within the Model No. 6400 is as specified in Westinghouse Idaho Nuclear Company, Inc. Drawing No. 059888;

(viii) Krypton package must be enclosed within a tight fitting plywood box constructed in accordance with Westinghouse Idaho Nuclear Company, Inc. Drawing No. 059886.

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 polyurethane foam must be Instapak 200, or equivalent.

7. The maximum weight of the contents including secondary packaging, dunnage, shoring and bracing must not exceed 30,000 pounds.

8. Sufficient dunnage, shoring and/or bracing must be utilized to minimize secondary impact of the secondary packaging within the cavity under accident conditions.

9. Protrusions from secondary packaging such as lifting eyes, etc., must be positioned such that they will not contact the cavity walls, or shoring must be provided to prevent puncture of the cavity walls by the protrusions under the accident conditions.

10. Contents must be positioned in the cavity such that the center of gravity of the loaded package is substantially the same as the center of gravity of an empty package.

11. The cavity of the overpack must be vented through an absolute filter to equalize pressure between the outside and inside of the overpack.

12. Contents packaged under the conditions of this certificate of compliance are exempt from the requirements of 10 CFR 71.63. This condition expires on October 1, 2004.

13. Condition 5(c) of this certificate of compliance is not applicable where the fissile material is excluded as provided by 10 CFR 71.53 until October 1, 2004, and 10 CFR 71.15, thereafter.

14. In addition to the requirements of Subpart G of 10 CFR Part 71, the package must be prepared for shipment, operated, and maintained in accordance with "Operating Inspection and Maintenance Procedure No. CSK-003, Rev. 0," included in the Westinghouse Electric Corporation supplement dated April 14, 1992.

15. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.12 until October 1, 2004, and 10 CFR 71.17, thereafter.

16. Expiration date: November 30, 2007.

REFERENCES

Westinghouse Electric Corporation application dated August 7, 1981.

General Electric Company supplement dated: October 1, 1981.

Babcock and Wilcox Company supplements dated: March 8, 1982; and January 10, 1985.

Department of Energy, Division of Naval Reactors, supplements dated: April 22, and July 8, 1983; and March 5, 1984.

Department of Energy, Chicago Operations Office, supplement dated: April 15, 1983.

Department of Energy, Washington, DC, supplement dated: June 6, 1988.

Westinghouse Electric Corporation supplements dated: April 14, 1992; and April 14, 1997.

Westinghouse Electric Company, Division of CBS Corporation supplements dated: ecember 22, 1997; September 28, 1998; and February 22, 1999.

Westinghouse Electric Company, LLC supplement dated: June 14, 2002.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

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

Date: August 27, 2004

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
6406	12	71-6406	USA/6406/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)*

U.S. Department of Energy
Division of Naval Reactors
Washington, DC 20585

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

U.S. Energy Research and Development Administration application dated July 19, 1977, 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.: None specified

(2) Description

Specific packaging is not required. Safety is independent of packaging.

(b) Contents

(1) Type and form of material

Unirradiated fuel assemblies of the following type:

(i) D2W rodded fuel cell or unrodded corner type D2W fuel module in a Model No. 658H1AB shipping and storage container. Rodded type fuel module shall have a control rod and control rod holddown device installed.

(2) Maximum quantity of material per package

One fuel assembly as described in 5(b)(1)(i).

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
6406	12	71-6406	USA/6406/AF	2	OF	2

(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): 100

6. Expiration date: March 31, 2008.

REFERENCES

U.S. Energy Research and Development Administration application dated July 19, 1977.

Supplements: Department of Energy letters G#5868 dated January 4, 1978, with enclosures; #6291 dated July 13, 1979; G#7609 dated September 30, 1983; G#C85-0435 dated April 19, 1985; G#C87-8027 dated December 23, 1987; G#92-03690 dated September 11, 1992; G#97-03513 dated June 11, 1997; G#C02-0700 dated February 8, 2002; G#02-0755 dated April 8, 2002; and G#02-4039 dated September 5,2002.

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: March 20, 2003,

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
6441	8	71-6441	USA/6441/B( )F	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)*

U.S. Department of Energy
Division of Naval Reactors
Washington, DC 20585

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Safety Analysis Report for D2G Power Unit Shipping Container dated August 4, 1969, 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.: D2G Power Unit

(2) Description

The D2G Power Unit shipping container assembly consists of five main assemblies; (1) the barrel assembly, (2) the upper cover, (3) the lower cover, (4) the main shipping skid, and (5) the barrel trunnion supports. To prepare the power unit shipping container for shipment of a power unit, the container barrel is rotated to the vertical position, the upper cover is removed and the power unit is loaded into the barrel and secured in the container with eight (8) shipping studs. The upper cover is then installed and the container is rotated to the horizontal position for shipment. The container assembly is 31 feet long and 8-1/2 feet wide and it is attached to a government owned permanently assigned depressed center railroad car; the maximum height above the rails is 13 feet, 10 inches in the shipping configuration. The power unit is shipped complete with design control rods and mechanisms installed.

The Type D or E power unit are retained in the container by means of eight shipping bolts. A special shipping ring is used to clamp the closure head and core cartridge assembly to the barrel upper flange of the shipping container. The control rods are restrained in the unit by means of rebound and outmotion latches located in the latching portion of the control rod drive mechanisms. The container assembly weighs about 100,000 pounds empty and about 270,000 pounds loaded.

(3) Drawings

The packaging is constructed in accordance with Baldwin-Lima-Hamilton Corporation Drawing Nos. R-126361, Rev. E, and R-126347, Rev. K, and Westinghouse Electric Corporation Drawing Nos. 955F632, Rev. 5, and 972D940, Rev. 5.

5. (b) Contents

(1) Type and form of material

Unirradiated enriched uranium as contained in Naval Reactors Type D or E power units consisting of core barrel, unirradiated fuel assemblies, closure head, mechanisms and associated hardware, with all design control rods and mechanisms installed.

(2) Maximum quantity of material per package

One power unit as described in 5(b)(1).

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

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

6. Expiration date: August 31, 2007

REFERENCES

Safety Analysis Report for D2G Power Unit Shipping Container, ONP-74252-13, dated August 4, 1969.

Supplements: Bettis Atomic Power Laboratory letters WAPD-DP(CH)-1252, dated November 30, 1973; WAPD-DP(CH)-1466, dated October 18, 1974; Knolls Atomic Power Laboratory letter CGN 85542-250, dated February 5, 1981; Naval Reactors letter NR:RR:ESSNIDER G#92-03731, dated October 7, 1992; Naval Reactors letter NR:RR:SLDUNN G#97-03543, dated July 10, 1997; and Naval Reactors letter NR:RR:MSHonea G#02-0735, dated March 13, 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: September 05, 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
6553	21	71-6553	USA/6553/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)*

United States Enrichment Corp.
6903 Rockledge Drive
Bethesda, MD 20817

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Safety Analysis Report on the "Paducah Tiger" Protective Overpack for 10-Ton Cylinders of Uranium Hexafluoride, Report No. KY-665, Revision 1, dated October 28, 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 No.: Paducah Tiger

(2) Description

A protective overpack which provides impact and thermal resistance for the Model No. 48X 10-ton cylinder. The cylinder is welded steel, and is 48 inches in diameter, 121 inches long, and has a 5/8-inch thick wall. The cylinder has a 108.9 ft3 volume, and is rated at 200 psig service pressure. The protective overpack has overall dimensions of approximately 153 inches x 76 inches x 72 inches. The overpack consists of two parts, a body and a lid, which are clamped and secured by four, 1-3/8-inch ratchet type binders, and eight, 1-3/4-inch guide pins, fitted with 3/4-inch high strength latch pins. The closed, assembled overpack consists of an outer 1/8-inch steel shell backed on both long sides, top and bottom by two, 10-gauge stainless steel breakaway plates. The valve end is protected by a 3/8-inch stainless steel breakaway plate and a 2-inch thick aluminum stiffening plate. A centrally located 3/16-inch steel shell, 60 inches in diameter x 128 inches long is separated from the outer shell by fire retardant polyurethane foam. The cylinder is held in the overpack by rubber shock isolators. Four mild steel brackets are provided on the body for lifting. Four, 2-inch bolts are used in conjunction with the ISO corner fittings for tie-down. The maximum gross weight of the package is 40,000 pounds.

(3) Drawings

The Paducah Tiger overpack is constructed in accordance with Martin Marietta Energy Systems, Inc., Drawing Nos. M-1209-NRC-1, Rev. 0, M-1209-NRC-2, Rev. 0, M-1209-NRC-3, Rev. A, M-1209-NRC-4, Rev. 1, and M-1209-NRC-5, Rev. 0.

5.(b) Contents

(1) Type and form of material

Solid uranium hexafluoride (UF6) at not more than 4.5 w/o U-235 isotope enrichment, and an H/U ratio of no more than 0.088.

(2) Maximum quantity of material per package

The maximum weight of UF6 not to exceed 21,030 pounds (9,540 kg). The maximum U-235 content not to exceed 640 pounds (290 kg).

(3) Criticality Safety Index 0.0

6. Each Model No. 48X cylinder must be inspected, tested, maintained, assembled, and used in accordance with American National Standards Institute (ANSI) N14.1-2001. The cylinders must be designed and fabricated in accordance with ANSI N14.1-2001 or an earlier version of ANSI N14.1 in effect at the time of fabrication. The cylinders must be fabricated in accordance with Section VIII, Division I, of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code and must be ASME Code stamped. Except that the 48X cylinders manufactured by W.H. Stewart Company in accordance with ANSI N14.1-1971 after ANSI N14.1-1982 was approved may be used for shipment in the Paducah Tiger package provided that they are inspected, tested and re-certified in accordance with ANSI N14.1-2001.

7. In addition to the requirements of Subpart G of 10 CFR Part 71, each package shall be maintained, repaired, operated and prepared for shipment in accordance with Operating Instructions and Acceptance Tests and Maintenance Program in the application dated October 28, 1998, as supplemented December 21, 1998, June 7, 1999, and February 29, 2000.

8. Use of Model No. 48A cylinders is not authorized.

9. Use of Model No. 48X cylinders made of A-285 steel is not authorized.

10. The Model 48X 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. Paducah Tiger overpacks previously constructed in accordance with Martin Marietta Energy Systems, Inc., Drawing Nos. M-1209-NRC-1, Rev. C; M-1209-NRC-2, Rev. A, M-1209-NRC-3, Rev. A; and M-1209-NRC-4, Rev. A, may be used until September 10, 1999. For the overpacks authorized by this condition, the clearance distance between the end of the cylinder valve and the plane of the end of the cylinder skirt must be measured prior to each shipment. The clearance distance must be at least 3/8 inch.

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

13. Revision No. 20 of this certificate may be used until December 30, 2006.

14. Expiration date: October 1, 2008. This package is not renewable.

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
6553	21	71-6553	USA/6553/AF	3	OF	3

REFERENCES

Safety Analysis Report on the "Paducah Tiger" Protective Overpack for 10-Ton Cylinders of Uranium Hexafluoride, Union Carbide Corporation Report No. KY-665, Revision 1, Dated October 28, 1998.

Supplements dated: December 21, 1998; January 12 and June 7, 1999; February 29, 2000; June 12, 2000; November 1, 2001; June 18, 2004; and September 6, 2005.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Robert A. Nelson, Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: December 22, 2005

UNITED STATES NUCLEAR REGULATORY COMMISSION

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
6574	32	71-6574	USA/6574/B( )	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)*
NUKEM Corporation
3800 Fernandina Road, Suite 200
Columbia, SC 29210-3854

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION
Scientific Ecology Group, Inc., application dated December 27, 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.: 3-82B

(2) Description

The packaging consists of a steel-lead-steel annulus cask fabricated in the form of a right circular cylinder and three different types of inner containers. The shielded cask, closed at one end and a lid closure at the other, is 66.25-inches in diameter by 74.5-inches in height. The cask wall consists of a 3/8-inch inner steel shell, 3-3/4-inches of lead shielding, one-inch outer steel shell, and a steel flange connecting the two shells. The cask outer shell is surrounded by a one-inch layer of insulating material and canned in 11-gauge steel.

The lid, sealed by a silicone flat gasket, is bolted to the cask body. A cylindrical shield plug is located in the center of the cask lid and is sealed by a silicone flat gasket. Lifting and tie-down devices are attached to the cask body. Impact skirts, consisting of removable rings of shock absorbing foam, are attached to the ends of the cask.

(3) Drawings

The package is fabricated in accordance with the following RWE NUKEM Corporation Drawing No.: STD-02-076, Sheets 1 through 3, Revision 8.

5. (b) Contents

(1) Type and form of material

Byproduct material consisting of dewatered, solid radioactive waste, including spent ion exchange resins, filter sludges, solidified evaporator concentrates, spent filter cartridges, and contaminated or irradiated solid materials.

(2) Maximum quantity of material per package

Greater than Type A quantity of byproduct material, which may contain not more than a Type A quantity of fissile material, provided the fissile material does not exceed the limits specified in 10 CFR 71.15. The cask contents must be contained within one of the following inner containers and limited as follows:

(a) Single disposable cylindrical containers constructed of metal or high integrity plastic with tightly fitted covers. A maximum decay heat load of 205 Btu/hr.

(b) Two pallets with four, 30-gallon drum size containers per pallet. Drums to be constructed of metal or high integrity plastic with a tightly fitted cover. A maximum decay heat load of 84 Btu/hr.

(c) One pallet with three, 55-gallon drum size containers. Drums to be constructed of metal or high integrity plastic with tightly fitted covers. A maximum decay heat load of 116 Btu/hr.

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 a hydrogen concentration 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.

6. (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. The total weight of the package must not exceed 50,000 pounds and the weight of the contents (including dunnage, etc.) must not exceed 8,195 pounds.

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 Section 7.0 of the application, as supplemented.

   (b) The package shall be maintained in accordance with the maintenance program in the supplement dated March 13, 1991.

9. Except for close fitting contents, sufficient dunnage, shoring, and/or bracing must be utilized to minimize secondary impact of the contents within the cavity under accident conditions of transport.

10. Prior to each shipment, the seal on the main cover and the seal on the shield plug cover, if opened, or if the security seal is broken, must be inspected. The seals must be replaced if the inspection shows any visible defects or every 12 months, whichever occurs first.

11. The packaging must be leak tested in accordance with Section 8.2.2 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 71.14(b)(3)(i), the pre-shipment leak test is not required.

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

13. Revision No. 31 of this certificate may be used until September 30, 2007.

14. Expiration date: October 1, 2008. This certificate is not renewable.

REFERENCES

Scientific Ecology Group Incorporated application dated December 27, 1990.

Supplements dated: March 13, 1991; March 7, 1996; and October 10, 1997.

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

ATG, Inc. supplements dated March 29, 2001; and May 10, 2001.

RWE NUKEM Corporation supplements dated May 8, 2003, May 13, 2005, and March 28, 2006.

NUKEM Corporation supplement dated September 6, 2006.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Robert A. Nelson, Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: September 27, 2006

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
6581	35	71-6581	USA/6581/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

Framatome ANP, Incorporated Consolidated License Application dated January 20, 2003, 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.: 51032-1

(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. Steel separator blocks are bolted between fuel assemblies. The separator blocks are a minimum 6 inches wide by approximately 8 inches high and 9 inches long, with a minimum nominal 3/8-inch thick wall. The outer container is approximately 43 inches in diameter by 216 inches 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 Siemens Power Corporation Drawing Nos.:

EMF-309,813, Rev. 2, Sheets 1 and 2
EMF-303,359, Rev. 7
EMF-303,360, Rev. 6
EMF-303,898, Rev. 5
EMF-300,607, Rev. 3
EMF-309,582, Rev. 0

5.(b) Contents

(1) Type and Form of material

(i) Unirradiated fuel rods consisting of uranium dioxide fuel pellets clad in zirconium alloy or stainless steel tubes. Fuel rods must be in one of the following configurations:

Type	15x15	17x17[1]	GEN1[2]	Rod Container[3]	T15x15 Square Array Assemblies[4]	T15 x15 Cruciform Assemblies
Maximum Enrichment (%U-235)	5.0	5.0	5.0	5.0	5.0	2.8
Rods Per Assembly	204	264	any number	any number	208	28
Nominal Rod Pitch (in.)	0.563	0.496	NA	NA	0.527	0.556
Maximum Pellet Density (%TD)	95	95	95	95	95	95
Maximum Clad OD (in.)	0.430	0.380	0.500	0.500	0.400	0.500
Minimum Clad OD (in.)	0.410	0.355	0.260	0.260	0.364	0.260
Minimum sum of clad thickness and pellet-clad gap[5] (in.)	0.023	0.023	0.023	0.023	0.016	0.023
Assembly Cross Section (in.)	8.445	8.432	8.25	NA	7.91	8.25
Active Fuel Length (in.)	196	196	196	196	196	116
Fuel Rod Arrangement (Figure Number in Application)	11.1	11.2	NA	NA	VII-1	VII-3

Table Notes

1 Fuel assemblies consisting of a maximum 264 fuel rods in a 17x17 square array with any number of edge rods missing.

2 Fuel assemblies consisting of any number of fuel rods in a square array with a maximum assembly cross section of 8.25 inches square.

3 Any number of fuel rods positioned in a rod container. The rod container consists of a schedule 40 steel pipe with a maximum nominal diameter of 5 inches.

4 Fuel assemblies consisting of a maximum of 208 fuel rods in a 15x15 square array, with any number of edge rods missing.

5 Minimum sum of the cladding wall thickness and the pellet-clad radial gap, ((Min Clad OD - Max Pellet OD)/2), in.

5.(b) Contents (Continued)

5.(b) Contents (Continued)

(ii) Unirradiated fuel assemblies, composed of uranium dioxide fuel pellets clad in zirconium alloy or stainless steel tubes. Uranium is enriched to a maximum of 5.05 wt% in the U-235 isotope. The fuel assemblies may contain inserted control rod assemblies. The fuel assemblies have the following specifications:

Type	L1	L2	L4
Array Size	15x15	15x15	17x17
Fueled Rods Per Assembly	208	208	264
Minimum No. of Non-Fueled Rods	17	17	25
Nominal Rod Pitch (in.)	0.568	0.568	0.496
Maximum Pellet Diameter (in.)	0.3707	0.3742	0.3232
Maximum Pellet Density (%TD)	97.5	97.5	97.5
Nominal Clad OD (in.)	0.430	0.430	0.374
Minimum sum of clad thickness and pellet-clad gap[1] (in.)	0.023	0.023	0.023
Assembly Cross Section (in.)	8.52	8.52	8.432
Active Fuel Length (in.)	196	196	196
Fuel Rod Arrangement (Figure Number in Application)	VIII-1	VIII-1	VIII-2

Table Notes:

1 Minimum sum of the cladding wall thickness and the pellet-clad radial gap, ((Min Clad OD - Max Pellet OD)/2), in.

5.(b) Contents (Continued)

(iii) Unirradiated fuel rods consisting of uranium dioxide fuel pellets clad in zirconium alloy or stainless steel tubes. Fuel rods must be in one of the following configurations.

Type	15x15	17x17[1]	GEN1[2]
Maximum Enrichment (%U-235)	4.87	4.87	4.87
Rods Per Assembly	204	264	any number
Nominal Rod Pitch (in.)	0.563	0.496	NA
Maximum Pellet Density (%TD)	97.5	97.5	97.5
Maximum Clad OD (in.)	0.430	0.380	0.500
Minimum Clad OD (in.)	0.410	0.355	0.260
Minimum sum of clad thickness and pellet-clad gap[3] (in.)	0.023	0.023	0.023
Assembly Cross Section (in.)	8.445	8.432	8.25
Active Fuel Length (in.)	196	196	196
Fuel Rod Arrangement (Figure Number in Application)	11.1	11.2	NA

Table Notes

1 Fuel assemblies consisting of a maximum 264 fuel rods in a 17x17 square array with any number of edge rods missing.

2 Fuel assemblies consisting of any number of fuel rods in a square array with a maximum assembly cross section of 8.25 inches square.

3 Minimum sum of the cladding wall thickness and the pellet-clad radial gap, ((Min Clad OD - Max Pellet OD)/2), in.

(2) Maximum quantity of material per package

Maximum quantity of material within a package may not exceed a Type A quantity. Total weight of fuel assemblies, or fuel rods, and rod containers, not to exceed 3400 pounds, and

(i) For the contents described in 5(b)(1)(i), the total weight of fuel assemblies:

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 the maximum fuel length for a full length assembly; or

Two rod containers.

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

Two fuel assemblies.

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

Two fuel assemblies.

(c) Criticality Safety Index 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. Separator blocks, shock mounts, and fuel element clamp assemblies must be in accordance with Tables 2.2, 2.3, 2.4, 2.5, and VII-3 of the application.

9. Each separator block must be attached to the strongback by one of the following methods, as shown in Drawing No. EMF-309,813, Rev. 2, Sheet 2:

(a) Two, 5/8-11 UNC Grade 5 steel cap screws and nuts. A 5/8-11 UNC Grade 2 (or better) steel stud may be substituted for one of the cap screws.

(b) Two, 1-8 UNC Grade 8 steel cap screws and nuts. A 1-8 UNC Grade 8 steel stud may be substituted for one of the cap screws.

The fuel assembly cross section is defined as the rod pitch times the number of rods on the edge of the assembly.

11. Rods containing gadolinia or other neutron poison are authorized but not required.

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE	PAGES
6581	35	71-6581	USA/6581/AF	6 OF	6

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 the procedures in Chapter 3.0 of the application.

    (b) Each packaging shall be maintained in accordance with the procedures in Section 3.4 of the application.

    (c) Each packaging shall meet the acceptance tests in Chapter 4.0 of the application.

    (d) Each fuel rod shall be welded closed and certified to be leak-tight prior to shipment.

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

14. Expiration date: October 1, 2008. This certificate is not renewable.

REFERENCES

Framatome ANP, Incorporated consolidated application dated January 20, and its supplements, May 8, June 18, July 7, and November 26, 2003; March 22 and August, 2005.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Robert J. Lewis, Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: 27 Sept. 2005

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
6613	12	71-6613	USA/6613/B(U)-96	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)*

QSA Global Inc.
40 North Avenue
Burlington, MA 01803

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

AEA Technology, QSA Inc., application dated July 19, 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.: 702

(2) Description

The cask system overall dimensions are 19" x 21" x 20". The cask is a stainless steel weldment containing depleted uranium shielding. The cask has a central cavity which is 2.26 inches in diameter by 3.25 inches long. Closure is accomplished by a neoprene gasket, six, 3/8-inch bolts and a stainless steel stepped plug containing depleted uranium shielding. The closure is equipped with an eye bolt. The cask is mounted on a 19" x 21" rectangular steel skid with four, 1/2-inch bolts and a tie-down system consisting of four, 1/2-inch diameter threaded rods which connect a clamp ring at the top of the cask to channel brackets welded to the skid. A protective cage constructed of 1-1/4-inch square steel tubing and perforated 18 gauge steel sheets tack welded to the tubular frame surrounds the cask and is bolted to the skid by four, 1/2-inch bolts. Maximum gross weight of the packaging is 410 pounds.

(3) Drawings

The cask and other system components are constructed in accordance with AEA Technology, QSA, Inc., Drawing Nos.: 70290, Sheets 1 to 10, Rev. M.

5. (b) Contents

(1) Type and form of material

Sources which meet the requirements of special form radioactive material. Authorized isotopes include Cs-137, Ir-192, Se-75, and Yb-169.

(2) Maximum quantity of material per package:

Isotope	Output Curies
Cs-137	500
Ir-192	15,000
Se-75	10,000
Yb-169	10,000

Output curies are determined by measuring the source output at 1 meter from the device and expressing its activity in curies. (Procedure reference: American National Standards Institute N432-1980, "Radiological Safety for the Design and Construction of Apparatus for Gamma Radiography.")

(3) Maximum decay heat per package:

129 watts.

6. The name plate must be fabricated of materials capable of resisting the fire test of 10 CFR Part 71 and maintaining their legibility.

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 Section 7.0 of the application, as supplemented.

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

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

Packages may be marked with Package Identification Number USA/6613/B(U)-85 until September 30, 2006.

9. Revision No. 11 of this certificate may be used until November 30, 2006.

Expiration date: June 30, 2008.

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

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6613	12	71-6613	USA/6613/B(U)-96	3	OF	3

REFERENCES

AEA Technology, QSA Inc., application dated July 19, 2001.

Supplements dated: March 12 and July 19, 2002; April 3, 2003; June 24 and October 25, 2005.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Robert A. Nelson, Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: 11/18/05

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
6642	7	71-6642	USA/6642/B( )	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)*
U.S. Department of Energy
Washington, DC 20585

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION
Safety Analysis Report - Packages SRL 4.5 Ton Californium Shipping Cask, DPSPU 74-124-6, December 1974, Rev. 1, March 1976, 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.: 4.5-Ton Cf

(2) Description

A shielded packaging for special form materials. The outer container is a 3/4-inch thick, 61-1/2-inch OD spherical steel shell filled with borated water extended polyester (WEP) shielding. Outer shell is fitted with nine (9) fusible plugs and a vent valve for relief of gases generated in the WEP material. The cylindrical containment cavity approximately 4-inch diameter by 6-3/8 inches high is centrally located in the sphere and surrounded by lead of 2 inches, 1.9 inches and 1.75 inches thickness on the bottom, sides and top, respectively. The containment vessel is an integral part of the outer container, and is held by a 31-1/2-inch long 4-1/2-inch OD tube welded to a 3/4-inch thick 22-1/2-inch diameter top plate mounted to the outer container closure assembly. Closure of the containment vessel is accomplished by a flange plate and sleeve insert assembly. The sleeve is a 27-inch long, 4-inch OD tube filled with lead and water extended polyester and is gasketed and bolted to the top closure assembly of the container. A 22-1/2-inch diameter protective cover bolts to the closure assembly sleeve. A hexagonal shaped assembly, approximately 5 feet across the flats mounts, to the spherical shell as a base. Four equally spaced lifting lugs are provided around the upper hemisphere. The cask gross weight is approximately 9,500 pounds.

(3) Drawings

The SRL 4.5-Ton Californium shipping cask is as described, and is constructed in accordance with E.I. duPont de Nemours Company Drawing Nos.: ST5-15813, Rev. 33; ST5-15814, Rev. 29; ST5-15815, Rev. 0; ST5-15816, Rev. 0; ST5-15817, Rev. 0; and ST5-15818, Rev. 5.

5.(b) Contents

(1) Type and form material.

Californium 252, as sealed source which meets the requirements of special form radioactive material.

(2) Maximum quantity of material per package.

46 curies (85 mg).

6. Prior to each shipment, the WEP shielding space shall be vented, using the1/4-inch angle valve which is then closed.

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 Procedure described in the application, as supplemented dated September 18, 1991.

(b) The package must be maintained in accordance with the Maintenance Program described in the application, as supplemented dated September 18, 1991.

8. Use of packaging fabricated after August 31, 1986, is not authorized.

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: February 28, 2007.

REFERENCES

Safety Analysis Report - Packages SRL 4.5-Ton Californium Shipping Cask, DPSPU 74-124-6, December 1974, Revision 1, March 1976.

Supplements dated: September 18, 1991; July 17, 1996; and January 25, 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: 5/14/02

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
6703	7	71-6703	USA/6703/B( )	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)*
General Atomics
3550 General Atomics Court
San Diego, CA 92121-1122

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION
General Atomic Company application dated December 16, 1974, 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.: RG-1

(2) Description:

The package, a thermoelectric generator, is 18 inches high and has a base diameter of 14 inches. The components include the main housing, uranium and tungsten shield, housing flange, electrical connector and lifting lugs. A notch at the base provides the tie-down flange. The 1.75-inch thick cover flange is bolted to the housing by 16 or 20 steel bolts depending on the generator configuration. The electrical receptacle is bolted to the cover flange with an O-ring being provided between the interfaces and on the lateral surface of the feed plug. The lifting lug is threaded into the cover flange and is removable if necessary for an operational installation. Package weight is approximately 800 pounds.

(3) Drawings:

The packaging is constructed in accordance with the detailed drawings listed on Gulf General Atomics Generator Assembly Drawing Nos.: D346-3000, Rev. K, and J346-3000, Rev. K.

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
6703	7	71-6703	USA/6703/B( )	2	OF	2

5. (b) Contents

(1) Type and form of material

Strontium-90 titanate doubly encapsulated in a Type 304L stainless steel liner and Hastelloy C capsule.

(2) Maximum quantity of material per package

8,300 curies.

6. In addition to the requirements of Subpart G of 10 CFR Part 71, the package must be prepared for shipment in accordance with the supplement dated February 26, 2004.

7. This certificate authorizes a one-time shipment from General Atomics Company site, San Diego, California, to the Los Alamos National Laboratory, Los Alamos, New Mexico, and additional shipments by the U.S. Department of Energy as needed for storage, disposition, and disposal.

8. This approval is limited to two packages (Serial Nos. -001 and -002).

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

10. Expiration date: September 30, 2008. This certificate is not renewable.

<u>REFERENCES</u>

General Atomic Company application dated December 16, 1974.

Supplements dated February 26, 2004; and January 31, 2005.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Robert J. Lewis, Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date 24 March 2005

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
6786	8	71-6786	USA/6786/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)*

Department of the Navy
NRSC Technical Support Center
Naval Sea Systems Command Detachment
Radiological Affairs Support Office
PO Drawer 0260
NWS Yorktown, VA 23691-0260

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Aerojet Application dated February 18, 1971, 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.: URIPS-8A and URIPS-8B

(2) Description

The packages, thermoelectric generators, are 28.5 inches in overall height, with an outer diameter of 19.14 inches, and total weight of approx. 1,600 pounds. The components include a depleted uranium shield (470 lbs.), a steel housing, cover bolts (recessed and caulked over), an electrical adaptor, cooling fin system, and cylindrical fin guard, stiffened by eight ribs on the inside surface. The housings are equipped with lifting and tie down devices. The Model No. URIPS-8B differs from Model No. URIPS-8A in the electric converter system. The thermoelectric generator may be secured in a shipping frame identified in Drawing No. 1138459, Rev. A.

(3) Drawings

The package is constructed in accordance with the following Aerojet Company Drawing Nos.:

1138441	8-Watt URIPS-8A Assembly
1138442, Rev. C	Generator Housing
1138457	Cooling Fins
1139240, Rev. A	Fin Guard
1139245, Rev. A	Shipping Package URIPS-8
1139246	8-Watt URIPS Assembly
1138459, Rev. A	Shipping Frame-URIPS-8
1138443, Rev. B	Top Cover
1138444	Bottom Cover
1138436	Fuel Capsule
1138437, Rev. B	Shield Uranium
1138435	Fuel Liner
1138440, Rev. A	W-2 Shield Plug
1138453	Insulation
1138455, Rev. B	Copper Plug

(b) Contents

(1) Type and form of material

Strontium 90 titanate doubly encapsulated which meets the requirements of special form radioactive material.

(2) Maximum quantity of material per package

56,850 ci.

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 specified in the supplement dated August 6, 1998.

(b) The package must be maintained in accordance with the maintenance procedures specified in the supplement dated August 6, 1998.

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

8. Expiration date: October 1, 2008. This package is not renewable.

REFERENCES

Aerojet Nuclear Systems Company application dated February 18, 1971.

Supplemented by Naval Nuclear Power Unit letter dated: December 10, 1971, and Oak Ridge

National Laboratory dated: December 28, 1972; and February 27 and March 27, 1973.

Department of the Navy application dated: June 8, 1990.

Supplements: Department of the Navy letter 5104 Ser 455/1U599998 dated June 18, 1991; Department of the Navy letter 5104 Ser N455C/8U595525 dated April 16, 1998; and Department of the Navy letter 5104 Ser N455C/8U595912 dated August 6, 1998; Department of the Navy letter 5104 Ser N455C/3U574771 dated August 25, 2003.

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: March 11, 2005

UNITED STATES NUCLEAR REGULATORY COMMISSION

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
9001	39	71-9001	USA/9001/B( )F	1 OF 9

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

VECTRA Technologies, Inc., application dated March 30, 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.: IF-300

(2) Description

A stainless steel encased, depleted uranium shielded cask. The cask is cylindrical in shape, 64 inches in diameter, and a maximum of 210 inches long with maximum cavity dimensions of 37-1/2 inches in diameter by 180-1/4 inches long. Shielding is provided by 4 inches of depleted uranium, 2-1/8 inches of stainless steel, and a minimum of 4-1/2 inches (550 gallons) of a water ethylene glycol mixture.

Two closure heads are provided for the shipment of BWR and PWR fuel assemblies. The heads are 304 stainless steel forgings and end plates which encase the 3-inch thick depleted uranium shielding. Either closure head may be used for packaging solid irradiated hardware.

The closure heads are secured to the cask body by means of 32, 1-3/4 inch studs and nuts. The cask is sealed with a metallic ring gasket.

The cavity is penetrated by a vent line at the top and a drain line at the bottom. These lines are sealed by bellows stainless steel globe valves and valved quick-disconnect couplings. Stainless steel pipe caps or pipe plugs may be used in lieu of the quick-disconnect couplings. The vent line is also equipped with a 350-400 psig rated rupture disk. All valves are housed in protected boxes on the cask exterior.

5.(a) Packaging (continued)

(2) Description (continued)

Neutron shielding is provided by a liquid-filled, thin-walled, corrugated containment on the cask exterior. This cylindrical structure is separated into two longitudinal compartments, each equipped with two expansion tanks, fill and relief valves. The fill line from each compartment is terminated by a stainless steel globe valve in a protected box (separate from cavity boxes) on the cask exterior. The stainless steel globe valves may be replaced by stainless steel blind flanges. The vent line from each compartment goes to an expansion tank which is provided with a pressure relief valve set at 200 psig.

The cask has three types of fuel baskets which can be interchanged to accommodate various fuels. The PWR basket holds seven assemblies (except for Group III PWR contents, where six assemblies are authorized and the center cell does not contain a fuel assembly), the unchanneled BWR basket holds eighteen assemblies, and the channeled BWR basket holds seventeen assemblies. The channeled and unchanneled BWR fuel baskets may be provided with supplementary shielding (depleted uranium) near the cask closure.

The cask is shipped horizontally with the bottom supported in a tipping cradle between two pedestals and the upper end resting in a semi-circular saddle; the upper end is pinned to the saddle. The cask supports are welded to the framing of a 37-1/2-foot long by 8-foot wide structural steel skid. The skid may also have installed on it an auxiliary cooling system, consisting of two diesel engines driving two blowers which discharge cooling air to the corrugated surface of the cask via common ducting. Neither installation nor operation of all or part of this auxiliary cooling system is a requirement of this package approval.

The entire cask and cooling system is covered by a retractable aluminum enclosure. Access to the enclosure is via locked panels in the side and a locked door in one end. Although the Model No. IF-300 cask can be transported for short distances on the highway, its principal mode of transportation is by railroad.

The gross weight of the cask is approximately 140,000 pounds. The skid and other external components weigh approximately 45,000 pounds.

(3) Drawings

The Model No. IF-300 shipping cask is described by the following General Electric Company Drawing Nos.: 159C5238 - Sheet 1, Rev. 9; Sheet 2, Rev. 3; Sheet 4, Rev. 8; Sheet 5, Rev. 5; Sheet 6, Rev. 8; Sheet 7, Rev. 4; Sheet 8, Rev. 5; Sheet 9, Rev. 8; Sheet 10, Rev. 5; and Sheet 11, Rev. 2, GTS Duratek Drawing No.: C-110-B-57915-001, Rev. 1, Duratek Drawing No. C-002-044125-001, Rev. 0, and Pacific Nuclear Systems, Inc. Drawing Nos.: 420-11-3000, Sheets 1 through 9, Rev. 1; 420-11-3001, Sheet 1, Rev. 1; 420-11-3002, Sheets 1 and 2, Rev. 1; 420-11-3003, Sheets 1 and 2, Rev. 1; 420-11-3004, Sheets 1 and 2, Rev. 1; 420-11-3005, Sheets 1 and 2, Rev. 1; and 420-11-3006, Sheet 1, Rev 1.

5.(a)(4) Basic Components

The basic components of the Model No. IF-300 shipping cask that are important to nuclear safety are listed in Section IX, Table IX-1.

(b) Contents

(1) Type and form of material

(i) Irradiated PWR and BWR uranium oxide fuel assemblies. PWR assemblies may be shipped with or without control rods. *Partial fuel assemblies, that is, assemblies from* which fuel pins are missing, **must not be** shipped unless dummy fuel pins are used to displace an amount of water equal to that displaced by the original pins. The specific power of each fuel assembly must not exceed 40 kW/kgU. The BWR and PWR fuel assemblies must have the following dimensions and specifications:

Group Ia fuel assemblies

	PWR	BWR
Fuel form	Clad UO_2 pellets	Clad UO_2 pellets
Cladding material	Zr or SS	Zr or SS
Maximum initial U content/assembly, kg	465	198
Maximum initial U-235 enrichment, weight percent	4.0	4.0
Maximum assembly average burnup, MWd/MTU	35,000	35,000
Minimum cooling time, days	120	120
Maximum initial bundle cross section, in	8.75	5.75
Fuel pin array	14x14/15x15	7x7
Initial fuel diameter, in	0.380-0.460	0.500-0.600
Initial fuel pin pitch range, in	0.502-0.582	0.647-0.809
Maximum initial active fuel length, in	145	146

5.(b)(1)(i) Contents - Type and form of material (continued)

<u>Group Ib fuel assemblies</u>

	PWR
Fuel form	Clad UO_2 pellets
Cladding material	Zr or SS
Maximum initial U content/assembly, kg	439
Maximum initial U-235 enrichment, weight percent	4.0
Maximum assembly average burnup, MWd/MTU	45,000
Minimum cooling time, years	5
Maximum initial bundle cross section, in	8.75
Fuel pin array	15x15
Initial fuel diameter, in	0.380-0.460
Initial fuel pin pitch range, in	0.502-0.582
Maximum initial active fuel length, in	144

5.(b)(1)(i) Contents - Type and form of material (continued)

<u>Group II fuel assemblies</u>

	PWR	BWR
Fuel form	Clad UO_2 pellets	Clad UO_2 pellets
Cladding material	Zr or SS	Zr or SS
Maximum initial U content/assembly, kg	475	198
Maximum initial U-235 enrichment, weight percent	4.0	4.0
Maximum assembly average burnup, MWd/MTU	35,000	35,000
Minimum cooling time, days	120	120
Maximum initial bundle cross section, in	8.75	5.75
Fuel pin array	16x16/17x17	8x8
Initial fuel diameter, in	0.376-0.400	0.475-0.505
Initial fuel pin pitch range, in	0.496-0.507	0.630-0.645
Maximum initial active fuel length, in	150	150

5.(b)(1)(i) Contents - Type and form of material (continued)

Group III fuel assemblies

	PWR[a]	BWR[b]
Fuel form	Clad UO_2 pellets	Clad UO_2 pellets
Cladding material	Zr	Zr
Maximum initial U content/assembly, kg	442	187
Maximum initial U-235 enrichment, weight percent	4.25	4.25
Maximum assembly average burnup, MWd/MTU	45,000	45,000
Minimum cooling time, years	5	4
Maximum initial bundle cross section, in	8.75	5.75 (8x8) 5.75 (9x9)
Fuel pin array	15x15	8x8/9x9
Initial fuel diameter, in	0.424	0.483 (8x8) 0.440 (9x9)
Initial fuel pin pitch, in	0.563	0.640 (8x8) 0.566 (9x9)
Maximum initial active fuel length, in	144	150 (8x8) 146 (9x9)
Minimum initial top/bottom blanket length, in[c]	6	6 (8x8) 6 (9x9)

Notes:

[a] The center fuel assembly location in the PWR basket must not contain a fuel assembly, with the six PWR assemblies being placed in the six peripheral basket positions.

[b] This note is no longer applicable.

[c] Length of natural UO_2 fuel above and below the enriched portion of the active fuel.

5.(b)(1) Contents - Type and form of material (continued)

(ii) Solid irradiated hardware, which may include fissile material, provided the quantity of fissile material does not exceed a Type A quantity and does not exceed the mass limits of 10 CFR 71.15. As needed, appropriate component spacers must be used when loading irradiated hardware into the cask cavity to limit movement of the contents during accident conditions of transport. Use of a steel liner is authorized provided: (1) its outside dimensions are approximately those of the cask cavity inside dimensions, (2) constructed of single thickness of steel plate with full penetration welds, (3) thickness of steel plate does not exceed one inch, and (4) the liner is provided with a drain and vent to insure water removal.

(2) Maximum quantity of material per package

Maximum decay heat per package not to exceed 40,000 Btu/hr. Maximum 5,725 Btu/hr/PWR assembly. Maximum 2,225 Btu/hr/BWR assembly.

(i) Seven PWR fuel assemblies for Groups Ia, Ib and II as described in 5.(b)(1)(i).

(ii) Six PWR fuel assemblies for Group III as described in 5.(b)(1)(i). The center fuel assembly location in the PWR basket for Group III PWR contents must not contain a fuel assembly, with the six PWR assemblies being placed in the six peripheral basket positions.

(iii) Seventeen channeled BWR assemblies (for Groups Ia, II and III), or eighteen unchanneled BWR fuel assemblies (for Groups Ia and II), as described in 5.(b)(1)(i).

(iv) Above fuel assemblies to be contained in their respective fuel baskets as shown in GE Drawing No. 159C5238 - Sheet 6, Rev. 8 and GTS Duratek Drawing No. C-110-B-57915-001, Rev. 1, or PNSI Drawing No. 420-11-3000, Sheets 1 through 9, Rev. 1.

5. (c) Unloaded package - contents and maximum quantity of material

Greater than a Type A quantity of residual radioactive material consisting of mixed-fission and activation products adhering to interior cavity and fuel basket surfaces.

(d) Criticality Safety Index

For Groups Ia, Ib and II PWR and BWR fuel assemblies as described in 5.(b)(1)(i)	0.4
For Group III PWR and BWR fuel assemblies as described in 5.(b)(1)(i)	0.0

6. The end of life total calculated residual gas that could become available from the fuel pins must not exceed 0.50 lb moles for content 5.(b).

7. The maximum gross weight of the cavity contents must not exceed 21,000 pounds.

8. For the shipment of irradiated fuel assemblies, the cask cavity (containment vessel) must be promptly inerted following removal of the water from the cavity. The cask cavity must be purged at least three times with argon, nitrogen, or helium. Each purge volume must be equivalent to or greater than the cask cavity volume. After the final purge, the cavity must be promptly filled with argon, nitrogen, or helium at 1.0 atm pressure.

9. Known or suspected failed fuel assemblies (rods) and fuel with cladding defects greater than pin holes and hairline cracks are not authorized.

10. Prior to loading Group III PWR contents, a plug must be inserted into the center assembly location of the PWR basket and there must not be a Group III PWR assembly in the center basket location at any time.

11. Prior to each shipment, the licensee must confirm that the cask contains no more than 1 cubic foot of water in the cavity and the licensee must prepare the cask for shipment, in accordance with Subsection 10.1 of the application.

12. (a) The cask contents shall be so limited that under normal conditions prior to transport, 62 times the neutron dose rate plus 6.3 times the gamma dose rate will not exceed 560 mrem/hr at a distance of six feet from the side of the cask (ten feet from the cask center-line).

    (b) The cask content limitation of 12.(a) does not apply to:

    (1) Group II BWR fuel in the channeled fuel basket with a minimum planar average enrichment of 2.65 wt% ^{235}U.

    (2) Group III BWR fuel in the channeled fuel basket with a minimum planar average enrichment of 3.19 wt% ^{235}U.

13. The neutron shielding tanks must be filled with approximately a 50/50 volume percent mixture of ethylene glycol and water during the months of October through May.

14. Replacement globe valves other than the valve specified on Drawing No. 159C5238-Sheet 4, Rev. 8, must be tested as stated in Subsection 6.6.3.2 of the application.

15. The packaging must be maintained in accordance with the requirements of Subsection 10.2 of the application. During inactive periods, the maintenance and testing frequency may be disregarded provided that the package is brought into full compliance with these requirements prior to the next use of the package.

16. The cask cavity must be equipped with a rupture disk device with a burst pressure within the range of 350-400 psig (443°F) including all tolerances.

17. The uranium shielding material must be separated from all steel surfaces with a minimum copper thickness of 4-mils, except that the stud bolts attaching the shield assemblies to top of the unchanneled BWR basket must be coated with a minimum of ½-mil of copper.

18. A shutoff valve must not be installed between each neutron shield tank and its respective thermal expansion tank.

19. The cask may be wrapped with reinforced plastic during shipment, provided that the decay heat of the contents does not exceed 1.5 KW. The reinforced plastic used to wrap the cask must not be greater than 0.015 inches thick or have a thermal conductivity less than 0.0242 Btu/hr-ft-°F. The reinforced plastic wrapping cannot be used as the cask surface for purposes of complying with 10 CFR 71.87.

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

21. Expiration date: October 1, 2008. This certificate is not renewable.

REFERENCES

VECTRA Technologies, Inc., application dated March 30, 1995.

VECTRA Technologies, Inc., supplements dated: April 27, and August 18, 1995; November 25, 1997;

Chem-Nuclear Systems supplements dated January 9, 1998; June 8 and June 21, 1999; January 14, February 17, March 16, June 16, July 14, October 11, October 20, and November 9, 2000; and April 23, 2001.

Duratek supplements dated February 4, September 9, October 21, 2002; August 25, 2004 and March 4, 2005.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Robert J. Lewis, Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: April 04, 2005

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
9010	41	71-9010	USA/9010/B( )F	1 OF	10

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 February 27, 1996.

4. CONDITIONS

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

)) Packaging

(1) Model No.: NLI-1/2

(2) Description

A depleted uranium, water, and lead shielded shipping cask, encased in stainless steel, and equipped with balsa impact limiters. The cylindrical cask body is 195-1/4 inches long by 47-1/8 inches OD. The principal shielding consists of 2-3/4 inches of depleted uranium, 2-1/8 inches of lead, and 5 inches of (borated) water-ethylene glycol mixture.

A 7/8-inch thick stainless steel outer shell is welded to a solid stainless steel forging at each end of the cask. The outer shell of the cask is surrounded by a 1/4-inch thick steel water jacket that is also attached to the end forgings. A water expansion tank is welded to the water jacket shell. The inner cask cavity is formed by a 1/2-inch thick, stainless steel cylindrical shell; welded at its top end to the upper cask forging and its bottom end to a circular plate.

There are four separate configurations of the cask.

Configuration (A): The containment vessel is a right circular stainless steel shell, 12-5/8 inches ID by 178 inches inside length by 1/4-inch thick, located within the inner cask cavity. The containment vessel is closed and sealed by a 5-inch thick, composite steel and uranium closure head, twelve, 1-inch diameter bolts, and silver plated, metallic O-ring. Eight of the twelve closure bolts are used to secure the containment vessel to the upper cask forging. Closure of the cask cavity is by a 1-1/2-inch thick steel closure head, eight, 1-inch diameter bolts, and elastomer O-ring. The radioactive contents are positioned and supported within the containment vessel (inner container) by an aluminum basket and internal support structure.

5.(a) Packaging (continued)

(2) Description (continued)

Configuration (B): The containment vessel is the 1/2-inch thick inner cavity shell. The 1/4-inch thick inner container is not used. The cask cavity is closed by two closure heads. The inner head is a 6-inch thick, composite steel and uranium plate secured to the upper cask forging by twelve, 1-inch diameter bolts and sealed with a silver plated, metallic O-ring. The outer head is 1-1/2-inch thick steel plate secured to the top of the upper cask forging by eight, 1-inch diameter bolts and sealed with an elastomer O-ring. The radioactive contents are positioned and supported within the containment vessel (inner cask cavity) by a modified aluminum basket and internal support structure.

Configuration (C): Same as Configuration (B), above, except the radioactive contents are positioned and supported within the containment vessel (inner cask cavity) in a stainless steel structure containing Boral sheets positioned so as to provide necessary neutron absorption.

Configuration (D): Same as Configuration (B) above, except that the radioactive contents are positioned and supported within the containment vessel (inner cask cavity) in a 3-element stainless steel structure as shown in NAC Drawing No. 347-291-F12, sheet 1, Rev. 2, and the cask must be enclosed in a closed shipping container.

The package, including impact limiters, has an overall length of 237 inches and an outside diameter of 75 inches. The maximum weight of the contents is 3,000 pounds. The weight of the package is approximately 49,250 pounds.

(3) Drawings

The Model No. NLI-1/2 shipping cask is constructed in accordance with the following National Lead Company Drawing Nos.:

General

70514F, Sheet 1, Rev. 8, Cask and Trailer General Arrangement
70514F, Sheet 2, Rev. 8, Cask and Trailer General Arrangement
70885F, Sheet 1, Rev. 3, Spent Fuel Cask Details
70885F, Sheet 2, Rev. 2, Spent Fuel Cask Details
70885F, Sheet 3, Rev. 2, Spent Fuel Cask Details
70885F, Sheet 4, Rev. 1, Spent Fuel Cask Details
70887F, Sheet 1, Rev. 1, Outer Closure Head
70888F, Sheet 1, Rev. 3, Spent Fuel Cask General Assembly

Configuration (A)

70516F, Sheet 1, Rev. 8, Spent Fuel Cask General Assembly
70562F, Sheet 1, Rev. 11, Inner Container
70562F, Sheet 2, Rev. 7, Inner Container
70562F, Sheet 3, Rev. 0, Inner Container*
70562F, Sheet 4, Rev. 0, Inner Container*

5.(a) Packaging (continued)

(3) Drawings (continued)

Configuration (B)

70886F, Sheet 1, Rev. 2, Basket Concept
70884F, Sheet 1, Rev. 2, Inner Closure Head

Configuration (C)

460-052-F8, Sheet 1, Rev. 4, Rockwell Fuel Basket-NLI-1/2 Cask*
460-052-F9, Sheet 1, Rev. 3, Container - Fermi Fuel, Rockwell Basket, NLI-1/2 Cask, Assembly of*

Configuration (D)

347-291-F12, sheet 1, Rev. 2, Liner - 3 Element, NLI-1/2 Cask, Fuel Movement Project*

*Nuclear Assurance Corporation drawings.

(b) Contents

(1) Type and form of material

(i) Irradiated PWR or BWR uranium oxide fuel assemblies of the following specifications:

	PWR	BWR	Consolidated Fuel Rods
Fuel form	Clad UO_2 pellet	Clad UO_2 pellet	Clad UO_2 pellets
Cladding material	Zr or SS	Zr or SS	Zr or SS
Maximum initial fuel pin pressure at 100°F, psig	550	200	550
Maximum initial U content/assembly, kg	475	197	950
Maximum average initial U-235 enrichment, w/o	3.70	2.65	3.70
Maximum bundle cross section, inches	8.75	5.75	8.75
Fuel pin array size	14x14/15x15 16x16/17x17	7x7 8x8	Pins from 7x7, 8x8, 14x14, 15x15, 16x16, 7x17 in triangular pitch

5.(b)(1)(i) (continued)

	PWR	BWR	Consolidated Fuel Rods
Maximum active fuel length, inches	144	145.25	144
Maximum specific power, kW/kgU	40	27	40
Maximum average burnup, MWD/MTU	40,000**	34,000	40,000
Maximum decay heat, kW	10.6	10.6	0.6
Minimum cooling time, days	150*	120	4,380

The PWR type assembly may be shipped either with or without burnable poison rods or control rods.

*Four (4) fuel rods may have a minimum cooling time of 120 days.

**PWR fuel assembly may have a maximum average burnup of 56,000 MWD/MTU provided the minimum cooling time prior to shipment is 450 days and the neutron shield fluid contains 1.0 weight percent boron. (The borated fluid may be left in the shielding tanks during the shipment of other contents.)

(ii) Irradiated metallic fuels of the following specifications:

	Fermi-1	EBR-II Blanket
Fuel form	Uranium-molybdenum alloy pins	Uranium metal cylindrical slugs
Cladding material	Zr	Aluminum containers
Max. initial U content/assembly, kg	18.7/assy. 300/16 assy. cask load	292/container
Max. avg. initial U-235 enrichment, w/o	26.0	0.21 (3.88 kg Pu/canister)
Max. bundle cross section, inches	2.93 sq	4.875 dia
Fuel rods per canister	140	41
Max. active fuel length, inches	30.5/assy 122/cask	157
Max. average burnup, MWD/MTU	2,840	2,400
Max. decay heat, watts	20	300
Min. cooling time, days	5,000	365

5.(b)(1)(continued)

(iii)

	Research Reactor
Fuel form	Uranium metal rods
Cladding material	Aluminum
Maximum initial U content/assembly-kg	54.5
Maximum average initial U-235 enrichment	Natural
Maximum bundle cross-section, inches	1.36
Intact fuel rods per canister, maximum	7
Canisters per cask	3 intact fuel
Max. active fuel length, inches	120.5
Maximum average burnup MWD/MTU	1,600
Maximum decay heat, watts	750
Minimum cooling time, days	365

(iv) Irradiated PWR* or BWR uranium oxide fuel rods of the following specifications:

	PWR Rods	BWR Rods
Fuel form	Clad UO_2 pellets	Clad UO_2 pellets
Cladding material	Zr or SS	Zr or SS
Maximum initial fuel pin pressure at 100°F, psig	550	200
Maximum initial U content, kg	58.2	75
Maximum average initial U-235 enrichment, w/o	4.9	5.0
Maximum bundle cross section, inches	8.75	5.75
Maximum active fuel length, inches	150	150
Maximum specific power, kW/kgU	44	60
Maximum average burnup, MWD/MTU	60,000	75,000

5.(b)(1)(iv) (continued)

	PWR Rods	BWR Rods
Maximum decay heat, kW	1.65	4.0
Minimum cooling time, days	150	150

* For the shipments of irradiated PWR fuel rods, the neutron shield fluid must contain 1.0 weight percent boron (the borated fluid may be left in the shielding tanks during the shipment of other contents).

(v) Solid, non-fissile, irradiated hardware and neutron source components.

(vi) Byproduct and special nuclear material in the form of irradiated uranium and plutonium oxide fuel rods. Prior to irradiation, the maximum average enrichment in U-235 plus plutonium not to exceed 3.70 w/o and the maximum enrichment not to exceed 4.0 w/o.

(vii) Irradiated PWR uranium oxide fuel assemblies including additional irradiated fuel rods inserted and secured in the guide thimbles. The fuel assemblies must conform to the maximum active dimensions as described in Item 5(b)(i) except that maximum initial U content must be 495 kg and the maximum average initial U-235 enrichment shall be 3.35 w/o.

(viii) Irradiated Connecticut Yankee fuel assembly with a maximum average initial U-235 enrichment of 4.0 w/o and each of the 15 x 15 fuel rods clad by stainless steel. 204 rods/assembly; active length of 121.4 inches.

(ix) Irradiated MARK 42 fuel assemblies consisting of three concentric fuel tubes with PuO_2-Al powder metallurgy cores clad with type 6063 aluminum, containing a total of 3.35 kg of plutonium. The plutonium was initially enriched to contain 78.28 w/o Pu-239, 2.27 w/o Pu-241 and 0.15 w/o Pu-238.

(x) Irradiated MARK 22 fuel assemblies consisting of two concentric fuel tubes with uranium-aluminum cores clad with type 8001 aluminum, containing a total of 3.2 kg of uranium-235. The uranium was initially enriched to contain 66 w/o to 80 w/o uranium-235. The irradiated MARK 22 fuel assembly has an active length of 150 inches, a maximum burn-up of 1226 MWD and a minimum cooling time of 150 days.

(2) Maximum quantity of material per package

(i) Items 5(b)(1)(i) or 5(b)(1)(vii) above: one PWR fuel assembly; two BWR fuel assemblies; or one consolidated fuel canister. Fuel assemblies to be contained in their respective fuel baskets as shown on National Lead Company Drawing No. 70562F, Sheet 1, Rev. 11, or 70886F, Sheet 1, Rev. 2. The consolidated fuel canister to be contained in Configuration (A) fuel basket as shown on National Lead Company Drawing No. 70562F, Sheet 1, Rev. 11.

(ii) Item 5(b)(1)(ii) above: four canisters per cask. The fuel canisters and fuel basket must be in accordance with Configuration (C) above.

5.(b)(2)(continued)

(iii) Item 5(b)(1)(iii) above:

(a) three canisters of unfailed fuel containing up to seven fuel rods per canister. The fuel canisters and fuel basket must be in accordance with Configuration (D) above; or

(b) up to six canisters containing one defective 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 the six-rod capacity liner as shown on Nuclear Assurance Corporation Drawing No. 347-029-20, Rev. 1. The maximum decay heat load for a defective fuel rod is limited to 5 watts; or

(c) up to three canisters containing either one defective fuel rod 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. The fuel basket is in accordance with Configuration (D) above. The weight of the filters is limited to 125 pounds per canister. The maximum decay heat load for the defective fuel rods and the failed fuel filters is limited to 5 watts per canister. Plutonium content of the filters not to exceed 20 curies plutonium per package.

(iv) Item 5(b)(1)(iv) above, the fuel rods will be shipped in Configuration (A) or (B). PWR fuel rods with burnup in excess of 45,000 MWD/MTU and BWR fuel rods with burnup in excess of 50,000 MWD/MTU will be shipped in Configuration (A) only. The maximum initial uranium content is limited to 58.2 kg per package for PWR rods and 75 kg per package for BWR rods; and

(a) up to 25 PWR fuel rods or up to 25 BWR fuel rods per cask. Up to 2 of the 25 PWR rods may have a maximum burnup of 65,000 MWD/MTU; or

(b) up to 18 PWR fuel rods, with a maximum specific power of 60 kW/kgU and a minimum cooling time of 300 days, per cask.

(v) Item 5(b)(1)(v) above, weight not to exceed 1,600 pounds.

(vi) Item 5(b)(1)(vi) above, the maximum mass of U-235 plus plutonium must not exceed 4.0 kg. Fuel rods must be contained in fuel baskets as shown on National Lead Company Drawing No. 70562F, Sheet 1, Rev. 11, or 70886F, Sheet 1, Rev. 2.

(vii) Item 5(b)(1)(viii) above: One Connecticut Yankee intact irradiated fuel assembly.

(viii) Item 5(b)(1)(ix) above: One irradiated MARK 42 fuel assembly in either intact or sectioned form, using Configuration (C) above. If sectioned, each section must be seal welded in a shipping can as shown on Martin Marietta Energy Systems Drawing Nos. M-12821-CP-105E, Rev. 0, and M-12821-CP-106E, Rev. 1. Four shipping cans will be loaded into a MARK 42 Segment Dry Shipping Canister as shown on Martin Marietta Energy Systems Drawing No. M-12821-CP-102, Rev. 1, along with a shipping canister spacer, as shown on Martin Marietta Energy Systems Drawing No. M-12821-CP-103, Rev. 1. The shipping canister will be loaded on top of a carrier spacer as shown on Martin Marietta Energy Systems Drawing No. M-12821-CP-112, Rev. 0. A maximum of 2 shipping canisters may be loaded into a cask.

5.(b)(2)(viii)(continued)

Intact fuel assemblies will be shipped in a MARK 42 Element Wet Shipping Canister as shown on Martin Marietta Energy Systems Drawing No. M-12821-CP-114, Rev. 0. A maximum of one intact assembly may be loaded into a cask.

(ix) Item 5(b)(1)(x) above: Two MARK 22 fuel assemblies or one MARK 22 fuel assembly with the two cores separated, using Configuration (C) above. Each assembly or core will be shipped in a shipping canister as shown on Sandia National Laboratory Drawing No. R21563, Sheet 1, Iss. B.

(c) Criticality Safety Index 100

6. Irradiated fuels described in items 5(b)(1)(i), 5(b)(1)(ii), 5(b)(1)(iii), and 5(b)(1)(iv) above may not have a maximum burnup which exceeds 1.25 times the specified maximum average burnup.

7. The cask cavity and containment vessel (inner container) must be dry (no free water) when delivered to a carrier for transport. Residual moisture must be promptly removed from the cask cavity and containment vessel by the methods described in Section XV of the application. Removal of the residual moisture from cask cavity when package is used in Configurations (B), (C), or (D) is not required providing the decay heat load does not exceed 2.0 kW.

8. For the shipment of irradiated fuel assemblies or a canister of consolidated irradiated fuel, the cask cavity canister of consolidated irradiated fuel (if present), and containment vessel must be promptly inerted following removal of the water from the cavity. For contents not vacuum dried, the cask cavity and containment vessel must be purged at least three times with argon, nitrogen, or helium. Each purge volume must be equivalent to or greater than the cask cavity and containment vessel volume. After the final purge, or following vacuum drying, the cavity and containment vessel must be promptly filled with argon, nitrogen, or helium at 1.0 atm pressure.

9. Known or suspected failed fuel assemblies (rods) and fuel with cladding defects greater than pin holes and hairline cracks must be shipped in Configuration (A).

10. The consolidated fuel canister must be provided with vent and drain lines (openings) to permit free draining of the canister. No valves can be installed on the vent and drain lines.

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

12. 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 75.0 inches.

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

14. The neutron shielding tank must be filled with a mixture of water and ethylene glycol (52% by volume). This mixture must not freeze or precipitate in a temperature range from -40°F to 330°F. The neutron shield tank may be empty when the cask is in Configuration D.

15. The structures used to support the package on the transport vehicle must be as described in the application.

16. Any system used for cooling down the package must be provided with a pressure relief device set so that during the cool-down process, the maximum pressure in the containment vessel cannot exceed 310 psig when the package is used in Configuration (A) or 365 psig when the package is used in Configuration (B).

17. As needed, appropriate component spacers must be used in the cask cavity to limit movement of contents during shipment.

18. Shipping cans used for sectioned MARK 42 irradiated fuel assemblies must be seal welded and must be leak tested to $1x10^{-7}$ std cm^3/sec.

19. 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 Section XV of the application, as supplemented.

    (b) The package shall be maintained and tested in accordance with the maintenance program in Section XVI of the application, as supplemented.

    (c) When the package is to be used for the transport of authorized contents having a decay heat load of greater than 4.0 kW, a 220 psig hydrostatic test of the containment cavity, and a 405 psig hydrostatic test of the water jacket and expansion tank shall be performed as part of the maintenance program as specified in Section XVI of the application.

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

21. Revision No. 40 of this certificate may be used until April 30, 2007.

21. Expiration date: October 1, 2008. This certificate is not renewable.

REFERENCES

Nuclear Assurance Corporation application dated February 27, 1996, as supplemented March 26, 1996; June 9, 1998; March 29, May 20 and August 13, 1999; February 15, 2001; and March 21, 2006.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Robert A. Nelson, Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date April 13, 2006

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
9015	21	71-9015	USA/9015/B ( )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)*

Transnuclear, Inc.
7135 Minstrel Way
Columbia, MD 21045

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Transnuclear, Inc., application dated March 25, 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.: TN-8 AND TN-8L

(2) Description

The TN-8 and TN-8L are lead, steel and resin-shielded irradiated fuel shipping casks. The cask approximates a right circular cylinder 1,718 mm in diameter and 5,516 mm long. The cavity consists of three stainless steel square pressure vessels welded to an end plate and a circular stepped top flange, separated by a T-shaped copper plate and surrounded with B4C + Cu plates. Each cavity is 230 x 230 mm and 4,280 mm long. The main shielding consists of 135 mm of lead, 26 mm of steel, and 150 mm of resin. A wet cement layer is located between the lead and the outer shell. Radial copper fins are welded to the outer shell and cover the surface of the cask between each end drum. The Model No. TN-8 has 150 rows of fins and the Model No. TN-8L has 104 rows of fins.

The lid is a welded stainless steel shell containing lead and resin shields. The pressure vessel is closed and sealed by sixteen, 1-1/4-inch diameter bolts and two silicone rubber or Viton O-rings located within recessed grooves on the top flange. Each extremity of the cask is surrounded by circular stainless steel drums reinforced by radial gusset plates and filled with balsa wood. A disk shaped impact limiter, constructed of carbon steel and balsa wood, is fastened to each drum with four, 1-1/4-inch bolts. The vent and drain lines which penetrate the inner cavity are equipped with positive closures. In addition, all access ports are protected by the impact limiters.

The lid of the cask may be replaced with a modified lid which increases the cavity length to 4,362 mm or to 4,394 mm with the lid plate removed. This arrangement is referred to as "Configuration X."

5.(a)(2) (Continued)
Trunnions are used for lifting and tie-down of the package. The package weighs approximately 36,000 kg.

(3) Drawings

The Model No. TN-8 packaging is constructed in accordance with Transnuclear Drawing No. 9317.01, Rev. J. The Model No. TN-8L is constructed in accordance with Transnuclear Drawing No. 9317.138, Rev. A. The materials of construction and welds shall be in accordance with Annexes A, B, and C to Chapter II of the application.

The lid for Configuration X is constructed in accordance with Transnuclear Drawing Nos. 9040-500-1, Rev. 1, 9040-500-2, Rev. 1 and 9040-500-3, Rev. 0.

(b) Contents

(1) Type and form of material

(i) Irradiated PWR uranium oxide fuel assemblies of the following specifications:

Fuel form	Clad UO2 Pellets
Cladding material	Zr or SS
Maximum initial U content/assembly, kg	469
Maximum average initial U-235 enrichment with Zr cladding, w/o	3.2
Maximum average initial U-235 enrichment with SS cladding, w/o	4.0
Maximum bundle cross section, in	8.5
Maximum active fuel length, in	146
Minimum cooling time, day	150
Maximum weight/fuel assembly, kg	733; and

Group I fuel assemblies

Initial fuel pin pressure at 100°F, psig	250
Maximum average burnup, MWD/MTU	38,500; or

Group II fuel assemblies

Maximum average burnup, MWD/MTU	36,000

For the casks in Configuration X, the minimum cooling time of the fuel assemblies shall be 1,460 days with the lid plate installed and 2,190 days with the lid plate removed.

(ii) Solid non-fissile irradiated hardware. As needed, appropriate component spacers must be used when loading irradiated hardware into the cask cavity to limit movement of the contents during accident conditions of transport.

(iii) Intact BWR and PWR fuel rods. The rods shall be constrained by a basket or grid structure; initial U-235 content shall be less than 15.0 kg per rod bundle; cross sectional area of the rods, tubes, and full length structural material shall not be less than 29.6 square inches; and the bundle cross section shall not be greater than 8.5 inches. Maximum weight per bundle shall not exceed 733 kg. The Group I and Group II burnup limits of paragraph 5.(b)(1)(i) apply.

(2) Maximum quantity of material per package

(i) For the contents described in Item 5.(b)(1)(i), Group I fuel assemblies:

Three PWR assemblies. The maximum decay heat load is not to exceed 35.5 kilowatts per package and 12 kilowatts per assembly for the Model No. TN-8 packaging and 23.7 kilowatts per package and 7.9 kilowatts per assembly for the Model No. TN-8L packaging.

(ii) For the contents described in Item 5.(b)(1)(i), Group II fuel assemblies:

Three PWR assemblies. The maximum decay heat load and the maximum free gas volume are not to exceed the limits listed in the table below:

Decay Heat per Shipment, kw(a)	Maximum Free Gas for 3 Assemblies m^3 (NTP)(b)	Configuration X Maximum Free Gas for 3 Assemblies m^3 (NTP)(b)
1.5	0.558	0.601
3.0	0.543	0.585
9.0	0.483	0.520
15.0	0.441	0.475
21.0	0.408	0.439
27.0	0.384	0.413

Notes: (a) Decay heat load per assembly must not exceed 7.9 kilowatts for Model No. TN-8L packaging.

(b) NTP conditions are 25°C and one (1) bar.

5.(b)(2) (Continued)

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

Three rod bundles. The maximum decay heat load and maximum free gas volume are not to exceed the limits listed in Paragraph 5.(b)(2)(ii).

(c) Criticality Safety Index: 100

6. Group I and Group II fuel assemblies, either Zr or SS clad, and bundles of PWR and/or BWR fuel rods that individually meet all the appropriate specifications of 5.(b)(1)(i), 5.(b)(2)(i), 5.(b)(1)(iii), and 5.(b)(2)(iii) above may be packaged in any combination.

7. PWR assemblies may be shipped either with or without burnable poison rod, thimble plug, or control rod assemblies.

8. As needed, appropriate component spacers may be used in the cask cavity to properly position the fuel assemblies.

9. The maximum weight of the contents (fuel assemblies, component spacers, inserts, irradiated hardware, etc.) must not exceed 2,200 kg.

10. The cask cavity must be dry (no free water) when delivered to a carrier for transport. Residual moisture must be promptly removed from the cask cavity by the methods described in Annex I to Chapter VIII of the application. For contents 5.(b)(1)(i) and 5.(b)(1)(iii), the cavity must be promptly backfilled with 1.0 atm of helium, nitrogen, or argon gas.

11. Known or suspected failed fuel assemblies (rods) and fuel cladding defects greater than pin holes and hairline cracks are not authorized.

12. For contents 5.(b)(1)(ii), the dryness verification test is required but leakage tests for containment assembly verification are not required.

13. The package contents must be so limited that under normal conditions of transport, the total dose rates must not exceed 17 mrem/hr at one meter from the surface of the package.

14. Any system used for cooling down the package must be provided with a pressure relief device set so that the maximum pressure in the containment vessel cannot exceed 7 atmospheres during the cool-down process.

15. The systems and components of each packaging must meet the periodic tests and criteria specified in Chapter VIII of the application. The Keff verification and shielding efficiency verification tests in Chapter VIII of the application must be performed on each packaging within the two year period preceding any shipment of contents listed in 5.(b)(1)(i) and 5.(b)(1)(iii). The Keff verification and shielding efficiency verification tests need not be performed on packaging during periods (which may exceed two years) when only irradiated hardware as specified in 5.(b)(1)(ii) is shipped.

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

    (a) The package must be prepared for shipment and operated in accordance with the Operating Procedures in the application dated March 25, 1991.

    (b) Each package must be tested, repaired, and maintained in accordance with the Acceptance Tests and Maintenance Procedures in the application dated March 25, 1991.

17. All valves, fittings, seals, and relief devices must be of the type, size, model and manufacture as indicated on the design drawings. The resin material must be of the specifications stated in Annex A to Chapter II of the application.

18. In accordance with Annex L to Chapter VIII, at periodic intervals not to exceed two years, the thermal performance of the cask must be analyzed to verify that the cask operation has not degraded below that which is licensed*. Following the initial acceptance tests, the heat source may be that provided by the decay heat from the loading of the package, provided that the heat source is equal to at least 25% of the design heat load for the package. Each cask that fails to meet the thermal acceptance criteria given in Annex L of the application must be withdrawn from service until corrective action can be completed or the license amended to limit the package to a lower heat load.

    *The thermal performance test is not required at periodic intervals when the maximum decay heat load per package does not exceed 25% of the design heat load.

19. The Configuration X lid shall be operated and maintained in accordance with Annex N to Chapter VIII, in the application dated March 25, 1991.

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

21. Revision 20 of this certificate may be used until May 31, 2007.

22. Expiration date: October 1, 2008.

REFERENCES

Transnuclear, Inc., application dated March 25, 1991, and supplements dated April 22, 1991; April 22, 1996; March 22, 2001 and April 26, 2006.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

James R. Hall for

Robert A. Nelson, Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: May 9, 2006

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9016	14	71-9016	USA/9016/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)*

Transnuclear, Inc.
7135 Minstrel Way
Columbia, MD 21045

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Transnuclear, Inc., application dated March 25, 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.: TN-9

(2) Description

The TN-9 is a lead, steel and resin shielded irradiated fuel shipping cask. The cask approximates a right circular cylinder 1,718 mm in diameter and 5,756 mm long. The cavity consists of three rectangular, stainless steel pressure vessels welded to end plates and a circular stepped top flange, separated by thin copper plates. The bays are divided into a total of seven square compartments, 150 x 150 mm and 4,520 mm long. The main shielding consists of 128 mm of lead, 26 mm of steel, and 150 mm of resin. A wet cement layer is located between the lead and the outer shell. Radial copper fins are welded to the outer shell and cover the surface of the cask between each end drum.

The lid is a welded stainless steel shell containing lead and resin shields. The pressure vessel are closed and sealed by sixteen, 1-1/4-inch diameter bolts and two silicone rubber or Viton O-rings located within recessed grooves on the top flange. Each extremity of the cask is surrounded by circular stainless steel drums reinforced by radial gusset plates and filled with balsa wood. A disk shaped impact limiter, constructed of carbon steel and balsa wood, is fastened to each drum with four, 1-1/4-inch bolts. The vent and drain lines which penetrate the inner cavity are equipped with positive closures. In addition, all access ports are protected by the impact limiters. Trunnions are used for lifting and tie-down of the package. The weight of the package is approximately 36,000 kg.

(3) Drawings

The package is constructed in accordance with Transnuclear Drawing No. 9317.03, Rev. J. The materials of construction and welds must be in accordance with Annex A, B, and C to Chapter II of the application.

5.(b) Contents

(1) Type and form of material

(i) Irradiated BWR uranium oxide fuel assemblies of the following specifications:

Fuel form	Clad UO2 Pellets
Cladding material	Zr or SS
Initial fuel pin pressure at 100°F, psig	200
Maximum initial U content/assembly, kg	201
Maximum average initial U-235 enrichment, w/o	2.65
Maximum bundle cross section, in	5.52
Maximum active fuel length, in	144
Average burnup, MWD/MTU	36,500
Minimum cooling time, day	150
Maximum weight/fuel assembly, kg	300

(ii) Solid non-fissile irradiated hardware. As needed, appropriate component spacers must be used when loading irradiated hardware into the cask cavity to limit movement of the contents during accident conditions of transport.

(2) Maximum quantity of material per package

(i) Seven BWR assemblies. The maximum decay heat load per package is not to exceed 24.4 kilowatts and 3.5 kilowatts per assembly. As needed, appropriate component spacers may be used in the cask cavity to properly position the fuel assemblies.

(ii) The maximum weight of the contents (fuel assemblies, component spacers, inserts, irradiated hardware, etc.) must not exceed 2,110 kg.

(c) Criticality Safety Index: 100

6. The cask cavity must be dry (no free water) when delivered to a carrier for transport. Residual moisture must be promptly removed from the cask cavity by the methods described in Annex I to Chapter VIII of the application. For contents 5.(b)(1)(i), the cavity must be promptly backfilled with 1.0 atm of helium, nitrogen, or argon gas.

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 VIII of the application.

   (b) Each package must be tested and maintained in accordance with the Acceptance Test and Maintenance Procedures in Chapter VIII of the application.

8. Known or suspected failed fuel assemblies (rods) and fuel with cladding defects greater than pin holes and hairline cracks are not authorized.

9. For contents 5.(b)(1)(ii), the dryness verification test is required but leakage tests for assembly verification are not required.

10. The package contents must be so limited that under normal conditions of transport, the total dose rates must not exceed 14 mrem/hr at one meter from the surface of the package.

11. Any system used for cooling down the package must be provided with a pressure relief device set so that the maximum pressure in the containment vessel cannot exceed 7 atmospheres during the cool-down process.

12. The systems and components of each packaging must meet the periodic tests and criteria specified in Chapter VIII of the application. Each packaging that fails to meet these criteria must be withdrawn from service until corrective action has been completed.

13. All valves, fittings, seals, and relief devices must be of the type, size, model, and manufacture as indicated on the design drawings. The resin material must be of the specifications stated in Annex A to Chapter II of the application.

14. In accordance with Annex L to Chapter VIII, at periodic intervals not to exceed two years, the thermal performance of the cask must be analyzed to verify that the cask operation has not degraded below that which is licensed*. Following the initial acceptance tests, the heat source may be that provided by the decay heat from the loading of the package, provided that the heat source is equal to at least 25% of the design heat load for the package. Each cask that fails to meet the thermal acceptance criteria given in Annex L of the application must be withdrawn from service until corrective action can be completed or the license amended to limit the package to lower heat load.

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* The thermal performance test is not required at periodic intervals when the maximum decay heat load per package does not exceed 25% of the design heat load.

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

16. Revision No. 12 and Revision No. 13 of this certificate may be used until May 31, 2007.

17. Expiration date: October 1, 2008.

REFERENCES

Transnuclear, Inc., application dated March 25, 1991, and supplements dated April 22, 1991; April 22, 1996; March 22, 2001, and April 26, 2006.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

for Robert A. Nelson, Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: May 19, 2006

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
9023	9	71-9023	USA/9023/B( )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)*

NAC International, Inc.
3930 East Jones Bridge Road
Norcross, GA 30092-2107

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Nuclear Assurance Corporation, application dated November 18, 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.: NLI-10/24

(2) Description

A lead, water, depleted uranium and high temperature polymer shielded shipping cask, encased in stainless steel, equipped with balsa impact limiters, and mounted to a railcar which is considered to be an integral part of the packaging for normal conditions of transport. The cask body is 204.5 inches long by 96 inches in OD. The principal shielding consists of 6 inches of lead and 9 inches of water. Depleted uranium plates are encased in the bottom end forging and cask inner closure head. High temperature polymer sheet is encased in the bottom end and positioned between the inner and outer closure heads at the top end.

The lead shield is bonded between 0.75-inch stainless steel inner shell and a 2-inch stainless steel outer shell. The outer shell is surrounded by a 0.75-inch stainless steel water jacket shell. The three shells are welded to stainless steel forgings at both ends. Four water expansion tanks are mounted to the railcar and are connected to the water jacket by a flexible metal hose.

The primary containment vessel is comprised of the 0.75-inch inner shell and the inner closure head. It is 179.5 inches long and has a 45-inch inside diameter. The inner closure head is held in place by sixteen bolts and is sealed with a metallic O-ring. Secondary containment is provided by the outer closure head which is bolted and has a Viton or silicone O-ring seal. There is no direct penetration between the containment cavity and the ambient. The two penetrations into the containment cavity are from the space between the inner and outer closure heads, which has a single penetration through the cask body connecting it with the ambient. The two lid penetrations are sealed with 1.5-inch quick-disconnect valves and metal O-ring seals each in a valve box arrangement.

5.(a)(2) con't

The radioactive contents are positioned within the containment cavity using neutron poisoned aluminum baskets and internal support structures. The PWR and BWR fuel basket cavities are lined with neutron absorber sleeves composed of a silver-indium-cadmium (80-15-5 w/o) alloy.

An auxiliary cooling system, mounted to the railcar, is used to maintain the cask and fuel temperatures so as to facilitate handling and cooldown.

The fully loaded cask, excluding the railcar, is approximately 194,000 pounds, which includes a maximum gross weight of the cavity contents of 34,100 pounds (fuel, spacers, fuel basket, etc.).

(3) Drawings

The Model No. NLI-10/24 shipping cask is constructed in accordance with the NL Industries, Inc., and National Lead Company Drawing Nos. as specified on page XVIII-1, Rev. 9, and page XVIII-2, Rev. 8, in Section XVIII of the application.

(b) Contents

(1) Type and form of material

Irradiated PWR and BWR uranium oxide fuel assemblies of the following specifications:

	PWR	BWR
Fuel form	Clad UO_2 pellets	Clad UO_2 pellets
Cladding material	Zr or SS	Zr or SS
Maximum initial U content/assembly, kg	475	200
Maximum average initial U-235 enrichment, w/o	3.5	2.8
Maximum initial U-235 content/assembly, kg	16.6	5.6
Maximum bundle cross section, inches	9.00	5.75
Fuel pin array size, number of pins	14x14/15x15 16x16/17x17	7x7/8x8
Maximum active fuel length, inches	144	144

5.(b)(1) con't

Irradiated PWR and BWR uranium oxide fuel assemblies of the following specifications:

	PWR	BWR
Maximum specific power, kw/kgU	40	27
Maximum average burnup, MWD/MTU	35,500	29,700
Minimum cooling time, days	150	150

The PWR type assemblies may be shipped either with or without control rods.

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

The maximum decay heat load per package not to exceed 70 kilowatts, and:

Ten PWR fuel assemblies or twenty-four BWR fuel assemblies.

Above assemblies must be contained in their respective fuel baskets as shown on NL Industries, Inc., and National Lead Company Drawing Nos.:

70652F, Sheet 1, Rev. 7 PWR Fuel Basket,
Sheet 2, Rev. 5 10/24 Rail Cask
70653F, Sheet 1, Rev. 7 BWR Fuel Basket,
Sheet 2, Rev. 5 10/24 Rail Cask

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 maximum gross weight of the cavity contents must not exceed 34,100 pounds (fuel, spacers, basket, etc.).

7. The containment vessel must be dry (no free water) when delivered to a carrier for transport. Residual moisture must be promptly removed from the containment vessel by the methods described in Section XVI of the application. The containment vessel must be promptly filled with helium to 1.0 atm pressure.

8. Known or suspected failed fuel assemblies (rods) and fuel with cladding defects greater than pin holes and hairline cracks are not authorized.

The cask contents must be so limited under normal conditions of transport that the following measured dose rates be satisfied:

a) at one meter from the external radial midplane surface of the package: 625 times the neutron dose rate plus 2.5 times the gamma dose rate will not exceed 1,000 millirems per hour; and

b) at one meter from the external surface of the bottom of the package: 115 times the neutron dose rate plus 2.0 times the gamma dose rate will not exceed 1,000 millirems per hour.

10. The neutron shielding system and auxiliary cooling system must be filled with a mixture of water and ethylene glycol (53% to 58% by weight ethylene glycol).

11. The neutron shielding system must be equipped with two pressure relief valves (one on the cask and one on an expansion tank) set at 220 psig.

12. Any system used for cooling down the package must be provided with a pressure relief device set so that the maximum pressure in the containment vessel cannot exceed 233 psig during the cooldown process.

13. The systems and components of each packaging must meet the criteria for the periodic tests specified in Section XVII of the application.

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

(i) Each packaging must meet the acceptance tests and be maintained in accordance with Section XVII of the application, and

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

15. Prior to first use, each packaging shall meet the criteria for the acceptance tests specified in Sections XIV and XV of the application, except that the prototype railcar test, meeting the stated design criteria, need be performed only once.

16. Packaging is authorized for rail mode of transport only.

17. Fabrication of new packages or major packaging components, including the fuel basket, is not authorized.

18. Expiration date: July 31, 2008.

REFERENCES

Nuclear Assurance Corporation application dated November 18, 1991.

Supplements dated: February 7, 1992; February 28 and November 25, 1997; and June 11, 2003.

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 July 2, 2003

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
9027	18	71-9027	USA/9027/B(U)-96	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)*

QSA Global Inc.
40 North Avenue
Burlington, MA 01803

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

AEA Technology/QSA Inc. application dated August 31, 2005, 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.: 741-OP

(2) Description

The Model No. 741-OP consists of a gamma ray projector within a protective carbon steel container. The protective container is of welded steel construction and is approximately 32 inches long, 19 inches wide, and 18.5 inches high. Polyurethane foam and wood inserts locate the Model No. 741 series projectors in the center of the container and provide impact protection.

The 741 series projectors include the Model Nos. 741, 741E, 741A, 741AE, 741B and 741BE. The primary components of the projector consist of an outer steel shell, internal bracing, polyurethane foam, depleted uranium shield, and an "S" tube. The radioactive contents are securely positioned in the "S" tube by a source cable locking device and shipping plug. A 1/4-inch thick steel shipping plate is bolted over the source locking mechanism for additional protection during transport. Tamper-proof seals are provided on the outer steel container. The dimensions of the projector are approximately 19 1/8 inches long, 13 7/8 inches wide, and 11 3/8 inches in height. The maximum weight of the package is 510 pounds, and the maximum weight of the projector is 360 pounds.

(3) Drawings

The package is constructed in accordance with QSA Global Inc. Drawing Nos. R74190, Rev. G, Sheets 1-7; and R741-OP, Rev. E, Sheets 1-7.

5. (b) Contents
   (1) Type and form of material

   Cobalt-60 or iridium-192 as sealed sources which meet the requirements of special form radioactive material.

   (2) Maximum quantity of material per package.

   33 curies of cobalt-60; or
   240 curies of iridium-192 (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."

6. The source shall be secured in the shielded position of the packaging by the source assembly lock, lock cap and safety plug assembly. The source assembly lock, lock cap and safety plug assembly must be fabricated of materials capable of resisting a 1475°F fire environment for one half hour and maintaining their positioning function. The locking ball 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. The 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:

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

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

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

10. Revision No. 17 of this certificate may be used until August 31, 2007.

11. Expiration date: August 31, 2011.

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9027	18	71-9027	USA/9027/B(U)-96	3	OF	3

REFERENCES

AEA Technology/QSA, Inc. application dated August 31, 2005.

Supplements dated: October 25, 2005, February 20, July 17, August 11, and August 15, 2006.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

James R. Hall for

Christopher M. Regan, Acting Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: August 25, 2006

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9030	10	71-9030	USA/9030/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)*

Department of the Navy
Naval Sea Systems Command
Detachment
Radiological Affairs Support Office
PO Drawer 0260
NWS Yorktown, VA 23691-0260

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Teledyne Energy Systems application
dated November 12, 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.

Packaging

(1) Model Nos.: MW-3000 and Sentinel-8

(2) Description

The packages are thermoelectric generators. The major components include: the main housing, tungsten shield, housing flange, and electrical connectors. The approximate dimensions and weights for the Model Nos. are as follows:

Model No.	Dimension (inch)	Weight (lb)
MW-3000	24 OD x 23	2,700
Sentinel-8	24 OD x 25	3,200

(3) Drawings

The packagings are constructed in accordance with the following Drawing Nos.:

Model No.	Drawing Nos.
MW-3000	Martin Co. Drawing No. 471A1000000
Sentinel-8	Isotopes, Inc. Drawing No. J-30856-003-10000

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
9030	10	71-9030	USA/9030/B( )	2	OF	3

(b) Contents

(1) Type and form of material

Strontium 90 titanate doubly encapsulated in Hastelloy fuel capsule which meet the requirements of special form radioactive material.

(2) The maximum quantity of material per package

Model No.	Quantity
MW-3000	25,000 Curies
Sentinel-8	40,000 Curies

6. Eye-bolts shall be removed or covered during transportation to prevent their use as tie-down devices of packages.

7. The MW-3000 and Sentinel-8 shall have their top steel cover plate bolted to the outer wrought steel shield at all times except when maintenance operations are being performed on the generator which require removal of the top steel cover plate.

8. Fabrication of additional units is not authorized.

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 supplement dated February 1, 1991.

   (b) The package shall be maintained in accordance with the maintenance program in the supplement dated February 1, 1991.

10. The packages authorized by this certificate are hereby approved for use under the general license provisions of 10 CFR 71.17.

11. Expiration date: October 1, 2008. This certificate is not renewable.

12. Revision No. 9 of this certificate may be used until November 1, 2006.

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
9030	10	71-9030	USA/9030/B( )	3	OF	3

REFERENCES

Teledyne Energy Systems application dated November 12, 1990.

Teledyne supplement dated: February 1, 1991.

Department of the Navy supplement dated: February 7, 1994; September 20, 1995; April 16, 1998; April 27, 2000; and September 27, 2005.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Robert A. Nelson, Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: October 19, 2005

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
9034	13	71-9034	USA/9034/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)*

General Atomics
P.O. Box 85608
San Diego, CA 92186-9784

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

General Atomics application dated October 4, 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.: TRIGA-I

(2) Description

TRIGA fuel element shipping container. The outer packaging is a steel drum, approximately 22.5-inches in diameter by 39-1/4 inches high. The inner vessel is a 5-inch Schedule 40 carbon steel pipe. Dimensions of the inner vessel are approximately 31 inches in height with a 1/4-inch thick wall and a 5-inch inside diameter. The top of the inner vessel is a threaded pipe cap and the bottom is a welded 1/4-inch thick flat disc. The inner vessel is centered and supported within the outer packaging by eight, 3/8-inch diameter braced, support spacer rods. The void between the inner vessel and the outer packaging is filled with vermiculite tamped to a minimum density of 4.5 lbs/ft^3. Maximum gross weight including contents is approximately 235 pounds.

(3) Drawing

The packaging is constructed in accordance with General Atomic Company Drawing No. TOS396C160, Rev. G.

5. (b) Contents

(1) Type and form of material

TRIGA fuel elements containing uranium-zirconium-hydride or erbium-uranium-zirconium-hydride with nominal fuel composition (excluding erbium content) as described in Table A.1-1 of the October 4, 1995 application, and clad with stainless steel, aluminum or incoloy. Uranium enriched to a maximum 93.5 w/o in the U-235 isotope. The H to Zr atomic ratio within the fuel meat must not exceed 1.65.

(2) Maximum quantity of material per package

U-235 content not to exceed 1.39 kg, contained in a maximum of 7 1.5-inch diameter fuel elements, or a maximum of 25 0.5-inch diameter fuel elements, with nominal fuel composition (excluding erbium content) as described in Table A.1-2 (Rev. 1) of the October 4, 1995, application. For enrichments greater than 5 weight percent U-235, uranium content not to exceed an A_2 quantity.

(c) Criticality Safety Index 0.4

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 of Chapter 8 of the application.

(b) The packaging must meet the Acceptance Tests and Maintenance Program of Chapter 9 of the application.

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

8. Expiration date: December 31, 2010.

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
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REFERENCES

General Atomic Company application dated October 4, 1995.

Supplements dated: December 5, 1995, October 16, 2000, and November 16, 2005.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Robert A. Nelson, Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: December 8, 2005

UNITED STATES NUCLEAR REGULATORY COMMISSION

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
9035	20	71-9035	USA/9035/B(U)-96	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)*
QSA Global Inc.
40 North Avenue
Burlington, MA 01803

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION
AEA Technology/QSA, Inc., application dated August 29, 2005.

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.: 680-OP

(2) Description

The Model No. 680-OP consists of a gamma ray projector within a protective steel container. The protective container is of welded steel construction and is approximately 32 inches long, 19 inches wide, and 18-1/2 inches high. Polyurethane foam and wood inserts locate the Model 680 series projectors in the center of the container and provide impact protection.

The 680 series projectors include the Model Nos. 680, 680E, 680A, 680AE, 680B and 680BE. The primary components of the projector consist of an outer steel shell, internal bracing, polyurethane foam, depleted uranium shield, and an "S" tube. The radioactive contents are securely positioned in the "S" tube by a source cable locking device and shipping plug. A 1/4-inch thick steel shipping plate is bolted over the source locking mechanism for additional protection during transport. Tamper-proof seals are provided on the outer steel container. The dimensions of the projector are approximately 21 inches long, 14-5/8 inches wide, and 11-13/16 inches high. The maximum weight of the package is 615 pounds, and the maximum weight of the projector is 465 pounds.

(3) Drawings

The packaging is constructed in accordance with QSA Global Inc., Drawing Nos. R68090, Sheets 1-7, Rev. H, and R680-OP, Sheets 1-7, Rev. G.

5.(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:

110 curies (4.1 TBq) (output)

Output curies are determined by measuring the source output at 1 meter and expressing its activity in curies derived from the following: 1.30 R/h-Ci cobalt-60 at 1 meter. (Ref: American National Standards Institute, 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 safety plug assembly. The source assembly lock, lock cap and safety plug assembly must be fabricated of materials capable of resisting a 1475°F fire environment for one half hour and maintaining their positioning function. The locking ball 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. The 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:

(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 Section 7 of the application.

9. Revision No. 19 of this certificate may be used until August 31, 2007.

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

11. Expiration date: June 30, 2010.

REFERENCES

AEA Technology/QSA, Inc., application dated August 29, 2005.

Supplements dated: October 25, 2005, February 20, August 1, August 11, and August 15, 2006.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Christopher M. Regan, Acting Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: September 8, 2006

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
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2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

Source Production & Equipment Co.
113 Teal Street
St. Rose, LA 70087-9691

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Source Production & Equipment Company application dated February 28, 2001.

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.: C-1

(2) Description

The packaging consists of a steel inner unit inside an outer overpack. The inner unit is a rectangular box approximately 9" high x 7.5" wide x 7.5" deep around a depleted uranium shield. All fittings and source locking components are protected and enclosed within the 1/8" carbon steel outer shell. The inner receptacle consists of a uranium shield equipped with two closed bottom Zircalloy or titanium "J" tubes, each of which may house one "pigtail type" special form source. The overpack is a 12-gallon, 20- or 22-gage steel drum partially filled with foam. The weight of the inner unit is 51 to 70 lbs. The weight of the overpack is 19 to 22 lbs. Up to 8 lbs. of ancillary equipment may be included within the overpack. The maximum gross weight of the package is 100 lbs.

(3) Drawings

The package is constructed in accordance with Source Production & Equipment Company Inc. Drawing Nos. B322000, Rev. (3); B311000, Rev. (2); B311001, Rev. (1); and B311002, Rev. (0).

(b) Contents

(1) Type and form of material

Iridium-192, Selenium-75, and Ytterbium-169 as sealed sources that meet the requirements of special form radioactive material.

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
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(b) Contents cont'd

(2) Maximum quantity of material per package

Two sealed sources with a combined activity not to exceed 300 curies.

6. Tungsten shield pads, with dimensions up to approximately 2-inches diameter and 1/2-inch thick, may be welded to the inside surface of the source changer housing.

7. The nameplate shall 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 shall be prepared for shipment and operated in accordance with the Operating Procedures of Section 7.0 of the consolidated application dated February 28, 2001, as supplemented June 23, 2006.

   b. The package must meet the Acceptance Tests and Maintenance Program of Section 8.0 of the consolidated application dated February 28, 2001, as supplemented June 23, 2006.

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

10. Revision No. 10 of this certificate may be used until October 31, 2007.

11. Expiration date: October 31, 2011.

REFERENCES

Source Production & Equipment Company applications dated September 27, 2000, and February 28, 2001.

Supplements dated: April 11 and May 11, 2001; and May 1, June 14 and June 23, 2006.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Christopher Regan, Acting Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: August 7, 2006.

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
9037	13	71-9037	USA/9037/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)*

General Atomics
P.O. Box 85608
San Diego, CA 92186-9784

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

General Atomics application dated October 4, 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.: TRIGA-II

(2) Description

TRIGA fuel element shipping container. The outer packaging is a steel drum, approximately 22.5 inches in diameter by 57.5 inches high. The inner vessel is a 5-inch Schedule 40 carbon steel pipe. Dimensions of the inner vessel are approximately 50 inches in height with a 1/4-inch thick wall and a 5-inch inside diameter. The top of the inner vessel is a threaded pipe cap and the bottom is a welded 1/4-inch thick flat disc. The inner vessel is centered and supported within the outer packaging by eight, 3/8-inch diameter braced, support spacer rods. The void between the inner vessel and the outer packaging is filled with vermiculite tamped to a minimum density of 4.5 lbs/ft^3. Maximum gross weight including contents is approximately 330 pounds.

(3) Drawing

The packaging is constructed in accordance with General Atomic Company Drawing No. TOS396C161, Rev. F.

5. (b) Contents

(1) Type and form of material

Special function TRIGA fuel elements containing uranium-zirconium-hydride or erbium-uranium-zirconium-hydride whose fuel portion has nominal compositions (except erbium content) as described in Table A.1-1 of the October 4, 1995, application, and clad with stainless steel, aluminum or incoloy. Uranium enriched to a *maximum* 93.5 *w/o in the* U-235 *isotope. The* H *to* Zr *atomic ratio within the fuel* meat must not exceed 1.65.

(2) Maximum quantity of material per package

U-235 content not to exceed 1.39 kg, contained in a maximum of 7 1.5-inch diameter fuel elements, or a maximum of 25 0.5-inch diameter fuel *elements, whose fuel portion has nominal compositions (except erbium content) as* described in Table A.1-2 (Rev. 1) of the October 4, 1995, application. For enrichments greater than 5 weight percent U-235, uranium content not to exceed an A_2 quantity.

(c) Criticality Safety Index 0.4

6. 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 of Chapter 8 of the application.

(b) The packaging must meet the Acceptance Tests and Maintenance Program of Chapter 9 of the application.

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

8. Expiration date: December 31, 2010.

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
9037	13	71-9037	USA/9037/AF	3	OF	3

REFERENCES

General Atomic Company application dated October 4, 1995.

Supplements dated: December 5, 1995, October 16, 2000, and November 16, 2005.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Robert A. Nelson, Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: December 7, 2005

NRC FORM 618
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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
9056	12	71-9056	USA/9056/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)*

Source Production and
Equipment Company, Inc.
113 Teal Street
St. Rose, LA 70087

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Source Production and Equipment Company, Inc. application dated March 24, 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.

) Packaging

(1) Model No.: SPEC 2-T

(2) Description

A steel encased, uranium shielded Gamma Ray Projector. Primary components consist of an outer steel shell, internal bracing, depleted uranium shield, and a Zircalloy "S" tube. The contents are securely positioned in the Zircalloy "S" tube by a source cable locking device and shipping plug. The unit resembles a rectangular box approximately 13-3/8" long by 4-11/16" high by 4-3/8" wide with a maximum gross weight of 56 pounds.

(3) Drawings

The packaging is constructed in accordance with Source Production and Equipment Company, Inc. Drawing Nos. 12688-1, Rev. (2); 788-1, Rev. (4); and 788-2, Rev. (0).

The packaging may also be as shown in Source Production and Equipment Company Drawing No. 1000; Rev. (0), provided fabrication was completed prior to June 8, 1989.

The overpack is a 12 gallon open head 20 or 22 gauge National Motor Freight Classification 100-H, or succeeding issues, Item 260 steel drum constructed in accordance with Source Production and Equipment Company, Inc. Drawing No. 53189-2, Rev. (2).

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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
9056	12	71-9056	USA/9056/B(U)	2	OF	3

5.(b) Contents

(1) Type and form of material

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

(2) Maximum quantity of material per package

225 curies

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 and 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 source assembly ball stop 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.

The nameplates must be fabricated of materials capable of resisting the fire test of 10 CFR Part 71 and maintaining their legibility.

8. For transportation of more than 45 curies per package in private carriage the shipment must be in accordance with 49 CFR 173.441(b).

9. For transportation of more than 45 curies per package by a common carrier, the package must be within a protective overpack as described and constructed in accordance with 5(a)(3).

10. 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 of Section 7.0 of the application, as supplemented; and

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

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

12. Expiration date: April 30, 2010.

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
9056	12	71-9056	USA/9056/B(U)	3	OF	3

REFERENCES

Source Production and Equipment Company, Inc. application dated March 24, 2000.

Supplements dated: March 30, 2000, and March 14, 2005.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Robert J. Lewis, Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: 04 April 2005

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
9067	7	71-9067	USA/9067/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-3

(2) Description

Steel encased, lead shielded shipping package. The packaging is provided with a recessed, plug-type lid and a 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 dimensions, weight, and shielding are as follows:

Exterior height, in.	26.4
Exterior diameter, in.	19.0
Cavity height, in.	10.5
Cavity diameter, in.	4.5
Lead shielding, in.	6.0
Loaded weight, lb.	2,800 (Incl 110-1b. skid)

(3) Drawings

The packaging is constructed in accordance with Battelle Memorial Institute Drawing No. BCL3-01, Sheets 1 & 2, Rev. C.

The inner can assembly is constructed in accordance with Battelle Memorial Institute Drawing No. BCL3-38, 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 300 watts decay heat, and

(i) Fissile material not to exceed 100 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. 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.

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

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 three (3) feet from the external surface of the package.

9. The maximum gross weight of the cavity contents must not exceed 40 pounds (inner can assembly, radioactive material, etc.).

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9067	7	71-9067	USA/9067/B( )F	3	OF	3

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) Each package shall be operated and prepared for shipment 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: September 30, 2007.

REFERENCES

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

Supplement dated: April 10, 1992; January 27 and August 18, 1997; and July 29, 2002.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

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

Date: September 20, 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
9070	17	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.
1102 Broadway Plaza, Suite 300
Tacoma, WA 98402-3526

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.

(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 71.15. 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.

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

13. Expiration date: January 31, 2010.

NRC FORM 618 (8-2000) 10 CFR 71		CERTIFICATE OF COMPLIANCE FOR RADIOACTIVE MATERIAL PACKAGES		U.S. NUCLEAR REGULATORY COMMISSION
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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, and December 17, 2004.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

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

Date: January 18, 2005

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	14	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 26-1/2" 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.

5. (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

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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, 2008.

REFERENCES

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

Supplements dated: November 24, 1992; October 31, 1997; July 28, 1999; January 5, 2000; April 23, 2001; and December 17, 2002.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

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

ate: January 27, 2003

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
9099	10	71-9099	USA/9099/B(U)F-85	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)*

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
Application dated January 27, 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.: 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.

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

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, 2009.

UNITED STATES NUCLEAR REGULATORY COMMISSION

REFERENCES

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

Supplements dated: February 18, 1999, April 27, 2000, and December 5, 2003.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

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

'te: 12/30/2003

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

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2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

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.

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 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 3/8-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 closure 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 package 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. 240116, Rev G.

(b) Contents

(1) Type and form of material

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

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5. (b) Contents (continued)

(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.

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 October 7, 2003.

(b) The package must meet the Acceptance Test and Maintenance program in the supplement dated October 7, 2003.

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

10. Expiration date: October 1, 2008. This package is not renewable.

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; September 23, 1998; September 29 and October 7, 2003.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Meraj Rahimi

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

ate: March 11, 2005

NRC FORM 618 (3-2000) 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
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2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

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.

(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.

(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% $(Pu\text{-}U_2)C_3$. 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) Criticality Safety Index: 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
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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-4-06230, Sheet 1, Revision No. 0) must be torqued to 70 ± 10 ft-lbs.

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

12. Revision No. 14 of this certificate may be used until April 30, 2007.

13. Expiration Date: April 30, 2011.

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; February 22, 2001; and February 16, 2006.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Robert A. Nelson, Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: March 17, 2006

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
9148	7	71-9148	USA/9148/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)*

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

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

AEA Technology QSA, Inc. application dated December 21, 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.: 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 19.75 inches high. The radioactive source assembly is housed in a 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 970 pounds.

(3) Drawing

The packaging is constructed in accordance with AEA Technology QSA, Inc. Drawing No. R77090 - Sheets 1 through 6, Rev. D.

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

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(b) Contents

(1) Type and form of material

Sources which meet the requirements of special form radioactive material. Authorized isotopes include Ir-192, Co-60, Sc-46, and Cs-137

(2) Maximum quantity of material per package

Isotope	Output Curies
Ir-192	1,000
Co-60	800
Sc-46	800
Cs-137	1,000

(3) Maximum decay heat per package:

14 watts.

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

7. 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 the application.

8. The packaging authorized by this certificate is hereby approved for use under the general license provision of 10 CFR 71.17.

NRC FORM 618
(8-2000)
10

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
9148	7	71-9148	USA/9148/B(U)-85	3	OF	3

9. Expiration date: March 31, 2008.

REFERENCES

AEA Technology QSA, Inc. application dated December 21, 2001.

Supplements dated: September 13, November 7, and November 14, 2002; September 14, 2004.

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: February 23, 2005

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
9150	7	71-9150	USA/9150/B(U)-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)*

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 matting 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 grams 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 maybe 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 provision of 10 CFR 71.17.

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

16. Revision 6 of this Certificate may be used until September 30, 2007.

17. Expiration date: September 30, 2011.

REFERENCES

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; May 24, 2001; and June 1, 2006.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Robert A. Nelson, Chief
Licensing Branch
Division of Spent Fuel Storage
and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: 10/10/06

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9152	14	71-9152	USA/9152/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)*
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

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.

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 4.4.2-1 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) Criticality Safety Index

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:

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

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.

(b) 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.

11. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.12, until October 1, 2004, and under provisions of 10 CFR 71.17 thereafter.

12. Expiration date: October 1, 2008. This certificate is not renewable.

REFERENCES

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

Department of Energy supplements dated: May 12, 1989; April 11, 1994; March 24, 1999; October 14, 2003; March 3, 2004, and May 26, 2004.

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 June 14, 2004

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
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2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

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.

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 50 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. 4 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.

5. (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 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.

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 until October 1, 2004, and 10 CFR 71.17, thereafter.

10. Expiration date: September 30, 2009.

REFERENCES

Industrial Nuclear Company application dated June 8, 1999.

Supplements dated: June 9, August 6 and September 14, 1999, October 24, 2003, and August 20, 2004.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

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

/RA/

Date: September 24, 2004

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	15	71-9168	USA/9168/B(U)	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 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.

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(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. 12.

(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 71.15 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 71.14(b)(3)(i), 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 71.17.

12. Expiration date: June 30, 2010.

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
9168	15	71-9168	USA/9168/B(U)	4	OF	4

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 May 10, 2005.

Duratek supplements dated: April 23, 2001; October 31 and November 26, 2002; April 4 and November 6, 2003.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Robert Lewis, Chief
Spent Fuel Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

ate 25 May 2005

UNITED STATES 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
9184	6	71-9184	USA/9184/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.
1102 Broadway Plaza, Suite 300
Tacoma, WA 98402-3526

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

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

5.(a) Packaging Continued

(2) Description continued

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.

(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, until October 1, 2004, and under the provisions of 10 CFR 71.17 thereafter.

8. Expiration date: July 31, 2009.

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 OF PAGES
9184	6	71-9184	USA/9184/B(U)	3 OF 3

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 and March 16, 2004.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

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

UNITED STATES NUCLEAR REGULATORY COMMISSION

Date: May 10, 2004

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
9185	6	71-9185	USA/9185/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)*

Industrial Nuclear Company
14320 Wicks Blvd.
San Leandro

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Industrial Nuclear Company application dated July 1, 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.: 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. 4, IR 50-1A, Rev. 2, IR 50-1B, Rev. 1, IR 100-1A, Rev. 4, and IR 100-1B, Rev. 2.

5. (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 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: December 31, 2008.

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
9185	6	71-9185	USA/9185/B(U)-85	3	OF	3

REFERENCES

Industrial Nuclear Company application dated July 1, 1999.

Supplements dated: September 14 and December 29, 1999; and October 24, 2003.

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 03, 2003

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 (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
9187	7	71-9187	USA/9187/B(U)-96	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)*

QSA Global, Inc.
40 North Avenue
Burlington, MA 01803

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

QSA Global, Inc. application dated March 6, 2006, 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.: 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) Drawing

The packaging is constructed in accordance with QSA Global Drawing No. R86590, Sheets 1 through 8, Rev. G.

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9187	7	71-9187	USA/9187/B(U)-96	2	OF	2

(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."

6. 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 Section 7 of the application; and,

(b) Each packaging shall be maintained in accordance with the Acceptance Tests and Maintenance Program in Section 8 of the application.

7. The packaging authorized by this certificate is hereby approved for use under the general license provision of 10 CFR 71.17.

8. Revision No. 6 of this certificate may be used until October 31, 2007.

9. Expiration date: December 31, 2008.

REFERENCES

QSA Global, Inc. application dated March 6, 2006.

Supplement(s) dated: August 24, and September 28, 2006.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Robert A. Nelson, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: October 24, 2006

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9196	22	71-9196	USA/9196/AF-96	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

Duratek, Inc., application dated June 9, 2005, 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) an ANSI N14.1 Standard 30C cylinder.

The ANSI N14.1 Standard 30C 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 Duratek, Inc., Drawing No. C-110-B-57922-0002, Sheets 1 through 3, Rev. 3.

(b) Contents

(1) Type and form of material

UF_6 enriched in the U-235 isotope.

(2) Maximum quantity of material per package

ANSI Standard N14.1 30B or 30C cylinder: 5,020 pounds UF_6 enriched to not more than 5 weight percent in the U-235 isotope. The maximum H/U atomic ratio for the UF_6 is 0.088.

(c) Criticality Safety Index (CSI)

Criticality safety index for the UX-30 overpack containing a standard ANSI N14.1 30B cylinder	5.0
Criticality safety index for the UX-30 overpack containing a standard ANSI N14.1 30C 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 N14.1-2001 or an earlier version of ANSI N14.1 in effect at the time of fabrication or b) American National Standard N14.1-2001 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 ANSI N14.1 Standard 30C cylinder (new or retrofitted cylinders) must be fabricated, inspected, tested, and maintained in accordance with ANSI N14.1-2001 Addendum 2-2004.

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 weather/dust seal gasket between the upper and lower shells must be inspected and must be replaced if inspection shows excessive wear or any defects to the gasket.

(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 and plug threads 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. Transport by air is not authorized.

12. Packagings may be marked with Package Identification Number USA/9196/AF-85 until October 31, 2007 and must be marked with Package Identification Number USA/9196/AF-96 after October 31, 2007.

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

14. Revision No. 21 of this certificate may be used until November 1, 2006.

15. Expiration date: February 28, 2010.

REFERENCES

Duratek Inc., application dated: June 9, 2005.

Duratek Inc., supplements dated: June 30 and September 9, 2005.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Robert Nelson, Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

10/14/05

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	12	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) Criticality Safety Index: 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, as supplemented.

(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 this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

14. Revision No. 11 of this certificate may be used until June 30, 2007

15. Expiration date: June 30, 2011.

REFERENCES

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

Supplements dated: April 9 and 15, 1991.
Department of Energy supplements dated: February 21, 1996; February 1, 2001; October 14, 2003; March 3, 2004 and February 16, 2006.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Robert A. Nelson, Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date June 20, 2006

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
9203	14	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 January 20, 2006.

4. CONDITIONS

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

5.

(a) Packaging

(1) Model No.: 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) Criticality Safety Index 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. 5.

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.

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 71.17.

12. Revision No. 13 of this certificate may be used until January 31, 2007.

Expiration date: February 28, 2011.

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
9203	14	71-9203	USA/9203/AF	4 OF	4

REFERENCES

Framatome Cogema Fuels applications dated October 5, 2005, and January 20, 2006.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Robert A. Nelson, Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: January 27, 2006

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9204	11	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. 2.

(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 71.15, 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 71.14(b)(3)(I), 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 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;

(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 Transuranic Content Codes," for each site specific appendix to Appendix 4.10.2 as listed below:

Appendix 4.10.2.1	Compliance Methodology for TRU Waste From Battelle Colombus Laboratories,
Appendix 4.10.2.2	Compliance Methodology for TRU Waste From Missouri University Research Reactor,
Appendix 4.10.2.3	Compliance Methodology for TRU Waste Form Energy Technology Engineering Center,
Appendix 4.10.2.4	Compliance Methodology for TRU Waste From Lawrence Livermore National Laboratory,

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
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Appendix 4.10.2.5 *Compliance Methodology for TRU Waste From Idaho National Engineering and Environmental Laboratory;*

(3) One or more filter vents must be installed in the drum 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 containers authorized for shipment of TRU in the Model No. CNS 10-160B are the 30-gallon and the 55-gallon drum. Up to ten payload containers of TRU waste may be packaged in the cask.

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

13. Revision No. 10 of this certificate may be used until December 9, 2006.

14. Expiration date: October 31, 2010.

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, June 14, 2002, August 20, 2004, and March 7, April 8, October 26, December 2 and 7, 2005.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Robert A. Nelson, Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: 12/9/05

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE	PAGES
9208	16	71-9208	USA/9208/B( )	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)*

   NUKEM Corporation
   3800 Fernandina Road, Suite 200
   Columbia, SC 29210-3854

   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 3-½-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 until October 1, 2004, and 10 CFR 71.15, thereafter. 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 71.14(b)(3)(i), the pre-shipment leak test is not required.

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

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

12. Revision No. 15 of this certificate may be used until August 31, 2007. Use of Revision No. 15 beyond August 31, 2007, is authorized only if this certificate has been renewed or is under timely renewal as specified in 10 CFR 71.38(b), or has otherwise not been terminated.

13. Expiration date: August 31, 2007.

REFERENCES

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

RWE NUKEM Corporation supplement dated May 8, 2003.

NUKEM Corporation supplement dated September 6, 2006.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Robert A. Nelson, Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: September 27, 2006

NRC FORM 618 (8-2000) 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
9212	4	71-9212	USA/9212/B(M)F-96	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 TRU Solutions, LLC application dated November 27, 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 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 1-8, Rev. 4.

The fixed lid waste canister is constructed and assembled in accordance with Packaging Technology Drawing No. X-106-501-SNP, Rev. 4. The removable lid waste canister is constructed and assembled in accordance with Packaging Technology Drawing No. X-106-502-SNP, Rev. 2.

(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 RH-TRAMPAC (Revision 0).

(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 limits described in Section 3.1, "Nuclear Criticality" of RH-TRAMPAC (Revision 0). Pu-239 equivalent is determined in accordance with RH-TRAMPAC (Revision 0). 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 RH-TRAMPAC (Revision 0).

(c) Criticality Safety Index: 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 RH-TRAMPAC (Revision 0).

7. Each waste canister must not exceed the decay heat limits determined as specified in RH-TRAMPAC (Revision 0), or must be tested for gas generation in accordance with RH-TRAMPAC (Revision 0), 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 Section 2.4, Filter Vents, of RHTRAMPAC (Revision 0).

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.

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
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11. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

12 Packages may be marked with Package Identification Number USA/9212/B(M)F-85 until July 31, 2007 and must be marked with Package Identification Number USA/9212/B(M)F-96 after July 31, 2007.

13. Revision No. 3 of this certificate may be used until July 31, 2007.

14. This package may not be used for transport by aircraft.

15. Expiration date: February 28, 2010.

REFERENCES

Westinghouse TRU Solutions, LLC, application dated November 27, 2002.

Supplements dated: Washington TRU Solutions, LLC, November 1, 2004; October 14, 2005; and June 5, 2006.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Christopher M. Regan, Acting Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: July 28, 2006

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
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2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

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.

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 wooden overpack with a steel outer shell. 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 overpack is approximately 49 inches in diameter and 59 inches high, including the lid lifting eye and the base support structure. 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. G; and 240122, Sheet 1 of 2, Rev. H, Sheet 2 of 2, Rev. H.

(b) Contents

(1) Type and form of material

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

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5.(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. H:

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. H:

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. H:

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 be maintained in accordance with Teletherapy Shipping Packaging Maintenance Procedure R-2019-G, Revision 0, provided in the supplement dated May 1, 2003.

(b) The package shall be prepared for shipment and operated in accordance with Teletherapy Shipping/Transfer Cask Unloading and Loading Procedures R-2014-G, Revision 0, provided in the supplement dated May 1, 2003.

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 within the wooden overpack.

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U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

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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: May 31, 2008.

<u>REFERENCES</u>

Neutron Products, Incorporated, application dated September 14, 1992.

Supplements dated: October 29, 1992; November 17, 1993; September 8, 1997; September 5, 2002; May 1 and October 7, 2003.

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 18, 2003

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	9	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 until October 1, 2004, and 10 CFR 71.15, thereafter.

(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) Criticality Safety Index

(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.

   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.

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 until October 1, 2004, and 10 CFR 71.17, thereafter.

16. Expiration date: January 31, 2008.

REFERENCES

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

Supplement dated: November 24, 1992; October 31, 1997; March 31, 1999; April 23, 2001; and December 17, 2002.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Stuart W. Brown for

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

Date: July 23, 2004

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
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2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

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)(iii).

(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)(ii).

(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)(ii).

(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)(iii).

5.(c) Criticality Safety Index

Minimum criticality safety 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 71.17.

8. Expiration date: June 30, 2010.

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
9217	13	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; February 6 and August 21, 2001; and December 16, 2004

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Robert Lewis, Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: 24 March 2005

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
9218	18	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

Washington TRU Solutions LLC application dated October 4, 2004, 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. V. The contents are positioned within the packaging in accordance with the Contact-Handled Transuranic Waste Authorized Methods for Payload Control (CH-TRAMPAC), Rev. 2, Section 2.9, "Payload Container/Assembly Configuration Specifications." The standard pipe overpack is constructed and assembled in accordance with Packaging Technology, Inc., Drawing No. 163-001, Rev. 6. The S100 pipe overpack is constructed and assembled in accordance with Packaging Technology, Inc., Drawing No. 163-002, Rev. 4. The S200 pipe overpack is constructed and assembled in accordance with Packaging Technology, Inc., Drawing No. 163-003, Rev. 3. The S300 pipe overpack is constructed and assembled in accordance with Packaging Technology, Inc., Drawing No. 163-004, Sheet 1, Rev. 1. The compacted puck drum spacers needed for the purpose of maintaining subcriticality in 55-, 85-, and 100-gallon drums are constructed and assembled in accordance with Drawing No. 163-006, 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, an 85-gallon drum, a 100-gallon drum, a standard waste box (SWB), a standard pipe overpack, an S100 pipe overpack, an S200 pipe overpack, an S300 pipe overpack, or ten-drum overpack (TDOP). The payload containers are described in CH-TRAMPAC, Rev. 2, Section 2.9, "Payload Container/Assembly Configuration Specifications." 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 CH-TRAMPAC, Rev. 2. For payloads of content code LA 154 and SQ 154, the absence of flammable gas mixtures is ensured as described in Appendix 6.12 of the CH-TRU Payload Appendices, Rev. 1. For payload configurations with an unvented heat-sealed bag layer, the absence of flammable gas mixtures is ensured as described in Appendix 6.13 of the CH-TRU Payload Appendices, Rev. 1.

(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) 550 pounds per S100 pipe overpack,
- (v) 547 pounds per S200 pipe overpack,
- (vi) 547 pounds per S300 pipe overpack,
- (vii) 1,000 pounds per 85-gallon drum,

(viii) 1,000 pounds per 100-gallon drum,
(ix) 4,000 pounds per SWB, or
(x) 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) 14 S300 pipe overpacks,
(vi) 8 85-gallon drums,
(vii) 6 100-gallon drums,
(viii) 2 SWBs, or
(ix) 1 TDOP.

Fissile material not to exceed the limits specified in CH-TRAMPAC, Rev. 2, Section 3.1, "Nuclear Criticality."

The S100, S200, and S300 pipe overpack payloads shall meet the curie limits specified in CH-TRAMPAC, Rev. 2, Section 3.3, "Activity Limits."

Maximum decay heat per package not to exceed 40 watts. Decay heat per payload container not to exceed the values given in CH-TRAMPAC, Rev. 2, Table 5.2-1, "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 Section 5.2.3 of CH-TRAMPAC, Rev. 2. For content code LA 154 and SQ 154 payloads, decay heat per payload container not to exceed the values specified in Appendix 6.12 of CH-TRU Payload Appendices.

5. (c) Criticality Safety Index: 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 CH-TRAMPAC, Rev. 2.

7. Each payload container must be assigned to a shipping category in accordance with CH-TRAMPAC, Rev. 2, Section 5.1, "Payload Shipping Category." For a payload assembly made up of payload containers with the same shipping categories, each payload container and payload assembly must not exceed the allowable wattage in accordance with CH-TRAMPAC, Rev. 2, Section 5.2.3, "Hydrogen Gas Generation Rate and Decay Heat Limits for analytical category" or must be tested for gas generation in accordance with CH-TRAMPAC, Rev. 2, Section 5.2.5, "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 CH-TRAMPAC, Rev. 2, Section 6.2.4, "Mixing of Shipping Categories," and Appendix 2.4 of the CH-TRU Payload Appendices, "Mixing of Shipping Categories

and Determination of the Flammability Index." Each content code LA 154 and SQ 154 payload container must be assigned to a shipping category in accordance with Appendix 6.12 of CH-TRU Payload Appendices. 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 6.12 of CH-TRU Payload Appendices. For a payload of content code LA 154 or SQ 154 containers with different shipping categories, the flammability index of each payload container must not exceed 50,000 in accordance with Appendix 6.12 of CH-TRU Payload Appendices.

8. Payload containers within a package shall be selected in accordance with CH-TRAMPAC, Rev. 2, Section 6.0, "Payload Assembly Requirements." Payload containers of content code LA 154 shall be assembled in accordance with Appendix 6.12 of CH-TRAMPAC, Rev. 2.

9. Each payload container must be vented in accordance with Section 2.5, "Filter Vents," of the CH-TRAMPAC, Rev. 2. Drums which were not equipped with filtered vents during storage must be aspirated in accordance with CH-TRAMPAC, Rev. 2, Section 5.3, "Venting and Aspiration."

10. For close-proximity and controlled shipments meeting the conditions specified in Appendices 3.5 and 3.6, respectively, of CH-TRU Payload Appendices, shipping periods of 20 days and 10 days may be applicable. The shipping period for any mode of transport is not to exceed 60 days. For content code LA 154 and SQ 154 shipments, the shipping period as defined in Appendix 6.12 of the CH-TRU Payload Appendices is not to exceed 5 and 10 days, respectively.

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 6.12 of CH-TRU Payload Appendices.

(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 Sections 7.1.5 and 7.1.6 of the Safety Analysis Report.

(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 71.17.

Expiration date: August 31, 2009.

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
9218	18	71-9218	USA/9218/B(U)F-85	5	OF	5

<u>REFERENCES</u>

Washington TRU Solutions, LLC, October 4, 2004 and March 4 and June 8, 2005.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Michael T. Markley, Acting Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: 07/19/2005

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
9221	5	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.

For contents exceeding a Type A quantity, the radioactive material must be contained within a specimen with intact, undamaged cladding.

(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: 1.0 x grams U-235 + 1.4 x grams U-233 + 1.6 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.

The total quantity of radioactive material in the form of loose surface contamination within the package must not exceed a Type A quantity.

5. (c) Criticality Safety Index: 0.0

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

(a) The package must be 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: October 1, 2008.

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
9221	5	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, S#01-10827 dated March 16, 2001, S#06-01881 dated May 31, 2006, and S#06-03403 dated September 7, 2006.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Robert Nelson, Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: September 28, 2006

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
9225	42	71-9225	USA/9225/B(U)F-96	1	OF	23

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.
3930 East Jones Bridge Road
Norcross, GA 30092

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

NAC International, Inc., application dated August 8, 2005, 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, or two BWR assemblies. In addition, the cask may be used to transport metallic fuel rods, MTR and DIDO fuel assemblies and plates, individual PWR rods, high burnup PWR or BWR rods, TRIGA fuel elements, TRIGA fuel cluster rods, tritium-producing burnable absorber rods (TPBARs), PULSTAR fuel elements, spiral fuel assemblies, and MOATA fuel plate bundles. 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. 5	Cask Assembly
LWT 315-40-02, Rev. 17 (Sheets 1-2)	Body Assembly
LWT 315-40-03, Rev. 22 (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. 16 (Sheets 1-5)	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. 7 (Sheets 1-2)	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. 3 (Sheets 1-2)	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. 3 (Sheets 1-3)	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. 1 (Sheets 1-2)	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
LWT 315-40-120, Rev. 2 (Sheets 1-3)	Top Module, General Atomics IFM, LWT Cask
LWT 315-40-123, Rev. 1 (Sheets 1-2)	Spacer, General Atomics IFM, LWT Cask
LWT 315-40-124, Rev. 0	Transport Cask Assembly, General Atomics IFM, LWT Cask
LWT 315-40-125, Rev. 2 (Sheets 1-3)	Transport Cask Assembly, Framatome/EPRI, LWT Cask
LWT 315-40-126, Rev. 2 (Sheets 1-2)	Weldment, Framatome/EPRI, LWT Cask
LWT 315-40-127, Rev. 1 (Sheets 1-2)	Spacer Assembly, TPBAR Shipment

5.(a)(3)(ii) Drawings (continued)

LWT 315-40-129, Rev. 1	Canister Body Assembly, Failed Fuel Can, PULSTAR
LWT 315-40-130, Rev. 1	Assembly, Failed Fuel Can, PULSTAR
LWT 315-40-133, Rev. 0 (Sheets 1-2)	Transport Cask Assembly, PULSTAR Shipment, LWT Cask
LWT 315-40-134, Rev. 1	Body Weldment, Screened Fuel Can, PULSTAR Fuel
LWT 315-40-135, Rev. 1	Assembly, Screened Fuel Can, PULSTAR Fuel
LWT 315-40-139, Rev. 0	Transport Cask Assembly, ANSTO Fuel
LWT 315-40-140, Rev. 0 (Sheets 1-2)	Weldment, 7 Cell Basket, Top Module, ANSTO Fuel
LWT 315-40-141, Rev. 0 (Sheets 1-2)	Weldment, 7 Cell Basket, Intermediate Module, ANSTO Fuel
LWT 315-40-142, Rev. 0 (Sheets 1-2)	Weldment, 7 Cell Basket, Base Module, ANSTO 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 zirconium alloy type 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.

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

5.(b)(1)(i) Irradiated PWR fuel assemblies. (continued)

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

(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 zirconium alloy type 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.

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.

5.(b)(1) Type and form of material (continued)

(iii) Irradiated PWR rods, consisting of uranium dioxide pellets within zirconium alloy type 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 inch. 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.

(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	

5.(b)(1) Type and form of material (continued)

(iv) (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 element.

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).

5.(b)(1) Type and form of material (continued)

(iv) (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		

(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.

5.(b)(1) Type and form of material (continued)

(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 zirconium alloy type 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 zirconium alloy type 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 inch. 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	0.025 cm	0.025 cm	0.025 cm
Maximum Outer Diameter	9.535 cm	9.535 cm	9.535 cm
Minimum Inner Diameter	5.88 cm	5.88 cm	5.88 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)(1) Type and form of material (continued)

(xi) General Atomics (GA) Irradiated Fuel Material (IFM) consisting of two separate types of fuel materials: (a) High Temperature Gas Cooled Reactor (HTGR); and (b) Reduced-Enrichment Research and Test Reactor (RERTR) type TRIGA fuel entities.

(a) GA HTGR IFM comprised of four forms: fuel particles (kernels), fuel particles (coatings), fuel compacts (rods), and fuel pebbles. Fuel particles (kernels) are solid, spheridized, high-temperature sintered fully-densified, ceramic kernel substrate, composed of UO_2, UCO_2, $(Th,U)C_2$, or $(Th,U)O_2$. Fuel particles (coatings) are solid , spheridized, isotropic, discrete multi-layered fuel particle coatings with chemical composition including pyrolitic-carbon (PyC) and silicon carbide (SiC). Fuel compacts (rods) are multi-coated ceramic fuel particles, bound in solid, cylindrical, injection molded, high-temperature heat-treated compacts which are composed of carbonized graphite shim, coke, and graphite powder. Fuel pebbles are multi-coated fuel particles, bound in solid, spherical injection-molded, high-temperature heat-treated pebbles composed of carbonized graphite shim, coke and graphite powder. Initial enrichment of the HTGR IFM varies from 10.0 to 93.15 wt% ^{235}U.

(b) GA RERTR IFM comprised of irradiated TRIGA fuel elements which contain three distinct mass loadings of uranium of 20, 30, and 45 wt% U. The average mass of the fuel portion of the elements is 551 g with a maximum initial enrichment of 19.7 wt% U-235.

GA IFM content description:

	GA HTGR IFM	GA RERTR IFM
Fuel material	UC_2, UCO, UO_2 $(Th,U)C_2$, $(Th,U)O_2$	U-ZrH metal alloy
Maximum fuel weight, lbs	23.52	23.73
Maximum overall length, in	n/a	29.92
Maximum active fuel length, in	n/a	22.05
Fuel rod cladding	n/a	Incoloy 800
Maximum Uranium, kg U	0.21	3.86
Maximum initial ^{235}U, wt%	93.15	19.7
Maximum Activity, Ci	483	2920

5.(b)(1) *Type and form of material (continued)*

(xii) Irradiated tritium-producing burnable absorber rods (TPBARs), as described in Section 1.2.3.6 of the application. Each TPBAR is approximately 153 inches in length and 0.381 inches in diameter and is stainless steel clad. The TPBARs contain lithium aluminate annular pellets, with an inner zircaloy liner and an outer nickel-plated zircaloy tube. Each TPBAR contains a maximum of 1.2 grams tritium. The minimum cool time is 30 days.

(xiii) Intact or damaged PULSTAR fuel elements, including fuel debris, pieces and nonfuel components of PULSTAR fuel assemblies as specified below.

Description	Value
Maximum Pellet Diameter (inch)	0.423
Minimum Element (Rod) Cladding Thickness (inch)	0.0185
Minimum Element (Rod) Diameter (inch)	0.470
Maximum Active Fuel Height (inch)	24.1
Nominal Element (Rod) Length (inch)	26.2
Nominal Assembly Length (inch)	38
Maximum Assembly or Loaded Can Weight (lb)	80
Maximum PULSTAR Can Content Weight (lb)	39.6
Maximum Enrichment (wt % ^{235}U)	6.5
Maximum ^{235}U Content per Element (g)	33
No. of Elements (Rods) per Assembly	25
No. of Elements (Rods) per Can[1]	≤25
Maximum Depletion (% ^{235}U)	45
Minimum Cooling Time (yrs)	1.5
Maximum Heat Load per Assembly (W)	30
Maximum Heat Load per Element (W)	1.2

[1] Damaged PULSTAR fuel elements, including fuel debris, pieces and nonfuel components of PULSTAR fuel assemblies must be loaded into a PULSTAR can. The contents of a PULSTAR can are restricted to the equivalent of the fuel material in 25 intact PULSTAR fuel elements and of the displaced volume of 25 intact PULSTAR fuel elements.

5.(b)(1) Type and form of material (continued)

(xiv) Intact, irradiated ANSTO fuel consisting of spiral fuel assemblies and MOATA plate bundles.

Spiral fuel assemblies consist of 10 curved uranium-aluminum alloy fuel plates between an inner and an outer aluminum shell, with the following fuel parameters:

Parameter	Limiting Values
Number of fuel plates per assembly	10
Maximum ^{235}U content per assembly (g)	160
Maximum enrichment (wt % ^{235}U)	85
Maximum assembly weight (lb)	18
Minimum plate thickness (cm)	0.124
Minimum active fuel height (cm)	59.075

MOATA plate bundles consist of uranium-aluminum alloy fuel plates with aluminum cladding, with the following specifications:

Parameter	Limiting Values
Maximum number of fuel plates per assembly	14
Maximum ^{235}U content per plate (g)	22.3
Maximum enrichment (wt % ^{235}U)	92
Maximum plate spacer thickness (cm)	0.18
Maximum active fuel width (cm)	7.32
Maximum bundle weight (lb)	18

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.

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

(ii) For the contents described in Item 5.(b)(1)(ii): two BWR assemblies positioned within 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 Item 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 Item 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.

(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 intact fuel elements in the non-poisoned TRIGA fuel basket, and up to 140 intact 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. 2, 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)

(vii) (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. 1, 315-40-087, Rev. 5, 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.

(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 intact rods in the non-poisoned TRIGA fuel basket, and up to 560 intact 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. 1, 315-40-087, Rev. 5, 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.

(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.

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

(ix) For high burnup PWR fuel rods, as described in Item 5.(b)(1)(viii): up to 25 fuel rods. Maximum decay heat not to exceed 2.3 kilowatts per package.

Intact individual rods may be placed either in an irradiated or unirradiated fuel assembly lattice (skeleton) or in a fuel rod insert. The PWR fuel assembly lattice must be transported in the PWR basket.

Up to 14 of the 25 fuel rods may be classified as damaged. Damaged fuel rods may include fuel debris, particles, loose pellets, and fragmented rods. Damaged fuel rods must be placed in a fuel rod insert. Damaged fuel rods may also be placed in individual failed fuel rod capsules, as shown in Figure 1.2-11 of the application, prior to placement in the fuel rod insert. Irradiated guide tubes and guide tube segments may be placed in the fuel rod insert. The fuel rod insert must be transported in a PWR/BWR transport canister, which is positioned in the PWR insert in the PWR basket.

(x) For high burnup BWR fuel rods, as described in Item 5.(b)(1)(ix): up to 25 fuel rods. Maximum decay heat not to exceed 2.1 kilowatts per package.

Intact individual rods may be placed either in a fuel assembly lattice or in a fuel rod insert. The BWR fuel assembly lattice must be transported in the PWR insert in the PWR basket.

Up to 14 of the 25 fuel rods may be classified as damaged. Damaged fuel rods may include fuel debris, particles, loose pellets, and fragmented rods. Damaged fuel rods must be placed in a fuel rod insert. Damaged fuel rods may also be placed in individual failed fuel rod capsules, as shown in Figure 1.2-11 of the application, prior to placement in the fuel rod insert. The fuel rod insert must be transported in a PWR/BWR transport canister, which is positioned in the PWR insert in the PWR basket.

(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 the top basket fuel element active fuel region is spaced a minimum 3.7 inches from the bottom of the cask lid. Spacing of the active fuel may be accomplished by fuel element hardware, lid spacer, or a combination thereof. Maximum decay heat is 1.05 kilowatts per package. At a top basket active fuel region to cask lid spacing of less than 3.7 inches, the maximum decay heat not to exceed 18 watts per DIDO fuel element and a total of 756 watts per package.

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

(xii) For GA IFM as described in Item 5.(b)(1)(xi):

(a) Mixture of fuel particles (kernels and coatings), fuel compacts (rods), and fuel pebbles, packaged in its own Fuel Handling Unit (FHU).

GA HTGR FHU consists of two redundant canisters. GA HTGR IFM is packaged inside a primary canister with welded closure, as shown in General Atomics Drawing No. 032237, Rev. B, "HTGR Primary Enclosure." The primary canister is packaged inside a secondary canister with welded closure, as shown in General Atomics Drawing No. 032231, Rev. A, "HTGR Secondary Enclosure."

GA HTGR FHU total maximum decay heat not to exceed 2.05 watts, and maximum loaded weight not to exceed 71.5 lbs.

(b) Twenty irradiated TRIGA fuel elements; 13 of the elements are intact, and the remaining 7 are sectioned. GA RERTR IFM is packaged in its own FHU.

GA RERTR FHU consists of two redundant canisters. GA RERTR IFM is packaged inside a primary canister with welded closure, as shown in General Atomics Drawing No. 032236, Rev. B, "RERTR Primary Enclosure." The GA RERTR IFM primary canister is packaged inside a secondary canister with welded closure, as shown in General Atomics Drawing No. 032230, Rev. A, "RERTR Secondary Enclosure."

GA RERTR FHU total maximum decay heat not to exceed 11 watts, and maximum loaded weight not to exceed 76.0 lbs.

(xiii) For TPBARs as described in Item 5.(b)(1)(xii):

300 TPBARs, including a maximum of 2 damaged rods, positioned within a consolidation canister, as shown in Figure 1.2-10 of the application. The maximum decay heat is 2.31 watts per rod and 693 watts per package. The maximum weight of the TPBARs and the consolidation canister is 1,000 pounds. Consolidation canisters with fewer than 300 TPBARs may also contain stainless steel spacers of various geometries. The total weight and volume of the reduced TPBAR contents plus the spacers must be less than or equal to the weight and volume of 300 TPBARs.

(xiv) For PULSTAR fuel as described in Item 5.(b)(1)(xiii):

Up to 700 intact or damaged PULSTAR fuel elements in either assembly or element form, including fuel debris, pellets, pieces and nonfuel components of PULSTAR fuel assemblies. *The contents of a PULSTAR can are restricted to the equivalent of the fuel* material in 25 intact PULSTAR fuel elements and of the displaced volume of 25 intact PULSTAR fuel elements.

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

(xv) For ANSTO fuel as described in Item 5.(b)(1)(xiv):

Up to 42 spiral fuel assemblies, MOATA plate bundles, or any combination of spiral fuel assemblies and MOATA plate bundles. ANSTO fuel must be loaded within ANSTO basket modules. Spiral fuel assemblies may be cropped by removing nonfuel-bearing hardware to fit the ANSTO basket modules. Fuel assemblies that are cropped, but are otherwise intact, may be considered intact. For spiral fuel assemblies, the maximum decay heat per assembly is 15.7 watts. The minimum cool time as a function of burnup shall be consistent with the maximum decay heat limit and shall be determined using the procedures for medium enriched DIDO fuel in Section 7.1.4 of the application; the minimum cool time may not be less than 270 days. For MOATA plate bundles, the maximum heat load per bundle is 3 watts, and the minimum cool time is 10 years.

5.(c) Criticality Safety Index

(1)	For TRIGA fuel elements, TRIGA fuel cluster rods, metallic fuel rods, MTR fuel assemblies, up to 25 PWR fuel rods, up to 25 high burnup PWR or BWR rods, GA IFM, uncanned intact PULSTAR fuel assemblies and elements, 42 spiral fuel assemblies, and MOATA plate bundles:	0.0
(2)	For PWR fuel assemblies:	100
(3)	For BWR fuel assemblies:	5.0
(4)	For DIDO fuel assemblies:	12.5
(5)	For package with any number of canned PULSTAR fuel	33.4

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), 5.(b)(2)(viii)(b), 5.(b)(2)(ix), 5.(b)(2)(x), and 5.(b)(2)(xiv).

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 +/- 20 ft-lbs. The bolts used to secure the vent and drain port covers must be torqued to 100 +/- 10 inch-lbs. The bolts used to secure the Alternate B port covers must be torqued to 280 +/- 10 inch-lbs.

9. Prior to each shipment, the package must be leak tested to 1 x 10^{-3} std cm^3/sec, except that replaced seals must be leak tested to 5.5 x 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 x 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; and

    (c) The package shall be prepared for shipment and operated in accordance with the Package Operations of Chapter 7 of the application. 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 or TPBARs.

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, GA IFM, PULSTAR fuel elements, spiral fuel assemblies, and MOATA plate bundles, 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, high burnup PWR or BWR rods, TPBARs, PULSTAR fuel elements, spiral fuel assemblies, or MOATA plate bundles 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. For shipment of TPBARs:

    (a) Prior to first use for shipment of TPBARs, each packaging must be hydrostatic pressure tested to 450 +15/-0 psig, as described in Section 8.1.2 of the application;

    (b) The package must be marked with Package Identification Number USA/9225/B(M)-96;

    (c) The package must be configured as shown in NAC International Drawing No. 315-40-128, Rev. 1; and

    (d) Prior to each shipment, after loading, each cask containment seal must be tested to show no leakage greater than 2×10^{-7} std-cm^3/s (helium).

15. For shipment of PULSTAR fuel:

    (a) Intact fuel elements may be configured as PULSTAR fuel assemblies, may be placed into a TRIGA fuel rod insert (a 4 x 4 rod holder), or may be loaded into PULSTAR fuel cans. Intact PULSTAR fuel assemblies and PULSTAR fuel elements in a TRIGA fuel rod insert may be loaded in any module of the 28 MTR basket assembly. PULSTAR fuel cans may only be loaded into the top or base module of the 28 MTR basket assembly.

(b) Damaged PULSTAR fuel elements and nonfuel components of PULSTAR fuel assemblies must be loaded into PULSTAR cans. Damaged PULSTAR fuel, including fuel debris, pellets or pieces, may be placed in an encapsulating rod prior to loading into a PULSTAR fuel can. PULSTAR fuel cans may only be loaded into the top or base module of the 28 MTR basket assembly.

(c) Loading of modules with mixed PULSTAR payload configuration is allowed.

16. Transport by air is not authorized.

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

18. Revision 40 of this certificate may be used until July 31, 2007.

19. Expiration Date: February 28, 2010.

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9225	42	71-9225	USA/9225/B(U)F-96	23	OF	23

REFERENCES

NAC International, Inc., application dated August 8, 2005.

Supplements dated: December 15, 2005, April 17, 2006, June 9 and June 15, 2006.

FOR THE U. S. NUCLEAR REGULATORY COMMISSION

Christopher M. Regan, Acting Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: August 3, 2006

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9226	1	71-9226	USA/9226/B(U)F-85	1	OF	8

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 Atomics
3550 General Atomics Court
San Diego, California 92121-1122

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION
General Atomics application dated
August 31, 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.

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

5.a. (2) (continued)

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.

5.a. (2) (continued)

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 (Proprietary Version)
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.

5.b. (2) (continued)

(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

<u>Fuel Type</u> 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

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

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 $\times 10^{-3}$ 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, 2008.

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9226	1	71-9226	USA/9226/B(U)F-85	8	OF	8

REFERENCES
General Atomics Safety Analysis Report for the GA-4 Legal Weight Truck Spent Fuel Shipping Cask, August 31, 1994.

Supplements dated: August 5, 1998, General Atomics Safety Analysis Report for the GA-4 Legal Weight Truck Spent Fuel Shipping Cask, Revision G (Proprietary) and Revision H (Non-Proprietary); and June 12, 2003.

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 July 17, 2003

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
9228	23	71-9228	USA/9228/B(U)F-96	1	OF	10

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.

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9228	23	71-9228	USA/9228/B(U)F-96	2	OF	10

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.

(ix) The optional lead liner is constructed and assembled in accordance with General Electric Company Drawing No. 129D4922, Rev. 2.

5.(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.

(iv) Irradiated Tower Shielding Reactor (TSR) fuel elements, positioned within the TSR fuel basket specified in 5(a)(3).

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5.(b)(1) Type and form of material (continued)

(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.

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

(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.
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.

5.(c) Criticality Safety Index

For the contents described in 5(b)(1)(i), 5(b)(1)(ii) (except byproduct material), 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), 5(b)(1)(vi), and byproduct material from 5(b)(1)(ii); and limited in 5(b)(2)(iv), 5(b)(2)(v), 5(b)(2)(vi), and 5(b)(2)(ii): 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.

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)(i), 5(b)(1)(ii), 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. For horizontal transport of irradiated fuel and byproduct material contents described in 5(b)(1)(i) and 5(b)(1)(ii), the maximum decay heat is limited to 600 watts per package and the lead liner described in 5(a)(3)(ix) must be used.

17. Packagings may be marked with Package Identification Number USA/9228/B(U)F-85 until May 31, 2006, and must be marked with Package Identification Number USA/9228/B(U)F-96 after May 31, 2006.

18. Air transport of fissile material is not authorized.

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

20. Revision No. 22 of this certificate may be used until May 31, 2007.

21. Expiration date: May 31, 2011.

REFERENCES

General Electric Company application dated December 12, 2000.

Supplements dated: December 20, 2000; March 16 and 27, 2001; March 22, 2002; and March 25, May 4, 5, and 23, July 28, 2005, and January 25, 2006.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

James R. Hall for

Robert A. Nelson, Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: May 8, 2006

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
9233	7	71-9233	USA/9233/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)*

Transnuclear, Inc
Four Skyline Drive
Hawthorne, NY 10532-2120

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Transnuclear, Inc. application dated March 8, 2005.

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.

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 mass limits of 10 CFR 71.15. 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 this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

10. Expiration date: April 30, 2010.

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(8-2000)
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U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

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REFERENCES

Transnuclear, Inc., application dated December 2, 2004.

Supplements dated: March 8, 2005.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Robert J. Lewis, Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

ate: 14 Apr 2005

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
9234	19	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, LLC
1802 Fairfax Road
Greensboro, NC 27407

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.

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) Criticality Safety Index

Minimum transport index to be shown on label for nuclear criticality control: 5.0

The Model 30B cylinders must be fabricated, inspected, tested, and maintained in accordance with American National Standard ANSI N14.1-2001, or an earlier version of ANSI N14.1 in effect at the time of fabrication. 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.

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.

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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
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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 71.17.

13. Expiration date: December 31, 2008

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; April 11, 2002; August 27, 2003, January 28 and June 13, 2005.

United States Enrichment Corporation supplement dated: April 14, 1997.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Michael T. Markley, Acting Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: July 5, 2005

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9235	9	71-9235	USA/9235/B(U)F-96	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 Road, Suite 200
Norcross, Georgia 30092

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

NAC International, Inc., application dated March 1, 2004, 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

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

an outside diameter of 86.7 inches. The annulus between the inner and outer shells is filled with lead.

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 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

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

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.

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 is a right-circular cylinder formed by a series of 1.75-inch thick Type 304 stainless steel plates, laterally supported by 12 equally spaced welded 1.25-inch thick Type 304 stainless steel 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 1-3, Rev. 14
423-802, sheets 1-7, Rev. 20
423-803, sheets 1-2, Rev. 8
423-804, sheets 1-3, Rev. 8
423-805, sheets 1-2, Rev. 6
423-806, Rev. 7
423-807, sheets 1-3, Rev. 3
423-811, sheets 1-2, Rev. 11
423-812, Rev. 6
423-900, Rev. 6
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. 5
423-871, Rev. 5
423-872, Rev. 6
423-873, Rev. 2
423-874, Rev. 2
423-875, sheets 1-2, Rev. 7

(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, sheets 1-2, Rev. 2
455-870, Rev. 5
455-871, sheets 1-2, Rev. 8
455-871, sheets 1-3, Rev. 7P2[1]
455-872, sheets 1-2, Rev. 12
455-872, sheets 1-2, Rev. 11P1[1]
455-873, Rev. 4
455-881, sheets 1-3, Rev. 8
455-887, sheets 1-3, Rev. 4
455-888, sheets 1-2, Rev. 8
455-891, sheets 1-2, Rev. 1
455-891, sheets 1-3, Rev. 2PO[1]
455-892, sheets 1-2, Rev. 3
455-892, sheets 1-3, Rev. 3P0[1]
455-893, Rev. 3
455-894, Rev. 2
455-895, sheets 1-2, Rev. 5
455-895, sheets 1-2, Rev. 5P0[1]
455-901, Rev. 0P0[1]
455-902, sheets 1-5, Rev. 0P4[1]
455-919, Rev. 2

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[1]Drawing defines the alternate configuration that accommodates the Yankee-MPC damaged fuel can.

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. D3
YR-00-061, Rev. D4
YR-00-062, sheet 1, Rev. D4
YR-00-062, sheet 2, Rev. D2
YR-00-062, sheet 3, Rev. D1
YR-00-063, Rev. D4
YR-00-064, Rev. D4
YR-00-065, Rev. D2
YR-00-066, sheet 1, Rev. D5
YR-00-066, sheet 2, Rev. D3

(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. 3
414-871, sheets 1-2, Rev. 6
414-872, sheets 1-3, Rev. 6
414-873, Rev. 2
414-874, Rev. 0
414-875, Rev. 0
414-881, sheets 1-2, Rev. 4
414-882, sheets 1-2, Rev. 4
414-887, sheets 1-4, Rev. 4
414-888, sheets 1-2, Rev. 4
414-889, sheets 1-3, Rev. 7
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. 4

(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. 1
414-902, sheets 1-3, Rev. 3
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

Notes:
[(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)

Notes:
(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	Yankee DFC
Cladding Material	Zircaloy	Zircaloy	SS	Zircaloy	Zirc/SS	Zirc/SS
Maximum Number of Rods per Assembly	237	231	305	231	64	305
Maximum Initial Uranium Content (kg/assembly)	246	240	287	240	70	287
Maximum Initial Enrichment (wt% ^{235}U)	4.0	3.9	4.94	4.0	4.94	4.97 [3]
Minimum Initial Enrichment (wt% ^{235}U)	4.0	3.7	4.94	3.5	3.5	3.5 [3]
Maximum Assembly Weight (lbs)	≤950	≤950	≤950	≤950	≤950	≤950
Maximum Burnup (Mwd/MTU)	32,000	36,000	32,000	36,000	36,000	36,000
Maximum Decay Heat per Assembly (kW)	0.28	0.347	0.28	0.34	0.11	0.347
Minimum Cool Time (yrs)	11.0	8.1	22.0	10.0	8.0	8.0
Maximum Active Fuel Length (in)	91	91	92	91	92	N/A

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.
3. Stated enrichments are nominal values (fabrication tolerances are not included).

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 71.15.

(iv) Irradiated intact and damaged Connecticut Yankee (CY) Class PWR fuel assemblies (including optional stainless steel rods inserted into the CY intact and damaged fuel assembly reactor control cluster assembly (RCCA) guide tubes that do not contain RCCAs), 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	PWR [1] 15x15	PWR [2] 15x15	PWR [3]	CY-MPC RFA[4]	CY-MPC DFC[5]
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[6]	4.61[6]
Minimum Initial Enrichment (wt% ^{235}U)	3.0	2.95	2.95	2.95	2.95
Maximum Assembly Weight (lbs)	≤1,500	≤1,500	≤1,500	≤1,600	≤1,600
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. Stainless steel assemblies manufactured by Westinghouse Electric Co., Babcock & Wilcox Fuel Co., Gulf Gen. Atomics, Gulf Nuclear Fuel, & Nuclear Materials & Man. Co.
2. Zircaloy spent fuel assemblies manufactured by Gulf Gen. Atomics, Gulf Nuclear Fuel, & Nuclear Materials & Man. Co., and Babcock & Wilcox Fuel Co.
3. Westinghouse Vantage 5H zircaloy clad spent fuel assemblies have an initial uranium enrichment > 3.93 % wt. U^{235}.
4. Reconfigured Fuel Assemblies (RFA) must be loaded in one of the 4 oversize fuel loading positions.
5. Damaged Fuel Cans (DFC) must be loaded in one of the 4 oversize fuel loading positions.
6. 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) Criticality Safety Index: 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.

   (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.

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.

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
9235	9	71-9235	USA/9235/B(U)F-96	12	OF	12

12. Transport by air is not authorized.

13. Packagings may be marked with Package Identification Number USA/9235/B(U)F-85 until April 30, 2007, and must be marked with Package Identification Number USA/9235/B(U)F-96 after April 30, 2007.

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

15. Revision No. 8 of this certificate may be used until April 30, 2007.

16. Expiration date: March 31, 2009.

REFERENCES

NAC International, Inc., application dated: March 1, 2004.

NAC International, Inc., supplements dated: August 4, and November 1, 2005, and March 1, 2006.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Robert A. Nelson, Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: Apr. 25, 2006

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9239	14	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. (a) 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.

(b) For shipments of 17x17 STANDARD lattice fuel assemblies (17x17 STD and 17x17 XL) with U-235 enrichments of over 4.85 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.

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		CERTIFICATE OF COMPLIANCE FOR RADIOACTIVE MATERIAL PACKAGES			
a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE	PAGES
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11. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR §71.17.

12. Revisions Nos. 12 and 13 of this certificate may be used until March 31, 2007. |

13. Expiration date: March 31, 2007. |

<u>REFERENCES</u>
Westinghouse Electric Corporation application dated February 14, 2002.

Supplements dated: March 6, 2002, and January 25, 2006.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Robert A. Nelson, Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: April 25, 2006

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
9246	5	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.

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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) Criticality Safety Index 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 71.17.

8. Revision No. 4 of this certificate may be used until September 30, 2007.

9. Expiration date: November 30, 2011.

REFERENCES

National Institute of Standards and Technology application dated February 7, 1992.

Supplements dated: February 14, 1992; August 7, 1996; August 17, 2001; and September 5, 2006.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Robert A. Nelson, Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: 9/27, 2006

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE	PAGES
9248	18	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
   Framatome ANP, Inc. application dated September 5, 2003, 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 Framatome ANP, Inc., and Siemens Nuclear Power Corporation/Advanced Nuclear Fuels Corporation Drawing Nos.:

EMF-304,416, Rev. 14.
EMF-306,272, Rev. 10.
EMF-309,141, Rev. 1.

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_20_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.

(x) 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 uranium enrichment is 2.3 weight percent U-235. 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. 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 grams 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), 5(b)(1)(ix), and 5(b)(1)(x):

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), 5(b)(1)(x), 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.

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) and 5(b)(1)(x), 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 September 5, 2003.

    (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 September 5, 2003.

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, 2009.

REFERENCES

Framatome ANP, Inc., application dated September 5, 2003.

Supplements dated: September 24 and November 6, 2003.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

[signature] for

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

Date December 19 2003

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
9250	10	71-9250	USA/9250/B(U)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)*

BWX Technologies
Nuclear Products Division
P.O. Box 785
Lynchburg, VA 24505

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

BWX Technologies, Nuclear Products Division application dated June 13, 2005.

4. CONDITIONS

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

5.

Packaging

(1) Model No.: 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 BWX Technologies, Inc., Drawing Nos. 1220276 E, Rev. 4, and 1220277 E, Rev. 8.

5.(b) Contents

Type and form of material, maximum quantity of material per package, and Criticality Safety Index.

The weight of the contents, including secondary containers, inserts, and other materials in the inner vessel, shall not exceed 50 pounds.

(1) 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 enrichment. Carbide compounds are not authorized. The maximum H/U must consider all sources of moderation in the inner vessel.

Fissile Material	Maximum H/U	Maximum Fissile Mass per Package (kg)	Criticality Safety Index
U-235	3	9.0	2.0
U-235	3	1.6	0.5
U-235	20	4.0	2.0
U-233	20	0.5	1.8

(2) 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 53.5 to 56.0 percent by weight. 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 enrichment. The maximum H/U must consider all sources of moderation in the inner vessel.

Fissile Material	Maximum H/U	Maximum Fissile Mass per Package (kg)	Criticality Safety Index
U-235	3	9.0	2.0
U-235	3	1.6	0.5
U-235	20	4.0	2.0
U-233	20	0.5	1.8

5.(b) Contents (continued)

(3) Unirradiated uranium as solid metal. The uranium may be of any enrichment. The maximum H/U must consider all sources of moderation in the inner vessel.

Fissile Material	Maximum H/U	Maximum Fissile Mass per Package (kg)	Criticality Safety Index
U-235	3	9.0	2.5
U-235	3	1.6	0.5
U-235	20	4.0	2.0
U-233	20	0.5	1.8

(4) Unirradiated uranium as solid metal. The uranium may be of any enrichment. The packaging must include a solid aluminum disk insert positioned in the inner vessel, as shown on BWX Technologies, Inc., Drawing No. 1220277 E, Rev. 8 (Part No. 6). The maximum H/U must consider all sources of moderation in the inner vessel.

Fissile Material	Maximum H/U	Maximum Fissile Mass per Package (kg)	Criticality Safety Index
U-235	3	9.0	2.0

(5) 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 enrichment. The quantity of uranyl nitrate shall not exceed 1000 mL of solution.

Fissile Material	Maximum H/U	Maximum Fissile Mass per Package (kg)	Criticality Safety Index
U-235	N/A	0.4	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 71.17.

9. Expiration date: March 31, 2008.

REFERENCES

BWX Technologies, Inc., application dated June 13, 2005.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

S. Raynard Wharton

for Robert J. Lewis, Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: September 22, 2005

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9251	12	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 October 8, 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

(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 drum lid is closed with a 12-gauge locking ring with drop forged lugs and a 5/8-inch diameter bolt. In addition to the locking ring, three lid clamps are installed to secure the drum lid. 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. 5, and 1283759D, Rev. 0.

5.(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. The gasket shall be replaced if it is damaged, defective, or degraded.

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
9251	12	71-9251	USA/9251/AF	3	OF	3

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

Framatome ANP, Inc., application dated October 8, 2002.

Supplement dated: April 17, 2003.

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: May 16, 2003

UNITED STATES NUCLEAR REGULATORY COMMISSION

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
9252	5	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 weight percentage U-235. 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.

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

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.

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, 2008.

REFERENCES

B&W Fuel Company application dated March 9, 1993.

Supplements dated: May 10, and July 7, 1993; April 13, 1994; June 17, 1998; November 13, 2000; February 9, 2001; and August 26, 2003.

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: October 14, 2003

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
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2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

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.

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.

<u>For Configuration 1</u>:

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.

<u>For Configuration 2</u>:

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) For Configuration 2:

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) For Configuration 2:

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) 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.

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, until October 1, 2004, and under the provisions of 10 CFR 71.17 thereafter.

9. Expiration date: May 31, 2009.

<u>REFERENCES</u>

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, October 2, 2001, and April 22, 2004.

Transnuclear, Inc. supplements dated September 19, 2001; and March 1, May 17, June 14 and 21, 2002 ; June 3, and July 21, 2003.

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: May 21, 2004

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

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2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

Transnuclear, Inc.
Four Skyline Drive
Hawthorne, NY 10532

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Transnuclear, Inc. consolidated application dated August 4, 2003.

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 9,
Sheets 1 through 2
MP187 Multi-Purpose Cask
General Arrangement

NUH-05-4001, Revision 15,
Sheets 1 through 6
MP187 Multi-Purpose Cask
Main Assembly

NUH-05-4002, Revision 5
Sheets 1 and 2
MP187 Multi-Purpose Cask
Impact Limiters

NH-05-4003, Revision 10,
Sheets 1 and 2
NUHOMS® MP187 Multi-Purpose Cask
On-Site Transfer Arrangement

NUH-05-4004, Revision 16,
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 7,
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 3.36 ^{235}U 3.12 ^{235}U	45,000 40,000 35,000	38 years/14 kW/ 0.583 kW
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 (Criticality Safety Index)

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.

10. Expiration Date: October 31, 2008.

REFERENCES

Transnuclear, Inc. application dated August 4, 2003.

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: October 14, 2003

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
9258	2	71-9258	USA/9258/B(U)-96	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)*

MDS Nordion
447 March Road
Kanata, Ontario, Canada K2K 1X8

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

MDS Nordion consolidated application dated August 1, 2003, 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.: 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. There are two alternative source carriers. The F-313 source carrier holds forty special form sources in a single ring configuration. The F-457 source carrier holds eighty special form sources in a double ring configuration.

The cask body is constructed of a ½-inch thick inner stainless steel shell, and a ½-inch thick outer stainless steel shell. The annulus between the inner and outer shells is filled with lead, approximately 11 ¼ inches thick. The cask is closed by a 2 ½ 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 carbon 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.

5.(a) Packaging

(2) Description (continued)

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
Fireshield outer diameter	47 inches
Overall package dimensions (including shipping skid)	
width	78 inches
length	78 inches
height	80 1/2 inches
Maximum contents weight	40 pounds
Maximum package weight (including contents)	21,000 pounds

(3) Drawings

The packaging is constructed in accordance with MDS Nordion Drawing Nos.:

F629401-001, Sheet 1, Rev. F,
F629401-001, Sheet 2, Rev. F,
F629401-001, Sheet 3, Rev. D,
F629401-001, Sheet 4, Rev. F,
F629401-001, Sheet 5, Rev. F,
F631301-001, Rev. B, and
F645701-001, 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

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.

7. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.12, until October 1, 2004, and under provisions of 10 CFR 71.17 thereafter.

8. Packagings may be marked with Package Identification Number USA/9258/B(U)-85 until October 1, 2005, and must be marked with Package Identification Number USA/9258/B(U)-96 after October 1, 2005.

9. Expiration date: December 31, 2008.

REFERENCES

MDS Nordion application dated August 1, 2003.

Supplements dated: March 12, April 20, and May 20, 2004.

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: June 3, 2004

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9261	5	71-9261	USA/9261/B(U)F-85	1	OF	10

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)*

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 12, dated October 9, 2006.

4. CONDITIONS

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

5.

(a) Packaging

(1) Model No.: 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 that house the spent nuclear fuel and an overpack that 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 96 inches without impact limiters and approximately 128 inches with impact limiters. Maximum gross weight for transportation (including overpack, MPC, fuel, and impact limiters) is 282,000 pounds. Specific tolerances germane to the safety analyses are called out in the drawings listed below.

Multi-Purpose Canister

There are six Multi-Purpose Canister (MPC) models designated as the MPC-24, MPC-24E, MPC-24EF, MPC-32, MPC-68, and MPC-68F. All MPCs are designed to have identical exterior dimensions, except those MPC-24E/EFs custom-designed for the Trojan plant, which are approximately nine inches shorter than the generic Holtec MPC design. A single overpack design is provided that 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 series is designed to contain up to 24 Pressurized Water Reactor (PWR) fuel assemblies; the MPC-32 is designed to contain up to intact 32 PWR assemblies; and the MPC-68 and MPC-68F are designed to contain up to 68 Boiling Water Reactor (BWR) fuel assemblies. BWR fuel debris may be shipped only in the MPC-68F.

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

PWR spent fuel assemblies classified as fuel debris may be loaded only in MPC-24EF.

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 generic MPC is fixed. The outer diameter of the Trojan MPCs is the same as the generic MPC, but the height is approximately nine inches shorter than the generic MPC design. A steel spacer is used with the Trojan plant MPCs to ensure the MPC-overpack interface is bounded by the generic design. The fuel basket designs vary based on the MPC model. The MPC pressure boundary is a welded enclosure constructed entirely of a stainless steel alloy.

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 12:

(a) HI-STAR 100 Overpack	Drawing 3913, Sheets 1-9, Rev. 7
(b) MPC Enclosure Vessel	Drawing 3923, Sheets 1-5, Rev. 14
(c) MPC-24E/EF Fuel Basket	Drawing 3925, Sheets 1-4, Rev. 5
(d) MPC-24 Fuel Basket Assembly	Drawing 3926, Sheets 1-4, Rev. 5
(e) MPC-68/68F/68FF Fuel Basket	Drawing 3928, Sheets 1-4, Rev. 5

5.(a) (3) Drawings (continued)

(f) HI-STAR 100 Impact Limiter CoC No. 9261, Appendix B	Drawing C1765, Sheets 1 and 2, Rev. 2; Sheet 3, Rev. 1, Sheet 4, Rev. 2; Sheets 5 and 6, Rev. 1; and Sheet 7, Rev. 0.
(g) HI-STAR 100 Assembly for Transport	Drawing 3930, Sheets 1-3, Rev. 1
(h) Trojan MPC-24E/EF Spacer Ring	Drawing 4111, Sheets 1-2, Rev. 0
(i) Damaged Fuel Container for Trojan Plant SNF	Drawing 4119, Sheet 1-4, Rev. 1
(j) Spacer for Trojan Failed Fuel Can	Drawing 4122, Sheets 1-2, Rev. 0
(k) Failed Fuel Can for Trojan	SNC Drawings PFFC-001, Rev. 8 and PFFC-002, Sheets 1 and 2, Rev. 70
(l) HI-STAR 100 MPC-32	Drawing 3927, Sheets 1-4, Rev. 6

5.(b) Contents

(1) Type, 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)(i) 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, missing structural components such as grid spacers, assemblies whose structural integrity have been impaired, 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 (or Canisters)(DFCs) are specially designed fuel containers for damaged fuel assemblies or fuel debris that 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 the HI-STAR 100 System SAR, Rev. 12.

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. Fuel debris also includes certain Trojan plant-specific fuel material contained in Trojan Failed Fuel Cans.

5.(b)(1)(b) Definitions (continued)

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. Fuel assemblies without fuel rods in fuel rod locations 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). Trojan fuel assemblies not loaded into DFCs or FFCs are classified as intact assemblies.

Minimum Enrichment is the minimum assembly average enrichment. Natural uranium blankets are not considered in determining minimum enrichment.

Non-Fuel Hardware is defined as Burnable Poison Rod Assemblies (BPRA), Thimble Plug Devices (TDPs), and Rod Cluster Control Assemblies (RCCAs).

Planar-Average Initial Enrichment is the average of the distributed fuel rod initial enrichments within a given axial plane of the assembly lattice.

Trojan Damaged Fuel Containers (or Canisters) are Holtec damaged fuel containers custom-designed for Trojan plant damaged fuel and fuel debris as depicted in Drawing 4119, Rev. 1.

Trojan Failed Fuel Cans are non-Holtec designed Trojan plant-specific damaged fuel containers that may be loaded with Trojan plant damaged fuel assemblies, Trojan fuel assembly metal fragments (e.g., portions of fuel rods and grid assemblies, bottom nozzles, etc.), a Trojan fuel rod storage container, a Trojan Fuel Debris Process Can Capsule, or a Trojan Fuel Debris Process Can. The Trojan Failed Fuel Can is depicted in Drawings PFFC-001, Rev. 8 and PFFC-002, Rev. 7.

Trojan Fuel Debris Process Cans are Trojan plant-specific canisters containing fuel debris (metal fragments) and were used to process organic media removed from the Trojan plant spent fuel pool during cleanup operations in preparation for spent fuel pool decommissioning. Trojan Fuel Debris Process Cans are loaded into Trojan Fuel Debris Process Can Capsules or directly into Trojan Failed Fuel Cans. The Trojan Fuel Debris Process Can is depicted in Figure 1.2.10B of the HI-STAR100 System SAR, Rev. 12.

Trojan Fuel Debris Process Can Capsules are Trojan plant-specific canisters that contain up to five Trojan Fuel Debris Process Cans and are vacuumed, purged, backfilled with helium and then seal-welded closed. The Trojan Fuel Debris Process Can Capsule is depicted in Figure 1.2.10C of the HI-STAR 100 System SAR, Rev. 12.

5.(b)(1)(b) Definitions (continued)

ZR means any zirconium-based fuel cladding materials authorized for use in a commercial nuclear power plant reactor.

(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 decay heat limits for the stainless steel clad fuel assemblies or the applicable ZR clad fuel assemblies.

(d) For MPCs partially loaded with damaged fuel assemblies or fuel debris, all remaining ZR clad intact fuel assemblies in the MPC shall meet the more restrictive of the decay heat limits for the damaged fuel assemblies or the intact fuel assemblies.

(e) For MPC-68s partially loaded with array/class 6x6A, 6x6B, 6x6C, or 8x8A fuel assemblies, all remaining ZR clad intact fuel assemblies in the MPC shall meet the more restrictive of the decay heat limits for the 6x6A, 6x6B, 6x6C, and 8x8A fuel assemblies or the applicable Zircaloy clad fuel assemblies.

(f) PWR non-fuel hardware and neutron sources are not authorized for transportation except as specifically provided for in Appendix A to this CoC.

(g) BWR stainless-steel channels and control blades are not authorized for transportation.

(h) For spent fuel assemblies to be loaded into MPC-32s, core average soluble boron, assembly average specific power, and assembly average moderator temperature in which the fuel assemblies were irradiated, shall be determined according to Section 1.2.3.7.1 of the SAR, and the values shall be compared against the limits specified in Part VI of Table A.1 in Appendix A of this Certificate of Compliance.

(i) For spent fuel assemblies to be loaded into MPC 32s, the reactor records on spent fuel assemblies average burnup shall be confirmed through physical burnup measurements as described in Section 1.2.3.7.2 of the SAR.

5.(c) Criticality Safety Index (CSI)= 0.0

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:

6.(a) (continued)

(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 requirement 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 total 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) within the 12-month period prior to each shipment;
(ii) after detensioning one or more overpack lid bolts, drain port, or the vent port plug; and
(iii) after each seal replacement.

(b) Within 30 days before each shipment, all overpack 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 overpack containment boundary seal must be replaced after each use of the seal.

(4) The relief devices on the neutron shield vessel shall be replaced every 5 years.

(5) MPC-68F and MPC-24EF shall be leak tested prior to shipment to show a leak rate of no greater than 5×10^{-6} atm cm^3/sec (helium). The leak test shall have a minimum sensitivity of 2.5×10^{-6} atm cm^3/sec (helium).

(6) MPCs deployed at an ISFSI under 10 CFR Part 72 prior to transportation may be dried using the vacuum drying method or the Forced Helium Dehydration (FHD) method. MPCs placed directly into transportation service under 10 CFR 71 without first being deployed at an ISFSI must be dried using the FHD method. Water and residual moisture shall be removed from the MPC in accordance with the following specifications:

For those MPCs vacuum dried:

(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.

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6.(a) (continued)

For those MPCs dried using the FHD System:

(a) Following bulk moisture removal, the temperature of the gas exiting the demoisturizer shall be ≤ 21°F for ≥ 30 minutes.

(7) Following drying, the MPC shall be backfilled with 99.995% minimum purity helium: > 0 psig and ≤ 44.8 psig at a reference temperature of 70 °F.

(8) Water and residual moisture shall be removed from the HI-STAR 100 overpack in accordance with the following specifications:

(a) The overpack annulus shall be evacuated to a pressure of less than or equal to 3 torr.

(b) The overpack annulus 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 ≥ 10 psig and ≤ 14 psig.

(10) The following fasteners shall be tightened to the torque values specified below:

Fastener	Torque (ft-lbs)
Overpack Closure Plate Bolts	2895 ± 90
Overpack Vent and Drain Port Plugs	45 +5/-2
Top Impact Limiter Attachment Bolts	256 +10/-0
Bottom Impact Limiter Attachment Bolts	1500 +45/-0

(11) Verify that the appropriate fuel spacers, as necessary, are used to position the active fuel zone within the neutron absorber plates of the MPC, and limit axial movement of fuel assemblies in the MPC cavity.

(12) Appropriate monitoring for combustible gas concentration shall be performed prior to, and during MPC lid welding and weld cutting operations. The space below the MPC lid shall be exhausted or purged with inert gas prior to, and during MPC lid welding and weld cutting operations to provide additional assurance that flammable gas concentrations will not develop in this space.

(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 in accordance with ASME Section III, Subsection NB, Article NB-6110 and applicable sub-articles. If hydrostatic testing is used, the

6.(b) (continued)

MPC shall be pressure tested to 125% of the design pressure. The minimum hydrostatic test pressure shall be 125 psig. If pneumatic testing is used, the MPC shall be pressure tested to 120% of the design pressure. The minimum pneumatic test pressure shall be 120 psig.

(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 results, including all relevant 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 prior to each shipment. The periodic verification shall be performed by radiation measurements with either loaded contents or a check source. Measurements shall be taken at three cross sectional planes through the radial shield and at four points along each plane's circumference. The average measurement results from each sectional plane shall be compared to calculated values to assess the continued effectiveness of the neutron shield. The calculated values shall be representative of the loaded contents (i.e., fuel type, enrichment, burnup, cooling time, etc.) or the particular check source used for the measurements.

(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 5.(a)(3) of this Certificate of Compliance. 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.

6.(b) (continued)

(8) For each package, a periodic thermal performance test shall be performed within 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-24E, MPC-24EF, and 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) Flux trap sizes:

(a) The minimum flux trap size for the MPC-24 is 1.09 inches.

(b) The minimum flux trap sizes for the generic MPC-24E and MPC-24EF are 0.776 inch for cells 3, 6, 19, and 22; and 1.076 inch for the remaining cells.

(c) The minimum flux trap sizes for the Trojan MPC-24E and MPC-24EF are 0.526 inch for cells 3, 6, 19, and 22; and 1.076 inch for the remaining cells.

(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 14.5-1997.

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 9 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 71.17.

11. Revision No. 4 of this certificate may be used until October 12, 2007.

12. Expiration Date: March 31, 2009

Attachment: Appendix A

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9261	5	71-9261	USA/9261/B(U)F-85	10	OF	10

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 12, dated October 9, 2006.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Robert A. Nelson, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: October 12, 2006

APPENDIX A

CERTIFICATE OF COMPLIANCE NO. 9261, REVISION 5

MODEL NO. HI-STAR 100 SYSTEM

INDEX TO APPENDIX A

Page:	Table:	Description:
Page A-1 to A-19	Table A.1	Fuel Assembly Limits
Page A-1		MPC-24: Uranium oxide, PWR intact fuel assemblies listed in Table A.2.
A-2		MPC-68: Uranium oxide, BWR intact fuel assemblies listed in Table A.3 with or without Zircaloy channels.
A-3		MPC-68: Uranium oxide, BWR damaged fuel assemblies, with or without Zircaloy channels, placed in damaged fuel containers. Uranium oxide BWR damaged fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6A, 6X6C, 7x7A, or 8x8A.
A-4		MPC-68: Mixed oxide (MOX), BWR intact fuel assemblies, with or without Zircaloy channels. MOX BWR intact fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6B.
A-5		MPC-68: Mixed oxide (MOX), BWR damaged fuel assemblies, with or without Zircaloy channels, placed in damaged fuel containers. MOX BWR damaged fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6B.
A-6		MPC-68: Thoria rods (ThO_2 and UO_2) placed in Dresden Unit 1 Thoria Rod Canisters
A-7		MPC-68F: Uranium oxide, BWR intact fuel assemblies, with or without Zircaloy channels. Uranium oxide BWR intact fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6A, 6x6C, 7x7A, or 8x8A.
A-8		MPC-68F: Uranium oxide, BWR damaged fuel assemblies, with or without Zircaloy channels, placed in damaged fuel containers. Uranium oxide BWR damaged fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6A, 6x6C, 7x7A, or 8x8A.
A-9		MPC-68F: Uranium oxide, BWR fuel debris, with or without Zircaloy channels, placed in damaged fuel containers. The original fuel assemblies for the uranium oxide BWR fuel debris shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6A, 6x6C, 7x7A, or 8x8A.

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INDEX TO APPENDIX A

Table A.1 (Page 1 of 21)
Fuel Assembly Limits

I. MPC MODEL: MPC-24

A. Allowable Contents

1. Uranium oxide, PWR intact fuel assemblies listed in Table A.2 and meeting the following specifications:

a. Cladding type:	ZR or stainless steel (SS) as specified in Table A.2 for the applicable fuel assembly array/class
b. Maximum initial enrichment:	As specified in Table A.2 for the applicable fuel assembly array/class.
c. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly	
i. ZR clad:	An assembly post-irradiation cooling time, average burnup, and minimum initial enrichment as specified in Table A.4 or A.5, as applicable.
ii. SS clad:	An assembly post-irradiation cooling time, average burnup, and minimum initial enrichment as specified in Table A.6, as applicable.
d. Decay heat per assembly:	
i. ZR Clad:	≤833 Watts
ii. SS Clad:	≤488 Watts
e. Fuel assembly length:	≤ 176.8 inches (nominal design)
f. Fuel assembly width:	≤ 8.54 inches (nominal design)
g. Fuel assembly weight:	≤ 1,680 lbs

B. Quantity per MPC: Up to 24 PWR fuel assemblies.

C. Fuel assemblies shall not contain non-fuel hardware or neutron sources.

D. Damaged fuel assemblies and fuel debris are not authorized for transport in the MPC-24.

E. Trojan plant fuel is not permitted to be transported in the MPC-24.

Table A.1 (Page 2 of 21)
Fuel Assembly Limits

II. MPC MODEL: MPC-68

A. Allowable Contents

1. Uranium oxide, BWR intact fuel assemblies listed in Table A.3, with or without Zircaloy channels, and meeting the following specifications:

a. Cladding type:	ZR or stainless steel (SS) as specified in Table A.3 for the applicable fuel assembly array/class.
b. Maximum planar-average initial enrichment:	As specified in Table A.3 for the applicable fuel assembly array/class.
c. Initial maximum rod enrichment:	As specified in Table A.3 for the applicable fuel assembly array/class.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	
i. ZR clad:	An assembly post-irradiation cooling time, average burnup, and minimum initial enrichment as specified in Table A.7, except for (1) array/class 6x6A, 6x6C, 7x7A, and 8x8A fuel assemblies, which shall have a cooling time ≥ 18 years, an average burnup ≤ 30,000 MWD/MTU, and a minimum initial enrichment ≥ 1.8 wt% ^{235}U, and (2) array/class 8x8F fuel assemblies, which shall have a cooling time ≥ 10 years, an average burnup ≤ 27,500 MWD/MTU, and a minimum initial enrichment ≥ 2.4 wt% ^{235}U.
ii. SS clad:	An assembly cooling time after discharge ≥ 16 years, an average burnup ≤ 22,500 MWD/MTU, and a minimum initial enrichment ≥ 3.5 wt% ^{235}U.
e. Decay heat per assembly:	
i. ZR Clad:	≤272 Watts, except for array/class 8X8F fuel assemblies, which shall have a decay heat ≤183.5 Watts.
ii SS Clad:	≤83 Watts
f. Fuel assembly length:	≤ 176.2 inches (nominal design)
g. Fuel assembly width:	≤ 5.85 inches (nominal design)
h. Fuel assembly weight:	≤ 700 lbs, including channels

Table A.1 (Page 3 of 21)
Fuel Assembly Limits

II. MPC MODEL: MPC-68 (continued)

A. Allowable Contents (continued)

2. Uranium oxide, BWR damaged fuel assemblies, with or without Zircaloy channels, placed in damaged fuel containers. Uranium oxide BWR damaged fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6A, 6x6C, 7x7A, or 8x8A, and meet the following specifications:

a. Cladding type:	ZR
b. Maximum planar-average initial enrichment:	As specified in Table A.3 for the applicable fuel assembly array/class.
c. Initial maximum rod enrichment:	As specified in Table A.3 for the applicable fuel assembly array/class.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time $\geq$ 18 years, an average burnup $\leq$ 30,000 MWD/MTU, and a minimum initial enrichment $\geq$ 1.8 wt% ^{235}U.
e. Fuel assembly length:	$\leq$ 135.0 inches (nominal design)
f. Fuel assembly width:	$\leq$ 4.70 inches (nominal design)
g. Fuel assembly weight:	$\leq$ 550 lbs, including channels and damaged fuel container

Table A.1 (Page 4 of 21)
Fuel Assembly Limits

II. MPC MODEL: MPC-68 *(continued)*

A. Allowable Contents (continued)

3. Mixed oxide (MOX), BWR intact fuel assemblies, with or without Zircaloy channels. MOX BWR intact fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6B and meet the following specifications:

a. Cladding type:	ZR
b. Maximum planar-average initial enrichment:	As specified in Table A.3 for fuel assembly array/class 6x6B.
c. Initial maximum rod enrichment:	As specified in Table A.3 for fuel assembly array/class 6x6B.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time $\geq$ 18 years, an average burnup $\leq$ 30,000 MWD/MTIHM, and a minimum initial enrichment $\geq$ 1.8 wt% ^{235}U for the UO_2 rods.
e. Fuel assembly length:	$\leq$ 135.0 inches (nominal design)
f. Fuel assembly width:	$\leq$ 4.70 inches (nominal design)
g. Fuel assembly weight:	$\leq$ 400 lbs, including channels

Table A.1 (Page 5 of 21)
Fuel Assembly Limits

II. MPC MODEL: MPC-68 (continued)

A. Allowable Contents (continued)

4. Mixed oxide (MOX), BWR damaged fuel assemblies, with or without Zircaloy channels, placed in damaged fuel containers. MOX BWR damaged fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6B and meet the following specifications:

a. Cladding type:	ZR
b. Maximum planar-average initial enrichment:	As specified in Table A.3 for array/class 6x6B.
c. Initial maximum rod enrichment:	As specified in Table A.3 for array/class 6x6B.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time $\geq$ 18 years, an average burnup $\leq$ 30,000 MWD/MTIHM, and a minimum initial enrichment $\geq$ 1.8 wt% ^{235}U for the UO_2 rods.
e. Fuel assembly length:	$\leq$ 135.0 inches (nominal design)
f. Fuel assembly width:	$\leq$ 4.70 inches (nominal design)
g. Fuel assembly weight:	$\leq$ 550 lbs, including channels

Table A.1 (Page 6 of 21)
Fuel Assembly Limits

II. MPC MODEL: MPC-68 (continued)

A. Allowable Contents (continued)

5. Thoria rods (ThO_2 and UO_2) placed in Dresden Unit 1 Thoria Rod Canisters (as shown in Figure 1.2.11A of the HI-STAR 100 System SAR, Revision 12) and meeting the following specifications:

a. Cladding type:	ZR
b. *Composition:*	98.2 wt.% ThO_2, 1.8 wt. % UO_2 with an enrichment of 93.5 wt. % ^{235}U.
c. Number of rods per Thoria Rod Canister:	$\leq$ 18
d. Decay heat per Thoria Rod Canister:	$\leq$ 115 Watts
e. Post-irradiation fuel cooling time and average burnup per Thoria Rod Canister:	A fuel post-irradiation cooling time $\geq$ 18 years and an average burnup $\leq$ 16,000 MWD/MTIHM.
f. Initial heavy metal weight:	$\leq$ 27 kg/canister
g. Fuel cladding O.D.:	$\geq$ 0.412 inches
h. Fuel cladding I.D.:	$\leq$ 0.362 inches
i. Fuel pellet O.D.:	$\leq$ 0.358 inches
j. Active fuel length:	$\leq$ 111 inches
k. Canister weight:	$\leq$ 550 lbs, including fuel

B. Quantity per MPC: Up to one (1) Dresden Unit 1 Thoria Rod Canister plus any combination of damaged fuel assemblies in damaged fuel containers and intact fuel assemblies, up to a total of 68.

C. Fuel assemblies with stainless steel channels are not authorized for loading in the MPC-68.

D. Dresden Unit 1 fuel assemblies (fuel assembly array/class 6x6A, 6x6B, 6x6C, or 8x8A) with one Antimony-Beryllium neutron source are authorized for loading in the MPC-68. The Antimony-Beryllium source material shall be in a water rod location.

Table A.1 (Page 7 of 21)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F

A. Allowable Contents

1. Uranium oxide, BWR intact fuel assemblies, with or without Zircaloy channels. Uranium oxide BWR intact fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6A, 6x6C, 7x7A, or 8x8A and meet the following specifications:

a. Cladding type:	ZR
b. Maximum planar-average initial enrichment:	As specified in Table A.3 for the applicable fuel assembly array/class.
c. Initial maximum rod enrichment:	As specified in Table A.3 for the applicable fuel assembly array/class.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time ≥ 18 years, an average burnup ≤ 30,000 MWD/MTU, and a minimum initial enrichment ≥ 1.8 wt% ^{235}U.
e. Fuel assembly length:	≤ 176.2 inches (nominal design)
f. Fuel assembly width:	≤ 5.85 inches (nominal design)
g. Fuel assembly weight:	≤ 400 lbs, including channels

Table A.1 (Page 8 of 21)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (continued)

A. Allowable Contents (continued)

2. Uranium oxide, BWR damaged fuel assemblies, with or without Zircaloy channels, placed in damaged fuel containers. Uranium oxide BWR damaged fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6A, 6x6C, 7x7A, or 8x8A, and meet the following specifications:

a. Cladding type:	ZR
b. Maximum planar-average initial enrichment:	As specified in Table A.3 for the applicable fuel assembly array/class.
c. Initial maximum rod enrichment:	As specified in Table A.3 for the applicable fuel assembly array/class.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time $\geq$ 18 years, an average burnup $\leq$ 30,000 MWD/MTU, and a minimum initial enrichment $\geq$ 1.8 wt% ^{235}U.
e. Fuel assembly length:	$\leq$ 135.0 inches (nominal design)
f. Fuel assembly width:	$\leq$ 4.70 inches (nominal design)
g. Fuel assembly weight:	$\leq$ 550 lbs, including channels

Table A.1 (Page 9 of 21)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (continued)

A. Allowable Contents (continued)

3. Uranium oxide, BWR fuel debris, with or without Zircaloy channels, placed in damaged fuel containers. The original fuel assemblies for the uranium oxide BWR fuel debris shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6A, 6x6C, 7x7A, or 8x8A, and meet the following specifications:

a. Cladding type:	ZR
b. Maximum planar-average initial enrichment:	As specified in Table A.3 for the applicable original fuel assembly array/class.
c. Initial maximum rod enrichment:	As specified in Table A.3 for the applicable original fuel assembly array/class.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time ≥ 18 years, an average burnup ≤ 30,000 MWD/MTU, and a minimum initial enrichment ≥ 1.8 wt% ^{235}U for the original fuel assembly.
e. Fuel assembly length:	≤ 135.0 inches (nominal design)
f. Fuel assembly width:	≤ 4.70 inches (nominal design)
g. Fuel assembly weight:	≤ 550 lbs, including channels

Table A.1 (Page 10 of 21)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (continued)

A. Allowable Contents (continued)

4. Mixed oxide(MOX), BWR intact fuel assemblies, with or without Zircaloy channels. MOX BWR intact fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6B and meet the following specifications:

a. Cladding type:	ZR
b. Maximum planar-average initial enrichment:	As specified in Table A.3 for fuel assembly array/class 6x6B.
c. Initial maximum rod enrichment:	As specified in Table A.3 for fuel assembly array/class 6x6B.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time ≥ 18 years, an average burnup ≤ 30,000 MWD/MTIHM, and a minimum initial enrichment ≥ 1.8 wt% ^{235}U for the UO_2 rods.
e. Fuel assembly length:	≤ 135.0 inches (nominal design)
f. Fuel assembly width:	≤ 4.70 inches (nominal design)
g. Fuel assembly weight:	≤ 400 lbs, including channels

Table A.1 (Page 11 of 21)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (continued)

A. Allowable Contents (continued)

5. Mixed oxide (MOX), BWR damaged fuel assemblies, with or without Zircaloy channels, placed in damaged fuel containers. MOX BWR intact fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6B and meet the following specifications:

a. Cladding type:	ZR
b. Maximum planar-average initial enrichment:	As specified in Table A.3 for array/class 6x6B.
c. Initial maximum rod enrichment:	As specified in Table A.3 for array/class 6x6B.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time $\geq$ 18 years, an average burnup $\leq$ 30,000 MWD/MTIHM, and a minimum initial enrichment $\geq$ 1.8 wt% ^{235}U for the UO_2 rods.
e. Fuel assembly length:	$\leq$ 135.0 inches (nominal design)
f. Fuel assembly width:	$\leq$ 4.70 inches (nominal design)
g. Fuel assembly weight:	$\leq$ 550 lbs, including channels

Table A.1 (Page 12 of 21)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (continued)

A. Allowable Contents (continued)

6. Mixed oxide (MOX), BWR fuel debris, with or without Zircaloy channels, placed in damaged fuel containers. The original fuel assemblies for the MOX BWR fuel debris shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6B and meet the following specifications:

a. *Cladding type:*	ZR
b. Maximum planar-average initial enrichment:	As specified in Table A.3 for original fuel assembly array/class 6x6B.
c. Initial maximum rod enrichment:	As specified in Table A.3 for original fuel assembly array/class 6x6B.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time $\geq$ 18 years, an average burnup $\leq$ 30,000 MWD/MTIHM, and a minimum initial enrichment $\geq$ 1.8 wt% ^{235}U for the UO_2 rods in the original fuel assembly.
e. Fuel assembly length:	$\leq$ 135.0 inches (nominal design)
f. Fuel assembly width:	$\leq$ 4.70 inches (nominal design)
g. Fuel assembly weight:	$\leq$ 550 lbs, including channels

Table A.1 (Page 13 of 21)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (continued)

A. Allowable Contents (continued)

7. Thoria rods (ThO_2 and UO_2) placed in Dresden Unit 1 Thoria Rod Canisters (as shown in Figure 1.2.11A of the HI-STAR 100 System SAR, Revision 12) and meeting the following specifications:

a. Cladding Type:	ZR
b. Composition:	98.2 wt.% ThO_2, 1.8 wt. % UO_2 with an enrichment of 93.5 wt. % ^{235}U.
c. Number of rods per Thoria Rod Canister:	$\leq$ 18
d. Decay heat per Thoria Rod Canister:	$\leq$ 115 Watts
e. Post-irradiation fuel cooling time and average burnup per Thoria Rod Canister:	A fuel post-irradiation cooling time $\geq$ 18 years and an average burnup $\leq$ 16,000 MWD/MTIHM.
f. Initial heavy metal weight:	$\leq$ 27 kg/canister
g. Fuel cladding O.D.:	$\geq$ 0.412 inches
h. Fuel cladding I.D.:	$\leq$ 0.362 inches
i. Fuel pellet O.D.:	$\leq$ 0.358 inches
j. Active fuel length:	$\leq$ 111 inches
k. Canister weight:	$\leq$ 550 lbs, including fuel

Table A.1 (Page 14 of 21)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (continued)

B. Quantity per MPC:

Up to four (4) damaged fuel containers containing uranium oxide or MOX BWR fuel debris. The remaining MPC-68F fuel storage locations may be filled with array/class 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A fuel assemblies of the following type, as applicable:

1. Uranium oxide BWR intact fuel assemblies;
2. MOX BWR intact fuel assemblies;
3. Uranium oxide BWR damaged fuel assemblies placed in damaged fuel containers;
4. MOX BWR damaged fuel assemblies placed in damaged fuel containers; or
5. Up to one (1) Dresden Unit 1 Thoria Rod Canister.

C. Fuel assemblies with stainless steel channels are not authorized for loading in the MPC-68F.

D. Dresden Unit 1 fuel assemblies (fuel assembly array/class 6x6A, 6x6B, 6x6C or 8x8A) with one Antimony-Beryllium neutron source are authorized for loading in the MPC-68F. The Antimony-Beryllium neutron source material shall be in a water rod location.

Table A.1 (Page 15 of 21)
Fuel Assembly Limits

IV. MPC MODEL: MPC-24E

A. Allowable Contents

1. Uranium oxide, PWR intact fuel assemblies listed in Table A.2 and meeting the following specifications:

a. Cladding type:	ZR or stainless steel (SS) as specified in Table A.2 for the applicable fuel assembly array/class
b. Maximum initial enrichment:	As specified in Table A.2 for the applicable fuel assembly array/class.
c. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly	
i. ZR clad:	Except for Trojan plant fuel, an assembly post-irradiation cooling time, average burnup, and minimum initial enrichment as specified in Table A.4 or A.5, as applicable.
ii. SS clad:	An assembly post-irradiation cooling time, average burnup, and minimum initial enrichment as specified in Table A.6, as applicable.
iii Trojan plant fuel	An assembly post-irradiation cooling time, average burnup, and minimum initial enrichment as specified in Table A.8.
iv Trojan plant non-fuel hardware and neutron sources	Post-irradiation cooling time, and average burnup as specified in Table A.9
d. Decay heat per assembly	
i. ZR Clad:	Except for Trojan plant fuel, decay heat $\leq$ 833 Watts. Trojan plant fuel decay heat: $\leq$ 725 Watts
ii. SS Clad:	$\leq$ 488 Watts
e. Fuel assembly length:	$\leq$ 176.8 inches (nominal design)
f. Fuel assembly width:	$\leq$ 8.54 inches (nominal design)
g. Fuel assembly weight:	$\leq$ 1,680 lbs, including non-fuel hardware and neutron sources

Table A.1 (Page 16 of 21)
Fuel Assembly Limits

IV. MPC MODEL: MPC-24E

A. Allowable Contents (continued)

2. Trojan plant damaged fuel assemblies meeting the applicable criteria listed in Table A.2 and meeting the following specifications:

a. Cladding type:	ZR
b. Maximum initial enrichment:	3.7% ^{235}U
c. Fuel assembly post-irradiation cooling time, average burnup, decay heat, and minimum initial enrichment per assembly	An assembly post-irradiation cooling time, average burnup, and initial enrichment as specified in Table A.8 Decay Heat: $\leq$ 725 Watts
d. Fuel assembly length:	$\leq$ 169.3 inches (nominal design)
e. Fuel assembly width:	$\leq$ 8.43 inches (nominal design)
f. Fuel assembly weight:	$\leq$ 1,680 lbs, including DFC or Failed Fuel Can

B. Quantity per MPC: Up to 24 PWR intact fuel assemblies. For Trojan plant fuel only, up to four (4) damaged fuel assemblies may be stored in fuel storage locations 3, 6, 19, and/or 22. The remaining MPC-24E fuel storage locations may be filled with Trojan plant intact fuel assemblies.

C. Trojan plant fuel must be transported in the custom-designed Trojan MPCs with the MPC spacer installed. Fuel from other plants is not permitted to be transported in the Trojan MPCs.

D. Except for Trojan plant fuel, the fuel assemblies shall not contain non-fuel hardware or neutron sources. *Trojan intact fuel assemblies containing non-fuel hardware may be transported in any fuel storage location.*

E. Trojan plant damaged fuel assemblies must be transported in a Trojan Failed Fuel Can or a Holtec damaged fuel container designed for Trojan Plant fuel.

F. One (1) Trojan plant Sb-Be and /or up to two (2) Cf neutron sources in a Trojan plant intact fuel assembly (one source fuel assembly) may be transported in any one MPC. Each fuel assembly neutron source may be transported in any fuel storage location.

G. Fuel debris is not authorized for transport in the MPC-24E.

H. Trojan plant non-fuel hardware and neutron sources may not be transported in the same fuel storage location as a damaged fuel assembly.

Table A.1 (Page 17 of 21)
Fuel Assembly Limits

V. MPC MODEL: MPC-24EF

A. Allowable Contents

1. Uranium oxide, PWR intact fuel assemblies listed in Table A.2 and meeting the following specifications:

a. Cladding type:	ZR or stainless steel (SS) as specified in Table A.2 for the applicable fuel assembly array/class.
b. Maximum initial enrichment:	As specified in Table A.2 for the applicable fuel assembly array/class.
c. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly	
i. ZR clad:	Except for Trojan plant fuel, an assembly post-irradiation cooling time, average burnup, and minimum initial enrichment as specified in Table A.4 or A.5, as applicable.
ii. SS clad:	An assembly post-irradiation cooling time, average burnup, and minimum initial enrichment as specified in Table A.6, as applicable.
iii Trojan plant fuel:	An assembly post-irradiation cooling time, average burnup, and minimum initial enrichment as specified in Table A.8.
iv Trojan plant non-fuel hardware and neutron sources:	Post-irradiation cooling time, and average burnup as specified in Table A.9.
d. Decay heat per assembly:	
a. ZR Clad:	Except for Trojan plant fuel, decay heat ≤ 833 Watts. Trojan plant fuel decay heat: ≤ 725 Watts.
b. SS Clad:	≤ 488 Watts
e. Fuel assembly length:	≤ 176.8 inches (nominal design)
f. Fuel assembly width:	≤ 8.54 inches (nominal design)
g. Fuel assembly weight:	≤ 1,680 lbs, including non-fuel hardware and neutron sources.

Table A.1 (Page 18 of 21)
Fuel Assembly Limits

V. MPC MODEL: MPC-24EF

A. Allowable Contents (continued)

2. Trojan plant damaged fuel assemblies meeting the applicable criteria listed in Table A.2 and meeting the following specifications:

a. Cladding type:	ZR
b. Maximum initial enrichment:	3.7% ^{235}U
c. Fuel assembly post-irradiation cooling time, average burnup, decay heat, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time, average burnup, and initial enrichment as specified in Table A.8. Decay Heat: ≤ 725 Watts
d. Fuel assembly length:	≤ 169.3 inches (nominal design)
e. Fuel assembly width:	≤ 8.43 inches (nominal design)
f. Fuel assembly weight:	≤ 1,680 lbs, including DFC or Failed Fuel Can.

Table A.1 (Page 19 of 21)
Fuel Assembly Limits

V. MPC MODEL: MPC-24EF

A. Allowable Contents (continued)

3. Trojan Fuel Debris Process Can Capsules and/or Trojan plant fuel assemblies classified as fuel debris, for which the original fuel assemblies meet the applicable criteria listed in Table A.2 and meet the following specifications:

a. Cladding type:	ZR
b. Maximum initial enrichment:	3.7% ^{235}U
c. Fuel debris post-irradiation cooling time, average burnup, decay heat, and minimum initial enrichment per assembly:	Post-irradiation cooling time, average burnup, and initial enrichment as specified in Table A.8. Decay Heat: ≤ 725 Watts
d. Fuel assembly length:	≤ 169.3 inches (nominal design)
e. Fuel assembly width:	≤ 8.43 inches (nominal design)
f. Fuel assembly weight:	≤ 1,680 lbs, including DFC or Failed Fuel Can.

B. Quantity per MPC: Up to 24 PWR intact fuel assemblies. For Trojan plant fuel only, up to four (4) damaged fuel assemblies, fuel assemblies classified as fuel debris, and/or Trojan Fuel Debris Process Can Capsules may be stored in fuel storage locations 3, 6, 19, and/or 22. The remaining MPC-24EF fuel storage locations may be filled with Trojan plant intact fuel assemblies.

C. Trojan plant fuel must be transported in the custom-designed Trojan MPCs with the MPC spacer installed. Fuel from other plants is not permitted to be transported in the Trojan MPCs.

D. Except for Trojan plant fuel, the fuel assemblies shall not contain non-fuel hardware or neutron sources. Trojan intact fuel assemblies containing non-fuel hardware may be transported in any fuel storage location.

E. Trojan plant damaged fuel assemblies, fuel assemblies classified as fuel debris, and Fuel Debris Process Can Capsules must be transported in a Trojan Failed Fuel Can or a Holtec damaged fuel container designed for Trojan Plant fuel.

F. One (1) Trojan plant Sb-Be and /or up to two (2) Cf neutron sources in a Trojan plant intact fuel assembly (one source fuel assembly) may be transported in any one MPC. Each fuel assembly neutron source may be transported in any fuel storage location.

G. Trojan plant non-fuel hardware and neutron sources may not be transported in the same fuel storage location as a damaged fuel assembly.

Table A.1 (Page 20 of 21)
Fuel Assembly Limits

VI. MPC MODEL: MPC-32

A. Allowable Contents

1. Uranium oxide, PWR intact fuel assemblies in array/classes 15x15D, E, F, and H and 17x17A, B, and C listed in Table A.2 and meeting the following specifications:

a. Cladding type:	ZR
b. Maximum initial enrichment:	As specified in Table A.2 for the applicable fuel assembly array/class.
c. Post-irradiation cooling time, maximum average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time, average burnup, and minimum initial enrichment as specified in Table A.10 or A.11, as applicable.
d. Minimum average burnup per assembly (Assembly Burnup shall be confirmed per Subsection 1.2.3.7.2 of the SAR, which is hereby included by reference)	Calculated value as a function of initial enrichment. See Table A.12.
e. Decay heat per assembly:	≤ 625 Watts
f. Fuel assembly length:	≤ 176.8 inches (nominal design)
g. Fuel assembly width:	≤ 8.54 inches (nominal design)
h. Fuel assembly weight:	≤ 1,680 lbs

i. Operating parameters during irradiation of the assembly (Assembly operating parameters shall be determined per Subsection 1.2.3.7.1 of the SAR, which is hereby included by reference)

Core ave. soluble boron concentration:	≤ 1,000 ppmb
Assembly ave. moderator temperature:	≤ 601 K for array/classes 15x15D, E, F, and H ≤ 610 K for array/classes 17x17A, B, and C
Assembly ave. specific power:	≤ 47.36 kW/kg-U for array/classes 15x15D, E, F, and H ≤ 61.61 kW/kg-U for array/classes 17x17A, B, and C

Table A.1 (Page 21 of 21)
Fuel Assembly Limits

VI. MPC MODEL: MPC-32 (continued)

B. Quantity per MPC: Up to 32 PWR intact fuel assemblies.

C. Fuel assemblies shall not contain non-fuel hardware.

D. Damaged fuel assemblies and fuel debris are not authorized for transport in MPC-32.

E. Trojan plant fuel is not permitted to be transported in the MPC-32.

Table A.2 (Page 1 of 4)
PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	**14x14A**	**14x14B**	**14x14C**	**14x14D**	**14x14E**
Clad Material (Note 2)	ZR	ZR	ZR	SS	Zr
Design Initial U (kg/assy.) (Note 3)	$\leq$ 407	$\leq$ 407	$\leq$ 425	$\leq$ 400	$\leq$ 206
Initial Enrichment (MPC-24, 24E, and 24EF) (wt % ^{235}U)	$\leq$ 4.6 (24) $\leq$ 5.0 (24E/EF)	$\leq$ 4.6 (24) $\leq$ 5.0 (24E/EF)	$\leq$ 4.6 (24) $\leq$ 5.0 (24E/EF)	$\leq$ 4.0 (24) $\leq$ 5.0 (24E/EF)	$\leq$ 5.0
No. of Fuel Rod Locations	179	179	176	180	173
Fuel Clad O.D. (in.)	$\geq$ 0.400	$\geq$ 0.417	$\geq$ 0.440	$\geq$ 0.422	$\geq$ 0.3415
Fuel Clad I.D. (in.)	$\leq$ 0.3514	$\leq$ 0.3734	$\leq$ 0.3880	$\leq$ 0.3890	$\leq$ 0.3175
Fuel Pellet Dia. (in.)	$\leq$ 0.3444	$\leq$ 0.3659	$\leq$ 0.3805	$\leq$ 0.3835	$\leq$ 0.3130
Fuel Rod Pitch (in.)	$\leq$ 0.556	$\leq$ 0.556	$\leq$ 0.580	$\leq$ 0.556	Note 6
Active Fuel Length (in.)	$\leq$ 150	$\leq$ 150	$\leq$ 150	$\leq$ 144	$\leq$ 102
No. of Guide Tubes	17	17	5 (Note 4)	16	0
Guide Tube Thickness (in.)	$\geq$ 0.017	$\geq$ 0.017	$\geq$ 0.038	$\geq$ 0.0145	N/A

Table A.2 (Page 2 of 4)
PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	**15x15A**	**15x15B**	**15x15C**	**15x15D**	**15x15E**	**15x15F**
Clad Material (Note 2)	ZR	ZR	ZR	ZR	ZR	ZR
Design Initial U (kg/assy.) (Note 3)	≤ 464	≤ 464	≤ 464	≤ 475	≤ 475	≤ 475
Initial Enrichment (MPC-24, 24E, and 24EF) (wt % ^{235}U)	≤ 4.1 (24) ≤ 4.5 (24E/EF)	≤ 4.1 (24) ≤ 4.5 (24E/EF)	≤ 4.1 (24) ≤ 4.5 (24E/EF)	≤ 4.1 (24) ≤ 4.5 (24E/EF)	≤ 4.1 (24) ≤ 4.5 (24E/EF)	≤ 4.1 (24) ≤ 4.5 (24E/EF)
Initial Enrichment (MPC-32) (wt. % ^{235}U) (Note 5)	N/A	N/A	N/A	(Note 5)	(Note 5)	(Note 5)
No. of Fuel Rod Locations	204	204	204	208	208	208
Fuel Clad O.D. (in.)	≥ 0.418	≥ 0.420	≥ 0.417	≥ 0.430	≥ 0.428	≥ 0.428
Fuel Clad I.D. (in.)	≤ 0.3660	≤ 0.3736	≤ 0.3640	≤ 0.3800	≤ 0.3790	≤ 0.3820
Fuel Pellet Dia. (in.)	≤ 0.3580	≤ 0.3671	≤ 0.3570	≤ 0.3735	≤ 0.3707	≤ 0.3742
Fuel Rod Pitch (in.)	≤ 0.550	≤ 0.563	≤ 0.563	≤ 0.568	≤ 0.568	≤ 0.568
Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Guide and/or Instrument Tubes	21	21	21	17	17	17
Guide/Instrument Tube Thickness (in.)	≥ 0.015	≥ 0.015	≥ 0.0165	≥ 0.0150	≥ 0.0140	≥ 0.0140

Table A.2 (Page 3 of 4)
PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/ Class	**15x15G**	**15x15H**	**16x16A**	**17x17A**	**17x17B**	**17x17C**
Clad Material (Note 2)	SS	ZR	ZR	ZR	ZR	ZR
Design Initial U (kg/assy.) (Note 3)	≤ 420	≤ 475	≤ 443	≤ 467	≤ 467	≤ 474
Initial Enrichment (MPC-24, 24E, and 24EF) (wt % ^{235}U)	≤ 4.0 (24) ≤ 4.5 (24E/EF)	≤ 3.8 (24) ≤ 4.2 (24E/EF)	≤ 4.6 (24) ≤ 5.0 (24E/EF)	≤ 4.0 (24) ≤ 4.4 (24E/EF)	≤ 4.0 (24) ≤ 4.4 (24E/EF) (Note 7)	≤ 4.0 (24) ≤ 4.4 (24E/EF)
Initial Enrichment (MPC-32) (wt. % ^{235}U) (Note 5)	N/A	(Note 5)	N/A	(Note 5)	(Note 5)	(Note 5)
No. of Fuel Rod Locations	204	208	236	264	264	264
Fuel Clad O.D. (in.)	≥ 0.422	≥ 0.414	≥ 0.382	≥ 0.360	≥ 0.372	≥ 0.377
Fuel Clad I.D. (in.)	≤ 0.3890	≤ 0.3700	≤ 0.3320	≤ 0.3150	≤ 0.3310	≤ 0.3330
Fuel Pellet Dia. (in.)	≤ 0.3825	≤ 0.3622	≤ 0.3255	≤ 0.3088	≤ 0.3232	≤ 0.3252
Fuel Rod Pitch (in.)	≤ 0.563	≤ 0.568	≤ 0.506	≤ 0.496	≤ 0.496	≤ 0.502
Active Fuel Length (in.)	≤ 144	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Guide and/or Instrument Tubes	21	17	5 (Note 4)	25	25	25
Guide/Instrument Tube Thickness (in.)	≥ 0.0145	≥ 0.0140	≥ 0.0400	≥ 0.016	≥ 0.014	≥ 0.020

Table A.2 (Page 4 of 4)
PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Notes:

1. All dimensions are design nominal values. Maximum and minimum dimensions are specified to bound variations in design nominal values among fuel assemblies within a given array/class.

2. ZR Designates cladding material made of Zirconium or Zirconium alloys.

3. Design initial uranium weight is the nominal uranium weight specified for each assembly by the fuel manufacturer or reactor user. For each PWR fuel assembly, the total uranium weight limit specified in this table may be increased up to 2.0 percent for comparison with users' fuel records to account for manufacturer tolerances.

4. Each guide tube replaces four fuel rods.

5. Minimum burnup and maximum initial enrichment as specified in Table A.12.

6. This fuel assembly array/class includes only the Indian Point Unit 1 fuel assembly. This fuel assembly has two pitches in different sectors of the assembly. These pitches are 0.441 inches and 0.453 inches

7. Trojan plant-specific fuel is governed by the limits specified for array/class 17x17B and will be transported in the custom-designed Trojan MPC-24E/EF canisters. The Trojan MPC-24E/EF design is authorized to transport only Trojan plant fuel with a maximum initial enrichment of 3.7 wt.% ^{235}U.

Table A.3 (Page 1 of 5)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	**6x6A**	**6x6B**	**6x6C**	**7x7A**	**7x7B**	**8x8A**
Clad Material (Note 2)	ZR	ZR	ZR	ZR	ZR	ZR
Design Initial U (kg/assy.) (Note 3)	≤ 110	≤ 110	≤ 110	≤ 100	≤ 195	≤ 120
Maximum planar-average initial *enrichment* (wt.% ^{235}U)	≤ 2.7	≤ 2.7 for the UO_2 rods. See Note 4 for MOX rods	≤ 2.7	≤ 2.7	≤ 4.2	≤ 2.7
Initial Maximum Rod Enrichment (wt.% ^{235}U)	≤ 4.0	≤ 4.0	≤ 4.0	≤ 5.5	≤ 5.0	≤ 4.0
No. of Fuel Rod Locations	35 or 36	35 or 36 (up to 9 MOX rods)	36	49	49	63 or 64
Fuel Clad O.D. (in.)	≥ 0.5550	≥ 0.5625	≥ 0.5630	≥ 0.4860	≥ 0.5630	≥ 0.4120
Fuel Clad I.D. (in.)	≤ 0.5105	≤ 0.4945	≤ 0.4990	≤ 0.4204	≤ 0.4990	≤ 0.3620
Fuel Pellet Dia. (in.)	≤ 0.4980	≤ 0.4820	≤ 0.4880	≤ 0.4110	≤ 0.4910	≤ 0.3580
Fuel Rod Pitch (in.)	≤ 0.710	≤ 0.710	≤ 0.740	≤ 0.631	≤ 0.738	≤ 0.523
Active Fuel Length (in.)	≤ 120	≤ 120	≤ 77.5	≤ 80	≤ 150	≤ 120
No. of Water Rods (Note 11)	1 or 0	1 or 0	0	0	0	1 or 0
Water Rod Thickness (in.)	≥ 0	≥ 0	N/A	N/A	N/A	≥ 0
Channel Thickness (in.)	≤ 0.060	≤ 0.060	≤ 0.060	≤ 0.060	≤ 0.120	≤ 0.100

Table A.3 (Page 2 of 5)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	8x8B	8x8C	8x8D	8x8E	8x8F	9x9A
Clad Material (Note 2)	ZR	ZR	ZR	ZR	ZR	ZR
Design Initial U (kg/assy.) (Note 3)	≤ 185	≤ 185	≤ 185	≤ 185	≤ 185	≤ 177
Maximum planar-average initial enrichment (wt.% ^{235}U)	≤ 4.2	≤ 4.2	≤ 4.2	≤ 4.2	< 4.0	≤ 4.2
Initial Maximum Rod Enrichment (wt.% ^{235}U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0
No. of Fuel Rod Locations	63 or 64	62	60 or 61	59	64	74/66 (Note 5)
Fuel Clad O.D. (in.)	≥ 0.4840	≥ 0.4830	≥ 0.4830	≥ 0.4930	≥ 0.4576	≥ 0.4400
Fuel Clad I.D. (in.)	≤ 0.4295	≤ 0.4250	0.4230	≤ 0.4250	≤ 0.3996	≤ 0.3840
Fuel Pellet Dia. (in.)	≤ 0.4195	≤ 0.4160	≤ 0.4140	≤ 0.4160	≤ 0.3913	≤ 0.3760
Fuel Rod Pitch (in.)	≤ 0.642	≤ 0.641	≤ 0.640	≤ 0.640	≤ 0.609	≤ 0.566
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Water Rods (Note 11)	1 or 0	2	1 - 4 (Note 7)	5	N/A (Note 12)	2
Water Rod Thickness (in.)	≥ 0.034	> 0.00	> 0.00	≥ 0.034	≥ 0.0315	> 0.00
Channel Thickness (in.)	≤ 0.120	≤ 0.120	≤ 0.120	≤ 0.100	≤ 0.055	≤ 0.120

Table A.3 (Page 3 of 5)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	**9x9B**	**9x9C**	**9x9D**	**9x9E (Note 13)**	**9x9F (Note 13)**	**9x9G**
Clad Material (Note 2)	ZR	ZR	ZR	ZR	ZR	ZR
Design Initial U (kg/assy.) (Note 3)	≤ 177	≤ 177	≤ 177	≤ 177	≤ 177	≤ 177
Maximum planar-average initial enrichment (wt.% ^{235}U)	≤ 4.2	≤ 4.2	≤ 4.2	≤ 4.0	≤ 4.0	≤ 4.2
Initial Maximum Rod Enrichment (wt.% ^{235}U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0
No. of Fuel Rods	72	80	79	76	76	72
Fuel Clad O.D. (in.)	≥ 0.4330	≥ 0.4230	≥ 0.4240	≥ 0.4170	≥ 0.4430	≥ 0.4240
Fuel Clad I.D. (in.)	≤ 0.3810	≤ 0.3640	≤ 0.3640	≤ 0.3640	≤ 0.3860	≤ 0.3640
Fuel Pellet Dia. (in.)	≤ 0.3740	≤ 0.3565	≤ 0.3565	≤ 0.3530	≤ 0.3745	≤ 0.3565
Fuel Rod Pitch (in.)	≤ 0.572	≤ 0.572	≤ 0.572	≤ 0.572	≤ 0.572	≤ 0.572
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Water Rods (Note 11)	1 (Note 6)	1	2	5	5	1 (Note 6)
Water Rod Thickness (in.)	> 0.00	≥ 0.020	≥ 0.0300	≥ 0.0120	≥ 0.0120	≥ 0.0320
Channel Thickness (in.)	≤ 0.120	≤ 0.100	≤ 0.100	≤ 0.120	≤ 0.120	≤ 0.120

Table A.3 (Page 4 of 5)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	10x10A	10x10B	10x10C	10x10D	10x10E
Clad Material (Note 2)	ZR	ZR	ZR	SS	SS
Design Initial U (kg/assy.) (Note 3)	≤ 186	≤ 186	≤ 186	≤ 125	≤ 125
Maximum planar-average initial enrichment (wt.% ^{235}U)	≤ 4.2	≤ 4.2	≤ 4.2	≤ 4.0	≤ 4.0
Initial Maximum Rod Enrichment (wt.% ^{235}U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0
No. of Fuel Rod Locations	92/78 (Note 8)	91/83 (Note 9)	96	100	96
Fuel Clad O.D. (in.)	≥ 0.4040	≥ 0.3957	≥ 0.3780	≥ 0.3960	≥ 0.3940
Fuel Clad I.D. (in.)	≤ 0.3520	≤ 0.3480	≤ 0.3294	≤ 0.3560	≤ 0.3500
Fuel Pellet Dia. (in.)	≤ 0.3455	≤ 0.3420	≤ 0.3224	≤ 0.3500	≤ 0.3430
Fuel Rod Pitch (in.)	≤ 0.510	≤ 0.510	≤ 0.488	≤ 0.565	≤ 0.557
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 83	≤ 83
No. of Water Rods (Note 11)	2	1 (Note 6)	5 (Note 10)	0	4
Water Rod Thickness (in.)	≥ 0.0300	> 0.00	≥ 0.031	N/A	≥ 0.022
Channel Thickness (in.)	≤ 0.120	≤ 0.120	≤ 0.055	≤ 0.080	≤ 0.080

Table A.3 (Page 5 of 5)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Notes:

1. All dimensions are design nominal values. Maximum and minimum dimensions are specified to bound variations in design nominal values among fuel assemblies within a given array/class.
2. ZR designates cladding material made from Zirconium or Zirconium alloys.
3. Design initial uranium weight is the uranium weight specified for each assembly by the fuel manufacturer or reactor user. For each BWR fuel assembly, the total uranium weight limit specified in this table may be increased up to 1.5% for comparison with users' fuel records to account for manufacturer's tolerances.
4. $\leq$ 0.635 wt. % ^{235}U and $\leq$ 1.578 wt. % total fissile plutonium (^{239}Pu and ^{241}Pu), (wt. % of total fuel weight, i.e., UO_2 plus PuO_2).
5. This assembly class contains 74 total fuel rods; 66 full length rods and 8 partial length rods.
6. Square, replacing nine fuel rods.
7. Variable
8. This assembly class contains 92 total fuel rods; 78 full length rods and 14 partial length rods.
9. This assembly class contains 91 total fuel rods, 83 full length rods and 8 partial length rods.
10. One diamond-shaped water rod replacing the four center fuel rods and four rectangular water rods dividing the assembly into four quadrants.
11. These rods may be sealed at both ends and contain Zr material in lieu of water.
12. This assembly is known as "QUAD+" and has four rectangular water cross segments dividing the assembly into four quadrants.
13. For the SPC 9x9-5 fuel assembly, each fuel rod must meet either the 9x9E or 9x9F set of limits for clad O.D., clad I.D., and pellet diameter.

Table A.4

FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND MINIMUM ENRICHMENT
MPC-24/24E/24/EF PWR FUEL WITH ZIRCALOY CLAD AND
WITH NON-ZIRCALOY IN-CORE GRID SPACERS

Post-irradiation Cooling Time (years)	Assembly Burnup (MWD/MTU)	Assembly Initial Enrichment (wt. % U-235)
$\geq$ 9	$\leq$ 24,500	$\geq$ 2.3
$\geq$ 11	$\leq$ 29,500	$\geq$ 2.6
$\geq$ 13	$\leq$ 34,500	$\geq$ 2.9
$\geq$ 15	$\leq$ 39,500	$\geq$ 3.2
$\geq$ 18	$\leq$ 44,500	$\geq$ 3.4

Table A.5

FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND MINIMUM ENRICHMENT
MPC-24/24E/24EF PWR FUEL WITH ZIRCALOY CLAD AND
WITH ZIRCALOY IN-CORE GRID SPACERS

Post-irradiation Cooling Time (years)	Assembly Burnup (MWD/MTU)	Assembly Initial Enrichment (wt. % U-235)
$\geq$ 6	$\leq$ 24,500	$\geq$ 2.3
$\geq$ 7	$\leq$ 29,500	$\geq$ 2.6
$\geq$ 9	$\leq$ 34,500	$\geq$ 2.9
$\geq$ 11	$\leq$ 39,500	$\geq$ 3.2
$\geq$ 14	$\leq$ 44,500	$\geq$ 3.4

Table A.6

FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND MINIMUM ENRICHMENT
MPC-24/24E/24EF PWR FUEL WITH STAINLESS STEEL CLAD

Post-irradiation Cooling Time (years)	Assembly Burnup (MWD/MTU)	Assembly Initial Enrichment (wt. % U-235)
≥ 19	≤ 30,000	≥ 3.1
≥ 24	≤ 40,000	≥ 3.1

Table A.7

FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND MINIMUM ENRICHMENT
MPC-68

Post-irradiation Cooling Time (years)	Assembly Burnup (MWD/MTU)	Assembly Initial Enrichment (wt. % U-235)
≥ 8	≤ 24,500	≥ 2.1
≥ 9	≤ 29,500	≥ 2.4
≥ 11	≤ 34,500	≥ 2.6
≥ 14	≤ 39,500	≥ 2.9
≥ 19	≤ 44,500	≥ 3.0

Table A.8

TROJAN PLANT FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND MINIMUM ENRICHMENT LIMITS (Note 1)

Post-irradiation Cooling Time (years)	Assembly Burnup (MWD/MTU)	Assembly Initial Enrichment (wt.% ^{235}U)
≥16	≤42,000	≥3.09
≥16	≤37,500	≥2.6
≥16	≤30,000	≥2.1

NOTES:

1. Each fuel assembly must only meet one set of limits (i.e., one row)

Table A.9

TROJAN PLANT NON-FUEL HARDWARE AND NEUTRON SOURCES COOLING AND BURNUP LIMITS

Type of Hardware or Neutron Source	Burnup (MWD/MTU)	Post-irradiation Cooling Time (Years)
BPRAs	≤15,998	≥24
TPDs	≤118,674	≥11
RCCAs	≤125,515	≥9
Cf neutron source	≤15,998	≥24
Sb-Be neutron source with 4 source rods, 16 burnable poison rods, and 4 thimble plug rods	≤45,361	≥19
Sb-Be neutron source with 4 source rods, 20 thimble plug rods	≤88,547	≥9

Table A.10

FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND MINIMUM ENRICHMENT MPC-32 PWR FUEL WITH ZIRCALOY CLAD AND WITH NON-ZIRCALOY IN-CORE GRID SPACERS

Post-irradiation cooling time (years)	Assembly burnup (MWD/MTU)	Assembly Initial Enrichment (wt. % U-235)
≥12	≤24,500	≥2.3
≥14	≤29,500	≥2.6
≥16	≤34,500	≥2.9
≥19	≤39,500	≥3.2
≥20	≤42,500	≥3.4

Table A.11

FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND MINIMUM ENRICHMENT MPC-32 PWR FUEL WITH ZIRCALOY CLAD AND WITH ZIRCALOY IN-CORE GRID SPACERS

Post-irradiation cooling time (years)	Assembly burnup (MWD/MTU)	Assembly Initial Enrichment (wt.% U-235)
≥8	≤24,500	≥2.3
≥9	≤29,500	≥2.6
≥12	≤34,500	≥2.9
≥14	≤39,500	≥3.2
≥19	≤44,500	≥3.4

Table A.12

FUEL ASSEMBLY MAXIMUM ENRICHMENT AND MINIMUM BURNUP REQUIREMENTS FOR TRANSPORTATION IN MPC-32

Fuel Assembly Array/Class	**Configuration (Note 2)**	**Maximum Enrichment (wt.% U-235)**	**Minimum Burnup (B) as a Function of Initial Enrichment (E) (Note 1) (GWD/MTU)**
15x15D, E, F, H	A	4.65	$B = (1.6733)*E^3-(18.72)*E^2+(80.5967)*E-88.3$
	B	4.38	$B = (2.175)*E^3-(23.355)*E^2+(94.77)*E-99.95$
	C	4.48	$B = (1.9517)*E^3(21.45)*E^2+(89.1783)*E-94.6$
	D	4.45	$B = (1.93)*E^3-(21.095)*E^2+(87.785)*E-93.06$
17x17A,B,C	A	4.49	$B = (1.08)*E^3-(12.25)*E^2+(60.13)*E-70.86$
	B	4.04	$B = (1.1)*E^3-(11.56)*E^2+(56.6)*E-62.59$
	C	4.28	$B = (1.36)*E^3-(14.83)*E^2+(62.27)*E-72.93$
	D	4.16	$B = (1.4917)*E^3-(16.26)*E^2+(72.9883)*E-79.7$

NOTES:

1. E = Initial enrichment (e.g., for 4.05 wt.%, E = 4.05).
2. See Table A.13.
3. Fuel Assemblies must be cooled 5 years or more.

Table A.13

LOADING CONFIGURATIONS FOR THE MPC-32

CONFIGURATION	ASSEMBLY SPECIFICATIONS
A	• Assemblies that have not been located in any cycle under a control rod bank that was permitted to be inserted during full power operation (per plant operating procedures); or • Assemblies that have been located under a control rod bank that was permitted to be inserted during full power operation (per plant operating procedures), but where it can be demonstrated, based on operating records, that the insertion never exceeded 8 inches from the top of the active length during full power operation.
B	• Of the 32 assemblies in a basket, up to 8 assemblies can be from core locations where they were located under a control rod bank, that was permitted to be inserted more than 8 inches during full power operation. There is no limit on the duration (in terms of burnup) under this bank. • The remaining assemblies in the basket must satisfy the same conditions as specified for configuration A.
C	• Of the 32 assemblies in a basket, up to 8 assemblies can be from core locations where they were located under a control rod bank, that was permitted to be inserted more than 8 inches during full power operation. Location under such a control rod bank is limited to 20 GWD/MTU of the assembly. • The remaining assemblies in the basket must satisfy the same conditions as specified for configuration A.
D	• Of the 32 assemblies in a basket, up to 8 assemblies can be from core locations where they were located under a control rod bank, that was permitted to be inserted more than 8 inches during full power operation. Location under such a control rod bank is limited to 30 GWD/MTU of the assembly. • The remaining assemblies in the basket must satisfy the same conditions as specified for configuration 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 12, dated October 6, 2006.

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
9263	4	71-9263	USA/9263/B(U)-96	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)*

   Source Production and
   Equipment Company, Inc.
   113 Teal Street
   St. Rose, LA 70087

   b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

   Source Production and Equipment Company, Inc.
   Application dated April 22, 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.: SPEC-150

(2) Description

A welded titanium encased, uranium shielded, radiographic exposure device. Primary components consist of an outer titanium shell, internal supports, depleted uranium shield, and a titanium, titanium alloy or zircalloy S-tube. The contents are securely positioned in the S-tube by a source cable lock assembly and source safety plug assembly. The unit resembles a rectangular box approximately 5.4 inches wide, 5.6 inches high and 14.5 inches long. The maximum weight of the package is 53 pounds.

(3) Drawings

The packaging is constructed and assembled in accordance with Source Production and Equipment Company, Inc. Drawing Nos. 15B000, Rev. 6; 15B001-3, Rev. 2; 15B002A, Rev. 5; 15B008, Rev. 4; 15B625, Rev. 1; 19B005, Rev. 0; 19B006, Rev. 0; and 190909, Rev. 0.

(b) Contents

(1) *Type and form of material*

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

(2) Maximum quantity of material per package

150 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".

6. The source shall be secured in the shielded position of the packaging by the source assembly lock, lock cap and safety plug assembly. The safety plug assembly, lock cap and 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 locking ball of the source assembly must engage the locking device. The flexible cable of the source assembly and safety plug assembly must be of sufficient length and diameter to provide positive positioning of the source in the shielded position.

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

Packagings may be marked with Package Identification Number USA/9263/B(U)-85 until April 30, 2006, and must be marked with Package Identification Number USA/9263/B(U)-96 after April 30, 2006.

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 the Operating Procedures in Section 7, of the application, as supplemented, and

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

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

10. Expiration date: June 30, 2010.

REFERENCES

Source Production and Equipment Company, Inc., application dated April 22, 1999.

Supplements dated: May 6, 1999; March 22, June 6, and June 19, 2000; March 28, 2005.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Robert Lewis, Chief
Spent Fuel Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: 26 April 2005

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
9269	5	71-9269	USA/9269/B(U)-96	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)*

QSA Global, Inc.
40 North Avenue
Burlington, MA 01803

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

AEA Technology/QSA Inc., application dated July 23, 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.: 650L

(2) Description

A welded stainless steel encased uranium shielded, Iridium-192 or Selenium-75 source changer. Primary components consist of a steel or stainless steel housing, internal supports, depleted uranium shield and a titanium "U" tube. The tube is crimped in the middle of the "U" to provide a positive stop for the source assembly. Additionally, the Model No. 650L has two source locking assemblies mounted on the top cover plate. These assemblies are used to secure the radioactive source in a shielded position during transport. The unit resembles a rectangular box approximately 10-inches long, 13.25-inches high and 8.25-inches wide. The maximum weight of the package is 90 pounds.

(3) Drawings

The packaging is constructed in accordance with the AEA Technology/QSA Inc., Drawing No. R65006, Rev. H, Sheets 1-4.

(b) Contents

(1) Type and form of material

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

Selenium-75 as sealed sources which meet the requirements of special form radioactive material.

5. (b) Contents (continued)

(2) Maximum quantity of material per package

Ir-192: 240 curies (8.9 TBq) (output)

Se-75: 300 curies (11.1 TBq) (output)

Output curies are determined by measuring the source output at 1 meter and expressing its activity in curies derived from the following: 0.48 R/h-Ci Iridium-192 at 1 meter (Ref: American National Standard N432-1980, "Radiological Safety for the Design and Construction of Apparatus for Gamma Radiography") and 0.2 R/h-Ci Selenium-75 at 1 meter (Ref: U.S. Public Health Service, Bureau of Radiological Health, 1970. Radiological Health Handbook, rev. ed., Rockville, MD).

6. The source shall be secured in the shielded position of the packaging by the source assembly. The source assembly must be fabricated of materials capable of resisting a 1475°F fire environment for one-half hour and maintaining its positioning function. The cable of the source assembly must engage the source hold-down assembly. The flexible cable of the source assembly must be of sufficient length and diameter to provide positive positioning of the source at the crimp of the "U" tube.

7. The 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:

(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 71.17.

10. Revision No. 4 of this certificate may be used until July 31, 2007.

11. Expiration date: November 30, 2010.

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9269	5	71-9269	USA/9269/B(U)-96	3	OF	3

REFERENCES

AEA Technology/QSA Inc. application dated July 23, 1999.

Supplements dated November 19, 1999, October 2 and October 31, 2000, July 8, 2005, and March 1, June 6, June 30 and July 14, 2006.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Christopher M. Regan, Acting Chief
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: August 3, 2006

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE	PAGES
9270	2	71-9270	USA/9270/B(U)F-96	1 OF	20

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.
3930 East Jones Bridge Rd.
Norcross, Georgia 30092

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

NAC International, Inc. application dated April 30, 1997, 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.: UMS Universal Transport Cask Package

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

The UMS is a canister-based system for the storage and transportation of spent nuclear fuel. The transportation component of the UMS system, designated the Universal Transport System, consists of a Universal Transport cask body with a closure lid and energy-absorbing impact limiters loaded with a Transportable Storage Canister (TSC) containing either spent Pressurized Water Reactor (PWR) or Boiling Water Reactor (BWR) nuclear fuel or Maine Yankee site specific contents including Greater than Class C (GTCC) waste.

The NAC-UMS is designed to transport up to 24 intact PWR spent fuel assemblies, 56 intact BWR spent fuel assemblies, GTCC waste, or site specific spent nuclear fuel with associated component hardware. Based on the length of the fuel assemblies, PWR fuels are grouped into three classes (Classes 1 through 3), and BWR fuels are grouped into two classes (Classes 4 and 5). Class 1 and 2 PWR fuel assemblies include non-fuel-bearing inserts (components which include thimble plugs and burnable poison rods installed in the guide tubes). Class 4 and 5 BWR fuel assemblies include the zirconium alloy channels. The loading of site specific fuels that include control component hardware may require the use of a TSC that is longer than if the hardware were excluded. The spent fuel is loaded into a TSC which contains a stainless steel grid work referred to as a basket.

The cask body of the UMS is a right-circular cylinder of multi wall construction which consists of 304 stainless steel inner and outer shells separated by lead gamma radiation shielding which is poured in place. The inner and outer shells are welded to a 304 stainless steel top forging which mates to the cask lid. The inner shell is also welded to a 304 stainless steel bottom forging and the outer shell is welded to the bottom plate. The cask bottom consists of the bottom forging and bottom plate with neutron shield material sandwiched between them. Layers of 4.5 inches thick 304

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

stainless steel ring and two 0.75 inch stainless steel disks are located at the bottom lead annulus between the bottom forging and the outer shell.

Neutron shield material is also placed in an annulus that surrounds the cask outer shell along the length of the cask cavity and is enclosed by a stainless steel shell with top and bottom plates. The neutron shield material is a solid synthetic polymer (NS-4-FR). Twenty-four bonded copper and Type 304 stainless steel fins are located in the radial neutron shield to enhance the heat rejection capability of the cask and to support the neutron shield shell and end plates.

The containment boundary of the UMS consists of the inner shell; bottom forging; top forging; cask lid and lid inner O-ring; vent port cover plate and vent port cover plate inner O-ring; and drain port cover plate and drain port cover plate inner O-ring.

There are five TSCs of different lengths, each to accommodate a different class of PWR or BWR fuel assembly. Each TSC has an outside diameter of about 67 inches and the lengths vary from about 175 to 192 inches long. The TSC assembly consists of a right circular cylindrical shell with a welded bottom plate, a fuel basket, a shield lid, two penetration port covers, and a structural lid. The TSC contains the basket and fuel assemblies or GTCC waste. Spacers are placed below each Class 1, 2, 4 or 5 canisters to locate and support the canister in the cask cavity.

The spacers are free standing structures that are confined in place by the bottom of the canister and the cask bottom inner surface. The spacer(s) ensure that the canister lid is laterally supported by the cask top forging when the cask is horizontal and minimizes axial movement of the canister. Each Class 1 PWR canister is positioned by a stainless steel spacer that is 16.75 inches in length. Each Class 2 PWR canister is positioned by a stainless steel spacer that is 7.65 inches in length. No spacers are used with the Class 3 PWR canister. The Class 4 BWR canister is located by four 1.5 inch aluminum spacers and the Class 5 BWR canister is located with a 1.5 inch aluminum spacer.

The spent fuel basket design uses a series of high strength stainless steel PWR or carbon steel BWR support disks to support the fuel assemblies in stainless steel tubes. The PWR fuel tubes contain neutron absorber on all four sides of the tubes. Three types of fuel tubes are designed to contain the BWR fuel: (1) tubes containing neutron absorber on two sides of the tubes; (2) tubes containing neutron absorber on one side; and (3) tubes containing no neutron absorber. Aluminum heat transfer disks are provided in both the PWR and BWR fuel baskets to enhance thermal performance of the basket. The heat transfer disks are supported by stainless steel tie rods and split spacers that maintain the basket assembly configuration.

The GTCC waste canister is essentially identical to the Class 1 TSC, except for the placement of lifting lugs and the placement of a key way within the canister. The GTCC basket is constructed of Type 304 stainless steel and consists primarily of a cylinder with a 3-inch thick wall closed at the bottom end with a 3-inch thick plate. The cylinder is centered in the GTCC waste canister by 14 Type 304 stainless steel support plates along its length. A 3-inch thick 304 stainless steel separator fixture divides the cylinder into two vertically stacked components, each 77 inches deep with a diameter of 47.8 inches.

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

The package has impact limiters at each end of the cask body. The impact limiters consist of a combination of redwood and balsa wood encased in Type 304 stainless steel. The impact limiters limit the g-loads acting on the cask during a transport drop load condition due to crushing of the redwood and balsa wood. The upper and lower impact limiters are bolted to the cask body by 16 equally spaced attachment rods with nuts.

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

Overall length (with impact limiters, in)	273.3
Overall length (without impact limiters, in)	209.3
Impact Limiter Outside diameter (in)	124.0
Outside diameter (without impact limiters, in)	92.9
Cavity diameter (in)	67.6
Cavity length (in)	192.5
Cask lid thickness (in)	6.5
Bottom thickness (in)	10.3
Inner shell thickness (in)	2.0
Outer shell thickness (in)	2.75
Gamma shield thickness (in)	2.75
Radial neutron shield thickness (in)	4.50

Transportable Storage Canister

Shell thickness (in)	0.625
Shell bottom (in)	1.75
Shield lid thickness (in)	7
Structural lid thickness (in)	3
Outer diameter (in)	67
Internal cavity diameter (in)	65.8
Internal fuel cavity length (in), depending on class	163-180
Overall length (in), depending on class	175-192

Fuel Basket

Basket assembly length (in), depending on class	162-180
Basket assembly diameter (in)	65.5
Number of support disks, depending on class	30-41
Number of heat transfer disks, depending on class	17-33

Total weight (pounds) including cask, basket, impact limiters, fuel, canister with lids, cask lid, and spacers for each fuel class is approximately:

Class 1 (PWR)	251,000
Class 2 (PWR)	252,000
Class 3 (PWR)	249,000
Class 4 (BWR)	256,000
Class 5 (BWR)	255,000

5.(a)(3) Drawings

The package is constructed and assembled in accordance with NAC drawings:

790-209, Rev. 1	790-210, Rev. 1	790-500, Rev. 4	790-501, Rev. 3
790-502, Rev. 7	790-503, Rev. 3	790-504, Rev. 2	790-505, Rev. 2
790-508, Rev. 2	790-509, Rev. 3	790-516, Rev. 3	790-519, Rev. 2
790-520, Rev. 2	790-570, Rev. 4	790-571, Rev. 3	790-572, Rev. 4
790-573, Rev. 7	790-574, Rev. 3	790-575, Rev. 10	790-581, Rev. 9
790-582, Rev. 12	790-583, Rev. 8	790-584, Rev. 19	790-585, Rev. 19
790-587, Rev. 1	790-591, Rev. 6	790-592, Rev. 8	790-593, Rev. 7
790-594, Rev. 2	790-595, Rev. 10	790-605, Rev. 11	790-611, Rev. 6
790-612, Rev. 9	412-501, Rev. 4	412-502, Rev. 6	

5.(b) Contents

(1) Type and Form of Material

The package is designed to transport four types of contents as listed below:

i. 24 intact irradiated PWR fuel assemblies within a TSC;
ii. 56 intact irradiated BWR fuel assemblies within a TSC;
iii. 24 Intact and Damaged PWR assemblies, and Fuel Debris from Maine Yankee within a TSC; or
iv. GTCC waste from Maine Yankee within a TSC.

Each type of package contents is described in detail below.

(i) Intact PWR assemblies

The package is designed to transport 24 irradiated intact PWR fuel assemblies within the TSC. An intact fuel assembly is a spent nuclear fuel assembly without known or suspected cladding defects greater than pinhole leaks or hairline cracks. An empty fuel rod position must be filled with a solid filler rod, fabricated from either zirconium alloy or Type 304 stainless steel, which displaces an equal or greater volume than that occupied by a fuel rod.

The fuel assemblies consist of uranium dioxide pellets with zirconium alloy type cladding. Prior to irradiation, the fuel assemblies must be within the dimensions and specifications of Table 5.(b)(1)(i)-1 below. The combined maximum average burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 5.(b)(1)(i)-2 below. PWR fuel assemblies may include standard inserts such as guide tube thimble plugs and burnable poison rods.

The minimum and maximum allowable assembly average enrichment for loading is 1.9 wt% ^{235}U and 4.2 wt% ^{235}U respectively. Unenriched fuel assemblies are not authorized for loading into the TSC. The maximum burn up of the spent fuel assemblies is 45,000 MWD/GTU and the minimum cool time is 5 years. The maximum weight of UO_2 is 11.53 MTU per cask.

Table 5.(b)(1)(i)-1, Intact PWR Fuel Assembly Characteristics

TSC Class[1]	Vendor[2]	Array	Max. Length (in)	Max. Width (in)	Max. Assembly Weight	Max MTU	No of Fuel Rods	Max Pitch (in)	Min Rod Dia (in)	Min Clad Thick (in)	Max Pellet Dia (in)	Max Active Length (in)	Min Guide Tube Thickness (in)
1	CE	14x14	157.3	8.11	1292	0.404	176[4]	0.590	0.438	0.024	0.380	137.0	0.040
1	Ex/ANF	14x14	160.2	7.76	1271	0.369	179	0.556	0.424	0.030	0.351	142.0	0.034
1	WE	14x14	159.8	7.76	1177	0.362	179	0.556	0.400	0.024	0.345	144.0	0.034
1	WE	14x14	159.8	7.76	1302	0.415	179	0.556	0.422	0.022	0.368	145.2	0.034
1	WE, Ex/ANF	15x15	159.8	8.43	1472	0.465	204	0.563	0.422	0.024	0.366	144.0	0.015
1	Ex/ANF	17x17	159.8	8.43	1348	0.413	264	0.496	0.360	0.025	0.303	144.0	0.016
1	WE	17x17	159.8	8.43	1482	0.468	264	0.496	0.374	0.022	0.323	144.0	0.016
1	WE	17x17	160.1	8.43	1373	0.429	264	0.496	0.360	0.022	0.309	144.0	0.016
2	B&W	15x15	165.7	8.54	1515	0.481	208	0.568	0.430	0.026	0.369	144.0	0.016
2	B&W	17x17	165.8	8.54	1505	0.466	264	0.502	0.379	0.024	0.324	143.0	0.017
3	CE	16x16	178.3	8.10	1430	0.442	236[4]	0.506	0.382	0.023	0.3255	150.0	0.035
1	Ex/ANF[3]	14x14	160.2	7.76	1215	0.375	179	0.556	0.417	0.030	0.351	144.0	0.036
1	CE[3]	15x15	147.5	8.20	1360	0.432	216	0.550	0.418	0.026	0.358	132.0	---
1	Ex/ANF[3]	15x15	148.9	8.25	1339	0.431	216	0.550	0.417	0.030	0.358	131.8	---
1	CE[3]	16x16	158.2	8.10	1300	0.403	236[4]	0.506	0.382	0.023	0.3255	136.7	0.035

[1] Minimum and Maximum initial Enrichments are 1.9 wt% ^{235}U and 4.2 wt% ^{235}U, respectively. All fuel rods are zirconium alloy type clad.

[2] Vendor ID indicates the source of assembly base parameters. Loading of assemblies meeting dimensional limits is not restricted to the vendor(s) listed.

[3] 14x14, 15x15, and 16x16 fuel manufactured for Prairie Island, Palisades and St. Lucie 2 cores, respectively, These are not generic fuel assemblies provided to multiple reactors.

[4] Some fuel rod positions may be occupied by burnable poison rods or solid filler rods.

Table 5.(b)(1)(i)-2, Loading Table for Intact PWR Fuel

Minimum Initial Enrichment wt% ^{235}U (E)	Burnup ≤ 30 GWD/MTU Minimum Cooling Time (years)					30 < Burnup ≤ 35 GWD/MTU Minimum Cooling Time (years)				
	CE 14x14	14x14	15x15	16x16	17x17	CE 14x14	14x14	15x15	16x16	17x17
1.9 ≤ E < 2.1	6	8	8	7	8	8	10	11	9	10
2.1≤ E < 2.3	6	7	8	6	7	7	10	10	8	10
2.3≤ E < 2.5	6	7	7	6	7	7	9	10	8	9
2.5 ≤ E < 2.7	6	7	7	6	7	7	9	9	7	8
2.7≤ E < 2.9	6	7	7	6	7	6	8	9	7	8
2.9≤ E < 3.1	5	7	7	6	6	6	8	8	7	8
3.1≤ E < 3.3	5	6	7	6	6	6	8	8	7	7
3.3 ≤ E < 3.5	5	6	6	6	6	6	7	8	6	7
3.5 ≤ E < 3.7	5	6	6	6	6	6	7	7	6	7
3.7 ≤ E≤ 4.2	5	6	6	6	6	6	7	7	6	7
Minimum Initial Enrichment wt% ^{235}U (E)	**35 < Burnup ≤ 40 GWD/MTU Minimum Cooling Time (years)**					**40 < Burnup ≤ 45 GWD/MTU Minimum Cooling Time (years)**				
	CE 14x14	14x14	15x15	16x16	17x17	CE 14x14	14x14	15x15	16x16	17x17
1.9 ≤ E < 2.1	11	15	15	13	15	18	20	21	20	20
2.1≤ E < 2.3	10	13	14	12	13	15	19	19	18	19
2.3≤ E < 2.5	9	12	13	11	12	14	17	19	17	17
2.5 ≤ E < 2.7	9	12	12	10	11	12	16	18	15	17
2.7≤ E < 2.9	8	11	11	9	11	11	15	18	14	17
2.9≤ E < 3.1	8	10	10	9	10	10	14	18	13	15
3.1≤ E < 3.3	7	10	10	9	10	10	13	17	13	15
3.3 ≤ E < 3.5	7	9	10	8	9	9	12	17	13	15
3.5 ≤ E < 3.7	7	9	10	8	9	8	11	17	12	15
3.7 ≤ E≤ 4.2	7	8	10	8	8	8	11	15	12	14

5.(b)(1)(ii) Intact BWR assemblies

The package is designed to transport 56 irradiated intact BWR fuel assembles within the TSC. An intact fuel assembly is a spent nuclear fuel assembly without known or suspected cladding defects greater than pinhole leaks or hairline cracks.

For BWR fuel, the initial enrichment limit (the enrichment of the as-delivered fresh fuel assembly) represents the maximum peak planar-average enrichment allowed for loading into the TSC. The peak planar-average enrichment is defined to be the maximum planar-average enrichment at any height along the axis of the fuel assembly.

The fuel assemblies consist of uranium dioxide pellets with zirconium alloy type cladding. Prior to irradiation, the fuel assemblies must be within the dimension and specifications of Table 5.(b)(1)(ii)-1 below. The combined maximum average burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 5.(b)(1)(ii)-2.

BWR intact fuel assemblies are authorized with or without channels based on a maximum channel width of 120 mils. The minimum and maximum allowable assembly average enrichment for loading is 1.9 wt% ^{235}U and 4.0 wt% ^{235}U respectively. The maximum burn up of the spent fuel assemblies is 45,000 MWD/GTU and the minimum cool time is six years. The maximum weight of UO_2 is 11.08 MTU per cask. Unenriched fuel assemblies are not authorized for loading into the TSC. BWR fuel assemblies with unenriched axial blankets must have an enriched central fuel region and are acceptable for loading into a TSC if the minimum fuel enrichment of the central region is 1.9 wt% ^{235}U. Any empty fuel position must be filled with a solid filler rod fabricated from either zirconium alloy or Type 304 stainless steel.

Table 5.(b)(1)(ii)-1, Intact BWR Fuel Assembly Characteristics

Canister Class[1,5]	Vendor[4]	Array	Max. Length (in)	Max. Assembly Width (in)[5]	Max. Assembly Weight (lb)[6]	Max MTU	No of Fuel Rods	Max Pitch (in)	Min Rod Dia (in)	Min Clad Thick (in)	Max Pellet Dia (in)	Max Active Length (in)[2]
4	Ex/ANF	7x7	171.3	5.51	620	0.196	48	0.738	0.570	0.036	0.490	144
4	Ex/ANF	8x8	171.3	5.51	563	0.177	63	0.641	0.484	0.036	0.405	145.2
4	Ex/ANF	9x9	171.3	5.51	557	0.173	79	0.572	0.424	0.030	0.357	145.2
4	GE	7x7	171.1	5.51	681	0.199	49	0.738	0.570	0.036	0.488	144.0
4	GE	7x7	171.2	5.51	681	0.198	49	0.738	0.563	0.032	0.487	144.0
4	GE	8x8	171.1	5.51	639	0.173	60	0.640	0.484	0.032	0.410	145.2
4	GE	8x8	171.1	5.51	681	0.179	62	0.640	0.483	0.032	0.410	145.2
4	GE	8x8	171.1	5.51	681	0.186	63	0.640	0.493	0.034	0.416	144.0
5	Ex/ANF	8x8	176.1	5.51	588	0180	62	0.641	0.484	0.036	0.405	150.0
5	Ex/ANF	9x9	176.1	5.51	576	0.167	74[3]	0.572	0.424	0.030	0.357	150.0
5[5]	Ex/ANF	9x9	176.1	5.51	576	0.178	79[3]	0.572	0.424	0.030	0.357	150.0
5	GE	7x7	175.9	5.51	683	0.198	49	0.738	0.563	0.032	0.487	144.0
5	GE	8x8	176.1	5.51	665	0.179	60	0.640	0.484	0.032	0.410	150.0
5	GE	8x8	175.9	5.51	681	0.185	62	0.640	0.483	0.032	0.410	150.0
5	GE	8x8	175.9	5.51	681	0.188	63	0.640	0.493	0.034	0.416	146.0
5	GE	9x9	176.1	5.51	646	0.186	74[3]	0.566	0.441	0.028	0.376	150.0
5	GE	9x9	176.1	5.51	646	0.198	79[3]	0.566	0.441	0.028	0.376	150.0

[1] Maximum Peak Planar Average Enrichment 4.0 wt%^{235}U. Minimum enrichment is 1.9 wt%^{235}U. All fuel rods are zirconium alloy type clad.

[2] 150 inch active fuel length assemblies contain 6 inch natural uranium blankets on top and bottom.

[3] Shortened active fuel length in some rods.

[4] Vendor ID indicates the source of assembly base parameters. Loading of assemblies meeting dimensional limits is not restricted to the vendor(s) listed.

[5] Assembly width including channel. Unchanneled or channeled may be loaded based on a maximum channel thickness of 120 mils.

[6] Exxon/ANF assembly weights are listed without channel.

Table 5.(b)(1)(ii)-2, Loading Table for Intact BWR Fuel

Minimum Initial Enrichment wt% ^{235}U (E)	Burnup ≤ 30 GWD/MTU Minimum Cooling Time (years)			30 < Burnup ≤ 35 GWD/MTU Minimum Cooling Time (years)		
	9x9	8x8	7x7	9x9	8x8	7x7
1.9 ≤ E < 2.1	8	8	8	14	13	15
2.1 ≤ E < 2.3	7	7	8	12	12	13
2.3 ≤ E < 2.5	7	7	7	11	10	11
2.5 ≤ E < 2.7	7	6	7	9	9	10
2.7 ≤ E < 2.9	6	6	6	9	8	9
2.9 ≤ E < 3.1	6	6	6	8	8	8
3.1 ≤ E < 3.3	6	6	6	7	7	8
3.3 ≤ E < 3.5	6	6	6	7	7	7
3.5 ≤ E < 3.7	6	6	6	7	7	7
3.7 ≤ E ≤ 4.0	6	6	6	7	7	7
Minimum Initial Enrichment wt% ^{235}U (E)	35 < Burnup ≤ 40 GWD/MTU Minimum Cooling Time (years)			40 < Burnup ≤ 45 GWD/MTU Minimum Cooling Time (years)		
	9x9	8x8	7x7	9x9	8x8	7x7
1.9 ≤ E < 2.1	24	23	25	34	33	35
2.1 ≤ E < 2.3	21	20	22	31	30	32
2.3 ≤ E < 2.5	19	18	20	29	28	29
2.5 ≤ E < 2.7	17	16	17	26	25	27
2.7 ≤ E < 2.9	14	14	15	24	23	24
2.9 ≤ E < 3.1	13	12	13	21	20	22
3.1 ≤ E < 3.3	11	11	12	19	18	20
3.3 ≤ E < 3.5	10	10	11	17	16	18
3.5 ≤ E < 3.7	10	9	10	15	14	16
3.7 ≤ E ≤ 4.0	10	9	10	14	13	15

5.(b)(1)(iii) Intact and Damaged PWR assemblies, and Fuel Debris from Maine Yankee

The package is designed to transport 24 irradiated intact or damaged PWR fuel assemblies, canistered fuel debris, and GTCC waste within the TSC from the Maine Yankee Reactor. The standard Maine Yankee fuel assembly is the intact PWR CE 14x14 (see section 5.(b)(1)(i)).

In the course of reactor operations, some of the 14x14 assemblies were modified to change the standard configuration. These modifications included a) the removal of fuel rods without replacement; b) the replacement of removed fuel rods or burnable poison rods with rods of a different material, such as stainless steel, or with fuel rods of a different enrichment; and c) the insertion of control elements, or instruments or plug thimbles, in guide tube positions. In addition to the modified fuel assemblies, there are fuel assemblies that were designed with variable enrichment and axial blankets. These fuel assemblies are not modified, but differ from the cask design basis fuel assemblies.

Stainless steel spacers may be used in canisters to axially position PWR intact fuel assemblies that are shorter than the available cavity length. The minimum length of the PWR intact fuel assembly internal structure and bottom end fitting and/or spacers will ensure that the minimum distance to the fuel region for the base of the canister is 3.2 inches.

Unenriched fuel assemblies are not authorized for loading.

The following are the allowable Maine Yankee site specific contents:

5.(b)(1)(iii)(A) Maine Yankee's site specific contents not requiring preferential loading patterns:

(1) Standard Irradiated CE 14 x 14 intact PWR fuel assemblies meeting the PWR fuel assembly characteristics in Table 5.(b)(1)(i)-1. The maximum fuel assembly weight, including other associated hardware is 1,515 pounds. The combined maximum average burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 5.(b)(1)(iii)(A)-1.

(2) Irradiated Maine Yankee CE 14 x 14 PWR intact fuel assemblies may contain inserted control element assemblies (CEA), in-core instrument (ICI) thimbles or CEA plugs. CEAs or CEA plugs may not be inserted in damaged fuel assemblies, consolidated fuel assemblies or assemblies with irradiated stainless steel replacement rods. Fuel assemblies with a CEA or CEA plug inserted must be loaded in a Class 2 canister and cannot be loaded in a Class 1 canister. Fuel assemblies without an inserted CEA or CEA plug, including those with inserted ICI Thimbles, must be loaded in a Class 1 canister. The combined maximum average burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 5.(b)(1)(iii)(A)-1 except for those assemblies containing ICI thimbles which must meet the specifications of Table 5.(b)(1)(iii)(A)-2.

(3) PWR intact fuel assemblies with fuel rods replaced with stainless steel or zirconium alloy rods or with Uranium oxide rods nominally enriched up to 1.95 wt%. The combined maximum average burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 5.(b)(1)(iii)(A)-3.

(4) PWR intact fuel assemblies with fuel rods having variable enrichments with a maximum rod enrichment up to 4.21 wt% ^{235}U and that also have a maximum planar average enrichment up to 3.99 wt% ^{235}U. The combined maximum average burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 5.(b)(1)(iii)(A)-1.

(5) PWR intact fuel assemblies with annular axial end blanket enrichments up to 2.6 wt% ^{235}U. The combined maximum average burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 5.(b)(1)(iii)(A)-1.

(6) PWR intact fuel assemblies with burnable poison rods or solid filler rods may occupy up to 16 of 176 fuel rod positions. The combined maximum average burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 5.(b)(1)(iii)(A)-1.

(7) PWR intact fuel assemblies with one or more grid spacers missing or damaged such that the unsupported length of the fuel rods does not exceed 60 inches or with end fitting damage, including damaged or missing hold-down springs, as long as the assembly can be handled safely by normal means. The combined maximum average burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 5.(b)(1)(iii)(A)-1.

5.(b)(1)(iii)(B) Maine Yankee site-specific allowable contents requiring preferential loading based on shielding, criticality, or thermal constraints (Maine Yankee CE 14 x 14 intact PWR fuel assemblies). A PWR basket fuel diagram can be found on Figure 5.(b)(1)(iii)(B)-1.

(1) Maine Yankee CE 14 x 14 PWR intact fuel assemblies with a burn up between 45,000 and 50,000 MWD/MTU meeting the following requirements for verification of the oxide layer thickness and high burn up fuel requiring preferential loading in the peripheral PWR fuel basket positions:

A verification program is required to determine the oxide layer thickness on high burn up fuel by measurement or by statistical analysis. A fuel assembly having a burn up between 45,000 MWD/MTU and 50,000 MWD/MTU is classified as high burn up. The verification program shall be capable of classifying high burn up fuel as INTACT FUEL or DAMAGED FUEL based on the following criteria:

I. A HIGH BURN UP FUEL assembly may be stored as INTACT FUEL provided that no more than 1% of the fuel rods in the assembly have a peak cladding oxide thickness greater than 80 microns, and that no more than 3% of the fuel rods in the assembly have a peak oxide layer thickness greater than 70 microns, and that the fuel assembly is otherwise INTACT FUEL.

II. A HIGH BURN UP FUEL assembly not meeting the cladding oxide thickness criteria for INTACT FUEL or that has an oxide layer that is detached or spalled from the cladding is classified as DAMAGED FUEL.

The combined maximum average burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 5.(b)(1)(iii)(A)-1.

(2) PWR intact fuel assemblies with up to 176 fuel rods missing from the fuel assembly lattice. The combined maximum average burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 5.(b)(1)(iii)(A)-1. These assemblies must be placed in a corner loading position in the PWR fuel basket.

(3) PWR intact fuel assemblies with burnable poison rods replaced by hollow zirconium alloy rods. The combined maximum average burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 5.(b)(1)(iii)(A)-1. These assemblies must be placed in a corner PWR fuel basket loading position.

(4) Intact fuel assemblies with a start-up source in a center guide tube. The assembly must be loaded in a basket corner position and must be loaded in a Class 1 canister. Only one start-up source may be loaded in any fuel assembly or any canister. The combined maximum average burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 5.(b)(1)(iii)(A)-1. These assemblies must be placed in a corner PWR fuel basket loading position.

(5) PWR intact fuel assemblies with CEA ends (fingertips) and/or an ICI segment inserted in corner guide tube positions. The assembly must also have a CEA plug installed. The assembly must be loaded in a PWR fuel basket corner position and must be loaded in a Class 2 canister. The combined maximum average burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 5.(b)(1)(iii)(A)-1. CEA fingertips are not considered as CEAs for determination of minimum cool times.

5.(b)(1)(iii)(C) Maine Yankee CE 14 x 14 PWR fuel enclosed in a Maine Yankee Fuel Can (MYFC).

All Maine Yankee CE 14 x 14 PWR fuel enclosed in an MYFC must be loaded in a Class 1 fuel canister in a corner position of the PWR fuel basket. Up to 4 MYFC may be loaded into a TSC. Intact Maine Yankee CE 14 x 14 PWR fuel may be loaded into a MYFC. The contents that must be loaded in the MYFC are:

(1) PWR fuel assemblies with up to two intact or damaged fuel rods inserted in each fuel assembly guide tube or with up to two burnable poison rods inserted in each guide tube. The rods inserted in the guide tubes cannot be from a different fuel assembly. The maximum number of rods in the fuel assembly (fuel rods plus inserted rods, including burnable poison rods) is 176. The combined maximum average burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 5.(b)(1)(iii)(A)-1 for intact fuel rods and Table 5.(b)(1)(iii)(A)-4 for damaged fuel rods.

(2) A damaged fuel assembly with up to 100% of the fuel rods classified as damaged and/or damaged or missing assembly hardware components. A damaged fuel assembly cannot have an inserted CEA or other non-fuel component. The combined maximum average burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 5.(b)(1)(iii)(A)-4 for damaged fuel rods.

(3) Individual intact or damaged fuel rods in a rod type structure, which may be a guide tube, to maintain configuration control. The combined maximum average

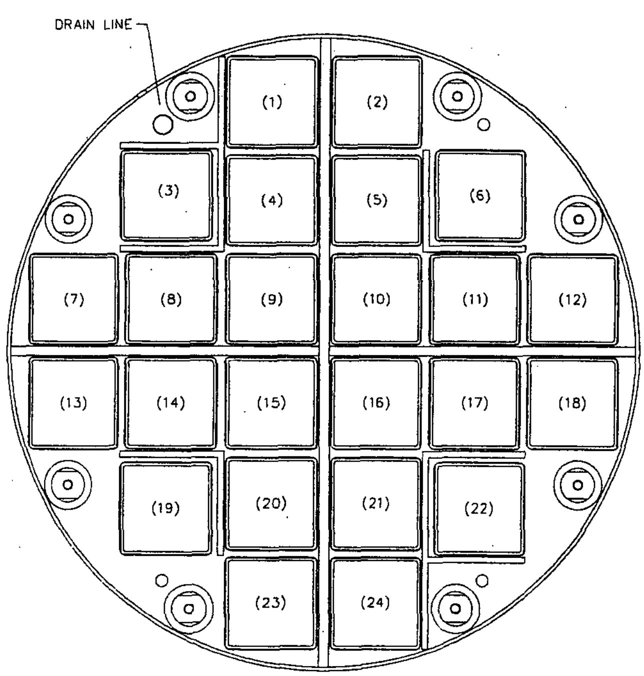
burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 5.(b)(1)(iii)(A)-1 for intact fuel rods and Table 5.(b)(1)(iii)(A)-4 for damaged fuel rods.

(4) Fuel debris consisting of fuel rods with exposed fuel pellets or individual intact or partial fuel pellets not contained in fuel rods. The combined maximum average burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 5.(b)(1)(iii)(A)-4 for damaged fuel rods.

(5) Consolidated Fuel lattice and structure with a 17 x 17 array formed by grids and top and bottom end fittings connected by four solid stainless steel rods. Maximum contents are 289 fuel rods having a total lattice weight less than or equal to 2,100 pounds. A consolidated fuel lattice cannot have an inserted CEA or other non-fuel component. Only one consolidated fuel lattice may be stored in any TSC. Fuel must be cooled a minimum of 24 years.

(6) High burn up fuel assemblies not meeting the oxide layer thickness criteria previously defined in Section 5.(b)(1)(iii)(B)(1). The combined maximum average burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 5.(b)(1)(iii)(A)-4 for damaged fuel rods.

PWR Basket Fuel Loading Position Diagram, Figure 5.(b)(1)(iii)(B)-1



Basket corner positions are positions 3, 6, 19, and 22. Corner positions are also periphery positions.

2. Basket periphery positions are positions 1, 2, 3, 6, 7, 12, 13, 18, 19, 22, 23, and 24. Periphery positions include the corner positions.

Table 5.(b)(1)(iii)(A)-1, Loading Table for Maine Yankee CE 14x14 Fuel with and without CEA Cooled to Indicated Time						
Burnup 30 GWD/MTU		Minimum Cool Time (Years) for				
Enrichment	No CEA (Class 1)	No CEA (Class 2)	5 Yr CEA	10 Yr CEA	15 Yr. CEA	20 Yr. CEA
1.9 ≤ E < 2.1	6	6	7	6	6	6
2.1≤ E < 2.3	6	6	7	6	6	6
2.3≤ E < 2.5	6	6	6	6	6	6
2.5 ≤ E < 2.7	6	6	6	6	6	6
2.7≤ E < 2.9	6	6	6	6	6	6
2.9≤ E < 3.1	5	6	6	6	6	6
3.1≤ E < 3.3	5	5	6	6	6	5
3.3 ≤ E < 3.5	5	5	6	6	5	5
3.5 ≤ E < 3.7	5	5	6	5	5	5
3.7 ≤ E≤ 4.2	5	6	5	5	5	5

Loading Table for Maine Yankee CE 14x14 Fuel with and without CEA Cooled to Indicated Time						
Burnup 35 GWD/MTU		Minimum Cool Time (Years) for				
Enrichment	No CEA (Class 1)	No CEA (Class 2)	5 Yr CEA	10 Yr CEA	15 Yr. CEA	20 Yr. CEA
1.9 ≤ E < 2.1	8	8	9	8	8	8
2.1≤ E < 2.3	7	7	9	8	8	8
2.3≤ E < 2.5	7	7	8	7	7	7
2.5 ≤ E < 2.7	7	7	8	7	7	7
2.7≤ E < 2.9	6	7	7	7	7	7
2.9≤ E < 3.1	6	6	7	7	6	6
3.1≤ E < 3.3	6	6	7	6	6	6
3.3 ≤ E < 3.5	6	6	7	6	6	6
3.5 ≤ E < 3.7	6	6	6	6	6	6
3.7 ≤ E≤ 4.2	6	6	6	6	6	6

Table 5.(b)(1)(iii)(A)-1, continued , Loading Table for Maine Yankee CE 14x14 Fuel with and without CEA Cooled to Indicated Time

Burnup 40 GWD/MTU		Minimum Cool Time (Years) for				
Enrichment	No CEA (Class 1)	No CEA (Class 2)	5 Yr CEA	10 Yr CEA	15 Yr. CEA	20 Yr. CEA
1.9 ≤ E < 2.1	11	12	14	13	12	12
2.1≤ E < 2.3	10	10	13	11	11	11
2.3≤ E < 2.5	9	9	12	10	10	10
2.5 ≤ E < 2.7	9	9	10	9	9	9
2.7≤ E < 2.9	8	8	10	9	8	8
2.9≤ E < 3.1	8	8	9	8	8	8
3.1≤ E < 3.3	7	7	8	8	8	8
3.3 ≤ E < 3.5	7	7	8	7	7	7
3.5 ≤ E < 3.7	7	7	8	7	7	7
3.7 ≤ E≤ 4.2	7	7	7	7	7	7

Loading Table for Maine Yankee CE 14x14 Fuel with and without CEA Cooled to Indicated Time

Burnup 45 GWD/MTU		Minimum Cool Time (Years) for				
Enrichment	No CEA (Class 1)	No CEA (Class 2)	5 Yr CEA	10 Yr CEA	15 Yr. CEA	20 Yr. CEA
1.9 ≤ E < 2.1	18	18	21	19	18	18
2.1≤ E < 2.3	15	16	19	17	17	16
2.3≤ E < 2.5	14	14	18	16	15	15
2.5 ≤ E < 2.7	12	13	16	14	14	13
2.7≤ E < 2.9	11	12	14	13	12	12
2.9≤ E < 3.1	10	11	13	12	11	11
3.1≤ E < 3.3	10	10	12	11	10	10
3.3 ≤ E < 3.5	9	9	11	10	10	10
3.5 ≤ E < 3.7	9	9	10	10	10	10
3.7 ≤ E≤ 4.2	9	9	10	10	10	10

Table 5.(b)(1)(iii)(A)-1, continued, Loading Table for Maine Yankee CE 14x14 Fuel with and without CEA Cooled to Indicated Time

Burnup 50 GWD/MTU		Minimum Cool Time (Years) for				
Enrichment	No CEA (Class 1)	No CEA (Class 2)	5 Yr CEA	10 Yr CEA	15 Yr. CEA	20 Yr. CEA
1.9 ≤ E < 2.1	27	27	29	27	27	27
2.1≤ E < 2.3	24	24	27	25	24	24
2.3≤ E < 2.5	22	22	25	23	22	22
2.5 ≤ E < 2.7	19	19	23	21	20	20
2.7≤ E < 2.9	17	17	21	19	18	18
2.9≤ E < 3.1	15	16	19	18	18	18
3.1≤ E < 3.3	15	15	18	17	17	17
3.3 ≤ E < 3.5	15	15	17	17	17	17
3.5 ≤ E < 3.7	14	14	15	15	15	15
3.7 ≤ E≤ 4.2	14	14	15	15	15	15

Table 5.(b)(1)(iii)(A)-2, Loading Table (Years) for Maine Yankee CE 14x14 fuel containing ICI Thimbles

Minimum Initial Enrichment wt% ^{235}U (E)	Burnup ≤ 30 GWD/MTU	30 < Burnup ≤ 35 GWD/MTU	35 < Burnup ≤ 40 GWD/MTU	40 < Burnup ≤ 45 GWD/MTU	45 < Burnup ≤ 50 GWD/MTU
1.9 ≤ E < 2.1	6	8	11	18	27
2.1≤ E < 2.3	6	7	10	16	24
2.3≤ E < 2.5	6	7	9	14	22
2.5 ≤ E < 2.7	6	7	9	13	19
2.7≤ E < 2.9	6	6	8	11	17
2.9≤ E < 3.1	5	6	8	10	15
3.1≤ E < 3.3	5	6	7	10	15
3.3 ≤ E < 3.5	5	6	7	9	15
3.5 ≤ E < 3.7	5	6	7	9	14
3.7 ≤ E≤ 4.2	5	6	7	9	14

Table 5.(b)(1)(iii)(A)-3, Required Cool Time for Maine Yankee Fuel Assemblies with Activated Stainless Steel Replacement Rods

Assy Number	Burnup (GWD/MTU)	Enrichment (wt %)	SSR Source (g/s/assy)	Cool Time (years)	Earliest Transportable
N420	45	3.3	2.1602E+13	10	Jan 2001
N842	35	3.3	3.1396E+12	6	Jan 2001
N868	40	3.3	5.2444E+12	7	Jan 2001
R032	45	3.5	1.4550E+13	9	Jan 2005
R439	50	3.5	1.3998E+13	14	Jan 2010
R444	50	3.5	5.5993E+13	19	Jan 2015

Table 5.(b)(1)(iii)(A)-4, Cool time (years) for Maine Yankee CE 14x14 damaged fuel

Minimum Initial Enrichment wt% ^{235}U (E)	Burnup ≤ 30 GWD/MTU	30 < Burnup ≤ 35 GWD/MTU	35 < Burnup ≤ 40 GWD/MTU	40 < Burnup ≤ 45 GWD/MTU	45 < Burnup ≤ 50 GWD/MTU
1.9 ≤ E < 2.1	7	11	19	28	37
2.1≤ E < 2.3	6	9	16	26	34
2.3≤ E < 2.5	6	8	14	23	32
2.5 ≤ E < 2.7	6	8	12	21	30
2.7≤ E < 2.9	6	7	11	19	27
2.9≤ E < 3.1	6	7	10	17	25
3.1≤ E < 3.3	5	7	9	15	23
3.3 ≤ E < 3.5	5	6	8	13	21
3.5 ≤ E < 3.7	5	6	8	12	19
3.7 ≤ E≤ 4.2	5	6	7	11	17

5.(b)(1)(iv) Greater Than Class C Waste from Maine Yankee

The package is designed to transport Maine Yankee Greater Than Class C Waste within a TSC. Maine Yankee GTCC waste consists of solid, irradiated, and contaminated hardware and solid, particulate debris or filter media, provided the quantity of fissile material does not exceed a Type A quantity and does not exceed the mass limits of 10 CFR 71.15. The maximum curie inventory shall not exceed the values shown in Table 5.(b)(1)(iv)-1.

Table 5.(b)(1)(iv)-1, Maine Yankee GTCC Curie Inventory Limits per TSC

Radionuclide	Curie Inventory (Ci)/ TSC
H-3	3.00E+02
C-14	1.50E+02
Mn-54	3.50E+02
Fe-55	2.00E+05
Co-58	1.00E+01
Co-60	2.90E+05
Ni-59	8.20E+02
Ni-63	9.00E+04
Nb-94	1.00E+01
Tc-99	1.00E+01

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

The maximum weight of the contents shall not exceed 77,500 pounds.

(i) For the contents described in 5.(b)(1)(i) and 5.(b)(1)(iii): 24 PWR fuel assemblies, including standard inserts such as burnable poison rods or guides or guide tube thimble plugs, with a maximum weight of 38,500 pounds and a maximum decay heat limit per package not to exceed the values in Table 5.(b)(2)-1. The individual PWR assembly decay heat is limited to 0.83 kW.

Table 5.(b)(2)-1, PWR Decay Heat Limits				
Cool Time (Years)	PWR Decay Heat Limit (kW) Burnup (MWD/MTU)			
	35,000	40,000	45,000	50,000[1]
5	20.0	20.0	19.9	19.3
6	19.5	19.3	19.2	18.7
7	17.8	17.8	17.7	17.2
10	17.4	17.3	17.2	16.8
15	16.8	16.8	16.7	16.5

[1]Maine Yankee PWR fuel assemblies

(ii) For the contents described in 5.(b)(1)(ii); 56 BWR assemblies with a maximum weight of 39,000 pounds and a maximum decay heat limit per package of 16 kW. The individual BWR assembly decay heat is limited to 0.29 kW.

(iii) For the contents described in 5.(b)(1)(iv): GTCC waste with a maximum weight per package of 20,000 pounds in total or 10,000 pounds per compartment. The maximum decay heat for the GTCC is 4.5 kW per package.

5.(c) Criticality Safety Index 0.0

6. The package must be transported as exclusive use in a closed transport vehicle as per 10 CFR 71.47(b).

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, 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.

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

Expiration date: October 31, 2007.

REFERENCES

NAC International, Inc., Application dated April 30, 1997.

NAC International, Inc., Supplements dated June 18, 1999, May 31, June 29, August 8, and September 20, 2000, February 28, March 14, March 31, June 1, and November 16, 2001, and January 31, March 13, August 12, September 27, and October 21, 2002, March 31, and September 28, 2004, and May 4, and June 6, 2005.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Robert J. Lewis, Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: 11 Aug 2005

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
9272	6	71-9272	USA/9272/AF-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)*

Westinghouse Electric Company LLC
P.O. Box 355
Pittsburgh, PA 15230-0355

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Westinghouse Electric Company LLC application dated January 4, 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 No.: CE-B1

(2) Description

A shipping container for unirradiated fuel assemblies. The package consists 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 11-1/4 inches by 18-1/8 inches by 182 inches long. There are two channel sections within the inner container, and each channel section holds one BWR fuel assembly. The inner container is equipped with a lid and an end cap that are closed by 18 bolts and fastening lugs. The overall dimensions of the wooden outer container are approximately 33-1/2 inches by 34-3/4 inches by 208-1/2 inches long. The cushioning material between the inner and outer containers is phenolic impregnated honeycomb and ethafoam. The inner container may be positioned on a series of vibration dampers mounted on the inside bottom of the wooden outer container.

The maximum weight of the package, including contents, is 2,964 pounds.

(3) Drawings

The packaging is constructed and assembled in accordance with Combustion Engineering Drawing Nos.:

L-9272-01, Sheets 1 and 2, Rev. 1, and
L-9272-02, Rev. 1.

5.(b) Contents

(1) Type and form of material

The package is designed to hold two unirradiated BWR fuel assemblies, comprised of UO_2 fuel rods in a 10 x 10 square array. The fuel cross-sectional area is 25 square inches. Each assembly is made up of 96 full-length fuel rods having a maximum active fuel length of 150 inches. The fuel pellet diameter is 0.819 ± 0.002 cm, encapsulated in 0.063 cm zirconium alloy cladding. There is a 0.0085 cm gap between the pellets and the cladding. The maximum U-235 enrichment of any fuel rod is 5.0 weight percent. Each assembly contains water holes in the four center rod positions of the assembly. Three different fuel package loadings have the following specifications:

(i) Maximum average U-235 enrichment is 4.0 weight percent within any axial zone of the assembly; Maximum U-235 content is 3.25 weight percent of any gadolinia-urania rod or axial zone of any gadolinia-urania fuel rod; Maximum number of fuel rods per assembly containing 5.0 weight percent U-235 enriched pellets is 36; Maximum U-235 enrichment is 4.0 weight percent for all edge rods, and 3.5 weight percent for all corner rods; Each assembly must include at least eight fuel rods with a minimum gadolinia content of 2.5 weight percent in all axial regions with enriched pellets. The eight gadolinia rods are arranged with two rods in each quadrant of the fuel assembly. The two gadolinia rods within each quadrant must be symmetric about the geometric diagonal of the fuel assembly, and must not be in an edge or corner rod location. Other fuel rods containing gadolinia may be present.

(ii) Maximum average U-235 enrichment is 4.725 weight percent within any axial zone of the assembly; Maximum U-235 content is 4.2 weight percent of any gadolinia-urania rod or axial zone of any gadolinia-urania fuel rod; Maximum number of fuel rods per assembly containing 5.0 weight percent U-235 enriched pellets is 52; Maximum U-235 enrichment is 4.5 weight percent for all edge rods, and 4.0 weight percent for all corner rods; Each assembly must include at least eight fuel rods with a minimum gadolinia content of 5.3 weight percent in all axial regions with enriched pellets. The eight gadolinia rods are arranged with two rods in each quadrant of the fuel assembly. The two gadolinia rods within each quadrant must be symmetric about the geometric diagonal of the fuel assembly, and must not be in an edge or corner rod location. Other fuel rods containing gadolinia may be present.

5.(b)(1) Type and form of material (continued)

(iii) Maximum average U-235 enrichment is 4.858 weight percent within any axial zone of the assembly; Maximum U-235 content is 4.2 weight percent of any gadolinia-urania rod or axial zone of any gadolinia-urania fuel rod; Maximum number of fuel rods per assembly containing 5.0 weight percent U-235 enriched pellets is 80; Maximum U-235 enrichment is 4.0 weight percent for all corner rods; Each assembly must include at least twelve fuel rods with a minimum gadolinia content of 2.43 weight percent in all axial regions with enriched pellets. The twelve gadolinia rods are arranged with three rods in each quadrant of the fuel assembly. The three gadolinia rods within each quadrant must be symmetric about the geometric diagonal of the fuel assembly, and must not be in an edge or corner rod location. Other fuel rods containing gadolinia may be present.

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

Two fuel assemblies. The total weight of contents not to exceed 1,184 pounds.

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

Minimum transport index to be shown on
label for nuclear criticality control: 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 inserts may be positioned between rods within the fuel assemblies. The quantity of polyethylene must not exceed 18.33 g polyethylene per centimeter length of the fuel assembly, and must not exceed a total of 6.99 kg per fuel assembly. The polyethylene may be borated.

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 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.

9. Only ABB/Combustion Engineering packagings with Serial Nos. CE-B1/001 through CE-B1/039, inclusive, are authorized for use.

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, 2007.

REFERENCES

Westinghouse Electric Company LLC application dated January 4, 2002.

Supplement dated: January 9, 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 January 25, 2002

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9274	7	71-9274	USA/9274/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)*

Westinghouse Electric Company, LLC
P.O. Box 355
Pittsburgh, PA 15230-0355

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Westinghouse Electric Company application dated May 15, 2003, 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.: ABB-2901

(2) Description

A shipping container for low-enriched uranium oxide pellets, composed of an inner container, surrounded by insulating material, and an outer drum. The inner container is 10.75 ± 1/4 inches square and approximately 30 inches long, 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 by asbestos or ceramic sheet, plywood, hardboard, and insulating material. The drum has a 16-gauge closure lid. The drum lid is closed with a 12-gauge locking ring with drop forged lugs and a 5/8-inch diameter bolt. In addition to the locking ring, three lid clamps are installed to secure the drum lid. The drum has approximate dimensions of 22.5-inch ID by 36-inch height. The uranium oxide pellets are packaged in boxes positioned within a steel insert. The maximum gross weight of the package is 660 pounds.

(3) Drawings

The packaging is constructed and assembled in accordance with Westinghouse Electric Company, LLC, Drawing Nos.

10004E01, Rev. 2;
10004E02, Sheets 1 and 2, Rev. 2; and
10004E03, Rev. 2.

5. (b) Contents

(1) Type and form of material

Sintered uranium oxide pellets enriched to a maximum 5.0 w/o in the U-235 isotope. The maximum pellet diameter is 0.969 cm, and the minimum pellet diameter is 0.818 cm.

(2) Maximum quantity of material per package

227 pounds of pellets, with the U-235 content not to exceed 4.54 kg. The pellets must be packaged on corrugated stainless steel trays, within shipping container boxes and a shipping container insert in accordance with ABB Combustion Engineering Nuclear Systems Drawing Nos. L-9274-02, Sheets 1 and 2, Rev. 0, and L-9274-03, Rev. 0.

Maximum weight of contents within the inner container is 427 pounds, including radioactive material, secondary containers, and other packaging material.

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

Minimum transport index to be shown on label for nuclear criticality control: 0.5

6. Corrugated stainless steel trays must be positioned between each layer of pellets, and on the top and bottom of the pellet stack. Spacers must be inserted in partially filled pellet shipping boxes to provide a snug fit.

7. The package may also contain stainless steel pellets, depleted uranium pellets, and neutron poisons such as gadolinia, erbium, and boron carbide.

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

(a) Prior to each shipment the insert (containment vessel) gasket shall be inspected. This gasket shall be replaced if inspection shows any defects.

(b) The package must be prepared for shipment and operated in accordance with the Operating Procedures of Chapter 7 and the 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: September 30, 2007.

REFERENCES

Westinghouse Electric Company application dated May 15, 2003.

Supplement dated November 21, 2003.

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: January 29, 2004

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9276	2	71-9276	USA/9276/B(U)F-85	1	OF	26

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)*

   BNFL Fuel Solutions
   2105 S. Bascom Ave., Suite 160
   Campbell, CA 95008

   b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

   BNFL Fuel Solutions application dated April 20, 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. FuelSolutions™ TS125 Transportation Package

(2) Description

The FuelSolutions™ TS125 Transportation Package consists of a TS125 Transportation Cask and impact limiters, together with a FuelSolutions™ W21 or W74 canister and its payload. The FuelSolutions™ canister and its payload are contained inside the TS125 Transportation Cask cavity. The TS125 Transportation Cask cavity is sized to accommodate one FuelSolutions™ long canister, or alternatively, one FuelSolutions™ short canister with a cask cavity spacer. The approximate dimensions and weights of the package are as follows:

Package Length: .... 342.4 inches
Package Outside Diameter: .... 143.5 inches
Cask Length (w/o impact limiters): .... 210.4 inches
Cask Outside Diameter (w/o impact limiters): .... 94.2 inches
Cask Cavity Length: .... 193.0 inches
Cask Cavity Diameter (section at rails): .... 66.88 inches
Canister Outside Diameter: .... 66.0 inches
Maximum Long Canister Length: .... 192.25 inches
Maximum Short Canister Length: .... 182.25 inches
Cask Cavity Spacer Length: .... 10.0 inches
Max. Package Weight: .... 285,000.0 pounds
Max. Cask Payload Weight (incl. canister and cavity spacer): ...... 85,000.0 pounds

The TS125 Transportation Cask body is an assembly composed of stainless steel components of an inner shell, an outer shell, a top ring forging, a closure lid with a seal test port and a cavity vent port, a bottom plate forging, and a cavity drain port. The inner and outer shells are welded to the bottom plate forging and the top ring forging. The cask body also includes an annular lead gamma shield; an annular neutron shield with cask tie-down rings, support angles, and jacket; a bottom end neutron shield with a support ring and jacket; a longitudinal shear block; and lifting trunnion mounting bosses. The inner and outer shells form the annular cavity for the lead gamma shield. The outer shell and the neutron shield jacket form the annular cavity for the solid neutron shield. The neutron shield support angles facilitate heat rejection through the solid neutron shielding material to the outer surface of the cask body. The cask closure lid includes a thick recessed plate with two concentric "Helicoflex" silver-jacketed metallic o-ring seals, the cavity vent port, and the seal test port. The closure lid is secured to the cask body during transport with 60 – 2 inch diameter closure bolts. The vent and drain ports are closed by a plug assembly to maintain containment integrity during transportation.

The Transportation Cask's containment boundary consists of: the inner cylindrical shell, the bottom plate forging (which forms the bottom closure of the cask), the top ring forging and sealing surfaces, the closure lid and sealing surfaces, the welds associated with the above components, the closure bolts, the innermost closure lid o-ring seal, the cavity vent port seal gland and o-ring seal, and the cavity drain port seal gland and o-ring seal. The package is designed to be "leaktight" as defined by ANSI N14.5 (leakage rate less than or equal to 1×10^{-7} ref-cm^3/s. The structural components of the Transportation Cask are made of high-strength austenitic stainless steel. The gamma shielding is made of lead and is completely enclosed within the annular region between the inner and outer steel shells. The neutron shielding is solid hydrogenous material that is completely enclosed within the annular region between the cask outer shell and neutron shield jacket with tie-down rings at each end.

The FuelSolutions™ TS125 Transportation Cask has identical energy-absorbing impact limiters at both ends. Each impact limiter assembly consists of crushable aluminum honeycomb energy-absorbing core segments that are encased in a sealed stainless steel shell. In addition to confining the aluminum honeycomb core segments in the event of a free drop, the impact limiter shell protects the aluminum honeycomb material from the weather. Both the top and bottom impact limiters are attached to the transportation cask body tie-down rings with 12, one inch diameter bolts. A tamper-indicating device is provided which connects each impact limiter to the transportation cask to assure that the package has not been opened by unauthorized personnel during transport.

A FuelSolutions™ canister consists of a steel shell assembly and an internal basket assembly. The shell assembly maintains a helium atmosphere for transport conditions. Credit is not taken for containment provided by the canister shell for transport conditions. The shell assembly also provides radiological shielding in both the radial and axial directions. The internal basket assembly provides geometric spacing, structural support, and criticality control for the spent nuclear fuel (SNF) assemblies for transport conditions.

There are two classes of W21 canisters (W21T and W21M), differing primarily in materials of construction. Each W21 canister class includes four different canister types, as follows. The W21T canister class includes a long canister with lead shield plugs (W21T-LL), a long canister with carbon steel shield plugs (W21T-LS), a short canister with lead shield plugs (W21T-SL), and a short canister with carbon steel shield plugs (W21T-SS). The W21M canister class includes a long canister with depleted uranium shield plugs (W21M-LD), a long canister with carbon steel shield plugs (W21M-LS), a short canister with depleted uranium shield plugs (W21M-SD), and a short canister with carbon steel shield plugs (W21M-SS). There are also two classes of W74 canisters (W74T and W74M), differing primarily in materials of construction. Both the W74T and W74M canister classes include only a long canister with carbon steel shield plugs.

A FuelSolutions™ canister shell assembly consists of a steel cylindrical shell, bottom end closure, bottom shield plug, bottom shell extension, bottom outer plate, top shield plug, top inner closure plate, and top outer closure plate. The closure plates at the top and bottom are welded to the cylindrical shell. All structural components of the canister shell assembly are constructed of austenitic stainless steel, with the exception of the shield plugs. The shield plug materials may be composed of lead, depleted uranium or carbon steel, depending upon the specific canister variant. To prevent any corrosion, galvanic, or chemical reactions between the shield plug materials and the cask environment or contents, the shield materials are isolated from the environment and cask interior. The lower shield plugs are encased within stainless steel. The upper shield plugs that are made of lead or depleted uranium are encased in stainless steel. The carbon steel upper shield plug is electroless nickel-plated.

A FuelSolutions™ W21 canister basket assembly consists of 21 guide tubes that are positioned and supported by a series of circular spacer plates, which are in turn positioned and supported by support rod assemblies. The W21 guide tubes include neutron absorber sheets on all four sides.

The W74 canister includes two stackable basket assemblies with a capacity to accommodate up to 64 Big Rock Point fuel assemblies. Each basket includes 37 cell locations, with the center five cell locations mechanically blocked to prevent fuel loading in these locations. The W74 basket assembly consists of a series of circular spacer plates that are positioned and supported by four support tubes that run through the spacer plates and support sleeves between the spacer plates. Each basket cell location, with the exception of the four support tubes and the five blocked-out center cells, contain a guide tube assembly. The W74 guide tube assemblies include borated stainless steel neutron absorber sheets on either one side or two opposite sides. The guide tubes are arranged in the basket to position at least one poison sheet between adjacent fuel assemblies, with the exception of intact fuel assemblies placed in the support tubes.

In the W74 basket, damaged fuel is placed in damaged fuel cans that are accommodated in the support tube cell locations. The W74 damaged fuel cans are similar to the W74 guide tubes, but include a screened bottom end, a screened removal lid, and borated stainless steel neutron absorber sheets on all four sides.

(3) Drawings

The FuelSolutions™ TS125 Transportation Package is constructed and assembled in accordance with the following drawings:

FS-200, Revision 1, Sheets 1 through 3
FS-205, Revision 2, Sheets 1 through 3
FS-210, Revision 2, Sheets 1 through 9
FS-220, Revision 1, Sheets 1 through 7
FS-230, Revision 1, Sheets 1 and 2
W21-110, Revision 4, Sheets 1 through 9

W21-120, Revision 5, Sheets 1 through 10
W21-121, Revision 5, Sheet 1
W21-122, Revision 3, Sheets 1 and 2
W21-130, Revision 4, Sheets 1 through 9
W21-131, Revision 3, Sheets 1 and 2
W21-140, Revision 5, Sheets 1 through 4
W21-150, Revision 4, Sheets 1 and 2
W21-190, Revision 4, Sheet 1
W74-110, Revision 5, Sheets 1 and 2
W74-120, Revision 5, Sheets 1 through 6
W74-121, Revision 7, Sheet 1
W74-122, Revision 6, Sheet 1
W74-130, Revision 6, Sheets 1 and 2
W74-140, Revision 5, Sheets 1 through 4
W74-150, Revision 5, Sheets 1 and 2
3319, Revision 6, Sheets 1 through 5

(b) Contents

(1) Type and Form of Material

Shipment of spent fuel, with plutonium in excess of 20 curies per package, in the form of debris, particles, loose pellets, and fragmented rods or assemblies, is not authorized.

(i) W21 Canister

The contents of the W21 canister are limited to 21 pressurized water reactor (PWR) SNF assemblies meeting the requirements of Table 1 and Table 2. Two different loading configurations, designated as W21-1 and W21-2, are permitted in the W21 canister. The W21-2 loading configuration, which accommodates SNF with higher initial ^{235}U enrichments, consists of up to 20 PWR SNF assemblies meeting the requirements of Table 1 and Table 2. The W21-2 loading configuration requires that the center guide tube be mechanically blocked to prevent inadvertent loading of a SNF assembly. If less than the maximum number of PWR assemblies are loaded, dummy assemblies having a width, length, and weight similar to that of the PWR assemblies they are replacing, must be loaded in the empty guide tubes.

The SNF assemblies that are permitted in the W21 canister must meet all of the parameter requirements of at least one criticality class. Table 2 lists the dimensional and initial enrichment limits for each criticality class of PWR fuel assembly. Table 2 provides separate assembly initial ^{235}U enrichment limits for the W21-1 and W21-2 canister loading configurations. The initial enrichment limits presented in Table 2 are bounding for assemblies containing any type of control insert, including assemblies with fuel rods replaced with any type of rod of equal or greater diameter and height.

Table 3 lists minimum required cooling times, as a function of burnup, for PWR assemblies loaded into the W21 canister. For a given fuel burnup level, assembly radiation sources increase with decreasing initial enrichment. Table 3 lists two minimum initial enrichment values for each assembly burnup level. Table 3 also lists two different minimum allowable cooling times, corresponding to the two minimum initial enrichment levels. An assembly must have an initial enrichment level equal to or greater than the value shown in Table 3, to qualify for the corresponding minimum allowable cooling time also shown in Table 3. Assemblies with initial enrichment levels lower than the lowest values shown (for the assembly's burnup level) in Table 3 are not qualified for transportation in the W21 canister.

Table 3 also gives limits on the total amount of initial (pre-irradiation) cobalt that may be present in the assembly active fuel zone (including both assembly and control insert hardware). For assemblies with less than 11 grams of cobalt in the fuel zone, the shorter cooling times shown in Table 3 may be used (provided that the minimum initial enrichment requirement is also met). The longer cooling times shown in Table 3 must be used for assemblies with over 11 grams of cobalt in the fuel zone. Cobalt present in control components that do not extend into the assembly fuel zone (such as thimble plug assemblies) or that do not reside in the core during operation (such as control rod assemblies) do not need to be included in the total fuel zone cobalt content.

All PWR SNF assembly control inserts placed in the W21 canister must be intact, and may contain B_4C, borosilicate glass, silver-indium-cadmium, hafnium, or Gd_2O_3 poison materials. Control insert rod cladding, and other insert hardware may consist of any type of zircaloy, stainless steel, or inconel. Any PWR assembly control insert that meets these material requirements may be loaded into the W21 canister. Control inserts that employ solid inconel rods that reside in the core, such as the B&W Grey APSRA, are not qualified for transportation in the W21 canister. Any insert that contains significant quantities of inconel (such as Inconel rod cladding) requires an evaluation of total assembly fuel zone cobalt quantity. Fuel rods may also be replaced with solid steel or Inconel rods, or rods containing any of the above poison materials, provided that the fuel zone cobalt requirements are met. UO_2 fuel rods containing Gd_2O_3 poison material are also permissible, although the poison is not relied upon to increase allowable ^{235}U initial enrichment levels for the fuel rod or assembly in question.

(ii) W74 Canister

The W74 canister contents are limited to 64 Big Rock Point (BRP) SNF assemblies without channels, including intact, partial, and damaged UO_2 and mixed oxide (MOX) fuel assemblies meeting the applicable acceptance criteria specified in Table 4 through Table 9. Specifications W74-1 and W74-2 for intact UO_2 and MOX fuel assemblies are provided in Table 4 and Table 5, respectively. Specifications W74-3 and W74-4 for partial UO_2 and MOX fuel assemblies are provided in Table 6 and Table 7, respectively. Lastly, specifications W74-5 and W74-6 for damaged UO_2 and MOX fuel assemblies are provided in Table 8 and Table 9, respectively. All UO_2 rods may contain any quantity of Gd_2O_3 poison material, provided that the specified ^{235}U initial enrichment limits are satisfied. BRP assemblies containing any amount of plutonium fuel (before irradiation) must meet the requirements of the MOX fuel specifications given in Table 5, Table 7, or Table 9. If less than the maximum number of BRP assemblies are loaded, dummy assemblies having a width, length, and weight similar to that of the BRP assemblies they are replacing, must be loaded in the empty guide tubes or support tubes.

The BRP UO_2 fuel assembly types permitted in the W74 canister are identified in Table 10. Any BRP fuel assemblies that do not meet all of the parameter requirements given for any fuel assembly class in Table 10 may only be loaded into the W74 canister damaged fuel can, as long as the requirements given in the applicable damaged fuel loading specification (W74-5 or W74-6) are still met. Any BRP fuel assemblies that meet all of the parameter requirements shown in Table 10, except for the requirement for the number of non-corner water holes, are classified as partial assemblies. The lower initial enrichment limits given in Specification W74-3 apply for those assemblies.

The specific BRP intact MOX fuel assembly types accommodated in the W74 canister are shown in Figure 1 through Figure 4. The specific BRP partial MOX fuel assembly types accommodated in the W74 canister are shown in Figure 5 through Figure 8. These figures show the maximum initial ^{235}U enrichment levels for the uranium present in all UO_2 and MOX fuel rods in each MOX assembly array. The figures also show the maximum overall weight percentage of PuO_2 in the initial MOX fuel rod (metal-oxide) material composition, with one exception. For the two MOX rods shown in Figure 4, the maximum total plutonium (metal) content, rather than the maximum overall weight percent of PuO_2 , is specified. The limits on maximum burnup, maximum heavy metal loading, and minimum cooling time for each BRP MOX fuel type are shown in Table 11.

Table 1 - Generic Requirements for All W21 Canister PWR SNF Contents

Fuel Assembly Parameter	Requirement
Fuel Rod Cladding Material	Zircaloy 2, 4
Assembly Condition	Intact (1)
Maximum Assembly Width (inch)	8.54
Maximum Burnup Level (MWd/MTU)	60,000 (2)
Maximum Uranium Loading (MTU/assy)	0.471
Axial Uranium Loading (kg/assy-inch)	3.27
Maximum Fuel Zone Height (inch)	150
Maximum Fuel Pellet Stack Density	96.5% (3)
Minimum Bottom Nozzle Height (inch)	1.97 (4)

Notes:

(1) Intact assemblies have no known or suspected fuel rod cladding defects greater than pinhole leaks and hairline cracks. Intact fuel also has no detectable grid spacer damage, or axial shifting in grid spacer location. Fuel assemblies with missing fuel rods (from the standard rod array configuration) may be loaded if all missing fuel rods are replaced with dummy rods that have a height and diameter at least as great as that of a standard fuel rod (i.e., by rods that displace an equal or greater volume of water).

(2) For assembly burnups exceeding 45,000 MWd/MTU, it is necessary to verify that the cladding oxide layer thickness does not exceed 70 μm, by measurement of a statistical sample of limiting fuel assemblies. The exposure (burnup) of any inserted control component must not exceed that of the host fuel assembly.

(3) Defined as the average material density within the cylindrical envelope volume covered by the fuel pellets, relative to the theoretical UO_2 density of 10.97 g/cc. Thus, "smearing" over fuel pellet dishes and chamfers to determine the "stack" density is acceptable.

(4) The bottom nozzle height is defined as the distance between the assembly bottom and the bottom of the active fuel.

Table 2 - W21 Canister SNF Assembly Dimensional and Enrichment Limits

Fuel Assembly Class(1)	Criticality Class(1)	Max. Initial Enrichment (w/o ^{235}U)(2)		Number of Fuel Rods	Min. Clad O.D. (in.)	Min. Clad Thickness (in.)	Min. Pellet Diameter (in.)	Fuel Rod Pitch (in.)	No. Guide / Instrument Tube Locations(5)
		W21-1(3)	W21-2(4)						
B&W 15x15	B&W 15x15	4.70	5.00	208	0.4300	0.0265	0.3675	0.568	17
B&W 17x17	B&W 17x17	4.60	4.90	264	0.3770	0.0220	0.3232	0.502	25
CE 14x14	CE 14x14	5.00	5.00	176	0.4400	0.0260	0.3700	0.580	5(6)
	CE 14x14 A	5.00	5.00	176	0.4400	0.0260	0.3795	0.568	5(6)
Palisades	CE 15x15 P	5.00	5.00	208 - 216	0.4135	0.0240	0.3500	0.550	1-9
Yankee Rowe	15x16	5.00	5.00	231	0.3650	0.0240	0.3105	0.472	1
	15x16 A	5.00	5.00	237	0.3650	0.0240	0.3105	0.468	1
CE 16x16 CE System 80 St. Lucie 2	CE 16x16	5.00	5.00	236	0.3820	0.0250	0.3250	0.506	5(6)
WE 14x14	WE 14x14	5.00	5.00	179	0.4000	0.0243	0.3444	0.556	17
WE 15x15	WE 15x15	4.70	5.00	204	0.4200	0.0240	0.3569	0.563	21
	WE 15x15 A	4.90	5.00	204	0.4240	0.0300	0.3565	0.563	21
WE 17x17	WE 17x17	4.70	5.00	264	0.3740	0.0225	0.3195	0.496	25
	WE 17x17 A	4.60	4.90	264	0.3600	0.0225	0.3088	0.496	25
	WE 17x17 B	4.60	4.90	264	0.3600	0.0250	0.3030	0.496	25

Notes:

(1) Assembly class defined per Energy Information Administration, *Spent Nuclear Fuel Discharges from U.S. Reactors 1993*, U. S. Department of Energy, 1995. The fuel assembly criticality classes are arbitrary designations given to each set of assembly parameters that are evaluated for criticality.

(2) The maximum allowable enrichments apply for all assemblies that meet the specified physical parameter requirements for the defined assembly class. The maximum allowable enrichments are defined as the maximum planar average enrichment at any axial assembly location. An exception is the CE 15x15 P assembly class, for which the maximum allowable enrichment applies to each individual fuel pin within the assembly.

(3) This enrichment limit applies for up to 21 SNF assemblies, in any W21 canister guide tube.

(4) This enrichment limit applies for up to 20 SNF assemblies, with the center guide tube empty.

(5) Whereas the number of guide tube locations is a specified parameter, the materials and dimensions of the guide tubes are not specified, since any quantity of steel or zircaloy in the guide tube locations will reduce assembly reactivity. Guide tube locations may contain nothing, hollow zircaloy or stainless rods (or rod clusters), solid zircaloy or stainless rods (or rod clusters), or poison rods (or rod clusters).

(6) The CE 14x14 and CE 16x16 assembly guide tubes occupy four fuel rod locations within the assembly array.

Table 3 - W21 Canister Minimum PWR Assembly Cooling Time Requirements

Assembly Burnup Level (GWd/MTU)(1)	Assembly Initial Enrichment (w/o ^{235}U)(1)	Assembly Fuel Zone Cobalt Qty (g/assy)(2)	Required Cooling Time (years)
≤35	≥2.8 %	≤ 11	≥ 6
≤40	≥3.0 %	≤11	≥ 8
≤45	≥3.3 %	≤11	≥ 10
≤50	≥3.5 %	≤11	≥ 12
≤55	≥3.8 %	≤11	≥15
≤ 60	≥4.0 %	≤11	≥18
≤35	≥1.5 %	≤50	≥15
≤40	≥1.5 %	≤50	≥20
≤45	≥1.5 %	≤50	≥25
≤50	≥2.5 %	≤50	≥25
≤55	≥3.0 %	≤ 50	≥25
≤60	≥3.5 %	≤50	≥25

Notes:

(1) Assembly average values.
(2) Defined as the total initial (pre-irradiation) cobalt mass within the assembly fuel zone, including any cobalt present in inserted control components.

Table 4 - W74 Canister Contents Specification W74-1
Intact UO_2 Fuel Assemblies

SNF Parameter	Loading/Acceptance Criteria
Payload Description	≤64 Big Rock Point BWR intact UO_2 fuel assemblies. [1,2,3] Any remaining empty canister basket guide tubes and/or support tubes may be loaded with fuel assemblies meeting any of the acceptable payload specifications W74-2 through W74-6, subject to the limitations of those specifications.
Cladding Material/Condition	Zircaloy 2,4 cladding with no known or suspected cladding defects greater than hairline cracks or pinhole leaks.
Maximum Uranium Loading	≤142.1 kg/assembly.
Maximum Initial Enrichment[4]	≤4.10 w/o ^{235}U.
Minimum Assembly Average Initial Enrichment	≥3.0 w/o ^{235}U.
Maximum Burnup	≤32,000 MWd/MTU.
Minimum Cooling Time	≥6.0 years.[5]

W74-1 Notes:

(1) Loaded assemblies must meet all of the assembly geometry requirements specified in Table 10, for any one of the defined assembly classes.

(2) Intact fuel assemblies include those BRP fuel assemblies with 1 to 4 corner rods missing, and BRP 9x9 fuel assemblies with 1 rod missing from a non-corner location. This includes assemblies with partial length rods, or rod fragments inside stainless tubes, in any of the array corner locations. It also includes 9x9 assemblies with 11x11 assembly rods in corner locations.

(3) Intact UO_2 assemblies may have any number of fuel rods replaced with solid zircaloy or stainless steel rods, or with poison rods, given that the length and diameter of the replacement rod are at least as great as that of the fuel rod. The empty array or guide tube locations may contain nothing, hollow zircaloy or stainless steel rods, neutron source rods, or any similar non-fissile fuel assembly component.

(4) Defined as the maximum array-average enrichment, which is the peak planar average initial enrichment considering all elevations along the assembly axis.

(5) If an intact UO_2 assembly has been further irradiated after having fuel rods replaced by dummy stainless rods, an evaluation must be performed that shows that the active fuel region non-fuel gamma source strength is bounded by that described in Section 5.2.2.1 of the WSNF-123 SAR. A similar evaluation is required for any assembly containing over 2.9 grams of initial cobalt in the assembly fuel zone.

Table 5 - W74 Canister Contents Specification W74-2
Intact MOX Fuel Assemblies

SNF Parameter	Loading/Acceptance Criteria
Payload Description	≤64 Big Rock Point BWR intact MOX fuel assemblies. (1,2,3) Any remaining empty canister basket guide tubes and/or support tubes may be loaded with fuel assemblies meeting any of the acceptable payload specifications W74-1 and W74-3 through W74-6, subject to the limitations of those specifications.
Cladding Material/Condition	Zircaloy 2,4 cladding with no known or suspected cladding defects greater than hairline cracks or pinhole leaks.
Maximum Heavy Metal Loading	The heavy metal loading varies by MOX assembly type and must not exceed the maximum values defined in Table 11.
Allowable Fuel Composition	Maximum initial ^{235}U enrichment and maximum PuO_2 weight percentage is shown for every fuel rod location in the MOX assembly array in Figure 1 through Figure 4. (4,5)
Maximum Burnup	The burnup varies by MOX assembly type and must not exceed the maximum values defined in Table 11.
Minimum Cooling Time	The cooling time varies by MOX assembly type and must not be less than the minimum values defined in Table 11. (6)

W74-2 Notes:

(1) Intact MOX assemblies may have any number of fuel rods replaced with solid zircaloy or stainless steel rods, or with poison rods, given that the length and diameter of the replacement rod are at least as great as that of the fuel rod. They may also have hollow zircaloy or stainless steel rods, neutron source rods, or any similar non-fissile fuel assembly component placed in the empty array or guide tube locations, including all forms of inserts or control components.

(2) J2 (Figure 1) MOX assemblies must meet all of the assembly geometry requirements shown for Siemens 9x9 fuel in Table 10. DA and G-Pu (Figure 2 and Figure 3, respectively) MOX assemblies must meet all of the assembly geometry requirements shown for Siemens 11x11 fuel in Table 10. One exception is that J2 MOX assemblies with a cladding thickness of 0.05 inches and a fuel pellet diameter of 0.4515 inches are also acceptable. UO_2 9x9 assemblies with 2 inserted MOX rods (shown in Figure 4) must meet all of the assembly geometry requirements shown for Siemens 9x9 in Table 10.

(3) Intact G-Pu MOX assemblies may have 0 to 4 fuel rods in the array corner locations. G-Pu MOX assemblies may also have partial length rods, or rod fragments inside stainless tubes, in any of the array corner locations.

(4) The maximum ^{235}U enrichment shown in Figure 1 through Figure 4 is defined as the weight percentage of ^{235}U in any uranium that is present in the rod. The PuO_2 weight percentage is the overall mass of PuO_2 in the rod divided by the overall metal-oxide (UO_2 + PuO_2) mass in the rod. Fuel rods in candidate assemblies may have 235U enrichment levels and PuO_2 weight percentages that are equal to or less than the values shown in Figure 1 through Figure 4 for that fuel rod array location.

(5) Figure 4 specifies a maximum total MOX fuel rod plutonium metal mass as opposed to a maximum PuO_2 weight percentage.

(6) If an intact MOX assembly has been further irradiated after having fuel rods replaced by dummy stainless rods, an evaluation must be performed that shows that the active fuel region non-fuel gamma source strength is bounded by that described in Section 5.2.2.1 of the WSNF-123 SAR. A similar evaluation is required for any assembly containing over 2.9 grams of initial cobalt in the assembly fuel zone.

Table 6 - W74 Canister Contents Specification W74-3
Partial UO_2 Fuel Assemblies

SNF Parameter	Limit/Specification
Payload Description	≤64 Big Rock Point BWR partial UO_2 fuel assemblies.[1,2] Partial fuel assemblies are defined as those assemblies having one or more full-length fuel rods missing from the intact fuel assembly array (except as permitted by W74-1 Notes 2 and 3). The affected array locations may contain nothing, partial length rods, hollow zircaloy or stainless steel rods, neutron source rods, or any other non-fissile fuel assembly component with a lower length or diameter than a full-length fuel rod. Any remaining empty canister basket guide tubes and/or support tubes may be loaded with fuel assemblies meeting any of the acceptable loading specifications W74-1, W74-2, and W74-4 through W74-6, subject to the limitations of those specifications.
Cladding Material/Condition	Zircaloy 2,4 cladding with no known or suspected cladding defects greater than hairline cracks or pinhole leaks.
Maximum Uranium Loading	≤142.1 kg/assembly
Maximum Initial Enrichment[3]	≤3.55 w/o ^{235}U (9x9) ≤3.6 w/o ^{235}U (11x11)
Minimum Assembly Average Initial Enrichment	≥3.0 w/o ^{235}U
Maximum Burnup	≤32,000 MWd/MTU
Minimum Cooling Time	≥6.0 years[4]

<u>W74-3 Notes:</u>

(1) Partial UO_2 assemblies may have any number of fuel rods replaced with solid zircaloy or stainless steel rods, or with poison rods.

(2) Loaded partial assemblies must meet all of the geometry requirements shown (for any of the defined assembly classes) in Table 10, except for the "maximum number of non-corner water holes."

(3) Defined as the maximum array average initial enrichment, which is the peak planar average initial enrichment considering all elevations along the fuel assembly axis. The averaging is applied only to those fuel rods that are present in the partial array.

(4) *If a partial UO_2 assembly has been further irradiated after having fuel rods replaced by dummy stainless rods, an* evaluation must be performed that shows that the active fuel region non-fuel gamma source strength is bounded by that described in Section 5.2.2.1 of the WSNF-123 SAR. A similar evaluation is required for any assembly containing over 2.9 grams of initial cobalt in the assembly fuel zone.

Table 7 - W74 Canister Contents Specification W74-4
Partial MOX Fuel Assemblies

SNF Parameter	Loading/Acceptance Criteria
Payload Description	≤64 Big Rock Point BWR partial MOX fuel assemblies.(1,2,3) Partial MOX assemblies must conform exactly to one of the four partial assembly array configurations shown in Figure 5 through Figure 8, with respect to the number and location of missing fuel rods within the assembly array. The missing fuel rod array locations may contain nothing, hollow zircaloy or stainless steel rods, neutron source rods, or any other non-fissile fuel assembly component. Any remaining empty canister basket guide tubes and/or support tubes may be loaded with fuel assemblies meeting any of the acceptable loading specifications W74-1 through W74-3, W74-5, and W74-6, subject to the limitations of those specifications.
Cladding Material/Condition	Zircaloy-2,4 cladding with no known or suspected cladding defects greater than hairline cracks or pinhole leaks.
Maximum Heavy Metal Loading	The heavy metal loading varies by fuel assembly type and must not exceed the maximum values defined in Table 11.
Allowable Fuel Composition	Maximum initial ^{235}U enrichment and maximum PuO_2 weight percentage is shown for every fuel rod location (in each of the four allowable partial MOX assembly array configurations) in Figure 5 through Figure 8.(4)
Maximum Burnup	The burnup varies by MOX assembly type and must not exceed the maximum values defined in Table 11.
Minimum Cooling Time	The cooling time varies by MOX assembly type and must not be less than the minimum values defined in Table 11.

W74-4 Notes:

(1) Partial MOX assemblies may have any number of fuel rods replaced with solid zircaloy or stainless steel rods, or with poison rods, given that the length and diameter of the replacement rod are at least as great as that of the fuel rod.

(2) If a partial MOX assembly has been further irradiated after having fuel rods replaced by dummy stainless rods, an evaluation must be performed that shows that the active fuel region non-fuel gamma source strength is bounded by that described in Section 5.2.2.1 of the WSNF-123 SAR. A similar evaluation is required for any assembly containing over 2.9 grams of initial cobalt in the assembly fuel zone.

(3) Loaded partial assemblies must meet all of the geometry requirements shown (for any of the defined assembly classes) in Table 10, except for the "maximum number of non-corner water holes."

(4) The maximum ^{235}U enrichment shown in Figure 5 through Figure 8 is defined as the weight percentage of ^{235}U in any uranium that is present in the rod. The PuO_2 weight percentage is the overall mass of PuO_2 in the rod divided by the overall metal-oxide (UO_2 + PuO_2) mass in the rod. Fuel rods in candidate assemblies may have ^{235}U enrichment levels and PuO_2 weight percentages that are equal to or less than the values shown in Figure 5 through Figure 8 for that fuel rod array location.

Table 8 - W74 Canister Contents Specification W74-5
Damaged UO_2 Fuel Assemblies

SNF Parameter	Limit/Specification
Payload Description	≤8 Big Rock Point BWR damaged UO_2 fuel assemblies. Damaged fuel assemblies are defined as those with fuel cladding damage in excess of hairline cracks or pinhole leaks. Fuel assemblies with damaged grid spacers (defined as damaged to a degree where fuel rod structural integrity cannot be assured, or where grid spacers have moved from their design position) are also considered to be damaged fuel assemblies. Each fuel assembly designated as damaged must be placed within a damaged fuel can and loaded into a basket support tube in the upper or lower basket. The remaining empty canister basket guide tubes and support tubes may be loaded with fuel assemblies meeting any of the acceptable loading specifications W74-1 through W74-4 and W74-6, subject to the limitations of those specifications, for a total of ≤64 Big Rock Point BWR fuel assemblies. Any intact or partial UO_2 fuel assembly that does not meet all of the assembly geometry requirements shown in Table 10 (other than the number of water holes) must also be loaded into a damaged fuel can.
Cladding Material/Condition	Zircaloy 2,4 cladding with fuel rod damage in excess of hairline cracks or pinhole leaks.
Maximum Uranium Loading	≤142.1 kg/assembly.
Maximum Initial Enrichment	≤4.61 w/o ^{235}U peak fuel pellet initial enrichment.
Maximum Pellet Density	≤96.5% (as defined in Table 10, Note 1).
Minimum Assembly Average Initial Enrichment	≥3.0 w/o ^{235}U
Maximum Burnup	≤32,000 MWd/MTU.
Minimum Cooling Time	≥6.0 years.(1)

W74-5 Note:

(1) If a damaged UO_2 assembly has been further irradiated after having fuel rods replaced by dummy stainless rods, an evaluation must be performed that shows that the active fuel region non-fuel gamma source strength is bounded by that described in Section 5.2.2.1 of the WSNF-123 SAR. A similar evaluation is required for any assembly containing over 2.9 grams of initial cobalt in the assembly fuel zone.

Table 9 - W74 Canister Contents Specification W74-6
Damaged MOX Fuel Assemblies

SNF Parameter	Limit/Specification
Payload Description	≤ 8 Big Rock Point BWR damaged MOX fuel assemblies. Damaged fuel assemblies are defined as those with fuel cladding damage in excess of hairline cracks or pinhole leaks. Fuel assemblies with damaged grid spacers (defined as damaged to a degree where the fuel rod structural integrity cannot be assured, or where the grid spacers have shifted vertically from their design position) are also considered to be damaged fuel assemblies. Each fuel assembly designated as damaged must be placed within a damaged fuel can and loaded into a support tube locations in the upper and lower basket. The remaining empty canister basket guide tubes and support tubes may be loaded with fuel assemblies meeting any of the acceptable loading specifications W74-1 through W74-5, subject to the limitations of those specifications, for a total of ≤64 Big Rock Point BWR fuel assemblies. Any intact or partial MOX assembly that does not meet all of the assembly geometry requirements shown in Table 10 (other than the number of water holes) must also be loaded into a damaged fuel can.(1)
Cladding Material/ Condition	Zircaloy 2,4 cladding with fuel rod damage in excess of hairline cracks or pinhole leaks.
Maximum Pellet Density	96.5% (as defined in Table 10, Note 1)
Maximum Heavy Metal Loading	The heavy metal loading varies by MOX assembly type and must not exceed the maximum values defined in Table 11.
Allowable Fuel Composition	≤4.61 w/o ^{235}U for all UO_2 fuel pellets. All MOX fuel pellets must meet the maximum ^{235}U enrichment and PuO_2 weight percentage requirements for one of the four MOX fuel material compositions described in Figure 1 through Figure 3.
Maximum Burnup	The burnup varies by MOX assembly type and must not exceed the maximum values defined in Table 11.
Minimum Cooling Time	The cooling time varies by MOX assembly type and must not be less than the minimum values defined in Table 11.(2)

W74-6 Notes:

(1) The UO_2 9x9 assemblies with 2 inserted MOX rods (shown in Figure 4) may not be loaded into the W74 damaged fuel can.

(2) If a damaged MOX assembly has been further irradiated after having fuel rods replaced by dummy stainless rods, an evaluation must be performed that shows that the active fuel region non-fuel gamma source strength is bounded by that described in Section 5.2.2.1 of the WSNF-123 SAR. A similar evaluation is required for any assembly containing over 2.9 grams of initial cobalt in the assembly fuel zone.

Table 10 - W74 Canister Fuel Geometry Specifications

Fuel Assembly Parameter	Fuel Assembly Class			
	GE 9x9	Siemens 9x9	Siemens 11x11	Siemens 11x11A
Fuel Pellet Stack Density(1)	≤ 96.5%	≤96.5%	≤96.5%	≤96.5%
Number of Fuel Rods	≤81	≤ 81	≤121	≤ 121
Clad O.D. (in)	0.5625	0.5625	0.449	0.449
Clad Thickness (in)	0.040	0.040	0.034	0.034
Pellet Diameter (in)	0.471	0.4715(2)	0.3715	0.3735
Fuel Rod Pitch (in)	0.707	0.707	0.577	0.577
Active Fuel Length (in)	≤70	≤70	≤70	≤ 70
Number of Array Corner Rods(3)	0-4	0-4	0-4	0-4
Number of Non-Corner Water Holes(3)	≤ 1	0	0	0
Number of Inert Rods(3)	≥0	≥ 0	≥0	≥0
Bottom Tie Plate Height (in)(4)	≥1.25	≥1.25	≥1.25	≥1.25

Notes:

(1) The fuel pellet stack density is defined as the average density of the fuel pellet material (within the cylindrical envelope volume covered by the pellet stack) divided by the theoretical UO_2 density of 10.97 g/cc. Thus, smearing the fuel material over the dishing and chamfer voids in the pellet stack is acceptable for determining the stack density.

(2) Assemblies E65 and E72 may each contain two MOX fuel rods with either solid pellets or annular pellets with a 0.1 inch or 0.2 inch inside diameter. In any given MOX fuel rod, the entire pellet stack must contain the same pellet type (i.e., solid, 0.1-inch annular, or 0.2-inch annular).

(3) The definitions of corner rods, non-corner rods, and inert rods are given in the W74-1 and W74-3 assembly loading specifications.

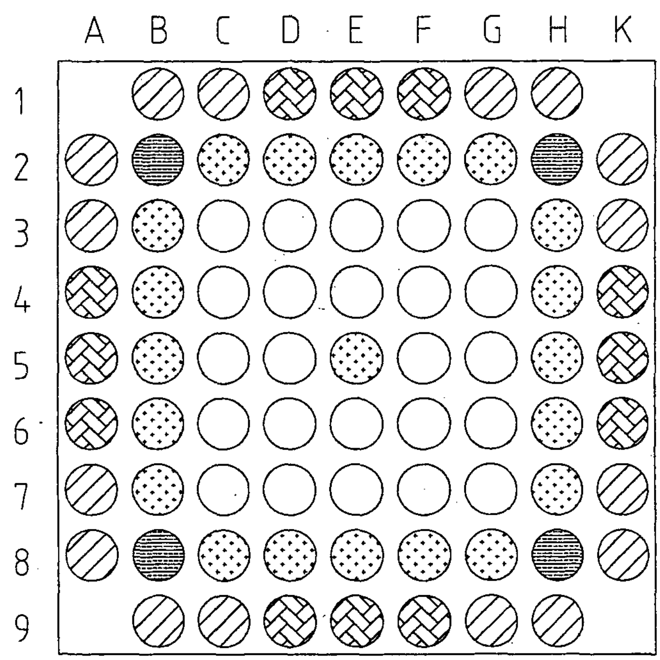
(4) Defined as the distance from the bottom of the assembly to the bottom of the active fuel.

Table 11 - W74 Canister Assembly Specific Requirements for Big Rock Point MOX Fuel

BRP MOX Assembly Type	Maximum Heavy Metal Loading (kg)	Maximum Burnup (MWd/MTIHM)(1)	Minimum Cooling Time (years)
J2 (9x9)	124	22,820	22
DA (11x11)	126	21,850	22
G-Pu (11x11)	127	34,220	15
UO_2 9x9 with 2 inserted MOX rods	142.1	32,000	6

Note:

(1) The exposure (burnup) of any inserted control component must not exceed that of the host fuel assembly.



Fuel Pin Compositions

2.55 Wt% U-235

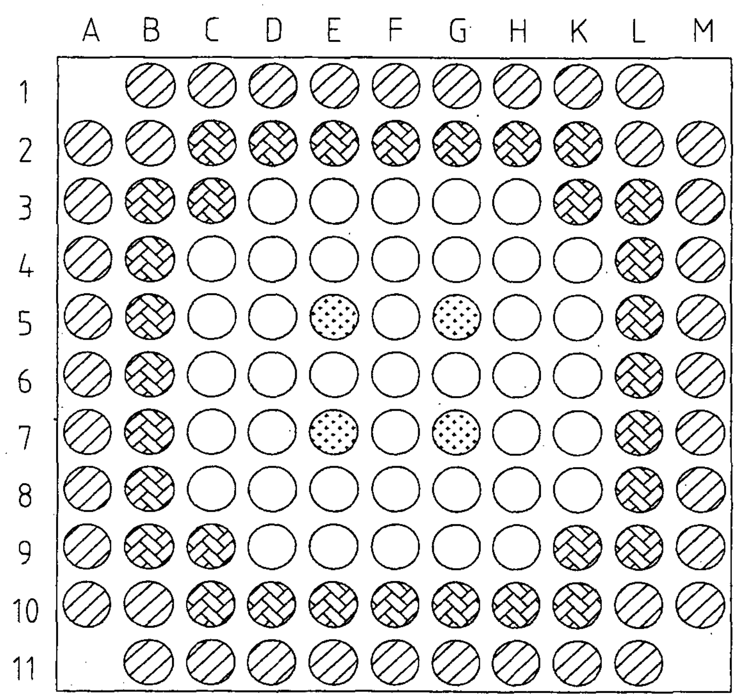
3.30 Wt% U-235

4.50 Wt% U-235

3.30 Wt% U-235 and 1.00 % Gd2O3 in UO2

0.711 Wt% U-235 3.65 % PuO2

Figure 1 - J2 (9x9) BRP MOX Assembly Array



Fuel Pin Compositions

2.40 Wt% U-235

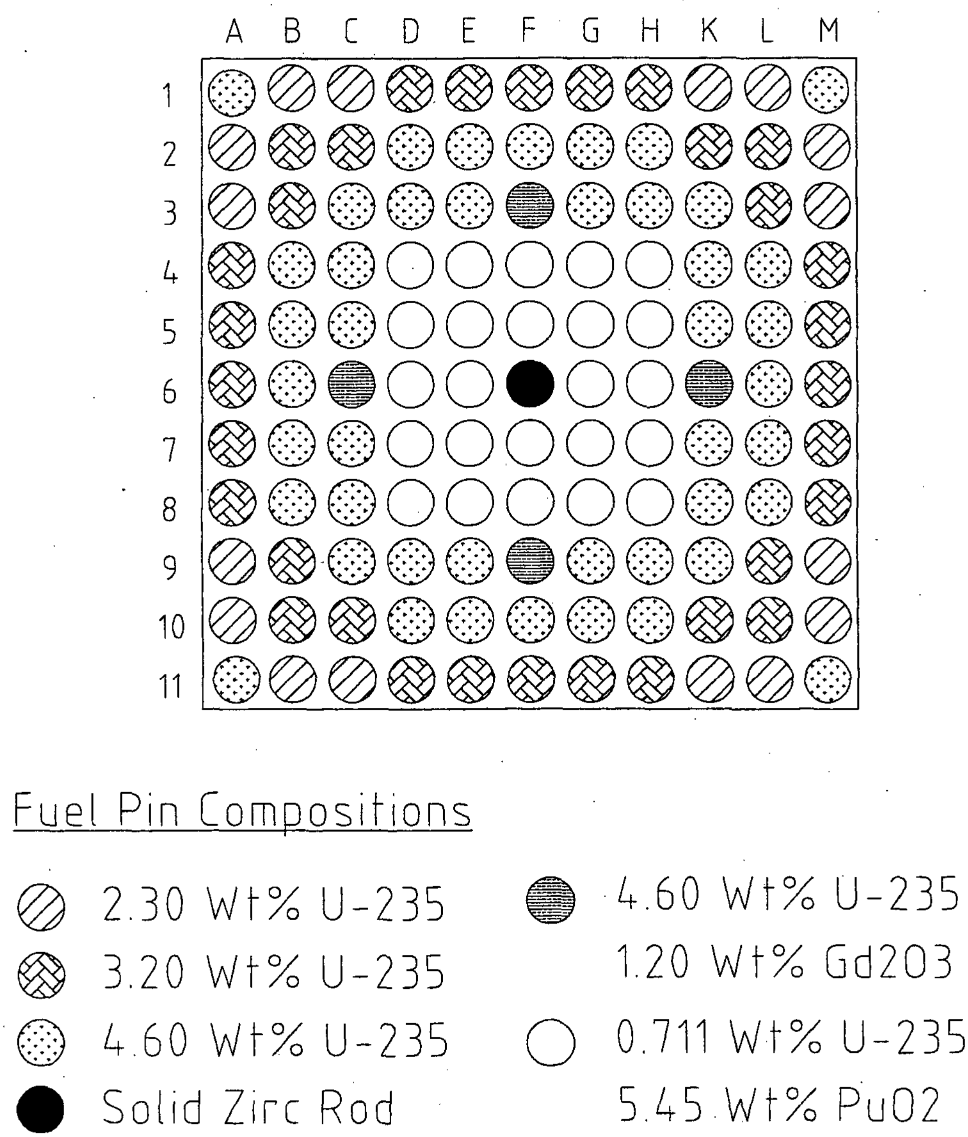
1.56 Wt% U-235
1.03 Wt% PuO2

2.40 Wt% U-235
2.45 Wt% PuO2

Water Rods

Note: Water rods are identical to the fuel rods (same diameter and cladding thickness), except that they contain no fuel pellets.

Figure 2 - DA (11x11) BRP MOX Assembly Array



Note: G-Pu assemblies may have any number of fuel rods missing (or present) in the four array corner locations

Figure 3 - G-Pu (11x11) BRP MOX Assembly Array

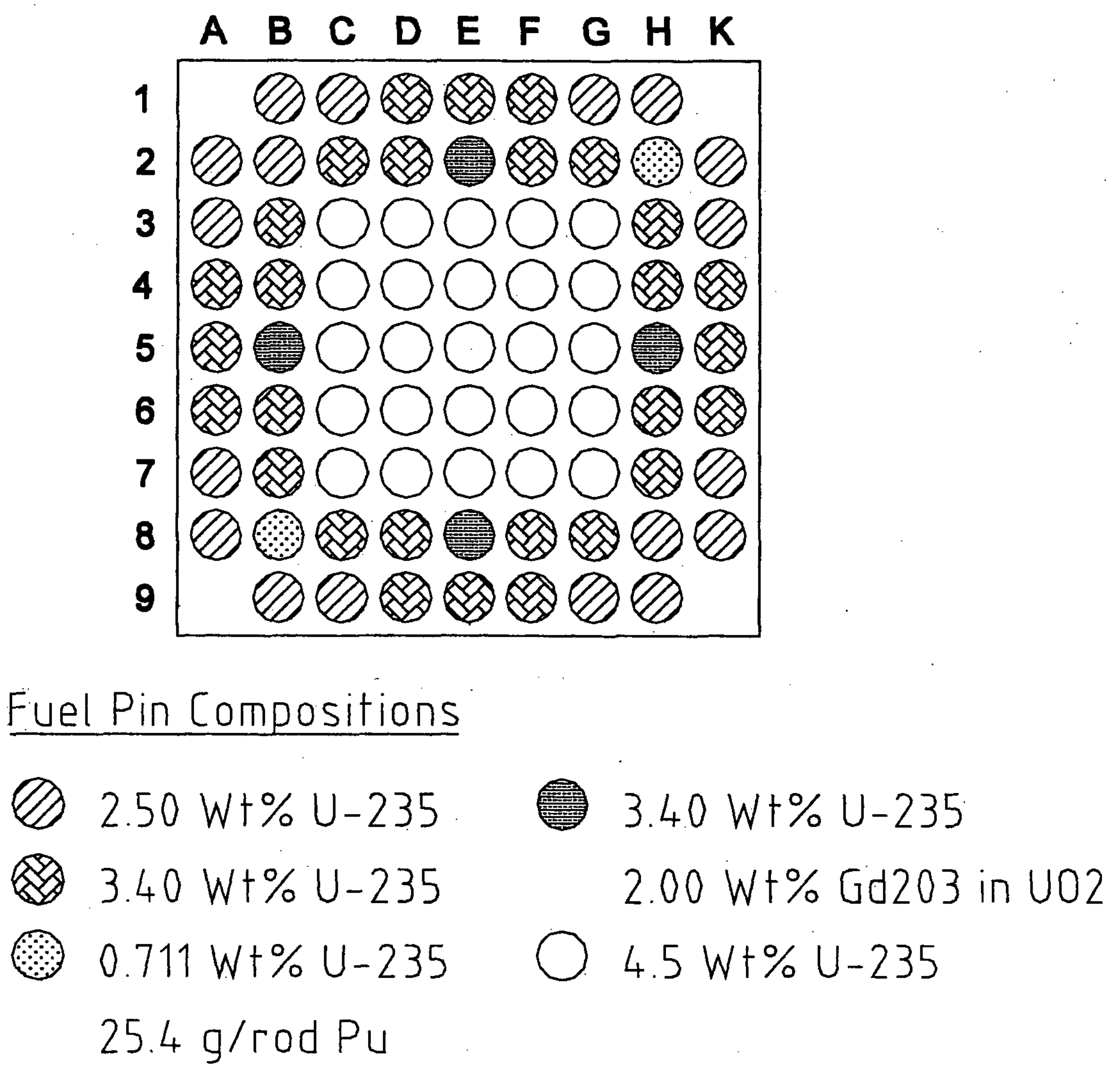


Figure 4 - UO2 9x9 BRP Assembly with Two Inserted MOX Rods

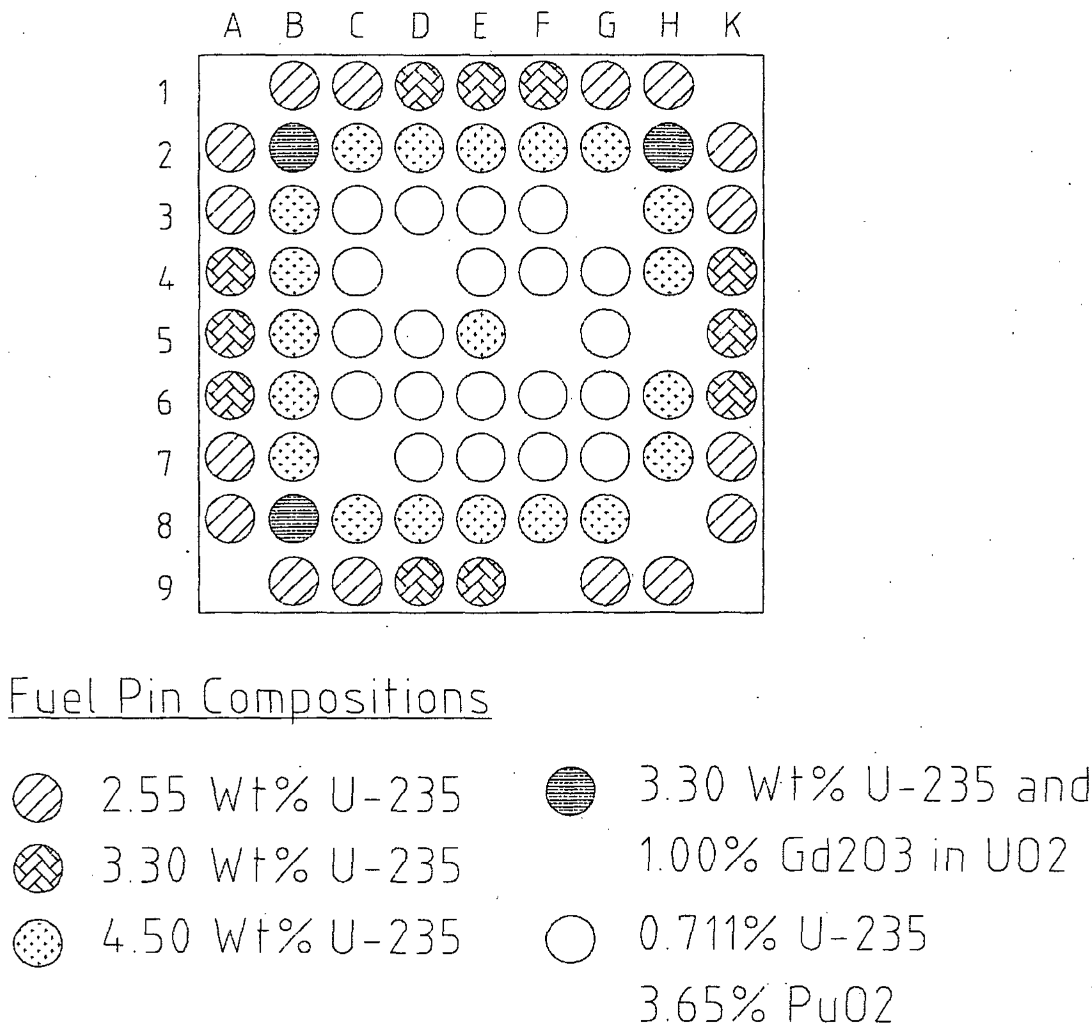


Figure 5 - J2 Partial MOX Assembly Array #1

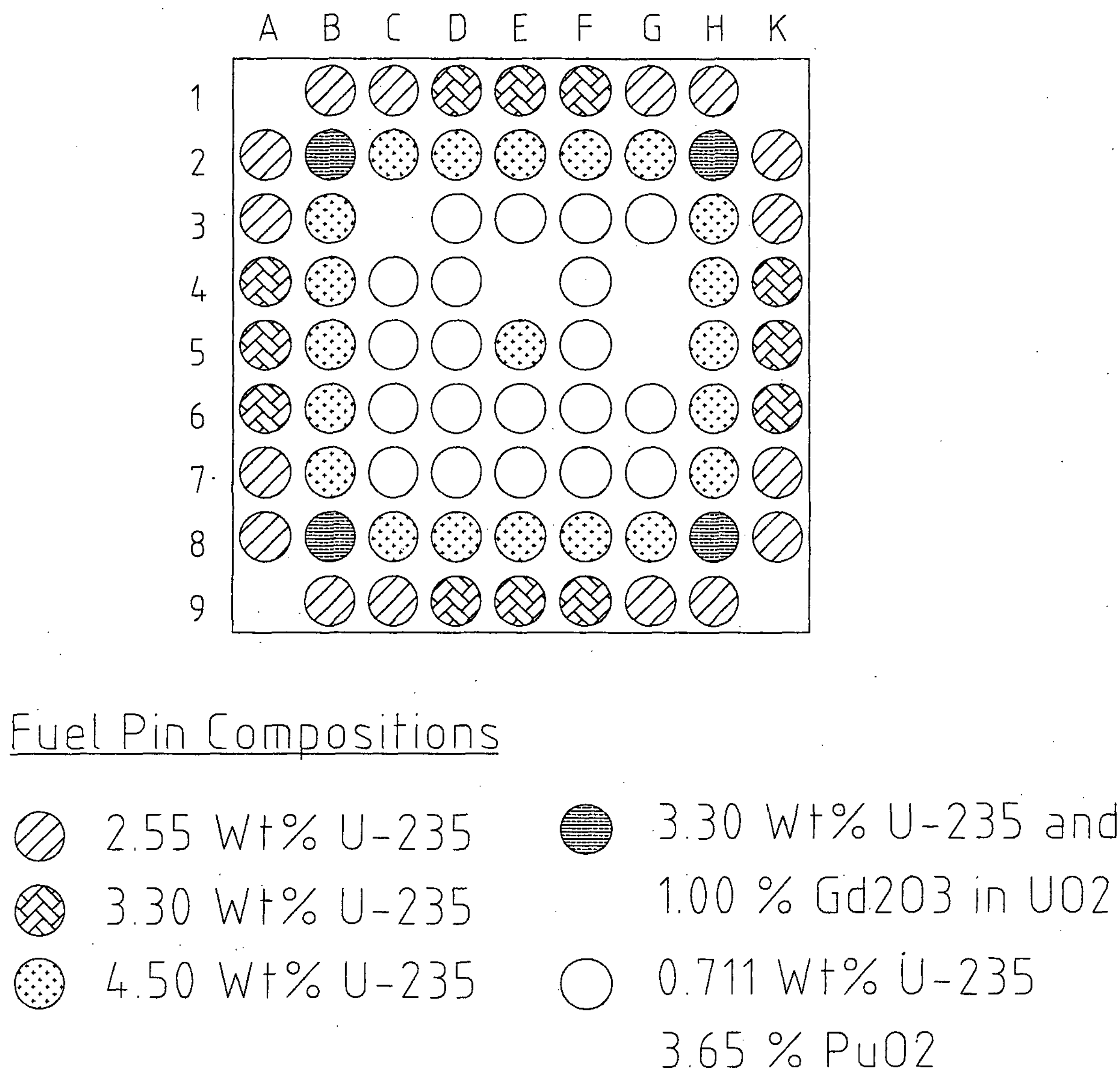


Figure 6 - J2 Partial MOX Assembly Array #2

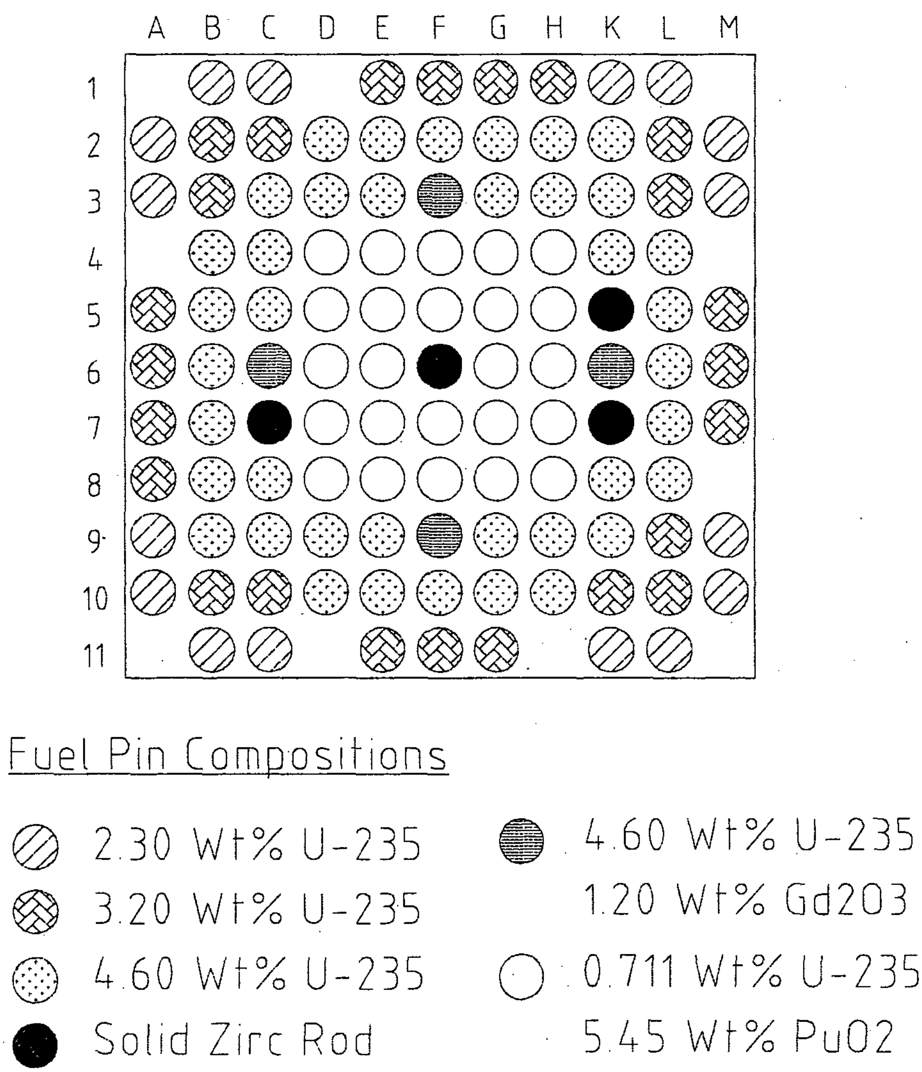


Figure 7 - G-Pu Partial MOX Assembly Array #1

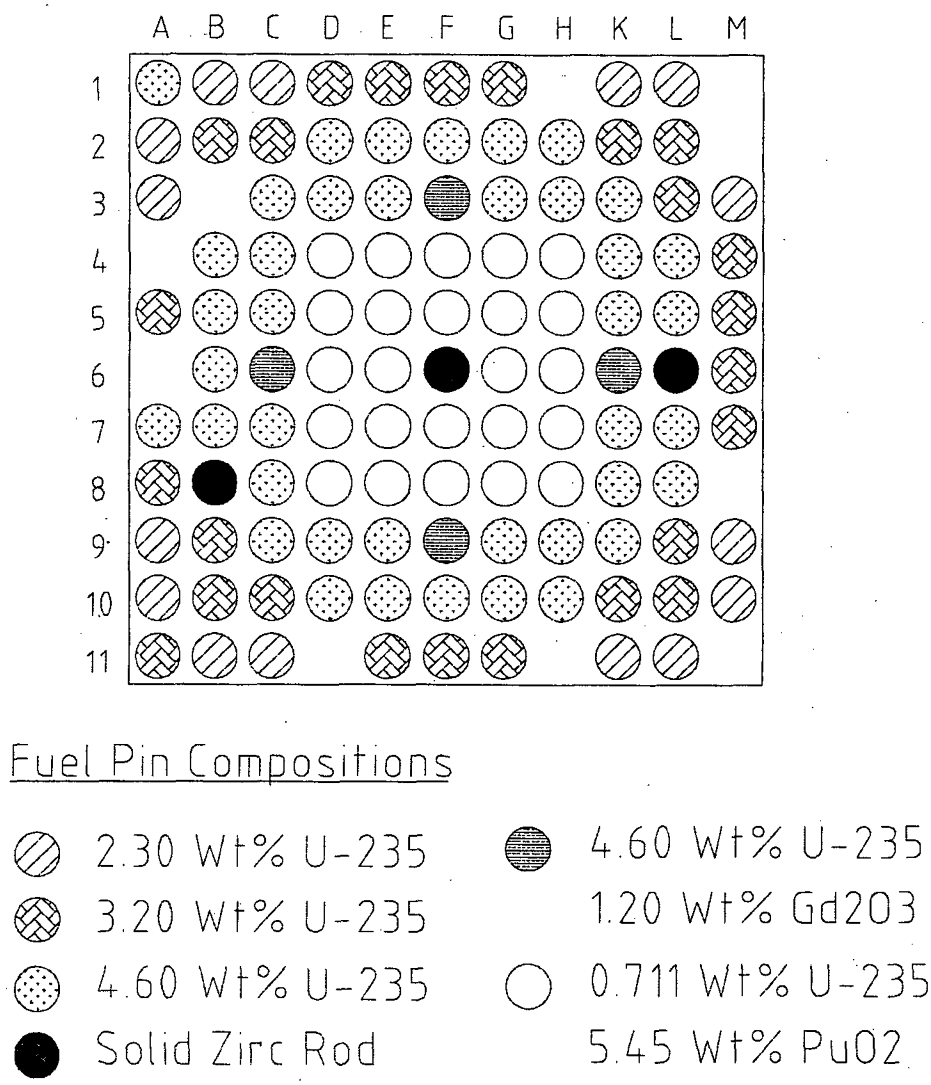


Figure 8 - G-Pu Partial MOX Assembly Array #2

(2) Maximum Quantity of Material Per Package

The maximum payload weight of the TS125 Transportation cask is 85,000 pounds. The payload weight includes the weight of the FuelSolutions™ canister and its SNF payload, plus the weight of the cask cavity spacer for short canisters.

(3) Decay Heat Limit

The W74 canister loading criteria can be described as follows:

A Big Rock Point spent fuel assembly is allowed to be shipped in the canister if Q (heat generation per assembly) $\leq$ 0.275 kW.

No decay heat limit is specified for the W21 canister. The PWR assembly fuel parameters requirements given in Table 3 ensure that assembly heat generation levels will not exceed the heat generation level that was analyzed in the thermal licensing evaluations (1.05 kW/assembly).

(c) Criticality Safety Index

(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:

(1) The package shall meet the Acceptance Tests and Maintenance Program of Chapter 8 of the application, as supplemented.

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

7. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.12, until October 1, 2004, and under the provisions of 10 CFR 71.17 thereafter.

8. Expiration date: September 30, 2007

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9276	2	71-9276	USA/9276/B(U)F-85	26	OF	26

REFERENCES

BNFL Fuel Solutions Corporation, application dated April 20, 2001.

Supplements dated June 7, 2001; January 22, February 5, February 28, April 11, and April 30,2002; January 17, August 7, and November 26, 2003; and April 20, April 28, April 29, May 7, and May 12, 2004.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

James R. Hall for

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

ate: May 28, 2004

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
9277	3	71-9277	USA/9277/B( )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)*

General Atomics
3550 General Atomics Court
San Diego, CA 92121

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Public Service Company of Colorado application dated March 28, 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.: FSV-1 Unit 3

(2) Description

The FSV-1 Unit 3 is a stainless steel-encased, depleted uranium-shielded cask. The cask body is a cylinder 208-inches long and 28-inches in diameter, except for the top flange area, which is 31-inches in diameter. The cavity is approximately 17.7-inches in diameter and 187.6-inches long.

The cask may be used in one of seven configurations (A through G) depending on contents. Configurations A, B, C, and D are used to ship solid, non-fissile irradiated hardware. These configurations use an outer lid consisting of a 3.75-inch thick stainless steel plate and a 2.25-inch thick depleted uranium shield. The lid is bolted to the cask body by 24 1.25-inch diameter fasteners. The primary seal is a silicone elastomeric seal ring between the outer lid and cask body. Configuration B does not require an inner container. Configuration C uses a supplemental stainless steel shield ring and cover plate. Configuration D uses a supplemental carbon steel shield ring and cover plate.

Configuration E is used to ship Fort St. Vrain (FSV) high temperature gas reactor (HTGR) fuel elements. This configuration uses the stainless steel inner container (as shown in General Atomic Drawing Nos. GADR 55-2-1, Rev. C, and GADR 55-2-2, Rev. A) as the containment vessel. The inner container lid is a stainless steel shell containing depleted uranium 4.15-inches thick. The inner lid is secured to the inner container body by 12 0.5-inch diameter fasteners. The primary seal is a silicone elastomeric seal ring between the inner lid and inner container body. Configuration E is equipped with an impact limiter on the upper end.

Configuration F and G are used to shop solid non-fissile irradiated and contaminated hardware from the FSV HGTR. These configurations use a 4.75-inch thick steel outer lid. The lid is secured to the cask body by 24 1.25-inch diameter fasteners. The primary seal is a molded silicone elastomeric seal ring between the outer lid and the cask body. Configurations F and G both use an impact limiter on the upper end. Configurations F and G also use a burial canister with a 12-inch thick carbon steel plug. The shielded spacer in the burial canister is used only in Configuration G.

The overall weight for the FSV-1 Unit 3 package is 46,025 pounds for Configurations A, B, C, and D and 47,600 pounds for Configurations E, F, and G.

(3) Drawings

The FSV-1 Unit 3 package is constructed in accordance with the following drawings:

Configuration A

National Lead Company Drawing Nos. : 70086F, Rev. 7; 70296F, Rev. 2; and General Atomics Drawing No. 1501-003, Rev. C

Configuration B

Same as for Configuration A except that an inner container is not required.

Configuration C and D

In addition to the drawings for Configuration A, General Atomics Drawing Nos. GADR 55-2-10, Issue D, and GADR 55-2-14, Issue N/C (optional). Configuration C uses a supplemental stainless steel shield ring and cover plate constructed in accordance with Drawing No. GADR 55-2-11, Issue B. Configuration D uses a supplemental carbon steel shield ring and cover plate constructed in accordance with Drawing No. GADR 55-2-11, Issue A.

Configuration E

In addition to the drawings for Configuration A, General Atomics Drawings Nos. GADR 55-2-1, Issue C; GADR 55-2-2, Issue A; and GADR 55-2-3, Issue B.

Configuration F and G

In addition to the drawings for Configuration A, General Atomic Drawings Nos. GADR 55-2-1, Issue C; GADR 55-2-2, Issue A; GADR 55-2-12, Issue C; and GADR 55-2-13, Issue A.

5. (b) Contents

(1) Type and form of material

(i) Irradiated fuel elements consisting of graphite body, hexagonal in horizontal cross section, approximately 31.2-inches high and 14.2-inches across the flats. Prior to irradiation, each fuel element contains thorium and uranium enriched to a maximum of 93.5 w/o in the U-235 isotope, or

(ii) Solid, irradiated, and contaminated hardware, which may include fissile material, 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 until October 1, 2004, and 10 CFR 71.15, thereafter and neutron source components, or

(iii) Solid, nonfissile, irradiated and contaminated hardware which has been removed from the Fort St. Vrain High Temperature Gas Cooled Reactor and the surface contamination does not exceed 51 millicuries per package.

(2) Maximum quantity of material per package

Decay heat not to exceed 4.1 kw and:

(i) Item 5(b)(1)(i) above:

Six fuel elements each containing a maximum of 1.4 kg of enriched uranium, having a thorium/uranium ratio greater than 8.1:1 and weighing approximately 300 pounds. The gross weight of the cask cavity contents, including the component spacers, inner container, and irradiated fuel elements shall not exceed 4,430 pounds. Contents must be shipped in Configuration E.

(ii) Item 5(b)(1)(ii) above:

The gross weight of the cask cavity contents, including appropriate component spacers, liners, inner containers, shield rings and solid, non-fissile, irradiated and contaminated hardware shall not exceed 3,720 pounds. Contents must be shipped in Configurations A, B, C, or D.

(iii) Item 5(b)(1)(iii) above:

The gross weight of all of the cask cavity contents, including burial canister and spacers, with or without supplemental shielding shall not exceed 4,430 pounds. Contents must be shipped in Configurations F or G.

5. (c) Criticality Safety Index

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

6. As needed, appropriate component spacers must be used in the cask cavity when shipping the contents described in paragraph 5(b) to limit movement of contents during shipment.

7. For transport of the contents of Item 5(b)(1)(ii) in Configuration D, the dose rate measured on the surface of the package must not exceed 200 mr/hr. For the purpose of this requirement, the surface of any personnel barrier may not be considered the surface of the package.

8. The Model No. FSV-1 Unit 3 cask may be wrapped with reinforced plastic when shipping the contents described in Item 5(b)(1)(ii) or (iii) provided the heat generation rate does not exceed 500 watts. The applicable requirements of 10 CFR 71.87 must be satisfied prior to wrapping the cask.

9. Use of packaging fabricated after August 31, 1986, is not authorized.

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

(a) Configurations A, B, C, and D of the Model FSV-1 Unit 3 shipping cask shall be prepared for shipment and operated in accordance with the Operating Procedures of Section 7.0, Volume I, of the application, as supplemented. The package shall be maintained in accordance with the Maintenance Program in Section 8.0, Volume I, of the application, as supplemented.

(b) Configurations E, F, and G of the Model FSV-1 Unit 3 shipping cask shall be prepared for shipment and operated in accordance with the Operating Procedures of Section 7.0, Volume II, of the application, as supplemented. The packages shall be maintained in accordance with the Maintenance Program in Section 8.0, Volume II, of the application, as supplemented.

(c) The main flange seals must be replaced within twelve (12) months prior to any use of the packaging and must be replaced if inspection shows any defect.

(d) The silicone O-ring on the inner container primary plug in Configuration E must be replaced within the twelve (12) months prior to any use of the packaging and must be replaced if inspection shows any defect.

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

12. Revision 2 of this certificate may be used until May 31, 2007.

13. Expiration date: October 1, 2008. This certificate is not renewable.

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
9277	3	71-9277	USA/9277/B( )F	5	OF	5

REFERENCE

Public Service Company of Colorado application dated March 28, 1996, as supplemented by Chem-Nuclear Systems, L.L.C., letter dated May 19, 1997, and General Atomics letter dated June 6, 1997, as supplemented April 11, 2001, June 7, 2001, and May 5, 2006

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Robert A. Nelson, Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: 5/31/06

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
9279	4	71-9279	USA/9279/B(U)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 20586

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Washington TRU Solutions LLC application dated October 4, 2004, 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.: HalfPACT Waste Shipping Container

(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), an approximate 8-inch thick layer of polyurethane foam, a 1/4-inch thick layer of ceramic fiber paper 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 92 inches height. The package weighs not more than 18,100 pounds when loaded with the maximum allowable contents of 7,600 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. 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 74 inches diameter and 69 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. 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 45 inches height.

5.(a)(3) Drawings

The package is constructed and assembled in accordance with Packaging Technology, Inc., Drawing 707-SAR Sheets 1-12, Rev. 6. The standard pipe overpack is constructed and assembled in accordance with Packaging Technology, Inc. Drawing No. 163-001, Sheets 1-3, Rev. 6. The S100 pipe overpack is constructed and assembled in accordance with Packaging Technology, Inc., Drawing No. 163-002, Sheets 1 and 2, Rev. 4. The S200 pipe overpack is constructed and assembled in accordance with Packaging Technology, Inc., Drawing No. 163-003, Sheets 1 and 2, Rev. 3. The S300 pipe overpack is constructed and assembled in accordance with Packaging Technology, Inc., Drawing No. 163-004, Sheet 1, Rev. 1. The compacted puck drum spacers needed for the purpose of maintaining subcriticality in 55-, 85-, and 100-gallon drums are constructed and assembled in accordance with Drawing No. 163-006, Rev. 0.

(b) Contents

(1) Type and form of material

Byproduct, source, and special nuclear material in the form of dewatered, solid or solidified materials and wastes. Materials must be packaged in one of the following payload containers: a 55-gallon drum, standard waste box (SWB), 85-gallon drum, standard pipe overpack, S100 pipe overpack, S200 pipe overpack, S300 pipe overpack, or 100-gallon drum. The payload containers are described in Section 2.9, "Payload Container/Assembly Configuration Specifications," of the CH-TRAMPAC, Rev. 2. Explosives, corrosives (pH less than 2 or greater than 12.5), nonradioactive pyrophorics, and compressed gases are prohibited. Within a payload container radioactive pyrophorics must not exceed 1 weight 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 the CH-TRAMPAC, Rev. 2. For payload configurations with an unvented heat-sealed bag layer, the absence of flammable gas mixtures is ensured as described in Appendix 6.13 of the CH-TRU Payload Appendices, Rev. 1.

(2) Maximum quantity of material per package

The package contents are limited to 7,600 pounds, including the weight of the payload containers and any other components of the payload assembly. The maximum gross weight for a payload container not to exceed the following:

(i) 328 pounds per 6-inch standard pipe overpack,
(ii) 547 pounds per 12-inch standard pipe overpack,
(iii) 550 pounds per S100 pipe overpack,
(iv) 547 pounds per S200 pipe overpack,
(v) 547 pounds per S300 pipe overpack,
(vi) 1,000 pounds per 100-gallon drum,
(vii) 1,000 pounds per 55-gallon drum,
(viii) 1,000 pounds per 85-gallon drum, or
(ix) 4,000 pounds per SWB.

5.(b)(2) Maximum number of payload containers per package and authorized packaging configurations as follows:

(i) 7 55-gallon drums,
(ii) 7 standard pipe overpacks,
(iii) 7 S100 pipe overpacks,
(iv) 7 S200 pipe overpacks,
(v) 7 S300 pipe overpacks,
(vi) 4 85-gallon drums,
(vii) 3 100-gallon drums, or
(viii) 1 SWB.

Fissile material not to exceed the limits specified in CH-TRAMPAC, Rev. 2, Section 3.1, "Nuclear Criticality."

The S100, S200, and S300 pipe overpack payloads shall meet the curie limits specified in CH-TRAMPAC, Rev. 2, Section 3.3, "Activity Limits."

Maximum decay heat per package not to exceed 30 watts. Decay heat per payload container not to exceed the values in Table 5.2-1 of the CH-TRAMPAC, Rev. 2, "List of Approved Alphanumeric 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 Section 5.2.3 of the CH-TRAMPAC, Rev. 2.

5. (c) Criticality Safety Index: 0.0

6. Physical form, chemical properties, chemical compatibility, configuration of waste containers and contents, isotopic inventory, fissile content, decay heat, weight and center of gravity; and radiation dose rate must be determined and limited in accordance with CH-TRAMPAC, Rev. 2.

7. Each payload container must be assigned to a shipping category in accordance with Section 5.1, "Payload Shipping Category" of CH-TRAMPAC, Rev. 2. Each payload container and payload assembly must not exceed the allowable wattage in accordance with Section 5.2.3, "Hydrogen Gas Generation Rate and Decay Heat Limits for Analytical Category," or must be tested for gas generation in accordance with Section 5.2.5, "Unified Flammable Gas Test Procedure," of CH-TRAMPAC, Rev. 2. 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 CH-TRAMPAC, Rev. 2, Section 6.2.4, "Mixing of Shipping Categories," and Appendix 2.4 of the CH-TRU Payload Appendices, "Mixing of Shipping Categories and Determination of the Flammability Index."

8. Payload containers within a package shall be selected in accordance with Section 6.0, "Payload Assembly Requirements" of CH-TRAMPAC, Rev. 2.

9. Each payload container must be vented in accordance with Section 2.5, "Filter Vents" of CH-TRAMPAC, Rev. 2. Drums which were not equipped with filtered vents during storage must be aspirated in accordance with Section 5.3, "Venting and Aspiration" of CH-TRAMPAC, Rev. 2.

10. For close-proximity and controlled shipments meeting the conditions specified in Appendices 3.5 and 3.6, respectively, of CH-TRU Payload Appendices, shipping periods of 20 days and 10 days may be applicable. The shipping period for any mode of transport is not to exceed 60 days. The content code LA 154 and SQ 154 are not authorized for loading and shipment in the HalfPACT packagings.

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.

    (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) All free standing water must be removed from the inner containment vessel cavity and the outer containment vessel cavity before shipment.

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

13. Expiration date: October 31, 2010.

14. Revision No. 3 of this certificate may be used until October 31, 2006.

REFERENCES

Washington TRU Solutions, LLC, October 4, 2004 and March 4, June 8, and August 19, 2005.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Robert A. Nelson, Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: 10/19/05

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
9280	2	71-9280	USA/9280/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)*

BWX Technologies
Nuclear Products Division
P.O. Box 785
Lynchburg, VA 24505-0785

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

BWX Technologies, Inc. application dated November 11, 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.: UBE-1

(2) Description

A steel drum for the transport of solid uranium and uranium-beryllium waste materials. The packaging is a 55-gallon, open-head steel drum with a minimum 18-gauge shell and bottom head, and a minimum 16-gauge closure lid. The lid is closed by a 12-gauge bolted locking ring with drop forged lugs, one of which is threaded, having a 5/8 inch bolt and nut. The closure includes a gasket. The gross weight of the package, including the maximum weight of contents, is approximately 600 pounds.

(3) Drawings

The packaging is constructed and assembled in accordance with Babcock & Wilcox Company Drawing. No. LP3023C, Rev. 4.

(b) Contents

(1) Type and form of material

Uranium and uranium-beryllium mixtures in the form of solids, and solid waste materials.

5.(b) Contents (continued)

(2) Maximum quantity of material per package

550 pounds. The uranium may be of any enrichment, and the beryllium may be present in any concentration. The maximum fissile mass is 100 grams U-235 per package, and the maximum average fissile mass density in the package is 0.5 gram U-235 per liter. Fission and activation products may be present, provided that the total quantity is less than 1×10^{-3} A_2 per package.

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

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

Maximum Fissile Mass Per Package (grams U-235 per package)	Minimum Transport Index
2.0	0.5
5.0	1.0
6.0	1.2
10.0	2.0
20.0	4.0
25.0	5.0
50.0	10.0
100.0	20.0

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

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

(b) Each packaging must be acceptance tested in accordance with the Acceptance Tests in Section 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: February 28, 2008.

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9280	2	71-9280	USA/9280/AF-85	3	OF	3

REFERENCES

BWX Technologies, Inc., application dated November 11, 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 February 6, 2003

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
9281	4	71-9281	USA/9281/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)*

BWX Technologies, Inc.
Nuclear Products Division
P.O. Box 785
Lynchburg, VA 24505-0785

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

BWX Technologies, Inc., application dated April 21, 2003.

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.: UBE-2

(2) Description

A steel drum for the transport of uranium and uranium-beryllium waste materials of solid form. The waste may be contained within compacted 55-gallon drums. The packaging is a 70-gallon, open-head steel drum with a minimum 18-gauge shell and bottom head, and a minimum 18-gauge closure lid. The lid is closed by a 12-gauge bolted locking ring with drop forged lugs, one of which is threaded, having a 5/8 inch bolt and nut. The closure includes a gasket. The gross weight of the package, including the maximum weight of contents, is approximately 1000 pounds.

(3) Drawings

The packaging is constructed and assembled in accordance with Babcock & Wilcox Company Drawing. No. LP3024C, Rev. 1.

(b) Contents

(1) Type and form of material

Uranium and uranium-beryllium waste of solid form. The waste may be contained within compacted 55-gallon drums.

(2) Maximum quantity of material per package:

950 pounds, including compacted secondary containers. The uranium may be of any enrichment, and the beryllium may be present in any concentration. The maximum fissile mass is 100 grams U-235 per package, and the maximum average fissile mass density in the package is 0.5 gram U-235 per liter. Fission and activation products may be present, provided that the total quantity is less than 1×10^{-3} A_2 per package.

(c) Criticality Safety Index

Maximum Fissile Mass Per Package (grams U-235 per package)	Criticality Safety Index
2.0	0.5
5.0	1.0
6.0	1.2
10.0	2.0
20.0	4.0
25.0	5.0
50.0	10.0
100.0	20.0

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

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

(b) Each packaging must be acceptance tested in accordance with the Acceptance Tests in Section 8 of the application.

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

8. Revision No. 3 of this certificate may be used until January 31, 2007.

9. Expiration date: May 31, 2008.

REFERENCES

BWX Technologies, Inc., application dated April 21, 2003.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Robert A. Nelson, Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: January 17, 2006

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
9282	1	71-9282	USA/9282/B(U)-96	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)*

Source Production
and Equipment Company, Inc.
113 Teal Street
St. Rose, LA 70087-9691

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Source Production and Equipment Company, Inc. application dated June 28, 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.: SPEC-300

(2) Description

The SPEC-300 is a radiographic device that consists of a source assembly, a depleted uranium shield, and a stainless steel enclosure. The radioactive source assembly is housed in a zircaloy or titanium "S" tube that is surrounded by the depleted uranium shield. The depleted uranium shield is secured in the stainless steel enclosure. The void space between the depleted uranium shield and the enclosure is filled with high density polyurethane foam. The package is approximately 26 inches long, 14 inches wide, and 15 inches high. The maximum gross weight of the package is 780 pounds.

(3) Drawings

The packaging is constructed and assembled in accordance with Source Production and Equipment Co., Inc. General Arrangement drawings: 19B000 sheets 1-8, Rev. 4 and B190700 sheet 1, Rev. 3.

(b) Contents

(1) Type and form of material

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

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9282	1	71-9282	USA/9282/B(U)-96	2 OF	3

5.(b) Contents (continued)

(2) Maximum quantity of material per package

300 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."

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

7. The name plate 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 shall be prepared for shipment in accordance with the Operating Procedures in Chapter 7.0 of the application, as supplemented; and

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

9. Packagings may be marked with Package Identification Number USA/9282/B(U)-85 until April 30, 2006, and must be marked with Package Identification Number USA/9282/B(U)-96 after April 30, 2006.

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

11. Expiration date: April 30, 2010.

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
9282	1	71-9282	USA/9282/B(U)-96	3	OF	3

REFERENCES

Source Production and Equipment Company, Inc., application dated June 28, 1999.

Supplements dated: October 6, November 4, November 22, and December 15, 1999; February 29 and March 27, 2000; and March 14, 2005.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Robert J. Lewis, Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: 28 April 2005

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
9283	2	71-9283	USA/9283/B(U)-96	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)*

QSA Global, Inc.
40 North Avenue
Burlington, MA 01803

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

AEA Technology/QSA Inc. application dated May 29, 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.

Packaging

(1) Model No.: OPL-660 and OP-660

(2) Description

The Model Nos. OPL-660 and OP-660 consist of a radiography camera within a protective container. The protective container is a 20 mm Cartridge Shipping and Storage Box fabricated according to military specification MIL-S-23389B. The protective container is of welded steel construction and is approximately 18½ inches long, 14½ inches high, and 8¼ inches wide. The protective container is fitted with foam and wood inserts and a lid that is secured by latches. The Model 660 series projector fits snugly in the center of the foam inserts within the protective container. The Model No. OPL-660 container has thin lead sheets to provide extra shielding at the ends and bottom. The maximum weight of the package is 88 pounds.

The Model 660 series projector is a radiography device. The projector's overall dimensions are approximately 12⅞ inches long, 5¼ inches wide, and 9⅝ inches high. The projector weighs a maximum of 56 pounds. The principal components of the 660 series projectors include an outer steel shell, polyurethane foam, a depleted uranium shield, an "S" tube, and end plugs. The sealed source contents are securely positioned in the "S" tube by a source cable locking device and shipping plug.

(3) Drawings

The packaging is constructed in accordance with the following AEA Technology QSA, Inc., Drawings:

R66050, Rev. C, Sheets 1 & 2, and R66060, Rev. A, Sheets 1-3.

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

(i) 140 Curies (output) for the Model No. 660B or 660BE projectors.

(ii) 120 Curies (output) for the Model No. 660, 660E, 660A or 660AE projectors.

Output curies are determined by measuring the source output at 1 meter and expressing its activity in curies derived from the following: 0.48 R/h-Ci Iridium-192 at 1 meter. (Ref: American National Standard N432-1980, "Radiological Safety for the Design and Construction of Apparatus for Gamma Radiography.")

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

7. The name plate 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 Test and Maintenance Program of Chapter 8.0 of the application, as supplemented; and

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

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

10. Packages may be marked with Package Identification Number USA/9283/B(U)-85 until July 31, 2007, and must be marked with Package Identification Number USA/9283/B(U)-96 after July 31, 2007

11. Revision No. 1 of this certificate may be used until July 31, 2007.

12. Expiration date: June 30, 2008.

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
9283	2	71-9283	USA/9283/B(U)-96	3	OF	3

REFERENCES

AEA Technology QSA, Inc., application dated May 21, 1998.

Supplements dated: June 15, 1998; March 6, 2003; and May 30, 2006.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

for

Christopher M. Regan, Acting Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: July 25, 2006

UNITED STATES NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

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, LLC
1802 Fairfax Road
Greensboro, NC 27407

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Eco-Pak Specialty Packaging application dated June 19, 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 No.: ESP-30X Protective Shipping Package for 30-inch UF_6 Cylinders

(2) Description

An overpack for the transport of 30-inch enriched uranium hexafluoride (UF_6) cylinders. The shape of the overpack is a right circular cylinder constructed of two 11 gauge carbon steel shells. The area between the shells is filled with fire retardant phenolic foam per ESP specification ESP-PF-1. The volume between the 1/2" inch thick end plates of the two shells is also filled with phenolic foam. A stepped horizontal joint permits the top half of the overpack to be removed from the base. The horizontal joint of each half of the overpack is covered with steel and a 5/8" thick silicone gasket seals the joint. The overpack halves are secured with ten 3/4" diameter steel bolts and nuts.

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

Outer shell inside diameter	43"
Outer shell length	96"
Inner shell inside diameter	30 7/8"
Inner shell length	82 5/8"
Overpack weight	2,955 pounds
30B Cylinder weight	1,390 pounds
UF_6 maximum load	5,020 pounds
Maximum package gross weight (including contents)	9,365 pounds

(3) Drawings

The packaging is constructed and assembled in accordance with ESP Drawing Nos.:

30X-1 SAR, Rev. 2, Sheets 1-4

5.(b) Contents

(1) Type and form of material

The UF_6 must be packaged in Model 30B UF_6 cylinders which have been fabricated, inspected, tested and maintained in accordance with the requirements of ANSI N14.1. The UF_6, which may contain either virgin or recycled uranium, must not contain more than the following maximum quantities of radionuclides and impurities:

U^{232}	5.0E-09 g/gU
U^{234}	2.0E-03 g/gU
U^{235}	5.0E-02 g/gU
U^{236}	2.5E-02 g/gU
U^{238}	balance of total uranium content
Pu and Np	Alpha activity not exceed 3.3 Bq/gU
Tc^{99}	5.0E-06 g/gU
Th^{228}	1.17E-09 g/gU

Fission Products 4.4 X 10^5 Mev Bq/d kgU (total contribution from gamma emitting fission products); this results in the following individual maximum activities:

Ru^{106}/Rh^{106}	2095 Bq/gU
Ru^{103}/Rh^{103}	885 Bq/gU
$Ce^{144}/Pr^{144}/Pr^{144}$	8349 Bq/gU
Sb^{125}	1030 Bq/gU
Cs^{134}	283 Bq/gU
Cs^{137}/Ba^{137}	778 Bq/gU
Zr^{95}	598 Bq/gU
Nb^{95}	574 Bq/gU

The total concentration of elements that form non-volatile fluorides (including Al, Ba, Bi, Cd, Co, Cr, Cu, Fe, Pb, Li, Mg, Mn, Ni, K, Ag, Na, Sr, Th, Sn, Zn, and Zr) must not exceed 3.0E-03 g/gU.

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The contents of other elements must not exceed the following concentrations in g/gU.

Sb<1	As<3	B<1	Bi<5	Cl<100
Cr<10	Nb<1	P<50	Ru<1	Si<100
Ta<1	Ti<1	Mo<1.4	W<1.4	V<1.4

Additionally, for reprocessed UF_6, the maximum total activity present in the package is limited to 957 mixture A_2 values.

(2) Maximum quantity of material per package

The package contents are limited to a maximum of 5,020 pounds of UF_6 enriched to not more than 5 wt%U^{235}. The maximum H/U atomic ratio for the UF_6 is 0.088.

5. (c) Criticality Safety Index 5.0

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

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

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

7. The 30-inch diameter UF_6 cylinder must be fabricated, inspected, tested and maintained in accordance with American National Standard N14.1-1995 or an earlier version of ANSI N14.1 in effect at the time of fabrication. 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.

8. 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.

9. The leak tightness of the 30B UF_6 cylinder shall be verified using a test having a sensitivity of at least 1 x 10^{-3} std-cc/sec per ANSI Standard N14.5-1997 prior to loading into the ESP-30X overpack.

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

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
9284	4	71-9284	USA/9284/B(U)F-85	4	OF	4

11. Expiration date: May 31, 2010.

REFERENCES

ESP application dated June 19, 1998.

Supplements dated: August 27, 1999; March 22, May 12, and May 18, 2000; April 11, 2002; January 28, and April 12, 2005.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Robert J. Lewis, Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date 20 April 2005

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
9285	2	71-9285	USA/9285/AF-85	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)*

Global Nuclear Fuel - Americas, L.L.C.
P.O. Box 780
Wilmington, NC 28402

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

General Electric Company application dated August 4, 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 No.: SRP-1

(2) Description

A steel drum for the transport of solid uranium contaminated residues. The packaging is a 55-gallon, open-head steel drum with a minimum 18-gauge shell and bottom head, and a minimum 16-gauge closure lid. The lid is closed by a 12-gauge bolted locking ring with drop forged lugs, one of which is threaded, having a 5/8 inch bolt and nut. The closure includes a gasket. The gross weight of the package, including the maximum weight of contents, is 825 pounds.

(3) Drawings

The packaging is constructed and assembled in accordance with General Electric Company Drawing. No. 0025E98, Rev. 1.

(b) Contents

(1) Type and form of material

Uranium-contaminated solid residues.

(2) Maximum quantity of material per package

775 pounds. The maximum uranium enrichment is 5.0 weight percent U-235. The maximum fissile mass is 104 grams U-235 per package, and the maximum average fissile mass density in the package is 0.5 gram U-235 per liter. In addition, the uranium may not exceed 0.05 weight percent U-234 and 0.025 weight percent U-236.

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

Minimum transport index to be shown on label for nuclear criticality control: 0.6

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 Section 7 of the application.

   (b) Each packaging must be acceptance tested in accordance with the Acceptance Tests in Section 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: October 31, 2008.

REFERENCES

General Electric Company application dated August 4, 1998.

Supplements dated: October 2, 1998, October 14, 1999, and August 6, 2003.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

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

Date: August 18, 2003

NRC FORM 618 (8-2000) 10 CFR 71	CERTIFICATE OF COMPLIANCE FOR RADIOACTIVE MATERIAL PACKAGES		U.S. NUCLEAR REGULATORY COMMISSION	
CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE / PAGES
9287	1	71-9287	USA/9287/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)*

Packaging Technology, Inc.
1102 Broadway Plaza, Suite 300
Tacoma, WA 98402-3526

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Packaging Technology, Inc., application dated November 18, 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 No.: SteriGenics Eagle

(2) Description

A stainless steel, lead shielded shipping cask for special form cobalt-60 sealed sources. The package consists of a cylindrical cask body with closure lid, and removable toroidal impact limiters, and a basket that carries and positions the cobalt-60 sealed source capsules. The packaging is constructed primarily of ASTM Type 304 stainless steel. The package is designed to transport up to 330,000 curies of cobalt-60.

The outside diameter of the cask body is approximately 37-11/16 inches. The diameter of the inner cavity is approximately 10-3/4 inches. The stainless steel inner shell has a minimum thickness of 1 inch and the stainless steel outer shell is 1 inch thick. The region between the two shells is filled with lead shielding. The closure lid and cask bottom end each consist of two stainless steel plates with lead between the two plates. The lead shielding thickness is approximately 10-3/8 inches on the side, 14-3/8 inches in the closure lid, and 11-7/8 inches on the cask bottom. The closure lid is secured by 12, 3/4-inch bolts. The closure lid is equipped with a Viton O-ring seal. The lid has a drain port and a vent port, and the cask body has a drain port. Each port is closed by a plug.

A double stainless steel thermal radiation shield is provided on the outside of the cask body in the region between the two impact limiters. The inner thermal shield is about 3/4-inches thick and is radially separated from the cask outer shell by 12 gauge spacers at each end. The outer shield is a sheet of 10 gauge material separated from the inner shield by a spiral wrap of 12 gauge wire.

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

The top and bottom impact limiters are toroidal stainless steel shells. They are attached to either end of the cask body using 12, 1-inch diameter ball-lock pins orientated radially around the cask body. One pin on each limiter is installed with a lockwire to provide a tamper-indicating device.

The cask lifting attachments thread into the upper cask body. The cask lid is also equipped with removable lid-lifting attachments. The cask rests on a steel pallet and is held down to the pallet by means of a steel frame placed on the top impact limiter. This steel frame is used to tie the cask to the conveyance. The maximum weight of the package, including contents is 20,000 lbs.

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

Cask Body Outer Diameter	37-11/16 inches
Cask Body Height	49-7/8 inches
Cask Cavity Inner Diameter	10-3/4 inches
Cask Cavity Inner Height	19 inches
Lead Shield Sidewall Thickness	10-3/8 inches
Overall Package Dimension	
Diameter at Impact Limiters	60 inches
Diameter at Body	37-11/16 inches
Height with Impact Limiters	76 inches
Maximum Contents Weight	50 pounds
Maximum Package Weight (Including Contents)	20,000 pounds

(3) Drawings

The packaging is constructed and assembled in accordance with Packaging Technology, Incorporated, Drawing No. 98003-SAR, Rev.1, Sheets 1 through 8.

(b) Contents

(1) 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:

330,000 curies. Not to exceed 18,400 curies per special form source.

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
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6. 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 Chapter 7.0 of the application, as supplemented.

   (b) Each packaging must meet the Acceptance Tests and Maintenance Program of Chapter 8.0 of the application, as supplemented.

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

8. Expiration date: December 31, 2009.

REFERENCES

Packaging Technology, Inc., application dated November 18, 1998.

Supplements dated: August 20, 1999, and November 29, 2004.

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 21, 2004

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE	PAGES
9288	7	71-9288	USA/9288/B(U)F-96	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)*
Columbiana Hi Tech, LLC
1802 Fairfax Road
Greensboro, NC 27407

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION
Columbiana Hi Tech, LLC, consolidated application dated February 27, 2006, 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.: CHT-OP-TU

(2) Description

A shipping container for uranium oxide pellets, powder, and uranium-bearing materials. The package is roughly cubical and is approximately 45-inches x 45-inches x 62-inches high. The package has four internal sleeves in which Oxide Vessels are inserted.

The outer shell of the package is constructed of 11-gauge mild or stainless steel and the space between the outer shell and the sleeves are filled with fire retardant, closed cell phenolic or polyurethane foam.

The sleeves are constructed of 11-gauge mild or stainless steel with an inner diameter of 10-1/4 inches. The sleeves are closed with twelve 1/2-inch-diameter bolts using an outer lid assembly on a 1/16-inch-thick neoprene or silicone gasket. The outer lid assembly is filled with fire-retardant, closed cell phenolic or polyurethane foam.

The Oxide Vessel is constructed of series 300 stainless steel, with an inner diameter of either 6, 7.5, or 8 inches. The Oxide Vessel is closed by eight 1/2-inch-diameter bolts on a 5/8-inch-thick stainless steel lid with a double O-ring seal. The O-ring seal material is either silicon rubber, fluorosilicon or fluorocarbon (viton).

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

Sleeve inside diameter	10 1/4-inches
Oxide Vessel inside diameter	6, 7.5, or 8 inches
Oxide Vessel inside height	40 3/4-inches
Overall package dimensions	
width	45 inches
length	45 inches
height	62 inches
Maximum contents weight per Oxide Vessel	402 pounds
Maximum empty transport weight including four empty Oxide Vessels	2576 pounds
Maximum loaded package weight (with four filled Oxide Vessels)	3757 pounds

(3) Drawings

The packaging is constructed and assembled in accordance with Columbiana Hi Tech Drawing Nos.:

OP-TU-SAR, Rev. 12, Sheets 1 of 2 and 2 of 2;
OP-TU-A2, Rev. 12, Sheet 1 of 1;
OP-TU-A3, Rev. 12, Sheet 1 of 1;
OP-TU-A4, Rev. 12, Sheet 1 of 1; and,
OPTU-V-AB1, Rev. 8, Sheets 1 of 2 and 2 of 2.

5.(b) Contents

(1) Type and form of material

Uranium-bearing compounds in solid form, heterogenous or homogenous (i.e., pellets and powder). The contents may include up to 1000 grams of polyethylene or other plastics as packaging, waste or impurities per Oxide Vessel (4000 grams per package), provided that: (1), the total water equivalent of the plastic is less than 1307 grams per Oxide Vessel (5228 grams per package); and, (2) the decay heat is less than 0.068 W/m^3. Materials with a decay heat greater than 0.068 W/m^3 may not be packaged using hydrogen bearing plastics, and may only use non-hydrogen bearing plastics such as Teflon™ (polytetrafluoroethylene or PTFE) or metallic containers. In addition, the contents are limited to:

A. Unirradiated uranium oxide powder enriched to no more than 5.0 weight percent in the U-235 isotope.

B. Unirradiated uranium oxide pellets or a mixture of pellets and powder enriched to no more than 5.0 weight percent in the U-235 isotope.

C. Reprocessed uranium oxide powder enriched to no more than 5.0 weight percent in the U-235 isotope, with limits specified in Table 1.

Table 1: Allowable Content for Shipment of Reprocessed Uranium Oxide

	Maximum Content		
Isotope	Type A	Type B Level I	Type B Level II
U-232 (g/gU)	Mixtures of isotopes shall be evaluated and designated as a Type A quantity per 10 CFR Part 71 Appendix A. The maximum enrichment per package is 5 weight per cent ^{235}U.	2.00E-09	5.00E-09
U-234 (g/gU)		2.00E-03	2.00E-03
U-235 (g/gU)		5.00E-02	5.00E-02
U-236 (g/gU)		2.50E-02	2.50E-01
Np-237 (g/gU)		1.66E-06	5.00E-03
Pu-238 (g/gU)		6.20E-11	4.00E-08
Pu-239 (g/gU)		3.04E-09	3.04E-09
Pu-240 (g/gU)		3.04E-09	6.00E-09
Gamma Emitters (MeV-Bq/kgU)		6.38E+05	1.91E+06

D. Reprocessed uranium oxide pellets or a mixture of pellets and powder enriched to no more than 5.0 weight percent in the U-235 isotope, with the limits specified in Table 1.

E. Homogeneous (powder or crystalline form) uranium-bearing materials enriched to 5.0 weight percent in the U-235 isotope in the form of solids, or solidified or dewatered materials.

Uranium compounds must have a ratio of non-fissile atoms to uranium atoms greater than two (2) and the density of these compounds is less than 10.96 g/cm^3 (density of UO_2). Material such as U-metal, U-metal alloys, or uranium hydrides (e.g., UH_X) may not be shipped. Uranium-bearing materials may include oxides, carbides, silicates or other compounds of uranium. Uranium-bearing materials may be moderated by graphite to any degree. Compounds may be mixed with other non-fissile materials with the exception of beryllium or hydrogenous material enriched in deuterium. Materials with a hydrogen density greater than water must be excluded, except for the allowance provided by Condition No. 5.(b)(1).

F. Heterogeneous (pellets or previously pelletized materials) uranium-bearing materials enriched to 5.0 weight percent in the U-235 isotope in the form of solids, or solidified or dewatered materials.

Uranium compounds must have a ratio of non-fissile atoms to uranium atoms greater than two (2) and the density of these compounds is less than 10.96 g/cm^3 (density of UO_2). Material such as U-metal, U-metal alloys, or uranium hydrides (e.g., UH_X) may not be shipped. Uranium-bearing materials may include oxides, carbides, silicates or other compounds of uranium. Uranium-bearing materials may be moderated by graphite to any degree. Compounds may be mixed with other non-fissile materials with the exception of beryllium or hydrogenous material enriched in deuterium. Materials with a hydrogen density greater than water must be excluded, except for the allowance provided by Condition No. 5.(b)(1).

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

For the contents described in Condition Nos. 5.(b)(1)(A), 5.(b)(1)(C), and 5.(b)(1)(E), the maximum quantity of material is 402 pounds of uranium compounds per 8-inch, 7.5-inch, or 6-inch diameter Oxide Vessel and a maximum load of 1608 pounds per package.

For the contents described in Condition Nos. 5.(b)(1)(B), 5.(b)(1)(D), and 5.(b)(1)(F), the maximum quantity of material is 402 pounds of uranium compounds per 7.5-inch or 6-inch diameter Oxide Vessel and a maximum load of 1608 pounds per package. The 8-inch diameter Oxide Vessel is not authorized for these contents.

The maximum allowable contents heat generation rate is 1.0 BTU/hr/ft^3 (10.3 W/m^3).

5.(c) Criticality Safety Index 2.0

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

(a) The package must be prepared for shipment and operated in accordance with the Operating Procedures in Section 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 Section 8 of the application, as supplemented.

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

8. Packagings may be marked with Package Identification Number USA/9288/B(U)F-85 until April 30, 2007, and must be marked with Package Identification Number USA/9288/B(U)F-96 after April 30, 2007.

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
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9. Transport by air of fissile material is not authorized.

10. Revision No. 6 of this certificate may be used until April 30, 2007.

11. Expiration date: March 31, 2010.

REFERENCES

Columbiana Hi Tech, LLC, consolidated application dated February 27, 2006.

Supplement dated April 10, 2006.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

For

Robert A. Nelson, Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date 4/24/06

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
9289	3	71-9289	USA/9289/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)*

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 1, 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 No.: WE-1

(2) Description

A fresh fuel assembly shipping container. The package has two shipping configurations: one for shipping a single BW 17x17 fuel assembly composed of uranium dioxide pellets within zircalloy cladding; and the other for shipping up to 48 Pathfinder fuel assemblies within a steel canister which functions as a secondary containment vessel. The package consists of a cylindrical outer container and a rectangular inner container bolted to a strongback. The outer container is constructed of 11 gauge carbon steel and opens into two semi-cylindrical halves. The inner container is comprised of 1-inch thick carbon steel plates that are bolted together. The inner container is secured to the strongback by bolts and clamp arms. Wood blocks surround the region between the inner container and the strongback. The strongback is supported by 14 shock mounts attached to the outer container.

<u>For BW 17x17 Fuel Shipment Configuration:</u>

The BW 17x17 fuel assembly shipment configuration consists of the fuel assembly placed into the inner container. The fuel assembly is surrounded by thermal insulation and secured inside the inner container with nine integral clamp frames.

5. (2) Description (Continued)

<u>For Pathfinder Fuel Shipment Configuration</u>:

Pathfinder Fuel shipment configuration consists of the Pathfinder fuel in the Pathfinder Canister, which is placed into the inner container. The Pathfinder Canister is a sealed cylindrical canister which houses up to 48 Pathfinder fuel assemblies. Wood dunnage or empty sheaths may be used to fill empty spaces in the canister. The canister is made of austenitic stainless steel and has a welded body and a bolted closure lid. The Pathfinder Canister is surrounded by thermal insulation, and secured inside the inner container with five integral clamp frames. The clamp frames, which consist of bolted clamp arms, are bolted to the inner rectangular container. Wood blocks surround both ends of the Pathfinder Canister. A stainless steel spacer tube is used to fill the space between the Pathfinder Canister and the inner container.

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

Inner container length	165 inches
Inner container width (sq)	16 ½ inches
Outer container length	216 inches
Outer container diameter	44 inches
Maximum content weight	1610 pounds
Maximum package weight (including contents)	9090 pounds

(3) Drawings

The packaging is constructed in accordance with the following Framatome Cogema Fuels Drawing Nos.:

1273964, Rev. 0
1273965, Rev. 1
1273966, Rev. 0
1273967, Rev. 0
1273968, Rev. 0

The Pathfinder Canister Configuration is constructed in accordance with the following Framatome ANP Drawing Nos.:

5016270, Rev. 1
5021426, Sheets 1 and 2, Rev. 0.

(b) Contents

(1) Type and form of material

(i) For BW 17x17 Fuel Shipment Configuration:

A fuel assembly composed of uranium dioxide pellets within zircalloy cladding. The fuel assembly has the following specifications:

Assembly type	BW 17x17
No. fuel rods	264
No. non-fuel tubes	25
Nominal fuel rod pitch, in.	0.496
Maximum fuel pellet OD, in.	0.3232
Nominal clad OD, in.	0.374
Nominal clad thickness, in.	0.022
Nominal guide and instrument tube OD, in.	0.48
Nominal guide and instrument tube ID, in.	0.452
Nominal active fuel length, in.	144
Maximum uranium enrichment, weight percent U-235	4.6
Maximum U-235 mass, kg	22.14

(ii) For Pathfinder Fuel Shipment Configuration:

An unirradiated fuel assembly composed of six fuel pins clustered around a center absorber pin in a hexagonal pattern. The fuel pins consist of uranium dioxide pellets inside Incoloy 800 cladding. The absorber pin consists of Incoloy 800 cladding with or without poison material. Fuel pins and absorber pins are separated by spacer wires and enclosed in a cylindrical sheath made of stainless steel, incoloy or incoloy alloy. The fuel assembly has the following specifications:

Assembly type	Pathfinder
No. fuel pins per assembly	6
No. non-fuel pins per assembly	1
Maximum uranium enrichment, weight percent U-235	7.51
Maximum uranium mass per assembly, kg U	2.2281

5. (b) Contents (continued)

Maximum UO_2 density, g/cm^3	10.61
Fuel pellet outer diameter (OD), in.	0.207 ± 0.0005
Nominal active fuel length, in.	72.0
Minimum clad OD, in.	0.246
Maximum clad inner diameter (ID), in.	0.212
Nominal center-to-center pin pitch, in.	0.289
Nominal sheath ID, in.	0.945
Nominal sheath OD, in.	1.00

(2) Maximum quantity of material per package

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

One BW 17x17 fuel assembly contents, not to exceed 1610 pounds. The radioactive material may not exceed any of the following limits:

U-232	0.01 microgram per gram of uranium
U-234	0.001 gram per gram of uranium
U-236	0.013 gram per gram of uranium
Tc-99	5 micrograms per gram of uranium
Fission Products	4.4 x 10^5 MeV-Becquerel per kilogram of uranium
Np and Pu	35 Becquerels per gram of uranium

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

Up to 48 unirradiated Pathfinder fuel assembles inside a Pathfinder Canister. The weight of the fully loaded canister not to exceed 800 pounds.

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

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

6. 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 of Chapter 7 of the application, as supplemented.

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

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: February 28, 2009.

REFERENCES

Framatome ANP, Inc. application dated: May 1, 2002.

Supplement dated: November 12, 2002, and January 8, 2004.

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: January 26, 2004

UNITED STATES 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
9290	3	71-9290	USA/9290/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)*

MDS Nordion
447 March Road
Kanata, Ontario
Canada K2K 1X8

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

MDS Nordion application dated February 20, 2003, 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. F-430/GC-40 Transport Package

(2) Description

The Model No. F-430/GC-40 Transport package is designed to transport MDS Nordion's Gammacell-40 (GC-40) irradiator containing cesium-137 sealed sources in special form. The F-430 overpack provides impact and thermal protection for the radioactive contents. Containment is provided by the special form sealed source and shielding is provided by the GC-40 irradiator body.

The F-430 is stainless steel cylindrical package with a 50" diameter and a height of 50" that is placed on a removable mild steel skid. The maximum weight of the package is 7000 pounds. The maximum weight of the GC-40 contents is 4000 pounds.

The overpack consists of nested cylindrical shells. The shells are made from stainless steel and the volume between the shells is filled with rigid foam. This foam provides insulation during an accidental fire. Vent holes, plugged with material designed to melt in a fire, are provided between the shells to prevent pressure buildup and allow a pathway for escape of gases from foam during an accidental fire.

The package contents consists of a Cesium-137 sealed source contained within an MDS Nordion GC-40 irradiator (upper or lower heads). The GC-40 is a research irradiator with lead shielding and a lead filled source drawer.

5.(a)(2) (continued)

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

Package outside diameter	50 inches
Package height	50 inches
Cavity diameter	36 inches
Cavity height	35.25 inches
Removable skid	50 inches x 50 inches x 8 inches (height)
Overpack weight	2640 pounds
Contents weight	4000 pounds
Maximum package weight	7000 pounds

(3) Drawings

The packaging is constructed in accordance with the MDS Nordion drawings F643001-001, Rev. K, Sheet 1 of 2, and F643001-001, Rev. D, Sheet 2 of 2.

(b) Contents

(1) Type and form of material

Cesium-137 as a sealed source which meet the requirements of special form radioactive material.

(2) Maximum quantity of material per package

2,000 Curies.

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

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
9290	3	71-9290	USA/9290/B(U)-85	3	OF	3

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

8. Revision No. 2 of this certificate may be used until February 28, 2007.

9. Expiration date: February 28, 2007.

REFERENCES

MDS Nordion application dated February 20, 2003.

Supplements dated: July 21, August 25, and December 18, 2003; January 16, July 16, July 21, and July 23, 2004; and April 21, 2006.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

For

Robert A. Nelson, Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: May 24, 2006

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
9291	6	71-9291	USA/9291/B(U)F-96	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, LLC
1802 Fairfax Road
Greensboro, NC 27407

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Columbiana Hi Tech, LLC, consolidated application dated February 17, 2006, 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.: Liqui-Rad (LR) Transport Unit Package

(2) Description

The LR Package is designed to transport Type B quantities of fissile uranyl nitrate solutions. The package uses thermal and impact limiting systems to protect the containment vessel and prevent the contents from being released. The primary structural components of the LR packaging consist of a stainless steel containment vessel, a carbon steel outer vessel and a carbon steel framing system. The containment vessel is built in accordance with ASME Pressure Vessel Code (Section VIII, Division 1) but does not require an ASME stamp. Double O-ring seals on the containment vessel's primary and secondary lids provide a leak tight seal which is leak testable. A closed-cell phenolic foam or polyurethane foam surrounds the top and bottom head area of the containment vessel and ceramic fiber blanket and board insulation are used in the sidewalls and outer lid for thermal insulation and impact absorption. The maximum volume of the contents is limited to 230 gallons which maintains a minimum ullage of 33 gallons.

The LR is a cylindrical package set in a rectangular angle frame. The dimensions of the package are approximately 56"(l) x 56"(w) x 73"(h). The maximum weight of the package is 5692 pounds. The outer vessel is constructed of 10 gauge carbon steel. The containment vessel is constructed of 1/4 inch stainless steel with 1/4 inch thick flanged and dished heads. The containment vessel is rated at 50 psig pressure. Closed-cell phenolic or polyurethane foam and ceramic fiber insulation are sandwiched between the containment vessel and the package's outer shell.

The package is designed to be leak-tight (maximum allowable leakrate of 1×10^{-7} ref-cm^3/sec). The containment vessel is closed using a double O-ring and is secured by sixteen 5/8 inch stainless steel studs. The outer lid is closed with four 5/8 inch

5.(a)(2) Continued

stainless steel bolts and nuts. The package is also equipped with plastic plugs to vent any gases that may be generated by the insulation during a fire event. All valves and fittings are provided within sealed enclosures to contain any leakage during valve failure.

(3) Drawings

The packaging is constructed and assembled in accordance with Columbiana Hi Tech Drawing Nos. LR-SAR, Sheets 1 through 4, Rev. 7.

5.(b) Contents

(1) Type and form of material

Low enriched Uranyl Nitrate solutions with the specifications shown in Table 1 below. The uranium concentration must be less than or equal to 125 gU/liter with an enrichment less than or equal to 5.0 wt% U-235. Non-fissile chemical impurities may be present up to the chemical impurity specification in Table 1. Additionally, fissile isotopes are also limited to the quantities in Table 1.

(2) Maximum quantity of material per package

230 gallons of Uranyl Nitrate solution with limits as shown in table 1.

Table 1

ITEM	SPECIFICATION
Solution Density	≤1.17 g/cc
Chemical Impurities	≤1500 μg/gU
Nitric Acid Normality	0.1 - 0.7
Uranium Concentration	≤125 gU/l
U-232	≤2.0E-03 μg/gU
U-234	≤2.0E+03 μg/gU
U-235	≤0.05 g/gU (12 pounds maximum quantity of U-235 per LR)
U-236	≤2.5E+04 μg/gU
U-238	remainder of uranium
Pu/Np Alpha Activity	≤93 Bq/gU
Gamma Emitters	0.515E-01 Ci

5. (c) Criticality Safety Index 0.0

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

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

   (b) Each packaging must be acceptance tested and maintained in accordance with 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 71.17.

8. Packagings may be marked with Package Identification Number USA/9291/B(U)F-85 until March 31, 2007, and must be marked with Package Identification Number USA/9291/B(U)F-96 after March 31, 2007.

9. Transport by air of fissile material is not authorized.

10. Revision No. 5 of this certificate may be used until August 31, 2007.

11. Expiration date: October 31, 2011.

REFERENCES

Columbiana Hi Tech, LLC, consolidated application dated February 17, 2006.

Supplement dated: July 25, 2006

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Christopher M. Regan, Acting Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: August 3, 2006

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
9292	5	71-9292	USA/9292/AF-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)*

Westinghouse Electric Company, LLC
P.O. Box 355
Pittsburgh, PA 15230-0355

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Westinghouse Electric Company, LLC application dated September 16, 2004, 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.: PATRIOT

(2) Description

A shipping container for unirradiated fuel assemblies. The package consists of a right rectangular metal inner container and a wooden outer container, with cushioning material between the inner and outer containers.

There are two versions of the metal inner container. Both versions measure approximately 11-1/4 inches high by 18-1/8 inches wide by 182 inches long. There are two channel sections within the inner container, and each channel section holds one BWR fuel assembly. The inner container is equipped with a lid and an end cap that are closed by 18 bolts and fastening lugs. The overall dimensions of the wooden outer container are approximately 30-1/4 inches wide by 31-1/4 inches high by 207-3/4 inches long. The cushioning material between the inner and outer containers is phenolic impregnated honeycomb and ethafoam. The inner container may be positioned on a series of vibration dampers mounted on the inside bottom of the wooden outer container.

The maximum weight of the package, including contents, is 2,988 pounds with the version #1 inner container and 2,964 pounds with the version #2 (optional) inner container.

5.(a)(3) Drawings

The packaging is constructed and assembled in accordance with Westinghouse Electric Company, LLC Drawing Nos.:

10014E27, Rev. 1,
10014E28, Sheets 1 and 2, Rev. 2,
10015E58, Sheets 1 and 2, Rev. 2

5.(b) Contents

(1) Type and form of material

The package is designed to hold two unirradiated BWR fuel assemblies, comprised of UO_2 fuel rods in a 10 x 10 square array. The fuel cross-sectional area is 25 square inches.

(i) Description of Assembly Type #1:

Each assembly is made up of 96 full-length fuel rods having a maximum active fuel length of 150 inches. The fuel pellet diameter is 0.819 ± 0.002 cm, encapsulated in 0.063 cm zirconium alloy cladding. There is a 0.0085 cm gap between the pellets and the cladding. The maximum U-235 enrichment of any fuel rod is 5.0 weight percent. Each assembly contains water holes in the four center rod positions of the assembly. Three different fuel package loadings have the following specifications:

(A) Maximum average U-235 enrichment is 4.0 weight percent within any axial zone of the assembly; Maximum U-235 content is 3.25 weight percent of any gadolinia-urania rod or axial zone of any gadolinia-urania fuel rod; Maximum number of fuel rods per assembly containing 5.0 weight percent U-235 enriched pellets is 36; Maximum U-235 enrichment is 4.0 weight percent for all edge rods, and 3.5 weight percent for all corner rods; Each assembly must include at least eight fuel rods with a minimum gadolinia content of 2.5 weight percent in all axial regions with enriched pellets. The eight gadolinia rods are arranged with two rods in each quadrant of the fuel assembly. The two gadolinia rods within each quadrant must be symmetric about the geometric diagonal of the fuel assembly, and must not be in an edge or corner rod location. Other fuel rods containing gadolinia may be present.

(B) Maximum average U-235 enrichment is 4.725 weight percent within any axial zone of the assembly; Maximum U-235 content is 4.2 weight percent of any gadolinia-urania rod or axial zone of any gadolinia-urania fuel rod; Maximum number of fuel rods per assembly containing 5.0 weight percent U-235 enriched pellets is 52; Maximum U-235 enrichment is 4.5 weight percent for all edge rods, and 4.0 weight percent for all corner rods; Each assembly must include at least eight fuel rods with a minimum gadolinia content of 5.3 weight percent in all axial regions with enriched pellets. The eight gadolinia rods are arranged with two rods in each quadrant of the fuel assembly. The two gadolinia rods within each quadrant must be symmetric about the geometric diagonal of the fuel assembly, and must not be in an edge or corner rod location. Other fuel rods containing gadolinia may be present.

.(b) Contents (continued)

(C) Maximum average U-235 enrichment is 4.858 weight percent within any axial zone of the assembly; Maximum U-235 content is 4.2 weight percent of any gadolinia-urania rod or axial zone of any gadolinia-urania fuel rod; Maximum number of fuel rods per assembly containing 5.0 weight percent U-235 enriched pellets is 80; Maximum U-235 enrichment is 4.0 weight percent for all corner rods; Each assembly must include at least twelve fuel rods with a minimum gadolinia content of 2.4 weight percent in all axial regions with enriched pellets. The twelve gadolinia rods are arranged with three rods in each quadrant of the fuel assembly. The three gadolinia rods within each quadrant must be symmetric about the geometric diagonal of the fuel assembly, and must not be in an edge or corner rod location. Other fuel rods containing gadolinia may be present.

(ii) Description of Assembly Type #2

Each assembly is made up of 96 fuel rods having a maximum active fuel length of 150 inches. Each assembly contains four one-third length fuel rods and eight two-thirds length fuel rods. The four one-third length fuel rods are located on the outside corners of the assembly. The eight two-thirds length fuel rods, arranged as two rods in each quadrant of the assembly, are located symmetric to the geometric diagonal, toward the center of the assembly. The fuel pellet diameter is 0.848 cm nominal, encapsulated in 0.061 cm nominal zirconium alloy cladding. There is a 0.0075 cm gap between the pellets and the cladding. The maximum U-235 enrichment of any fuel rod is 5.0 weight percent. Each assembly contains water holes in the four center rod positions of the assembly. The fuel assembly must be transported in channels. The specifications for each one-third length axial section of the fuel assembly are as follows:

(A) Upper section must contain 84 fuel rods, arranged as 21 rods per quadrant. Maximum U-235 enrichment of any rod is 5.0 weight percent. This section of the assembly must include at least eight fuel rods with a minimum gadolinia content of 4.0 weight percent in all axial regions with enriched pellets. The eight gadolinia rods are arranged with two rods in each quadrant of the fuel assembly, arranged symmetrically along the geometric diagonal of the assembly, and must not be in an edge or corner rod location. The section must contain 12 water holes, arranged as three water holes in each quadrant of the assembly. One of the three water holes within each quadrant must be located on the outside corner location of the assembly, and the other two water holes must be located on the geometric diagonal of the fuel assembly. Other fuel rods containing gadolinia may be present.

(B) Middle section must contain 92 fuel rods, arranged as 23 rods per quadrant. Maximum U-235 enrichment of any rod is 5.0 weight percent. This section of the assembly must include at least ten fuel rods with a minimum gadolinia content of 4.0 weight percent in all axial regions with enriched pellets. The ten gadolinia rods must be arranged symmetrically along the geometric diagonal of the assembly, and must not be in an edge or corner rod location. The section must contain four water holes, arranged as one water hole in each quadrant of the assembly. Each water hole within each quadrant must be located on the outside corner location of the assembly. Other fuel rods containing gadolinia may be present.

(C) Lower section must contain 96 fuel rods, arranged as 24 rods per quadrant. Maximum U-235 enrichment of any rod is 5.0 weight percent. This section of the assembly must include at least twelve fuel rods with a minimum gadolinia content of 4.0 weight percent in all axial regions with enriched pellets. The twelve gadolinia rods must be arranged symmetrically along the geometric diagonal of the assembly, and must not be in an edge or corner rod location. Other fuel rods containing gadolinia may be present.

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

Two fuel assemblies. The total weight of contents not to exceed 1,320 pounds.

.(c) Criticality Safety Index: 1.0

. 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.

. For the contents described in 5.(b)(1)(i), polyethylene inserts may be positioned between rods within the fuel assemblies. The quantity of polyethylene must not exceed 18.33 g polyethylene per centimeter length of the fuel assembly, and must not exceed a total of 6.99 kg per fuel assembly. The polyethylene may be borated. No polyethylene inserts may be used for the contents described in 5.(b)(1)(ii).

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.

For packagings fabricated in accordance with Drawing No. 10015E58, Rev. 1 (referred to as version #2 inner containers), only Serial Nos. 001 through 039, inclusive, are authorized for use.

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

. Revision No. 4 of this certificate may be used until August 31, 2007. Revision No. 3 of this certificate may be used until January 31, 2007.

. Expiration date: August 31, 2010.

REFERENCES

Vestinghouse Electric Company, LLC consolidated application dated: September 16, 2004.

upplements dated: April 14, June 14, August 9, and September 22, 2005; January 6, and May 13, 2006.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Christopher Regan, Acting Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

ate: August 10, 2006

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE	PAGES
9293	2	71-9293	USA/9293/B(U)F-85	1 OF	9

2. PREAMBLE

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

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

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

   a. ISSUED TO *(Name and Address)*

   Transnuclear, Inc.
   7135 Minstrel Way
   Columbia, MD 21045

   b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

   Transnuclear, Inc., application dated May 19, 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. Packaging

   (1) Model No. TN-68 Transport Package

   (2) Description

   The TN-68 is predominantly a steel package that is used to transport up to 68 intact BWR fuel assemblies with or without channels. The overall dimensions of the package are 271 inches long and 144 inches in diameter with the impact limiters installed.

   The package generally consists of four components, the fuel basket assembly, a containment vessel within a forged steel cask body, a radial neutron shield, and impact limiters.

   The basket assembly locates and supports the fuel assemblies, transfers heat to the cask body wall and provides neutron absorption to satisfy sub-criticality requirements. The basket structure consists of an assembly of stainless steel cells, joined by fusion welding of 1.75 inch wide stainless steel plates. Above and below the plates are slotted borated aluminum (or boron carbide/aluminum) metal matrix composite neutron poison plates which form an egg-crate structure. This construction forms a honey-comb like structure of cell liners which provides compartments for 68 fuel assemblies. The nominal dimensions of each cell is 6.0 inches x 6.0 inches.

   A thick-walled (6.0 inch), forged steel cask body for gamma shielding surrounds the containment vessel, by an independent shell and bottom plate of carbon steel. The gamma shield completely surrounds the containment vessel inner shell and bottom closure. The thickness of the bottom of the cask body is 8.25 inches. A 4.5 inch thick steel gamma shield is also welded to the inside of the containment lid.

5.(a)(2) continued

The containment boundary consists of the inner shell and bottom plate, shell flange, lid outer plates, lid bolts, penetration cover plate and bolts and the inner metallic O-rings of the lid seal and the two lid penetrations (vent and drain). The containment vessel length is approximately 189 inches with a wall thickness of 1.5 inches. The cylindrical cask cavity has a nominal diameter of 69.5 inches and a length of 178 inches. The containment lid is 5 inches thick and is fastened to the cask body with 48 bolts. Double metallic O-ring seals are provided for lid closure. To preclude air in-leakage, the cask cavity is pressurized with helium to above atmospheric pressure. There are two penetrations through the containment vessel which are located in the lid. These penetrations are for draining and venting. Double metallic seals are also used on these two lid penetrations. The OP port provides access to the interspace lid seals for leak testing purposes. The OP transport cover is not part of the containment boundary.

Neutron shielding is provided by a borated polyester resin compound surrounding the gamma shield. The resin compound is cast into long, slender aluminum containers. The total thickness of the resin and aluminum is approximately 6 inches. The array of resin-filled containers is enclosed within a smooth 0.75 inch outer steel shell constructed of two half cylinders.

The package has impact limiters at each end of the cask body. The impact limiters consist of balsa wood and redwood blocks, encased in sealed stainless steel shells that maintain the wood in a dry atmosphere and provide wood confinement when crushed during a free drop. The impact limiters have internal radial gussets for added strength and confinement. The impact limiters are attaching to each other using 13 tie rods and to the cask by eight bolts attaching to brackets welded to the outer shell in eight locations (four bolting locations per impact limiter).

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

Overall length (with impact limiters, in)	271
Overall length (without impact limiters, in)	197
Impact Limiter Outside diameter, (in)	144
Outside diameter (without impact limiters, in)	98
Cavity diameter (in)	69.5
Cavity length (in)	178
Containment shell thickness (in)	1.5
Containment vessel length (in)	184
Body wall thickness (in)	7.5
Containment lid thickness (in)	5
Overall lid thickness (in)	9.5
Bottom thickness (in)	9.75
Resin and aluminum box thickness (in)	6
Outer shell thickness (in)	0.75
Overall basket length (in)	164
Maximum weight of package (pounds)	272,000
Maximum weight of BWR fuel contents (pounds)	47,900
Maximum weight of impact limiters and attachments (pounds)	32,000

5.(a)(3) Drawings

The package is constructed and assembled in accordance with TN drawings:

972-71-1, Revision 1
972-71-2, Revision 2
972-71-3, Revision 4
972-71-4, Revision 2
972-71-5, Revision 1
972-71-6, Revision 1
972-71-7, Revision 3
972-71-8, Revision 2
972-71-9, Revision 2
972-71-10, Revision 1
972-71-11, Revision 1
972-71-12, Revision 0
972-71-13, Revision 0
972-71-14, Revision 1

(b) Contents

(1) Type and form of material

Contents are limited to 68 unconsolidated intact irradiated GE BWR fuel assemblies with zircalloy cladding. An intact fuel assembly is a spent nuclear fuel assembly without known or suspected cladding defects greater than pinhole leaks or hairline cracks. Partial fuel assemblies (i.e. spent 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 equal to that displaced by the original rod(s).

Spent nuclear fuel may be transported with or without channels. Any fuel channel thickness up to 0.120 is acceptable on any of the fuel designs shown below. The maximum initial rod pressurization is 155 psig. The maximum fuel assembly length is 176.2 inches and the maximum fuel assembly width is 5.44 inches.

Permissible fuel assemblies are limited as stated in table 1 (fuel types may be C, D, or S lattice):

5.(b)(1) continued

Table 1, Fuel characteristics

GE fuel generation	model	array	rod pitch	fuel rods	rod od	clad thick	pellet dia.	water rods	water rod od	water rod id	U content (MTU/ Assembly)	Max active fuel length
2A	2a	7x7	0.738	49	0.570	0.036	0.488	0	x	x	0.1977	144
2, 2B	2	7x7	0.738	49	0.563	0.032	0.487	0	x	x	0.1977	144
3, 3A, 3B	3	7x7	0.738	49	0.563	0.037	0.477	0	x	x	0.1896	144
4, 4A,4B	4	8x8	0.640	63	0.493	0.034	0.416	1	0.493	0.425	0.1880	146
5	5	8x8	0.640	62	0.483	0.032	0.410	2	0.591	0.531	0.1876	150
6, 6B	5	8x8	0.640	62	0.483	0.032	0.410	2	0.591	0.531	0.1876	150
7, 7B	5	8x8	0.640	62	0.483	0.032	0.410	2	0.591	0.531	0.1876	150
8, 8B -2w	82	8x8	0.640	62	0.483	0.032	0.411	2	0.591	0.531	0.1885	150
[illegible]B-4W*	84	8x8	0.640	60	0.483	0.032	0.411	4	0.591	0.531	0.1824	150
[illegible]3-4W**	84	8x8	0.640	60	0.483	0.032	0.411	4	0.483	0.431	0.1824	150
9, 9B	9	8x8	0.640	60	0.483	0.032	0.411	1	1.34	1.26	0.1824	150
10	9	8x8	0.640	60	0.483	0.032	0.411	1	1.34	1.26	0.1824	150
11	11	9x9	0.566	74	0.440	0.028	0.376	2	0.98	0.92	0.1757	146 full, 90 partial
13	11	9x9	0.566	74	0.440	0.028	0.376	2	0.98	0.92	0.1757	146 full, 90 partial
12	12	10x10	0.510	92	0.404	0.026	0.345	2	0.98	0.92	0.1857	150 full, 93 partial

*2 large water rods
**2 small water rods

Notes on table 1:

1. All dimensions in inches.
2. All fuel channels 5.278 inches inside, and from 0.065 to 0.120 inches thick.
3. All fuels are evaluated with 96.5% theoretical density and 3.7 wt% U-235 average enrichment.
4. The fuel pitch is for C and D lattice designs. The S lattice fuels have a smaller pitch, which is less reactive.
5. The fuel designs designated by GE as 6, 6B, 7, and 7B are sometimes referred to as "P" (pressurized) and "B" (barrier).

Provided all of the requirements of this section are met, the bounding fuel characteristics are: a) maximum initial lattice-average enrichment is 3.7%; b) the mnimum initial bundle average enrichment is 3.3%; c) the maximum assembly average burnup is 40,000

5.(b)(1) continued

MWD/MTU; d) the minimum cool time is 10 years; and e) the maximum heat load per assembly is 0.313 Kw.

Fuel assemblies are categorized into three types, Type I, Type II and Type III. There are two basic loading configurations for the package. The first configuration is a mixture of Type I and Type II fuel assemblies. The second configuration is Type III fuel assemblies. The maximum burnup, minimum initial enrichments and cooling times for each of the three fuel assembly types is contained in the tables below.

In the mixed Type I and Type II configuration, Type I assemblies shall be placed only into the interior compartments of the fuel basket as shown in figure 5.3-3 of the application. Type II fuel assemblies may be placed in any basket fuel compartment.

In the second configuration, Type III fuel assemblies may be placed in any basket fuel compartment.

Acceptable cooling time as a function of maximum burnup and minimum initial enrichment and BWR Cooling times (years)
TYPE I BWR Fuel

Burnup (GWd/MTU)

Initial Enrichment (bundle ave %w)	15	20	30	32	33	34	35	36	37	38	39	40
1.0	10	10										
1.1	10	10										
1.2	10	10										
1.3	10	10										
1.4	10	10										
1.5	10	10	10	10	11	11	11					
1.6	10	10	10	10	10	11	11	11				
1.7	10	10	10	10	10	11	11	11	12			
1.8	10	10	10	10	10	11	11	11	11	12		
1.9	10	10	10	10	10	11	11	11	11	12		
2.0	10	10	10	10	10	10	11	11	11	12	12	
2.1	10	10	10	10	10	10	11	11	11	12	12	12
2.2	10	10	10	10	10	10	11	11	11	12	12	12
2.3	10	10	10	10	10	10	11	11	11	11	12	12
2.4	10	10	10	10	10	10	10	11	11	11	12	12
2.5	10	10	10	10	10	10	10	11	11	11	12	12
2.6	10	10	10	10	10	10	10	11	11	11	12	12
2.7	10	10	10	10	10	10	10	10	11	11	11	12
2.8	10	10	10	10	10	10	10	10	10	11	11	12
2.9	10	10	10	10	10	10	10	10	10	11	11	12
3.0	10	10	10	10	10	10	10	10	10	10	11	12
3.1	10	10	10	10	10	10	10	10	10	10	11	12
3.2	10	10	10	10	10	10	10	10	10	10	10	11
3.3	10	10	10	10	10	10	10	10	10	10	10	10
3.4	10	10	10	10	10	10	10	10	10	10	10	10
3.5	10	10	10	10	10	10	10	10	10	10	10	10
3.6	10	10	10	10	10	10	10	10	10	10	10	10
3.7	10	10	10	10	10	10	10	10	10	10	10	10

Acceptable cooling time as a function of maximum burnup and minimum initial enrichment and BWR Cooling times (years)
TYPE II BWR Fuel

Burnup (GWd/MTU)

Initial Enrichment (bundle ave %w)	15	20	30	32	33	34	35	36	37	38	39	40
1.0	18	21	X	X	X	X	X	X	X	X	X	X
1.1	17	20	X	X	X	X	X	X	X	X	X	X
1.2	17	20	X	X	X	X	X	X	X	X	X	X
1.3	17	20	X	X	X	X	X	X	X	X	X	X
1.4	17	20	X	X	X	X	X	X	X	X	X	X
1.5	16	19	25	26	26	X	X	X	X	X	X	X
1.6	16	19	25	26	26	X	X	X	X	X	X	X
1.7	16	19	25	25	26	26	27	X	X	X	X	X
1.8	16	19	24	25	26	26	27	27	X	X	X	X
1.9	16	19	24	25	25	26	27	27	X	X	X	X
2.0	16	18	24	25	25	26	26	27	28	X	X	X
2.1	15	18	23	25	25	26	26	27	27	X	X	X
2.2	15	18	23	25	25	25	26	27	27	X	X	X
2.3	15	18	23	24	25	25	26	26	27	27	X	X
2.4	15	18	22	24	24	25	26	26	27	27	X	X
2.5	15	17	22	24	24	25	25	26	26	27	X	X
2.6	15	17	22	24	24	24	25	26	26	27	X	X
2.7	15	17	22	24	24	24	25	26	26	26	27	27
2.8	14	17	22	23	24	24	25	25	26	26	27	27
2.9	14	17	22	23	23	24	24	25	26	26	27	27
3.0	14	17	21	23	23	23	24	25	25	26	27	27
3.1	14	17	21	23	23	23	24	25	25	26	27	27
3.2	13	16	21	23	23	23	24	24	25	25	26	27
3.3	13	16	21	23	22	23	23	24	25	25	26	26
3.4	13	16	21	23	22	23	23	24	25	25	26	26
3.5	13	16	21	22	22	23	23	24	25	25	26	26
3.6	13	16	21	21	22	22	23	24	25	25	26	26
3.7	12	15	20	21	22	22	23	24	25	25	25	26

Acceptable cooling time as a function of maximum burnup and minimum initial enrichment and BWR Cooling times (years)
TYPE III BWR Fuel

Burnup (GWd/MTU)

Initial Enrichment (bundle ave %w)	15	20	30	32	33	34	35	36	37	38	39	40
1.0	10	11										
1.1	10	11										
1.2	10	10										
1.3	10	10										
1.4	10	10										
1.5	10	10	15	16	16	17	17					
1.6	10	10	14	16	16	17	17	17				
1.7	10	10	14	15	16	16	17	17	17			
1.8	10	10	14	15	15	16	16	17	17	18		
1.9	10	10	14	15	15	16	16	17	17	18		
2.0	10	10	14	15	15	16	16	16	17	17	18	
2.1	10	10	14	15	15	15	16	16	16	17	18	18
2.2	10	10	13	14	15	15	16	16	16	17	17	18
2.3	10	10	13	14	15	15	16	16	16	17	17	18
2.4	10	10	13	14	15	15	15	16	16	17	17	18
2.5	10	10	13	14	14	15	15	16	16	16	17	18
2.6	10	10	13	14	14	15	15	16	16	16	17	17
2.7	10	10	13	14	14	15	15	15	16	16	17	17
2.8	10	10	13	13	14	14	15	15	16	16	17	17
2.9	10	10	13	13	14	14	15	15	15	16	16	17
3.0	10	10	12	13	14	14	14	15	15	16	16	17
3.1	10	10	12	13	14	14	14	15	15	15	16	16
3.2	10	10	12	13	14	14	14	15	15	15	16	16
3.3	10	10	12	13	13	14	14	14	15	15	16	16
3.4	10	10	12	13	13	13	14	14	15	15	16	16
3.5	10	10	12	13	13	13	14	14	14	15	15	16
3.6	10	10	12	12	13	13	14	14	14	15	15	15
3.7	10	10	12	12	13	13	14	14	14	15	15	15

5.(b) continued

(2) Maximum quantity of material per package

The maximum contents weight is 75,600 pounds. The maximum weight of the irradiated fuel contents is 47,900 pounds.

(3) Decay Heat Limit

Maximum decay heat per package not to exceed 21.2kW. The maximum heat load per assembly is 0.313 kW/assembly.

(c) Criticality Safety Index 0.0

6. 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, as supplemented.

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

7. Known or suspected fuel assemblies with cladding defects greater than pin hole leaks and or hairline cracks are not authorized.

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

9. Revision No. 1 of this certificate may be used until February 28, 2007.

10. Expiration date: February 28, 2011.

REFERENCES

Transnuclear, Inc., application dated May 19, 1999.

Supplements dated March 2, October 18, and November 13, 2000, January 12, 2001, and January 20, 2006.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Robert A. Nelson, Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: February 10, 2006

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
9294	4	71-9294	USA/9294/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)*

Global Nuclear Fuel - Americas, LLC
P.O. Box 780
Wilmington, NC 28402

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Global Nuclear Fuel - Americas, LLC, application dated January 29, 2001.

4. CONDITIONS

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

5.

Packaging

(1) Model No.: NPC

(2) Description

A cubic stainless steel and foam outer packaging with nine cylindrical containment vessels for the transport of unirradiated, low-enriched uranium oxide powder, pellets, and compounds of uranium as defined in 5(b). The overall package dimensions are approximately 45 inches wide, 45 inches deep, and 44 inches high.

The outer packaging consists of a 10-gage stainless steel outer shell with a ceramic fiber board liner and rigid polyurethane foam filler. The foam filler has a three-by-three array of vertical cylindrical cutouts that accommodate stainless steel sleeves for placement of the containment vessels. The outer packaging is equipped with a top cover that is secured to the outer packaging body by a combination of 16 closure cap screws and four closure strips secured by 24 bolts.

The containment vessel is a maximum 8.515 inches in inner diameter and approximately 32 inches in overall length. The containment vessel is constructed of 18-gage stainless steel, surrounded by a cadmium sheet and polyethylene wrap within a 24-gage stainless steel jacket. The containment vessel is closed by a 16-gage closure lid, a silicone rubber gasket, and a band clamp assembly, which is composed of a 0.063-inch thick strap and retainer, a T-bolt, and a nut.

The gross weight of the package (packaging and contents) is 1,302 kg (2,870 pounds). The maximum weight of the contents is 540 kg (1,190 pounds).

5.(a) (3) Drawings

The packaging is fabricated and assembled in accordance with the following Global Nuclear Fuel - Americas, LLC, Drawing Nos.:

177D4970, Sheet 1, Revision 0
177D4970, Sheet 2, Revision 0
177D4970, Sheet 3, Revision 0
177D4970, Sheet 4, Revision 0
177D4970, Sheet 5, Revision 0
177D4970, Sheet 6, Revision 0
177D4970, Sheet 7, Revision 0
177D4970, Sheet 8, Revision 0

(b) Contents

Type, Form, and Maxium Quantity of Material Per Package

Material Forms[1] (≤5.00 wt.% U-235)	Particle Size Restriction: Minimum OD (Inches)	Maximum Loading per ICCA (kgs)		Maximum Loading per NPC (kgs)	
		Net[4]	Uranium	Net[4]	Uranium
Homogenous Uranium Oxide/Compounds[2]	N/A	60.0	52.89	540.0	476.1
Heterogenous UO_2 Pellets (BWR)	0.342	60.0	48.48	540.0	436.3
Heterogenous UO_2 Pellets(PWR)	0.300	60.0	46.71	540.0	420.4
Heterogenous Uranium Compounds[3]	Unrestricted particle size	60.0	40.54	540.0	364.8

[1]The Material Form within any NPC must be the same.
[2]Homogenous compounds limited to UO_2, U_3O_8, $UO_{x,\ x>2}$, dried calcium-containing sludges, $UO_2(NO3)_2 \cdot 6H_2O$, and uranium oxide bearing ash.
[3]Heterogenous compounds limited to UO_2, U_3O_8, and $UO_{x,\ x>2}$.
[4]Maximum content weight of any ICCA including plastic or metal receptacles (e.g., bags, bottles, cans).
Note: Uranium-bearing contents may be moderated by water or carbon to any degree and may be mixed with other non-fissile materials with the exception of deuterium, tritium, and beryllium. Materials such as uranium metal and uranium metal alloys are not covered by this certificate.

(c) Criticality Safety Index 0.7

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9294	4	71-9294	USA/9294/AF-85	3	OF	3

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) Each packaging must be acceptance tested and maintained in accordance with 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 71.17.

8. Revision No. 3 of this certificate may be used until November 30, 2006.

9. Expiration date: November 30, 2010.

REFERENCES

Global Nuclear Fuel - Americas, LLC, application dated January 29, 2001.

Supplements dated: August 1, 23 and 27, 2001; March 4 and September 30, 2002; June 30 and October 3, 2005.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Robert A. Nelson, Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date November 21, 2005

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9295	0	71-9295	USA/9295/B(U)F-96	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)*

   Packaging Technology, Inc.
   1102 Broadway Plaza, Suite 300
   Tacoma, WA 98402-3526

   b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

   Packaging Technology, Inc., application dated June 25, 2004, 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.: MFFP

(2) Description

The MFFP package is designed to transport three unirradiated mixed oxide (MOX) fuel assemblies. The MFFP body is made of a 9/16-inch thick XM-19 austenitic stainless steel cylindrical shell with the flange section and a 1-1/2 inch bottom end plate welded to it. A circumferentially continuous doubler plate, constructed of Type XM-19 austenitic stainless steel, is welded to each end of the shell, near the end of each impact limiter. Welded to the doubler plate are the impact limiter attachment lugs, six per impact limiter. The doubler plate also serves to provide a tiedown interface with the transportation skid.

The seal flange is located at the open end of the body, and consists of a locally thicker wall section to accommodate the closure lid sealing area and the closure bolt threaded holes. The transition between the shell and the seal flange section is a 3:1 taper. Polyurethane foam is used to build the outer diameter of the body out to the full diameter of the sealing flange and closure lid.

The closure lid is a weldment constructed of Type XM-19 3/4-inch outer plate and 5/8-inch thick inner plate, stiffened with eight 1/2-inch thick radial ribs that are three inches deep. A 1/2-inch thick, 6 inch inner diameter cylinder forms a hub at the inner end of the radial ribs. The ribs are welded on all four edges to the adjacent structure. Each rib has a projection that passes through a slot in the outer plate, and the ribs and outer plate are welded together.

5.(a)(2) continued

The closure lid inner plate is welded to the outer ring. The seal flange of the closure lid has a minimum thickness of one inch, and provides location for three O-ring bore seals with the middle seal providing the containment seal. The seals are 3/8-inch diameter butyl rubber O-ring.

Up to three unirradiated fuel assemblies are held in place inside the overpack by a strongback assembly which is constructed from 1/4-inch thick Type 304 stainless steel weldment, a series of clamp arm assemblies, a top, and a bottom plate assemblies. For shipping less than three fuel assemblies, non-fuel dummy assemblies are used in the strongback locations not occupied by the fuel assemblies. The physical size and weight of the non-fuel dummy assemblies are nominally the same as the MK-BW/MOX1 17 x 17 design. Neutron poison plates are placed inside the weldment. A series of fuel control structure (FCS) limits lateral expansion of fuel rods during vertical and near vertical hypothetical accident condition (HAC) free drops and also hold neutron poison plates.

A pair of conical-shaped impact limiters filled with polyurethane foam provide thermal and impact protections. The closure lid end impact limiter has 1/4-inch thick shells to resist perforation from the HAC puncture drop, and to protect the closure lid and sealing area from puncture and HAC fire damage.

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

Overall package outside dimensions (inches)	
Without Impact Limiters	
Diameter	30
Length	171
With Impact Limiters	
Diameter	60
Length	201
Maximum content weight	4,740 lbs
Maximum package weight (Including contents)	14,130 lbs

(3) Drawings

The packaging shall be constructed and assembled in accordance with Packaging Technology, Inc., drawing numbers:

(a) Shipping Package	99008-10, Rev. 2, Sheet 1
(b) Body Assembly	99008-20, Rev. 2, Sheets 1 through 5
(c) Strongback Assembly	99008-30, Rev. 3, Sheets 1 through 7

5.(a)(3) continued

(d) Top Plate Assembly	99008-31, Rev. 1, Sheets 1 through 3
(e) Bottom Plate Assembly	99008-32, Rev. 0, Sheets 1 and 2
(f) Clamp Arm Assembly	99008-33, Rev. 1, Sheets 1 through 4
(g) Fuel Control Structure Assembly	99008-34, Rev. 3, Sheets 1 and 2
(h) Impact Limiter	99008-40, Rev. 1, Sheets 1 through 3

(b) Contents

(1) Type and Form of Material

Unirradiated 17 x 17 fuel assemblies with solid PuO_2+UO_2 pellets in zirconium based alloy (M5) tubes. The fuel assemblies are based on the MK-BW/MOX1 17 x 17 PWR design. The fuel assemblies may contain Burnable Poison Rod Assemblies (BPRA). The physical specifications for the unirradiated fuel assemblies and the burnable poison rod assemblies are provided in Tables 1 and 2. For shipping less than three fuel assemblies, non-fuel dummy assemblies are used in the strongback locations not occupied by the fuel assemblies. The physical size and weight of the non-fuel dummy assemblies are nominally the same as the MK-BW/MOX1 17 x 17 design.

5.(b)(1) continued

Table 1 - Fuel Assembly Physical Parameters
(nominal values unless stated otherwise)

Parameter	Values
Fuel Rod Cladding Material	M5
Fuel Rod Array	17 x 17
Fuel Rods per Fuel Assembly	264
Guide Tubes per Fuel Assembly	24
Instrument Tubes per Fuel Assembly	1
Guide/Instrument Tube Thickness (inches)	0.016
Fuel Assembly Length (inches)	161.61
Fuel Assembly Maximum Width (inches)	8.565
Fuel Rod Pitch (inches)	0.496
Fuel Rod Length (inches)	152.4
Fuel Rod Outside Diameter (inches)	0.374
Fuel Rod Clad Thickness (inches)	0.023
Active Fuel Length (inches)	144.0
PuO_2 + UO_2 Weight (pounds)	1,157
Heavy Metal Weight (pounds)	1,020
Maximum Fuel Assembly Weight including Burnable Poison Rod Assembly (pounds)	1,580
Maximum Initial Pu Loading (weight% of heavy metal)	6.0

Table 2 - Burnable Poison Rod Assembly Parameters

Parameter	Value
Poison Rod Cladding Material	Zircaloy-4
Poison/Thimble Plug Rod Array	Up to 24 rods
Burnable Poison Material	Al_2O_3-B_4C

5. (b) (2) Maximum Quantity of Material per Package

Three unirradiated fuel assemblies with the specifications on the fuel pellets and their enrichment provided in Table 3.

Table 3 - Nuclear Design Parameters for Fuel Assemblies

Parameter	Value
Nominal Pellet Diameter (inches)	0.323
Maximum Effective Pellet Density (gram/cm^3)	10.85
Maximum Total Plutonium (Pu) Content	0.06 g Pu/g Heavy Metal (Pu+U)
Plutonium Isotopic Contents	Pu-238: Up to 0.0005 g/g Pu Pu-239: 0.90 to 0.95 g/g Pu Pu-240: 0.05 to 0.09 g/g Pu Pu-241: Up to 0.01 g/g Pu Pu-242: Up to 0.001 g/g Pu
Minimum Total Uranium (U) Content	0.94 g U/g Heavy Metal (Pu+U)
Uranium Isotopic Contents	U-234: Up to 0.0005 g/g U U-235: Up to 0.003 g/g U U-238: Remainder of U content

(c) Criticality Safety Index 0.0

6. 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 Package Operations of Chapter 7 of the application, as supplemented.

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

(c) The boron-10 areal density within each of the internal neutron poison plates shall be verified as described in Section 8.1.5.2 of the application, as supplemented.

(d) If wrapping is used on the unirradiated fuel assemblies, the ends must be assured to be open during shipment in the package.

(e) Non-fuel dummy assemblies with the same nominal size and weight as the MK-BW/MOX1 17 x 17 design shall be used in the case of loading less than three fuel assemblies in a MFFP packaging.

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

8. Expiration date: June 30, 2010.

REFERENCES

Packaging Technology, Inc., application dated June 25, 2004.

Supplement dated: February 4 and 10, April 8, and June 3, 2005.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Stewart W. Brown for

Robert J. Lewis, Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: June 24, 2005

UNITED STATES NUCLEAR REGULATORY COMMISSION

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
9296	6	71-9296	USA/9296/B(U)-96	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)*

QSA Global, Inc.
40 North Avenue
Burlington, MA 01803

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

QSA Global, Inc., consolidated application dated October 20, 2005.

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. 880 Series Packages

(2) Description

The Model No. 880 series packages are designed for use as a radiography exposure device and a transport package for Type B quantities of radioactive material in special form. The Model No. 880 has three versions called the 880 Delta, 880 Sigma and the 880 Elite. The 880 Delta has a maximum capacity of 150 Curies of Iridium-192 or 150 Curies of Selenium-75, the 880 Sigma has a maximum capacity of 130 Curies of Iridium-192 or 150 Curies of Selenium-75, and the 880 Elite has a maximum capacity of 50 Curies of Iridium-192 or 150 Curies of Selenium-75. The Delta and Sigma versions are identical and the Elite has a lighter weight depleted uranium shield. An optional jacket can be placed on the packages when they are used as an industrial radiography exposure device or a transport package.

All versions of the package, without the jacket, are cylindrical in shape with a diameter of 5 inches and a length of 13 5/16 inches. With the jacket, the shape of the packages is an extruded triangle 9 inches high, 7 ½ inches wide, and 13 5/16 inches long. The weight of the Delta and Sigma versions is 46 pounds (52 pounds with the jacket) and the Elite version is 37 pounds (42 pounds with the jacket).

The major components of the packages consist of a welded stainless steel cylindrical body, a depleted uranium shield, a stainless steel rear plate with a locking assembly, a stainless steel front plate with a shielded port, and an optional jacket.

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

The welded cylindrical body consists of a five inch diameter, 0.06 inch wall tube shell with 0.12 inch end-plates. A U-bracket is welded to each end-plate and is located on the inside cavity of the shell tube. The depleted uranium shield is centrally located within the welded body between the end-plate and is fastened to each U-bracket by a 0.37 inch diameter titanium shield pin. A U-shaped copper spacer fills the gap between the shield and the U-bracket. An S-shaped titanium source tube is cast into the center of the shield to provide a cavity for the source wire assembly to travel through during use.

The front and rear plates are attached to the welded body with four tamperproof screws through rivnuts assembled into end-plates. The rear plate assembly consists of a source locking mechanism fastened to the rear plate. The front plate assembly consists of a shielded port mechanism contained within the front plate.

An optional polyurethane jacket covers the package cylinder and provides a handle and a stable base. The jacket handle contains a wire molded in for additional reinforcement.

(3) Drawings

The packaging is constructed in accordance with the AEA Technology/QSA, Inc., drawings R88000, Rev. J, Sheets 1-5.

(b) Contents

(1) Type and form of material

Iridium-192 as a sealed source which meets the requirements of special form radioactive material.

Selenium-75 as a sealed source which meets the requirements of special form radioactive material.

(2) Maximum quantity of material per package

150 Curies (5.55 TBq) (output) Ir-192 for the Model No. 880 Delta.
150 Curies (5.55 TBq) (output) Se-75 for the Model No. 880 Delta.

130 Curies (4.81 TBq) (output) Ir-192 for the Model No. 880 Sigma.
150 Curies (5.55 TBq) (output) Se-75 for the Model No. 880 Sigma.

50 Curies (1.85 TBq) (output) Ir-192 for the Model No. 880 Elite.
150 Curies (5.55 TBq) (output) Se-75 for the Model No. 880 Elite.

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5. (b) (2) Contents (continued)

Output curies are determined by measuring the source output at 1 meter and expressing its activity in curies derived from the following: 0.48 R/hr - Ci Iridium-192 at 1 meter and 0.20 R/hr - Ci Selenium-75 at 1 meter. (Ref: Radiological Health Handbook, rev. ed., U.S. Public Health Service, Bureau of Radiological Health, Rockville, MD, 1970.)

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

7. The name plate 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 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.

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

10. Revision No. 5 of this certificate may be used until August 31, 2007.

11. Expiration date: March 31, 2011.

REFERENCES

QSA Global, Inc., consolidated Safety Analysis Report dated October 20, 2005.

Supplement dated July 19, 2006.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

[signature] for

Christopher Regan, Acting Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: September 1, 2006

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
9297	1	71-9297	USA/9297/AF-96	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)*

Westinghouse Electric Company
P.O. Drawer R
Columbia, SC 29250

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Westinghouse Electric Company application dated April 1, 2004, 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.: Traveller STD and Traveller XL

(2) Description

The Traveller package is designed to transport non-irradiated uranium fuel assemblies or rods with enrichment up to 5.0 weight percent. The package is designed to carry one fuel assembly or one container for loose rods. The package consists of three components: 1) an outerpack, 2) a clamshell, and 3) a fuel assembly or rod container.

The outerpack is a structural component that serves as the primary impact and thermal protection for the fuel assembly or rod container. The outerpack has a long horizontal tubular design consisting of a top and bottom half. At each end of the package are thick limiters consisting of two sections of foam of different densities sandwiched between three layers of sheet metal. The impact limiters are integral parts of the outerpack and reduce damage to the contents during an end, or high-angle drop. The outerpack also provides for lifting, stacking, and tie down during transportation.

The clamshell is a horizontal structural component that serves to protect the contents during routine handling and in the event of an accident. The clamshell consists of an aluminum "V" extrusion, two aluminum door extrusions, and a small access door. Each extruded aluminum door is connected to the "V" extrusion with piano-type hinges (continuous hinges). These doors are held closed with a latching mechanism and quarter-turn bolts. Neutron absorber plates are installed in each leg of the "V" extrusion and in each of the doors. The "V" extrusion and the bottom plate are lined with a cork rubber pad to cushion and protect the contents during normal handling and transport conditions.

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

The Traveller package is designed to carry loose rods using either of two types of rod containers: a rod box or rod pipe. The rod box is an ASTM, Type 304 stainless steel container of rectangular cross section with stiffening ribs located approximately every 60 centimeters (cm) (23.6 inches (in.)) along its length. It is secured by fastening a removable top cover to the container body using socket head cap screws. The rod pipe consists of a 15.2 cm (6 in.) standard 304 stainless steel, Schedule 40 pipe, and standard 304 stainless steel closures at each end. The closure is a 0.635 cm (0.25 in.) thick cover secured with Type 304 stainless steel hardware to a flange fabricated from 0.635 cm (0.25 in.) thick plate.

There are two models of the Traveller packaging, the Traveller STD and the Traveller XL.

Traveller STD:

Package gross weight	2,041 kilograms (kg) (4,500 pounds (lbs))
Packaging gross weight	1,293 kg (2,850 lbs)
Contents gross weight	748 kg (1,650 lbs)
Outer dimensions	
Length	500 cm (197 in.)
Width	68.6 cm (27.1 in.)
Height	100 cm (39.3 in.)

Traveller XL:

Package gross weight	2,313 kg (5,100 lbs)
Packaging gross weight	1,419 kg (3,129 lbs)
Contents gross weight	894 kg (1,971 lbs)
Outer dimensions	
Length	574 cm (226.1 in.)
Width	68.6 cm (27.1 in.)
Height	100 cm (39.3 in.)

(3) Drawings

The packagings are fabricated and assembled in accordance with the following Westinghouse Electric Company's Drawing Nos.:

10004E58, Rev. 3 (Sheets 1-8)
10006E58, Rev. 5
10006E59, Rev. 1 (Sheets 1-2)

(b) Contents (Type and Form of Material)

(1) Fuel Assembly

(i) Unirradiated PWR uranium dioxide fuel assemblies with a maximum uranium-235 enrichment of 5.0 weight percent. The parameters of the fuel assemblies that are permitted are as follows:

5.(b)(1)(i) Fuel Assembly (Continued)

Parameters for 14 x 14 Fuel Assemblies

Fuel Assembly Description	14 x 14	14 x 14	14 x 14
Fuel Assembly Type	W-STD	W-OFA	CE-1/CE-2
No. of Fuel Rods per Assembly	179	179	176
No. of Non-Fuel Rods	17	17	20
Nominal Guide Tube Wall Thickness	0.043 cm (0.017 in.)	0.043 cm (0.017 in.)	0.097 cm (0.038 in.)
Nominal Guide Tube Outer Diameter	1.369 cm (0.539 in.)	1.336 cm (0.526 in.)	2.822 cm (1.111 in.)
Nominal Pellet Diameter	0.929 cm (0.366 in.)	0.875 cm (0.344 in.)	0.956/0.966 cm (0.376/0.381 in.)
Nominal Clad Outer Diameter	1.072 cm (0.422 in.)	1.016 cm (0.400 in.)	1.118 cm (0.440 in.)
Nominal Clad Thickness	0.062 cm (0.024 in.)	0.062 cm (0.024 in.)	0.071/0.066 cm (0.028/0.026 in.)
Clad Material	Zirconium alloy	Zirconium alloy	Zirconium alloy
Nominal Assembly Envelope	19.70 cm (7.76 in.)	19.70 cm (7.76 in.)	20.60 cm (8.11 in.)
Nominal Lattice Pitch	1.412 cm (0.556 in.)	1.412 cm (0.556 in.)	1.473 cm (0.580 in.)

Parameters for 15 x 15 Fuel Assemblies

Fuel Assembly Description	15 x 15	15 x 15
Fuel Assembly Type	STD/OFA	B&W
No. of Fuel Rods per Assembly	205	208
No. of Non-Fuel Rods	20	17
Nominal Guide Tube Wall Thickness	0.043/0.043 cm (0.017/0.017 in.)	0.043 cm (0.017 in.)
Nominal Guide Tube Outer Diameter	1.387/1.354 cm (0.546/0.533 in.)	1.354 cm (0.533 in.)
Nominal Pellet Diameter	0.929 cm (0.366 in.)	0.929 cm (0.366 in.)
Nominal Clad Outer Diameter	1.072 cm (0.422 in.)	1.072 cm (0.422 in.)
Nominal Clad Thickness	0.062 cm (0.024 in.)	0.062 cm (0.024 in.)
Clad Material	Zirconium alloy	Zirconium alloy
Nominal Assembly Envelope	21.39 cm (8.42 in.)	21.66 cm (8.53 in.)
Nominal Lattice Pitch	1.430 cm (0.563 in.)	1.443 cm (0.568 in.)

5.(b)(1)(i) Fuel Assembly (Continued)

Parameters for 16 x 16 Fuel Assemblies

Fuel Assembly Description	16 x 16	16 x 16	16 x 16	16 x 16
Fuel Assembly Type	W-STD	CE	NGF	ATOM
No. of Fuel Rods per Assembly	235	236	235	236
No. of Non-Fuel Rods	21	20	21	20
Nominal Guide Tube Wall Thickness	0.046 cm (0.018 in.)	0.102 cm (0.040 in.)	0.041 cm (0.016 in.)	0.057 cm (0.023 in.)
Nominal Guide Tube Outer Diameter	1.196 cm (0.471 in.)	2.489 cm (0.980 in.)	1.204 cm (0.474 in.)	1.354 cm (0.533 in.)
Nominal Pellet Diameter	0.819 cm (0.323 in.)	0.826 cm (0.325 in.)	0.784 cm (0.309 in.)	0.914 cm (0.360 in.)
Nominal Clad Outer Diameter	0.950 cm (0.374 in.)	0.970 cm (0.382 in.)	0.914 cm (0.360 in.)	1.075 cm (0.423 in.)
Nominal Clad Thickness	0.057 cm (0.023 in.)	0.064 cm (0.025 in.)	0.057 cm (0.023 in.)	0.072 cm (0.029 in.)
Clad Material	Zirconium alloy	Zirconium alloy	Zirconium alloy	Zirconium alloy
Nominal Assembly Envelope	19.72 cm (7.76 in.)	20.63 cm (8.12 in.)	19.72 cm (7.76 in.)	22.95 cm (9.03 in.)
Nominal Lattice Pitch	1.232 cm (0.485 in.)	1.285 cm (0.506 in.)	1.232 cm (0.485 in.)	1.430 cm (0.563 in.)

Parameters for 17 x 17 and 18 x 18 Fuel Assemblies

Fuel Assembly Description	17 x 17	17 x 17	18 x 18
Fuel Assembly Type	W-STD/XL	W-OFA	ATOM
No. of Fuel Rods per Assembly	264	264	300
No. of Non-Fuel Rods	25	25	24
Nominal Guide Tube Wall Thickness	0.041/0.051 cm (0.016 /0.020 in.)	0.041 cm (0.016 in.)	0.065 cm (0.026 in.)
Nominal Guide Tube Outer Diameter	1.204/1.224/1.24 cm (0.474/0.482/0.488 in.)	1.204 cm (0.474 in.)	1.240 cm (0.488 in.)
Nominal Pellet Diameter	0.819 cm (0.323 in.)	0.784 cm (0.309 in.)	0.805 cm (0.317 in.)
Nominal Clad Outer Diameter	0.950 cm (0.374 in.)	0.914 cm (0.360 in.)	0.950 cm (0.374 in.)
Nominal Clad Thickness	0.057 cm (0.023 in.)	0.057 cm (0.023 in.)	0.064 cm (0.025 in.)
Clad Material	Zirconium alloy	Zirconium alloy	Zirconium alloy
Nominal Assembly Envelope	21.39 cm (8.42 in.)	21.39 cm (8.42 in.)	22.94 cm (9.03 in.)
Nominal Lattice Pitch	1.260 cm (0.496 in.)	1.260 cm (0.496 in.)	1.270 cm (0.500 in.)

5.(b)(1) Fuel Assembly (Continued)

(ii) Non-fissile base-plate mounted core components and spider-body core components are permitted.

(iii) Neutron sources or other radioactive material are not permitted.

(iv) Materials with moderating effectiveness greater than full density water are not permitted.

(v) There is no restriction on the length of top and bottom annular blankets.

(2) Loose Fuel Rods

Unirradiated uranium dioxide fuel rods with a maximum uranium-235 enrichment of 5.0 weight percent. Fuel rods shall be transported in the Traveller package inside either a rod pipe or rod box as specified in License Drawings 10006E58 or 10006E59, specified in Section 5(a)(3). The fuel rods shall meet the parametric requirements given below:

Parameter	Limit
Maximum Enrichment	5.0 weight percent uranium-235
Pellet diameter	0.508 – 1.524 cm (0.20 – 0.60 in.)
Maximum stack length	Up to rod container length
Cladding	Zirconium alloy
Integral absorber	Gadolinia, erbia, and boron
Wrapping or sleeving	Plastic or other material with moderating effectiveness no greater than full density water
Maximum number of rods per container	Up to rod container capacity

5.(c) Criticality Safety Index

(1) When transporting fuel assemblies: 0.7

(2) When transporting loose rods in a rod container: 0.0

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

    (a) The package must be prepared for shipment and operated in accordance with the Operating Procedures in Chapter 7 of the Traveller License Application, Revision 4.

    (b) Each packaging must be acceptance tested and maintained in accordance with the Acceptance Tests and Maintenance Program in Chapter 8 of the Traveller License Application, Revision 4.

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

8. The package is not authorized by this certificate for air transport.

9. Revision No. 0 of this certificate may be used until April 30, 2007.

10. Expiration date: March 15, 2010.

REFERENCES

Westinghouse Electric Company application dated April 1, 2004.

Supplements dated: October 15 and November 16, 2004, and February 16, March 4, and March 10, 2005, and March 17 and April 12, 2006.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Robert A. Nelson, Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: April 15 2006

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
9299	1	71-9299	USA/9299/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)*

MDS Nordion
447 March Road
Kanata, Ontario, Canada K2K 1X8

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

MDS Nordion application dated
June 28, 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.: F-423

(2) Description

A double-walled welded stainless steel overpack for shipping sealed sources within the Gammacell 220 (GC220) gamma irradiator. The packaging consists of concentric box-like stainless steel shells separated by an annulus of rigid polyurethane foam. The overall overpack wall thickness is eight inches on the sides, twelve inches on the front and rear, and four inches on the base. The overpack lid is constructed of a sheet of 1/2-inch thick stainless steel on top, a sheet of 1/4-inch thick cold-rolled steel on the bottom, and 4-inches of polyurethane foam in between. The package is closed by bolting the lid to the body with 40 one-inch diameter bolts.

The GC220 irradiator is positioned inside the cavity formed by the inner stainless steel shell, along with an inner steel frame and a rigid polyurethane foam bonnet and lower crush pad. Shielding is provided by the GC220 irradiator, which is a welded steel lead-filled device. The GC220 is a lead-filled shielding head mounted on a steel stand. The GC220 shielding head consists of inner and outer steel shells with lead in between. The nominal lead thickness is 10 inches. The GC220 has an irregular shape, however, the base is 60-inches long by 40-inches wide. In its shipping configuration, the GC220 is 58-inches high. The GC220 shielding plug is welded from 304 stainless steel and lead filled. The GC220 drawer is welded from 304 stainless steel and is lead filled.

5(a) (2) (continued)

The maximum package weight (including contents) is 21,000 lbs (9,530 kgs). The approximate package component dimensions and weights are as follows:

Component	Weight (lbs / kg)	Nominal Dimensions (L x W x H inches)
Overpack Lid	1,046 / 474	67.50 x 55.00 x 4.75
Inner Frame	1,038 / 470	60.50 x 48.00 x 54.13
Bonnet	846 / 384	52.00 x 41.50 x 36.75
GC220	8,575 / 3,890	60.00 x 40.00 x 58.00
Overpack Body	8,750 / 3,969	86.50 x 66.00 x 80.37
Lower Crush Pad	354 / 160	47.00 x 31.00 x 7.00

(3) Drawings

The packaging is constructed in accordance with MDS Nordion Drawing No. F642301-001, Sheet 1, Revision G, and Sheet 2, Revision D.

(b) Contents

(1) Type and form of material

i. Cobalt-60 as sealed sources that meet the requirements of special form radioactive material.
ii. Cobalt-60 as sealed sources described in Condition No. 6 below.

(2) Maximum quantity of material per package

26,000 curies, a maximum of 48 sources per package, and a maximum of 5,000 curies per source.

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U.S. NUCLEAR REGULATORY COMMISSION

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FOR RADIOACTIVE MATERIAL PACKAGES

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6. Sealed sources limited to MDS Nordion sealed source capsules manufactured before February 19, 1973: C-166, C-167, and C-185. In addition, these sources must meet the following:

   (a) Sources must conform to the specifications identified in the application in Figure 4.2 for the C-166 source, Figure 4.3 for the C-167 source, and Figure 4.4 for the C-185 source;

   (b) Sources must be shown to not be leaking within six months prior to shipment; and

   (c) Sources must not have been damaged during their service life.

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. Revision No. 0 of this certificate may be used until August 31, 2006.

10. Expiration date: August 31, 2006.

REFERENCES:

MDS Nordion application dated June 28, 2000.

Supplements dated: May 11 and June 28, 2001; September 1 and November 17, 2005; and May 25, 2006.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Robert A. Nelson, Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date June 2, 2006

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
9301	1	71-9301	USA/9301/AF-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)*

   Packaging Technology, Inc.
   1102 Broadway Plaza, Suite 300
   Tacoma, WA 98402

   b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

   Packaging Technology, Inc. application dated July 24, 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 No.: TNF-XI

(2) Description

A shipping container for unirradiated enriched forms of homogenous and heterogenous uranium oxides. The packaging body is a parallelepiped and is approximately 44 inches x 44 inches x 37 inches. The package contents are enclosed in pails which each have a borated stainless steel ring. Three pails are stacked inside four inner wells of the packaging body. Each inner well is closed by a primary lid and an upper plug.

The packaging body is constructed of an outer stainless steel envelope which is 0.08 inches thick. The space between the outer shell and the inner wells is filled with fire-retardant, open cell phenolic foam.

The four inner wells each have an inside diameter of 14 inches and height of 27 inches. The inner wells are constructed of (1) an outer shell of stainless steel sheet 0.04 inches thick, with a diameter of 17 inches, (2) an inner shell of stainless steel sheet 0.04 inches thick with a diameter of 14 inches, and (3) a flat bottom 0.04 inch thick stainless steel sheet with a 0.08 inch thick borated stainless steel plate glued to it. A molded annular layer of neutron-poison BORA resin is inserted between the inner and outer steel shells of the inner well.

5(a)(2) Description (continued)

Each upper plug consists of two thermal insulating disks of phenolic foam, with an internal stiffener disk made of aluminum alloy. The upper plug assembly is encapsulated inside a 0.03 inch thick stainless steel envelope.

The four primary lids closing off the inner wells are stainless steel circular plates 0.2 inches thick on the center part, and 0.4 inches thick on the periphery. Four bayonet teeth are welded to the primary lid to lock in the well flanges. A primary lid locker is located between the well flange and the primary lid to prevent the rotation of the primary lid during transport. The primary lid and the inner well are sealed by an elastomer gasket set in a rectangular groove machined on the inner face of the primary lid.

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

Inner well inside diameter	14 inches
Overall package dimensions	
Width	44 inches
Length	44 inches
Height	41 inches
Maximum weight of contents in any pail	25 kg
Maximum content weight	300 kg
Maximum package weight (including contents)	1050 kg

(3) Drawing

The packaging is constructed in accordance with the Packaging Technology Inc. Drawing No. 10799-SAR, Rev. 3, Sheets 1 through 7.

(b) Contents

(1) Type and form of material

Uranium oxide pellets, powder, and scrap meeting the requirements of Enriched Commercial Grade Uranium, as defined in ASTM C996-96. U_3O_8 or $UO_{x,\ x>2}$ are authorized provided that the equivalent UO_2 mass is less than the limits specified below:

5(b)(1) Contents (continued)

Max ^{235}U Enrichment (weight percent)	Homogenous UO_2 Powder Maximum Loading (kg)	Heterogenous UO_2 Pellet Maximum Loading (kg)
≤4.05	300	300
4.1	300	293
4.15	300	287
4.25	300	271
4.35	300	259
4.45	300	247
4.55	294	238
4.65	281	228
4.75	265	219
4.85	255	208
4.95	244	202
5.0	239	197

(2) Maximum quantity of material per package

No more than 25 kg of contents per pail. No more than 300 kg of contents per package.

(c) Transport Index for Criticality Control (Criticality Safety Index): 0.5

6. 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 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; and,

(c) 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.

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
9301	1	71-9301	USA/9301/AF-85	4	OF	4

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

8. Expiration date: August 30, 2008.

<u>REFERENCES</u>

Packaging Technology, Inc. application dated July 24, 2002.

Supplements dated: October 29, 2002, March 7, April 3, May 6, June 26, and July 21, 2003.

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: August 14, 2003

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE	PAGES
9302	1	71-9302	USA/9302/B(U)F-85	1 OF	8

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

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

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Transnuclear Inc., application dated May 2, 2001.

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®-MP197

(2) Description

The NUHOMS®-MP197 package consists of an outer cask, into which a NUHOMS®-61BT transportable dry shielded canister (DSC) is placed. During shipment, energy-absorbing impact limiters are utilized for additional package protection. Additionally, a personnel barrier is mounted to the transportation frame to prevent access to the cask body.

Cask

The NUHOMS®-MP197 transport cask is fabricated primarily of stainless steel. Non-stainless steel members include the cask lead shielding between the containment boundary inner shell and the structural shell, the o-ring seals, the neutron shield, and carbon steel closure bolts. The body of the cask consists of a 1.25 inch thick, 68 inch inside diameter, stainless steel inner (containment) shell and a 2.5 inch thick, 82 inch outside diameter stainless steel structural shell, without impact limiters, which sandwich the 3.25 inch thick cast lead shielding. The overall external dimensions of the cask are 208 inches long and 91.5 inches in outer diameter. The weight of cask body is 148,840 pounds, including about 10,000 pounds of neutron shield and 60,000 pounds of cast lead.

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

The containment system of the NUHOMS®-MP197 transportation cask consists of the inner shell, a 6.50 inch thick bottom plate, 2.5 inch thick RAM access closure with a diameter of approximately 24 inches, a top closure flange, a 4.5 inch thick top closure lid with closure bolts, drain port closures and bolts, and double o-ring seals for each penetration. The containment vessel prevents leakage of radioactive material from the cask cavity. The cask cavity is pressurized to above atmospheric pressure with an inert gas (helium). Helium assists in the heat removal. Shielding is provided by about 4 inches of stainless steel, 3.25 inches of lead, and about 4.5 inches of neutron shielding. Four removable trunnions are provided for handling and lifting of the cask.

Dry Shielded Canister (DSC)

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 shell has an outside diameter of about 67 inches and an external length of about 200 inches. The DSC basket assembly provides criticality control and contains a storage position for each fuel assembly. No credit is given to the DSC as a containment boundary. The basket is designed to accommodate 61 intact BWR fuel assemblies with or without fuel channels. The basket structure consists of a welded assembly of stainless steel tubes (fuel compartments) separated by poison plates and surrounded by larger stainless steel boxes and support rails.

The poison plates are constructed from borated aluminum, and provide a heat conduction path from the fuel assemblies to the canister wall, as well as the necessary criticality control.

Impact Limiters

The impact limiter shells are fabricated from stainless steel. Within that shell is a laminate of balsa wood and redwood. Each impact limiter is attached to the cask top (front) and bottom (rear) by 12 bolts. The impact limiters are provided with seven fusible plugs that are designed to melt during a fire accident, thereby relieving excessive internal pressure. Each impact limiter has two hoist rings for handling. The hoist rings are threaded into the impact limiter shell. During transportation, the impact limiter hoist rings are removed. An aluminum thermal shield is added to the bottom impact limiter to reduce the impact limiter wood temperature. The weight of the impact limiters, the thermal shield, and attachment bolts, is approximately 28,000 lbs.

(3) Drawings

The package shall be constructed and assembled in accordance with the following Transnuclear Inc., drawing numbers:

1093-71-1, Revision 0,
NUHOMS®-197 Packaging
Transport Configuration

1093-71-2, Revision 1,
NUHOMS®-197 Packaging
General Arrangement

1093-71-3, Revision 1,
NUHOMS®-MP197 Packaging
Parts List

1093-71-4, Revision 1,
NUHOMS®-MP197 Packaging
Cask Body Assembly

1093-71-5, Revision 0,
NUHOMS®-MP197 Packaging
Cask Body Details

1093-71-6, Revision 0,
NUHOMS®-MP 197 Packaging
Cask Body Details

1093-71-7, Revision 0,
NUHOMS®-MP197 Packaging
Lid Assembly & Details

1093-71-8, Revision 0,
NUHOMS®-MP197 Packaging
Impact Limiter Assembly

1093-71-9, Revision 0,
NUHOMS®-MP197 Packaging
Impact Limiter Details

1093-71-10, Revision 0,
NUHOMS®-61BT Transportable
Canister for BWR Fuel Basket
Assembly

1093-71-11, Revision 1,
NUHOMS®-61BT Transportable
Canister for BWR Fuel Basket Details

1093-71-12, Revision 0,
NUHOMS®-61BT Transportable
Canister for BWR Fuel Basket Details

1093-71-13, Revision 1,
NUHOMS®-61BT Transportable
Canister for BWR Fuel General
Assembly

1093-71-14, Revision 1,
NUHOMS®-61BT Transportable
Canister for BWR Fuel General
Assembly

1093-71-15, Revision 2,
NUHOMS®-61BT Transportable
Canister for BWR Fuel Shell Assembly

1093-71-16, Revision 0,
NUHOMS®-61BT Transportable
Canister for BWR Fuel Shell Assembly

1093-71-17, Revision 2,
NUHOMS®-61BT Transportable
Canister for BWR Fuel Canister Details

1093-71-18, Revision 1,
NUHOMS®-61BT Transportable
Canister for BWR Fuel Canister Details

1093-71-20, Revision 0,
NUHOMS®-MP197 Packaging
Regulatory Plate

1093-71-21, Revision 0,
NUHOMS®-MP197 Packaging
on Transport Skids

5. (b) Contents of Packaging

(1) Type and Form of Material

(a) Intact irradiated BWR fuel assemblies, with or without fuel channels, with uranium oxide pellets and zircaloy cladding. Channel thickness is limited to 0.065 to 0.120 inches. Prior to irradiation, the fuel assemblies must meet the dimensions and specifications of Table 1. Assemblies containing fuel rods with no known or suspected cladding defects greater than hairline cracks or pinhole leaks are authorized when contained in the NUHOMS®-61BT DSC.

(b) The maximum burn-up and minimum cooling times for the individual 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)(c). The maximum total allowable cask heat load is 15.86 kW.

(c) The maximum assembly decay heat of an individual assembly is 260 watts.

(d) BWR fuel assembly poison material shall meet the design requirements of Table 3.

TABLE 1[1]

Assembly Type	7x7 49/0	8x8 63/1	8x8 62/2	8x8 60/4	8x8 60/1	9x9 74/2	10x10 92/2
Maximum Initial Enrichment (wt% ^{235}U)	See Table 3	See Table 3	See Table 3	See Table 3	See Table 3	See Table 3	See Table 3
Rod Pitch (in)	0.738	0.640	0.640	0.640	0.640	0.566	0.510
Number of Fuel Rods per Assembly	49	63	62	60	60	66-full 8-partial	78-full 14-partial
Fuel Rod OD (in)	0.563	0.493	0.483	0.483	0.483	0.440	0.404
Minimum Cladding Thickness (in)	0.032	0.034	0.032	0.032	0.032	0.028	0.026
Pellet Diameter	0.487	0.416	0.410	0.410	0.411	0.376	0.345
Maximum Active Fuel Length (in)	144	146	150	150	150	146-full 90-partial	150-full 93-partial

[1)] Maximum Co-59 content in the Top End Fitting region is 4.5 gm per assembly
Maximum Co-59 content in the Plenum region is 0.9 gm per assembly
Maximum Co-59 content in the In-Core region (including the whole fuel channel) is 4.5 gm per assembly
Maximum Co-59 content in the Bottom region is 4.1 gm per assembly

TABLE 2

Intact BWR Fuel Assembly Characteristics	
Physical Parameters:	
Fuel Design	7x7, 8x8, 9x9, or 10x10 BWR fuel assemblies manufactured by General Electric or equivalent reload fuel
Cladding Material	Zircaloy
Fuel Damage	Cladding damage in excess of pinhole leaks or hairline cracks is not authorized to be stored as "Intact BWR fuel"
Channels	Fuel may be stored with or without fuel channels
Maximum assembly weight	705lbs
Radiological Parameters:	
Group 1:	
Maximum Burnup:	27,000 MWd/MTU
Minimum Cooling Time:	6-Years
Maximum Initial Enrichment:	See Table 3
Minimum Initial Bundle Average Enrichment:	2.0 wt. % U-235
Maximum Initial Uranium Content:	198 kg/assembly
Maximum Decay Heat:	260 W/assembly
Group 2:	
Maximum Burnup:	35,000 MWd/MTU
Minimum Cooling Time:	12-Years
Maximum Initial Enrichment:	See Table 3
Minimum Initial Bundle Average Enrichment:	2.65 wt. % U-235
Maximum Initial Uranium Content:	198 kg/assembly
Maximum Decay Heat:	260 W/assembly

Intact BWR Fuel Assembly Characteristics	
Radiological Parameters:	
Group 3:	
Maximum Burnup:	37,200 MWd/MTU
Minimum Cooling Time:	12-Years
Maximum Initial Enrichment:	See Table 3
Minimum Initial Bundle Average Enrichment:	3.38 wt. % U-235
Maximum Initial Uranium Content:	198 kg/assembly
Maximum Decay Heat:	260 W/assembly
Group 4:	
Maximum Burnup:	40,000 MWd/MTU
Minimum Cooling Time:	15-Years
Maximum Initial Enrichment:	See Table 3
Minimum Initial Bundle Average Enrichment:	3.4 wt. % U-235
Maximum Initial Uranium Content:	198 kg/assembly
Maximum Decay Heat:	260 W/assembly

TABLE 3
Minimum Boron-10 Areal Density as a Function of Maximum Fuel Assembly Lattice Average Enrichment

NUHOMS® -61BT DSC Basket Type	Maximum Fuel Assembly Lattice Average Enrichment(wt.% U-235)	Minimum Boron-10 Areal Density for Boral® (g/cm^2)	Minimum Boron-10 Areal Density for Borated Aluminum, Metamic®, and Boralyn® (g/cm^2)	Areal Density Used in the Criticality Evaluation [75% Credit for Boral®] (g/cm^2)
	Intact Fuel Assemblies			
A	3.7	0.025	0.021	0.019
B	4.1	0.038	0.032	0.029
C	4.4	0.048	0.040	0.036

5. (b) Contents of Packaging (continued)

(2) Maximum quantity of material per package

(a) The quantity of material authorized for transport is 61 intact standard BWR fuel assemblies with or without fuel channels. 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.

(b) For material described in 5(b)(1) the approximate maximum payload is 21,500 lbs.

(c) Criticality Safety Index "0"

6. Fuel assemblies with missing fuel rods shall not be shipped unless the missing fuel rods are replaced by dummy rods that displace an equal or greater amount of water.

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. In addition this will include:

(1) verification of the basket type A, B, or C, by inspection of the last digit of the serial number on the grapple ring at the bottom of the DSC,

(2) verification that the fuel assemblies to be placed in the DSC meet the maximum burnup, maximum initial enrichment, minimum cooling time, and maximum decay heat limits for fuel assemblies as specified in Tables 2 and 3. The enrichment limit must correspond to the basket type determined in 7(a)(1) above.

(b) All fabrication acceptance tests and maintenance shall be performed in accordance with Acceptance Tests and Maintenance Program in Chapter 8 of the application, as supplemented. In addition this will include replacement of the cask lid bolts after 85, or fewer, round trip shipments to ensure that the allowable fatigue damage factor will not be exceeded during normal conditions of transport.

8. This package is approved for exclusive use by 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 71.17.

Revision No. 0 of this certificate may be used until January 31, 2007.

11. Expiration Date: July 31, 2007.

CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9302	1	71-9302	USA/9302/B(U)F-85	8	OF	8

REFERENCES

Transnuclear Inc., Safety Analysis Report for the NUHOMS®-MP197 Transport Packaging, dated May 2, 2001.

Transnuclear Inc., letters dated January 29, 2002, January 31, 2002, March 1, 2002, March 20, 2002, April 29, 2002, and May 16, 2002.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Robert A. Nelson, Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

February 2, 2006

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
9309	6	71-9309	USA/9309/B(U)F-96	1	OF	9

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)*

Global Nuclear Fuel - Americas, LLC
P.O. Box 780
Wilmington, NC 28402

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Global Nuclear Fuel - Americas, LLC, application dated March 31, 2004, 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.: RAJ-II

(2) Description

The RAJ-II package is a rectangular box that is 742 mm (29.21 in) high by 720 mm (28.35 in) wide by 5,068 mm (199.53 in) long to transport a maximum of two Boiling Water Reactor (BWR) fuel assemblies or individual rods that meet the ASTM C996-96 standard of enriched commercial grade uranium or enriched reprocessed uranium.

It is comprised of one inner container and one outer container both made of stainless steel. The inner container is comprised of a double-wall stainless steel sheet structure with alumina silicate thermal insulator filling the gap between the two walls to reduce the flow of the heat into the contents in the event of a fire. Foam polyethylene cushioning material is placed on the inside of the inner container for protection of the fuel assembly. The outer container is comprised of a stainless steel angular framework covered with stainless steel plates. Inner container clamps are installed inside the outer container with a vibro-isolating device between to alleviate vibration occurring during transportation. Wood and honeycomb resin impregnated kraft paper are placed as shock absorbers to reduce shock in the event of a drop of the package. The fuel rod clad and ceramic nature of the fuel pellets provide primary containment of the radioactive material.

5.(a)(2) continued

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

Maximum gross shipping weight	1,614 kg (3,558 lbs)
Maximum weight of inner container	308 kg (679 lbs)
Maximum weight of outer container	622 kg (1,371 lbs)
Maximum weight of packaging	930 kg (2,050 lbs)
Dimensions of inner container	
Length	4,686 mm (184.49 in)
Width	459 mm (18.07 in)
Height	286 mm (11.26 in)
Dimensions of outer container	
Length	5,068 mm (199.53 in)
Width	720 mm (28.35 in)
Height	742 mm (29.21 in)

(3) Drawings

This packaging is constructed in accordance with the Global Nuclear Fuel (GNF) Drawing Nos.:

Outer Container Drawings	Inner Container Drawings	Contents Containers
105E3737, Rev. 6	105E3745, Rev. 8	105E3773, Rev. 1
105E3738, Rev. 7	105E3746, Rev. 1	0028B98, Rev. 1
105E3739, Rev. 4	105E3747, Rev. 4	
105E3740, Rev. 4	105E3748, Rev. 2	
105E3741, Rev. 1	105E3749, Rev. 6	
105E3742, Rev. 3		
105E3743, Rev. 4		
105E3744, Rev. 5		

(b) Contents

(1) *Type and form of material*

Enriched commercial grade uranium or enriched reprocessed uranium, as defined in ASTM C996-96, oxide fuel rods enriched to no more than 5.0 weight percent in the U-235 isotope, with limits specified in Table 1 and Table 2 below.

5.(b)(1) continued

Table 1: Maximum weight of uranium dioxide pellets per fuel assembly

Type 8x8 fuel assembly	**Type 9x9 fuel assembly**	**Type 10x10 fuel assembly**
235 kg	240 kg	275 kg

Table 2: Maximum Authorized Concentrations

Isotope	**Maximum content**
U-232	2.00×10^{-9} g/gU
U-234	2.00×10^{-3} g/gU
U-235	5.00×10^{-2} g/gU
U-236	2.50×10^{-2} g/gU
Np-237	1.66×10^{-6} g/gU
Pu-238	6.20×10^{-11} g/gU
Pu-239	3.04×10^{-9} g/gU
Pu-240	3.04×10^{-9} g/gU
Gamma Emitters	5.18×10^{5} MeV - Bq/kgU

5.(b)(1)(i) 8 x 8 fuel assemblies comprised of 60 to 64 rods in a square array with a maximum active fuel rod length of 381 cm. The maximum pellet diameter, minimum clad thickness, rod pitch, water rod specifications, and poison rod specification are in accordance with Table 3 below.

(ii) 9 x 9 fuel assemblies comprised of 72 to 81 rods in a square array with a maximum active fuel rod length of 381 cm. The maximum pellet diameter, minimum clad thickness, rod pitch, water rod specifications, and poison rod specification are in accordance with Table 3 below.

5.(b)(1) continued

(iii) 10 x 10 fuel assemblies comprised of 91 to 100 rods in a square array with a maximum active fuel rod length of 385 cm. The maximum pellet diameter, minimum clad thickness, rod pitch, water rod specifications, and poison rod specification are in accordance with Table 3 below.

(iv) Oxide fuel rods configured loose, in a 5 inch diameter schedule 40 stainless steel pipe/protective case or strapped together. When fuel rods are placed in polyethylene sleeves, each polyethylene sleeve shall not exceed 0.0152 cm in thickness. The maximum pellet diameter, minimum clad thickness, and rod specifications are in accordance with Table 4 below.

5.(b)(1) continued

Table 3: Fuel Assembly Parameters

Parameter	**Units**	**Type**	**Type**	**Type**	**Type**
Fuel Assembly Type	Rods	8x8	9x9	FANP 10x10	GNF 10x10
UO_2 Density		≤98% Theoretical	≤98% Theoretical	≤98% Theoretical	≤98% Theoretical
Number of water rods (see Condition 8)	#	0, 2x2	0, 2 - 2x2 off-center diagonal, 3x3	0, 2 - 2x2 off-center diagonal, 3x3	0, 2 - 2x2 off-center diagonal, 3x3
Number of fuel rods	#	60 -64	72 - 81	91 - 100	91 - 100
Fuel Rod OD	cm	≥1.176	≥1.093	≥1.000	≥1.010
Fuel Pellet OD	cm	≤1.05	≤0.96	≤0.895	≤0.895
Cladding Type		Zirconium Alloy	Zirconium Alloy	Zirconium Alloy	Zirconium Alloy
Cladding ID	cm	≤1.10	≤1.02	≤0.933	≤0.934
Cladding Thickness	cm	≥0.038	≥0.036	≥0.033	≥0.038
Active Fuel Length	cm	≤381	≤381	≤385	≤385
Fuel Rod Pitch	cm	≤1.692	≤1.51	≤1.350	≤1.350
U-235 Pellet Enrichment	wt%	≤5.0	≤5.0	≤5.0	≤5.0
Max. Lattice Avg. Enrich.	wt%	≤5.0	≤5.0	≤5.0	≤5.0
Channel Thickness[a]	cm	0.17 - 0.3048	0.17 - 0.3048	0.17 - 0.3048	0.17 - 0.3048
Partial Fuel Rods (1/3 through 2/3 normal length)	#	None	12	14	14

5.(b)(1) *continued*

Table 3: Fuel Assembly Parameters (continued)

Parameter	**Units**	**Type**	**Type**	**Type**	**Type**
Gadolinia Requirements Lattice Avg. Enrichment[b]	# @ wt% Gd_2O_3				
≤5.0 wt% U-235		7 @ 2wt %	10 @ 2wt %	12 @ 2wt %	12 @ 2wt %
≤4.7 wt% U-235		6 @ 2wt %	8 @ 2wt %	12 @ 2wt %	12 @ 2wt %
≤4.6 wt% U-235		6 @ 2wt %	8 @ 2wt %	10 @ 2wt %	10 @ 2wt %
≤4.3 wt% U-235		6 @ 2wt %	8 @ 2wt %	9 @ 2wt %	9 @ 2wt %
≤4.2 wt% U-235		6 @ 2wt %	6 @ 2wt %	8 @ 2wt %	8 @ 2wt %
≤4.1 wt% U-235		4 @ 2wt %	6 @ 2wt %	8 @ 2wt %	8 @ 2wt %
≤3.9 wt% U-235		4 @ 2wt %	6 @ 2wt %	6 @ 2wt %	6 @ 2wt %
≤3.8 wt% U-235		4 @ 2wt %	4 @ 2wt %	6 @ 2wt %	6 @ 2wt %
≤3.7 wt% U-235		2 @ 2wt %	4 @ 2wt %	6 @ 2wt %	6 @ 2wt %
≤3.6 wt% U-235		2 @ 2wt %	4 @ 2wt %	4 @ 2wt %	4 @ 2wt %
≤3.5 wt% U-235		2 @ 2wt %	2 @ 2wt %	4 @ 2wt %	4 @ 2wt %
≤3.3 wt% U-235		2 @ 2wt %	2 @ 2wt %	2 @ 2wt %	2 @ 2wt %
≤3.1 wt% U-235		None	2 @ 2wt %	2 @ 2wt %	2 @ 2wt %
≤3.0 wt% U-235		None	None	2 @ 2wt %	2 @ 2wt %
≤2.9 wt% U-235		None	None	None	None
Polyethylene Equivalent Mass (Maximum per assembly)[c]	kg	11	11	10.2	10.2

a. Transport with or without channels is acceptable
b. Required gadolinia rods must be distributed symmetrically about the major diagonal
c. *Polyethylene equivalent mass calculation, refer to 6.3.2.2 of the application*

Table 4: Fuel Rod Parameters

Parameter	**Units**	**Type**	**Type**	**Type**
Fuel Assembly Type		8 x 8	9 x 9	10 x 10
UO_2 Density		≤98% theoretical	≤98% theoretical	≤98% theoretical
Fuel Rod OD	cm	≥1.10	≥1.02	≥1.00
Fuel Pellet OD	cm	≤1.05	≤0.96	≤0.90
Cladding Type		Zirc. Alloy	Zirc. Alloy	Zirc. Alloy

5.(b)(1) continued

Table 4: Fuel Rod Parameters (continued)

Parameter	**Units**	**Type**	**Type**	**Type**
Cladding ID	cm	≤1.10	≤1.02	≤1.00
Cladding Thickness	cm	≥ 0.00	≥ 0.00	≥ 0.00
Active Fuel Length	cm	≤381	≤381	≤385
Maximum U-235 Pellet Enrichment	wt%	≤5.0	≤5.0	≤5.0
Maximum Average Fuel Rod Enrichment	wt%	≤5.0	≤5.0	≤5.0

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

Total weight of payload contents (fuel assemblies, or fuel rods and rod shipping containers) not to exceed 684 kg (1508 pounds).

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

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

Allowable number of fuel rods per compartment (2 compartments per package).

	8 x 8 assembly type	9 x 9 assembly type	10 x 10 assembly type
Configured loose	≤25	≤25	≤25
Configured in 5-inch SS pipe/ protective case	≤22	≤26	≤30
Configured strapped together	≤25	≤25	≤25

(c) Criticality Safety Index 1.0

6. 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 Package Operations of Chapter 7 of the application, as supplemented.

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

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

   (d) If wrapping is used on the unirradiated fuel assemblies, the ends must be assured to be open during the shipment in the package.

7. Cluster separators are optional and may be comprised of polyethylene or other plastics. Polyethylene or plastic mass limits shall be determined in accordance with Section 6.3.2.2 (Material Specifications) of the application, as supplemented.

8. Water rods are limited as shown in Table 3 above.

   For 8 x 8 fuel assembly designs, there can be either 0 or 1 water rod, and the water rod location occupies a space equivalent to 2 x 2 fuel rods. This is designated as 0, 2 x 2 in the table.

   For 9 x 9 and 10 x 10 fuel assembly designs, there can be either 0, 1, or 2 water rods in the assembly, and the water rod location occupies a space equivalent to (a) two 2 x 2 fuel rod equivalent spaces on a diagonal at the center of the assembly, or (b) one 3 x 3 fuel rod equivalent space (9 fuel rods space) in the center of the assembly. These configurations are designated as 0, 2 - 2x2 off-center diagonal, 3x3 in the table.

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

10. Transport by air of fissile material is not authorized.

11. Revision No. 5 of this certificate may be used until May 31, 2007.

12. Expiration date: November 30, 2009.

REFERENCES

Global Nuclear Fuel - Americas, LLC, application dated March 31, 2004.

Supplement dated: April 22, September 3, September 16, October 28, November 8 and 29, 2004; and April 8, May 25, June 6, August 3, 2005; and January 27, 2006; and February 16 and April 21, 2006.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Meraj Rahimi
for Robert A. Nelson, Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: May 17, 2006

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
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2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

MDS Nordion
447 March Road
Ottawa, ON K2K 1X8
Canada

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

MDS Nordion application dated May 27, 2003, 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. F-431 Transport Package

(2) Description

The Model No. F-431 Transport Package is designed to transport Cesium-137 in either special form or RAMCO-50 non-special form sealed sources. The F-431 Transport Package consist of: (1) the overpack which provides impact and thermal protection; (2) either the MDS Nordion Gammacell-1000 irradiator (GC-1000), or the MDS Nordion Gammacell-3000 irradiator (GC-3000) which provides shielding protection; and (3) the radioactive contents in either special form or RAMCO-50 non-special form sealed sources which provides containment.

The F-431 Transport Package is a stainless steel cylindrical package with a 1,067-millimeter (mm) (42-inch (in.)) outside diameter and a height of 1,283 mm (50.5 in.) that is placed on a removable mild steel skid. The maximum weight of the package is 2,270 kilograms (kg) (5000 pounds (lb)).

The overpack consists of nested cylindrical shells. The shells are made from stainless steel and the volume between the shells is filled with rigid foam. This foam provides insulation during an accidental fire. Vent holes, plugged with material designed to melt in a fire, are provided between the shells to prevent pressure buildup and allow a pathway for escape of gases from foam during an accidental fire.

The GC-1000 and the GC-3000 are lead-shielding casks each with a source cavity. The package contents may consists of up to eight cesium-137 special form sealed sources or RAMCO-50 non-special form sealed sources (provided Condition 5.(b)(1)(ii) is met) inside a source holder, within the source cavity. The maximum total activity of cesium-137 is 113

5.(a)(2) continued

tera-Becquerels (TBq)(3,050 Curies (Ci)). The following are the features of the GC-1000 and GC-3000:

Irradiator Model	Rated Capacity	Diameter*	Height*	Lead Thickness*	Steel Shell Thickness*	Weight*
GC-1000	113 TBq (3,050 Ci)	457 mm (18 in.)	610 mm (24 in.)	150 mm (6 in.)	9.5 mm (0.375 in.)	1,035 kg (2,280 lb)
GC-3000	113 TBq (3,050 Ci)	457 mm (18 in.)	610 mm (24 in.)	110 mm (4.3 in.)	9.5 mm (0.375 in.)	1,035 kg (2,280 lb)

* Nominal Values

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

Package outside diameter	1,067mm (42 inches)
Package height	1,283 mm (50.5 inches)
Cavity diameter	559 mm (22 inches)
Cavity height	813 mm (32 inches)
Removable skid	1,118 mm (44 inches) x 1,003 mm (39.5 inches) x 203 mm (8 inches)
Overpack weight	1044 kg (2300 lbs)
Contents weight (max.)	1226 kg (2700 lbs)
Maximum package weight	2,270 kg (5000 lbs)

(3) Drawings

The packaging is constructed in accordance with the MDS Nordion drawing F643101-001, Sheet 1, Revision F and Sheet 2, Revision B.

(b) Contents

(1) Type and form of material

(i) Cesium-137 as a sealed source which meets the requirements of special form radioactive material. The sealed sources consist of the following special form sources: C-378, C-1000, C-1001, C-3000, C-3001, or ISO-1000.

(ii) Cesium-137 as the RAMCO-50 non-special form sealed source, provided the following conditions are met:

- Source must conform to the specifications given in Figure 4.8 of the Safety Analysis Report and sealed source registry Certificate No. NR-0880-S-804-S.
- Source must have been shown to not be leaking within six months prior to shipment.

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5.(b) continued

- Source must not have been damaged during its service in the GC-1000.

(2) Maximum quantity of material per package

113 TBq (3,050) Curies.

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) Each packaging must be acceptance tested and maintained in accordance with 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, until October 1, 2004, and under provisions of 10 CFR 71.17 thereafter.

Expiration date: June 30, 2009.

REFERENCES

MDS Nordion application dated May 27, 2003.

Supplements dated: April 16, July 16, July 21, and July 23, 2004.

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 July 27, 2004

NRC FORM 618 U.S. NUCLEAR REGULATORY COMMISSION
(8-2000)
10 CFR 71

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

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

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

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

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

a. ISSUED TO *(Name and Address)*

QSA Global, Inc.
40 North Avenue
Burlington, MA 01803

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

QSA Global, Inc., consolidated application dated December 6, 2005, 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.: 976 Series

(2) Description

The Model No. 976 Series packages are designed for use as transport packages for Type B quantities of radioactive material in special form. The Model No. 976 has six versions called the 976A, 976B, 976C, 976D, 976E and 976F. The Model 976A package contains a 855 shield container. The Model 976B package contains a 3015 shield container. The Model 976C package contains a 3056 shield container. The Model 976D package contains a 3018 shield container. The Model 976E package contains a 3078 shield container. The Model 976F package contains a 1911 shield container. All versions of the package include a 16 gauge stainless steel 20 gallon drum, four 3/8" - 16 UNC x ¾" long stainless steel lid closure bolts, a clamp band with M8 stainless steel bolt, and cork inserts to position and support the individual shield containers within the package. All Model No. 976 series packages measure 19 ¾" in diameter by 21 ¼" tall.

The shield containers are described as follows:

855 - 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 stainless steel hex head bolts. Approximately 11 ¼" in diameter at the base by 11 ¾" tall (without the eyebolt).

5(a) (2) (Description continued)

3015 - A lead shield container surrounded on the sides and partially on the top by an outer stainless steel jacket. The steel jacket incorporates two stainless steel lifting handles. The container includes a lower depleted uranium shielding insert encased in stainless steel, a tungsten capsule holder, an upper lead insert, a lead top shield plug with a stainless steel extension, and a gasketed shield lid which secures to the shield container body by two M10 stainless steel screws and washers. Measures approximately 7 ½" in diameter (including the handle bosses) by 10.1" tall.

3056 - A lead shield container which incorporates stainless steel strapping, handle bosses and lifting handles along with a combination lower depleted uranium insert and upper lead insert with ten stainless steel "J" tubes. The lead insert is partially enclosed by stainless steel. The "J" tubes are covered with tube caps and the tube caps are further covered by a stainless steel "top hat" or lid secured to the container by an M12 steel rod and retaining nut. Measures approximately 7.7" in diameter (including the handle bosses) by 10.4" tall.

3018 - A lead shield container surrounded on the sides and partially on the top by an outer stainless steel jacket. The steel jacket incorporates two stainless steel lifting handles. The container includes a lower depleted uranium shielding insert encased in stainless steel and upper lead insert with four stainless steel "J" tubes. The "J" tubes are covered with tube caps. The shield inserts are secured to the shield body by means of a stainless steel bracket and two M10 stainless steel bolts and washers. The metal bracket also incorporates a stainless steel disk above the "J" tubes which further protects the tube caps during shipment. Measures approximately 7 ½" in diameter (including the handle bosses) by 10.8" tall.

3078 - A stainless steel encased, depleted uranium shield container which includes two stainless steel lifting handles. The shield container incorporates a stainless steel encased depleted uranium upper shield plug that is inserted into the shield body over an optional stainless steel or aluminum source holder can. The upper shield insert is secured to the shield body by a stainless steel cover bolted above the shield insert by four M8 stainless steel screws. Measures approximately 6.1" in diameter by 8.4" tall.

1911 - A stainless steel encased, lead shield container which includes a bolted shield lid and an M10 stainless steel lifting eyebolt. The shield lid is secured to the shield container body by four stainless steel M8 bolts and washers. The inner shield cavity incorporates either a depleted uranium upper and lower shield insert, a tungsten upper and lower shield insert or a lead upper and lower shield insert. Additional handling source stainless steel, aluminum or tungsten capsule holders or cans may be used in the shield insert cavities. Measures approximately 8" in diameter by 8 ¾" tall (without the eyebolt).

5(a) (2) (Description continued)

The following table gives the maximum package weight.

Model No.	Maximum Package Weight (lbs)
976A	300
976B	190
976C	190
976D	190
976E	226
976F	263

(3) Drawings

This packaging is constructed in accordance with the following AEA Technology, Inc., Drawings.:

R97608, Rev. E, Sheet 1	20 gallon drum
RCLM009, Rev. B, Sheets 1-2	Band clamp
R97637, Rev. A, Sheet 1	Top inner cork spacer
97623, Rev. A, Sheet 1	Bottom inner cork insert
R97623A, Rev. A, Sheet 1	Bottom inner cork insert, alt.
R97615, Rev. B, Sheet 1	Top outer cork insert
R97616, Rev. B, Sheet 1	Bottom outer cork insert
R976A, Rev. D, Sheet 1	Model No. 976A with 855 shield container
R85590, Rev. E, Sheets 1-6	855 source changer
R976B, Rev. E, Sheet 1	Model No. 976B with 3015 shield container
R3015, Rev. C, Sheets 1-3	3015 shield container
R976C, Rev. E, Sheet 1	Model No. 976C with 3056 shield container
R3056, Rev. D, Sheets 1-4	3056 shield container
R976D, Rev. E, Sheet 1	Model No. 976D with 3018 shield container
R3018, Rev. C, Sheets 1-4	3018 shield container
R976E, Rev. E, Sheet 1	Model No. 976E with 3078 shield container
R3078, Rev. D, Sheets 1-4	3078 shield container
R976F, Rev. C, Sheet 1	Model No. 976F with 1911 shield container
R1911, Rev. C, Sheets 1-8	1911 shield container

(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

Model No.	Maximum Capacity - Output Activity* (Ci)	Maximum Capacity - Output Activity (TBq)
976A	1,000	37
976B	350	12.95
976C	800	29.6
976D	500	18.5
976E	1,000	37
976F	1,000	37

* Output curies are determined by measuring the source output at 1 meter and expressing its activity in curies derived from the following: 0.48 R/h-Ci Iridium-192 at 1 meter. (Ref: American National Standard N432-1980, "Radiological Safety for the Design and Construction of Apparatus for Gamma Radiography.")

6. Tensile and yield strength for the materials of construction must comply with the following values:

Material	Tensile Strength	Yield Strength
Depleted Uranium	65 ksi	30 ksi
Copper	25 ksi	9 ksi
Steel (nominal)	53 ksi	36 ksi
Stainless Steel	75 ksi	30 ksi
Tungsten	142 ksi	109 ksi
Cork (minimum)	80 psi	NA
Lead ($^{96}Pb/^{4}Sb$)	3,990 psi	NA

7. The sources shall be secured in the shielded positions of the packaging in accordance with the Package Loading requirements contained in Section 7 of the application, as supplemented. For "J" tube style shield containers, the flexible cable of the source assembly and source cap must be of sufficient length and diameter to provide positive positioning of the source in the shielded position.

8. The name plate must be fabricated of materials 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) The package shall be prepared for shipment in accordance with the Package Operations in Section 7 of the application, as supplemented, and,

   (b) Each packaging must be acceptance tested and maintained in accordance with the Acceptance Tests and Maintenance Program in Section 8 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 71.17.

11. Revision No. 1 of this certificate may be used until December 31, 2006.

12. Expiration date: June 30, 2010.

REFERENCES

QSA Global, Inc., consolidated application dated December 6, 2005.

Supplements dated December 13 and December 15, 2005.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Robert A. Nelson, Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: 12/15/05

NRC FORM 618 (8-2000) 10 CFR 71		CERTIFICATE OF COMPLIANCE FOR RADIOACTIVE MATERIAL PACKAGES		U.S. NUCLEAR REGULATORY COMMISSION
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2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

U.S. Department of Energy
Washington, DC 20585

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

BWXT Y-12, L.L.C., application dated February 25, 2005, 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.: ES-3100

(2) Description

The ES-3100 package is a cylindrical container that is approximately 110 cm (43 in) in overall height and 49 cm (19 in) in overall diameter and is composed of an outer drum assembly and an inner containment vessel. The containment vessel is placed inside the drum and surrounded by a cement based borated neutron absorber, Catalog 277-4. The purpose of the ES-3100 is to transport bulk high enriched uranium in oxide form, uranium metal and alloy, and uranyl nitrate crystals.

The outer drum assembly consists of a reinforced stainless steel, standard mil spec 30-gal drum with an increased length. The volume formed between the drum and the attached inner liner is filled with an inorganic, castable refractory material, Kaolite 1600™, which is comprised of concrete and vermiculite. The Kaolite 1600™ acts as both a thermal insulating and an impact limiting material.

The containment vessel is approximately 82 cm (32 in) in overall height and 13 cm (5 in) in overall diameter and is constructed of 304L stainless steel. The containment boundary consists of the 0.1 in thick containment vessel body and the lid assembly. The lid assembly consists of a sealing lid, a closure nut, and external retaining ring, which holds both the assembly and closure nut together. The double ethylene-propylene elastomer O-rings in the top flange of the containment vessel permit leak testing of the containment vessel. The maximum gross weight of the package, including contents, is 190.5 kg (420 lb).

5.(a) Packaging (continued)

(3) Drawings

The Model No. ES-3100 package is constructed and assembled in accordance with:

(i) BWXT Y-12, L.L.C., Drawing No. M2E801580A037, Sheets 1 through 6, Rev. A, "Consolidated Assembly Drawing."

(ii) BWXT Y-12, L.L.C., Drawing No. M2E801580A026, Rev. C, "Heavy Can Spacer Assembly."

(iii) Equipment Specification JS-YMN3-801580-A001, Rev. E, "ES-3100 Containment Vessel."

(iv) Equipment Specification JS-YMN3-801580-A002, Rev. D, "ES-3100 Drum Assembly."

(v) Equipment Specification JS-YMN3-801580-A003, Rev. B, "Manufacturing Process Specification for Casting Kaolite 1600™ into the ES-3100 Shipping Package."

(vi) Equipment Specification JS-YMN3-801580-A005, Rev. C, "Casting Catalog No. 277-4 Neutron Absorber for the ES-3100 Shipping Package."

5.(b) Contents (Type and form of material, maximum quantity of material per package, and Criticality Safety Index (CSI)).

The weight of the radioactive contents, convenience cans, can lift attachments, polyethylene bags, spacers, and other material in the containment vessel shall not exceed 90 lb. The maximum mass of hydrogenous packaging materials in the containment vessel (e.g., polyethylene containers or bagging, silicone rubber pads, etc.) shall not exceed 500 grams. The maximum content decay heat load shall not exceed 0.4 watts.

The concentration limits of uranium and transuranic constituents shall be the following:

Isotope	Maximum Concentration
U-232	0.040 μg/gU[a]
U-233	0.006 g/gU[b]
U-234	0.02 g/gU
U-235	1.00 g/gU
U-236	0.40 g/gU
Transuranics (except Np)	40.0 μg/gU
Np-237	0.003 g/gU

[a] μg/gU = 10^{-6} grams per gram of total uranium
[b] g/gU = grams per gram of total uranium

5.(b) Contents (continued)

(1) Uranium as solid metal or alloy, packaged in stainless-steel or tin-plated carbon steel convenience cans.

The maximum uranium enrichment is 100 weight percent U-235.

For contents that must be shipped with spacers, the spacers must be in accordance with BWXT Y-12, L.L.C., Drawing No. M2E801580A026 and Equipment Specification JS-YMN3-801580-A005, as specified in Condition No. 5.(a)(3). The quantity of fissile material in any convenience can shall not exceed one-third of the mass loading limit per package for that content. Spacers must be positioned between every two convenience cans.

(i) For metal and alloy in the form of solid geometric shapes, meeting the following restrictions; mass limits are listed in Table 1. Contents not meeting the following restrictions must be shipped as broken metal (see Condition No. 5.(b)(1)(ii)).

(A) Spheres having a diameter no larger than 3.24 in (maximum of two spheres per convenience can)

(B) Cylinders having a diameter no larger than 3.24 in (maximum of one cylinder per convenience can)

(C) Square bars having a cross section no larger than 2.29 in × 2.29 in (maximum of one bar per convenience can)

(D) Slugs having dimensions of 1.5 in diameter × 2 in tall (maximum of 10 slugs per convenience can)

Table 1: Loading Limits for Metal and Alloy in Solid Geometric Shapes

Solid uranium metal or alloy (specified geometric shapes)	Uranium Enrichment (weight percent U-235)	CSI	With Spacers Maximum Mass U-235 (kg)		No Spacers Maximum Mass U-235 Per Package (kg)
			Per Convenience Can	Per Package	
Spheres	≤ 100	0.0	10.979	32.938	16.946
Cylinders	≤ 100	0.0	6.000	18.000	12.000
Sq. Bars	≤ 100	0.0	10.000	30.000	18.000
Slugs	> 80	0.0	5.447	16.342	Spacer req'd
Slugs	≤ 80	0.0	8.738	26.213	Spacer req'd

5.(b)(1) Contents (continued)

(ii) For metal or alloy defined as broken metal, mass limits are specified in Table 2. Uranium metal and alloy pieces must have a surface-area-to-mass ratio of not greater than 1.00 cm^2/g or must have a mass not less than 50 g, whichever is most restrictive. Powders, foils, turnings, wires, and incidental small particles are not permitted, unless they are restricted to not more than 1 percent by weight of the content per convenience can, and they are either in a sealed, inerted container or are stabilized to an oxide prior to shipment.

Table 2: Loading Limits for Solid Metal or Alloy in the Form Defined as Broken Metal

Uranium Enrichment (weight percent U-235)	CSI	With Spacers Maximum Mass U-235 (kg)[a]		No Spacers Maximum Mass U-235 Per Package (kg)[a]
		Per Convenience Can	Per Package	
> 95 and ≤ 100	0.0	0.925	2.774	Spacer req'd
	0.4	1.849	5.548	Spacer req'd
	0.8	2.774	8.323	Spacer req'd
	2.0	3.699	11.097	Spacer req'd
> 90 and ≤ 95	0.0	0.879	2.637	Spacer req'd
	0.4	1.758	5.274	Spacer req'd
	0.8	3.516	10.549	Spacer req'd
	2.0	5.568	16.703	Spacer req'd
> 80 and ≤ 90	0.0	0.833	2.500	Spacer req'd
	0.4	2.500	7.500	Spacer req'd
	0.8	3.333	10.000	Spacer req'd
	2.0	5.278	15.834	Spacer req'd
> 70 and ≤ 80	0.0	0.742	2.225	2.225
	0.4	2.967	8.900	4.450
	0.8	6.181	18.542	14.092
	2.0	7.911	23.734	18.542

5.(b)(1) Contents (continued)

Table 2: Loading Limits for Solid Metal or Alloy in the Form Defined as Broken Metal (Continued)

Uranium Enrichment (weight percent U-235)	CSI	With Spacers Maximum Mass U-235 (kg)[a]		No Spacers Maximum Mass U-235 Per Package (kg)[a]
		Per Convenience Can	Per Package	
> 60 and ≤ 70	0.0	1.949	5.848	1.949
	0.4	4.115	12.346	7.797
	0.8	6.931	20.793	16.245
	2.0	8.231	24.692	24.692
≤ 60	0.0	3.718 kgU	11.153 kgU	5.576 kgU
	0.4	9.914 kgU	29.743 kgU	17.660 kgU
	0.8	11.773 kgU	35.320 kgU	35.320 kgU
	2.0	11.773 kgU	35.320 kgU	35.320 kgU

[a] All limits are expressed in kg U-235 unless specified as kgU, which means kilograms of total uranium.

(2) *Uranium as oxide, which may include UO_2, UO_3, and U_3O_8, packaged in stainless-steel or* tin-plated carbon-steel convenience cans. The physical form of all contents is dense, loose powder which may contain clumps and pellets. Moisture content in oxide is limited to 3 weight percent water. The mass limit shall be 24.0 kg of oxide, with a maximum mass of 21.124 kg U-235, with a CSI of 0.0. No spacers are required in the containment vessel.

(3) Solid uranyl nitrate in the form of uranyl nitrate crystals, [$UO_2(NO_3)_2 \cdot xH_2O$, where x is ≤ 6]. Uranyl nitrate crystals must be contained in a non-metallic convenience container (such as Teflon or polyethylene bottle). The mass limit shall be 24.0 kg of uranyl nitrate crystals, with a maximum mass of 11.303 kg U-235, with a CSI of 0.0. No spacers are required in the containment vessel.

6. The vent holes on the outer steel drum shall be capped closed during transport and storage to preclude entry of rain water into the insulation cavity of the drum.

7. Content forms may not be mixed in a single ES-3100 containment vessel.

8. Any combination of convenience can sizes is allowed in a single package, as long as the total height of the can stack (including silicone rubber pads and spacers, if required) does not exceed the inside working height of the containment vessel (31 in). Any closure on the convenience can is allowed.

9. Empty convenience cans, spacers, silicone rubber pads, and/or stainless-steel scrubbers (i.e., stainless steel trimmings that act as dunnage) may be used to fill the void space in the containment vessel. Empty convenience cans must have a minimum 0.125 in diameter hole through the lid.

10. The contents and the convenience cans may be bagged or wrapped in polyethylene for contamination control provided the limits of Condition No. 5.(b) are met.

11. Transport by air is not authorized.

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 the Package Operations in Section 7 of the application, as supplemented.

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

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

14. Expiration date: April 30, 2011.

REFERENCES

BWXT Y-12, L.L.C., application dated February 25, 2005, as supplemented.

BWXT Y-12, L.L.C., supplements dated April 27, May 26, August 15, 2005; and January 9, February 6, and March 20, 2006.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Robert A. Nelson, Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: April 7, 2006

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
9511	3	71-9511	USA/9511/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)*
U.S. Department of Energy
Washington, DC 20545

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION
U. S. Department of Energy
application dated February 26, 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: BUSS R-1

(2) Description

The packaging is a cylindrical, forged stainless steel cask. The cask body is a one-piece forging, 54.25 inches OD by 49 inches high. The cask cavity is 20.25 inches in diameter by 23 inches high. A solid, stainless steel basket, 19.95 inches in diameter by 22.83 inches high, sits in the cask cavity. The basket has either four, six, twelve, or sixteen 2.875-inch diameter holes that serve as receptacles for the source capsules. Eleven 4-inch high, circumferential fins surround the cask body exterior. A covered vent port and a covered drain port are located on the side of the cask body. The cask lid is a one-piece forging, 28.78 inches in diameter by 12.84 inches thick. Twelve 1.5-inch diameter bolts fasten the cask lid to the cask body through a 3.8-inch thick flange. The cask lid and port covers each have concentric, double O-rings. The inner O-ring is metallic and retains the helium coolant which fills the cask cavity. The outer O-ring is elastomeric and provides an annular test volume for leak testing the metallic O-ring. The cask has an impact limiter on each end. The impact limiter is polyurethane foam in a stainless steel shroud.

The overall dimensions of the packaging with impact limiters are 84.7 inches in diameter by 107 inches high. The maximum total weight of the contents is 400 pounds. The maximum weight of the package, including contents, is 30,000 pounds. The shipping skid and personnel barrier, which are not part of the package, weigh an additional 3,700 pounds.

5.(a) Packaging (continued)

(3) Drawings

The packaging is constructed in accordance with the following drawings:

Drawing No.	Title
S54773, Sht. 1, Rev. B	Cask with Impact Limiters
S48981, Sht. 1, Rev. H	Cask Assembly
T73684, Sht. 1, Rev. N, and Sht. 2, Rev. M	Body, Cask, 304 (BUSS)
R44382, Sht. 1, Rev. B	Alternate Detail N for Upper Port of Unit 1, Heat No. 82V65-1-1
T73693, Sht. 1, Rev. M	Cask Lid (BUSS) 304 SST
S66574, Sht. 1, Rev. B	Bolt, Tension, 12 Point External Wrenching , Flanged
T99946, Sht. 1, Rev. E	Seal, Helicoflex, Cask Lid (BUSS)
T73685, Sht. 1, Rev. E	Plug, Drain (BUSS)
T99945, Sht. 1, Rev. D	Seal, Helicoflex, Drain Plug (BUSS)
R44676, Sht. 1, Rev. A	Bore Plug Assembly, Cask Body
R43728, Sht. 1, Rev. A	Bore Plug, Cask Body
S48929, Shts. 1 and 2, Rev. G, Sht. 3, Rev. E	Impact Limiter BUSS
R44381, Sht. 1, Rev. B, and Sht. 2, Rev. A	Impact Limiter BUSS, Non Conformance
S50032, Sht. 1, Rev. D	Cradle, BUSS Cask
S52606, Shts. 1 and 2, Rev. C	Pallet
S52608, Sht. 1, Rev. C	Block, Mounting
S50052, Sht. 1, Rev. F	Basket, Cask Body, 4 Hole (BUSS)
S50053, Sht. 1, Rev. E	Basket, Cask Body, 6 Hole (BUSS)
S50054, Sht. 1, Rev. D	Basket, Cask Body, 12 Hole (BUSS)
S50055, Sht. 1, Rev. E	Basket, Cask Body, 16 Hole (BUSS)

5.(b) Contents

(1) Type and form material.

Melt-cast cesium chloride (CsCl) or pressed-filled strontium fluoride (SrF_2) capsules meeting the requirements of special form radioactive material. The capsules are as described in supplement dated February 28, 1992.

5.(b) Contents (continued)

(2) Maximum quantity of material per package.

Basket Type	Capsule Type	Maximum Capsule Thermal Power (Watts)	Maximum Cask Thermal Power (Watts)	Maximum Cask Activity (million Ci)
16-Hole	CsCl	250	4000	0.85
12-Hole	CsCl	333	4000	0.85
6-Hole	SrF_2	650	3900	0.65
4-Hole	SrF_2	850	3400	0.56

6. For shipments of CsCl capsules, the shipment period must be completed within thirty (30) days following the placement of the cask lid on the cask body.

7. The lifting lugs must not be used as tie-downs, and the lifting lug holes must be plugged or covered during transit.

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

   (a) The 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, as supplemented.

9. The package authorized by this certificate must be transported on a conveyance assigned for the exclusive use of this package.

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: July 31, 2007.

REFERENCES

U. S. Department of Energy application dated February 26, 1991.

Supplements dated February 28, 1992; May 6, 1994; August 26, 1994; July 29, 1997; and June 7, 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: 07/26/02

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
9786	6	71-9786	USA/9786/B(U)	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, D.C. 20585

   b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

   S3G Core Basket Disposal Container
   Safety Analysis Report for Packaging
   dated June 1980, 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.: S3G Core Basket Disposal Container Assembly

(2) Description

The package consists of either one irradiated S3G, S1C or S7G core basket packaged in an inner, lead-filled container (S5W Core Basket Removal Container (CBRC)) which is placed inside an outer container (S3G Core Basket Disposal Container (CBDC)). The package weighs approximately 172,000 pounds.

The S3G CBDC is a 4-inch thick steel cylinder, 89 inches in outside diameter, 131 inches long, with an 8-inch thick top end plate and a 5-inch thick bottom end plate. Both end plates are welded to the cylinder with full penetration welds.

The S5W CBRC, which will be disposed of along with the outer S3G CBDC and inner core basket, is basically a cylindrical shaped container comprised of lead shielding sandwiched between two 304 stainless steel shells. The 1-inch thick inner shell is 60 inches O.D. and 107.5 inches long. The outer shell is made up of two geometries, a 72.5-inch O.D., 0.5-inch thick cylindrical shell that measures 66 inches long and joins a truncated conical shell which has a 64-inch O.D. at the small end. The two shells are joined by a full thickness penetration weld and a weld backup strap on the inside shell surface. Full penetration welds are also made on both ends of the shells to the top canning and shield ring.

The S5W CBRC will contain either an S3G, S1C or S7G core basket. The irradiated S3G core basket is an Inconel 600 cylindrical shell. Three, 3-inch thick 304 stainless steel plates are positioned in the core basket prior to removal to provide overhead radiation shielding. The lower plate is 46.2 inches in diameter. The upper plates have the same diameter but contain six extensions that fit inside recessed cutouts within the core basket. The total core basket weight is approximately 9,650 pounds.

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

The S1C core basket is a 304 stainless steel cylindrical shell positioned inside a 304 stainless steel thermal shield. The overhead shielding consists of a set of 2-inch thick 304 stainless steel plates attached to the S1C core basket to provide radiation shielding during handling. The core basket weight is approximately 8,523 pounds.

The S7G core basket is an Inconel 600 cylindrical shell. A 304 stainless steel laminated plate (8-inches thick) with lifting attachments is attached to the top of the S7G core basket to provide radiation shielding during handling. The core basket weight is approximately 8,873 pounds.

The package may alternatively consist of S8G irradiated components positioned within an irradiated components discharge rack (ICDR) which is placed in an S3G CBDC. The ICDR is a steel rack approximately 128 inches high and 80 inches in diameter, and is designed to fit inside the S3G CBDC. The ICDR consists of a center cylinder assembly surrounded by 23 storage tubes, a top plate and a cylinder support base. The center cylinder is HY-80 steel, has a 36-inch outer diameter and a 4.5-inch wall thickness, and is 117 inches high. There are 9 storage tubes positioned inside the center cylinder. The total weight of the irradiated components, the ICDR, and the S3G CBDC is approximately 125,000 pounds.

(3) Drawings

The packaging is constructed in accordance with Bettis Drawing No. 1527E40 for the S3G Core Basket Assembly and KAPL Drawing No. 152D7009 for the S1C Core Basket Assembly and KAPL Drawing No. 232B4874 for the S7G Core Basket Assembly and KAPL Drawing No. 978E644 for the S8G Irradiated Components.

(b) Contents

(1) Type and form of material

(i) An irradiated core basket either the S3G, S1C or S7G and S5W CBRC. The shipment may include surface contamination in the form of activated corrosion products and for the S3G core basket approximately 8 gallons of residual water.

(ii) S8G irradiated components within an ICDR. The shipment may include surface contamination in the form of activated corrosion products.

(2) Quantity of material per package

(i) Item 5(b)(1)(i) above:

One irradiated core basket and S5W CBRC as described in 5(b)(1). Surface contamination not to exceed 20.6 curies for the S3G core basket, 7.45 curies for the S1C core basket or 1.2 curies for the S7G core basket. The activation level of the irradiated S3G core basket is not to exceed 131,000 curies; the irradiated S1C core basket not to exceed 20,000 curies; and the activation level of the irradiated S7G core basket is not to exceed 140,000 curies.

(ii) Item 5(b)(1)(ii) above:

Irradiated components, including 141 instrument lines, 18 lower control drive mechanism assemblies, 4 filled sleeves, and 1 instrumentation stalk. Surface contamination not to exceed 65.5 curies. Activation level of the irradiated components not to exceed 2,440 curies.

6. Shipment of an irradiated S3G core basket must be made no earlier than 75 days after reactor shutdown.

7. Shipment of an irradiated S1C core basket must be made no earlier than 60 days after reactor shutdown.

8. Shipment of an irradiated S7G core basket must be made no earlier than 180 days after reactor shutdown.

9. Shipment of S8G irradiated components must be made no earlier than 100 days after reactor shutdown.

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

(a) Each packaging must meet the following Acceptance Tests and Maintenance Program:

<u>S3G Core Basket</u>

Section 8.0 of application dated June 1980

<u>S1C Core Basket</u>

Section 8.0 of application dated August 1983

<u>S7G Core Basket</u>

Section 8.0 of application dated May 1987

<u>S8G Irradiated Components</u>

Section 8.0 of application dated September 1991

(b) The package shall be prepared for shipment and operated in accordance with the following operating procedures:

S3G Core Basket

Section 7.0 of application dated June 1980

S1C Core Basket

Section 7.0 of application dated August 1983

S7G Core Basket

Section 7.0 of application dated May 1987

S8G Irradiated Components

Section 7.0 of application dated September 1991

11. Revision No. 5 of this certificate may be used until April 30, 2007.

12. Expiration date: August 31, 2011

REFERENCES

S3G Core Basket Disposal Container Safety Analysis Report for Packaging, WAPD-REO(C)-122, dated June 1980, as revised (Revision 2, dated May 5, 1986).

Safety Analysis Report for Packaging an S1C Core Basket-Thermal Shield Assembly in the S3G Core Basket Disposal Container, S1C CB-TS, dated August 1983.

S7G Core Basket in the S3G Core Basket Disposal Container Safety Analysis Report for Packaging, dated May 1987.

S8G Irradiated Components in the S3G Core Basket Disposal Container Safety Analysis Report for Packaging, Revision 2, dated September 1991.

DOE memorandums G#7627 dated November 16, 1983; G#C86-3736 dated May 24, 1986; G#C86-3750 dated July 15, 1986; G#87-5663 dated July 7, 1987; G#91-10937 dated July 31, 1991; G#C91-11007 dated September 18, 1991; G#96-03385 dated February 16, 1996; G#01-03414 dated January 31, 2001; and G#06-01024 dated March 7, 2006.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

James R. Hall for

Robert A. Nelson, Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material
Safety and Safeguards

UNITED STATES NUCLEAR REGULATORY COMMISSION

Date: May 12, 2006

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
9788	13	71-9788	USA/9788/B(U)-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)*

U.S. Department of Energy
Division of Naval Reactors
Washington, DC 20585

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Deactivated S5W Reactor Compartment Safety Analysis Report for packaging, dated 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 Nos.: S5W Reactor Compartment and SSN 688 Class Reactor Compartment

(2) Description

The package consists of a deactivated and defueled S5W or SSN 688 Class reactor compartment which has been separated from the remainder of the submarine hull and prepared for shipment by sealing all openings and attaching handling fixtures. For each package model, the reactor compartment itself is between two containment bulkheads which are added to the package before shipping. The ship's hull and the containment bulkheads define the package containment boundaries. The containment bulkheads are either installed at the ends of the package or recessed. The strength of all package boundary closures is at least equivalent to the strength of the bulkheads. The deactivated reactor plant remains in place within the reactor compartment during shipment. The plant is defueled and drained except for small inaccessible pockets of liquid, primarily water. Potentially radioactively contaminated components and piping from other locations in the ship may be placed within the package and secured.

5. (a) Packaging (Continued)

(2) Description (Continued)

The S5W Reactor Compartment package is between 35 and 45 feet long and approximately cylindrical with a maximum diameter of approximately 33 feet. The containment bulkheads are made of HS steel. The bulkheads may be installed at the ends of the package or may be recessed. The forward containment bulkhead may include existing ship structure which has been sealed to form a watertight bulkhead. The hull is constructed of HY-80 steel. The maximum weight of the S5W package is 2,160,000 pounds for the 598 and 585 classes and is 2,262,400 pounds for all other classes.

The SSN 688 Class Reactor Compartment package is approximately 46 feet long and approximately cylindrical with a maximum diameter of approximately 33 feet. The containment bulkheads are made of HS steel. The bulkheads may be installed at the ends of the package or may be recessed. The hull is constructed of HY-80 steel. The maximum weight of the package is 3,360,000 pounds.

(3) Drawings

The package is constructed in accordance with the drawings, figures, and sketches included in the application, as supplemented (see References, below).

(b) Contents

Activated structural components associated with the S5W and SSN 688 Class reactor vessel complex, plant piping, ion exchanger resin, purification filter media (SSN 688 Class only), residual liquid and other miscellaneous components and materials contaminated with radioactive corrosion products (crud).

6. Residual liquids contained within plant systems must be removed prior to transport to the maximum extent practical, in accordance with established procedures, methods, and controls, as described in submittal dated April 5, 1996, or in the Safety Analysis Report for the individual submarine class reactor compartment packages. Not more than 660 gallons of residual liquids remain in the S5W Reactor Compartment package, and not more than 1,200 gallons of residual liquids remain in the SSN 688 Class Reactor Compartment package.

7. For packages with recessed containment bulkheads, the aft containment bulkheads and stiffeners, horizontal divider plate, and any structure between the pressure hull and the outer non-pressure hull must be recessed at least 7 inches from the aft end of the S5W package. The forward containment bulkhead and stiffeners, existing tank stiffeners, deck structure, and horizontal girders must be recessed at least 15 inches from the forward end of the S5W package. For SSN 688 Class packages with recessed containment bulkheads, both the aft and forward containment bulkheads, stiffeners and horizontal girders must be recessed at least 15 inches from the end of the package.

8. The Lowest Service Temperature (LST) must be determined for each package. The package shall not be shipped unless its LST is less than or equal to the normal daily minimum temperature expected during the shipment of the package as determined on the basis of weather forecasts.

9. Ion exchanger resin with up to 3.1 curies (1.1×10^{11} becquerels) of Co-60 may be shipped in the S5W package. Shipment of the S5W packages shall not occur before 180 days after final reactor shutdown.

10. For SSN 688 Class packages, the Co-60 curie content of the ion exchanger resin and purification filter media shall be limited as described in Table 4-2 of the supplement dated March 15, 2002. For packages that have supplemental shielding as described in Appendix 2.10.14 of the supplement dated March 15, 2002, alternative radioactivity limits for the ion exchanger resin and purification filter media may apply, as specified in Table 2.10.14-1 of Appendix 2.10.14.

11. Shipment of the SSN 688 Class packages shall not occur before 365 days after final reactor shutdown.

12. Additional shielding may be provided on the exterior of the package by steel plates securely welded to the package surface so as to remain in place under the hypothetical accident conditions in 10 CFR Part 71.

13. 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.

    (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.

14. Expiration date: September 30, 2008.

REFERENCES

Deactivated S5W Reactor Compartment Safety Analysis Report for Packaging, WAPD-REO(C)-250, dated July 1981.

Supplements: Naval Reactors Memorandum Z#C90-14416 dated March 29, 1990, supplement dated July 6, 1990;
Naval Reactors Memorandum Z#C90-14456 dated August 30, 1990;
Naval Reactors Memorandum Z#C92-14438 dated August 3, 1992;
Naval Reactors Memorandum Z#C93-00069 dated October 14, 1993;
Naval Reactors Memorandum Z#C95-00113 dated March 16, 1995;
Naval Reactors Memorandum Z#96-14430 dated April 5, 1996;
Naval Reactors Memorandum Z#96-14434 dated April 10, 1996;
Naval Reactors Memorandum Z#C95-00191 dated December 14, 1995;
Naval Reactors Memorandum Z#96-14457 dated June 20, 1996;
Naval Reactors Memorandum Z#C96-14520 dated November 22, 1996;
Naval Reactors Memorandum Z#C96-14549 dated December 19, 1996;
Naval Reactors Memorandum Z#C97-14698 dated October 31, 1997;
Naval Reactors Memorandum Z#C98-00021 dated February 27, 1998; and
Naval Reactors Memorandum Z#C02-03057 dated March 15, 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: 02/28/2003

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
9791	6	71-9791	USA/9791/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
   Division of Naval Reactors
   Washington, DC 20585

   b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

   PWR-2 Lower Core Barrel Safety Analysis Report for Packaging dated January 1982, 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.: PWR-2 Lower Core Barrel Shipping and Disposal Container

(2) Description

The PWR-2 Lower Core Barrel Shipping and Disposal Container package consists of an inner burial container and a reusable outer container. The inner container is loaded with a D1G prototype pressure vessel assembly. The package weighs approximately 400,000 pounds.

The outer container is a 4-inch thick steel cylinder, 127 inches in outside diameter, 212 inches long, with two 6-inch thick end plates. The bottom end plate is welded to the cylinder with a full penetration weld and the top end plate is bolted with 107, 2-inch diameter fasteners.

The package is equipped with two 2.5-inch thick by 10-inch long circumferential impact limiter rings on the side, two concentric impact limiter rings on the ends, and aluminum honeycomb crush blocks in the top and bottom spaces between the inner and outer containers.

The container is supported horizontally on the railroad car by eight gussets attached to two horizontal plates. Each plate is bolted to the top flange of an I-beam. The bottom flange of the I-beam is bolted to a 300-ton railroad car.

5. (a) Packaging (continued)

The inner disposal container (liner) is of the following design:

The D1G prototype pressure vessel assembly has an inner burial container that consists of two cylinders constructed of HY-80 steel connected by a transition ring that is welded to the two cylinders. The maximum outer diameter of the cylinder is approximately 118 inches at the upper flange. The overall length of the inner container is 184.5 inches. The container wall is 3.12 inches in the upper cylinder and 4 inches in the bottom cylinder. The bottom plate varies in thickness from 6 to 2.4 inches and is attached to the container by 12, 4.5-inch thick gussets. The cover plate is approximately 10 inch thick and is attached to the container by a 3.25-inch thick closure weld. The container is axially positioned within the outer container by aluminum honeycomb energy absorbers.

(3) Drawings

The packaging is constructed in accordance with Westinghouse Drawing Nos. 1575E12, 1574E96, 6236E43, Sheets 1 through 3, Rev. A, and 6236E44, General Electric Drawing Nos. 977E709 and 977E467, and KAPL, Inc. Drawing Nos. 108E6847and 108E6846.

(b) Contents

(1) Type and form of material

An irradiated D1G prototype pressure vessel assembly, including pressure vessel, core barrel, thermal shields, and two surveillance train assemblies. In addition, the contents may include surface contamination in the form of activated corrosion products and 119 gallons of residual water.

(2) Quantity of material in package

One irradiated D1G prototype pressure vessel assembly. Surface contamination not to exceed 4.61 curies. Displaced material from cutting operations not to exceed 10.6 curies. The irradiated components not to exceed 60,000 curies.

6. The package will be operated in accordance with the procedures described in Chapter 7 of the application and in accordance with Naval Reactors letter G#C97-03596 dated August 28, 1997, or G#C98-10723 dated February 13, 1998. The package will be tested and maintained in accordance with the procedures in Chapter 8 of the application and in accordance with Naval Reactors letter G#C97-03596 dated August 28, 1997, or G#C98-10723 dated February 13, 1998.

7. Expiration date: July 31, 2007.

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
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REFERENCES

PWR-2 Lower Core Barrel Safety Analysis Report for Packaging, WAPD-LP(CES)CS-670 dated January 1982.

Supplements: Naval Reactors letters G#7241 dated December 2, 1982, G#84-452 dated March 28, 1984, G#C92-03331 dated January 29, 1992, G#92-03546 dated June 5, 1992, G#92-03589 dated July 2, 1992, G#97-053513 dated June 11, 1997, G#C97-03596 dated August 28, 1997, G#C98-10723 dated February 13, 1998, G#98-10801 dated May 5, 1998, and G#02-0688 dated January 16, 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

ate: 1 May, 2002

UNITED STATES NUCLEAR REGULATORY COMMISSION

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
9792	7	71-9792	USA/9792/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)*

U.S. Department of Energy
Division of Naval Reactors
Washington, DC 20585

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Department of Energy application dated April 22, 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.: Model 1 D1G Core Basket-Thermal Shield Shipping and Storage Container

(2) Description

The Model 1 D1G Core Basket-Thermal Shield (CB-TS) Shipping and Storage Container is a right circular cylinder approximately 115 inches in diameter and either 209 inches long (D1G design including impact limiter assembly) or 216 inches long (D2W design including impact limiter assembly). Access for loading is provided by a removable closure head. The container, consisting of the cylindrical side walls and the bottom end, has a three layer construction with a steel inner vessel approximately eight inches thick covered with approximately nine inches of reinforced concrete which is encased by a 3/8-inch thick outer shell. The CB-TS is secured in place inside the container with an 8-inch thick steel preload ring which is bolted to the inner vessel with 72 high strength bolts.

Closure of the containment vessel is provided by the 6-inch thick steel closure head which is fastened to the inner vessel with 72 high strength bolts. A steel closure ring is welded over the bolts and provides containment. A carbon steel inner impact limiter is welded to the top end of the closure ring. A wood outer impact limiter is bolted to the top plate of the container outer shell.

For land transport, the shipping container is transported with its axis horizontal and is supported by a shipping skid. For sea transport, the shipping container is transported with its axis vertical and is supported by a shipping frame assembly. The loaded container weighs up to 185 tons.

5. (a) (3) Drawings

Packagings for which fabrication was begun before March, 1991, are constructed in accordance with the General Electric Company Drawings contained in Appendix 2.10.4 of the application, and packagings for which fabrication was begun after March, 1991, are constructed in accordance with the KAPL Drawings for the redesign configuration in Appendix 2.10.4 of the application.

(b) Contents

One irradiated D1G core basket-thermal shield assembly, and not more than one core's worth of irradiated D1G support assemblies, D1G lower control rod drive mechanisms, and D1G upper support assemblies; surface contamination in the form of activated corrosion products; and not more than 3.5 gallons of residual water.

6. (a) Preloading of the preload plate and the closure head and sealing the container must be done with a temperature at or above +40 °F.

(b) Shipment of containers S/N 0000001 through 0000007 shall be made only when the average daily temperature is expected to be above +40 °F. Shipment of containers S/N 000008 through 000019 and S/N N00020 through N00031 shall be made only when the average daily temperature is expected to be above +10 °F, subject to the following exception: shipment of any container with the closure head identified as 04241-171D6617 P5, SER N00031 (Forging S/N BG-7140) shall be made only when the average daily temperature is expected to be above +30 °F.

(c) The D1G CB-TS Shipment shall be made no earlier than 150 days after shutdown of the reactor.

7. The package shall be prepared for shipment and operated in accordance with the Operating Procedures in Chapter 7.0 of the application, and each packaging shall be tested and maintained in accordance with the Acceptance Tests and Maintenance Program in Chapter 8.0 of the application.

8. For sea transport, the supplemental operating procedures and acceptance tests in Sections 11.0 and 12.0 of the submittal dated April 5, 2002, shall be used.

9. Expiration Date: September 30, 2007.

<u>REFERENCES</u>

Department of Energy, Division of Naval Reactors, application dated April 22, 1991.

Supplements dated: Naval Reactors Letters G#92-03668, dated August 27, 1992; G#C95-10762, dated April 10, 1995; G#C96-03576, dated November 1, 1996; and G#C02-0751, dated April 5, 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

ate: February 5, 2003

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
9793	12	71-9793	USA/9793/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
Division of Naval Reactors
Washington, D.C. 20585

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

"Core Independent M-140 Safety Analysis Report for Packaging," transmitted February 27, 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.: M-140

(2) Description

The M-140 is a stainless steel cask for transporting spent fuel. The cask is a right circular cylinder and is transported in the upright position. The package's approximate dimensions and weights are as follows:

Cavity diameter	70 inches
Cavity height	46 inches
Body outer diameter	98 inches
Body steel wall thickness	14 inches
Package overall outer diameter	126 inches
Package overall height	194 inches
Packaging weight, including standard internals	315,000 pounds
Maximum package weight, including contents	375,000 pounds

The cask body is made from 304 stainless steel forgings. The cask walls are 14-inches thick and the bottom plate is 12-inches thick. The cask body flange provides a seating surface for the closure head and its protective dome. The flange contains 36 wedge assemblies located radially around the inside diameter. Retention of the closure head is achieved by engaging the wedges in a tapered groove in the circumferential edge of the closure head. The cask body has 180 external cooling fins welded to the exterior wall. A support ring is welded to the external cooling fins at a point above the center of gravity. The support ring seats on, and is

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

bolted to, the railcar mounting ring during transport. For sea transport, the center section of the railcar is shipped with the package and serves as the package support structure. The cask bottom is equipped with an energy absorber which is composed of five concentric stainless steel rings varying in thickness and height.

The closure head is made from forged 304 stainless steel and is approximately 13-inches thick and 81.7 inches in diameter. The closure head is equipped with an access port, which is approximately 24 inches in diameter, and is offset from the center of the closure head. The access port plug is a stepped design with a maximum diameter of approximately 31 inches and is attached to the closure head by 24 bolts. The closure head and access port are sealed with double ethylene propylene O-ring seals. Seal test ports are provided for the closure head and access port seals. A stainless steel protective dome is positioned over the closure head and is secured to the cask body flange by 12, 1.38-inch diameter, 38.5-inch long studs installed in a vertical direction and 6, 2.5-inch diameter, 9-inch long shear bolts installed in the radial direction.

The containment system is composed of the cask body, the closure head, and the closure head access port plug. There are seven penetrations in the standard containment system: a closure head, a drain port, a vent port, and an access port in the closure head, a thermocouple penetration, a water inlet penetration, and a water outlet penetration in the cask body. Each penetration is sealed with a plug and a double ethylene propylene O-ring seal and is equipped with a leak test port. For some shipping configurations, two additional penetrations may be present in the closure head: a removable fuel assembly (RFA) access port and another vent penetration.

The spent fuel modules are positioned in an internals assembly. The internals assembly is composed of stacked internal spacer plates which have openings for the spent fuel modules. The internals assembly has a top plate or top plate subassembly which is preloaded by springs against a retaining ring fitted in a groove in the cask cavity wall. The internals assembly may be a standard, Type 1, Type 2 or Type 3 internals assembly.

(3) Drawings

The packaging is constructed and assembled in accordance with the Westinghouse Electric Corporation Drawings in Appendix 1.3.2 of the application. Internals assemblies and fuel modules are constructed and assembled in accordance with drawings in Chapter 1 of the applicable Safety Analysis Reports for Packaging.

5.(b) *Contents*

(1) Type and form of material

Spent fuel, limited to the following types, including associated activated corrosion products:

(i) S3G-3 spent fuel.

(ii) S8G spent fuel.

(iii) D1G Core 2 spent fuel.

(iv) D2W spent fuel.

(v) A1G spent fuel.

(vi) S6W spent fuel.

(vii) S9G spent fuel.

(2) Maximum quantity of material per package

Total package weight, including spent fuel and internals assembly, not to exceed 375,000 pounds; and

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

S3G-3 spent fuel modules, not to exceed 62,300 Btu/hr decay heat per package.

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

S8G spent fuel, not to exceed 51,609 Btu/hr decay heat per package (prototype spent fuel modules), or 45,713 Btu/hr decay heat per package (shipboard modules).

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

D1G-2 spent fuel with thermal limits as determined either by calculation of the wet hold time using Curve B from Figure 3-5 of the Safety Analysis Report for Packaging or by use of a shielding hold time from 8(b) below, whichever hold time is greater.

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

(iv) For contents described in 5(b)(1)(iv):

D2W spent fuel modules, not to exceed 63,000 Btu/hr decay heat per package for prototype spent fuel, 53,000 Btu/hr decay heat per package for shipboard Type 3 spent fuel modules, or 45,900 Btu/hr decay heat per package for shipboard Type 5 spent fuel modules.

(v) For contents described in 5(b)(1)(v):

A1G spent fuel with thermal limits as determined either by calculation of the wet hold time using Curve C from Figure 3-5 of the Safety Analysis Report for Packaging or an administrative hold time of 50 days, whichever hold time is greater.

(vi) For contents described in 5(b)(1)(vi):

S6W spent fuel modules, not to exceed 46,011 Btu/hr decay heat per package for a shipboard core or 47,160 Btu/hr for a prototype core at the time of container draining.

(vii) For contents described in 5(b)(1)(vii):

S9G spent fuel modules, not to exceed 55,002 BTU/hr decay heat per package at the time of container draining.

(c) Criticality Safety Index

Spent fuel module	Criticality Safety Index
S3G-3	100
S8G	100
D1G Core 2	100
D2W	100
A1G	0
S6W	100
S9G	0

6. For S3G-3 spent fuel shipments:

(a) Authorized fuel loadings, internals assembly, and other loading restrictions are specified in Section 1.2 of the Safety Analysis Report for Packaging.

(b) Minimum fuel cooling time is 130 days after shutdown.

(c) Core age must be at least 4,000 logging corrected full power hours.

(d) Control rod hold-down devices must be installed on cells which have control rods. Module grapple adapters serve as poison shipping rod holddown devices for refueling shipments.

7. For S8G spent fuel shipments:

(a) Authorized fuel loadings, internals assembly, and other loading restrictions are specified in Section 1.2 of the Safety Analysis Report for Packaging.

(b) Minimum fuel cooling time is 248 days after shutdown for prototype modules and 157 days after shutdown for shipboard modules.

(c) Full and partial fuel modules may be shipped in any combination, but all modules must be shipped with control rods.

(d) Control rod holddown devices must be installed on the cells. Module grapple adapters serve as control rod holddown devices.

8. For D1G Core 2 spent fuel shipments:

(a) Authorized fuel loadings, internals assembly, and other loading restrictions are specified in Section 1.2 of the Safety Analysis Report for Packaging.

(b) The minimum cooling time shall be the greater of 90 days for rail transport, 105 days for ship transport, or that calculated from Curve B of Figure 3-5 of the Safety Analysis Report for Packaging.

(c) Control rod holddown devices must be installed on rodded modules. The universal grapple adapters serve as the control rod holddown devices.

9. For D2W spent fuel shipments:

(a) Authorized fuel loadings, internals assembly, and other loading restrictions are specified in Section 1.2 of the Safety Analysis Report for Packaging.

(b) Minimum fuel cooling time is 180 days after shutdown.

(c) Control rod holddown devices must be installed on all rodded modules. The universal grapple adapters serve as the rod holddown devices.

10. For A1G spent fuel shipments:

    (a) Authorized fuel loadings, internals assembly, and other loading restrictions are specified in Section 1.2 of the Safety Analysis Report for Packaging.

    (b) All fuel clusters must be shipped with either control rods or poison shipping rods, with rod holddown devices installed.

    (c) Minimum fuel cooling time shall be the greater of 50 days after shutdown or that calculated using Curve C from Figure 3-5 of the Safety Analysis Report for Packaging.

11. For S6W spent fuel shipments:

    (a) Authorized fuel loadings, internals assembly, and other loading restrictions are specified in Section 1.2 of the Safety Analysis Report for Packaging.

    (b) The minimum fuel cooling time before container draining shall be 300 days after shutdown for a shipboard core or 450 days after shutdown for a prototype core.

    (c) All fuel modules must be shipped with control rods, control rod restraints, and grapple adapters installed. A lower pedestal must be installed in each module holder port.

12. For S9G spent fuel shipments:

    (a) Authorized fuel loadings, internals assembly, and other loading restrictions are specified in Section 1.2 of the Safety Analysis Report for Packaging.

    (b) The minimum fuel cooling time is 100 days.

    (c) All S9G spent fuel modules must have control rods, control rod holddown devices, and grapple adapters installed.

13. The package must contain no more than 6 gallons of residual water, except that shipments of D2W recoverable irradiated fuel may contain up to 11 gallons of residual water.

14. Failed fuel or fuel with defective cladding is not authorized for shipment.

15. Each packaging must meet the Acceptance Tests and Maintenance Program of Chapter 8 of the application, except:

    All containment seals, including the main closure head seal, must be replaced with new seals within the 12-month period prior to each shipment, or earlier if inspection shows any defect.

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

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16. The package must be prepared for transport and operated in accordance with Chapter 7 of the application, except:

    The containment seals, excluding the main closure head seal, must pass a leak test after final closure prior to each shipment. The leak test must have a sensitivity of at least 1×10^{-3} std-cm^3/sec.

17. Prior to first use, and within the 12-month period prior to each shipment, all containment seals, including the main closure head seal, must be leak tested to show a leak rate no greater than 1×10^{-4} std-cm^3/sec. The leak test must have a sensitivity of at least 5×10^{-5} std-cm^3/sec.

18. Revision No. 11 of this certificate may be used until May 31, 2007.

19. Expiration date: October 31, 2011.

REFERENCES

"Core Independent M-140 Safety Analysis Report For Packaging," transmitted February 27, 1991.

Supplements dated: May 28, June 21, and July 17, 1991; February 4 and 7, August 17, and December 2, 1992; October 14, 1994; September 1, and November 16, 1995; May 13, August 7, September 26, and November 26, 1996; February 10, 1997; June 11, 1998; April 11, 2001; March 5 and November 27, 2002; and April 18, 2006.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

James R. Hall for

Robert A. Nelson, Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: May 12, 2006

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
9794	4	71-9794	USA/9794/B(U)-96	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 Report for Packaging for CGN Reactor Compartment Disposal, dated July 12, 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.: CGN Reactor Compartment Disposal Package

(2) Description

The package consists of one deactivated and defueled CGN 36, 37, 38, 39, or 40 (36-40) Reactor Compartment that has been separated from the remainder of the cruiser hull and prepared for shipment by enclosing the entire reactor compartment within a welded steel container. The package is approximately cylindrical, about 40-feet high and about 32-feet in diameter. The entire package is a sixteen-sided polyhedron with an enlarged base containing support fixtures, which extend approximately 10 feet beyond the diameter of the package and provide lift points for the package. The container is constructed of high strength steel (MIL-S-22698). The reactor compartment decks, inner-bottom tank structure, secondary shield, and primary shield tank provide internal support and are fastened to the container by welding. The reactor compartment components are drained of water, except for small inaccessible pockets. The maximum weight of the CGN 36-40 package is 5,000,000 pounds. Potentially radioactive contaminated components and piping from areas outside the reactor compartment may be secured within the package.

(3) Drawings

The packaging is constructed in accordance with the drawings in Chapter 1 of the application.

5.(b) Contents

(1) Type and form of material

Activated structural components associated with the CGN 36-40 reactor, plant piping, ion exchanger resin, purification filter media (which may be solidified), and other components contaminated with radioactive corrosion products (crud). Residual liquid, primarily water, some of which contains low level radioactivity, may be present in quantities up to 850 gallons in the CGN 36-40 package.

(2) Maximum quantity of material per package

The maximum quantity of radioactive material contents (crud and activation) shall not exceed the quantities specified in Section 1.2.3.1 of the application.

6. (a) The shipment of a CGN 36-37 package shall be no earlier than 639 days after shutdown.

(b) The shipment of a CGN 38-40 package shall be no earlier than 365 days after shutdown.

7. The Lowest Service Temperature (LST) must be determined for each package. The package shall not be shipped unless its LST is less than or equal to the daily minimum temperature expected during shipment of the package, as determined on the basis of weather forecasts.

8. (a) For CGN 36-37 packages, the Co-60 curie content of ion exchanger resin shall be less than 6.8 curies. The Co-60 curie content of purification filter media (which has not been solidified) shall be less than 4.1 curies. The combined Co-60 curie content of ion exchanger resin and unsolidified purification filter media shall be less than 10.6 curies.

(b) For CGN 38-40 packages, the Co-60 curie content of ion exchanger resin shall be less than 6.8 curies. The Co-60 curie content of purification filter media (which has not been solidified) shall be less than 5.3 curies. The combined Co-60 curie content of ion exchanger resin and unsolidified purification filter media shall be less than 9.58 curies.

9. (a) CGN 36-37 reactor vessels shall have been operated for less than 18,683 effective full power hours.

(b) CGN 38-40 reactor vessel shall have been operated for less than 14,300 effective full power hours.

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 Chapter 7 of the application.

(b) The package must be acceptance tested in accordance with Chapter 8 of the application.

11. Revision No. 3 of this certificate may be used until February 28, 2007.

12. Expiration date: February 28, 2011

NRC FORM 618 (8-2000) 10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
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REFERENCES

"Safety Analysis Report for Packaging for CGN Reactor Compartment Disposal," dated July 12, 1994.

Supplements Dated: November 10, 1994; July 14, 1995; November 22, 1996; June 16 and July 17, 1998; December 22, 1999; and August 30, 2005.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Robert A. Nelson, Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety and Safeguards

Date: February 8, 2006

NRC FORM 618 (8-2000) 10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
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9795	2	71-9795	USA/9795/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
   Division of Naval Reactors
   Washington, D.C. 20585

   b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

   Irradiated Component Disposal Container
   Safety Analysis Report for Packaging
   dated July 10, 1997, 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.: Irradiated Component Disposal Container (ICDC)

(2) Description

The Model No. ICDC is stainless steel cask with an impact limiter at the upper end. The cask body is cylindrical in shape with overall dimensions of approximately 134.6 inches long by 122 inches diameter at the container body flange. The cask cavity is approximately 134.6 inches long by 91 inches diameter. The wall of the cask is 304 stainless steel, 10 inches thick at the bottom and 5 inches thick at the top. The bottom of the cask is an 11 inch thick circular steel plate. The cask lid is closed by a full penetration weld. The upper impact limiter is a stainless steel ring attached with 21 studs to the cask body. A centering plate and pedestals, welded to the bottom end plate, are used to position the contents within the package. The maximum weight of the package is 200,000 pounds. The maximum weight of the contents is approximately 36,300 pounds.

(3) Drawings

The package is constructed in accordance with the drawings, figures and sketches included in the application documents (see References, below).

5. (b) Contents

The contents of the package are cell support housings and other miscellaneous core components from a spent reactor core. The maximum number of these components per package is specified in Section 1.1 of the application. The other contents of the package include potential residual water not greater than 6 gallons, diatomaceous earth desiccant to absorb the residual water and a stainless steel pumpdown lance which may be left in the package. The maximum radioactivity of the contents is 5,600 curies. The total radioactivity is based on transport no earlier than 50 days after core shutdown.

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.

(c) The package may contain no more than 6 gallons of residual water.

(d) The ICDC shall be shipped no earlier that 50 days after core shutdown.

(e) The total number of cluster joint stud remnants loaded into each ICDC must not exceed 25.

(f) The gross weight of the package shall not exceed 200,000 pounds.

7. Expiration date: April 30, 2008.

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U.S. NUCLEAR REGULATORY COMMISSION

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9795	2	71-9795	USA/9795/B(U)-85	3	OF	3

REFERENCES

Irradiated Component Disposal Container Safety Analysis Report for Packaging dated July 10, 1997.

Supplements: U.S. Department of Energy, Division of Naval Reactors Memorandum G#C98-11009 dated December 2, 1998.
U.S. Department of Energy, Division of Naval Reactors Memorandum G#99-03507 dated May 3, 1999.
U.S. Department of Energy, Division of Naval Reactors Memorandum G#C02-4083 dated October 23, 2002.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

UNITED STATES NUCLEAR REGULATORY COMMISSION

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

Date: April 30, 2003.

Type of Packaging: BYPROD. NORM. FORM

Model	Package ID #	Expiration Date
BCL-3	USA/9067/B( )F	09/30/2007
CI-20WC-2A	USA/9098/B( )	10/01/2008
PAS-1	USA/9184/B(U)	07/31/2009

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Type of Packaging: BYPROD. SPEC. FORM

Model	Package ID #	Expiration Date
A-0109	USA/6280/B( )	02/28/2005
BUSS R-1	USA/9511/B(U)	07/31/2007
C-1	USA/9036/B(U)-96	10/31/2011
EAGLE	USA/9287/B(U)-85	12/31/2009
F-294	USA/9258/B(U)-96	12/31/2008
F-423	USA/9299/B(U)-85	08/31/2006
F-430/GC-40	USA/9290/B(U)-85	02/28/2007
F-431	USA/9310/B(U)-96	06/30/2009
IR-100	USA/9157/B(U)-85	09/30/2009
LCG-25A	USA/4888/B( )	01/31/2007
LCG-25B	USA/4888/B( )	01/31/2007
LCG-25C	USA/4888/B( )	01/31/2007
MW-3000	USA/9030/B( )	10/01/2008
NPI-20WC-6	USA/9102/B( )	10/01/2008
NPI-20WC-6 MKII	USA/9215/B(U)	05/31/2008
OP-100	USA/9185/B(U)-85	12/31/2008
OP-660	USA/9283/B(U)-96	06/30/2008
OPL-660, OP-660	USA/9283/B(U)-96	06/30/2008
ORNL TRU CALIF	USA/5740/B( )	10/01/2008
RG-1	USA/6703/B( )	09/30/2008
SENTINEL-100F	USA/5862/B( )	10/01/2008
SENTINEL-25A	USA/4888/B( )	01/31/2007
SENTINEL-25B	USA/4888/B( )	01/31/2007
SENTINEL-25C	USA/4888/B( )	01/31/2007

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List of Packages by Package Type
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Type of Packaging: BYPROD. SPEC. FORM

Model	Package ID #	Expiration Date
SENTINEL-25C3	USA/4888/B( )	01/31/2007
SENTINEL-25D	USA/4888/B( )	01/31/2007
SENTINEL-25E	USA/4888/B( )	01/31/2007
SENTINEL-25F	USA/4888/B( )	01/31/2007
SENTINEL-8	USA/9030/B( )	10/01/2008
SNAP-21	USA/5830/B( )	10/01/2008
SPEC 2-T	USA/9056/B(U)	04/30/2010
SPEC-150	USA/9263/B(U)-96	06/30/2010
SPEC-300	USA/9282/B(U)-96	04/30/2010
URIPS-8A & -8B	USA/6786/B( )	10/01/2008
URIPS-8B	USA/6786/B( )	10/01/2008
1500	USA/5939/B()F	10/01/2008
181375	USA/5796/B(U)	08/31/2007
181375 & 181361	USA/5796/B(U)	08/31/2007
4.5-TON CF	USA/6642/B( )	02/28/2007
5979	USA/5979/B( )	10/01/2008
5984	USA/5984/B( )	08/31/2007
650L	USA/9269/B(U)-96	11/30/2010
680-OP	USA/9035/B(U)-96	06/30/2010
702	USA/6613/B(U)-96	06/30/2008
741-OP	USA/9027/B(U)-96	08/31/2011
770	USA/9148/B(U)-85	03/31/2008
865	USA/9187/B(U)-96	12/31/2008
880 SERIES PKG	USA/9296/B(U)-85	03/31/2011
976 SERIES	USA/9314/B(U)-96	06/30/2010

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List of Packages by Package Type
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Type of Packaging: FISSILE URANIUM

Model	Package ID #	Expiration Date
ABB-2901	USA/9274/AF	09/30/2007
ANF-250	USA/9217/AF	06/30/2010
ATR	USA/9099/B(U)F-85	01/31/2009
BW-2901	USA/9251/AF	10/31/2007
CE-B1	USA/9272/AF-85	01/31/2007
CHT-OP-TU	USA/9288/B(U)F-96	03/31/2010
DHTF	USA/9203/AF	02/28/2011
D2G POWER UNIT	USA/6441/B( )F	08/31/2007
ES-3100	USA/9315/B(U)F-96	04/30/2011
ESP-30X	USA/9284/B(U)F-85	05/31/2010
FSV-3	USA/6347/AF	09/30/2007
INNER HFIR UN	USA/5797/B(U)F	09/30/2007
LIQUI-RAD	USA/9291/B(U)F-85	10/31/2011
MCC-3 -4 & -5	USA/9239/AF	03/31/2007
MCC-4	USA/9239/AF	03/31/2007
MCC-5	USA/9239/AF	03/31/2007
MFFP	USA/9295/B(U)F-96	06/24/2005
MODEL B	USA/6206/AF	10/01/2008
MODEL 1 S-6213	USA/9186/B(U)F	05/31/2007
MODEL 2 S-6213	USA/9186/B(U)F	05/31/2007
NCI-21PF-1	USA/9234/B(U)F	12/31/2008
NFS-URANYL NIT.	USA/5059/AF	05/31/2007
NNFD-10	USA/6357/AF	10/01/2008
NONE SPECIFIED	USA/6406/AF	03/31/2008

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List of Packages by Package Type
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Type of Packaging: FISSILE URANIUM

Model	Package ID #	Expiration Date
NPC	USA/9294/AF-85	11/30/2010
OUTER HFIR UN	USA/5797/B(U)F	09/30/2007
PADUCAH TIGER	USA/6553/AF	10/01/2008
PATRIOT	USA/9292/AF-85	08/31/2010
RA-3	USA/4986/AF	03/31/2008
RAJ-II	USA/9309/B(U)F-96	11/30/2009
SP-1 SP-2 SP-3	USA/9248/AF	02/28/2009
SP-2	USA/9248/AF	02/28/2009
SP-3	USA/9248/AF	02/28/2009
SRP-1	USA/9285/AF-85	10/31/2008
ST	USA/9246/AF	11/30/2011
S5W POWER UNIT	USA/5580/B( )F	12/31/2007
TNF-XI	USA/9301/AF-85	08/30/2008
TRAVELLER STD	USA/9297/AF-96	03/15/2010
TRAVELLER XL	USA/9297/AF-96	03/15/2010
TRIGA-I	USA/9034/AF	12/31/2010
TRIGA-II	USA/9037/AF	12/31/2010
UBE-1	USA/9280/AF-85	02/28/2008
UBE-2	USA/9281/AF-85	05/31/2008
UNC-2600	USA/5086/B(U)F	02/28/2009
UX-30	USA/9196/AF-96	02/28/2011
WE-1	USA/9289/B(U)F-85	02/28/2009
235R001	USA/6386/B(U)F	04/30/2010
5X22	USA/9250/B(U)F-85	03/31/2008
51032-1	USA/6581/AF	10/01/2008

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List of Packages by Package Type
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Type of Packaging: FISSILE URANIUM

Model	Package ID #	Expiration Date
51032-2	USA/9252/AF	10/31/2008
814A	USA/5149/B( )F	10/01/2008
927A1 & 927C1	USA/6078/AF	10/01/2008
927C1	USA/6078/AF	10/01/2008

Type of Packaging: IRRADIATED FUEL

Model	Package ID #	Expiration Date
BMI-1	USA/5957/B( )F	10/01/2008
CNS 1-13G	USA/9216/B( )F	01/31/2008
FSV-1 UNIT 3	USA/9277/B( )F	10/01/2008
GA-4	USA/9226/B(U)F-85	10/31/2008
GE-100	USA/5926/B( )F	05/31/2008
HI-STAR 100	USA/9261/B(U)F-85	03/31/2009
IF-300	USA/9001/B( )F	10/01/2008
M-130	USA/6003/B( )F	09/30/2007
M-140	USA/9793/B(U)F-85	10/31/2011
NAC-LWT	USA/9225/B(U)F-96	02/28/2010
NAC-STC	USA/9235/B(U)F-96	03/31/2009
NAC-1	USA/9183/B( )F	09/30/2004
NLI-1/2	USA/9010/B( )F	10/01/2008
NLI-10/24	USA/9023/B( )F	07/31/2008
NUHOMS MP187	USA/9255/B(U)F-85	10/31/2008
NUHOMS-MP197	USA/9302/B(U)F-85	07/31/2007
T-2	USA/5607/B( )F	10/01/2008
T-3	USA/9132/B(M)F	04/30/2011
TN-FSV	USA/9253/B(U)F-85	05/31/2009
TN-68	USA/9293/B(U)F-85	02/28/2011
TN-8	USA/9015/B( )F	10/01/2008
TN-8L	USA/9015/B( )F	10/01/2008
TN-9	USA/9016/B( )F	10/01/2008
TS125	USA/9276/B(U)F-85	09/30/2007

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List of Packages by Package Type
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Type of Packaging: IRRADIATED FUEL

Model	Package ID #	Expiration Date
UMS UNIVERSAL	USA/9270/B(U)F-96	10/31/2007
125-B	USA/9200/B(M)F	06/30/2011
2000	USA/9228/B(U)F-96	05/31/2011

Type of Packaging: PU AIR

Model	Package ID #	Expiration Date
PAT-1	USA/0361/B(U)F-96	03/31/2009
PAT-2	USA/9150/B(U)-85	09/30/2011

Type of Packaging: PU NORM. FORM

Model	Package ID #	Expiration Date
HALFPACT	USA/9279/B(U)F-85	10/31/2010
NRBK-41	USA/9221/B( )F	10/01/2008
RH-TRU 72-B	USA/9212/B(M)F-85	02/28/2010
TRUPACT-II	USA/9218/B(U)F-85	08/31/2009
6400	USA/6400/B( )F	11/30/2007

Type of Packaging: PU SPEC. FORM

Model	Package ID #	Expiration Date
NEUTRON SOURCE	USA/5757/B( )F	06/30/2008

Type of Packaging: WASTE, B

Model	Package ID #	Expiration Date
CGN RCDP	USA/9794/B(U)-96	02/28/2011
CNS 1-13C	USA/9081/B( )	01/31/2008
CNS 1-13C II	USA/9152/B( )F	10/01/2008
CNS 10-160B	USA/9204/B(U)-85	10/31/2010
CNS 3-55	USA/5805/B( )	10/01/2008
CNS 8-120B	USA/9168/B(U)	06/30/2010
D1G CB-TS	USA/9792/B(U)	09/30/2007
ICDC	USA/9795/B(U)-85	04/30/2008
N-55	USA/9070/B(U)	01/31/2010
PWR-2 CORE BAR.	USA/9791/B(U)-85	07/31/2007
SSN 688	USA/9788/B(U)-85	09/30/2008
S3G CBDCA	USA/9786/B(U)	08/31/2011
S5W REC. COMPT.	USA/9788/B(U)-85	09/30/2008
TN-RAM	USA/9233/B(U)	04/30/2010
10-142	USA/9208/B( )	08/31/2007
3-82B	USA/6574/B( )	10/01/2008

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10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

The purpose of this directory is to make available a convenient source of information on packaging approved by the U.S. Nuclear Regulatory Commission. To assist in identifying packaging, an index by Model Number and corresponding Certificate of Compliance Number is included at the front of Volumes 1 and 2. An alphabetical listing by user name is included in the back of Volume 3 of approved Quality Assurance programs. The reports include a listing of all users of each package design and approved Quality Assurance programs prior to the publication date of the directory.

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