NRC FORM 618
(3-96)
10 CFR 71

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIALS PACKAGES

U.S. NUCLEAR REGULATORY COMMISSION

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. PACKAGE IDENTIFICATION NUMBER	d. PAGE NUMBER	e. TOTAL NUMBER PAGES
9228	18	USA/9228/B(U)F-85	1	8

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

General Electric Company
Vallecitos Nuclear Center
P.O. Box 460, Vallecitos Road
Pleasanton, CA 94566

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION:

General Electric Company application dated May 19, 1988, as supplemented.

c. DOCKET NUMBER 71-9228

4. CONDITIONS

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

5.

(a) Packaging

(1) Model No.: 2000

(2) Description

A steel encased lead shielded shipping cask. The cask is within a double-walled overpack with toroidal shell impact limiters at each end. The overall dimensions are approximately 131.5 inches in height and 72.0 inches in diameter. The cask is transported in the upright or horizontal position. The gross weight of the package is approximately 33,550 lbs.

The cask is constructed of two concentric 1-inch thick 304 stainless steel cylindrical shells (ASTM A 240) joined at the bottom end to a 6-inch thick 304 stainless steel forging (ASTM A 182). The annulus between the two shells is filled with lead approximately 4 inches thick. The cask is approximately 71.0 inches in height and has an outer diameter of 38.5 inches. The cask cavity is approximately 26.5 inches in diameter and 54.0 inches deep.

The cask lid is 304 stainless steel and lead, has a stepped design, and is fully recessed into the cask top flange. The lid is secured to the cask body by 15, 1.25-inch diameter socket head screws. The cask is sealed by elastomeric O-rings bonded to a thin aluminum disc-shaped ring. The cask is equipped with a seal test port on the side of the cask body, a vent port in the cask lid, and a drain port near the bottom of the cask.

The cask is positioned within an overpack constructed from two 0.5-inch thick concentric 304 stainless steel cylindrical shells (ASTM A 240). The shells are separated radially by eight equally spaced tubes and horizontally by two tube sections. A 304 stainless steel toroidal shell impact limiter is attached to each end of the overpack. The overpack opens just above the lower impact limiter for access to the cask. The top of the overpack is joined to the base by 15, 1-3/8-inch diameter shoulder screws.

5(a)(2) Description (Continued)

Gussets on the top and bottom impact limiters provide tie-down points for the package. The cask body is equipped with attachment plates for lifting devices. The cask lifting devices are detached during transport.

(3) Drawings

(i) The packaging is constructed and assembled in accordance with General Electric Company Drawing Nos. 129D4946, Rev. 10; 105E9520, Rev. 4; and 105E9521, Rev. 5.

(ii) Packaging Serial No. 2001 is constructed and assembled in accordance with General Electric Company Drawing Nos. 129D4946, Rev. 10; 101E8718, Rev. 12; and 101E8719, Rev. 12.

(iii) The HFIR fuel basket and liner are constructed and assembled in accordance with General Electric Company Drawing No. 105E9523, Rev. 3.

(iv) The multifunctional rack is constructed and assembled in accordance with General Electric Company Drawing No. 105E9555, Rev. 2.

(v) The barrel rack is constructed and assembled in accordance with General Electric Company Drawing No. 166D8066, Rev. 1.

(vi) The material basket is constructed in accordance with General Electric Company Drawing No. 183C8356, Rev. 1. The material basket may be used with the multifunctional rack and the barrel rack.

(vii) The TSR fuel basket is constructed and assembled in accordance with General Electric Company Drawing No. 105E9560, Rev. 2.

(viii) The MTR fuel basket is constructed and assembled in accordance with General Electric Company Drawing No. 105E9557, Rev. 9.

(b) Contents

(1) Type and form of material

(i) Irradiated fuel rods, which may be cut or segmented.

(ii) Byproduct, source, or special nuclear material in solid form.

(iii) Irradiated High Flux Isotope Reactor (HFIR) fuel assembly, positioned within the HFIR fuel basket and liner as specified in 5(a)(3). The HFIR fuel assembly is fabricated in accordance with Oak Ridge National Laboratory Drawing Nos. M-11524-OH-101-D, Rev. 0, and M-11524-OH-102-D, Rev. 0.

(iv) Irradiated Tower Shielding Reactor (TSR) fuel elements, positioned within the TSR fuel basket specified in 5(a)(3).

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.

(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	1/2	1-1/2	1-1/2	1/2
Graphite reflectors	With or without reflectors	With or without reflectors	With reflectors	With reflectors	Without reflectors
Uranium concentration in fuel (w/o U)	8 - 45	10 - 45	8.5 min.	8.5 min.	10 min.
Max. rod length (in)	30	30	30	30	30
Max. active fuel length (in)	15	22	15	15	22
Min. clad thickness (in)	0.02	0.016	0.02	0.02	0.016
Max. uranium enrichment (w/o U-235)	20.0	20.0	70.0	94.0	94.0
Max. active fuel diameter (in)	1.435	0.51	1.435	1.435	0.51
Max. U-235 per rod (g)	165	44 (max. 15 rods per basket cell) 33 (max. 20 rods per basket cell)	140	220	44 (max. 15 rods per basket cell) 33 (max. 20 rods per basket cell)
Max. U-235 mass per fuel basket cell (g)	560	660	560	660	660
Max. burnup (GWd/MTU)	427	427	427	568	568
Min. cool time (days)	120	120	120	120	120

Note: The enrichments, masses, and dimensions shall be based on values prior to irradiation.

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

Not to exceed 5,450 lbs, including fuel baskets, carrier racks, shoring, secondary containers, and shielding liner.

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

600 watts decay heat; and

Fissile contents not to exceed 1175 grams U-235 equivalent mass with initial enrichment not to exceed 5 weight percent in the fissile isotope; minimum pellet diameter of 0.3 inch, maximum burnup of 45 GWd/MTU, and minimum cooling time of 120 days; or

Fissile contents not to exceed 1750 grams U-235 equivalent mass with initial enrichment not to exceed 5 weight percent in the fissile isotope; minimum pellet diameter of 0.35 inch, maximum burnup of 38 GWd/MTU, and minimum cooling time of 120 days. Fuel rods must be contained in closed, 5-inch schedule 40 pipe, with a maximum of 437.5 grams U-235 equivalent per pipe; or

Fissile contents not to exceed 242 grams U-235 equivalent mass with initial enrichment not to exceed 5 weight percent in the fissile isotope; minimum pellet diameter of 0.3 inch, maximum burnup of 52 GWd/MTU, and minimum cooling time of 180 days.

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

2000 watts decay heat. Fissile contents not to exceed 500 grams U-235 equivalent mass. Carrier racks specified in 5(a)(3)(iv) or 5(a)(3)(v) must be used for contents exceeding 600 watts decay heat per package.

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

One HFIR fuel assembly. The fuel assembly is composed of one inner fuel element, with up to 2628 grams U-235, and one outer fuel element, with up to 6872 grams U-235. The maximum uranium enrichment is 93.2 weight percent U-235. The maximum burnup per assembly is 2300 MWd, the minimum cool time is two years. Decay heat not to exceed 600 watts per package.

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

A maximum of 4393 grams U-235 per package. The maximum uranium enrichment is 94.0 weight percent U-235. Decay heat not to exceed 35 watts per package. The TSR fuel elements must be positioned and limited within the TSR fuel basket as follows:

Lower fuel basket section -	Up to 4 upper or lower fuel elements, or a combination of upper and lower fuel elements, for a total U-235 mass of 1412 grams.

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

Middle fuel basket section - Up to 4 fuel cover (lune) plates, for a total U-235 mass of 304 grams.

Upper fuel basket section - Up to 6 annular fuel elements plus one cylindrical fuel element, for a total U-235 mass of 2677 grams.

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

Weight of contents, including fuel elements, spacers, shoring, and hardware, not to exceed 42.8 lbs per fuel basket cell.

Decay heat not to exceed any of the following: 1500 watts per package, 120 watts per cell, 35 watts per cell in the upper half of the fuel basket, 85 watts per cell in the lower half of the fuel basket, 765 watts in the lower half of the fuel basket (i.e., the lower half of all 21 cells combined).

Failed fuel elements are permitted provided the damage is limited to cladding defects due to corrosion, nicks, and scratches. Failed fuel elements must be structurally and geometrically intact.

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

Weight of contents, including fuel elements, spacers, shoring, and hardware, not to exceed 42.8 lbs per fuel basket cell.

For stainless steel and inconel clad fuel, decay heat not to exceed any of the following: 1500 watts per package, 120 watts per cell, 35 watts per cell in the upper half of the fuel basket, 85 watts per cell in the lower half of the fuel basket, 765 watts in the lower half of the fuel basket (i.e., the lower half of all 21 cells combined).

For aluminum clad fuel, decay heat not to exceed either of the following: 630 watts per package, 30 watts per cell.

(c) Transport Index for Criticality Control

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

For the contents described in 5(b)(1)(i), 5(b)(1)(ii), and 5(b)(1)(iii); and limited in 5(b)(2)(i), 5(b)(2)(ii), and 5(b)(2)(iii):	100
For the contents described in 5(b)(1)(iv), 5(b)(1)(v), and 5(b)(1)(vi); and limited in 5(b)(2)(iv), 5(b)(2)(v), and 5(b)(2)(vi):	0.0

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

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

8. Bolt torque:

   The cask lid bolts must be torqued to 690 ft-lbs (lubricated).

   The bolts used to secure the top of the overpack to the overpack base must be torqued to 100 ft-lbs (dry).

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

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

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

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

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

10. Prior to each shipment (except for contents meeting the requirements of special form radioactive material), the package must be leak tested to 1×10^{-3} std cm^3/sec. Prior to first use, after the third use, and at least once within the 12-month period prior to each subsequent use, the package must be leak tested to 1×10^{-7} std cm^3/sec.

11. The cask must be vacuum dried prior to shipment if contents are loaded under water, or if water is introduced into the cask cavity. During shipments for which vacuum drying is performed, the cask cavity must be filled with helium.

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

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

   (b) Each package must meet the Acceptance Tests and Maintenance Program of Chapter 8 of the application, except that inspections in Section 8.2 of the application must be performed at least once within the 12-month period prior to each use; and

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

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

13. Appropriate carrier racks or shoring must be provided to minimize movement of contents during accident conditions of transport. A lead liner, as shown in General Electric Company Drawing No. 129D4922, Rev. 2, which was included in the March 29, 1989, supplement, may be used inside the cask.

14. Each batch of ethylene propylene seals must be tested in accordance with Section 8.1.4.2 of the application.

15. Fissile mass limits for reactor fuel are based on fissile mass prior to irradiation.

16. For the contents described in 5(b)(1)(v) and 5(b)(1)(vi), the package may be transported horizontally. For horizontal transport, the package must be secured to the truck bed with the top end of the package (closure end) facing the front (cab) of the truck.

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

18. Expiration date: July 31, 2000.

REFERENCES

General Electric Company application dated May 19, 1988.

Supplements dated: March 29 and August 24, 1989; May 30, October 11, and December 12, 1990; May 22, 1991; June 8, July 27, August 4, and December 9, 1993; April 29 and July 28, 1994; May 19 and 30, 1995; October 31, 1996; May 9, June 3, July 14, August 14, and October 7, 1997; May 15, June 2 (3), June 24, October 6, November 10, 1998; and February 3, April 29, May 26 and 27, June 9, 12, 18, 22, 25, July 27 and 29, October 18, and November 1, 1999.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

E William Brach

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

Date: November 3, 1999

NRC FORM 618
(3-96)
10 CFR 71

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIALS PACKAGES

U.S. NUCLEAR REGULATORY COMMISSION

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. PACKAGE IDENTIFICATION NUMBER	d. PAGE NUMBER	e. TOTAL NUMBER PAGES
9233	5	USA/9233/B(U)	1	3

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

Transnuclear, Inc.
Four Skyline Drive
Hawthorne, NY 10532-2120

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION:

Transnuclear, Inc. application
dated November 22, 1988, as supplemented.

c. DOCKET NUMBER 71-9233

4. CONDITIONS

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

5.

(a) Packaging

(1) Model No.: TN-RAM

(2) Description

The package is a steel encased lead shielded cask with wood impact limiters attached at both ends. The cask is a right circular cylinder. The overall dimensions of the packaging are approximately 178 inches long and 92 inches diameter with the impact limiters installed. The cask body is approximately 129 inches long with an outer diameter of 51 inches. The cask cavity has a length of approximately 111 inches and an inside diameter of 35 inches. The cask body is made of a 0.75-inch stainless steel inner shell, a 5.88-inch thick lead annulus, a 1.5-inch thick stainless steel outer shell, a 0.5-inch thick inner bottom plate and a 2.5-inch thick outside bottom plate. The lead shielding is 6 inches thick in the bottom end of the cask. The outer shell of the cask body is covered with a stainless steel thermal shield. The closure lid consists of a 2.5-inch thick outer stainless steel plate and a 0.5-inch thick inner stainless steel plate separated by 6 inches of lead shielding. The lid is secured by sixteen 1.5-inch diameter closure bolts. Two concentric silicone O-rings are installed in grooves on the underside of the lid. The cask is equipped with a sealed leak test port between the O-rings, a vent port in the closure lid and a sealed drain port in the bottom of the cask.

Each impact limiter is attached to the cask by eight 1.75-inch diameter bolts. The cask is equipped with 6 trunnions, four at the top and two at the bottom.

The gross weight of the package is approximately 80,000 pounds, including maximum contents of 9,500 pounds.

5.(a) Packaging (continued)

(3) Drawings

The packaging is constructed in accordance with Transnuclear, Inc. Drawing Nos. 990-701, Rev. 6; 990-702, Rev. 6; 990-703, Rev. 6; 990-704, Rev. 3; 990-705, Rev. 4; 990-706, Rev. 3; 990-707, Rev. 3; 990-708, Rev. 5; and 990-709, Rev. 1.

(b) Contents

(1) Type and Form of Material

Dry irradiated and contaminated non-fuel-bearing solid materials contained within a secondary container.

(2) Maximum quantity of material per package

Greater than Type A quantities of radioactive material which may include fissile material provided that the fissile material does not exceed the generally licensed mass limits specified in 10 CFR 71.18, 71.20 and 71.22. The contents may not exceed 2,000 times an A_2 quantity. The decay heat of the contents may not exceed 300 watts. The maximum gross weight of the contents, secondary container and shoring is limited to 9,500 pounds.

6. As appropriate, shoring must be used in the secondary container sufficient to prevent significant movement of the contents under accident conditions.

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

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

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

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

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

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

10. Expiration date: January 31, 2005

Page 3 - Certificate No. 9233 - Revision No. 5 - Docket No. 71-9233

REFERENCES

Transnuclear, Inc. application dated November 22, 1988.

Supplements dated: January 13, May 18, June 5, July 21, July 28, and August 11, 1989; January 4, 1990; December 18, 1997; August 20, 1998; and December 7, 1999.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

E. William Brach

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

Date: February 4, 2000

NRC FORM 618
(3-96)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIALS PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. PACKAGE IDENTIFICATION NUMBER	d. PAGE NUMBER	e. TOTAL NUMBER PAGES
9234	13	USA/9234/B(U)F	1	3

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

Eco-Pak Specialty Packaging
125 Iodent Way, Suite B
Elizabethton, TN 37643

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION:

Nuclear Containers, Inc. application dated January 11, 1993, as supplemented.

c. DOCKET NUMBER 71-9234

4. CONDITIONS

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

5.

(a) Packaging

(1) Model No.: NCI-21PF-1

(2) Description

Overpack for 30-inch enriched uranium hexafluoride (UF_6) cylinders. The valve end of the cylinder may be equipped with a valve protection device. The overpack is a right circular cylinder constructed of two stainless steel shells with the volume between the shells filled with fire resistant, phenolic-foam per USAEC Specification SP-9, Rev. 1, and Supplement K/TL-729. The volume between the 1/4-inch thick end closure plates of the two shells is filled with oak wood blocks which are cross-laminations of 3 layers of boards glued and nailed together. A stepped and gasketed horizontal joint permits the top half of the overpack to be removed from the base. The package "halves" are secured with ten, 1-inch stainless steel toggle closures. The overpack is 43-5/8 inches O.D. by 92 inches long. The maximum gross weight of the package, including the valve protection device, is 8875 pounds.

(3) Drawing

The Model No. NCI-21PF-1 packaging is fabricated in accordance with Nuclear Containers, Inc. Drawing No. DED-206-B, Sheets 1 through 11, Rev. 5. The valve protection device and the valve protection device gauge are fabricated and assembled in accordance with United States Enrichment Corporation Drawing Nos. VPD-0001, Rev. 1, VPD-0002, Rev. 2, and VPD-0003, Rev. 1.

5.(b) Contents

(1) Type and form of material

Uranium hexafluoride contained within a Model 30B cylinder.

(2) Maximum quantity of material per package

5,020 pounds uranium hexafluoride. Uranium enriched to not more than 5 w/o in the U-235 isotope. The total quantity of radioactive material within a package may not exceed a Type A quantity.

(c) Transport Index for Criticality Control

Minimum transport index to be shown on label for nuclear criticality control: 5.0

6. The Model 30B cylinders must be fabricated, inspected, tested, and maintained in accordance with American National Standard N14.1 (1990 Edition). Cylinders must be fabricated in accordance with Section VIII, Division I, of the ASME (American Society of Mechanical Engineers) Boiler and Pressure Vessel Code and be ASME code stamped.

7. At least once every five years, each packaging must be inspected to verify the presence and condition of the insulation. The inspection shall consist of inserting a probe through each vent hole in both the lid and base to confirm the presence and rigidity of the insulation. For packagings which require drying, the inspection must be performed after drying.

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

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

(b) Each packaging must meet the Acceptance Tests and Maintenance Program of Chapter 8 of the application, as supplemented July 9 and August 11, 1997.

(c) The package shall be prepared for shipment and operated in accordance with the Operating Procedures of Chapter 7 of the application, as supplemented July 9 and August 11, 1997.

(d) 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.

9. 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.

10. 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 supplements dated July 9 and August 11, 1997.

11. 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.

12. 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.

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

14. Expiration date: December 31, 2003.

REFERENCES

Nuclear Containers, Inc. application dated January 11, 1993.

Supplements dated: September 10, 1993; July 21, 1994; November 19, 1996; and February 26, April 21, May 15, July 9, August 11, 1997, and September 9, 1998.

United States Enrichment Corporation supplement dated: April 14, 1997.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Cass R. Chappell

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

Date: 11/25/98

NRC FORM 618
(3-96)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIALS PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. PACKAGE IDENTIFICATION NUMBER	d. PAGE NUMBER	e. TOTAL NUMBER PAGES
9235	2	USA/9235/B(U)F-85	1	11

2. PREAMBLE

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

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

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

a. ISSUED TO (Name and Address)

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

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION:

NAC International, Inc. application dated December 30, 1996, as supplemented

c. DOCKET NUMBER 71-9235

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 irradiated PWR fuel assemblies in a canister, and (c) non-fissile, solid radioactive (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 250,000 lbs.

The cask body is made of two concentric stainless steel shells. The inner shell is 1.5 inches thick and has an inside diameter of 71 inches. The outer shell is 2.65 inches thick and has an outside diameter of 86.7 inches. The annulus between the inner and outer shells is filled with lead.

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

The inner and outer shells are welded to steel forgings at the top and bottom ends of the cask. The bottom end of the cask consists of two stainless steel circular plates which are welded to the bottom end forging. The inner bottom plate is 6.2 inches thick and the outer bottom plate is 5.45 inches thick. The space between the two bottom plates is filled with a 2-inch thick disk of a synthetic polymer (NS4FR) neutron shielding material.

The cask is closed by two steel lids which are bolted to the upper end forging. The inner lid (containment boundary) is 9 inches thick and is made of Type 304 stainless steel. The outer lid is 5.25 inches thick and is made of SA-705 Type 630 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 metallic O-rings. The outer lid is equipped with a single metallic O-ring. The inner lid is fitted with a vent and drain port which are sealed by metallic O-rings and cover plates.

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. The package is equipped at each end with an impact limiter made of redwood and balsa.

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 contents are transported either directly loaded (uncanistered) into a stainless steel fuel basket or within a stainless steel transportable storage canister (TSC). The TSC, including its welded shield and structural lids, represents the separate inner container for the purposes of meeting 10 CFR 71.63.

The directly loaded fuel basket within the cask cavity can accommodate up to 26 PWR fuel assemblies. The fuel assemblies are positioned within square sleeves made of stainless steel. Boral sheets are encased outside the walls of the sleeves. The sleeves are laterally supported by 31, 1/2-inch thick, 71-inch diameter stainless steel disks. The basket also has 20 heat transfer disks made of Type 6061-T6 aluminum alloy. The support disks and heat transfer disks are connected by six, 1-5/8-inch diameter by 161-inch long threaded rods made of Type 17-4 PH stainless steel.

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

The TSC shell, bottom plate, and welded shield and structural lids are fabricated from stainless steel. The bottom is a 1-inch thick steel plate, and the shell is constructed of 5/8-inch thick rolled steel plate. The shell is 70 inches in diameter with a height of 122 inches. 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), with a maximum weight limit of 30,600 lbs. Alternatively, a stainless steel GTCC waste basket is used for up to 24 containers of waste.

The TSC fuel basket positions 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 and 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-T6 aluminum alloy.

An RFA can accommodate up to 64 Yankee Class fuel rods, as intact or damaged fuel or fuel debris, in an 8x8 array of stainless steel tubes. An RFA is a stainless steel square shell, with a stainless steel basket assembly that supports 64 fuel tubes. Intact and damaged Yankee Class 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 fuel assembly.

The TSC GTCC basket positions up to 24 Yankee Class waste containers within square stainless steel sleeves. The basket is supported laterally by eight 1-inch thick, 69-inch diameter stainless steel disks. The 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 GTCC waste containers accommodate radiation activated and surface contaminated steel, plasma cutting debris (dross) or filter media, and have the same external dimensions of Yankee Class fuel assemblies.

The TSC, in either the GTCC waste or Yankee Class fuel configuration, is axially positioned in the cask cavity by two aluminum honeycomb spacers. The spacers, which are enclosed in a Type 6061-T6 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.

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5.(a)(3) Drawings

(i) The cask is constructed and assembled in accordance with the following Nuclear Assurance Corporation (now NAC International) Drawing Nos.:

423-800, sheets 1-2,	Rev. 6	423-807, sheets 1-2,	Rev. 1
423-802, sheets 1-6,	Rev. 8	423-811, sheets 1-2,	Rev. 5
423-803,	Rev. 1	423-812,	Rev. 0
423-804, sheets 1-3,	Rev. 2	423-809, sheets 1-2,	Rev. 1
423-805,	Rev. 1	423-810, sheets 1-2,	Rev. 1
423-806,	Rev. 1	423-900,	Rev. 3

(ii) For the directly loaded configuration, the basket is constructed and assembled in accordance with the following Nuclear Assurance Corporation (now NAC International) Drawing Nos.:

423-870,	Rev. 2	423-873,	Rev. 1
423-871,	Rev. 1	423-874,	Rev. 2
423-872,	Rev. 3	423-875,	Rev. 2

(iii) For the 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. 1	455-887, sheets 1-2,	Rev. 1
455-801, sheets 1-2,	Rev. 1	455-888,	Rev. 4
455-820,	Rev. 1	455-891,	Rev. 0
455-870,	Rev. 3	455-892,	Rev. 1
455-871, sheets 1-2,	Rev. 3	455-893,	Rev. 3
455-872,	Rev. 5	455-894,	Rev. 1
455-873,	Rev. 2	455-895,	Rev. 2
455-881,	Rev. 2		

(iv) For the TSC configuration, RFAs are constructed and assembled in accordance with the following Yankee Atomic Electric Company Drawing Nos.:

YR-00-060	Rev. 1	YR-00-064	Rev. 1
YR-00-061	Rev. 1	YR-00-065	Rev. 1
YR-00-062	Rev. 1	YR-00-066	Rev. 1
YR-00-063	Rev. 1		

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 40,000 MWD/MTU when cooled for at least 6.5 years, or 45,000 MWD/MTU when cooled for at least 10 years. 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)
Cladding Material	Zirc-4	Zirc-4	Zirc-4	Zirc-4	Zirc-4
Maximum Initial Uranium Content (kg/assembly)	407	469	426	464	426
Maximum Initial Enrichment (wt% ^{235}U)	4.2	4.2	4.2	4.2	4.2
Minimum Initial Enrichment (wt% ^{235}U)	3.7	3.7	3.7	3.7	3.7
Assembly Cross-Section (in)	7.76 to 8.11	8.20 to 8.54	8.10 to 8.14	8.43 to 8.54	8.43
Number of Fuel Rods per Assembly	176 to 179	204 to 216	236	264	264
Fuel Rod OD (in)	0.422 to 0.440	0.418 to 0.430	0.382	0.374 to 0.379	0.360
Minimum Cladding Thickness (in)	0.023	0.024	0.025	0.023	0.023
Pellet Diameter	0.344 to 0.377	0.358 to 0.390	0.325	0.3225 to 0.3232	0.3088
Maximum Active Fuel Length (in)	146	144	150	144	144

5.(b) (1) Contents - Type and Form of Material (Continued)

(ii) Irradiated intact Yankee Class PWR fuel assemblies or RFAs within the TSC. The maximum initial fuel pin pressure is 315 psig. The fuel assemblies consist of uranium oxide pellets with the specifications, based on design nominal or operating history record values, listed below:

Assembly Manufacturer/Type	UN 16x16	CE [1] 16x16	West. 18x18	Exxon [2] 16x16	Yankee RFA
Cladding Material	Zircaloy	Zircaloy	SS	Zircaloy	Zirc/SS
Maximum Number of Rods per Assembly	237	231	305	231	64
Maximum Initial Uranium Content (kg/assembly)	246	240	287	240	70
Maximum Initial Enrichment (wt% ^{235}U)	4.0	3.9	4.94	4.0	4.94
Minimum Initial Enrichment (wt% ^{235}U)	4.0	3.7	4.94	3.5	3.5
Maximum Assembly Weight (lbs)	850	850	900	850	850
Maximum Burnup (Mwd/MTU)	32,000	36,000	32,000	36,000	36,000
Maximum Decay Heat per Assembly (kW)	0.28	0.347	0.28	0.34	0.11
Minimum Cool Time (yrs)	11.0	8.1	19.0	9.0	8.0
Maximum Active Fuel Length (in)	91	91	92	91	92

Notes:

[1] Combustion Engineering (CE) fuel with a maximum burnup of 32,000 Mwd/MTU, a minimum enrichment of 3.5 wt percent ^{235}U, a minimum cool time of 8.0 years, and a maximum decay heat per assembly of 0.304 kW is authorized.

[2] Exxon assemblies with stainless steel in-core hardware shall be cooled a minimum of 16.0 years with a maximum decay heat per assembly of 0.269 kW.

5.(b) (1) Contents - Type and Form of Material (Continued)

(iii) Solid, irradiated, and contaminated hardware and solid, particulate debris (dross) or filter media placed in a GTCC waste container, provided the quantity of fissile material does not exceed a Type A quantity and does not exceed the mass limits of 10 CFR 71.53.

(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): 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.

5.(c) Transport Index for Criticality Control

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

6. The maximum heat load within the packaging at any time (transport, storage, or testing) shall not exceed the decay heat limits in Item 5.(b)(2).

7. 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).

8. For contents placed in a GTCC waste container and described in Item 5.(b)(1)(iii): for any contents containing 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.

9. For damaged fuel rods and fuel debris of the quantity described in Item 5.(b)(2)(iii): 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. 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).

10. 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 contents to be loaded and independent verification that the contents meet 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 the surface removable contamination levels meet the requirements of 49 CFR 173.443 and 10 CFR 71.87.
        (c) Backfill the cask cavity with helium having a minimum purity of 99.9%.
        (d) Leak test all containment boundary penetrations with a helium mass spectrometer leak detector for the appropriate containment boundary condition (A or B). 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).

    (3) During direct loading of fuel 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 15 mbar (absolute) or less for a minimum of 1 hour.
        (b) Verification that the cask pressure rise is less than 12 mbar in 10 minutes.

(4) The separate inner container (TSC) shall be pneumatically pressure tested after loading. The minimum test pressure shall be 15 psig.

(5) During canistered fuel or GTCC waste loading operations, the removal of water and residual moisture from the containment vessel in accordance with the following specifications:
(a) Cask evacuation to a pressure of 3 mm of mercury or less for a minimum of 30 minutes.
(b) Verification that there is no cask pressure rise in 30 minutes.

(6) During loading and closing of the TSC, the removal of water and residual moisture from the canister in accordance with the following specifications at the time of closure:
(a) Canister evacuation to a pressure of 3 mm of mercury or less for a minimum of 30 minutes.
(b) Verification that there is no canister pressure rise in 30 minutes.

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 shall be radiographed and liquid-penetrant examined in accordance with ASME Code Section III, Division 1, Subsection NB.

(2) All separate inner container (TSC) boundary welds, with the exception of field installed shield and structural lid welds at the time of loading, shall be volumetrically and liquid-penetrant examined in accordance with ASME Code Section III, Division 1, Subsection NB.

(3) For separate inner container (TSC) welds that are installed after loading:
(a) The root and final surfaces of the shield lid weld shall be liquid-penetrant examined in accordance with ASME Code Section III, Division 1, Subsection NB.
(b) The root and final surfaces of the vent and drain port cover to shield lid welds shall be liquid-penetrant examined in accordance with ASME Code Section III, Division 1, Subsection NB.
(c) The structural lid weld shall be either ultrasonically examined with the final weld surface liquid penetrant examined, or progressively liquid-penetrant examined. The weld examinations shall be in accordance with ASME Code Section III, Division 1, Subsections NB-5332 (for ultrasonic) or NB-5350 (for liquid-penetrant).
(d) For the structural lid weld, if progressive liquid-penetrant examination alone is used, it must include at a minimum the root, each successive 3/8-inch weld thickness, and the final layer. 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.

(4) The cask primary containment boundary shall be pressure tested to 150% of the design pressure per 10 CFR 71.85(b). The minimum test pressure shall be 76 psig.

(5) Each pair of lifting trunnions shall be load tested, in the cask axial direction, to 300% (750,000 lbs. minimum) of the maximum service load to diametrically opposite trunnion pairs in accordance with the requirements of ANSI N14.6.

(6) A fabrication leakage test shall be performed on all containment components, including metallic closure 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 2.0 X 10^{-7} cm^3/sec (helium) and shall have a test sensitivity of at least 1.0 X 10^{-7} cm^3/sec (helium).

(7) All containment vessel metallic O-rings shall be replaced with new O-rings after each use. The BTFE O-rings of the interlid and pressure ports shall be visually inspected prior to each use and replaced if necessary. The BTFE O-rings shall be replaced within the 2-year period prior to shipment.

(8) Prior to the first use of each package, a thermal performance test shall be conducted to ensure that the cask, in the horizontal position, possesses the heat rejection capabilities predicted by the thermal analysis. The thermal test shall be performed with a minimum cask heat load of 22.1 kW supplied by electric heaters in each fuel tube.

(9) For each package, a periodic thermal performance test shall be performed every 5 years, or prior to the 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 bases.

(10) Prior to the completion of fabrication for each package, a gamma scan test of the steel and lead shielding shall be conducted over 100% of all accessible cask surfaces to verify shield integrity.

(11) For the fabrication of each package, the material content of the NS-4-FR neutron shielding material shall be tested and certified.

(12) Prior to the first shipment of each package, neutron and gamma shield effectiveness tests shall be performed. Measurement points shall be established to adequately demonstrate, for the entire package, the effectiveness of the neutron and gamma shielding materials.

(13) Each lot of Boral shall be tested using wet chemistry and/or neutron attenuation techniques to verify the minimum ^{10}B content. The test shall be representative of each Boral panel. The minimum allowable ^{10}B content is 0.01 g/cm^2 for the Yankee Class fuel tubes (Drawing No. 455-881) and 0.02 g/cm^2 ^{10}B for the directly loaded fuel tubes (Drawing No. 423-875).

11. 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.

12. 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.

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

14. Expiration date: March 31, 2004.

REFERENCES

NAC International, Inc., application dated December 30, 1996.

NAC International, Inc. supplements dated April 30, May 7, July 28 and 31, 1997; August 7, December 5, 12, 19, and 30, 1998; and January 15, February 12, 23, and 27, and March 1 and 22, 1999.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

E. William Brach

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

Date March 25, 1999

NRC FORM 618
(3-96)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIALS PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. PACKAGE IDENTIFICATION NUMBER	d. PAGE NUMBER	e. TOTAL NUMBER PAGES
9239	10	USA/9239/AF	1	4

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

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 January 31, 1991, as supplemented.

c. DOCKET NUMBER 71-9239

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 $Gd_2 0_3$ neutron absorber plates that are mounted on the center wall of the strongback. Additional horizontal $Gd_2 0_3$ 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 $Gd_2 0_3$ neutron absorber plates. Additional vee-shaped, guided $Gd_2 0_3$ 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.

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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 and 3, Rev. 5.

The MCC-4 packaging is constructed in accordance with Westinghouse Electric Corporation Drawing No. MCCL401, Sheets 1, 2, 3, and 4, Rev. 7.

The MCC-5 packaging is constructed in accordance with Westinghouse Electric Corporation Drawing No. MCCL501, Sheets 1 through 9, Rev. 4.

(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. 9, dated October 11, 1999	Fuel Assembly Parameters 14x14 Type Fuel Assemblies
Table 1-4.2, Rev. 9, dated October 11, 1999	Fuel Assembly Parameters 15x15 Type Fuel Assemblies
Table 1-4.3, Rev. 9, dated October 11, 1999	Fuel Assembly Parameters 16x16 Type Fuel Assemblies*
Table 1-4.4, Rev. 9, dated October 11, 1999	Fuel Assembly Parameters 17x17 Type Fuel Assemblies*
Table 1-4.5, Rev. 4, dated January 14, 1994	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.

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5. (b) Contents (continued)

(2) Maximum quantity of material per package

Two (2) fuel assemblies

(c) Transport Index for Criticality Control

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

6. For shipments of 14x14, 15x15, 16x16, and 17x17 fuel assemblies with U-235 enrichments of over 4.65 wt% and up to 5.0 wt%, horizontal Gd_2O_3 neutron absorber plates shall be positioned underneath each assembly. The horizontal absorber plates shall be placed horizontally on the underside of the strongback, as shown on Westinghouse Electric Corporation Drawing No. MCCL301, Sheet 1, Rev. 5, or Westinghouse Electric Corporation Drawing No. MCCL401, Sheet 1, Rev. 7.

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 shown on Westinghouse Electric Corporation Drawing No. MCCL501, Rev. 4.

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. 2, dated January 14, 1994, of the application. 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) The MCC-3 packaging shall be acceptance tested in accordance with Notes 3, 4, and 5 of Westinghouse Electric Corporation Drawing No. MCCL301, Sheet 1, Rev. 5, and with the Acceptance Tests in supplement dated March 24, 1997.

(b) The MCC-4 packaging shall be acceptance tested in accordance with Notes 4, 5, and 6 of Westinghouse Electric Corporation Drawing No. MCCL401, Sheet 2, Rev. 7, and with the Acceptance Tests in supplement dated March 24, 1997.

(c) The MCC-5 packaging shall be acceptance tested in accordance with the Acceptance Tests in supplement dated March 24, 1997.

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(d) The packages shall be maintained in accordance with the Maintenance Program in supplement dated March 24, 1997.

(e) The packages shall be operated and prepared for shipment in accordance with the Operating Procedures in supplement dated January 14, 1994, as revised in supplement dated August 2, 1994.

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

12. Expiration date: March 31, 2002.

REFERENCES

Westinghouse Electric Corporation application dated January 31, 1991.

Supplements dated: October 2, October 9, November 1, and November 13, 1991; January 27, March 30, May 12, and June 18, 1992; August 18, 1993; January 14, April 22, May 24, July 26, and August 2, 1994; October 1, 1996; March 24 and December 22, 1997; September 28, 1998, February 19, February 22, July 28, 1999, August 2, October 13, and December 3, 1999.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Susan J. Shankman for

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

Date: December 28, 1999

NRC FORM 618
(8-85)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIALS PACKAGES

1.a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. PACKAGE IDENTIFICATION NUMBER	d. PAGE NUMBER	e. TOTAL NUMBER PAGES
9243	0	USA/9243/B(U)	1	2

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

Amersham Corporation
40 North Avenue
Burlington, MA 01803

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION:

Amersham Corporation application dated September 13, 1994, as supplemented.

c. DOCKET NUMBER 71-9243

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.: 934

(2) Description

A welded stainless steel encased, uranium shielded, radiographic exposure device. Primary components consist of an outer stainless steel shell, internal supports, depleted uranium shield, and a bronze source tube. The contents are securely positioned in the source tube by a padlocked shipping plug. The unit resembles a right cylinder approximately 20.5 inches long and 6 inches in diameter. The maximum weight of the package is 68 pounds.

(3) Drawings

The package is constructed and assembled in accordance with Amersham Corporation Drawing No. 93400D, Rev. D, Sheets 1 through 8.

(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

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".

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6. The source shall be secured in the shielded position of the packaging by the operational locking plunger, shipping plug and padlock. The operational locking plunger, 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 operational locking plunger must engage the locking device during transport. The source rod assembly and the shipping plug must be of sufficient length and diameter to provide positive positioning of the source in the shielded position.

7. The nameplate 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, 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 §71.12.

10. Expiration date: January 31, 200[illegible].

REFERENCES

Amersham Corporation application dated September 13, 1994.

Amersham Corporation supplements dated March 30, and October 16, 1995; and February 1, 1996.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

William Travers

William D. Travers, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: 2/6/96

NRC FORM 618
(3-96)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIALS PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. PACKAGE IDENTIFICATION NUMBER	d. PAGE NUMBER	e. TOTAL NUMBER PAGES
9245	4	USA/9245/B(U)	1	2

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

Burnley Technology, Inc.
Post Office Box 1226
Plaistow, NH 03865

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION:

RTS Technology, Inc., application dated August 20, 1992, as supplemented.

c. DOCKET NUMBER 71-9245

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. 420

(2) Description

Radiographic device within a protective overpack. The overpack consists of an outer container which is a 10-gallon open head steel drum (14 inches in diameter and 17.25 inches in height) having a minimum 20-gauge body and cover, welded seams and a clamp-ring type head closure. The void space between the inner and outer container is filled with 1-1/2 inch thick molded asbestos free liner on the top, bottom, and sides, plus molded polyurethane fill to position and secure the radiographic device within the drum. Maximum gross weight of the package not to exceed 75 pounds. The maximum weight of the radiography devices within the package shall not exceed 51 pounds.

(3) Drawings

The overpack must be constructed in accordance with Burnley Technology Inc., Drawing Nos. 42400, Rev. 0; 42500, Rev. 0; and 42600, Rev. 0.

The radiographic devices, as secondary packaging, authorized for use in the overpack are constructed in accordance with the following Drawing Nos.:

For the Model No 424: RTS Technology, Inc., Drawing Nos. 42401, Sheets 1 & 2, Rev. 1; 42402, Rev. 3; 42403, Rev. 3; 42404, Sheets 1, 2, and 3, Rev. 0; 42407, Rev. 0; 42408, Rev. 0; 42415, Rev. 3; 42416, Rev. 0; 42417, Rev. 1; 42421, Rev.0; 42422, Rev. 0; 42423, Sheets 1 & 2, Rev. 1; 42424, Rev. 1; 42425, Rev. 0; and 42426, Rev. 0.

For the Model No. 425: RTS Technology, Inc., Drawing Nos. 42501, Rev. 0; 42503, Rev. 0; 42505, Sheets 1 & 2, Rev. 0; 42506, Rev. 2; 42551, Rev. 1; 42552, Rev. 0; and 42558, Rev. 0.

For the Model No. 426: RTS Technology, Inc., Drawing Nos. 42601, Rev. 0; 42605, Rev. 0; 42606, Rev. 0; and 42609, 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

200 Curies.

6. The sources shall be secured in the shielded position of the radiographic device 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 ball stop 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. 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 Operating Procedures in Chapter 7 of the application.

(b) Each packaging must meet the Maintenance Program and Acceptance Tests in Chapter 8 of the application, except the container should be visually examined to assure that the container meets the specifications as described in the drawings specified in 5(a)(3) of the certificate.

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

9. Expiration date: June 30, 2002.

REFERENCES

RTS Technology, Inc., application dated August 20, 1992.

RTS Technology, Inc., supplements dated: July 31, 1991, and October 16, 1992.

Burnley Technology, Inc., supplements dated: October 16, 1992, June 17, 1994, and May 14, 1997.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Cass R. Chappell

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

Date: July 8, 1997

NRC FORM 618
(3-96)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIALS PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. PACKAGE IDENTIFICATION NUMBER	d. PAGE NUMBER	e. TOTAL NUMBER PAGES
9246	2	USA/9246/AF	1	2

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

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.

c. DOCKET NUMBER 71-9246

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 lbs., 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.

Page 2 - Certificate No. 9246 - Revision No. 2 - Docket No. 71-9246

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 360 grams U-235. The total quantity of radioactive material within a package may not exceed a Type-A quantity.

(c) Transport Index for Criticality Control

Maximum transport index to be shown on Label for nuclear criticality control: 50.0

6. In addition to the requirements of Subpart G of 10 CFR Part 71, the package shall be prepared for shipment, operated, and maintained in accordance with the loading, unloading, and quality assurance procedures in the application. Prior to each shipment, the shipper shall make the determinations specified in the NIST "ST" Series Shipping Container Shipper's Checklist in the application.

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

8. Expiration date: September 30, 2001.

REFERENCES

National Institute of Standards and Technology application dated February 7, 1992.

Supplements dated: February 14, 1992; and August 7, 1996.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Cass R. Chappell

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

Date: October 1, 1996

NRC FORM 618
(3-96)
10 CFR 71

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIALS PACKAGES

U.S. NUCLEAR REGULATORY COMMISSION

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. PACKAGE IDENTIFICATION NUMBER	d. PAGE NUMBER	e. TOTAL NUMBER PAGES
9248	14	USA/9248/AF	1	5

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

Siemens Power Corporation
2101 Horn Rapids Road
Richland, WA 99352-0130

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION:

Siemens Power Corporation application dated November 25, 1998, as supplemented.

c. DOCKET NUMBER 71-9248

4. CONDITIONS

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

5.

(a) Packaging

(1) Model Nos.: SP-1, SP-2, and SP-3

(2) Description

Fuel assembly and fuel rod shipping containers. The packages consist of a right rectangular metal inner container and a wooden outer container, with cushioning material between the inner and outer containers.

The metal inner container is approximately 11-1/2 inches by 18 inches by 179-1/2 inches long and is positioned within a wooden outer container approximately 30 inches by 31 inches by 207 inches long. The SP-1 and SP-2 packagings differ in the length of the metal inner container and end piece. The SP-3 packagings have a reduced spacing between the fuel assembly channels and the outer surface of the metal inner container. Cushioning is provided between the inner and outer containers by phenolic impregnated honeycomb and ethafoam, or equivalent. Closure of the metal inner container and the wooden outer container is accomplished by bolts. A pressure relief (breather) valve is provided on the inner container, and is set for 0.5 psi differential. The maximum weight of the packaging and contents is 2,800 pounds.

(3) Drawings

The packagings are fabricated and assembled in accordance with the following Siemens Nuclear Power Corporation/Advanced Nuclear Fuels Corporation Drawing Nos.:

EMF-304,416, Rev. 13.
EMF-306,272, Rev. 9.
EMF-308,257, Rev. 5.
EMF-309,141, Rev. 1.
EMF-309,818, Rev. 0.

5.(a) (4) Product Containers

(i) Five-inch, Schedule 40, stainless steel pipe fitted with screw type or flange closure. The product container shall be vented if it contains materials which decompose at less than 1475 °F.

(ii) Rod shipping container as shown on Siemens Power Corporation Drawing No. EMF-309,141, Rev. 1.

(b) Contents

(1) Type and form of material

(i) UO_2 fuel assemblies in a 7 x 7, an 8 x 8, or a 9 x 9 square array with a maximum fuel cross-section area of 25 square inches, maximum fuel length of 174 inches and maximum average enrichment of 3.3 w/o U-235. Minimum zircaloy clad thickness is 0.025 inches; maximum pellet diameter is 0.555 inches. Any number of water rods in any arrangement is permitted.

(ii) UO_2 fuel assemblies in a 7 x 7, an 8 x 8, or a 9 x 9 square array with a maximum fuel length of 174 inches, and a maximum average enrichment between 3.3 to 4.0 w/o U-235. The maximum pellet diameter is 0.555 inch, and the minimum clad thickness is 0.025 inch. Any number of water rods in any arrangement is permitted, including part length rods. Each assembly contains at least 4 rods with nominal 2 weight percent Gd_2O_3, which are in non-perimeter locations and are symmetric about the diagonal.

(iii) UO_2 fuel assemblies with a maximum U-235 enrichment of 5.0 percent by weight, and a maximum average U-235 enrichment of 4.0 percent by weight. Each fuel assembly is made up of fuel rods in a 10 x 10 square array, with a maximum fuel cross section of 5.022 inches square, a nominal pitch of 0.511 inch, and a maximum fuel length of 174 inches. The maximum pellet diameter is 0.3356 inch, the minimum clad thickness is 0.0225 inch, and the maximum U-235 enrichment in any edge rod is 4.0 percent by weight. Each assembly contains at least 6 rods with nominal 2 weight percent Gd_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) (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), and 5(b)(1)(viii):

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.

(c) Transport Index for Criticality Control

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), and 5(b)(1)(vii), and limited in 5(b)(2)(i): 1.0

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

7. Polyethylene shipping shims may be inserted between rods within fuel assemblies as follows:

(a) For contents described in 5(b)(1)(i) and 5(b)(1)(ii), up to a maximum of 0.20 gram H_2O hydrogen equivalent per cubic centimeter averaged over the assembly.

(b) For contents described in 5(b)(1)(v), up to a maximum of 0.25 gram H_2O hydrogen equivalent per cubic centimeter averaged over the assembly.

(c) For contents described in 5(b)(1)(viii), up to a maximum volume fraction of 0.13 averaged over the void volume of the assembly.

(d) For contents described in 5(b)(1)(iii), 5(b)(1)(vi), and 5(b)(1)(vii), polyethylene shipping shims are not permitted.

8. Only contents described in 5(b)(1)(viii) are authorized for transport in Model No. SP-3 packages.

9. Maximum average enrichment means the highest average enrichment through any cross sectional plane of the assembly.

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

    (a) The package must be prepared for shipment and operated in accordance with the Operating Procedures in Chapter 7 of the application dated November 25, 1998.

    (b) Each packaging must be acceptance tested and maintained in accordance with the Acceptance Tests and Maintenance Program in Chapter 8 of the application dated November 25, 1998.

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

12. Expiration date: February 28, 2004.

REFERENCES

Siemens Power Corporation application dated November 25, 1998.

Supplements dated: December 2, and 15, 1998; and February 23, April 12, and July 27, 1999.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

E. William Brach

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

Date 8/30/99

NRC FORM 618
(3-96)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIALS PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. PACKAGE IDENTIFICATION NUMBER	d. PAGE NUMBER	e. TOTAL NUMBER PAGES
9250	7	USA/9250/B(U)F-85	1	5

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

Babcock and Wilcox Company
P. O. Box 785
Lynchburg, VA 24505

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION:

Babcock and Wilcox Company
application dated December 17, 1997.

c. DOCKET NUMBER 71-9250

4. CONDITIONS

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

5.

(a) Packaging

(1) Model No.: NNFD 5X22

(2) Description

A shipping container for unirradiated uranium of any enrichment. The outer packaging is a 16-gauge steel drum, approximately 22-1/2 inches in diameter and 34-3/4 inches high, with a heavy-duty clamp ring and forged lugs. The inner vessel (containment vessel) is a Schedule 40S stainless steel pipe with a welded bottom cap and a top weldneck flange. The inner vessel lid is a blind flange which is bolted to the weldneck flange with eight hex-head bolts. The closure includes double silicone O-ring seals and a leak-test port. The dimensions of the inner vessel are approximately 5 inches ID by 22 inches high. The inner vessel is centered within the outer drum by fiberboard and supported by plywood disks. The maximum weight of the package, including contents, is 300 pounds.

(3) Drawings

The packaging is constructed in accordance with Babcock & Wilcox Company Drawing Nos. 1220276E, Rev. 2, and 1220277E, Rev. 5.

(b) Contents

(1) Type and form of material

(i) Unirradiated uranium as solid compounds or alloys which do not decompose at temperatures up to 250 °F, and uranium oxides as powder or pellets. The uranium may be of any U-235 or U-233 enrichment. Carbide compounds are not authorized.

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(b) Contents (Continued)

(1) Type and form of material (Continued)

(ii) Unirradiated solid uranyl nitrate in the form of uranyl nitrate dihydrate crystals, which may have small amounts of uranyl trihydrate crystals interspersed. The uranyl nitrate crystals shall have a uranium content that is from 52.5 to 56.0 weight percent. The uranyl nitrate shall be packaged in Teflon primary containers that will not melt at temperatures up to 94 °C. The uranium may be of any U-235 enrichment.

(iii) Unirradiated uranium as solid metal. The uranium may be of any U-235 enrichment.

(iv) Unirradiated liquid uranyl nitrate solution in sealed glass containers or screw top plastic vials, each within one or more additional plastic vials with taped lids, and within a sealed product can or polyethylene bottle containing a sufficient amount of vermiculite to absorb twice the liquid contents present. The uranium may be of any U-235 enrichment. U-233 greater than a Type A quantity is not permitted.

(2) Maximum quantity of material per package and transport index for criticality control

The weight of the contents, including secondary containers, inserts, and other materials in the inner vessel, shall not exceed 50 pounds, and:

(i) For the material described in Items 5(b)(1)(i) and 5(b)(1)(ii), above, with a maximum H/U of 3, considering all sources of moderation in the inner vessel:

Fissile Material	Maximum Fissile Material per Package (kg)	Minimum Transport Index to be Shown on Label for Nuclear Criticality Control
U-235	9.0	2.0
U-235	1.6	0.5

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5.(b) Contents (continued)

(2) Maximum quantity of material per package and transport index for criticality control (continued)

(ii) For the material described in Items 5(b)(1)(i) and 5(b)(1)(ii), above, with a maximum H/U of 20, considering all sources of moderation in the inner vessel:

Fissile Material	Maximum Fissile Material per Package (kg)	Minimum Transport Index to be Shown on Label for Nuclear Criticality Control
U-233	0.5	1.8
U-235	4.0	2.0

(iii) For the material described in Item 5(b)(1)(iii), above, with a maximum H/U of 3, considering all sources of moderation in the inner vessel:

Fissile Material	Maximum Fissile Material per Package (kg)	Minimum Transport Index to be Shown on Label for Nuclear Criticality Control
U-235	9.0	2.5
U-235	1.6	0.5

(iv) For the material described in Item 5(b)(1)(iii), above, with a maximum H/U of 3, considering all sources of moderation in the inner vessel, and with a solid aluminum disk insert positioned in the inner vessel, as shown on Babcock & Wilcox Company Drawing No. 1220277E, Rev. 5 (Part No. 6).

Fissile Material	Maximum Fissile Material per Package (kg)	Minimum Transport Index to be Shown on Label for Nuclear Criticality Control
U-235	9.0	2.0

Page 4 - Certificate No. 9250 - Revision No. 7 - Docket No. 71-9250

5.(b) Contents (continued)

(2) Maximum quantity of material per package and transport index for criticality control (continued)

(v) For the material described in Item 5(b)(1)(iii), above, with a maximum H/U of 20, considering all sources of moderation in the inner vessel:

Fissile Material	Maximum Fissile Material per Package (kg)	Minimum Transport Index to be Shown on Label for Nuclear Criticality Control
U-235	4.0	2.0
U-233	0.5	1.8

(vi) For the material described in Item 5(b)(1)(iv), above:

Fissile material shall not exceed 400 grams U-235. The quantity of uranyl nitrate shall not exceed 1000 mL of solution.

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

6. The vent holes on the outer steel drum shall be capped or taped closed during transport and storage to preclude entry of rain water into the packaging.

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

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

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

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

9. Expiration date: January 31, 2003.

Page 5 - Certificate No. 9250 - Revision No. 7 - Docket No. 71-9250

REFERENCES

Babcock and Wilcox Company application dated December 17, 1997.

Supplement dated: March 25, 1998

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Cass R. Chappell

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

Date: May 14, 1998

NRC FORM 618
(3-96)
10 CFR 71

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIALS PACKAGES

U.S. NUCLEAR REGULATORY COMMISSION

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. PACKAGE IDENTIFICATION NUMBER	d. PAGE NUMBER	e. TOTAL NUMBER PAGES
9251	9	USA/9251/AF	1	3

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

Framatome Cogema Fuels
P.O. Box 11646
Lynchburg, VA 24506-1646

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION:

B&W Fuel Company application dated May 26, 1992, as supplemented.

c. DOCKET NUMBER 71-9251

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

(I) Model No.: BW-2901

(2) Description

A shipping container for low-enriched uranium oxide powder and pellets, composed of an inner container, surrounded by insulating material, and an outer drum. The inner cross sectional dimensions of the inner container are a maximum 11.15-inch square by 29.5-inch long. The inner container is constructed of minimum 14-gauge steel, with bolted and gasketed top flange closure and welded bottom sheet. The inner container is centered and supported in an 18-gauge steel drum with 16-gauge head and DOT Specification 17H or an equivalent DOT UN1A2/Y1.5/100 closure by asbestos or ceramic sheet, plywood, hardboard, and insulating material. The drum has approximate inner cross sectional dimensions of 22.5-inch by 34-inch height. The uranium oxide is packaged in boxes, and wood boards position the boxes within the inner container. Three borated aluminum plates (approximately 25 inches by 9.25 inches by 0.375 inch) are positioned within the inner container. The maximum gross weight of the package is 660 pounds.

(3) Drawings

The packaging is constructed in accordance with B&W Fuel Company Drawing Nos. 1215597D, Rev. 5, 1215598B, Rev. 1, 1215599E, Rev. 4, and 1283759D, Rev. 0.

(b) Contents

(l) 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 1283759D, 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

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:

(i) Each packaging must be maintained and acceptance tested in accordance with Chapter 8 of the application; and

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

(iii) Prior to each shipment the insert (containment vessel) gasket shall be inspected. This gasket shall be replaced if inspection shows any defects or every twelve (12) months, whichever occurs first.

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

10. Expiration date: September 30, 2002.

REFERENCES

B&W Fuel Company application dated May 26, 1992.

Supplements dated: August 3 and October 30, 1992; April 30, 1993; May 24 and September 22, 1995; February 29, April 22, and July 1, 1996; July 30, 1997; and March 26, 1999.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Cass R. Chappell

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

Date: 5/6/99

NRC FORM 618
(3-96)
10 CFR 71

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIALS PACKAGES

U.S. NUCLEAR REGULATORY COMMISSION

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. PACKAGE IDENTIFICATION NUMBER	d. PAGE NUMBER	e. TOTAL NUMBER PAGES
9252	3	USA/9252/AF	1	3

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

Framatome Cogema Fuels
PO 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.

c. DOCKET NUMBER 71-9252

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.: 1215926C, Rev. 1; 1215929D, Rev. 2; 1215930D, Rev. 2; 1215931D, Rev. 2; 1215932D, Rev. 2; 1215933D, Rev. 2; 1215934C, Rev. 1; 1215935D, Rev. 2; 1216010D, Rev. 1.

(b) Contents

(1) Type and form of material

Unirradiated fuel assemblies, composed of uranium dioxide fuel pellets clad in zircaloy tubes. Uranium is enriched to a maximum of 5.05 w/o in the U-235 isotope. The fuel assemblies may contain inserted control rod assemblies. The fuel assemblies have the following specifications:

Type	15x15	15x15	17x17	17x17	15x15
Rods Per Assembly	208	204	264	264	204
Nominal Rod Pitch (in.)	0.568	0.563	0.501	0.496	0.5625
Maximum Pellet Diameter (in.)	0.3707	0.3671	0.3252	0.3232	0.3672
Maximum Pellet Density (%TD)	97.5	97.5	97.5	97.5	97.5
Nominal Clad OD (in.)	0.430	0.422	0.379	0.374	0.422
Nominal Clad ID (in.)	0.377	0.370	0.332	0.326	0.368
Assembly Cross Section (in.)*	8.520	8.445	8.517	8.432	8.438
Active Fuel Length (in.)	144	144	144	144	120
Maximum U-235 Loading (kg)	25.20	24.24	24.62	24.32	20.20

* Assembly cross section is the product of the nominal rod pitch and the number of rods per edge.

(2) Maximum quantity of material per package

Two fuel assemblies. Total weight of fuel assemblies, including control rod assemblies, not to exceed 3400 pounds. Maximum quantity of radioactive material within a package may not exceed a Type A quantity.

5. (c) Transport Index for Criticality Control

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

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

7. Hydrogenous shims are not permitted within the fuel assemblies.

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

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

   (b) Each packaging shall be maintained in accordance with Section 8.2 of the application.

   (c) Each packaging shall meet the acceptance tests in Section 8.1 of the application.

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, 2003.

REFERENCES

B&W Fuel Company application dated March 9, 1993.

Supplements dated: May 10, and July 7, 1993; April 13, 1994; August 6, 1998.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Cass R. Chappell

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

Date: 09/25/98

NRC FORM 618
(3-96)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIALS PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. PACKAGE IDENTIFICATION NUMBER	d. PAGE NUMBER	e. TOTAL NUMBER PAGES
9253	4	USA/9253/B(U)F	1	3

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

U.S. Department of Energy
Washington, DC 20585

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION:

Safety Analysis Report for the TN-FSV Packaging dated March 31, 1993, as supplemented

c. DOCKET NUMBER 71-9253

4. CONDITIONS

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

5.

(a) Packaging

(1) Model No.: TN-FSV

(2) Description

A steel and lead shielded shipping cask for irradiated high temperature gas cooled reactor (HTGR) fuel elements. 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	42,000 pounds
Gross package weight, including contents	47,000 pounds

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 silicone O-rings, equipped with a leak test port. A vent port and drain port are sealed with single silicone O-rings and cover plates. 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.

Page 2 - Certificate No. 9253 - Revision No. 4 - Docket No. 71-9253

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

The cask has two lifting sockets bolted to the cask top flange. Two rear trunnions are provided for cask tie-down.

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.

(3) Drawings

The packaging is constructed and assembled in accordance with the following Transnuclear, Inc. Drawing Nos.:

1090-SAR-1, Rev. 2	1090-SAR-6, Rev. 2
1090-SAR-2, Rev. 2	1090-SAR-7, Rev. 2
1090-SAR-3, Rev. 2	1090-SAR-8, Rev. 2
1090-SAR-4, Rev. 2	1090-SAR-9, Rev. 2
1090-SAR-5, Rev. 2	1090-SAR-10, Rev. 1

(b) Contents

(1) Type and form of material

Irradiated HTGR fuel elements. 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.

(2) Maximum quantity of material per package

Six fuel elements, with decay heat not to exceed 60 watts per fuel element. The fuel elements are contained within a fuel storage container. Total weight of contents not to exceed 5,000 pounds, including fuel elements, fuel storage container, and depleted uranium shield plug.

(c) Transport Index for Criticality Control

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

Page 3 - Certificate No. 9253 - Revision No. 4 - Docket No. 71-9253

6. The package must be leak tested as follows:

   (a) Within the 12-month period prior to shipment, and after seal replacement, the package must be tested to show a leak rate no greater than 1×10^{-3} std-cm^3/sec. The leak test must have a sensitivity of at least 5×10^{-4} std-cm^3/sec.

   (b) 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 application. The acceptance criterion is a leak rate no greater than 1×10^{-3} std-cm^3/sec. The test must have a sensitivity of at least 1×10^{-3} std-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.

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 application.

   (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 application.

   (c) Prior to each shipment, 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.

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

9. Expiration date: May 31, 2004.

REFERENCES

Public Service Company of Colorado application dated March 31, 1993, as supplemented February 24, June 2, and June 14, 1994; September 11, and December 7, 1995.

U.S. Department of Energy supplements dated: March 24, 1997, and March 24, 1999.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Cass R. Chappell

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

Date: May 3, 1999

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIALS PACKAGES

U.S. NUCLEAR REGULATORY COMMISSION

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. PACKAGE IDENTIFICATION NUMBER	d. PAGE NUMBER	e. TOTAL NUMBER PAGES
9255	4	USA/9255/B(U)F-85	1	7

2. PREAMBLE

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

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

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

a. ISSUED TO (*Name and Address*)

Transnuclear West Inc.
39300 Civic Center Drive
Suite 280
Fremont, California 94538-2324

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION:

Pacific Nuclear Systems Inc. application dated October 8, 1993, as supplemented (Application and Technology are owned By Transnuclear, Inc.)

c. DOCKET NUMBER 71-9255

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.: NUHOMS® MP187 Multi-Purpose Cask

(2) Description:

The NUHOMS® MP187 Multi-Purpose Cask (package) consists of an outer cask, into which one of three 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, three 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.

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 7,
Sheets 1 through 2
MP187 Multi-Purpose Cask
General Arrangement

NUH-05-4001, Revision 13,
Sheets 1 through 6
MP187 Multi-Purpose Cask
Main Assembly

NUH-05-4002, Revision 4,
Sheets 1 and 2
MP187 Multi-Purpose Cask
Impact Limiters

NH-05-4003, Revision 8,
Sheets 1 and 2
NUHOMS® MP187 Multi-Purpose Cask
On-Site Transfer Arrangement

NUH-05-4004, Revision 13,
Sheets 1 through 4
NUHOMS® FO-DSC & FC-DSC
PWR Fuel Main Assembly

NUH-05-4005, Revision 11,
Sheets 1 through 4
NUHOMS® FF-DSC
PWR Fuel Main Assembly

NUH-05-4006NP, Revision 6,
Sheets 1 and 2
NUHOMS® MP187 Multi-Purpose
Transportation Skid/Personnel Barrier

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 or the FC-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 the FF-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) The fuel authorized for shipment in the NUHOMS®-MP187 package 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

(d) Intact fuel assemblies without control components shall be shipped only in the FO-DSC.

(e) Intact fuel assemblies with control components shall be only shipped in the FC-DSC.

(f) The maximum burn-up and minimum cooling times for the individual 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.

5.b Contents of Packaging:

(1) Type and Form of Material Continued:

(g) The maximum assembly decay heat (including control components when present) of an individual assembly is 0.764 kW, referred to as Type I, or 0.563 kW, referred to as Type II.

(h) Control components shall be cooled for at least 8 years.

(2) Maximum quantity of material per package:

(a) For material described in 5.b(1): 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.

(b) For material described in 5.b(1): the approximate maximum payload (including control components when present) is 81,100 lbs.

Table 1

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			

5.c. Transport Index for Criticality Control

Minimum transport index to be shown on the label for nuclear criticality control: "0"

6. Type I fuel assemblies shall be loaded only into the four innermost cells of a DSC, while Type II assemblies may be loaded into any cell when using the FO-DSC or the FC-DSC. The FF-DSC has no Type I or II placement restrictions.

7. Fuel assemblies with missing fuel rods shall not be shipped unless dummy fuel pins that displace an equal amount of water have been installed in the fuel assembly.

8. 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 include the following provisions:

      (1) a loading plan which has been independently verified and approved by a qualified individual other than the developer(s) which shall include:

         (a) hold points to verify that all fuel movements are performed under strict verbatim compliance with the fuel movement schedule;

         (b) videotaping and independent verification by ID number of each fuel assembly loaded; and

         (c) a final independent verification of the fuel placement.

      (2) procedures requiring 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 measurement instruments are calibrated for the energy spectrum of neutrons being emitted from the package;

         (b) verify that the surface removable contamination levels meet the requirements of 49 CFR 173.443 and 10 CFR 71.87; and

         (c) leak test containment vessel seals to verify a leak rate of less than 1×10^{-7} standard cubic centimeters per second of helium (std-cc/sec). The leak test shall have a test sensitivity of at least 5×10^{-8} std-cc/sec and shall be conducted:

            1) before first use of each package,

            2) within the 12-month period prior to each shipment, and

            3) after seal replacement.

      (3) procedures that require that the package metallic seals be replaced after each use.

   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:

8. For operating controls and procedures, in addition to the requirements of Subpart G of 10 CFR Part 71:
b.(1) continued

(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-5000.

(2) The DSC outer top cover plate weld shall be verified 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) Before joining the structural shell to the inner shell, the upper lifting trunnions shall be load tested to 150% of their maximum working load or 188,000 lbs. minimum per trunnion, in accordance with the requirements of ANSI N14.6-1986.

(4) 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 75 psig.

(5) The fabrication verification leak test for the inner shell shall be performed after initial fabrication, but before the lead pour, to verify that the leak rate from the cylindrical containment shell is less than 1 X 10^{-7} std-cc/sec. A second fabrication verification leak test shall be performed on the finished cask to demonstrate a leak rate of less than 1 x 10^{-7} std-cc/sec for the package. The results of both tests shall have at least, a sensitivity of 5 x 10^{-8} std-cc/sec.

(6) The poured-lead shielding integrity of the MP187 cask body shall be confirmed via gamma scanning prior to installation of the neutron shield. The scan shall utilize, at a maximum, a 6x6-inch test grid. The minimum lead thickness in the main cask body, away from the trunnions and the top and bottom forgings, shall be 3.90 inches

(7) The neutron shield shall have a minimum thickness of 4.31 inches. Its integrity shall be confirmed through a strict combination of fabrication process control and verification by measurement. This may be done either at first use or with a check source using, at a maximum, a 6x6-inch test grid.

(8) The complete cask shall be subjected to a thermal heat rejection test to demonstrate satisfactory operation of the as-built shells, top lid, and shielding materials. This test may be performed without the ram closure installed. Acceptance criteria shall be calculated for the change in temperature across the cask wall based on the applied heat load and existing environmental conditions using the same analytical methods used to predict the cask performance for the normal and accident conditions.

8. For operating controls and procedures, in addition to the requirements of Subpart G of 10 CFR Part 71: b. continued

   (9) Foam shall be installed within the cask impact limiters and tested to ensure conformance with the required foam material properties.

   (10) The neutron absorber plate's minimum acceptable areal boron content loading is 0.025 g/cm ^{2} Boron 10. The minimum Boron 10 content per unit area and the uniformity of dispersion within the sandwiched material shall be verified by testing each sheet with a sufficient sensitivity (at least to the 95/95 confidence level) to assure compliance with the drawings.

   (11) The impact limiters shall be visually inspected within 1 year of use for water absorption or degradation. Each impact limiter shall also be weighed at the time of inspection. If the weight has increased more than 3%, the impact limiter shall be repaired or replaced.

9. This package is approved for exclusive use rail, truck or marine transport.

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: September 10, 2003.

REFERENCES

Transnuclear West Inc., consolidated Safety Analysis Report for the NUHOMS® MP187 Multi-Purpose Cask, dated August 28, 1998, as supplemented February 2, and March 9, 1999.

Transnuclear West Inc., amendment requests dated December 18, 1998, as supplemented May 20, and October 1, 1999; and March 17, 2000.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

M. Wayne Hodges
for

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

Date: March 30, 2000

NRC FORM 618
(3-96)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIALS PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. PACKAGE IDENTIFICATION NUMBER	d. PAGE NUMBER	e. TOTAL NUMBER PAGES
9258	0	USA/9258/B(U)-85	1	3

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

MDS Nordion
447 March Road
Kanata, Ontario, Canada, K2K 1X8

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION:

MDS Nordion application dated June 30, 1998, as supplemented.

c. DOCKET NUMBER 71-9258

4. CONDITIONS

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

5.

(a) Packaging

(1) Model No.: F-294

(2) Description

A steel encased, lead shielded shipping cask for special form sources. The package consists of a cylindrical cask body with cooling fins, a closure plug, a cylindrical external fireshield, a top crush shield, a permanent skid, and a removable shipping skid. The special form sources are positioned by a source carrier within the cask cavity.

The cask body is constructed of a 1/2-inch thick inner stainless steel shell, and a 1/2-inch thick outer stainless steel shell. The annulus between the inner and outer shells is filled with lead, approximately 11 1/4 inches thick. The cask is closed by a 2 1/2 inch thick stainless steel closure lid and 16 one-inch diameter bolts. A lead radiation protection plug is fitted to the cask closure plate. Stainless steel fins are welded onto the exterior of the cask to dissipate heat. The cask is surrounded by a cylindrical fireshield which is constructed of ceramic fiber thermal insulation encased in mild steel shells. A composite assembly consisting of a finned crush shield that acts as an impact limiter and a fireshield is bolted to the top end of the cask. The cask is equipped with a fixed skid and a shipping skid composed of steel beams. The fixed skid includes a sheet of thermal insulation enclosed in steel.

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5(a)(2) cont. The approximate dimensions and weights of the package are as follows:

Cask body outer diameter (excluding cooling fins)	36 inches
Cask body height	52 1/4 inches
Cask cavity inside diameter	11 1/2 inches
Cask cavity inside height	19 3/4 inches
Lead shield thickness	11 1/4 inches
Fire shield outer diameter	47 inches
Overall package dimensions (including shipping skid)	
width	78 inches
length	78 inches
height	80 1/2 inches
Maximum contents weight	20 pounds
Maximum package weight (including contents)	21,000 pounds

(3) Drawings

The packaging is constructed in accordance with MDS Nordion drawing Nos.:

F629401-001, Sheets 1-5, Rev. D, and
F631301-001, Rev. B.

(b) Contents

(1) Type and form of material

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

(2) Maximum quantity of material per package

360,000 Curies

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

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

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

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

8. Expiration date: December 31, 2003.

REFERENCES

MDS Nordion application dated June 30, 1998.

Supplement dated: December 11, 1998.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Cass R. Chappell

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

Date: 1/6/99

NRC FORM 618
(3-96)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIALS PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. PACKAGE IDENTIFICATION NUMBER	d. PAGE NUMBER	e. TOTAL NUMBER PAGES
9261	1	USA/9261/B(U)F-85	1	7

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

Holtec International
Holtec Center
555 Lincoln Drive West
Marlton, NJ 08053

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION:

Holtec International Report No. HI-951251, *Safety Analysis Report for the Holtec International Storage, Transport, And Repository Cask System (HI-STAR 100 Cask System),* Revision 9, dated April 20, 2000.

c. DOCKET NUMBER 71-9261

4. CONDITIONS

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

5.

5 a. Packaging

(1) Model No.: HI-STAR 100 System

(2) Description

The HI-STAR 100 System is a canister system comprising a Multi-Purpose Canister (MPC) inside of an overpack designed for both storage and transportation (with impact limiters) of irradiated nuclear fuel. The HI-STAR 100 System consists of interchangeable MPCs which house the spent nuclear fuel and an overpack which provides the containment boundary, helium retention boundary, gamma and neutron radiation shielding, and heat rejection capability. The outer diameter of the overpack of the HI-STAR 100 is approximately 203 1/8 inches without impact limiters and approximately 305 7/8 inches with impact limiters. Maximum gross weight for transportation (including overpack, MPC, fuel, and impact limiters) is approximately 282,000 pounds. Specific tolerances are called out in drawings listed below.

Multi-Purpose Canister

There are three Multi-Purpose Canister (MPC) models, designated the MPC-24, MPC-68, and MPC-68F. All MPCs are designed to have identical exterior dimensions. A single overpack design is provided which is capable of containing each type of MPC. The two digits after the MPC designate the number of reactor fuel assemblies for which the respective MPCs are designed. The MPC-24 is designed to contain up to 24 Pressurized Water Reactor (PWR) fuel assemblies and the MPC-68 and MPC-68F are designed to contain up to 68 Boiling Water Reactor (BWR) fuel assemblies. Any MPC-68 loaded with material classified as fuel debris is designated as MPC-68F.

The HI-STAR 100 MPC is a welded cylindrical structure with flat ends. Each MPC is an assembly consisting of a honeycombed fuel basket, baseplate, canister shell, lid, and closure ring. The outer diameter and cylindrical height of each MPC is fixed. However, the number of spent fuel storage locations in each of the MPCs depends on the fuel assembly characteristics. For the HI-STAR 100 System transporting fuel debris in a MPC-68F, the MPC provides the second inner container, in accordance with 10 CFR 71.63. The MPC pressure boundary is a strength-welded enclosure constructed entirely of a stainless steel alloy.

5. a. (2) Description (continued)

Overpack

The HI-STAR 100 overpack is a multi-layer steel cylinder with a welded baseplate and bolted lid (closure plate). The inner shell of the overpack forms an internal cylindrical cavity for housing the MPC. The outer surface of the overpack inner shell is buttressed with intermediate steel shells for radiation shielding. The overpack closure plate incorporates a dual O-ring design to ensure its containment function. The containment system consists of the overpack inner shell, bottom plate, top flange, top closure plate, top closure inner O-ring seal, vent port plug and seal, and drain port plug and seal.

Impact Limiters

The HI-STAR 100 overpack is fitted with two impact limiters fabricated of aluminum honeycomb completely enclosed by an all-welded austenitic stainless steel skin. The two impact limiters are attached to the overpack with 20 and 16 bolts at the top and bottom, respectively.

(3) Drawings

The package shall be constructed and assembled in accordance with the following drawings or figures in Holtec International Report No. HI-951251, *Safety Analysis Report for the Holtec International Storage, Transport, And Repository Cask System (HI-STAR 100 Cask System)*, Revision 9:

(a) HI-STAR 100 MPC-24	Drawing C1395, Sheets 1-4, Rev. 1 Drawing C1396, Sheets 1-4, 6, Rev. 1; and Sheet 5, Rev. 0 Drawing BM-C1478, Sheets 1& 2, Rev. 1
(b) HI-STAR 100 MPC-68 and MPC-68F	Drawing C1401, Sheets 1-4, Rev. 1 Drawing C1402, Sheets 1-4, 6, Rev. 1; and Sheet 5, Rev. 0 Drawing BM-C1479, Sheets 1& 2, Rev. 1
(c) HI-STAR 100 Overpack	Drawing C1397, Sheet 1, Rev. 2; and Sheets 2-7, Rev. 1 Drawing C1398, Sheets 1-3, Rev. 1 Drawing C1399, Sheets 1-2, Rev. 1; and Sheet 3, Rev. 2 Drawing BM-C1476, Sheet 1, Rev. 1; and Sheet 2, Rev. 2
(d) HI-STAR 100 Impact Limiters	Drawing C1765, Sheets 1-6, Rev. 1; and Sheet 7, Rev. 0
(e) HI-STAR 100 Assembly for Transport	Drawing C1782, Rev. 1

5. b. Contents

(1) Type and Form, and Quantity of Material

(a) Fuel assemblies meeting the specifications and quantities provided in Appendix A to this Certificate of Compliance and meeting the requirements provided in Conditions 5.b(1)(b) through 5.b(1)(g) below are authorized for transportation.

(b) The following definitions apply:

Damaged Fuel Assemblies are fuel assemblies with known or suspected cladding defects, as determined by review of records, greater than pinhole leaks or hairline cracks, missing fuel rods that are not replaced with dummy fuel rods, or those that cannot be handled by normal means. Fuel assemblies which cannot be handled by normal means due to fuel cladding damage are considered fuel debris.

Damaged Fuel Containers (DFCs) are specially designed fuel containers for damaged fuel assemblies or fuel debris which permit gaseous and liquid media to escape while minimizing dispersal of gross particulates. The DFC designs authorized for use in the HI-STAR 100 are shown in Figures 1.2.10 and 1.2.11 of Holtec International Report No. HI-951251, Rev. 9.

Fuel Debris is ruptured fuel rods, severed rods, loose fuel pellets, and fuel assemblies with known or suspected defects which cannot be handled by normal means due to fuel cladding damage.

Incore Grid Spacers are fuel assembly grid spacers located within the active fuel region (i.e., not including top and bottom spacers).

Intact Fuel Assemblies are fuel assemblies without known or suspected cladding defects greater than pinhole leaks or hairline cracks and which can be handled by normal means. Partial fuel assemblies, that is fuel assemblies from which fuel rods are missing, shall not be classified as intact fuel assemblies unless dummy fuel rods are used to displace an amount of water greater than or equal to that displaced by the original fuel rod(s).

Minimum Enrichment is the minimum assembly average enrichment. Natural uranium blankets are not considered in determining minimum enrichment.

Planar-Average Initial Enrichment is the average of the distributed fuel rod initial enrichments within a given axial plane of the assembly lattice.

(c) For MPCs partially loaded with stainless steel clad fuel assemblies, all remaining fuel assemblies in the MPC shall meet the more restrictive of the two limits for the stainless steel clad fuel assemblies or the applicable Zircaloy clad fuel assemblies.

(d) For MPCs partially loaded with damaged fuel assemblies or fuel debris, all remaining Zircaloy clad intact fuel assemblies in the MPC shall meet the more restrictive of the two limits for the damaged fuel assemblies or the intact fuel assemblies.

5. b. (1) Type and Form, and Quantity of Material (continued)

(e) For MPC-68s partially loaded with array/class 6x6A, 6x6B, 6x6C, or 8x8A fuel assemblies, all remaining Zircaloy clad intact fuel assemblies in the MPC shall meet the more restrictive of the two limits for the 6x6A, 6x6B, 6x6C, and 8x8A fuel assemblies or the applicable Zircaloy clad fuel assemblies.

(f) PWR control rods, burnable poison rod assemblies, thimble plugs, and other non-fuel hardware are not authorized for transportation.

(g) BWR stainless-steel channels and control blades are not authorized for transportation.

c. Transport Index for Criticality Control

The minimum transport index to be shown on the label for nuclear criticality control: 0

6. For operating controls and procedures, in addition to the requirements of Subpart G of 10 CFR Part 71:

a. Each package shall be both prepared for shipment and operated in accordance with detailed written operating procedures. Procedures for both preparation and operation shall be developed. At a minimum, those procedures shall include the following provisions:

(1) Identification of the fuel to be loaded and independent verification that the fuel meets the specifications of Condition 5.b above.

(2) Before each shipment, the licensee or shipper shall verify and document that each of the requirements of 10 CFR 71.87 has been satisfied.

(3) The package must satisfy the following leak testing requirements:

(a) All overpack containment boundary seals shall be leak tested to show a leak rate of not greater than 4.3×10^{-6} atm cm^3/sec (helium). The leak test shall have a minimum sensitivity of 2.15×10^{-6} atm cm^3/sec (helium) and shall be performed:

(i) before the first shipment;
(ii) within the 12-month period prior to each successive shipment;
(iii) after detensioning one or more overpack lid bolts or the vent port plug; and
(iv) after each seal replacement.

(b) Before each shipment, all containment boundary seals shall be leak tested using a test with a minimum sensitivity of 1×10^{-3} atm cm^3/sec. If leakage is detected on a seal, then the seal must be replaced and leak tested per Condition 6.a(3)(a) above.

(c) Each containment boundary seal must be replaced after each use of the seal.

(4) The rupture discs on the neutron shield vessel shall be replaced every 5 years.

6. a. (continued)

(5) All MPCs shall be leak tested at the time of closure to show a leak rate of no greater than 5×10^{-6} atm cm^3/sec (helium).

(6) Water and residual moisture shall be removed from the MPC in accordance with the following specifications:

(a) The MPC shall be evacuated to a pressure of less than or equal to 3 torr.
(b) The MPC cavity shall hold a stable pressure of less than or equal to 3 torr for at least 30 minutes.

(7) Following vacuum-drying, the MPC shall be backfilled with 99.995% minimum purity helium: $\geq$ 1 atm and $\leq$ 28.3 psig for the MPC-24, and $\geq$ 1 atm and $\leq$ 28.5 psig for the MPC-68 and MPC-68F.

(8) Water and residual moisture shall be removed from the HI-STAR 100 overpack in accordance with the following specifications:

(a) The MPC shall be evacuated to a pressure of less than or equal to 3 torr.

(b) The overpack cavity shall hold a stable pressure of less than or equal to 3 torr for at least 30 minutes.

(9) Following vacuum drying, the overpack shall be backfilled with helium to $\geq$ 10 psig and $\leq$ 14 psig.

(10) The following fasteners shall be tightened to the torque values specified below:

Fastener	Torque (ft-lbs)
Overpack Closure Plate Bolts	2895 $\pm$ 90
Overpack Vent and Drain Port Plugs	45 +5/-0
Top Impact Limiter Attachment Bolts	256 +10/-0
Bottom Impact Limiter Attachment Bolts	1500 +45/-0
Tie-down Bolts	250 +20/-0
Transport Frame Bolts	250 +20/-0

(11) Verify that the appropriate fuel spacers, as necessary, are used to position the fuel in the MPC cavity.

b. All acceptance tests and maintenance shall be performed in accordance with detailed written procedures. Procedures for fabrication, acceptance testing, and maintenance shall be developed and shall include the following provisions:

(1) The overpack lifting trunnions shall be tested at 300% of the maximum design lifting load.

(2) The MPC shall be pressure tested to 125% of the design pressure. The minimum test pressure shall be 125 psig.

6. b. (continued)

(3) The overpack shall be pressure tested to 150% of the Maximum Normal Operating Pressure (MNOP). The minimum test pressure shall be 150 psig.

(4) The MPC lid-to-shell (LTS) weld shall be verified by either volumetric examination using the ultrasonic (UT) method or multi-layer liquid penetrant (PT) examination. The root and final weld layers shall be PT examined in either case. If PT alone is used, additional intermediate PT examination(s) shall be conducted after each approximately 3/8 inch of the weld is completed. The inspection of the weld must be performed by qualified personnel and shall meet the acceptance requirements of ASME B&PV Section III, NB-5350. The inspection process, including findings (indications) shall be made a permanent part of the licensee's records by video, photographic, or other means providing an equivalent retrievable record of weld integrity.

(5) The radial neutron shield shall have a minimum thickness of 4.3 inches and the impact limiter neutron shields shall have a minimum thickness of 2.5 inches. Before first use, the neutron shielding integrity shall be confirmed through a combination of fabrication process control and radiation measurements with either loaded contents or a check source. Measurements shall be performed over the entire exterior surface of the radial neutron shield and each impact limiter using, at a maximum, a 6 x 6 inch test grid.

(6) Periodic verification of the neutron shield integrity shall be performed within 5 years of each shipment. The periodic verification shall be performed by radiation measurements with either loaded contents or a check source. Measurements shall be performed at a minimum of 12 locations on the radial neutron shield and at a minimum of 4 locations on each impact limiter

(7) The first fabricated HI-STAR 100 overpack shall be tested to confirm its heat transfer capability. The test shall be conducted after the radial channels, enclosure shell panels, and neutron shield material have been installed and all inside and outside surfaces are painted per the Design Drawings specified in Section 1.4 of the SAR, Rev. 9. A test cover plate shall be used to seal the overpack cavity. Testing shall be performed in accordance with written and approved procedures. The test must demonstrate that the overpack is fabricated adequately to meet the design heat transfer capability.

(8) For each package, a periodic thermal performance test shall be performed every 5 years or prior to next use, if the package has not been used for transport for greater than 5 years, to demonstrate that the thermal capabilities of the cask remain within its design basis.

(9) The neutron absorber's minimum acceptable ^{10}B loading is 0.0267 g/cm^2 for the MPC-24 and 0.0372 g/cm^2 for the MPC-68, and 0.01 g/cm^2 for the MPC-68F. The ^{10}B loading shall be verified by chemistry or neutron attenuation techniques.

(10) The minimum flux trap size for the MPC-24 is 1.09 inches.

(11) The minimum fuel cell pitch for the MPC-68 and MPC-68F is 6.43 inches.

(12) The package containment verification leak test shall be per ANSI 14.5.

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7. The maximum gross weight of the package as presented for shipment shall not exceed 282,000 pounds.

8. The package shall be located on the transport vehicle such that the bottom surface of the bottom impact limiter is at least 6 feet (along the axis of the overpack) from the edge of the vehicle.

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

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

11. Expiration Date: March 31, 2004

Attachment: Appendix A

REFERENCES:

Holtec International Report No. HI-951251, *Safety Analysis Report for the Holtec International Storage, Transport, And Repository Cask System (HI-STAR 100 Cask System)*, Revision 9, dated April 20, 2000.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

M. Wayne Hodges
for

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

Date: May 11, 2000

APPENDIX A

CERTIFICATE OF COMPLIANCE NO. 9261, REVISION 1

MODEL NO. HI-STAR 100 SYSTEM

INDEX TO APPENDIX A

Page:	Table:	Description:
Page A-1 to A15	Table A1	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-8		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-9		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.

INDEX TO APPENDIX A

Table A.1 (Page 1 of 15)
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:	Zircaloy (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, decay heat and minimum initial enrichment per assembly	
i. Zr clad:	An assembly post-irradiation cooling time, average burnup, decay heat, 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, decay heat, and minimum initial enrichment as specified in Table A.6, as applicable.
d. Fuel assembly length:	$\leq$ 176.8 inches (nominal design)
e. Fuel assembly width:	$\leq$ 8.54 inches (nominal design)
f. Fuel assembly weight:	$\leq$ 1,680 lbs

B. Quantity per MPC: Up to 24 PWR fuel assemblies.

C. Fuel assemblies shall not contain control components.

D. Damaged fuel assemblies and fuel debris are not authorized for loading into the MPC-24.

Table A.1 (Page 2 of 15)
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:	Zircaloy (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, decay heat and minimum initial enrichment per assembly:	
i. Zr clad:	An assembly post-irradiation cooling time, average burnup, decay heat and minimum initial enrichment as specified in Table A.7, except for (1) array/class 6x6A, 6x6C, 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, a decay heat ≤ 183.5 Watts, 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. Fuel assembly length:	≤ 176.2 inches (nominal design)
f. Fuel assembly width:	≤ 5.85 inches (nominal design)
g. Fuel assembly weight	≤ 700 lbs, including channels

Table A.1 (Page 3 of 15)
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:	Zircaloy (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	$\leq$ 400 lbs, including channels

Table A.1 (Page 4 of 15)
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:	Zircaloy (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 15)
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:	Zircaloy (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	$\leq$ 400 lbs, including channels

Table A.1 (Page 6 of 15)
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 Holtec International Report No. HI-951251, Revision 9) and meeting the following specifications:

a. Cladding type:	Zircaloy (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 7 of 15)
Fuel Assembly Limits

II. MPC MODEL: MPC-68 (continued)

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 8 of 15)
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:	Zircaloy (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 9 of 15)
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:	Zircaloy (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	$\leq$ 400 lbs, including channels

Table A.1 (Page 10 of 15)
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:	Zircaloy (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 $\geq$ 18 years, an average burnup $\leq$ 30,000 MWD/MTU, and a minimum initial enrichment $\geq$ 1.8 wt% ^{235}U for 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	$\leq$ 400 lbs, including channels

Table A.1 (Page 11 of 15)
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:	Zircaloy (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 12 of 15)
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:	Zircaloy (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	$\leq$ 400 lbs, including channels

Table A.1 (Page 13 of 15)
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:	Zircaloy (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	$\leq$ 400 lbs, including channels

Table A.1 (Page 14 of 15)
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 Holtec International Report No. HI-951251, Revision 9) and meeting the following specifications:

a.	Cladding Type:	Zircaloy (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 15 of 15)
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.2 (Page 1 of 4)
PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	14x14A	14x14B	14x14C	14x14D	15x15A
Clad Material (Note 2)	Zr	Zr	Zr	SS	Zr
Design Initial U (kg/assy.) (Note 3)	≤ 407	≤ 407	≤ 425	≤ 400	≤ 464
Initial Enrichment (wt % ^{235}U)	≤ 4.6	≤ 4.6	≤ 4.6	≤ 4.0	≤ 4.1
No. of Fuel Rods	179	179	176	180	204
Clad O.D. (in.)	≥ 0.400	≥ 0.417	≥ 0.440	≥ 0.422	≥ 0.418
Clad I.D. (in.)	≤ 0.3514	≤ 0.3734	≤ 0.3880	≤ 0.3890	≤ 0.3660
Pellet Dia. (in.)	≤ 0.3444	≤ 0.3659	≤ 0.3805	≤ 0.3835	≤ 0.3580
Fuel Rod Pitch (in.)	≤ 0.556	≤ 0.556	≤ 0.580	≤ 0.556	≤ 0.550
Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 144	≤ 150
No. of Guide Tubes	17	17	5 (Note 4)	16	21
Guide Tube Thickness (in.)	≥ 0.017	≥ 0.017	≥ 0.038	≥ 0.0145	≥ 0.0165

Table A.2 (Page 2 of 4)
PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	15x15B	15x15C	15x15D	15x15E	15x15F
Clad Material (Note 2)	Zr	Zr	Zr	Zr	Zr
Design Initial U (kg/assy.) (Note 3)	$\leq$ 464	$\leq$ 464	$\leq$ 475	$\leq$ 475	$\leq$ 475
Initial Enrichment (wt % ^{235}U)	$\leq$ 4.1	$\leq$ 4.1	$\leq$ 4.1	$\leq$ 4.1	$\leq$ 4.1
No. of Fuel Rods	204	204	208	208	208
Clad O.D. (in.)	$\geq$ 0.420	$\geq$ 0.417	$\geq$ 0.430	$\geq$ 0.428	$\geq$ 0.428
Clad I.D. (in.)	$\leq$ 0.3736	$\leq$ 0.3640	$\leq$ 0.3800	$\leq$ 0.3790	$\leq$ 0.3820
Pellet Dia. (in.)	$\leq$ 0.3671	$\leq$ 0.3570	$\leq$ 0.3735	$\leq$ 0.3707	$\leq$ 0.3742
Fuel Rod Pitch (in.)	$\leq$ 0.563	$\leq$ 0.563	$\leq$ 0.568	$\leq$ 0.568	$\leq$ 0.568
Active Fuel Length (in.)	$\leq$ 150	$\leq$ 150	$\leq$ 150	$\leq$ 150	$\leq$ 150
No. of Guide Tubes	21	21	17	17	17
Guide Tube Thickness (in.)	$\geq$ 0.015	$\geq$ 0.0165	$\geq$ 0.0150	$\geq$ 0.0140	$\geq$ 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 (wt % ^{235}U)	≤ 4.0	≤ 3.8	≤ 4.6	≤ 4.0	≤ 4.0	≤ 4.0
No. of Fuel Rods	204	208	236	264	264	264
Clad O.D. (in.)	≥ 0.422	≥ 0.414	≥ 0.382	≥ 0.360	≥ 0.372	≥ 0.377
Clad I.D. (in.)	≤ 0.3890	≤ 0.3700	≤ 0.3320	≤ 0.3150	≤ 0.3310	≤ 0.3330
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 Tubes	21	17	5 (Note 4)	25	25	25
Guide 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 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.

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)	$\leq$ 110	$\leq$ 110	$\leq$ 110	$\leq$ 100	$\leq$ 195	$\leq$ 120
Maximum planar-average initial enrichment (wt.% ^{235}U)	$\leq$ 2.7	$\leq$ 2.7 for the UO_2 rods. See Note 4 for MOX rods	$\leq$ 2.7	$\leq$ 2.7	$\leq$ 4.2	$\leq$ 2.7
Initial Maximum Rod Enrichment (wt.% ^{235}U)	$\leq$ 4.0	$\leq$ 4.0	$\leq$ 4.0	$\leq$ 5.5	$\leq$ 5.0	$\leq$ 4.0
No. of Fuel Rods	35 or 36	35 or 36 (up to 9 MOX rods)	36	49	49	63 or 64
Clad O.D. (in.)	$\geq$ 0.5550	$\geq$ 0.5625	$\geq$ 0.5630	$\geq$ 0.4860	$\geq$ 0.5630	$\geq$ 0.4120
Clad I.D. (in.)	$\leq$ 0.5105	$\leq$ 0.4945	$\leq$ 0.4990	$\leq$ 0.4204	$\leq$ 0.4990	$\leq$ 0.3620
Pellet Dia. (in.)	$\leq$ 0.4980	$\leq$ 0.4820	$\leq$ 0.4880	$\leq$ 0.4110	$\leq$ 0.4910	$\leq$ 0.3580
Fuel Rod Pitch (in.)	$\leq$ 0.710	$\leq$ 0.710	$\leq$ 0.740	$\leq$ 0.631	$\leq$ 0.738	$\leq$ 0.523
Active Fuel Length (in.)	$\leq$ 120	$\leq$ 120	$\leq$ 77.5	$\leq$ 80	$\leq$ 150	$\leq$ 120
No. of Water Rods (Note 11)	1 or 0	1 or 0	0	0	0	1 or 0
Water Rod Thickness (in.)	$\geq$ 0	$\geq$ 0	N/A	N/A	N/A	$\geq$ 0
Channel Thickness (in.)	$\leq$ 0.060	$\leq$ 0.060	$\leq$ 0.060	$\leq$ 0.060	$\leq$ 0.120	$\leq$ 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	9x9B
Clad Material (Note 2)	Zr	Zr	Zr	Zr	Zr	Zr	Zr
Design Initial U (kg/assy.) (Note 3)	≤ 185	≤ 185	≤ 185	≤ 185	≤ 185	≤ 177	≤ 177
Maximum planar-average initial enrichment (wt.% ^{235}U)	≤ 4.2	≤ 4.2	≤ 4.2	≤ 4.2	< 3.6	≤ 4.2	≤ 4.2
Initial Maximum Rod Enrichment (wt.% ^{235}U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0
No. of Fuel Rods	63 or 64	62	60 or 61	59	64	74/66 (Note 5)	72
Clad O.D. (in.)	≥ 0.4840	≥ 0.4830	≥ 0.4830	≥ 0.4930	≥ 0.4576	≥ 0.4400	≥ 0.4330
Clad I.D. (in.)	≤ 0.4295	≤ 0.4250	0.4230	≤ 0.4250	≤ 0.3996	≤ 0.3840	≤ 0.3810
Pellet Dia. (in.)	≤ 0.4195	≤ 0.4160	≤ 0.4140	≤ 0.4160	≤ 0.3913	≤ 0.3760	≤ 0.3740
Fuel Rod Pitch (in.)	≤ 0.642	≤ 0.641	≤ 0.640	≤ 0.640	≤ 0.609	≤ 0.566	≤ 0.572
Design Active Fuel Length (in.)	≤ 150	≤ 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	1 (Note 6)
Water Rod Thickness (in.)	≥ 0.034	> 0.00	> 0.00	≥ 0.034	≥ 0.0315	> 0.00	> 0.00
Channel Thickness (in.)	≤ 0.120	≤ 0.120	≤ 0.120	≤ 0.100	≤ 0.055	≤ 0.120	≤ 0.120

Table A.3 (Page 3 of 5)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	**9x9C**	**9x9D**	**9x9E (Note 13)**	**9x9F (Note 13)**	**10x10A**
Clad Material (Note 2)	Zr	Zr	Zr	Zr	Zr
Design Initial U (kg/assy.) (Note 3)	≤ 177	≤ 177	≤ 177	≤ 177	≤ 186
Maximum planar-average initial enrichment (wt.% ^{235}U)	≤ 4.2	≤ 4.2	≤ 4.1	≤ 4.1	≤ 4.2
Initial Maximum Rod Enrichment (wt.% ^{235}U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0
No. of Fuel Rods	80	79	76	76	92/78 (Note 8)
Clad O.D. (in.)	≥ 0.4230	≥ 0.4240	≥ 0.4170	≥ 0.4430	≥ 0.4040
Clad I.D. (in.)	≤ 0.3640	≤ 0.3640	≤ 0.3640	≤ 0.3860	≤ 0.3520
Pellet Dia. (in.)	≤ 0.3565	≤ 0.3565	≤ 0.3530	≤ 0.3745	≤ 0.3455
Fuel Rod Pitch (in.)	≤ 0.572	≤ 0.572	≤ 0.572	≤ 0.572	≤ 0.510
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Water Rods (Note 11)	1	2	5	5	2
Water Rod Thickness (in.)	≥ 0.020	≥ 0.0300	≥ 0.0120	≥ 0.0120	≥ 0.0300
Channel Thickness (in.)	≤ 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	**10x10B**	**10x10C**	**10x10D**	**10x10E**
Clad Material (Note 2)	Zr	Zr	SS	SS
Design Initial U (kg/assy.) (Note 3)	$\leq$ 186	$\leq$ 186	$\leq$ 125	$\leq$ 125
Maximum planar-average initial enrichment (wt.% ^{235}U)	$\leq$ 4.2	$\leq$ 4.2	$\leq$ 4.0	$\leq$ 4.0
Initial Maximum Rod Enrichment (wt.% ^{235}U)	$\leq$ 5.0	$\leq$ 5.0	$\leq$ 5.0	$\leq$ 5
No. of Fuel Rods	91/83 (Note 9)	96	100	96
Clad O.D. (in.)	$\geq$ 0.3957	$\geq$ 0.3780	$\geq$ 0.3960	$\geq$ 0.3940
Clad I.D. (in.)	$\leq$ 0.3480	$\leq$ 0.3294	$\leq$ 0.3560	$\leq$ 0.3500
Pellet Dia. (in.)	$\leq$ 0.3420	$\leq$ 0.3224	$\leq$ 0.3500	$\leq$ 0.3430
Fuel Rod Pitch (in.)	$\leq$ 0.510	$\leq$ 0.488	$\leq$ 0.565	$\leq$ 0.557
Design Active Fuel Length (in.)	$\leq$ 150	$\leq$ 150	$\leq$ 83	$\leq$ 83
No. of Water Rods (Note 11)	1 (Note 6)	5 (Note 10)	0	4
Water Rod Thickness (in.)	> 0.00	$\geq$ 0.031	N/A	$\geq$ 0.022
Channel Thickness (in.)	$\leq$ 0.120	$\leq$ 0.055	$\leq$ 0.080	$\leq$ 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 75 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 PWR FUEL WITH ZIRCALOY CLAD AND
WITH NON-ZIRCALOY IN-CORE GRID SPACERS (Note 1)

Post-irradiation Cooling Time (years)	Assembly Burnup (MWD/MTU)	Assembly Minimum Enrichment (wt. % U-235)	Decay Heat (Watts)
≥ 10	≤ 24,500	≥ 2.3	≤ 411
≥ 12	≤ 29,500	≥ 2.6	≤ 473
≥ 14	≤ 34,500	≥ 2.9	≤ 540
≥ 15	≤ 37,500	≥ 3.2	≤ 579

Note 1: Linear interpolation between points is permitted.

Table A.5

FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND MINIMUM ENRICHMENT
MPC-24 PWR FUEL WITH ZIRCALOY CLAD AND
WITH ZIRCALOY IN-CORE GRID SPACERS (Note 1)

Post-irradiation Cooling Time (years)	Assembly Burnup (MWD/MTU)	Assembly Minimum Enrichment (wt. % U-235)	Decay Heat (Watts)
≥ 7	≤ 24,500	≥ 2.3	≤ 496
≥ 8	≤ 29,500	≥ 2.6	≤ 562
≥ 10	≤ 34,500	≥ 2.9	≤ 610
≥ 12	≤ 39,500	≥ 3.2	≤ 667
≥ 15	≤ 44,100	≥ 3.4	≤ 704

Note 1: Linear interpolation between points is permitted.

Table A.6

FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND MINIMUM ENRICHMENT
MPC-24 PWR FUEL WITH STAINLESS STEEL CLAD (Note 1)

Post-irradiation Cooling Time (years)	**Assembly Burnup (MWD/MTU)**	**Assembly Minimum Enrichment (wt. % U-235)**	**Decay Heat (Watts)**
≥ 19	≤ 30,000	≥ 3.1	≤ 377
≥ 24	≤ 40,000	≥ 3.1	≤ 475

Note 1: Linear interpolation between points is permitted.

Table A.7

FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND MINIMUM ENRICHMENT
MPC-68 (Note 1)

Post-irradiation Cooling Time (years)	**Assembly Burnup (MWD/MTU)**	**Assembly Minimum Enrichment (wt. % U-235)**	**Decay Heat (Watts)**
≥ 8	≤ 24,500	≥ 2.1	≤ 179
≥ 9	≤ 29,500	≥ 2.4	≤ 208
≥ 12	≤ 34,500	≥ 2.6	≤ 222
≥ 15	≤ 39,100	≥ 2.9	≤ 238

Note 1: Linear interpolation between points is permitted.

REFERENCE:

Holtec International Report No. HI-951251, *Safety Analysis Report for the Holtec International Storage, Transport, And Repository Cask System (HI-STAR 100 Cask System)*, Revision 9, dated April 20, 2000.

NRC FORM 618
(3-96)
10 CFR 71

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIALS PACKAGES

U.S. NUCLEAR REGULATORY COMMISSION

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. PACKAGE IDENTIFICATION NUMBER	d. PAGE NUMBER	e. TOTAL NUMBER PAGES
9263	3	USA/9263/B(U)-85	1	2

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

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 22, 1999, as supplemented.

c. DOCKET NUMBER 71-9263

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.

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 71.12.

10. Expiration date: June 30, 2005.

<u>REFERENCES</u>

Source Production and Equipment Company, Inc., application dated April 22, 1999.

Supplements dated: May 6, 1999; March 22, June 6, and June 19, 2000.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Susan F. Shankman for

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

Date: June 23, 2000

NRC FORM 618
(3-96)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIALS PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. PACKAGE IDENTIFICATION NUMBER	d. PAGE NUMBER	e. TOTAL NUMBER PAGES
9269	2	USA/9269/B(U)-85	1	2

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

AEA Technology/QSA 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.

c. DOCKET NUMBER 71-9269

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 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 wide, 13.25 inches high and 8.25 inches long. 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. F, Sheets 1-4. Packaging constructed in accordance with Amersham Corporation Drawing No. R65006, Rev. C, Sheets 1-4 may be shipped until June 30, 2000.

(b) Contents

(1) Type and form of material

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

Page 2 - Certificate No. 9269 - Revision No. 2 - Docket No. 71-9269

(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. 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, as supplemented, 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. The Model No. 650L may be marked with Package Identification Number USA/9269B(U) until June 30, 2000.

11. Expiration date: November 30, 2000.

REFERENCES

AEA Technology/QSA Inc. application dated July 23, 1999.

Supplement dated November 19, 1999.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

E. William Brach

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

Date: 2/7/00

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
9272	5	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

Combustion Engineering, Inc. application dated July 30, 1996, as supplemented.

4. CONDITIONS

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

5.

(a) Packaging

(1) Model No.: 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.

(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;

5.(b) Contents (iii) Con't.

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

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, 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
9272	5	71-9272	USA/9272/AF-85	4	OF	4

REFERENCES

Combustion Engineering, Inc. application dated July 30, 1996.

Supplements dated: December 12, 1996; January 9, 1997; January 11, and March 19, 1999.

ABB Combustion Engineering Nuclear Power, Inc. supplement dated June 10, 1999.

ABB C-E Nuclear Power, Inc. supplements dated: March 28, and April 4 and 12, 2000.

CE Nuclear Power, LLC supplement dated: September 14, 2000.

Westinghouse Electric Company LLC supplement dated: September 18, 2000.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

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

Date October 10, 2000

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
9274	5	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

Combustion Engineering, Inc. application dated April 9, 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.: 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 head and 12-gage closure ring with drop forged lugs and a 5/8-inch diameter bolt. 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 ABB Combustion Engineering Nuclear Systems Drawing No. L-9274-01, Rev. 0.

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9274	5	71-9274	USA/9274/AF	2	OF	3

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

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: July 31, 2002.

REFERENCES

Combustion Engineering, Inc. application dated April 9, 1997.

Supplement dated: May 20, 1997.

ABB Combustion Engineering Nuclear Power, Inc. supplements dated: June 10 and 21, and July 6, 1999.

ABB C-E Nuclear Power, Inc. supplements dated: March 28, and April 4 and 12, 2000.

CE Nuclear Power, LLC supplement dated: September 14, 2000.

Westinghouse Electric Company LLC supplement dated: September 18, 2000.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

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

Date: October 10, 2000

NRC FORM 618
(3-96)
10 CFR 71

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIALS PACKAGES

U.S. NUCLEAR REGULATORY COMMISSION

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. PACKAGE IDENTIFICATION NUMBER	d. PAGE NUMBER	e. TOTAL NUMBER PAGES
9277	0	USA/9277/B( )F	1	5

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

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

c. DOCKET NUMBER 71-9277

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.

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Configurations F and G are used to ship 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 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.

Configurations 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 Atomic Drawings Nos. GADR 55-2-1, Issue C; GADR 55-2-2, Issue A; and GADR 55-2-3, Issue B.

Configurations 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 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, nonfissile, 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.

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5. (c) Transport Index for Criticality Control

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 (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 package 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.

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

12. Effective date: July 29, 1997 Expiration date: May 31, 2001.

REFERENCES

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.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Charles J. Haughney

for William F. Kane, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: July 29, 1997

NRC FORM 618
(3-96)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIALS PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. PACKAGE IDENTIFICATION NUMBER	d. PAGE NUMBER	e. TOTAL NUMBER PAGES
9280	1	USA/9280/AF-85	1	2

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

Babcock & Wilcox Company
P. O. Box 785
Lynchburg, VA 24505-0785

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION:

BWX Technologies, Inc. application dated August 20, 1997, as supplemented.

c. DOCKET NUMBER 71-9280

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.

(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.

5.(c) Transport Index for Criticality Control

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: December 31, 2002.

REFERENCES

BWX Technologies, Inc. application dated August 20, 1997.

Supplements dated: October 6 and December 8, 1997; and February 6, 1998.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Cass R. Chappell

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

Date February 10, 1998

NRC FORM 618
(3-96)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIALS PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. PACKAGE IDENTIFICATION NUMBER	d. PAGE NUMBER	e. TOTAL NUMBER PAGES
9281	2	USA/9281/AF-85	1	2

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

BWX Technologies, Inc.
Naval Nuclear Fuel Division
P. O. Box 785
Lynchburg, VA 24505-0785

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION:

BWX Technologies, Inc., application dated December 9, 1997, as supplemented.

c. DOCKET NUMBER 71-9281

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 $1x10^{-3}$ A_2 per package.

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5.(c) Transport Index for Criticality Control

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: May 31, 2003.

REFERENCES

BWX Technologies, Inc., application dated December 9, 1997.
Supplement dated: March 6, 2000.

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: May 12, 2000

NRC FORM 618
(3-96)
10 CFR 71

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIALS PACKAGES

U.S. NUCLEAR REGULATORY COMMISSION

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. PACKAGE IDENTIFICATION NUMBER	d. PAGE NUMBER	e. TOTAL NUMBER PAGES
9282	0	USA/9282/B(U)-85	1	2

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

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

c. DOCKET NUMBER 71-9282

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.

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

10. Expiration date: April 30, 2005.

REFERENCES

Source Production and Equipment Company, Inc., application dated June 28, 1999.

Supplements dated: October 6, November 4, November 22, and December 15, 1999; and February 29 and March 27, 2000.

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: April 10, 2000

NRC FORM 618
(3-96)
10 CFR 71

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIALS PACKAGES

U.S. NUCLEAR REGULATORY COMMISSION

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. PACKAGE IDENTIFICATION NUMBER	d. PAGE NUMBER	e. TOTAL NUMBER PAGES
9283	0	USA/9283/B(U)-85	1	3

'REAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

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

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION:

AEA Technology/QSA Inc. application dated May 21, 1998, as supplemented.

c. DOCKET NUMBER 71-9283

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.: 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.

(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 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 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 §71.12.

10. Expiration date: June 30, 2003.

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REFERENCES

AEA Technology QSA, Inc., application dated May 21, 1998.

Supplement dated: June 15, 1998

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Cass R. Chappell

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

Date: 6/19/98

NRC FORM 618
(3-96)
10 CFR 71

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIALS PACKAGES

U.S. NUCLEAR REGULATORY COMMISSION

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. PACKAGE IDENTIFICATION NUMBER	d. PAGE NUMBER	e. TOTAL NUMBER PAGES
9284	1	USA/9284/B(U)F-85	1	3

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

Eco-Pak Specialty Packaging (ESP)
(A division of Columbiana Boiler Co.)
200 West Railroad Street
P.O. Box 68
Columbiana, OH 44408

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION:

Eco-Pak Specialty Packaging application dated June 19, 1998, as supplemented.

c. DOCKET NUMBER 71-9284

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: 4.4×10^5 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.

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.

Page 3 - Certificate No. 9284 - Revision No. 1 - Docket No. 71-9284

5. (c) Transport Index for Criticality Control

Minimum transport index to be shown on label for nuclear criticality control: 5.0

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 Operating Procedures in Chapter 7 of the application.

   (b) 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 1×10^{-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 71.12.

11. Expiration date: May 31, 2005.

REFERENCES

ESP application dated June 19, 1998.

Supplements dated: August 27, 1999, and March 22, May 12, and May 18, 2000.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Susan F. Shankman for

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

Date June 22, 2000

NRC FORM 618
(3-96)
10 CFR 71

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIALS PACKAGES

U.S. NUCLEAR REGULATORY COMMISSION

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. PACKAGE IDENTIFICATION NUMBER	d. PAGE NUMBER	e. TOTAL NUMBER PAGES
9285	1	USA/9285/AF-85	1	2

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

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.

c. DOCKET NUMBER 71-9285

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.

(c) Transport Index for Criticality Control

Minimum transport index to be shown on label for nuclear criticality control: 0.6

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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 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, 2003.

REFERENCES

General Electric Company application dated August 4, 1998.

Supplements dated: October 2, 1998; and October 14, 1999.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

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

Date: May 10, 2000

NRC FORM 618
(3-96)
10 CFR 71

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIALS PACKAGES

U.S. NUCLEAR REGULATORY COMMISSION

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. PACKAGE IDENTIFICATION NUMBER	d. PAGE NUMBER	e. TOTAL NUMBER PAGES
9287	0	USA/9287/B(U)-85	1	3

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

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

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION:

Packaging Technology, Inc., application dated November 18, 1998, as supplemented.

c. DOCKET NUMBER 71-9287

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.

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.

5(a)(2) cont.

The approximate dimensions 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.

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.

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

8. Expiration date: December 31, 2004.

REFERENCES

Packaging Technology, Inc., application dated November 18, 1998.

Supplement dated: August 20, 1999.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

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

Date: 2/2/00

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
9288	2	71-9288	USA/9288/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)*

Eco-Pak Specialty Packaging (ESP)
200 West Railroad Street
P.O. Box 68
Columbiana, OH 44408

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Eco-Pak Specialty Packaging application dated August 9, 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.: Eco-Pak OP Uranium Oxide Transport Unit Package

(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 cylinders in which storage vessels are inserted.

The outer shell of the package is constructed of 11 gauge mild steel and the space between the outer shell and the internal cylinders are filled with fire-retardant, closed cell, phenolic foam.

The internal cylinders are constructed of 11 gauge mild steel with an inner diameter of 10 1/4-inches. The cylinders are closed with twelve 1/2-inch bolts on a 1/2-inch thick 14-inch diameter inner silicone-gasketed carbon steel lid and a 1/2-inch thick 16 5/8-inch diameter outer silicone-gasketed carbon steel lid with the same number of bolts.

The inner storage vessel is a double walled container constructed of A-240, series 300, stainless steel with the annulus filled with fire retardant, closed-cell phenolic foam. The diameter of the outer annulus is 10-inches. The storage vessel has an inner diameter of either 7-inches or 7 3/4-inches. The storage vessel is closed by twelve 1/2-inch diameter bolts on a 5/8-inch thick stainless steel lid with a silicone gasket.

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

Containment vessel inside diameter	10 1/4-inches
Storage vessel outside diameter	10-inches
Storage vessel inside diameter	7-inches or 7 3/4-inches
Overall package dimensions	
width	45-inches
length	45-inches
height	62-inches
Maximum contents weight (UO_2)	1423 pounds
OP Transport unit	1648 pounds
OP Storage vessels (4)	686 pounds
Maximum package weight (including contents)	3757 pounds

(3) Drawings

The packaging is constructed and assembled in accordance with ESP Drawing Nos.:

OP-TU-SAR, Rev. 6, Sheet 1, OP-TU-SAR, Rev. 3, Sheets 2-4

5.(b) Contents

(1) Type and form of material

Uranium oxide pellets and powder. The contents may include up to 1000 grams of polyethylene per storage vessel (4000 grams per package), or other plastics, provided that the total water equivalent of the plastic is less than 1307 grams per storage vessel (5228 grams per package). In addition, the contents are limited to:

A. Uranium oxide powder enriched to no more than 4.5 weight percent in the U-235 isotope with a maximum of 356 pounds per 7-3/4-inch diameter storage vessel (shown in ESP Drawing No. OP-TU-SAR, Sheet 4, Rev. No. 3) and a maximum load of 1423 pounds per package .

B. Uranium oxide powder enriched to no more than 5.0 weight percent in the U-235 isotope with a maximum load of 356 pounds per 7-inch diameter storage vessel (shown in ESP Drawing No. OP-TU-SAR, Sheet 3, Rev. No. 3) and a maximum load of 1423 pounds per package.

C. 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 a maximum load of 356 pounds per 7-inch diameter oxide vessel (shown in ESP Drawing No. OP-TU-SAR, Sheet 3, Rev. No. 3) and a maximum load of 1423 pounds per package .

5.(b)(1)D. continued

Uranium-bearing materials 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 hydrites (e.g., UH) may not be shipped. Uranium-bearing materials may include oxides, carbides, silicates or other compounds of uranium. Pellets or previously pelletized materials are not allowed for transport in the 7-3/4-inch diameter storage vessel. Uranium-bearing materials may be moderated by graphite to any degree. Compounds may be mixed with other non-fissile materials with the exception of oils, deuterium, tritium and beryllium. Materials with a hydrogen density greater than water must be excluded. Uranium enriched to 5.0 weight percent in the U-235 isotope with a maximum load of 356 pounds per storage vessel and a maximum load of 1423 pounds per package can only be loaded into the 7-inch diameter oxide vessel shown in ESP Drawing No. OP-TU-SAR, Sheet 3, Rev. No. 2. Uranium enriched to 4.5 weight percent in the U-235 isotope with a maximum load of 356 pounds per storage vessel and a maximum load of 1423 pounds per package can only be loaded into the 7-3/4-inch diameter oxide vessel shown in ESP Drawing No. OP-TU-SAR, Sheet 4, Rev. No. 3.

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

1423 pounds of UO_2 material.

5. (c) Transport Index for Criticality Control

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

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
9288	2	71-9288	USA/9288/AF-85	4 OF 4

8. Expiration date: March 31, 2005.

REFERENCES

ESP application dated August 9, 1998.

Supplements dated: September 8 and October 29, 1999; and January 12, February 9, March 2, March 17, June 20, June 29, July 12, and July 18, 2000.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

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

Date August 21, 2000

NRC FORM 618
(3-96)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIALS PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. PACKAGE IDENTIFICATION NUMBER	d. PAGE NUMBER	e. TOTAL NUMBER PAGES
9289	0	USA/9289/B(U)F-85	1	3

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

Framatome Cogema Fuels
Route 726, Mt. Athos Road
P.O. Box 11646
Lynchburg, VA 24504

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION:

Framatome Cogema Fuels application dated January 8, 1999, as supplemented.

c. DOCKET NUMBER 71-9289

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 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 fuel assembly is surrounded by thermal insulation and secured inside the inner container with nine integral clamp frames. 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. 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

5. (a) Packaging (continued)

(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

(b) Contents

(1) Type and form of material

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

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5. (b) Contents (continued)

(2) Maximum quantity of material per package

One fuel assembly weighing no more than 9,090 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	4.4×10^5 MeV-Becquerel per kilogram of uranium
Np and Pu	35 Becquerels per gram of uranium

(c) Transport Index for Criticality Control

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 29, 2004.

REFERENCES

Framatome Cogema Fuels application dated: January 8, 1999.

Framatome Cogema Fuels supplements dated: February 9, 11, and 25, 1999.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Cass R. Chappell

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

Date: February 26, 1999

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
9292	2	71-9292	USA/9292/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

Combustion Engineering, Inc. application dated August 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.: 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.

The metal inner container is approximately 11-1/4 inches high by 18-1/8 inches wide by 179-3/4 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 31-3/4 inches high by 207-1/2 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 2,988 pounds.

(3) Drawings

The packaging is constructed and assembled in accordance with ABB CE Nuclear Power, Inc. Drawing Nos.

L-9292-01, Sheets 1 and 2, Rev. 2, and
L-9292-02, Rev. 3.

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.

(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 2.4 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.

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

Two fuel assemblies. The total weight of contents not to exceed 1,320 pounds.

(c) Transport Index for Criticality Control

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

10. Expiration date: January 31, 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
9292	2	71-9292	USA/9292/AF-85	4 OF 4

REFERENCES

ABB C-E Nuclear Power, Inc. application dated August 23, 1999.

Supplements dated: October 19 and December 10, 1999; and January 10, March 28, and April 4 and 12, 2000.

CE Nuclear Power, LLC supplement dated: September 14, 2000.

Westinghouse Electric Company LLC supplement dated: September 18, 2000.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

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

Date October 10, 2000

NRC FORM 618
(3-96)
10 CFR 71

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIALS PACKAGES

U.S. NUCLEAR REGULATORY COMMISSION

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. PACKAGE IDENTIFICATION NUMBER	d. PAGE NUMBER	e. TOTAL NUMBER PAGES
9511	2	USA/9511/B(U)	1	3

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

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

c. DOCKET NUMBER 71-9511

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)

(b) Contents

(1) Type and form of 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.

(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, 2002

REFERENCES

U. S. Department of Energy application dated February 26, 1991.

Supplements dated February 28, 1992, May 6, 1994, August 26, 1994, and July 29, 1997.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

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

Date: July 30, 1997

NRC FORM 618
(3-96)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIALS PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. PACKAGE IDENTIFICATION NUMBER	d. PAGE NUMBER	e. TOTAL NUMBER PAGES
9781	7	USA/9781/B( )F	1	3

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

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-160 Shipping Container dated October 18, 1968, as supplemented.

c. DOCKET NUMBER 71-9781

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-160

(2) Description

The packaging is a right circular cylinder, 79 inches in diameter by 199 inches in overall height. The packaging outer shell consists of 84 evenly spaced vertical fins 151.5-inches long, attached to a 1.5-inch thick wall (fabricated from carbon steel and clad with stainless steel on the outer surface). The inner shell, the containment vessel, is 1-inch thick (having a 1/8-inch thick rollbonded stainless steel cladding) whose base is 7-inches thick. The 9 7/16-inch annulus between the outer and inner shells is filled with lead. The top of the container is covered with a rotatable closure head fabricated of stainless steel 15-inches thick which is bolted to the container and seals the containment vessel. An oblong access plug in the cover allows for individual spent fuel cell loading or unloading.

The containment vessel has an inside diameter of 55 inches. The central region contains a secondary heat exchanger which is supported by the closure head. (This heat exchanger is not used during shipment.) An inner backup cylinder, 21 inches in diameter, occupies the central region of the containment vessel. The annulus between the backup cylinder and the containment vessel shell provides a space 17-inches wide by 160-inches high for spent fuel. The spent fuel is contained in the annulus by aluminum module holders designed for the particular spent fuel to be shipped.

The packaging has external penetrations to the containment vessel for a steam and water vent line, which is capped during shipment. Shipments are by rail. The packaging is cradled in a support which permits the packaging to be nearly horizontal during shipment. The maximum loaded shipping weight is 237,000 pounds.

Page 2 - Certificate No. 9781 - Revision No. 7 - Docket No. 71-9781

5.(a) Packaging (continued)

(3) Drawings

The packaging is constructed in accordance with the description and Drawing Nos. contained in the Bettis Atomic Power Laboratory Safety Analysis Reports (WAPD-OP(R)C-243, WAPD-OP(R)C-558, and WAPD-OP(R)C-621) dated May 1973, October 1, 1976, and March 1977.

5.(b) Contents

(1) Type and form of material

Irradiated fuel assemblies and blanket modules of the following type:

(i) PWR Core 2 Seed 1 fuel assembly.
(ii) PWR Core 2 Seed 2 fuel assembly.
(iii) PWR Core 2 Blanket fuel assembly.
(iv) S5G Fuel Module, rodded or unrodded.
(v) S5G Center Cell.

All shipments shall be made dry and shall use one holddown device per PWR module. Each PWR Core 2 Seed 1 or Seed 2 fuel assembly shall contain a poison rod or a control rod.

(2) Maximum quantity of material per package

(i) 12 fuel assemblies as described in 5(b)(1)(i) or 11 fuel assemblies and 1 specific blanket fuel assembly, Serial No. G2A-W01-67. Shipment shall not be made prior to 1,614 days after last power operation of the fuel and shall not exceed 12,846 Btu/hr of decay heat per shipment.

(ii) 12 fuel assemblies as described in 5(b)(1)(ii) which shall not exceed 1,100 Btu/hr per fuel assembly of decay heat or 13,200 Btu/hr per shipment.

(iii) 12 blanket fuel assemblies as described in 5(b)(1)(iii) which shall not exceed 21,300 Btu/hr of decay heat per shipment. Shipment shall not be made prior to 1,123 days after last power operation of the fuel.

(iv) 8 fuel assemblies as described in 5(b)(1)(ii) and 4 specific blanket fuel assemblies, Serial Numbers G2A-F01-26B, G2A-F01-02, G2A-F01-10, and G2A-W01-73, which shall not exceed 12,016 Btu/hr of decay heat per shipment. Shipment shall not be made prior to 1,487 days after last power operation of the fuel, with the four blanket fuel assemblies located adjacent to each other in a clockwise or counter-clockwise direction as specified by the serial numbers previously stated.

(v) 4 fuel assemblies as described in 5(b)(1)(iv) or 3 fuel assemblies and 1 center cell as described in 5(b)(1)(v). Shipment shall not be made prior to 168 days after last power operation of the fuel rod and shall not exceed 12,800 Btu/hr of decay heat per shipment.

5.(c) Transport Index for Criticality Control

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

6. Expiration date: January 31, 2003.

REFERENCES

Safety Analysis Report for M-160 Shipping Container: Core Independent Analyses SRSD-106 dated October 18, 1968, as transmitted by Naval Reactors letter G#2097 dated June 3, 1969, as supplemented by Knolls Atomic Power Laboratory letter ONP-74520-414 dated November 26, 1969; Naval Reactors letter G#3742 dated May 15, 1973; Bettis Atomic Power Laboratory letters WAPD-OP(R)C-284 dated August 23, 1973, and WAPD-OP(R)C-297 dated October 8, 1973; Naval Reactor letters G#5582 dated December 17, 1976; and G#5671 dated April 15, G#5702 dated May 23, G#5792 dated September 22, G#5793 dated September 29, and G#5872 dated December 20, 1977; G#5897 dated January 11, 1978; G#6658 dated April 14, 1980; G#92-03424 dated March 20, 1992, and G#97-03421 dated February 6, 1997.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Charles J. Haughney

Charles J. Haughney, Acting Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date February 2, 1998

NRC FORM 618
(3-96)
10 CFR 71

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIALS PACKAGES

U.S. NUCLEAR REGULATORY COMMISSION

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. PACKAGE IDENTIFICATION NUMBER	d. PAGE NUMBER	e. TOTAL NUMBER PAGES
9786	4	USA/9786/B(U)	1	4

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

U.S. Department of Energy
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

c. DOCKET NUMBER 71-9786

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

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(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. Expiration date: August 31, 2001.

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#96-03335 dated February 16, 1996.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

William Travers

William D. Travers, Director
Spent Fuel Project Office
Office of Nuclear Material
Safety and Safeguards

Date: 3/21/96

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIALS PACKAGES

U.S. NUCLEAR REGULATORY COMMISSION

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. PACKAGE IDENTIFICATION NUMBER	d. PAGE NUMBER	e. TOTAL NUMBER PAGES
9787	4	USA/9787/B(U)F	1	3

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

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

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION:

A1W-3 Power Unit Shipping Container Safety Analysis Report for Packaging dated August 1980, as supplemented.

c. DOCKET NUMBER 71-9787

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.: A1W-3 Power Unit Shipping Container (PUSC)

(2) Description

The package is a right circular cylindrical steel weldment. A module support device, a puncture protection cover, and an energy absorber are attached to the top end of the package. The module support device is secured to the power unit adapter flange with two 3-inch diameter shipping studs and is bolted to the upper clamp and cradle assembly with forty 2.25-inch diameter assembly studs. Module support cylinders (with locking nuts) and control rod holddown devices prevent fuel module or control rod motion. The top puncture protection cover, with 4-inch thick steel side walls and a 5-inch thick steel top plate, fits down over the eggcrate assembly and attaches to the bottom plate of the model support device with forty 2-inch diameter bolts. The top energy absorber, with an 80-inch outer diameter, a 54-inch inner diameter, and a height of 25 inches is welded to the top puncture protection cover.

At the bottom end of the A1W-3 PUSC, a 4-inch thick steel puncture protection plate, with a diameter of 50.5 inches, is attached to the bottom of the power unit thermal shield by eight sets of U-bolts. A bottom energy absorber, with an 89.25-inch outer diameter, a 52.5-inch inner diameter, and a height of 36 inches covers the bottom puncture protection plate and is attached to the thermal shield of the power unit with twelve 0.875-inch diameter bolts. The support barrel, which is part of the power unit assembly, and a stainless steel plug/band assembly, which extends around the outside of the support barrel, provide puncture protection for the sides of the power unit.

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

For shipment, the loaded A1W-3 PUSC is secured horizontally in a shipping pedestal. The shipping pedestal consists of a base on which two support beams are mounted horizontally with rubber shock mounts. The base is bolted to the deck of a 300-ton railcar and to four stop blocks which are welded to the railcar deck. The A1W-3 PUSC is secured to the shipping pedestal using the upper clamp and cradle assembly and the center and lower yoke/saddle assemblies. The yokes span across the upper side of the A1W-3 PUSC and are attached to the support beams while the saddles suspend from the yokes and support the weight of the loaded A1W-3 PUSC. A 0.25-inch thick railcar cover is used to enclose the entire A1W-3 PUSC for shipment.

The loaded A1W-3 PUSC (excluding the shipping pedestal) is 349 inches long, has a maximum diameter of 134 inches, and weighs approximately 297,000 pounds.

(3) Drawings

The package is constructed in accordance with the drawings, figures, and sketches included in the application documents (see References, below).

(b) Contents

One unirradiated A1W-3 power unit assembly.

(c) Transport Index for Criticality Control

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

6. Control rods and control rod holddown devices must be installed in the power unit as described in the application.

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

8. Expiration date: January 31, 2005

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REFERENCES

A1W-3 Power Unit Shipping Container Safety Analysis Report for Packaging, and WAPD-REO(c)-118, dated August 1980.

Supplements: A1W-3 Power Unit Shipping Container Modified Top Energy Absorber Revised Safety Analysis Report for Packaging, Attachment 1 to WAPD-REO(c)-118, dated February 1985, Naval Reactors Memorandum G#94-03572 dated November 4, 1994, and November 10, 1999.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

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

Date: 1/19/00

NRC FORM 618
(3-96)
10 CFR 71

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIALS PACKAGES

U.S. NUCLEAR REGULATORY COMMISSION

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. PACKAGE IDENTIFICATION NUMBER	d. PAGE NUMBER	e. TOTAL NUMBER PAGES
9788	12	USA/9788/B(U)-85	1	3

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

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.

c. DOCKET NUMBER 71-9788

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 and S6G Reactor Compartment

(2) Description

The package consists of a deactivated and defueled S5W or S6G 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.

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 S6G Reactor Compartment package is approximately 43 feet long and approximately cylindrical with a maximum diameter of approximately 33 feet. The containment bulkheads are made of HS steel. The bulkheads may 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.

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5.(a) (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 S6G reactor vessel complex, plant piping, ion exchanger resin, purification filter media which may be solidified (S6G only), residual liquid and other miscellaneous components and materials contaminated with radioactive corrosion products (crud).

Ion exchanger resins with up to 3.1 curies of Co-60 may be shipped in the S5W package. Ion exchanger resins and/or purification filter media with up to a combined total of 4.5 curies of Co-60 may be shipped in the S6G package.

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 an S5W package, and not more than 1,200 gallons of residual liquids remain in an S6G 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 S6G package 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. Ion exchanger resin and purification media with up to a combined total of 4.5 curies (1.6×10^{11} becquerels) of Co-60 may be shipped in the S6G package provided ion exchanger resin does not exceed 3.5 curies (1.3×10^{11} becquerels) of Co-60 and purification media does not exceed 3.3 curies (1.2×10^{11} becquerels) of Co-60. Shipment of the S6G packages shall not occur before 365 days after final reactor shutdown.

10. 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.

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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.

(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.

12. Expiration date: September 30, 2003.

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; and
Naval Reactors Memorandum Z#C98-00021 dated February 27, 1998.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Cass R. Chappell

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

Date: 09/25/98

NRC FORM 618
(3-96)
10 CFR 71

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIALS PACKAGES

U.S. NUCLEAR REGULATORY COMMISSION

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. PACKAGE IDENTIFICATION NUMBER	d. PAGE NUMBER	e. TOTAL NUMBER PAGES
9791	5	USA/9791/B(U)-85	1	4

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

U.S. Department of Energy
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

c. DOCKET NUMBER 71-9791

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 the S3G prototype reactor vessel assembly or 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.

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

The inner disposal container (liner) is one of the following designs:

(I) For the S3G prototype reactor vessel assembly, the inner burial container consists of two cylinders constructed of HY-80 steel. The two cylinders are connected by a transition ring. The transition ring, which is welded to the two cylinders, provides radial support for the inner container. The maximum outer diameter of the cylinders is approximately 118 inches at the upper flange. The overall length of the inner container is 174.5 inches. The container wall is 3 inches thick in the upper cylinder and 4 inches thick in the lower cylinder. The bottom plate is 5 inches thick and attached to the container by 12, 2.5-inch thick steel gussets. The cover plate is 5 inches thick and is attached to the container by a 3-inch thick closure weld. The container is axially positioned within the outer container by aluminum honeycomb energy absorbers.

(ii) For the D1G prototype pressure vessel assembly, the inner burial container 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.

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(b) Contents

(1) Type and form of material

(i) An irradiated S3G prototype reactor vessel assembly, including reactor vessel, core basket, thermal shield, closure head, closure mechanism, and three Materials Irradiation Facilities assemblies. In addition, the contents may include surface contamination in the form of activated corrosion products and approximately 19 gallons of residual water.

(ii) 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

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

One irradiated S3G prototype reactor vessel assembly. Surface contamination not to exceed 0.97 curies. The activated displaced metal not to exceed 0.2 curies. The irradiated components not to exceed 113,000 curies.

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

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, 2002.

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, and G#98-10801 dated May 5, 1998.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Cass R. Chappell

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

Date: July 30, 1998

NRC FORM 618
(3-96)
10 CFR 71

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIALS PACKAGES

U.S. NUCLEAR REGULATORY COMMISSION

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. PACKAGE IDENTIFICATION NUMBER	d. PAGE NUMBER	e. TOTAL NUMBER PAGES
9792	5	USA/9792/B(U)	1	3

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

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.

c. DOCKET NUMBER 71-9792

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/S1C 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. The S1C Pressure Vessel Assembly (PVA) is secured in place inside the container by a steel preload plate that 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.

The shipping container is transported with its axis horizontal and is supported by a shipping skid. The loaded container weighs up to 185 tons.

(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.

5. (b) Contents

(1) 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.

(2) One S1C PVA containing up to one core's worth of S1C core support assemblies, S1C main closure hardware, non-irradiated shipping hardware and not more than 70 gallons of residual water. Except for the shipping hardware, the components are irradiated and the surfaces are contaminated with activated corrosion products.

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.

(d) The S1C PVA shipment shall be made no earlier that 1460 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.

Page 3 - Certificate No. 9792 - Revision No. 5 - Docket No. 71-9792

8. Expiration Date: September 30, 2002.

REFERENCES

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 and G#C96-03576, dated November 1, 1996.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Cass R. Chappell

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

Date: January 16, 1998

NRC FORM 618
(3-96)
10 CFR 71

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIALS PACKAGES

U.S. NUCLEAR REGULATORY COMMISSION

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. PACKAGE IDENTIFICATION NUMBER	d. PAGE NUMBER	e. TOTAL NUMBER PAGES
9793	9	USA/9793/B(U)F	1	7

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

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

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION:

"Core Independent M-140 Safety Analysis Report for Packaging" transmitted February 27, 1991, as supplemented.

c. DOCKET NUMBER 71-9793

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	146 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 bolted to, the rail car mounting ring during transport. The cask bottom is equipped with an energy absorber which is composed of five concentric stainless steel rings varying in thickness and height.

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

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 the full-core D1G-2, two additional penetrations were added to 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 internals assembly or an S3G-3 internals assembly or a full D1G-2 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.

(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.

5.(b) Contents (continued)

(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 (SARP) or by use of a shielding hold time from 8(b) below, whichever hold time is greater.

(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 SARP or an administrative hold 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.

(c) Transport Index for Criticality Control

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

Spent fuel module	TI
S3G-3	100
S8G	100
D1G Core 2 (with standard internals)	100
D1G Core 2 (with D1G-2 internals)	0
D2W	100
A1G	0
S6W	100

6. For S3G-3 spent fuel shipments:

(a) Only a full load is authorized. A minimum of 12 fuel modules must have either control rods or poison shipping rods. All rodded and unrodded modules must be positioned as specified on page 6-11 (Rev. 1) of "S3G-3 Recoverable Irradiated Fuel in the M-140 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.

(e) All cells must have top and bottom energy absorbers.

(f) The weight of the spent fuel modules must be limited as specified on page 1-23 (Rev. 2) of "S3G-3 Recoverable Irradiated Fuel in the M-140 Safety Analysis Report for Packaging."

(g) S3G-3 internals assembly must be used for shipment of S3G-3 spent fuel modules.

7. For S8G spent fuel shipments:

(a) Only a full load is authorized. Full and partial fuel modules may be shipped in any combination. All full and partial fuel modules must have control rods.

(b) Minimum fuel cooling time is 248 days after shutdown for prototype modules and 157 days after shutdown for shipboard modules.

(c) All fuel modules must have lower supports and grapple adapters.

(d) Standard internals assembly must be used for shipment of S8G fuel modules. Full fuel modules must have two full (side) spacers; partial fuel modules must have two full (side) spacers and one partial (inside) spacer.

(e) The weight of the fuel modules must be limited as specified on page 1.23 (Rev. 4) of "S8G Recoverable Irradiated Fuel in the M-140 Safety Analysis Report For Packaging."

8. For D1G Core 2 spent fuel shipments:

(a) Either a full core or up to eight fuel modules may be shipped per package. Fuel modules of different types may be shipped in any combination.

(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 SARP.

(c) All normally rodded fuel modules must have control rods. Control rod hold-down devices must be installed on rodded modules.

(d) Rodded modules must have top and bottom energy absorbers. Unrodded modules must have top energy absorbers.

(e) For shipments of up to eight fuel modules, the standard internals assembly must be used. Fuel module cavity spacers must be used for all fuel modules.

(f) For full core shipments, the full core internals must be used.

9. For D2W spent fuel shipments:

(a) Up to one spent fuel core shipped per package. Fuel modules of different types may be shipped in any combination. Up to nine fuel modules may be shipped per package, provided that one of the fuel modules is the prototype RFA.

(b) Minimum fuel cooling time is 180 days after shutdown.

(c) All normally rodded fuel modules must have control rods. Control rod holddown devices must be installed on all rodded modules. The universal grapple adapters serve as the rod holddown devices.

(d) The standard internals assembly must be used for shipment of D2W fuel modules. All fuel modules must be shipped with the appropriate cell spacers, as shown in Appendix 1.4 of the application dated October 14, 1994.

10. For A1G spent fuel shipments:

(a) The A1G fuel modules may be shipped in any combination of fuel modules and cell support housings, and shall not exceed a total of eight items (partial shipments, with less than eight items, are acceptable).

(b) The A1G fuel modules shall be shipped with grapple adapters, support stands, rod holddown devices, and control rods installed. Fuel modules shall not be shipped with poison (shipping) rods installed.

(c) Cell support housings shall be shipped with grapple adapters and bottom spacers installed.

(d) 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 SARP.

11. For S6W spent fuel shipments:

(a) Up to eight fuel modules from a single core may be shipped.

(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.

(d) The standard internals assembly and the S6W guide spacers must be used for shipment of S6W fuel modules.

12. 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.

13. Failed fuel or fuel with defective cladding is not authorized for shipment.

14. 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.

15. 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.

16. 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.

17. Expiration date: October 31, 2001.

REFERENCES

"Core Independent M-140 Safety Analysis Report For Packaging," transmitted February 27, 1991.

Supplements dated: May 23, 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; and June 11, 1998.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

William F. Kane, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: August 10, 1998

NRC FORM 618
(3-96)
10 CFR 71

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIALS PACKAGES

U.S. NUCLEAR REGULATORY COMMISSION

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. PACKAGE IDENTIFICATION NUMBER	d. PAGE NUMBER	e. TOTAL NUMBER PAGES
9794	3	USA/9794/B(U)-85	1	3

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

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

c. DOCKET NUMBER 71-9794

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. Expiration date: September 30, 2005.

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; and December 22, 1999.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Susan F. Shankman for

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

Date: June 20, 2000

NRC FORM 618
(3-96)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIALS PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. PACKAGE IDENTIFICATION NUMBER	d. PAGE NUMBER	e. TOTAL NUMBER PAGES
9795	1	USA/9795/B(U)-85	1	2

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

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

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION:

A1G Irradiated Component Disposal Container
Safety Analysis Report for Packaging
dated July 10, 1997, as supplemented.

c. DOCKET NUMBER 71-9795

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.: A1G Irradiated Component Disposal Container

(2) Description

The Model No. A1G Irradiated Component Disposal Container 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).

(b) Contents

The contents of the package are A1G cell support housings and other miscellaneous core components from a spent A1G reactor core. The contents will not exceed 19 A1G cell support housing assemblies or will not exceed 18 A1G cell support housing assemblies and one miscellaneous component cylinder (MCC). 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.

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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 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 miscellaneous component cylinder is limited to a maximum of 39 upper lead screw segments (approximately 33 inches in length), 39 upper tie rod segments (approximately 33 inches in length), and a maximum of 25 cluster joint stud remnants. The maximum number of cluster upper joint stud nuts loaded in an ICDC, either inside the MCC or attached to cluster joint stud remnants at the bottom of cell support housings, must not exceed 250.

   (g) The gross weight of the package shall not exceed 200,000 pounds.

7. Expiration date: April 30, 2003

REFERENCES

A1G 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.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Cass R. Chappell

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

Date: 5/19/99

Type of Packaging: BYPROD. NORM. FORM

Model	Package ID #	Expiration Date
CI-20WC-2	USA/9098/B( )	05/31/2004
CI-20WC-2A	USA/9098/B( )	05/31/2004
PAS-1	USA/9184/B(U)	07/31/2004

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)	07/31/2002
C-1	USA/9036/B(U)	10/31/2000
EAGLE	USA/9287/B(U)-85	12/31/2004
F-294	USA/9258/B(U)-85	12/31/2003
GE-500	USA/9049/B( )	12/31/2000
IR-100	USA/9157/B(U)-85	09/30/2004
LCG-25A	USA/4888/B( )	01/31/2002
LCG-25B	USA/4888/B( )	01/31/2002
LCG-25C	USA/4888/B( )	01/31/2002
MW-3000	USA/9030/B( )	10/31/2005
NPI-20WC-6	USA/9102/B( )	10/31/2003
NPI-20WC-6 MKII	USA/9215/B(U)	10/31/2002
OP-100	USA/9185/B(U)-85	11/30/2003
OP-660	USA/9283/B(U)-85	06/30/2003
OPL-660	USA/9283/B(U)-85	06/30/2003
ORNL TRU CALIF	USA/5740/B( )	07/31/2001
SENTINEL-100F	USA/5862/B( )	09/30/2005
SENTINEL-25A	USA/4888/B( )	01/31/2002
SENTINEL-25B	USA/4888/B( )	01/31/2002
SENTINEL-25C	USA/4888/B( )	01/31/2002
SENTINEL-25C3	USA/4888/B( )	01/31/2002
SENTINEL-25D	USA/4888/B( )	01/31/2002
SENTINEL-25E	USA/4888/B( )	01/31/2002

U.S. Nuclear Regulatory Commission
List of Packages by Package Type

Type of Packaging: BYPROD. SPEC. FORM

Model	Package ID #	Expiration Date
SENTINEL-25F	USA/4888/B( )	01/31/2002
SENTINEL-8	USA/9030/B( )	10/31/2005
SNAP-21	USA/5830/B( )	11/30/2005
SPEC 2-T	USA/9056/B(U)	04/30/2005
SPEC-150	USA/9263/B(U)-85	06/30/2005
SPEC-300	USA/9282/B(U)-85	04/30/2005
URIPS-8A	USA/6786/B( )F	09/30/2003
URIPS-8B	USA/6786/B( )F	09/30/2003
181361	USA/5796/B(U)	07/31/2002
181375	USA/5796/B(U)	07/31/2002
4.5 TON CF	USA/6642/B( )	02/28/2002
420	USA/9245/B(U)	06/30/2002
5979	USA/5979/B( )	09/30/2005
5984	USA/5984/B( )	04/30/2001
650L	USA/9269/B(U)-85	11/30/2000
6717-B	USA/6717/B(U)	11/30/2003
680-OP	USA/9035/B(U)-85	05/31/2005
702	USA/6613/B(U)	06/30/2003
715	USA/9039/B(U)	12/31/2000
741-OP	USA/9027/B(U)-85	02/28/2001
770	USA/9148/B(U)	03/31/2002
771	USA/9107/B(U)	06/30/2003
855	USA/9165/B(U)	12/31/2003
865	USA/9187/B(U)	12/31/2003
934	USA/9243/B(U)	01/31/2001

U.S. Nuclear Regulatory Commission
List of Packages by Package Type

Type of Packaging: FISSILE URANIUM

Model	Package ID #	Expiration Date
ABB-2901	USA/9274/AF	07/31/2002
ANF-250	USA/9217/AF	06/30/2005
ATR	USA/9099/B(U)F-85	01/31/2004
A1W-3 PUSC	USA/9787/B(U)F	01/31/2005
BW-2901	USA/9251/AF	09/30/2002
CE-B1	USA/9272/AF-85	01/31/2002
DHTF	USA/9203/AF	01/31/2001
D2G POWER UNIT	USA/6441/B( )F	08/31/2002
ECO-PAK OP TU	USA/9288/AF-85	03/31/2005
ESP-30X	USA/9284/B(U)F-85	05/31/2005
FL 10-1	USA/9009/B( )F	09/30/2004
FPD-100	USA/9057/AF	09/30/2000
FSV-3	USA/6347/AF	05/31/2002
INNER HFIR UN	USA/5797/B(U)F	03/31/2002
MCC-3	USA/9239/AF	03/31/2002
MCC-4	USA/9239/AF	03/31/2002
MCC-5	USA/9239/AF	03/31/2002
MO-1	USA/9069/B( )F	12/31/2002
MODEL B	USA/6206/AF	09/30/2005
MODEL 1 S-6213	USA/9186/B(U)F	05/31/2002
MODEL 2 S-6213	USA/9186/B(U)F	05/31/2002
NCI-21PF-1	USA/9234/B(U)F	12/31/2003
NFS-URANYL NIT.	USA/5059/AF	08/31/1996
NNFD 5X22	USA/9250/B(U)F-85	01/31/2003

Type of Packaging: FISSILE URANIUM

Model	Package ID #	Expiration Date
NNFD-10	USA/6357/AF	04/30/2001
NONE SPECIFIED	USA/6406/AF	07/31/2002
OUTER HFIR UN	USA/5797/B(U)F	03/31/2002
PADUCAH TIGER	USA/6553/AF	07/31/2004
PATRIOT	USA/9292/AF-85	01/31/2005
RA-3	USA/4986/AF	03/31/2003
SP-1	USA/9248/AF	02/28/2004
SP-2	USA/9248/AF	02/28/2004
SP-3	USA/9248/AF	02/28/2004
SRP-1	USA/9285/AF-85	10/31/2003
ST	USA/9246/AF	09/30/2001
S5W POWER UNIT	USA/5580/B( )F	12/31/2002
TRIGA-I	USA/9034/AF	12/31/2000
TRIGA-II	USA/9037/AF	12/31/2000
UBE-1	USA/9280/AF-85	12/31/2002
UBE-2	USA/9281/AF-85	05/31/2003
UNC-2600	USA/5086/B(U)F	01/31/2004
UNC-2901	USA/6294/AF	03/31/2001
UX-30	USA/9196/AF-85	02/28/2001
WE-1	USA/9289/B(U)F-85	02/29/2004
235R001	USA/6386/B(U)F	04/30/2005
51032-1	USA/6581/AF	05/31/2004
51032-2	USA/9252/AF	09/30/2003
814A	USA/5149/B( )F	06/30/2005
927A1	USA/6078/AF	10/31/2005

U.S. Nuclear Regulatory Commission
List of Packages by Package Type

Type of Packaging: FISSILE URANIUM

Model	Package ID #	Expiration Date
927C1	USA/6078/AF	10/31/2005

Type of Packaging: IRRADIATED FUEL

Model	Package ID #	Expiration Date
BMI-1	USA/5957/B( )F	03/31/2001
CNS 1-13G	USA/9216/B( )F	12/31/2002
FSV-1	USA/6346/B( )F	05/31/2001
FSV-1 UNIT 3	USA/9277/B( )F	05/31/2001
GA-4	USA/9226/B(U)F-85	10/31/2003
GE-100	USA/5926/B( )F	05/31/2003
HI-STAR 100	USA/9261/B(U)F-85	03/31/2004
IF-300	USA/9001/B( )F	09/30/2005
M-130	USA/6003/B( )F	09/30/2002
M-140	USA/9793/B(U)F	10/31/2001
M-160	USA/9781/B( )F	01/31/2003
NAC-LWT	USA/9225/B(U)F-85	02/28/2005
NAC-STC	USA/9235/B(U)F-85	03/31/2004
NLI-1/2	USA/9010/B( )F	04/30/2001
NLI-10/24	USA/9023/B( )F	07/31/2003
NUHOMS-MP187	USA/9255/B(U)F-85	09/10/2003
T-2	USA/5607/B( )F	05/31/2003
T-3	USA/9132/B(M)F	04/01/2001
TN-BRP	USA/9202/B(U)F	06/30/2004
TN-FSV	USA/9253/B(U)F	05/31/2004
TN-REG	USA/9206/B(U)F	05/31/2005
TN-8	USA/9015/B( )F	05/31/2001
TN-8L	USA/9015/B( )F	05/31/2001
TN-9	USA/9016/B( )F	05/31/2001

U.S. Nuclear Regulatory Commission
List of Packages by Package Type

Type of Packaging: IRRADIATED FUEL

Model	Package ID #	Expiration Date
125-B	USA/9200/B(M)F	04/01/2001
2000	USA/9228/B(U)F-85	07/31/2000

U.S. Nuclear Regulatory Commission
List of Packages by Package Type

Type of Packaging: PU AIR

Model	Package ID #	Expiration Date
PAT-1	USA/0361/B(U)F-85	09/30/2003
PAT-2	USA/9150/B(U)-85	07/31/2001

U.S. Nuclear Regulatory Commission
List of Packages by Package Type

Type of Packaging: PU NORM. FORM

Model	Package ID #	Expiration Date
B-3	USA/6058/B( )	12/31/2000
NRBK-41	USA/9221/B( )F	09/30/2001
TRUPACT-II	USA/9218/B(U)F-85	06/30/2004
6400	USA/6400/B( )F	07/31/2002

Type of Packaging: PU SPEC. FORM

Model	Package ID #	Expiration Date
BCL-2	USA/9068/B( )F	05/31/2002
BCL-3	USA/9067/B( )F	09/30/2002
S5W REFUEL.SRCE	USA/5757/B( )F	03/31/2003
1500	USA/5939/B( )F	10/31/2003

U.S. Nuclear Regulatory Commission
List of Packages by Package Type

Type of Packaging: WASTE, B

Model	Package ID #	Expiration Date
AP-101	USA/9071/B( )	01/31/2002
A1G ICDC	USA/9795/B(U)-85	04/30/2003
CGN RCDP	USA/9794/B(U)-85	09/30/2005
CNS 1-13C	USA/9081/B( )	01/31/2003
CNS 1-13C II	USA/9152/B( )F	05/31/2004
CNS 10-160B	USA/9204/B(U)-85	10/31/2005
CNS 3-55	USA/5805/B( )	03/31/2004
CNS 8-120B	USA/9168/B(U)	06/30/2005
D1G CB-TS	USA/9792/B(U)	09/30/2002
N-55	USA/9070/B(U)	01/31/2005
NAC-1	USA/9183/B( )F	09/30/2004
PWR-2 CORE BAR.	USA/9791/B(U)-85	07/31/2002
RH-TRU 72-B	USA/9212/B(M)F-85	02/28/2005
S3G CBDCA	USA/9786/B(U)	08/31/2001
S5W REC. COMPT.	USA/9788/B(U)-85	09/30/2003
S6G REC. COMPT.	USA/9788/B(U)-85	09/30/2003
TN-RAM	USA/9233/B(U)	01/31/2005
10-135B	USA/9210/B(U)	01/31/2005
10-142	USA/9208/B( )	07/31/2001
3-82B	USA/6574/B( )	05/31/2001

NRC FORM 335 (2-89) NRCM 1102, 3201, 3202

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

(See instructions on the reverse)

1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

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Revision 23

2. TITLE AND SUBTITLE

Directory of Certificates of Compliance for Radioactive Materials Packages

Certificates of Compliance

3. DATE REPORT PUBLISHED

MONTH	YEAR
November	2000

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Spent Fuel Project Office
Office of Nuclear Material Safety and Safeguards
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555-0001

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Same as 8, above

10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

The purpose of this directory is to make available a convenient source of information on packagings 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.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

Transportation
Packaging
Radioactive Materials

13. AVAILABILITY STATEMENT

unlimited

14. SECURITY CLASSIFICATION

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unclassified

(This Report)

unclassified

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