NRC FORM 618 (8-2000) 10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

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

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

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

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

a. ISSUED TO *(Name and Address)*
Framatome ANP, Inc.
P.O. Box 11646
Lynchburg, VA 24506-1646

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION
B&W Fuel Company application dated March 9, 1993, as supplemented.

4. CONDITIONS

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

5.

(a) Packaging

(1) Model No.: 51032-2

(2) Description

A steel shipping container for fuel bundles, consisting of a strongback and fuel bundle clamping assembly, shock mounted to a steel outer container. Nine separator blocks, which are 6" x 8" x 8-1/2" long and have a 3/8" thick wall and a rectangular gusset plate welded inside, are bolted between fuel bundles. The outer container is composed of an 11 gauge steel shell approximately 43" diameter by 216" long. The maximum weight of the package, including contents, is 7,500 pounds.

(3) Drawings

The packaging is constructed and assembled in accordance with the following B&W Fuel Company Drawing Nos.: 1215926 C, Rev. 1; 1215929 D, Rev. 2; 1215930 D, Rev. 2; 1215931 D, Rev. 2; 1215932 D, Rev. 2; 1215933 D, Rev. 2; 1215934 C, Rev. 1; 1215935 D, Rev. 2; 1216010 D, Rev. 1.

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(b) Contents

(1) Type and form of material

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

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

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

(2) Maximum quantity of material per package

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

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

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

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

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

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

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

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

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

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

10. Expiration date: October 31, 2008.

REFERENCES

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

Supplements dated: May 10, and July 7, 1993; April 13, 1994; June 17, 1998; November 13, 2000; February 9, 2001; and August 26, 2003.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

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

Date: October 14, 2003

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CERTIFICATE OF COMPLIANCE
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2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*
U.S. Department of Energy
Washington, DC 20585

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION
Safety Analysis Report for the TN-FSV Package, dated March 31, 1993, as supplemented; Safety Analysis Report Addendum for the Oak Ridge Container in the TN-FSV Packaging, dated June 15, 2001, as supplemented.

4. CONDITIONS

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

5.

Packaging

(1) Model No.: TN-FSV

(2) Description

A steel and lead shielded shipping cask for irradiated nuclear fuel. The cask has two shipping configurations: Configuration 1 for shipping irradiated Fort St. Vrain high temperature gas cooled reactor (HTGR) fuel elements, and Configuration 2 for shipping irradiated fuel parts and intact irradiated Peach Bottom Unit 1 fuel elements within a secondary containment vessel. The cask is a right circular cylinder, with a balsa and redwood impact limiter at each end. The package has approximate dimensions and weights as follows:

Cavity diameter	18 inches
Cavity length	199 inches
Cask body outer diameter	31 inches
Lead shield thickness	3.44 inches
Package overall outer diameter, including impact limiters	78 inches
Package overall length, including impact limiters	247 inches
Packaging weight (Configuration 1)	42,000 pounds
Gross package weight, including contents (Configurations 1 and 2)	47,000 pounds

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

The cask body is made of two concentric shells of Type 304 stainless steel, welded to a bottom plate and a top closure flange. The inner shell has an ID of 18 inches and is 1.12 inches thick. The outer shell has an OD of approximately 30 inches and is 1.5 inches thick. The annular space between the inner and outer shells is filled with lead. The bottom plate is 5.5-inch thick Type 304 stainless steel. The closure lid is 2.5-inch thick Type 304 stainless steel, and is fully recessed into the cask top flange. The lid is fastened to the cask body by 12, 1-inch diameter closure bolts. The lid is sealed with double O-ring seals with a leak test port. A vent port and drain port are sealed with single O-rings and cover plates. Configuration 1 uses silicone O-ring seals and Configuration 2 uses butyl O-ring seals. The cask body is covered with a stainless steel thermal shield composed of 0.25-inch thick stainless steel plate over a wire wrap. The impact limiters are constructed of balsa and redwood encased in stainless steel shells.

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

<u>For Configuration 1</u>:

Irradiated hexagonal HTGR fuel elements are shipped in Configuration 1. The fuel elements are stacked in a carbon steel fuel storage container, which has an OD of approximately 17.6 inches and an overall length of 195 inches. The fuel storage container has a 0.5-inch thick shell, a 2.0-inch thick bottom plate, and a 1.5-inch thick lid. The lid accommodates a removable depleted uranium plug.

<u>For Configuration 2</u>:

Irradiated fuel parts and intact Peach Bottom Unit 1 fuel elements are shipped in Configuration 2. Canisters, containing either fuel parts or a single intact Peach Bottom fuel element, are loaded into a separate, secondary containment vessel, the Oak Ridge Container. The Oak Ridge Container is composed of a right circular cylindrical vessel and a basket assembly. The stainless steel vessel has a 10-gage (0.135-inch) wall thickness, an overall length of approximately 198 inches, and an outside diameter of approximately 20 inches at the lid end. The lid is approximately 7 inches thick and is closed by 12, 1/2-inch diameter bolts and two butyl O-ring seals. There is a single penetration through the lid which is closed by a bolted port cover and two butyl O-ring seals. The basket is composed of a series of discs, tie rods, and support tubes, with five fuel compartment tubes arranged in a star-like configuration. The basket incorporates fixed borated aluminum neutron poison plates. Flux trap spacers are positioned axially between stacked fuel parts canisters, and the canisters and spacers are positioned within a stainless steel sleeve that forms the fuel compartment. Canisters containing fuel parts (called Oak Ridge Canisters) and canisters containing intact Peach Bottom fuel elements may be shipped together.

5. (a) (3) Drawings

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

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

The Oak Ridge Container and internals are constructed and assembled in accordance with the following Transnuclear, Inc. Drawing Nos.:

3044-70-1, Rev. 5	3044-70-6, Rev. 2
3044-70-2, Rev. 3	3044-70-7, Rev. 2
3044-70-3, Rev. 2	3044-70-8, Rev. 1
3044-70-4, Rev. 2	3044-70-9, Rev. 0
3044-70-5, Rev. 2	

The Oak Ridge Canister is constructed and assembled in accordance with the following Lockheed Martin Energy Systems, Inc. Drawing No.:

X3E020566A175, Rev. 0

(b) Contents

(1) Type and form of material

(i) <u>For Configuration 1:</u>

Irradiated HTGR fuel elements within a fuel storage container. Each fuel element consists of a graphite block containing fuel rods. The fuel is composed of thorium/uranium carbide and thorium carbide fuel particles within the fuel rods. The graphite block is hexagonal in cross section and is approximately 14.2 inches across the flats and 31.2 inches long. Each fuel element contains a maximum of 1.4 kg of uranium enriched to a maximum of 93.5 weight percent U-235 and approximately 11.3 kg of thorium. The maximum burnup is approximately 70,000 MWd/MTIHM, and the minimum cool time is 1600 days.

5(b) (1) Type and form of material (Continued)

(ii) For Configuration 2:

Irradiated, intact Peach Bottom Unit 1, Core 2, fuel elements within aluminum canisters with steel liners. Each fuel element consists of stacked graphite annular rings, or compacts, with an inner diameter of approximately 1.75 inches and an outer diameter of approximately 2.75 inches. The fuel is composed of coated thorium/uranium carbide particles within the graphite. The active fuel length is approximately 90 inches. The fuel element may include associated hardware such as top plug, reflector apparatus, grappling hook, etc. Each fuel element contains a maximum of 0.25 kg of uranium enriched to a maximum of 93.15 weight percent U-235 and approximately 1.5 kg of thorium prior to irradiation. The maximum burnup is approximately 73,000 MWd/MTIHM and the minimum cool time is 27 years.

(iii) For Configuration 2:

Irradiated fuel parts within Oak Ridge Canisters, as described in Item No. 5(a)(3), above. The minimum fuel cool time is 15 years. The maximum fissile mass prior to irradiation per Oak Ridge Canister is limited as shown below:

Canister Group	Maximum mass U-235 per canister (grams)	Maximum mass Pu-239 + Pu-241 per canister (grams)
1	475	0
2	865	191
3	200	415
4	275	160
5	910	0

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

Total weight of contents and packaging material within the TN-FSV cavity not to exceed 5,000 pounds. For Configuration 1 this includes fuel elements, fuel storage container, and depleted uranium shield plug. For Configuration 2 this includes fuel materials, Oak Ridge Container, basket, Oak Ridge Canisters, Peach Bottom fuel canisters, flux trap spacers, and other packaging materials.

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

Six fuel elements, with decay heat not to exceed 60 watts per fuel element.

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

Total weight of fuel materials, canisters, and flux trap spacers within the Oak Ridge Container not to exceed 1,789 pounds. Decay heat not to exceed 120 watts per package. The maximum decay heat per Oak Ridge Canister is 35 watts, except that the maximum decay heat per Oak Ridge Canister in the position next to the lid is 7 watts. The maximum decay heat in any cross sectional region corresponding to the axial length of an Oak Ridge Canister is 55 watts, except that the maximum decay heat in the cross sectional region next to the lid is 35 watts.

Canisters containing intact Peach Bottom fuel elements and Oak Ridge Canisters containing irradiated fuel parts must be loaded into the Oak Ridge Container fuel compartments as follows:

Loading Pattern	One Fuel Compartment	Other Four Fuel Compartments
1	Four Group 2 Canisters	Four Group 1 Canisters
2	Four Group 5 Canisters	Four Group 1 Canisters
3	One Peach Bottom Element and One Group 4 Canister	One Peach Bottom Element and One Group 4 Canister
4	Two Group 3 Canisters and Two Group 4 Canisters	One Peach Bottom Element and One Group 4 Canister

Flux trap spacers, as shown in Transnuclear, Inc. Drawing No. 3044-70-3, must be positioned axially between any two Oak Ridge Canisters.

5. (c) Criticality Safety Index

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

6. The package must be leak tested as follows:

(a) <u>For Configuration 1</u>:

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

(2) Prior to each shipment, the package seals (main seal and vent seal) must be leak tested in accordance with Section 7.1.2 of the Safety Analysis Report. The acceptance criterion is a leak rate no greater than 1×10^{-3} ref-cm^3/sec. The test must have a sensitivity of at least 1×10^{-3} ref-cm^3/sec. The drain seal must also be tested if the drain port cover has been removed since the seal was last leak tested.

(b) <u>For Configuration 2</u>:

(1) In the 12-month period prior to shipment and after seal replacement, each containment seal of the outer cask and the Oak Ridge Container must be tested to show a leak rate no greater than 1×10^{-7} ref-cm^3/sec. The leak test must have a sensitivity of at least 5×10^{-8} ref-cm^3/sec.

(2) Prior to each shipment, the Oak Ridge Container containment seals (main seal and vent seal) and the outer cask containment seals (main seal and vent seal) must be leak tested in accordance with Section 7.1.2 of the Addendum. The seals must show no leakage greater than 1×10^{-7} ref-cm^3/sec or no leakage when tested to a sensitivity of at least 1×10^{-3} ref-cm^3/sec. The drain seal of the outer cask must also be tested if the drain port cover has been removed since the seal was last leak tested.

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

   (a) The package must be prepared for shipment and operated in accordance with the Operating Procedures of Chapter 7 of the Safety Analysis Report for Configuration 1, and Chapter 7 of the Addendum for Configuration 2.

   (b) Each packaging must meet the acceptance tests and must be maintained in accordance with the Acceptance Tests and Maintenance Program of Chapter 8 of the Safety Analysis Report. In addition, for Configuration 2, each packaging must meet the acceptance tests and must be maintained in accordance with the Acceptance Tests and Maintenance Program of Chapter 8 of the Addendum.

   (c) Prior to each shipment for Configuration 1 and Configuration 2, the cask main closure seal and vent seal must be inspected. The drain seal must be inspected if the drain port cover has been removed during preparation for shipment. All seals must be replaced within the 12-month period prior to shipment, or earlier if inspection shows any defect. In addition, prior to each shipment for Configuration 2, the Oak Ridge Container main closure seal and vent seal must be inspected. All seals must be replaced within the 12-month period prior to shipment, or earlier if inspection shows any defect.

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

9. Expiration date: May 31, 2009.

REFERENCES

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

U.S. Department of Energy supplements dated: March 24, 1997; March 24, 1999; June 15, September 18, October 2, 2001, and April 22, 2004.

Transnuclear, Inc. supplements dated September 19, 2001; and March 1, May 17, June 14 and 21, 2002 ; June 3, and July 21, 2003.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

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

Date: May 21, 2004

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

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2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

Transnuclear, Inc.
Four Skyline Drive
Hawthorne, NY 10532

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Transnuclear, Inc. consolidated application dated August 4, 2003.

4. CONDITIONS

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

5.

a. Packaging:

(1) Model No.: NUHOMS® MP187 Multi-Purpose Cask

(2) Description:

The NUHOMS® MP187 Multi-Purpose Cask (package) consists of an outer cask, into which one of the four different dry shielded canisters (DSC) is placed. During shipment, energy-absorbing impact limiters are utilized for additional package protection.

Cask

The purpose of the cask is to provide containment and shielding of the radioactive materials contained within the DSC during shipment. The cask is constructed of stainless steel and lead with a neutron shield of cementitious material. The inside cavity of the cask is a nominal 68 inches in diameter and 187 inches long. The bottom access closure is approximately 5 inches thick and 17 inches in diameter, secured by 12 1-inch diameter bolts. The top closure is approximately 6.5 inches thick and is secured by 36 2-inch diameter bolts. Both closures are sealed by redundant O-rings.

Containment is provided by a stainless steel closure lid bolted to the stainless steel cask. The containment system of the NUHOMS® MP187 transportation cask consists of (a) the inner shell, (b) the bottom end closure plate, (c) the top closure plate, (d) the top closure inner O-ring seal, (e) the ram closure plate, (f) the ram closure inner O-ring seal, (g) the vent port screw, (h) the vent port O-ring seal, (i) the drain port screw, and (j) the drain port O-ring seal. No credit is given to the DSC as a containment boundary.

Shielding is provided by 4 inches of stainless steel, 4 inches of lead, and approximately 4.3 inches of neutron shielding. The overall length of the cask is approximately 200 inches; the outer diameter is approximately 93 inches. The maximum gross weight of the package, with impact limiters, is approximately 282,000 lbs. The total length of the package with the impact limiters attached is approximately 308 inches. Four removable trunnions (two upper and two lower) are provided for handling and lifting.

Dry Shielded Canisters (DSCs)

The purpose of the DSC, which is placed within the transport cask, is to permit the transfer of spent fuel assemblies, into or out of a storage module, a dry transfer facility, or a pool as a unit. The DSC also provides additional axial biological shielding during handling and transport. The DSC consists of a stainless steel shell and a basket assembly. The approximately 5/8-inch thick shell has an outside diameter of about 67 inches and an external length of about 186 inches. The DSC basket assembly provides criticality control and contains a storage position for each fuel assembly. The basket is composed of circular spacer discs machined from thick carbon steel plates. Axial support for the DSC basket is provided by four high strength steel support rod assemblies. Carbon steel components of each DSC basket assembly are electrolytically coated with a thin layer of nickel to inhibit corrosion.

On the bottom of each DSC is a grapple ring, which is used to transfer a DSC horizontally from the cask into and out of dry storage modules. Because of the nature of the fuel that is to be transported, four different types of DSCs are designed for the package. Variations in the DSC configurations are summarized below:

- Fuel-Only Dry Shielded Canister (FO-DSC)

  The FO-DSC has a cavity length of approximately 167 inches and has solid carbon steel shield plugs at each end. The FO-DSC is designed to contain up to 24 intact Babcock and Wilcox (B&W) pressurized water reactor (PWR) spent fuel assemblies. The FO-DSC basket assembly consists of 24 guide sleeve assemblies with integral borated neutron absorbing plates, 26 spacer discs, and 4 support rod assemblies.

- Fuel/Control Components Dry Shielded Canister (FC-DSC)

  The FC-DSC has an internal cavity length of approximately 173 inches to accommodate fuel with the B&W control components installed. To obtain the increased cavity length, the shield plugs are fabricated from a composite of lead and steel. The FC basket is similar to the FO-DSC except that the support rod assemblies and guide sleeves are approximately 6-inches longer. The FC-DSC is also designed to contain up to 24 intact B&W PWR spent fuel assemblies with control components.

- Failed Fuel Dry Shielded Canister (FF-DSC)

  The FF-DSC has an internal cavity length of approximately 173 inches to accommodate 13 damaged B&W PWR spent fuel assemblies. Because the cladding has been locally degraded, individual (screened) fuel cans are provided to confine any gross loose material, maintain the geometry for criticality control, and facilitate loading and unloading operations. The FF-DSC is similar to FC-DSC in most respects with the exception of the basket assembly. The FF-DSC basket may be fabricated from austenitic stainless steel.

- 24PT1 Dry Shielded Canister (24PT1-DSC)

  The 24PT1-DSC has an internal cavity length of approximately 167 inches with a solid carbon steel shield plug at each end. The 24PT1-DSC will accommodate 22 to 24 Westinghouse (WE) 14 x14 PWR spent fuel assemblies, including control components. Control components authorized that are integral to WE 14x14 fuel assemblies include rod cluster control assemblies, thimble plug assemblies, and neutron source assemblies only. Fuel assemblies may be damaged or intact as described in 5.b(2)(a). The 24PT1-DSC basket assembly consists of 24 guide sleeve assemblies with integral borated neutron absorbing plates, 26 spacer discs, and 4 support rod assemblies. Up to four screened individual failed fuel cans are provided for storage of damaged fuel within the guide sleeve assemblies. These failed fuel cans are similar in configuration to the FF-DSC failed fuel cans.

Impact Limiters

The impact limiter shells are fabricated from stainless steel. Within that shell are closed-cell polyurethane foam and aluminum honeycomb material. The impact limiter is attached to the cask by carbon steel bolts. Each impact limiter is bolted to the cask body through the neutron shield top and bottom support rings. The weight of each impact limiter is approximately 15,800 lbs.

(3) Drawings

The package shall be constructed and assembled in accordance with the following Transnuclear West Drawing Numbers:

NUH-05-4000NP, Revision 9,
Sheets 1 through 2
MP187 Multi-Purpose Cask
General Arrangement

NUH-05-4001, Revision 15,
Sheets 1 through 6
MP187 Multi-Purpose Cask
Main Assembly

NUH-05-4002, Revision 5
Sheets 1 and 2
MP187 Multi-Purpose Cask
Impact Limiters

NH-05-4003, Revision 10,
Sheets 1 and 2
NUHOMS® MP187 Multi-Purpose Cask
On-Site Transfer Arrangement

NUH-05-4004, Revision 16,
Sheets 1 through 5
NUHOMS® FO-DSC & FC-DSC
PWR Fuel Main Assembly

NUH-05-4005, Revision 14,
Sheets 1 through 5
NUHOMS® FF-DSC
PWR Fuel Main Assembly

NUH-05-4006NP, Revision 7,
Sheets 1 and 2
NUHOMS® MP187 Multi-Purpose
Transportation Skid/Personnel Barrier

NUH-05-4010, Revision 2,
Sheets 1 through 6
NUHOMS® - 24PT1-DSC
Main Assembly

5.b Contents of Packaging

(1) Type and Form of Material:

(a) Intact fuel assemblies - Assemblies containing fuel rods with no known or suspected cladding defects greater than hairline cracks or pinhole leaks are authorized when contained in the FO-DSC, FC-DSC, or 24PT1-DSC.

(b) Damaged fuel assemblies - Assemblies containing fuel rods with known or suspected cladding defects greater than hairline cracks or pinhole leaks or with cracked, bulging, or discolored cladding are authorized when contained in a failed fuel can in the FF-DSC or the 24PT1-DSC. Spent fuel, with plutonium in excess of 20 curies per package, in the form of debris, particles, loose pellets, and fragmented rods or assemblies are not authorized. Damaged fuel assemblies may be shipped with or without control components.

(c) (i) The fuel authorized for shipment in the NUHOMS®-MP187 FO, FC, or FF DSC is B&W 15x15 uranium oxide PWR fuel assemblies with a maximum initial pellet enrichment of 3.43% by weight of U235, and a total uranium content not to exceed 466 Kg per assembly.

(ii) The fuel authorized for shipment in the NUHOMS®-MP187 24PT1-DSC is WE 14x14 stainless steel clad (SC) or zircaloy clad mixed oxide (MOX) PWR fuel assemblies as described in Table 2.

(d) Intact B&W 15x15 fuel assemblies without control components shall be shipped only in the FO-DSC. Intact B&W 15x15 fuel assemblies with control components shall be shipped only in the FC-DSC.

(e) Intact WE 14x14 fuel assemblies with or without control components shall be shipped only in the 24PT1-DSC. Control components authorized are integral to WE 14x14 fuel assemblies include rod cluster control assemblies, thimble plug assemblies, and neutron source assemblies only.

(f) (i) The maximum burn-up and minimum cooling times for the individual B&W 15x15 assemblies shall meet the requirements of Table 1. In addition, the fuel shall have been decayed for a time sufficient to meet the thermal criteria of 5.b(1)(g) and (h). The maximum total allowable cask heat load is 13.5 kW.

(ii) The maximum enrichment, burn-up and minimum cooling times for the individual WE 14x14 fuel assemblies shall meet the requirements of Table 2. In addition, the fuel shall have been decayed for a time sufficient to meet the thermal criteria of 5.b.(1)(g) and (h). The maximum total allowable cask heat load for the 24 PT1-DSC is per Table 2.

(g) (i) The maximum assembly decay heat (including control components when present) of B&W 15x15 individual fuel assembly is 0.764 kW, referred to as Type I, or 0.563 kW, referred to as Type II.

(ii) The maximum assembly decay heat (including control components when present) of WE 14x14 individual fuel assembly is per Table 2.

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5.b Contents of Packaging:

(1) Type and Form of Material Continued:

(h) (i) Control components for B&W 15x15 fuel assemblies stored in the FO, FC and FF-DSCs shall be cooled for at least 8 years.

(ii) Control components for WE 14x14 fuel assemblies stored in the 24PT1-DSC shall be cooled for at least 10 years.

(2) Maximum quantity of material per package:

(a) (i) For material described in 5.b(1) to be stored in the FO, FC or FF-DSCs: 24 PWR intact fuel assemblies or 13 damaged fuel assemblies, with no more than 15 damaged fuel rods per assembly. Where a DSC is to be loaded with fewer fuel assemblies than the DSC capacity, dummy fuel assemblies with the same nominal weight as a standard fuel assembly shall be installed in the unoccupied spaces.

(ii) For material described in 5.b(1) to be stored in the 24PT1-DSC: 22 to 24 PWR fuel assemblies of which up to four may be damaged WE 14x14 SC fuel assemblies with the balance intact WE 14x14 SC or MOX fuel assemblies. No more than one damaged WE 14x14 MOX fuel assembly can be stored per 24PT1-DSC with the balance intact WE 14x14 SC fuel assemblies. The damaged fuel assemblies shall have no more than 14 damaged fuel rods per assembly and shall be stored in the four outer corner fuel assembly locations along the 45°, 135°, 225°, 315° azimuth of the 24PT1-DSC. A DSC may include two empty slots if they are located on symmetrically opposite locations with respect to the 0° - 180° and 90°-270° DSC axes. Any additional empty fuel slots shall be loaded with dummy fuel assemblies that displace the same or greater amount of volume and with the same nominal weight as a standard fuel assembly. Fuel spacers shall be located at the bottom and top of each fuel assembly to center the fuel assemblies within the DSC. Failed fuel cans require only bottom spacers since a top spacer is integral to each failed fuel can.

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

Table 1- FO, FC and FF-DSC Fuel Assembly Burn-up vs. Cooling Time

Maximum Burn-up (MWD/MTIHM)*	Minimum Enrichment in the Active Fuel Region (w/o U-235)	Minimum Required Type I Cooling Time (years)	Minimum Required Type II Cooling Time (years)	Maximum Burn-up (MWD/MTIHM)*	Minimum Enrichment in the Active Fuel Region (w/o U-235)	Minimum Required Type I Cooling Time (years)	Minimum Required Type II Cooling Time (years)
<23,200	n/a	5	5	33,000	2.90	7	10
23,200	2.38	5	5	34,000	2.95	7	11
24,000	2.43	5	6	35,000	2.67	7	14
25,000	2.49	5	6	35,000	2.99	7	11
26,000	2.55	5	7	36,000	3.03	8	13
27,000	2.61	5	7	37,000	3.00	8	14
28,000	2.66	5	8	37,000	3.07	8	14
29,000	2.00	6	10	38,000	3.11	9	15
29,000	2.71	5	8	39,000	3.15	9	16
30,000	2.76	6	8	40,000	3.19	9	17
31,000	2.81	6	9				
32,000	2.86	6	10	* Megawatt Days per Metric Ton of Initial Heavy Metal			

Table 2 - 24PT1-DSC Fuel Assembly Burnup vs. Cooling Time

Fuel Type	Maximum Enrichment (Weight %)	Minimum Enrichment (Weight %)	Maximum Burnup (MWD/ MTU)	Minimum Cooling Time / Max Heat Load Per Cask / Max Assembly Heat Load (Incl. Control Components[1])
WE 14x14 Stainless Steel Clad (SC) (May Include Integral Fuel Burnable Absorber, boron coated fuel pellets)	4.05 ^{235}U	3.76 ^{235}U	45,000	38 years/14 kW/ 0.583 kW
		3.36 ^{235}U	40,000	
		3.12 ^{235}U	35,000	
WE 14x14 MOX	0.71 ^{235}U 2.84 fissile Pu (64 rods) 3.10 fissile Pu (92 rods) 3.31 fissile Pu (24 rods)	2.78 fissile Pu (64 rods) 3.05 fissile Pu (92 rods) 3.25 fissile Pu (24 rods)	25,000	30 years/13.706 kW/ 0.294 kW

Notes:

1 Control component cooling time must be a minimum of 10 years.

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

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

6. Type I fuel assemblies shall be loaded only into the four innermost cells of a DSC, while Type II assemblies may be loaded into any cell when using the FO-DSC or the FC-DSC. The FF-DSC has no Type I or II placement restrictions. The 24PT1-DSC has restrictions on the location of damaged fuel assemblies per Section 5.b.(2).

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

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

b. All fabrication acceptance tests and maintenance shall be performed in accordance with the Acceptance Tests and Maintenance Program in Chapter 8, as supplemented. In addition, this shall include:

(1) With the exception of the weld between the inner shell and top forging, all longitudinal and circumferential inner shell welds, which form the containment boundary of the cask, shall be radiographically inspected (RT) with acceptance standards in accordance with the ASME Code, Section III, Division 1, NB-5320. The weld between the inner shell and top forging shall be verified by RT or ultrasonically inspected (UT). The substitution of UT for the examination of the completed weld may be made provided the examination is performed using detailed written procedures, proven by actual demonstration to the satisfaction of the inspector as capable of detecting and locating defects described in ASME Code, Section III, Division 1 Subsection NB

(2) Verification of the DSC outer top cover plate weld by either volumetric or multilayer PT examination. If PT is used, at a minimum, it must include the root, each successive 1/4 inch weld thickness, and the final layer. The inspection of the weld must be performed by qualified personnel and shall meet the acceptance requirements of ASME B&PVC Section III, NB-5350. The inspection process, including findings (indications) shall be made a permanent part of the licensee's records by video, photographic, or other means providing an equivalent retrievable record of weld integrity.

(3) The minimum lead thickness in the main cask body, away from the trunnions and the top and bottom forgings, shall be 3.90 inches.

(4) The neutron shield shall have a minimum thickness of 4.31 inches.

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

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

10. Expiration Date: October 31, 2008.

REFERENCES

Transnuclear, Inc. application dated August 4, 2003.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

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

Date: October 14, 2003

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9258	3	71-9258	USA/9258/B(U)-96	1	OF	3

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

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

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

MDS Nordion consolidated application dated August 1, 2003, as supplemented.

4. CONDITIONS

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

5.
(a) Packaging

(1) Model No.: F-294

(2) Description

A steel encased, lead shielded shipping cask for special form sources. The package consists of a cylindrical cask body with cooling fins, a closure plug, a cylindrical external fireshield, a top crush shield, a permanent skid, and a removable shipping skid. The special form sources are positioned by a source carrier within the cask cavity. There are two alternative source carriers. The F-313 source carrier holds forty special form sources in a single ring configuration. The F-457 source carrier holds eighty special form sources in a double ring configuration.

The cask body is constructed of a ½-inch thick inner stainless steel shell, and a ½-inch thick outer stainless steel shell. The annulus between the inner and outer shells is filled with lead, approximately 11 ¼ inches thick. The cask is closed by a 2 ½ inch thick stainless steel closure lid and 16 one-inch diameter bolts. A lead radiation protection plug is fitted to the cask closure plate. Stainless steel fins are welded onto the exterior of the cask to dissipate heat. The cask is surrounded by a cylindrical fireshield which is constructed of ceramic fiber thermal insulation encased in carbon steel shells. A composite assembly consisting of a finned crush shield that acts as an impact limiter and a fireshield is bolted to the top end of the cask. The cask is equipped with a fixed skid and a shipping skid composed of steel beams. The fixed skid includes a sheet of thermal insulation enclosed in steel.

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9258	3	71-9258	USA/9258/B(U)-96	2	OF	3

5.(a) Packaging

(2) Description (continued)

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

Cask body outer diameter (excluding cooling fins)	36 inches
Cask body height	52 1/4 inches
Cask cavity inside diameter	11 1/2 inches
Cask cavity inside height	19 3/4 inches
Lead shield thickness	11 1/4 inches
Fireshield outer diameter	47 inches
Overall package dimensions (including shipping skid)	
width	78 inches
length	78 inches
height	80 1/2 inches
Maximum contents weight	40 pounds
Maximum package weight (including contents)	21,000 pounds

(3) Drawings

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

F629401-001, Sheet 1, Rev. F,
F629401-001, Sheet 2, Rev. F,
F629401-001, Sheet 3, Rev. D,
F629401-001, Sheet 4, Rev. F,
F629401-001, Sheet 5, Rev. F,
F631301-001, Rev. B, and
F645701-001, Rev. A.

(b) Contents

(1) Type and form of material

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

(1) Maximum quantity of material per package

360,000 Curies

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9258	3	71-9258	USA/9258/B(U)-96	3	OF	3

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

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

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

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

8. Revision 2 of this Certificate may be used until October 31, 2009.

9. Expiration date: December 31, 2013.

<u>REFERENCES</u>

MDS Nordion application dated August 1, 2003.

Supplements dated: March 12, April 20, May 20, 2004; and September 12, 2008.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Eric J. Benner

Eric J. Benner, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: 10/22/08

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9261	5	71-9261	USA/9261/B(U)F-85	1	OF	10

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*
Holtec International
Holtec Center
555 Lincoln Drive West
Marlton, NJ 08053

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION
Holtec International Report No. HI-951251, *Safety Analysis Report for the Holtec International Storage, Transport, And Repository Cask System (HI-STAR 100 Cask System)* Revision 12, dated October 9, 2006.

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

5.
(a) Packaging

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

(2) Description

The HI-STAR 100 System is a canister system comprising a Multi-Purpose Canister (MPC) inside of an overpack designed for both storage and transportation (with impact limiters) of irradiated nuclear fuel. The HI-STAR 100 System consists of interchangeable MPCs that house the spent nuclear fuel and an overpack that provides the containment boundary, helium retention boundary, gamma and neutron radiation shielding, and heat rejection capability. The outer diameter of the overpack of the HI-STAR 100 is approximately 96 inches without impact limiters and approximately 128 inches with impact limiters. Maximum gross weight for transportation (including overpack, MPC, fuel, and impact limiters) is 282,000 pounds. Specific tolerances germane to the safety analyses are called out in the drawings listed below.

Multi-Purpose Canister

There are six Multi-Purpose Canister (MPC) models designated as the MPC-24, MPC-24E, MPC-24EF, MPC-32, MPC-68, and MPC-68F. All MPCs are designed to have identical exterior dimensions, except those MPC-24E/EFs custom-designed for the Trojan plant, which are approximately nine inches shorter than the generic Holtec MPC design. A single overpack design is provided that is capable of containing each type of MPC. The two digits after the MPC designate the number of reactor fuel assemblies for which the respective MPCs are designed. The MPC-24 series is designed to contain up to 24 Pressurized Water Reactor (PWR) fuel assemblies; the MPC-32 is designed to contain up to intact 32 PWR assemblies; and the MPC-68 and MPC-68F are designed to contain up to 68 Boiling Water Reactor (BWR) fuel assemblies. BWR fuel debris may be shipped only in the MPC-68F.

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

PWR spent fuel assemblies classified as fuel debris may be loaded only in MPC-24EF.

The HI-STAR 100 MPC is a welded cylindrical structure with flat ends. Each MPC is an assembly consisting of a honeycombed fuel basket, baseplate, canister shell, lid, and closure ring. The outer diameter and cylindrical height of each generic MPC is fixed. The outer diameter of the Trojan MPCs is the same as the generic MPC, but the height is approximately nine inches shorter than the generic MPC design. A steel spacer is used with the Trojan plant MPCs to ensure the MPC-overpack interface is bounded by the generic design. The fuel basket designs vary based on the MPC model. The MPC pressure boundary is a welded enclosure constructed entirely of a stainless steel alloy.

Overpack

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

Impact Limiters

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

(3) Drawings

The package shall be constructed and assembled in accordance with the following drawings or figures in Holtec International Report No. HI-951251, *Safety Analysis Report for the Holtec International Storage, Transport, And Repository Cask System (HI-STAR 100 Cask System)*, Revision 12:

(a) HI-STAR 100 Overpack	Drawing 3913, Sheets 1-9, Rev. 7
(b) MPC Enclosure Vessel	Drawing 3923, Sheets 1-5, Rev. 14
(c) MPC-24E/EF Fuel Basket	Drawing 3925, Sheets 1-4, Rev. 5
(d) MPC-24 Fuel Basket Assembly	Drawing 3926, Sheets 1-4, Rev. 5
(e) MPC-68/68F/68FF Fuel Basket	Drawing 3928, Sheets 1-4, Rev. 5

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

(f) HI-STAR 100 Impact Limiter CoC No. 9261, Appendix B	Drawing C1765, Sheets 1 and 2, Rev. 2; Sheet 3, Rev. 1, Sheet 4, Rev. 2; Sheets 5 and 6, Rev. 1; and Sheet 7, Rev. 0.
(g) HI-STAR 100 Assembly for Transport	Drawing 3930, Sheets 1-3, Rev. 1
(h) Trojan MPC-24E/EF Spacer Ring	Drawing 4111, Sheets 1-2, Rev. 0
(i) Damaged Fuel Container for Trojan Plant SNF	Drawing 4119, Sheet 1-4, Rev. 1
(j) Spacer for Trojan Failed Fuel Can	Drawing 4122, Sheets 1-2, Rev. 0
(k) Failed Fuel Can for Trojan	SNC Drawings PFFC-001, Rev. 8 and PFFC-002, Sheets 1 and 2, Rev. 70
(l) HI-STAR 100 MPC-32	Drawing 3927, Sheets 1-4, Rev. 6

5.(b) Contents

(1) Type, Form, and Quantity of Material

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

(b) The following definitions apply:

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

Damaged Fuel Containers (or Canisters)(DFCs) are specially designed fuel containers for damaged fuel assemblies or fuel debris that permit gaseous and liquid media to escape while minimizing dispersal of gross particulates. The DFC designs authorized for use in the HI-STAR 100 are shown in Figures 1.2.10 and 1.2.11 of the HI-STAR 100 System SAR, Rev. 12.

Fuel Debris is ruptured fuel rods, severed rods, loose fuel pellets, and fuel assemblies with known or suspected defects which cannot be handled by normal means due to fuel cladding damage. Fuel debris also includes certain Trojan plant-specific fuel material contained in Trojan Failed Fuel Cans.

5.(b)(1)(b) Definitions (continued)

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

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

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

Non-Fuel Hardware is defined as Burnable Poison Rod Assemblies (BPRA), Thimble Plug Devices (TDPs), and Rod Cluster Control Assemblies (RCCAs).

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

Trojan Damaged Fuel Containers (or Canisters) are Holtec damaged fuel containers custom-designed for Trojan plant damaged fuel and fuel debris as depicted in Drawing 4119, Rev. 1.

Trojan Failed Fuel Cans are non-Holtec designed Trojan plant-specific damaged fuel containers that may be loaded with Trojan plant damaged fuel assemblies, Trojan fuel assembly metal fragments (e.g., portions of fuel rods and grid assemblies, bottom nozzles, etc.), a Trojan fuel rod storage container, a Trojan Fuel Debris Process Can Capsule, or a Trojan Fuel Debris Process Can. The Trojan Failed Fuel Can is depicted in Drawings PFFC-001, Rev. 8 and PFFC-002, Rev. 7.

Trojan Fuel Debris Process Cans are Trojan plant-specific canisters containing fuel debris (metal fragments) and were used to process organic media removed from the Trojan plant spent fuel pool during cleanup operations in preparation for spent fuel pool decommissioning. Trojan Fuel Debris Process Cans are loaded into Trojan Fuel Debris Process Can Capsules or directly into Trojan Failed Fuel Cans. The Trojan Fuel Debris Process Can is depicted in Figure 1.2.10B of the HI-STAR100 System SAR, Rev. 12.

Trojan Fuel Debris Process Can Capsules are Trojan plant-specific canisters that contain up to five Trojan Fuel Debris Process Cans and are vacuumed, purged, backfilled with helium and then seal-welded closed. The Trojan Fuel Debris Process Can Capsule is depicted in Figure 1.2.10C of the HI-STAR 100 System SAR, Rev. 12.

5.(b)(1)(b) Definitions (continued)

ZR means any zirconium-based fuel cladding materials authorized for use in a commercial nuclear power plant reactor.

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

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

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

(f) PWR non-fuel hardware and neutron sources are not authorized for transportation except as specifically provided for in Appendix A to this CoC.

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

(h) For spent fuel assemblies to be loaded into MPC-32s, core average soluble boron, assembly average specific power, and assembly average moderator temperature in which the fuel assemblies were irradiated, shall be determined according to Section 1.2.3.7.1 of the SAR, and the values shall be compared against the limits specified in Part VI of Table A.1 in Appendix A of this Certificate of Compliance.

(i) For spent fuel assemblies to be loaded into MPC 32s, the reactor records on spent fuel assemblies average burnup shall be confirmed through physical burnup measurements as described in Section 1.2.3.7.2 of the SAR.

5.(c) Criticality Safety Index (CSI)= 0.0

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

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

6.(a) (continued)

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

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

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

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

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

(b) Within 30 days before each shipment, all overpack containment boundary seals shall be leak tested using a test with a minimum sensitivity of 1×10^{-3} atm cm^3/sec. If leakage is detected on a seal, then the seal must be replaced and leak tested per Condition 6.(a)(3)(a) above.

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

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

(5) MPC-68F and MPC-24EF shall be leak tested prior to shipment to show a leak rate of no greater than 5×10^{-6} atm cm^3/sec (helium). The leak test shall have a minimum sensitivity of 2.5×10^{-6} atm cm^3/sec (helium).

(6) MPCs deployed at an ISFSI under 10 CFR Part 72 prior to transportation may be dried using the vacuum drying method or the Forced Helium Dehydration (FHD) method. MPCs placed directly into transportation service under 10 CFR 71 without first being deployed at an ISFSI must be dried using the FHD method. Water and residual moisture shall be removed from the MPC in accordance with the following specifications:

For those MPCs vacuum dried:

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

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

6.(a) (continued)

For those MPCs dried using the FHD System:

(a) Following bulk moisture removal, the temperature of the gas exiting the demoisturizer shall be ≤ 21°F for ≥ 30 minutes.

(7) Following drying, the MPC shall be backfilled with 99.995% minimum purity helium: > 0 psig and ≤ 44.8 psig at a reference temperature of 70 °F.

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

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

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

(9) Following vacuum drying, the overpack shall be backfilled with helium to ≥ 10 psig and ≤ 14 psig.

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

Fastener	Torque (ft-lbs)
Overpack Closure Plate Bolts	2895 ± 90
Overpack Vent and Drain Port Plugs	45 +5/-2
Top Impact Limiter Attachment Bolts	256 +10/-0
Bottom Impact Limiter Attachment Bolts	1500 +45/-0

(11) Verify that the appropriate fuel spacers, as necessary, are used to position the active fuel zone within the neutron absorber plates of the MPC, and limit axial movement of fuel assemblies in the MPC cavity.

(12) Appropriate monitoring for combustible gas concentration shall be performed prior to, and during MPC lid welding and weld cutting operations. The space below the MPC lid shall be exhausted or purged with inert gas prior to, and during MPC lid welding and weld cutting operations to provide additional assurance that flammable gas concentrations will not develop in this space.

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

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

(2) The MPC shall be pressure tested in accordance with ASME Section III, Subsection NB, Article NB-6110 and applicable sub-articles. If hydrostatic testing is used, the

6.(b) (continued)

MPC shall be pressure tested to 125% of the design pressure. The minimum hydrostatic test pressure shall be 125 psig. If pneumatic testing is used, the MPC shall be pressure tested to 120% of the design pressure. The minimum pneumatic test pressure shall be 120 psig.

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

(4) The MPC lid-to-shell (LTS) weld shall be verified by either volumetric examination using the ultrasonic (UT) method or multi-layer liquid penetrant (PT) examination. The root and final weld layers shall be PT examined in either case. If PT alone is used, additional intermediate PT examination(s) shall be conducted after each

approximately 3/8 inch of the weld is completed. The inspection of the weld must be performed by qualified personnel and shall meet the acceptance requirements of ASME B&PV Section III, NB-5350. The inspection results, including all relevant indications shall be made a permanent part of the licensee's records by video, photographic, or other means providing an equivalent retrievable record of weld integrity.

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

(6) Periodic verification of the neutron shield integrity shall be performed within 5 years prior to each shipment. The periodic verification shall be performed by radiation measurements with either loaded contents or a check source. Measurements shall be taken at three cross sectional planes through the radial shield and at four points along each plane's circumference. The average measurement results from each sectional plane shall be compared to calculated values to assess the continued effectiveness of the neutron shield. The calculated values shall be representative of the loaded contents (i.e., fuel type, enrichment, burnup, cooling time, etc.) or the particular check source used for the measurements.

(7) The first fabricated HI-STAR 100 overpack shall be tested to confirm its heat transfer capability. The test shall be conducted after the radial channels, enclosure shell panels, and neutron shield material have been installed and all inside and outside surfaces are painted per the Design Drawings specified in Section 5.(a)(3) of this Certificate of Compliance. A test cover plate shall be used to seal the overpack cavity. Testing shall be performed in accordance with written and approved procedures. The test must demonstrate that the overpack is fabricated adequately to meet the design heat transfer capability.

6.(b) (continued)

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

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

(10) Flux trap sizes:

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

(b) The minimum flux trap sizes for the generic MPC-24E and MPC-24EF are 0.776 inch for cells 3, 6, 19, and 22, and 1.076 inch for the remaining cells.

(c) The minimum flux trap sizes for the Trojan MPC-24E and MPC-24EF are 0.526 inch for cells 3, 6, 19, and 22, and 1.076 inch for the remaining cells.

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

(12) The package containment verification leak test shall be per ANSI 14.5-1997.

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

8. The package shall be located on the transport vehicle such that the bottom surface of the bottom impact limiter is at least 9 feet (along the axis of the overpack) from the edge of the vehicle.

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

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

11. Revision No. 4 of this certificate may be used until October 12, 2007.

12. Expiration Date: March 31, 2009

Attachment: Appendix A

REFERENCES:

Holtec International Report No. HI-951251, *Safety Analysis Report for the Holtec International Storage, Transport, And Repository Cask System (HI-STAR 100 Cask System)*, Revision 12, dated October 9, 2006.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Robert A. Nelson, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: October 12, 2006

APPENDIX A

CERTIFICATE OF COMPLIANCE NO. 9261, REVISION 5

MODEL NO. HI-STAR 100 SYSTEM

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Table A.1 (Page 1 of 21)
Fuel Assembly Limits

I. MPC MODEL: MPC-24

A. Allowable Contents

1. Uranium oxide, PWR intact fuel assemblies listed in Table A.2 and meeting the following specifications:

a. Cladding type:	ZR or stainless steel (SS) as specified in Table A.2 for the applicable fuel assembly array/class
b. Maximum initial enrichment:	As specified in Table A.2 for the applicable fuel assembly array/class.
c. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly	
i. ZR clad:	An assembly post-irradiation cooling time, average burnup, and minimum initial enrichment as specified in Table A.4 or A.5, as applicable.
ii. SS clad:	An assembly post-irradiation cooling time, average burnup, and minimum initial enrichment as specified in Table A.6, as applicable.
d. Decay heat per assembly:	
i. ZR Clad:	≤833 Watts
ii. SS Clad:	≤488 Watts
e. Fuel assembly length:	≤ 176.8 inches (nominal design)
f. Fuel assembly width:	≤ 8.54 inches (nominal design)
g. Fuel assembly weight:	≤ 1,680 lbs

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

C. Fuel assemblies shall not contain non-fuel hardware or neutron sources.

D. Damaged fuel assemblies and fuel debris are not authorized for transport in the MPC-24.

E. Trojan plant fuel is not permitted to be transported in the MPC-24.

Table A.1 (Page 2 of 21)
Fuel Assembly Limits

II. MPC MODEL: MPC-68

A. Allowable Contents

1. Uranium oxide, BWR intact fuel assemblies listed in Table A.3, with or without Zircaloy channels, and meeting the following specifications:

a. Cladding type:	ZR or stainless steel (SS) as specified in Table A.3 for the applicable fuel assembly array/class.
b. Maximum planar-average initial enrichment:	As specified in Table A.3 for the applicable fuel assembly array/class.
c. Initial maximum rod enrichment:	As specified in Table A.3 for the applicable fuel assembly array/class.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	
i. ZR clad:	An assembly post-irradiation cooling time, average burnup, and minimum initial enrichment as specified in Table A.7, except for (1) array/class 6x6A, 6x6C, 7x7A, and 8x8A fuel assemblies, which shall have a cooling time $\geq$ 18 years, an average burnup $\leq$ 30,000 MWD/MTU, and a minimum initial enrichment $\geq$ 1.8 wt% ^{235}U, and (2) array/class 8x8F fuel assemblies, which shall have a cooling time $\geq$ 10 years, an average burnup $\leq$ 27,500 MWD/MTU, and a minimum initial enrichment $\geq$ 2.4 wt% ^{235}U.
ii. SS clad:	An assembly cooling time after discharge $\geq$ 16 years, an average burnup $\leq$ 22,500 MWD/MTU, and a minimum initial enrichment $\geq$ 3.5 wt% ^{235}U.
e. Decay heat per assembly:	
i. ZR Clad:	$\leq$272 Watts, except for array/class 8X8F fuel assemblies, which shall have a decay heat $\leq$183.5 Watts.
ii SS Clad:	$\leq$83 Watts
f. Fuel assembly length:	$\leq$ 176.2 inches (nominal design)
g. Fuel assembly width:	$\leq$ 5.85 inches (nominal design)
h. Fuel assembly weight:	$\leq$ 700 lbs, including channels

Table A.1 (Page 3 of 21)
Fuel Assembly Limits

II. MPC MODEL: MPC-68 (continued)

A. Allowable Contents (continued)

2. Uranium oxide, BWR damaged fuel assemblies, with or without Zircaloy channels, placed in damaged fuel containers. Uranium oxide BWR damaged fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6A, 6x6C, 7x7A, or 8x8A, and meet the following specifications:

a. Cladding type:	ZR
b. Maximum planar-average initial enrichment:	As specified in Table A.3 for the applicable fuel assembly array/class.
c. Initial maximum rod enrichment:	As specified in Table A.3 for the applicable fuel assembly array/class.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time $\geq$ 18 years, an average burnup $\leq$ 30,000 MWD/MTU, and a minimum initial enrichment $\geq$ 1.8 wt% ^{235}U.
e. Fuel assembly length:	$\leq$ 135.0 inches (nominal design)
f. Fuel assembly width:	$\leq$ 4.70 inches (nominal design)
g. Fuel assembly weight:	$\leq$ 550 lbs, including channels and damaged fuel container

Table A.1 (Page 4 of 21)
Fuel Assembly Limits

II. MPC MODEL: MPC-68 (continued)

A. Allowable Contents (continued)

3. Mixed oxide (MOX), BWR intact fuel assemblies, with or without Zircaloy channels. MOX BWR intact fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6B and meet the following specifications:

a. Cladding type:	ZR
b. Maximum planar-average initial enrichment:	As specified in Table A.3 for fuel assembly array/class 6x6B.
c. Initial maximum rod enrichment:	As specified in Table A.3 for fuel assembly array/class 6x6B.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time $\geq$ 18 years, an average burnup $\leq$ 30,000 MWD/MTIHM, and a minimum initial enrichment $\geq$ 1.8 wt% ^{235}U for the UO_2 rods.
e. Fuel assembly length:	$\leq$ 135.0 inches (nominal design)
f. Fuel assembly width:	$\leq$ 4.70 inches (nominal design)
g. Fuel assembly weight:	$\leq$ 400 lbs, including channels

Table A.1 (Page 5 of 21)
Fuel Assembly Limits

II. MPC MODEL: MPC-68 (continued)

A. Allowable Contents (continued)

4. Mixed oxide (MOX), BWR damaged fuel assemblies, with or without Zircaloy channels, placed in damaged fuel containers. MOX BWR damaged fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6B and meet the following specifications:

a. Cladding type:	ZR
b. Maximum planar-average initial enrichment:	As specified in Table A.3 for array/class 6x6B.
c. Initial maximum rod enrichment:	As specified in Table A.3 for array/class 6x6B.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time ≥ 18 years, an average burnup ≤ 30,000 MWD/MTIHM, and a minimum initial enrichment ≥ 1.8 wt% ^{235}U for the UO_2 rods.
e. Fuel assembly length:	≤ 135.0 inches (nominal design)
f. Fuel assembly width:	≤ 4.70 inches (nominal design)
g. Fuel assembly weight:	≤ 550 lbs, including channels

Table A.1 (Page 6 of 21)
Fuel Assembly Limits

II. MPC MODEL: MPC-68 (continued)

A. Allowable Contents (continued)

5. Thoria rods (ThO_2 and UO_2) placed in Dresden Unit 1 Thoria Rod Canisters (as shown in Figure 1.2.11A of the HI-STAR 100 System SAR, Revision 12) and meeting the following specifications:

a. Cladding type:	ZR
b. Composition:	98.2 wt.% ThO_2, 1.8 wt. % UO_2 with an enrichment of 93.5 wt. % ^{235}U.
c. Number of rods per Thoria Rod Canister:	$\leq$ 18
d. Decay heat per Thoria Rod Canister:	$\leq$ 115 Watts
e. Post-irradiation fuel cooling time and average burnup per Thoria Rod Canister:	A fuel post-irradiation cooling time $\geq$ 18 years and an average burnup $\leq$ 16,000 MWD/MTIHM.
f. Initial heavy metal weight:	$\leq$ 27 kg/canister
g. Fuel cladding O.D.:	$\geq$ 0.412 inches
h. Fuel cladding I.D.:	$\leq$ 0.362 inches
i. Fuel pellet O.D.:	$\leq$ 0.358 inches
j. Active fuel length:	$\leq$ 111 inches
k. Canister weight:	$\leq$ 550 lbs, including fuel

B. Quantity per MPC: Up to one (1) Dresden Unit 1 Thoria Rod Canister plus any combination of damaged fuel assemblies in damaged fuel containers and intact fuel assemblies, up to a total of 68.

C. Fuel assemblies with stainless steel channels are not authorized for loading in the MPC-68.

D. Dresden Unit 1 fuel assemblies (fuel assembly array/class 6x6A, 6x6B, 6x6C, or 8x8A) with one Antimony-Beryllium neutron source are authorized for loading in the MPC-68. The Antimony-Beryllium source material shall be in a water rod location.

Table A.1 (Page 7 of 21)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F

A. Allowable Contents

1. Uranium oxide, BWR intact fuel assemblies, with or without Zircaloy channels. Uranium oxide BWR intact fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6A, 6x6C, 7x7A, or 8x8A and meet the following specifications:

a. Cladding type:	ZR
b. Maximum planar-average initial enrichment:	As specified in Table A.3 for the applicable fuel assembly array/class.
c. Initial maximum rod enrichment:	As specified in Table A.3 for the applicable fuel assembly array/class.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time $\geq$ 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:	$\leq$ 176.2 inches (nominal design)
f. Fuel assembly width:	$\leq$ 5.85 inches (nominal design)
g. Fuel assembly weight:	$\leq$ 400 lbs, including channels

Table A.1 (Page 8 of 21)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (continued)

A. Allowable Contents (continued)

2. Uranium oxide, BWR damaged fuel assemblies, with or without Zircaloy channels, placed in damaged fuel containers. Uranium oxide BWR damaged fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6A, 6x6C, 7x7A, or 8x8A, and meet the following specifications:

a. Cladding type:	ZR
b. Maximum planar-average initial enrichment:	As specified in Table A.3 for the applicable fuel assembly array/class.
c. Initial maximum rod enrichment:	As specified in Table A.3 for the applicable fuel assembly array/class.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time $\geq$ 18 years, an average burnup $\leq$ 30,000 MWD/MTU, and a minimum initial enrichment $\geq$ 1.8 wt% ^{235}U.
e. Fuel assembly length:	$\leq$ 135.0 inches (nominal design)
f. Fuel assembly width:	$\leq$ 4.70 inches (nominal design)
g. Fuel assembly weight:	$\leq$ 550 lbs, including channels

Table A.1 (Page 9 of 21)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (continued)

A. Allowable Contents (continued)

3. Uranium oxide, BWR fuel debris, with or without Zircaloy channels, placed in damaged fuel containers. The original fuel assemblies for the uranium oxide BWR fuel debris shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6A, 6x6C, 7x7A, or 8x8A, and meet the following specifications:

a. Cladding type:	ZR
b. Maximum planar-average initial enrichment:	As specified in Table A.3 for the applicable original fuel assembly array/class.
c. Initial maximum rod enrichment:	As specified in Table A.3 for the applicable original fuel assembly array/class.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time $\geq$ 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:	$\leq$ 550 lbs, including channels

Table A.1 (Page 10 of 21)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (continued)

A. Allowable Contents (continued)

4. Mixed oxide(MOX), BWR intact fuel assemblies, with or without Zircaloy channels. MOX BWR intact fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6B and meet the following specifications:

a. Cladding type:	ZR
b. Maximum planar-average initial enrichment:	As specified in Table A.3 for fuel assembly array/class 6x6B.
c. Initial maximum rod enrichment:	As specified in Table A.3 for fuel assembly array/class 6x6B.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time $\geq$ 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 11 of 21)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (continued)

A. Allowable Contents (continued)

5. Mixed oxide (MOX), BWR damaged fuel assemblies, with or without Zircaloy channels, placed in damaged fuel containers. MOX BWR intact fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6B and meet the following specifications:

a. Cladding type:	ZR
b. Maximum planar-average initial enrichment:	As specified in Table A.3 for array/class 6x6B.
c. Initial maximum rod enrichment:	As specified in Table A.3 for array/class 6x6B.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time $\geq$ 18 years, an average burnup $\leq$ 30,000 MWD/MTIHM, and a minimum initial enrichment $\geq$ 1.8 wt% ^{235}U for the UO_2 rods.
e. Fuel assembly length:	$\leq$ 135.0 inches (nominal design)
f. Fuel assembly width:	$\leq$ 4.70 inches (nominal design)
g. Fuel assembly weight:	$\leq$ 550 lbs, including channels

Table A.1 (Page 12 of 21)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (continued)

A. Allowable Contents (continued)

6. Mixed oxide (MOX), BWR fuel debris, with or without Zircaloy channels, placed in damaged fuel containers. The original fuel assemblies for the MOX BWR fuel debris shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6B and meet the following specifications:

a. Cladding type:	ZR
b. Maximum planar-average initial enrichment:	As specified in Table A.3 for original fuel assembly array/class 6x6B.
c. Initial maximum rod enrichment:	As specified in Table A.3 for original fuel assembly array/class 6x6B.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time $\geq$ 18 years, an average burnup $\leq$ 30,000 MWD/MTIHM, and a minimum initial enrichment $\geq$ 1.8 wt% ^{235}U for the UO_2 rods in the original fuel assembly.
e. Fuel assembly length:	$\leq$ 135.0 inches (nominal design)
f. Fuel assembly width:	$\leq$ 4.70 inches (nominal design)
g. Fuel assembly weight:	$\leq$ 550 lbs, including channels

Table A.1 (Page 13 of 21)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (continued)

A. Allowable Contents (continued)

7. Thoria rods (ThO_2 and UO_2) placed in Dresden Unit 1 Thoria Rod Canisters (as shown in Figure 1.2.11A of the HI-STAR 100 System SAR, Revision 12) and meeting the following specifications:

a. Cladding Type:	ZR
b. Composition:	98.2 wt.% ThO_2, 1.8 wt. % UO_2 with an enrichment of 93.5 wt. % ^{235}U.
c. Number of rods per Thoria Rod Canister:	$\leq$ 18
d. Decay heat per Thoria Rod Canister:	$\leq$ 115 Watts
e. Post-irradiation fuel cooling time and average burnup per Thoria Rod Canister:	A fuel post-irradiation cooling time $\geq$ 18 years and an average burnup $\leq$ 16,000 MWD/MTIHM.
f. Initial heavy metal weight:	$\leq$ 27 kg/canister
g. Fuel cladding O.D.:	$\geq$ 0.412 inches
h. Fuel cladding I.D.:	$\leq$ 0.362 inches
i. Fuel pellet O.D.:	$\leq$ 0.358 inches
j. Active fuel length:	$\leq$ 111 inches
k. Canister weight:	$\leq$ 550 lbs, including fuel

Table A.1 (Page 14 of 21)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (continued)

B. Quantity per MPC:

Up to four (4) damaged fuel containers containing uranium oxide or MOX BWR fuel debris. The remaining MPC-68F fuel storage locations may be filled with array/class 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A fuel assemblies of the following type, as applicable:

1. Uranium oxide BWR intact fuel assemblies;
2. MOX BWR intact fuel assemblies;
3. Uranium oxide BWR damaged fuel assemblies placed in damaged fuel containers;
4. MOX BWR damaged fuel assemblies placed in damaged fuel containers; or
5. Up to one (1) Dresden Unit 1 Thoria Rod Canister.

C. Fuel assemblies with stainless steel channels are not authorized for loading in the MPC-68F.

D. Dresden Unit 1 fuel assemblies (fuel assembly array/class 6x6A, 6x6B, 6x6C or 8x8A) with one Antimony-Beryllium neutron source are authorized for loading in the MPC-68F. The Antimony-Beryllium neutron source material shall be in a water rod location.

Table A.1 (Page 15 of 21)
Fuel Assembly Limits

IV. MPC MODEL: MPC-24E

A. Allowable Contents

1. Uranium oxide, PWR intact fuel assemblies listed in Table A.2 and meeting the following specifications:

a. Cladding type:	ZR or stainless steel (SS) as specified in Table A.2 for the applicable fuel assembly array/class
b. Maximum initial enrichment:	As specified in Table A.2 for the applicable fuel assembly array/class.
c. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly	
i. ZR clad:	Except for Trojan plant fuel, an assembly post-irradiation cooling time, average burnup, and minimum initial enrichment as specified in Table A.4 or A.5, as applicable.
ii. SS clad:	An assembly post-irradiation cooling time, average burnup, and minimum initial enrichment as specified in Table A.6, as applicable.
iii Trojan plant fuel	An assembly post-irradiation cooling time, average burnup, and minimum initial enrichment as specified in Table A.8.
iv Trojan plant non-fuel hardware and neutron sources	Post-irradiation cooling time, and average burnup as specified in Table A.9
d. Decay heat per assembly	
i. ZR Clad:	Except for Trojan plant fuel, decay heat ≤ 833 Watts. Trojan plant fuel decay heat: ≤ 725 Watts
ii. SS Clad:	≤ 488 Watts
e. Fuel assembly length:	≤ 176.8 inches (nominal design)
f. Fuel assembly width:	≤ 8.54 inches (nominal design)
g. Fuel assembly weight:	≤ 1,680 lbs, including non-fuel hardware and neutron sources

Table A.1 (Page 16 of 21)
Fuel Assembly Limits

IV. MPC MODEL: MPC-24E

A. Allowable Contents (continued)

2. Trojan plant damaged fuel assemblies meeting the applicable criteria listed in Table A.2 and meeting the following specifications:

a. Cladding type:	ZR
b. Maximum initial enrichment:	3.7% ^{235}U
c. Fuel assembly post-irradiation cooling time, average burnup, decay heat, and minimum initial enrichment per assembly	An assembly post-irradiation cooling time, average burnup, and initial enrichment as specified in Table A.8 Decay Heat: ≤ 725 Watts
d. Fuel assembly length:	≤ 169.3 inches (nominal design)
e. Fuel assembly width:	≤ 8.43 inches (nominal design)
f. Fuel assembly weight:	≤ 1,680 lbs, including DFC or Failed Fuel Can

B. Quantity per MPC: Up to 24 PWR intact fuel assemblies. For Trojan plant fuel only, up to four (4) damaged fuel assemblies may be stored in fuel storage locations 3, 6, 19, and/or 22. The remaining MPC-24E fuel storage locations may be filled with Trojan plant intact fuel assemblies.

C. Trojan plant fuel must be transported in the custom-designed Trojan MPCs with the MPC spacer installed. Fuel from other plants is not permitted to be transported in the Trojan MPCs.

D. Except for Trojan plant fuel, the fuel assemblies shall not contain non-fuel hardware or neutron sources. Trojan intact fuel assemblies containing non-fuel hardware may be transported in any fuel storage location.

E. Trojan plant damaged fuel assemblies must be transported in a Trojan Failed Fuel Can or a Holtec damaged fuel container designed for Trojan Plant fuel.

F. One (1) Trojan plant Sb-Be and /or up to two (2) Cf neutron sources in a Trojan plant intact fuel assembly (one source fuel assembly) may be transported in any one MPC. Each fuel assembly neutron source may be transported in any fuel storage location.

G. Fuel debris is not authorized for transport in the MPC-24E.

H. Trojan plant non-fuel hardware and neutron sources may not be transported in the same fuel storage location as a damaged fuel assembly.

Table A.1 (Page 17 of 21)
Fuel Assembly Limits

V. MPC MODEL: MPC-24EF

A. Allowable Contents

1. Uranium oxide, PWR intact fuel assemblies listed in Table A.2 and meeting the following specifications:

a. Cladding type:	ZR or stainless steel (SS) as specified in Table A.2 for the applicable fuel assembly array/class.
b. Maximum initial enrichment:	As specified in Table A.2 for the applicable fuel assembly array/class.
c. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly	
i. ZR clad:	Except for Trojan plant fuel, an assembly post-irradiation cooling time, average burnup, and minimum initial enrichment as specified in Table A.4 or A.5, as applicable.
ii. SS clad:	An assembly post-irradiation cooling time, average burnup, and minimum initial enrichment as specified in Table A.6, as applicable.
iii Trojan plant fuel:	An assembly post-irradiation cooling time, average burnup, and minimum initial enrichment as specified in Table A.8.
iv Trojan plant non-fuel hardware and neutron sources:	Post-irradiation cooling time, and average burnup as specified in Table A.9.
d. Decay heat per assembly:	
a. ZR Clad:	Except for Trojan plant fuel, decay heat ≤ 833 Watts. Trojan plant fuel decay heat: ≤ 725 Watts.
b. SS Clad:	≤ 488 Watts
e. Fuel assembly length:	≤ 176.8 inches (nominal design)
f. Fuel assembly width:	≤ 8.54 inches (nominal design)
g. Fuel assembly weight:	≤ 1,680 lbs, including non-fuel hardware and neutron sources.

Table A.1 (Page 18 of 21)
Fuel Assembly Limits

V. MPC MODEL: MPC-24EF

A. Allowable Contents (continued)

2. Trojan plant damaged fuel assemblies meeting the applicable criteria listed in Table A.2 and meeting the following specifications:

a. Cladding type:	ZR
b. Maximum initial enrichment:	3.7% ^{235}U
c. Fuel assembly post-irradiation cooling time, average burnup, decay heat, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time, average burnup, and initial enrichment as specified in Table A.8. Decay Heat: ≤ 725 Watts
d. Fuel assembly length:	≤ 169.3 inches (nominal design)
e. Fuel assembly width:	≤ 8.43 inches (nominal design)
f. Fuel assembly weight:	≤ 1,680 lbs, including DFC or Failed Fuel Can.

Table A.1 (Page 19 of 21)
Fuel Assembly Limits

V. MPC MODEL: MPC-24EF

A. Allowable Contents (continued)

3. Trojan Fuel Debris Process Can Capsules and/or Trojan plant fuel assemblies classified as fuel debris, for which the original fuel assemblies meet the applicable criteria listed in Table A.2 and meet the following specifications:

a. Cladding type:	ZR
b. Maximum initial enrichment:	3.7% ^{235}U
c. Fuel debris post-irradiation cooling time, average burnup, decay heat, and minimum initial enrichment per assembly:	Post-irradiation cooling time, average burnup, and initial enrichment as specified in Table A.8. Decay Heat: ≤ 725 Watts
d. Fuel assembly length:	≤ 169.3 inches (nominal design)
e. Fuel assembly width:	≤ 8.43 inches (nominal design)
f. Fuel assembly weight:	≤ 1,680 lbs, including DFC or Failed Fuel Can.

B. Quantity per MPC: Up to 24 PWR intact fuel assemblies. For Trojan plant fuel only, up to four (4) damaged fuel assemblies, fuel assemblies classified as fuel debris, and/or Trojan Fuel Debris Process Can Capsules may be stored in fuel storage locations 3, 6, 19, and/or 22. The remaining MPC-24EF fuel storage locations may be filled with Trojan plant intact fuel assemblies.

C. *Trojan plant fuel must be transported in the custom-designed* Trojan MPCs with the MPC spacer installed. Fuel from other plants is not permitted to be transported in the Trojan MPCs.

D. Except for Trojan plant fuel, the fuel assemblies shall not contain non-fuel hardware or neutron sources. Trojan intact fuel assemblies containing non-fuel hardware may be transported in any fuel storage location.

E. Trojan plant damaged fuel assemblies, fuel assemblies classified as fuel debris, and Fuel Debris Process Can Capsules must be transported in a Trojan Failed Fuel Can or a Holtec damaged fuel container designed for Trojan Plant fuel.

F. One (1) Trojan plant Sb-Be and /or up to two (2) Cf neutron sources in a Trojan plant intact fuel assembly (one source fuel assembly) may be transported in any one MPC. Each fuel assembly neutron source may be transported in any fuel storage location.

G. Trojan plant non-fuel hardware and neutron sources may not be transported in the same fuel storage location as a damaged fuel assembly.

Table A.1 (Page 20 of 21)
Fuel Assembly Limits

VI. MPC MODEL: MPC-32

A. Allowable Contents

1. Uranium oxide, PWR intact fuel assemblies in array/classes 15x15D, E, F, and H and 17x17A, B, and C listed in Table A.2 and meeting the following specifications:

a. Cladding type:	ZR
b. Maximum initial enrichment:	As specified in Table A.2 for the applicable fuel assembly array/class.
c. Post-irradiation cooling time, maximum average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time, average burnup, and minimum initial enrichment as specified in Table A.10 or A.11, as applicable.
d. Minimum average burnup per assembly (Assembly Burnup shall be confirmed per Subsection 1.2.3.7.2 of the SAR, which is hereby included by reference)	Calculated value as a function of initial enrichment. See Table A.12.
e. Decay heat per assembly:	≤ 625 Watts
f. Fuel assembly length:	≤ 176.8 inches (nominal design)
g. Fuel assembly width:	≤ 8.54 inches (nominal design)
h. Fuel assembly weight:	≤ 1,680 lbs

i. Operating parameters during irradiation of the assembly (Assembly operating parameters shall be determined per Subsection 1.2.3.7.1 of the SAR, which is hereby included by reference)

Core ave. soluble boron concentration:	≤ 1,000 ppmb
Assembly ave. moderator temperature:	≤ 601 K for array/classes 15x15D, E, F, and H ≤ 610 K for array/classes 17x17A, B, and C
Assembly ave. specific power:	≤ 47.36 kW/kg-U for array/classes 15x15D, E, F, and H ≤ 61.61 kW/kg-U for array/classes 17x17A, B, and C

Table A.1 (Page 21 of 21)
Fuel Assembly Limits

VI. MPC MODEL: MPC-32 (continued)

B. Quantity per MPC: Up to 32 PWR intact fuel assemblies.

C. Fuel assemblies shall not contain non-fuel hardware.

D. Damaged fuel assemblies and fuel debris are not authorized for transport in MPC-32.

E. Trojan plant fuel is not permitted to be transported in the MPC-32.

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

Fuel Assembly Array/Class	**14x14A**	**14x14B**	**14x14C**	**14x14D**	**14x14E**
Clad Material (Note 2)	ZR	ZR	ZR	SS	Zr
Design Initial U (kg/assy.) (Note 3)	≤ 407	≤ 407	≤ 425	≤ 400	≤ 206
Initial Enrichment (MPC-24, 24E, and 24EF) (wt % ^{235}U)	≤ 4.6 (24) ≤ 5.0 (24E/EF)	≤ 4.6 (24) ≤ 5.0 (24E/EF)	≤ 4.6 (24) ≤ 5.0 (24E/EF)	≤ 4.0 (24) ≤ 5.0 (24E/EF)	≤ 5.0
No. of Fuel Rod Locations	179	179	176	180	173
Fuel Clad O.D. (in.)	≥ 0.400	≥ 0.417	≥ 0.440	≥ 0.422	≥ 0.3415
Fuel Clad I.D. (in.)	≤ 0.3514	≤ 0.3734	≤ 0.3880	≤ 0.3890	≤ 0.3175
Fuel Pellet Dia. (in.)	≤ 0.3444	≤ 0.3659	≤ 0.3805	≤ 0.3835	≤ 0.3130
Fuel Rod Pitch (in.)	≤ 0.556	≤ 0.556	≤ 0.580	≤ 0.556	Note 6
Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 144	≤ 102
No. of Guide Tubes	17	17	5 (Note 4)	16	0
Guide Tube Thickness (in.)	≥ 0.017	≥ 0.017	≥ 0.038	≥ 0.0145	N/A

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

Fuel Assembly Array/Class	15x15A	15x15B	15x15C	15x15D	15x15E	15x15F
Clad Material (Note 2)	ZR	ZR	ZR	ZR	ZR	ZR
Design Initial U (kg/assy.) (Note 3)	≤ 464	≤ 464	≤ 464	≤ 475	≤ 475	≤ 475
Initial Enrichment (MPC-24, 24E, and 24EF) (wt % ^{235}U)	≤ 4.1 (24) ≤ 4.5 (24E/EF)	≤ 4.1 (24) ≤ 4.5 (24E/EF)	≤ 4.1 (24) ≤ 4.5 (24E/EF)	≤ 4.1 (24) ≤ 4.5 (24E/EF)	≤ 4.1 (24) ≤ 4.5 (24E/EF)	≤ 4.1 (24) ≤ 4.5 (24E/EF)
Initial Enrichment (MPC-32) (wt. % ^{235}U) (Note 5)	N/A	N/A	N/A	(Note 5)	(Note 5)	(Note 5)
No. of Fuel Rod Locations	204	204	204	208	208	208
Fuel Clad O.D. (in.)	≥ 0.418	≥ 0.420	≥ 0.417	≥ 0.430	≥ 0.428	≥ 0.428
Fuel Clad I.D. (in.)	≤ 0.3660	≤ 0.3736	≤ 0.3640	≤ 0.3800	≤ 0.3790	≤ 0.3820
Fuel Pellet Dia. (in.)	≤ 0.3580	≤ 0.3671	≤ 0.3570	≤ 0.3735	≤ 0.3707	≤ 0.3742
Fuel Rod Pitch (in.)	≤ 0.550	≤ 0.563	≤ 0.563	≤ 0.568	≤ 0.568	≤ 0.568
Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Guide and/or Instrument Tubes	21	21	21	17	17	17
Guide/Instrument Tube Thickness (in.)	≥ 0.015	≥ 0.015	≥ 0.0165	≥ 0.0150	≥ 0.0140	≥ 0.0140

Table A.2 (Page 3 of 4)
PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/ Class	**15x15G**	**15x15H**	**16x16A**	**17x17A**	**17x17B**	**17x17C**
Clad Material (Note 2)	SS	ZR	ZR	ZR	ZR	ZR
Design Initial U (kg/assy.) (Note 3)	≤ 420	≤ 475	≤ 443	≤ 467	≤ 467	≤ 474
Initial Enrichment (MPC-24, 24E, and 24EF) (wt % ^{235}U)	≤ 4.0 (24) ≤ 4.5 (24E/EF)	≤ 3.8 (24) ≤ 4.2 (24E/EF)	≤ 4.6 (24) ≤ 5.0 (24E/EF)	≤ 4.0 (24) ≤ 4.4 (24E/EF)	≤ 4.0 (24) ≤ 4.4 (24E/EF) (Note 7)	≤ 4.0 (24) ≤ 4.4 (24E/EF)
Initial Enrichment (MPC-32) (wt. % ^{235}U) (Note 5)	N/A	(Note 5)	N/A	(Note 5)	(Note 5)	(Note 5)
No. of Fuel Rod Locations	204	208	236	264	264	264
Fuel Clad O.D. (in.)	≥ 0.422	≥ 0.414	≥ 0.382	≥ 0.360	≥ 0.372	≥ 0.377
Fuel Clad I.D. (in.)	≤ 0.3890	≤ 0.3700	≤ 0.3320	≤ 0.3150	≤ 0.3310	≤ 0.3330
Fuel Pellet Dia. (in.)	≤ 0.3825	≤ 0.3622	≤ 0.3255	≤ 0.3088	≤ 0.3232	≤ 0.3252
Fuel Rod Pitch (in.)	≤ 0.563	≤ 0.568	≤ 0.506	≤ 0.496	≤ 0.496	≤ 0.502
Active Fuel Length (in.)	≤ 144	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Guide and/or Instrument Tubes	21	17	5 (Note 4)	25	25	25
Guide/Instrument Tube Thickness (in.)	≥ 0.0145	≥ 0.0140	≥ 0.0400	≥ 0.016	≥ 0.014	≥ 0.020

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

Notes:

1. All dimensions are design nominal values. Maximum and minimum dimensions are specified to bound variations in design nominal values among fuel assemblies within a given array/class.

2. ZR Designates cladding material made of Zirconium or Zirconium alloys.

3. Design initial uranium weight is the nominal uranium weight specified for each assembly by the fuel manufacturer or reactor user. For each PWR fuel assembly, the total uranium weight limit specified in this table may be increased up to 2.0 percent for comparison with users' fuel records to account for manufacturer tolerances.

4. Each guide tube replaces four fuel rods.

5. Minimum burnup and maximum initial enrichment as specified in Table A.12.

6. This fuel assembly array/class includes only the Indian Point Unit 1 fuel assembly. This fuel assembly has two pitches in different sectors of the assembly. These pitches are 0.441 inches and 0.453 inches

7. Trojan plant-specific fuel is governed by the limits specified for array/class 17x17B and will be transported in the custom-designed Trojan MPC-24E/EF canisters. The Trojan MPC-24E/EF design is authorized to transport only Trojan plant fuel with a maximum initial enrichment of 3.7 wt.% ^{235}U.

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

Fuel Assembly Array/Class	**6x6A**	**6x6B**	**6x6C**	**7x7A**	**7x7B**	**8x8A**
Clad Material (Note 2)	ZR	ZR	ZR	ZR	ZR	ZR
Design Initial U (kg/assy.) (Note 3)	≤ 110	≤ 110	≤ 110	≤ 100	≤ 195	≤ 120
Maximum planar-average initial enrichment (wt.% ^{235}U)	≤ 2.7	≤ 2.7 for the UO_2 rods. See Note 4 for MOX rods	≤ 2.7	≤ 2.7	≤ 4.2	≤ 2.7
Initial Maximum Rod Enrichment (wt.% ^{235}U)	≤ 4.0	≤ 4.0	≤ 4.0	≤ 5.5	≤ 5.0	≤ 4.0
No. of Fuel Rod Locations	35 or 36	35 or 36 (up to 9 MOX rods)	36	49	49	63 or 64
Fuel Clad O.D. (in.)	≥ 0.5550	≥ 0.5625	≥ 0.5630	≥ 0.4860	≥ 0.5630	≥ 0.4120
Fuel Clad I.D. (in.)	≤ 0.5105	≤ 0.4945	≤ 0.4990	≤ 0.4204	≤ 0.4990	≤ 0.3620
Fuel Pellet Dia. (in.)	≤ 0.4980	≤ 0.4820	≤ 0.4880	≤ 0.4110	≤ 0.4910	≤ 0.3580
Fuel Rod Pitch (in.)	≤ 0.710	≤ 0.710	≤ 0.740	≤ 0.631	≤ 0.738	≤ 0.523
Active Fuel Length (in.)	≤ 120	≤ 120	≤ 77.5	≤ 80	≤ 150	≤ 120
No. of Water Rods (Note 11)	1 or 0	1 or 0	0	0	0	1 or 0
Water Rod Thickness (in.)	≥ 0	≥ 0	N/A	N/A	N/A	≥ 0
Channel Thickness (in.)	≤ 0.060	≤ 0.060	≤ 0.060	≤ 0.060	≤ 0.120	≤ 0.100

Table A.3 (Page 2 of 5)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	**8x8B**	**8x8C**	**8x8D**	**8x8E**	**8x8F**	**9x9A**
Clad Material (Note 2)	ZR	ZR	ZR	ZR	ZR	ZR
Design Initial U (kg/assy.) (Note 3)	≤ 185	≤ 185	≤ 185	≤ 185	≤ 185	≤ 177
Maximum planar-average initial enrichment (wt.% ^{235}U)	≤ 4.2	≤ 4.2	≤ 4.2	≤ 4.2	< 4.0	≤ 4.2
Initial Maximum Rod Enrichment (wt.% ^{235}U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0
No. of Fuel Rod Locations	63 or 64	62	60 or 61	59	64	74/66 (Note 5)
Fuel Clad O.D. (in.)	≥ 0.4840	≥ 0.4830	≥ 0.4830	≥ 0.4930	≥ 0.4576	≥ 0.4400
Fuel Clad I.D. (in.)	≤ 0.4295	≤ 0.4250	0.4230	≤ 0.4250	≤ 0.3996	≤ 0.3840
Fuel Pellet Dia. (in.)	≤ 0.4195	≤ 0.4160	≤ 0.4140	≤ 0.4160	≤ 0.3913	≤ 0.3760
Fuel Rod Pitch (in.)	≤ 0.642	≤ 0.641	≤ 0.640	≤ 0.640	≤ 0.609	≤ 0.566
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Water Rods (Note 11)	1 or 0	2	1 - 4 (Note 7)	5	N/A (Note 12)	2
Water Rod Thickness (in.)	≥ 0.034	> 0.00	> 0.00	≥ 0.034	≥ 0.0315	> 0.00
Channel Thickness (in.)	≤ 0.120	≤ 0.120	≤ 0.120	≤ 0.100	≤ 0.055	≤ 0.120

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

Fuel Assembly Array/Class	**9x9B**	**9x9C**	**9x9D**	**9x9E (Note 13)**	**9x9F (Note 13)**	**9x9G**
Clad Material (Note 2)	ZR	ZR	ZR	ZR	ZR	ZR
Design Initial U (kg/assy.) (Note 3)	≤ 177	≤ 177	≤ 177	≤ 177	≤ 177	≤ 177
Maximum planar-average initial enrichment (wt.% ^{235}U)	≤ 4.2	≤ 4.2	≤ 4.2	≤ 4.0	≤ 4.0	≤ 4.2
Initial Maximum Rod Enrichment (wt.% ^{235}U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0
No. of Fuel Rods	72	80	79	76	76	72
Fuel Clad O.D. (in.)	≥ 0.4330	≥ 0.4230	≥ 0.4240	≥ 0.4170	≥ 0.4430	≥ 0.4240
Fuel Clad I.D. (in.)	≤ 0.3810	≤ 0.3640	≤ 0.3640	≤ 0.3640	≤ 0.3860	≤ 0.3640
Fuel Pellet Dia. (in.)	≤ 0.3740	≤ 0.3565	≤ 0.3565	≤ 0.3530	≤ 0.3745	≤ 0.3565
Fuel Rod Pitch (in.)	≤ 0.572	≤ 0.572	≤ 0.572	≤ 0.572	≤ 0.572	≤ 0.572
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Water Rods (Note 11)	1 (Note 6)	1	2	5	5	1 (Note 6)
Water Rod Thickness (in.)	> 0.00	≥ 0.020	≥ 0.0300	≥ 0.0120	≥ 0.0120	≥ 0.0320
Channel Thickness (in.)	≤ 0.120	≤ 0.100	≤ 0.100	≤ 0.120	≤ 0.120	≤ 0.120

Table A.3 (Page 4 of 5)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	**10x10A**	**10x10B**	**10x10C**	**10x10D**	**10x10E**
Clad Material (Note 2)	ZR	ZR	ZR	SS	SS
Design Initial U (kg/assy.) (Note 3)	≤ 186	≤ 186	≤ 186	≤ 125	≤ 125
Maximum planar-average initial enrichment (wt.% ^{235}U)	≤ 4.2	≤ 4.2	≤ 4.2	≤ 4.0	≤ 4.0
Initial Maximum Rod Enrichment (wt.% ^{235}U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0
No. of Fuel Rod Locations	92/78 (Note 8)	91/83 (Note 9)	96	100	96
Fuel Clad O.D. (in.)	≥ 0.4040	≥ 0.3957	≥ 0.3780	≥ 0.3960	≥ 0.3940
Fuel Clad I.D. (in.)	≤ 0.3520	≤ 0.3480	≤ 0.3294	≤ 0.3560	≤ 0.3500
Fuel Pellet Dia. (in.)	≤ 0.3455	≤ 0.3420	≤ 0.3224	≤ 0.3500	≤ 0.3430
Fuel Rod Pitch (in.)	≤ 0.510	≤ 0.510	≤ 0.488	≤ 0.565	≤ 0.557
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 83	≤ 83
No. of Water Rods (Note 11)	2	1 (Note 6)	5 (Note 10)	0	4
Water Rod Thickness (in.)	≥ 0.0300	> 0.00	≥ 0.031	N/A	≥ 0.022
Channel Thickness (in.)	≤ 0.120	≤ 0.120	≤ 0.055	≤ 0.080	≤ 0.080

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

Notes:

1. All dimensions are design nominal values. Maximum and minimum dimensions are specified to bound variations in design nominal values among fuel assemblies within a given array/class.

2. ZR designates cladding material made from Zirconium or Zirconium alloys.

3. Design initial uranium weight is the uranium weight specified for each assembly by the fuel manufacturer or reactor user. For each BWR fuel assembly, the total uranium weight limit specified in this table may be increased up to 1.5% for comparison with users' fuel records to account for manufacturer's tolerances.

4. $\leq$ 0.635 wt. % ^{235}U and $\leq$ 1.578 wt. % total fissile plutonium (^{239}Pu and ^{241}Pu), (wt. % of total fuel weight, i.e., UO_2 plus PuO_2).

5. This assembly class contains 74 total fuel rods; 66 full length rods and 8 partial length rods.

6. Square, replacing nine fuel rods.

7. Variable

8. This assembly class contains 92 total fuel rods; 78 full length rods and 14 partial length rods.

9. This assembly class contains 91 total fuel rods, 83 full length rods and 8 partial length rods.

10. One diamond-shaped water rod replacing the four center fuel rods and four rectangular water rods dividing the assembly into four quadrants.

11. These rods may be sealed at both ends and contain Zr material in lieu of water.

12. This assembly is known as "QUAD+" and has four rectangular water cross segments dividing the assembly into four quadrants.

13. For the SPC 9x9-5 fuel assembly, each fuel rod must meet either the 9x9E or 9x9F set of limits for clad O.D., clad I.D., and pellet diameter.

Table A.4

FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND MINIMUM ENRICHMENT
MPC-24/24E/24/EF PWR FUEL WITH ZIRCALOY CLAD AND
WITH NON-ZIRCALOY IN-CORE GRID SPACERS

Post-irradiation Cooling Time (years)	**Assembly Burnup (MWD/MTU)**	**Assembly Initial Enrichment (wt. % U-235)**
≥ 9	≤ 24,500	≥ 2.3
≥ 11	≤ 29,500	≥ 2.6
≥ 13	≤ 34,500	≥ 2.9
≥ 15	≤ 39,500	≥ 3.2
≥ 18	≤ 44,500	≥ 3.4

Table A.5

FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND MINIMUM ENRICHMENT
MPC-24/24E/24EF PWR FUEL WITH ZIRCALOY CLAD AND
WITH ZIRCALOY IN-CORE GRID SPACERS

Post-irradiation Cooling Time (years)	**Assembly Burnup (MWD/MTU)**	**Assembly Initial Enrichment (wt. % U-235)**
≥ 6	≤ 24,500	≥ 2.3
≥ 7	≤ 29,500	≥ 2.6
≥ 9	≤ 34,500	≥ 2.9
≥ 11	≤ 39,500	≥ 3.2
≥ 14	≤ 44,500	≥ 3.4

Table A.6

FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND MINIMUM ENRICHMENT
MPC-24/24E/24EF PWR FUEL WITH STAINLESS STEEL CLAD

Post-irradiation Cooling Time (years)	Assembly Burnup (MWD/MTU)	Assembly Initial Enrichment (wt. % U-235)
≥ 19	≤ 30,000	≥ 3.1
≥ 24	≤ 40,000	≥ 3.1

Table A.7

FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND MINIMUM ENRICHMENT
MPC-68

Post-irradiation Cooling Time (years)	Assembly Burnup (MWD/MTU)	Assembly Initial Enrichment (wt. % U-235)
≥ 8	≤ 24,500	≥ 2.1
≥ 9	≤ 29,500	≥ 2.4
≥ 11	≤ 34,500	≥ 2.6
≥ 14	≤ 39,500	≥ 2.9
≥ 19	≤ 44,500	≥ 3.0

Table A.8

TROJAN PLANT FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND MINIMUM ENRICHMENT LIMITS (Note 1)

Post-irradiation Cooling Time (years)	Assembly Burnup (MWD/MTU)	Assembly Initial Enrichment (wt.% ^{235}U)
≥16	≤42,000	≥3.09
≥16	≤37,500	≥2.6
≥16	≤30,000	≥2.1

NOTES:

1. Each fuel assembly must only meet one set of limits (i.e., one row)

Table A.9

TROJAN PLANT NON-FUEL HARDWARE AND NEUTRON SOURCES COOLING AND BURNUP LIMITS

Type of Hardware or Neutron Source	Burnup (MWD/MTU)	Post-irradiation Cooling Time (Years)
BPRAs	≤15,998	≥24
TPDs	≤118,674	≥11
RCCAs	≤125,515	≥9
Cf neutron source	≤15,998	≥24
Sb-Be neutron source with 4 source rods, 16 burnable poison rods, and 4 thimble plug rods	≤45,361	≥19
Sb-Be neutron source with 4 source rods, 20 thimble plug rods	≤88,547	≥9

Table A.10

FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND MINIMUM ENRICHMENT MPC-32 PWR FUEL WITH ZIRCALOY CLAD AND WITH NON-ZIRCALOY IN-CORE GRID SPACERS

Post-irradiation cooling time (years)	Assembly burnup (MWD/MTU)	Assembly Initial Enrichment (wt. % U-235)
≥12	≤24,500	≥2.3
≥14	≤29,500	≥2.6
≥16	≤34,500	≥2.9
≥19	≤39,500	≥3.2
≥20	≤42,500	≥3.4

Table A.11

FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND MINIMUM ENRICHMENT MPC-32 PWR FUEL WITH ZIRCALOY CLAD AND WITH ZIRCALOY IN-CORE GRID SPACERS

Post-irradiation cooling time (years)	Assembly burnup (MWD/MTU)	Assembly Initial Enrichment (wt.% U-235)
≥8	≤24,500	≥2.3
≥9	≤29,500	≥2.6
≥12	≤34,500	≥2.9
≥14	≤39,500	≥3.2
≥19	≤44,500	≥3.4

Table A.12

FUEL ASSEMBLY MAXIMUM ENRICHMENT AND MINIMUM BURNUP REQUIREMENTS FOR TRANSPORTATION IN MPC-32

Fuel Assembly Array/Class	**Configuration (Note 2)**	**Maximum Enrichment (wt.% U-235)**	**Minimum Burnup (B) as a Function of Initial Enrichment (E) (Note 1) (GWD/MTU)**
15x15D, E, F, H	A	4.65	$B = (1.6733)*E^3-(18.72)*E^2+(80.5967)*E-88.3$
	B	4.38	$B = (2.175)*E^3-(23.355)*E^2+(94.77)*E-99.95$
	C	4.48	$B = (1.9517)*E^3(21.45)*E^2+(89.1783)*E-94.6$
	D	4.45	$B = (1.93)*E^3-(21.095)*E^2+(87.785)*E-93.06$
17x17A,B,C	A	4.49	$B = (1.08)*E^3-(12.25)*E^2+(60.13)*E-70.86$
	B	4.04	$B = (1.1)*E^3-(11.56)*E^2+(56.6)*E-62.59$
	C	4.28	$B = (1.36)*E^3-(14.83)*E^2+(62.27)*E-72.93$
	D	4.16	$B = (1.4917)*E^3-(16.26)*E^2+(72.9883)*E-79.7$

NOTES:

1. E = Initial enrichment (e.g., for 4.05 wt.%, E = 4.05).
2. See Table A.13.
3. Fuel Assemblies must be cooled 5 years or more.

Table A.13

LOADING CONFIGURATIONS FOR THE MPC-32

CONFIGURATION	ASSEMBLY SPECIFICATIONS
A	• Assemblies that have not been located in any cycle under a control rod bank that was permitted to be inserted during full power operation (per plant operating procedures); or • Assemblies that have been located under a control rod bank that was permitted to be inserted during full power operation (per plant operating procedures), but where it can be demonstrated, based on operating records, that the insertion never exceeded 8 inches from the top of the active length during full power operation.
B	• Of the 32 assemblies in a basket, up to 8 assemblies can be from core locations where they were located under a control rod bank, that was permitted to be inserted more than 8 inches during full power operation. There is no limit on the duration (in terms of burnup) under this bank. • The remaining assemblies in the basket must satisfy the same conditions as specified for configuration A.
C	• Of the 32 assemblies in a basket, up to 8 assemblies can be from core locations where they were located under a control rod bank, that was permitted to be inserted more than 8 inches during full power operation. Location under such a control rod bank is limited to 20 GWD/MTU of the assembly. • The remaining assemblies in the basket must satisfy the same conditions as specified for configuration A.
D	• Of the 32 assemblies in a basket, up to 8 assemblies can be from core locations where they were located under a control rod bank, that was permitted to be inserted more than 8 inches during full power operation. Location under such a control rod bank is limited to 30 GWD/MTU of the assembly. • The remaining assemblies in the basket must satisfy the same conditions as specified for configuration A.

REFERENCES:

Holtec International Report No. HI-951251, *Safety Analysis Report for the Holtec International Storage, Transport, And Repository Cask System (HI-STAR 100 Cask System)*, Revision 12, dated October 6, 2006.

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
9263	4	71-9263	USA/9263/B(U)-96	1	OF	3

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

Source Production and
Equipment Company, Inc.
113 Teal Street
St. Rose, LA 70087

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Source Production and Equipment Company, Inc.
Application dated April 22, 1999,
as supplemented.

4. CONDITIONS

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

5.

(a) Packaging

(1) Model No.: SPEC-150

(2) Description

A welded titanium encased, uranium shielded, radiographic exposure device. Primary components consist of an outer titanium shell, internal supports, depleted uranium shield, and a titanium, titanium alloy or zircalloy S-tube. The contents are securely positioned in the S-tube by a source cable lock assembly and source safety plug assembly. The unit resembles a rectangular box approximately 5.4 inches wide, 5.6 inches high and 14.5 inches long. The maximum weight of the package is 53 pounds.

(3) Drawings

The packaging is constructed and assembled in accordance with Source Production and Equipment Company, Inc. Drawing Nos. 15B000, Rev. 6; 15B001-3, Rev. 2; 15B002A, Rev. 5; 15B008, Rev. 4; 15B625, Rev. 1; 19B005, Rev. 0; 19B006, Rev. 0; and 190909, Rev. 0.

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9263	4	71-9263	USA/9263/B(U)-96	2	OF	3

(b) Contents

(1) Type and form of material

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

(2) Maximum quantity of material per package

150 curies (output)

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

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

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

Packagings may be marked with Package Identification Number USA/9263/B(U)-85 until April 30, 2006, and must be marked with Package Identification Number USA/9263/B(U)-96 after April 30, 2006.

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

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

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

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

10. Expiration date: June 30, 2010.

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9263	4	71-9263	USA/9263/B(U)-96	3	OF	3

REFERENCES

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

Supplements dated: May 6, 1999; March 22, June 6, and June 19, 2000; March 28, 2005.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

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

Date: 26 April 2005

NRC FORM 618 (8-2000) 10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9269	5	71-9269	USA/9269/B(U)-96	1	OF	3

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

QSA Global, Inc.
40 North Avenue
Burlington, MA 01803

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

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

4. CONDITIONS

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

5.
(a) Packaging

(1) Model No.: 650L

(2) Description

A welded stainless steel encased, uranium shielded, Iridium-192 or Selenium-75 source changer. Primary components consist of a steel or stainless steel housing, internal supports, depleted uranium shield, and a titanium "U" tube. The tube is crimped in the middle of the "U" to provide a positive stop for the source assembly. Additionally, the Model No. 650L has two source locking assemblies mounted on the top cover plate. These assemblies are used to secure the radioactive source in a shielded position during transport. The unit resembles a rectangular box approximately 10-inches long, 13.25-inches high and 8.25-inches wide. The maximum weight of the package is 90 pounds.

(3) Drawings

The packaging is constructed in accordance with the AEA Technology/QSA Inc., Drawing No. R65006, Rev. H, Sheets 1-4.

(b) Contents

(1) Type and form of material

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

Selenium-75 as sealed sources which meet the requirements of special form radioactive material.

5. (b) Contents (continued)

(2) Maximum quantity of material per package

Ir-192: 240 curies (8.9 TBq) (output)

Se-75: 300 curies (11.1 TBq) (output)

Output curies are determined by measuring the source output at 1 meter and expressing its activity in curies derived from the following: 0.48 R/h-Ci Iridium-192 at 1 meter (Ref: American National Standard N432-1980, "Radiological Safety for the Design and Construction of Apparatus for Gamma Radiography") and 0.2 R/h-Ci Selenium-75 at 1 meter (Ref: U.S. Public Health Service, Bureau of Radiological Health, 1970: Radiological Health Handbook, rev. ed, Rockville, MD).

6. The source shall be secured in the shielded position of the packaging by the source assembly. The source assembly must be fabricated of materials capable of resisting a 1475°F fire environment for one-half hour and maintaining its positioning function. The cable of the source assembly must engage the source hold-down assembly. The flexible cable of the source assembly must be of sufficient length and diameter to provide positive positioning of the source at the crimp of the "U" tube.

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

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

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

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

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

10. Revision No. 4 of this certificate may be used until July 31, 2007.

11. Expiration date: November 30, 2010.

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9269	5	71-9269	USA/9269/B(U)-96	3	OF	3

REFERENCES

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

Supplements dated November 19, 1999, October 2 and October 31, 2000, July 8, 2005, and March 1, June 6, June 30 and July 14, 2006.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Christopher M. Regan, Acting Chief
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: August 3, 2006

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
9270	3	71-9270	USA/9270/B(U)F-96	1	OF	20

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*
NAC International, Inc.
3930 East Jones Bridge Rd.
Norcross, Georgia 30092

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION
NAC International, Inc. application dated April 30, 1997, as supplemented

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

5.
(a) Packaging

(1) Model No.: UMS Universal Transport Cask Package

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

The UMS is a canister-based system for the storage and transportation of spent nuclear fuel. The transportation component of the UMS system, designated the Universal Transport System, consists of a Universal Transport cask body with a closure lid and energy-absorbing impact limiters loaded with a Transportable Storage Canister (TSC) containing either spent Pressurized Water Reactor (PWR) or Boiling Water Reactor (BWR) nuclear fuel or Maine Yankee site specific contents including Greater than Class C (GTCC) waste.

The NAC-UMS is designed to transport up to 24 intact PWR spent fuel assemblies, 56 intact BWR spent fuel assemblies, GTCC waste, or site specific spent nuclear fuel with associated component hardware. Based on the length of the fuel assemblies, PWR fuels are grouped into three classes (Classes 1 through 3), and BWR fuels are grouped into two classes (Classes 4 and 5). Class 1 and 2 PWR fuel assemblies include non-fuel-bearing inserts (components which include thimble plugs and burnable poison rods installed in the guide tubes). Class 4 and 5 BWR fuel assemblies include the zirconium alloy channels. The loading of site specific fuels that include control component hardware may require the use of a TSC that is longer than if the hardware were excluded. The spent fuel is loaded into a TSC which contains a stainless steel grid work referred to as a basket.

The cask body of the UMS is a right-circular cylinder of multi wall construction which consists of 304 stainless steel inner and outer shells separated by lead gamma radiation shielding which is poured in place. The inner and outer shells are welded to a 304 stainless steel top forging which mates to the cask lid. The inner shell is also welded to a 304 stainless steel bottom forging and the outer shell is welded to the bottom plate. The cask bottom consists of the bottom forging and bottom plate with neutron shield material sandwiched between them. Layers of 4.5 inches thick 304

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

stainless steel ring and two 0.75 inch stainless steel disks are located at the bottom lead annulus between the bottom forging and the outer shell.

Neutron shield material is also placed in an annulus that surrounds the cask outer shell along the length of the cask cavity and is enclosed by a stainless steel shell with top and bottom plates. The neutron shield material is a solid synthetic polymer (NS-4-FR). Twenty-four bonded copper and Type 304 stainless steel fins are located in the radial neutron shield to enhance the heat rejection capability of the cask and to support the neutron shield shell and end plates.

The containment boundary of the UMS consists of the inner shell; bottom forging; top forging; cask lid and lid inner O-ring; vent port cover plate and vent port cover plate inner O-ring; and drain port cover plate and drain port cover plate inner O-ring.

There are five TSCs of different lengths, each to accommodate a different class of PWR or BWR fuel assembly. Each TSC has an outside diameter of about 67 inches and the lengths vary from about 175 to 192 inches long. The TSC assembly consists of a right circular cylindrical shell with a welded bottom plate, a fuel basket, a shield lid, two penetration port covers, and a structural lid. The TSC contains the basket and fuel assemblies or GTCC waste. Spacers are placed below each Class 1, 2, 4 or 5 canisters to locate and support the canister in the cask cavity.

The spacers are free standing structures that are confined in place by the bottom of the canister and the cask bottom inner surface. The spacer(s) ensure that the canister lid is laterally supported by the cask top forging when the cask is horizontal and minimizes axial movement of the canister. Each Class 1 PWR canister is positioned by a stainless steel spacer that is 16.75 inches in length. Each Class 2 PWR canister is positioned by a stainless steel spacer that is 7.65 inches in length. No spacers are used with the Class 3 PWR canister. The Class 4 BWR canister is located by four 1.5 inch aluminum spacers and the Class 5 BWR canister is located with a 1.5 inch aluminum spacer.

The spent fuel basket design uses a series of high strength stainless steel PWR or carbon steel BWR support disks to support the fuel assemblies in stainless steel tubes. The PWR fuel tubes contain neutron absorber on all four sides of the tubes. Three types of fuel tubes are designed to contain the BWR fuel: (1) tubes containing neutron absorber on two sides of the tubes; (2) tubes containing neutron absorber on one side; and (3) tubes containing no neutron absorber. Aluminum heat transfer disks are provided in both the PWR and BWR fuel baskets to enhance thermal performance of the basket. The heat transfer disks are supported by stainless steel tie rods and split spacers that maintain the basket assembly configuration.

The GTCC waste canister is essentially identical to the Class 1 TSC, except for the placement of lifting lugs and the placement of a key way within the canister. The GTCC basket is constructed of Type 304 stainless steel and consists primarily of a cylinder with a 3-inch thick wall closed at the bottom end with a 3-inch thick plate. The cylinder is centered in the GTCC waste canister by 14 Type 304 stainless steel support plates along its length. A 3-inch thick 304 stainless steel separator fixture divides the cylinder into two vertically stacked components, each 77 inches deep with a diameter of 47.8 inches.

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

The package has impact limiters at each end of the cask body. The impact limiters consist of a combination of redwood and balsa wood encased in Type 304 stainless steel. The impact limiters limit the g-loads acting on the cask during a transport drop load condition due to crushing of the redwood and balsa wood. The upper and lower impact limiters are bolted to the cask body by 16 equally spaced attachment rods with nuts.

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

Overall length (with impact limiters, in)	273.3
Overall length (without impact limiters, in)	209.3
Impact Limiter Outside diameter (in)	124.0
Outside diameter (without impact limiters, in)	92.9
Cavity diameter (in)	67.6
Cavity length (in)	192.5
Cask lid thickness (in)	6.5
Bottom thickness (in)	10.3
Inner shell thickness (in)	2.0
Outer shell thickness (in)	2.75
Gamma shield thickness (in)	2.75
Radial neutron shield thickness (in)	4.50

Transportable Storage Canister

Shell thickness (in)	0.625
Shell bottom (in)	1.75
Shield lid thickness (in)	7
Structural lid thickness (in)	3
Outer diameter (in)	67
Internal cavity diameter (in)	65.8
Internal fuel cavity length (in), depending on class	163-180
Overall length (in), depending on class	175-192

Fuel Basket

Basket assembly length (in), depending on class	162-180
Basket assembly diameter (in)	65.5
Number of support disks, depending on class	30-41
Number of heat transfer disks, depending on class	17-33

Total weight (pounds) including cask, basket, impact limiters, fuel, canister with lids, cask lid, and spacers for each fuel class is approximately:

Class 1 (PWR)	251,000
Class 2 (PWR)	252,000
Class 3 (PWR)	249,000
Class 4 (BWR)	256,000
Class 5 (BWR)	255,000

5.(a)(3) Drawings

The package is constructed and assembled in accordance with NAC drawings:

790-209, Rev. 1	790-210, Rev. 1	790-500, Rev. 4	790-501, Rev. 3
790-502, Rev. 7	790-503, Rev. 3	790-504, Rev. 2	790-505, Rev. 2
790-508, Rev. 2	790-509, Rev. 3	790-516, Rev. 3	790-519, Rev. 2
790-520, Rev. 2	790-570, Rev. 4	790-571, Rev. 3	790-572, Rev. 4
790-573, Rev. 7	790-574, Rev. 3	790-575, Rev. 10	790-581, Rev. 9
790-582, Rev. 12	790-583, Rev. 8	790-584, Rev. 19	790-585, Rev. 19
790-587, Rev. 1	790-591, Rev. 6	790-592, Rev. 8	790-593, Rev. 7
790-594, Rev. 2	790-595, Rev. 10	790-605, Rev. 11	790-611, Rev. 6
790-612, Rev. 9	412-501, Rev. 4	412-502, Rev. 6	

5.(b) Contents

(1) Type and Form of Material

The package is designed to transport four types of contents as listed below:

i. 24 intact irradiated PWR fuel assemblies within a TSC;
ii. 56 intact irradiated BWR fuel assemblies within a TSC;
iii. 24 Intact and Damaged PWR assemblies, and Fuel Debris from Maine Yankee within a TSC; or
iv. GTCC waste from Maine Yankee within a TSC.

Each type of package contents is described in detail below.

(i) Intact PWR assemblies

The package is designed to transport 24 irradiated intact PWR fuel assemblies within the TSC. An intact fuel assembly is a spent nuclear fuel assembly without known or suspected cladding defects greater than pinhole leaks or hairline cracks. An empty fuel rod position must be filled with a solid filler rod, fabricated from either zirconium alloy or Type 304 stainless steel, which displaces an equal or greater volume than that occupied by a fuel rod.

The fuel assemblies consist of uranium dioxide pellets with zirconium alloy type cladding. Prior to irradiation, the fuel assemblies must be within the dimensions and specifications of Table 5.(b)(1)(i)-1 below. The combined maximum average burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 5.(b)(1)(i)-2 below. PWR fuel assemblies may include standard inserts such as guide tube thimble plugs and burnable poison rods.

The minimum and maximum allowable assembly average enrichment for loading is 1.9 wt% ^{235}U and 4.2 wt% ^{235}U respectively. Unenriched fuel assemblies are not authorized for loading into the TSC. The maximum burn up of the spent fuel assemblies is 45,000 MWD/GTU and the minimum cool time is 5 years. The maximum weight of UO_2 is 11.53 MTU per cask.

Table 5.(b)(1)(i)-1, Intact PWR Fuel Assembly Characteristics													
TSC Class[1]	Vendor[2]	Array	Max. Length (in)	Max. Width (in)	Max. Assembly Weight	Max MTU	No of Fuel Rods	Max Pitch (in)	Min Rod Dia (in)	Min Clad Thick (in)	Max Pellet Dia (in)	Max Active Length (in)	Min Guide Tube Thickness (in)
1	CE	14x14	157.3	8.11	1292	0.404	176[4]	0.590	0.438	0.024	0.380	137.0	0.040
1	Ex/ANF	14x14	160.2	7.76	1271	0.369	179	0.556	0.424	0.030	0.351	142.0	0.034
1	WE	14x14	159.8	7.76	1177	0.362	179	0.556	0.400	0.024	0.345	144.0	0.034
1	WE	14x14	159.8	7.76	1302	0.415	179	0.556	0.422	0.022	0.368	145.2	0.034
1	WE, Ex/ANF	15x15	159.8	8.43	1472	0.465	204	0.563	0.422	0.024	0.366	144.0	0.015
1	Ex/ANF	17x17	159.8	8.43	1348	0.413	264	0.496	0.360	0.025	0.303	144.0	0.016
1	WE	17x17	159.8	8.43	1482	0.468	264	0.496	0.374	0.022	0.323	144.0	0.016
1	WE	17x17	160.1	8.43	1373	0.429	264	0.496	0.360	0.022	0.309	144.0	0.016
2	B&W	15x15	165.7	8.54	1515	0.481	208	0.568	0.430	0.026	0.369	144.0	0.016
2	B&W	17x17	165.8	8.54	1505	0.466	264	0.502	0.379	0.024	0.324	143.0	0.017
3	CE	16x16	178.3	8.10	1430	0.442	236[4]	0.506	0.382	0.023	0.3255	150.0	0.035
1	Ex/ANF[3]	14x14	160.2	7.76	1216	0.375	179	0.556	0.417	0.030	0.351	144.0	0.036
1	CE[3]	15x15	147.5	8.20	1360	0.432	216	0.550	0.418	0.026	0.358	132.0	—
1	Ex/ANF[3]	15x15	148.9	8.25	1339	0.431	216	0.550	0.417	0.030	0.358	131.8	—
1	CE[3]	16x16	158.2	8.10	1300	0.403	236[4]	0.506	0.382	0.023	0.3255	136.7	0.035

[1] Minimum and Maximum initial Enrichments are 1.9 wt% ^{235}U and 4.2 wt% ^{235}U, respectively. All fuel rods are zirconium alloy type clad.

[2] Vendor ID indicates the source of assembly base parameters. Loading of assemblies meeting dimensional limits is not restricted to the vendor(s) listed.

[3] 14x14, 15x15, and 16x16 fuel manufactured for Prairie Island, Palisades and St. Lucie 2 cores, respectively, These are not generic fuel assemblies provided to multiple reactors.

[4] Some fuel rod positions may be occupied by burnable poison rods or solid filler rods.

Table 5.(b)(1)(i)-2, Loading Table for Intact PWR Fuel

Minimum Initial Enrichment wt% ^{235}U (E)	Burnup ≤ 30 GWD/MTU Minimum Cooling Time (years)					30 < Burnup ≤ 35 GWD/MTU Minimum Cooling Time (years)				
	CE 14x14	14x14	15x15	16x16	17x17	CE 14x14	14x14	15x15	16x16	17x17
1.9 ≤ E < 2.1	6	8	8	7	8	8	10	11	9	10
2.1≤ E < 2.3	6	7	8	6	7	7	10	10	8	10
2.3≤ E < 2.5	6	7	7	6	7	7	9	10	8	9
2.5 ≤ E < 2.7	6	7	7	6	7	7	9	9	7	8
2.7≤ E < 2.9	6	7	7	6	7	6	8	9	7	8
2.9≤ E < 3.1	5	7	7	6	6	6	8	8	7	8
3.1≤ E < 3.3	5	6	7	6	6	6	8	8	7	7
3.3 ≤ E < 3.5	5	6	6	6	6	6	7	8	6	7
3.5 ≤ E < 3.7	5	6	6	6	6	6	7	7	6	7
3.7 ≤ E≤ 4.2	5	6	6	6	6	6	7	7	6	7
Minimum Initial Enrichment wt% ^{235}U (E)	35 < Burnup ≤ 40 GWD/MTU Minimum Cooling Time (years)					40 < Burnup ≤ 45 GWD/MTU Minimum Cooling Time (years)				
	CE 14x14	14x14	15x15	16x16	17x17	CE 14x14	14x14	15x15	16x16	17x17
1.9 ≤ E < 2.1	11	15	15	13	15	18	20	21	20	20
2.1≤ E < 2.3	10	13	14	12	13	15	19	19	18	19
2.3≤ E < 2.5	9	12	13	11	12	14	17	19	17	17
2.5 ≤ E < 2.7	9	12	12	10	11	12	16	18	15	17
2.7≤ E < 2.9	8	11	11	9	11	11	15	18	14	17
2.9≤ E < 3.1	8	10	10	9	10	10	14	18	13	15
3.1≤ E < 3.3	7	10	10	9	10	10	13	17	13	15
3.3 ≤ E < 3.5	7	9	10	8	9	9	12	17	13	15
3.5 ≤ E < 3.7	7	9	10	8	9	8	11	17	12	15
3.7 ≤ E≤ 4.2	7	8	10	8	8	8	11	15	12	14

5.(b)(1)(ii) Intact BWR assemblies

The package is designed to transport 56 irradiated intact BWR fuel assembles within the TSC. An intact fuel assembly is a spent nuclear fuel assembly without known or suspected cladding defects greater than pinhole leaks or hairline cracks.

For BWR fuel, the initial enrichment limit (the enrichment of the as-delivered fresh fuel assembly) represents the maximum peak planar-average enrichment allowed for loading into the TSC. The peak planar-average enrichment is defined to be the maximum planar-average enrichment at any height along the axis of the fuel assembly.

The fuel assemblies consist of uranium dioxide pellets with zirconium alloy type cladding. Prior to irradiation, the fuel assemblies must be within the dimension and specifications of Table 5.(b)(1)(ii)-1 below. The combined maximum average burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 5.(b)(1)(ii)-2.

BWR intact fuel assemblies are authorized with or without channels based on a maximum channel width of 120 mils. The minimum and maximum allowable assembly average enrichment for loading is 1.9 wt% ^{235}U and 4.0 wt% ^{235}U respectively. The maximum burn up of the spent fuel assemblies is 45,000 MWD/GTU and the minimum cool time is six years. The maximum weight of UO_2 is 11.08 MTU per cask. Unenriched fuel assemblies are not authorized for loading into the TSC. BWR fuel assemblies with unenriched axial blankets must have an enriched central fuel region and are acceptable for loading into a TSC if the minimum fuel enrichment of the central region is 1.9 wt% ^{235}U. Any empty fuel position must be filled with a solid filler rod fabricated from either zirconium alloy or Type 304 stainless steel.

Table 5.(b)(1)(ii)-1, Intact BWR Fuel Assembly Characteristics												
Canister Class [1,5]	Vendor[4]	Array	Max. Length (in)	Max. Assembly Width (in)[5]	Max. Assembly Weight (lb)[6]	Max MTU	No of Fuel Rods	Max Pitch (in)	Min Rod Dia (in)	Min Clad Thick (in)	Max Pellet Dia (in)	Max Active Length (in)[2]
4	Ex/ANF	7x7	171.3	5.51	620	0.196	48	0.738	0.570	0.036	0.490	144
4	Ex/ANF	8x8	171.3	5.51	563	0.177	63	0.641	0.484	0.036	0.405	145.2
4	Ex/ANF	9x9	171.3	5.51	557	0.173	79	0.572	0.424	0.030	0.357	145.2
4	GE	7x7	171.1	5.51	681	0.199	49	0.738	0.570	0.036	0.488	144.0
4	GE	7x7	171.2	5.51	681	0.198	49	0.738	0.563	0.032	0.487	144.0
4	GE	8x8	171.1	5.51	639	0.173	60	0.640	0.484	0.032	0.410	145.2
4	GE	8x8	171.1	5.51	681	0.179	62	0.640	0.483	0.032	0.410	145.2
4	GE	8x8	171.1	5.51	681	0.186	63	0.640	0.493	0.034	0.416	144.0
5	Ex/ANF	8x8	176.1	5.51	588	0180	62	0.641	0.484	0.036	0.405	150.0
5	Ex/ANF	9x9	176.1	5.51	576	0.167	74[3]	0.572	0.424	0.030	0.357	150.0
5[5]	Ex/ANF	9x9	176.1	5.51	576	0.178	79[3]	0.572	0.424	0.030	0.357	150.0
5	GE	7x7	175.9	5.51	683	0.198	48	0.738	0.563	0.032	0.487	144.0
5	GE	8x8	176.1	5.51	665	0.179	60	0.640	0.484	0.032	0.410	150.0
5	GE	8x8	175.9	5.51	681	0.185	62	0.640	0.483	0.032	0.410	150.0
5	GE	8x8	175.9	5.51	681	0.188	63	0.640	0.493	0.034	0.416	146.0
5	GE	9x9	176.1	5.51	646	0.186	74[3]	0.566	0.441	0.028	0.376	150.0
5	GE	9x9	176.1	5.51	646	0.198	79[3]	0.566	0.441	0.028	0.376	150.0

[1] Maximum Peak Planar Average Enrichment 4.0 wt%^{235}U. Minimum enrichment is 1.9 wt%^{235}U. All fuel rods are zirconium alloy type clad.

[2] 150 inch active fuel length assemblies contain 6 inch natural uranium blankets on top and bottom.

[3] Shortened active fuel length in some rods.

[4] Vendor ID indicates the source of assembly base parameters. Loading of assemblies meeting dimensional limits is not restricted to the vendor(s) listed.

[5] Assembly width including channel. Unchanneled or channeled may be loaded based on a maximum channel thickness of 120 mils.

[6] Exxon/ANF assembly weights are listed without channel.

Table 5.(b)(1)(ii)-2, Loading Table for Intact BWR Fuel

Minimum Initial Enrichment wt% ^{235}U (E)	Burnup ≤ 30 GWD/MTU Minimum Cooling Time (years)			30 < Burnup ≤ 35 GWD/MTU Minimum Cooling Time (years)		
	9x9	8x8	7x7	9x9	8x8	7x7
1.9 ≤ E < 2.1	8	8	8	14	13	15
2.1≤ E < 2.3	7	7	8	12	12	13
2.3≤ E < 2.5	7	7	7	11	10	11
2.5 ≤ E < 2.7	7	6	7	9	9	10
2.7≤ E < 2.9	6	6	6	9	8	9
2.9≤ E < 3.1	6	6	6	8	8	8
3.1≤ E < 3.3	6	6	6	7	7	8
3.3 ≤ E < 3.5	6	6	6	7	7	7
3.5 ≤ E < 3.7	6	6	6	7	7	7
3.7 ≤ E≤ 4.0	6	6	6	7	7	7
Minimum Initial Enrichment wt% ^{235}U (E)	35 < Burnup ≤ 40 GWD/MTU Minimum Cooling Time (years)			40 < Burnup ≤ 45 GWD/MTU Minimum Cooling Time (years)		
	9x9	8x8	7x7	9x9	8x8	7x7
1.9 ≤ E < 2.1	24	23	25	34	33	35
2.1≤ E < 2.3	21	20	22	31	30	32
2.3≤ E < 2.5	19	18	20	29	28	29
2.5 ≤ E < 2.7	17	16	17	26	25	27
2.7≤ E < 2.9	14	14	15	24	23	24
2.9≤ E < 3.1	13	12	13	21	20	22
3.1≤ E < 3.3	11	11	12	19	18	20
3.3 ≤ E < 3.5	10	10	11	17	16	18
3.5 ≤ E < 3.7	10	9	10	15	14	16
3.7 ≤ E≤ 4.0	10	9	10	14	13	15

5.(b)(1)(iii) Intact and Damaged PWR assemblies, and Fuel Debris from Maine Yankee

The package is designed to transport 24 irradiated intact or damaged PWR fuel assemblies, canistered fuel debris, and GTCC waste within the TSC from the Maine Yankee Reactor. The standard Maine Yankee fuel assembly is the intact PWR CE 14x14 (see section 5.(b)(1)(i)).

In the course of reactor operations, some of the 14x14 assemblies were modified to change the standard configuration. These modifications included a) the removal of fuel rods without replacement; b) the replacement of removed fuel rods or burnable poison rods with rods of a different material, such as stainless steel, or with fuel rods of a different enrichment; and c) the insertion of control elements, or instruments or plug thimbles, in guide tube positions. In addition to the modified fuel assemblies, there are fuel assemblies that were designed with variable enrichment and axial blankets. These fuel assemblies are not modified, but differ from the cask design basis fuel assemblies.

Stainless steel spacers may be used in canisters to axially position PWR intact fuel assemblies that are shorter than the available cavity length. The minimum length of the PWR intact fuel assembly internal structure and bottom end fitting and/or spacers will ensure that the minimum distance to the fuel region for the base of the canister is 3.2 inches.

Unenriched fuel assemblies are not authorized for loading.

The following are the allowable Maine Yankee site specific contents:

5.(b)(1)(iii)(A) Maine Yankee's site specific contents not requiring preferential loading patterns:

(1) Standard Irradiated CE 14 x 14 intact PWR fuel assemblies meeting the PWR fuel assembly characteristics in Table 5.(b)(1)(i)-1. The maximum fuel assembly weight, including other associated hardware is 1,515 pounds. The combined maximum average burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 5.(b)(1)(iii)(A)-1.

(2) Irradiated Maine Yankee CE 14 x 14 PWR intact fuel assemblies may contain inserted control element assemblies (CEA), in-core instrument (ICI) thimbles or CEA plugs. CEAs or CEA plugs may not be inserted in damaged fuel assemblies, consolidated fuel assemblies or assemblies with irradiated stainless steel replacement rods. Fuel assemblies with a CEA or CEA plug inserted must be loaded in a Class 2 canister and cannot be loaded in a Class 1 canister. Fuel assemblies without an inserted CEA or CEA plug, including those with inserted ICI Thimbles, must be loaded in a Class 1 canister. The combined maximum average burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 5.(b)(1)(iii)(A)-1 except for those assemblies containing ICI thimbles which must meet the specifications of Table 5.(b)(1)(iii)(A)-2.

(3) PWR intact fuel assemblies with fuel rods replaced with stainless steel or zirconium alloy rods or with Uranium oxide rods nominally enriched up to 1.95 wt%. The combined maximum average burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 5.(b)(1)(iii)(A)-3.

(4) PWR intact fuel assemblies with fuel rods having variable enrichments with a maximum rod enrichment up to 4.21 wt% ^{235}U and that also have a maximum planar average enrichment up to 3.99 wt% ^{235}U. The combined maximum average burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 5.(b)(1)(iii)(A)-1.

(5) PWR intact fuel assemblies with annular axial end blanket enrichments up to 2.6 wt% ^{235}U. The combined maximum average burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 5.(b)(1)(iii)(A)-1.

(6) PWR intact fuel assemblies with burnable poison rods or solid filler rods may occupy up to 16 of 176 fuel rod positions. The combined maximum average burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 5.(b)(1)(iii)(A)-1.

(7) PWR intact fuel assemblies with one or more grid spacers missing or damaged such that the unsupported length of the fuel rods does not exceed 60 inches, or with end fitting damage, including damaged or missing hold-down springs, as long as the assembly can be handled safely by normal means. The combined maximum average burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 5.(b)(1)(iii)(A)-1.

5.(b)(1)(iii)(B) Maine Yankee site-specific allowable contents requiring preferential loading based on shielding, criticality, or thermal constraints (Maine Yankee CE 14 x 14 intact PWR fuel assemblies). A PWR basket fuel diagram can be found on Figure 5.(b)(1)(iii)(B)-1.

(1) Maine Yankee CE 14 x 14 PWR intact fuel assemblies with a burn up between 45,000 and 50,000 MWD/MTU meeting the following requirements for verification of the oxide layer thickness and high burn up fuel requiring preferential loading in the peripheral PWR fuel basket positions:

A verification program is required to determine the oxide layer thickness on high burn up fuel by measurement or by statistical analysis. A fuel assembly having a burn up between 45,000 MWD/MTU and 50,000 MWD/MTU is classified as high burn up. The verification program shall be capable of classifying high burn up fuel as INTACT FUEL or DAMAGED FUEL based on the following criteria:

I. A HIGH BURN UP FUEL assembly may be stored as INTACT FUEL provided that no more than 1% of the fuel rods in the assembly have a peak cladding oxide thickness greater than 80 microns, and that no more than 3% of the fuel rods in the assembly have a peak oxide layer thickness greater than 70 microns, and that the fuel assembly is otherwise INTACT FUEL.

II. A HIGH BURN UP FUEL assembly not meeting the cladding oxide thickness criteria for INTACT FUEL or that has an oxide layer that is detached or spalled from the cladding is classified as DAMAGED FUEL.

The combined maximum average burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 5.(b)(1)(iii)(A)-1.

(2) PWR intact fuel assemblies with up to 176 fuel rods missing from the fuel assembly lattice. The combined maximum average burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 5.(b)(1)(iii)(A)-1. These assemblies must be placed in a corner loading position in the PWR fuel basket.

(3) PWR intact fuel assemblies with burnable poison rods replaced by hollow zirconium alloy rods. The combined maximum average burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 5.(b)(1)(iii)(A)-1. These assemblies must be placed in a corner PWR fuel basket loading position.

(4) Intact fuel assemblies with a start-up source in a center guide tube. The assembly must be loaded in a basket corner position and must be loaded in a Class 1 canister. Only one start-up source may be loaded in any fuel assembly or any canister. The combined maximum average burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 5.(b)(1)(iii)(A)-1. These assemblies must be placed in a corner PWR fuel basket loading position.

(5) PWR intact fuel assemblies with CEA ends (fingertips) and/or an ICI segment inserted in corner guide tube positions. The assembly must also have a CEA plug installed. The assembly must be loaded in a PWR fuel basket corner position and must be loaded in a Class 2 canister. The combined maximum average burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 5.(b)(1)(iii)(A)-1. CEA fingertips are not considered as CEAs for determination of minimum cool times.

5.(b)(1)(iii)(C) Maine Yankee CE 14 x 14 PWR fuel enclosed in a Maine Yankee Fuel Can (MYFC).

All Maine Yankee CE 14 x 14 PWR fuel enclosed in an MYFC must be loaded in a Class 1 fuel canister in a corner position of the PWR fuel basket. Up to 4 MYFC may be loaded into a TSC. Intact Maine Yankee CE 14 x 14 PWR fuel may be loaded into a MYFC. The contents that must be loaded in the MYFC are:

(1) PWR fuel assemblies with up to two intact or damaged fuel rods inserted in each fuel assembly guide tube or with up to two burnable poison rods inserted in each guide tube. The rods inserted in the guide tubes cannot be from a different fuel assembly. The maximum number of rods in the fuel assembly (fuel rods plus inserted rods, including burnable poison rods) is 176. The combined maximum average burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 5.(b)(1)(iii)(A)-1 for intact fuel rods and Table 5.(b)(1)(iii)(A)-4 for damaged fuel rods.

(2) A damaged fuel assembly with up to 100% of the fuel rods classified as damaged and/or damaged or missing assembly hardware components. A damaged fuel assembly cannot have an inserted CEA or other non-fuel component. The combined maximum average burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 5.(b)(1)(iii)(A)-4 for damaged fuel rods.

(3) Individual intact or damaged fuel rods in a rod type structure, which may be a ~~guide tube, to maintain configuration control. The combined maximum average~~

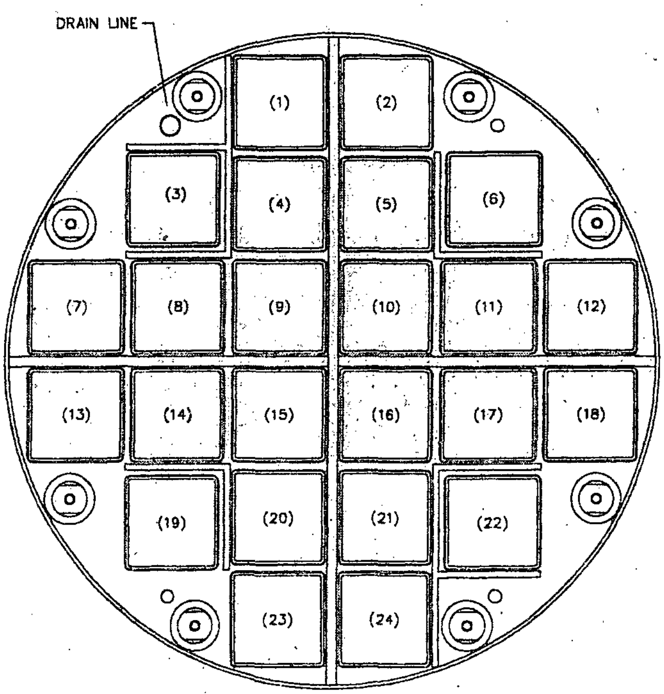
burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 5.(b)(1)(iii)(A)-1 for intact fuel rods and Table 5.(b)(1)(iii)(A)-4 for damaged fuel rods.

(4) Fuel debris consisting of fuel rods with exposed fuel pellets or individual intact or partial fuel pellets not contained in fuel rods. The combined maximum average burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 5.(b)(1)(iii)(A)-4 for damaged fuel rods.

(5) Consolidated Fuel lattice and structure with a 17 x 17 array formed by grids and top and bottom end fittings connected by four solid stainless steel rods. Maximum contents are 289 fuel rods having a total lattice weight less than or equal to 2,100 pounds. A consolidated fuel lattice cannot have an inserted CEA or other non-fuel component. Only one consolidated fuel lattice may be stored in any TSC. Fuel must be cooled a minimum of 24 years.

(6) High burn up fuel assemblies not meeting the oxide layer thickness criteria previously defined in Section 5.(b)(1)(iii)(B)(1). The combined maximum average burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 5.(b)(1)(iii)(A)-4 for damaged fuel rods.

PWR Basket Fuel Loading Position Diagram, Figure 5.(b)(1)(iii)(B)-1



1. Basket corner positions are positions 3, 6, 19, and 22. Corner positions are also periphery positions.
2. Basket periphery positions are positions 1, 2, 3, 6, 7, 12, 13, 18, 19, 22, 23, and 24. Periphery positions include the corner positions.

Table 5.(b)(1)(iii)(A)-1, Loading Table for Maine Yankee CE 14x14 Fuel with and without CEA Cooled to Indicated Time						
Burnup 30 GWD/MTU		Minimum Cool Time (Years) for				
Enrichment	No CEA (Class 1)	No CEA (Class 2)	5 Yr CEA	10 Yr CEA	15 Yr. CEA	20 Yr. CEA
1.9 ≤ E < 2.1	6	6	7	6	6	6
2.1≤ E < 2.3	6	6	7	6	6	6
2.3≤ E < 2.5	6	6	6	6	6	6
2.5 ≤ E < 2.7	6	6	6	6	6	6
2.7≤ E < 2.9	6	6	6	6	6	6
2.9≤ E < 3.1	5	6	6	6	6	6
3.1≤ E < 3.3	5	5	6	6	6	5
3.3 ≤ E < 3.5	5	5	6	6	5	5
3.5 ≤ E < 3.7	5	5	6	5	5	5
3.7 ≤ E≤ 4.2	5	6	5	5	5	5
Loading Table for Maine Yankee CE 14x14 Fuel with and without CEA Cooled to Indicated Time						
Burnup 35 GWD/MTU		Minimum Cool Time (Years) for				
Enrichment	No CEA (Class 1)	No CEA (Class 2)	5 Yr CEA	10 Yr CEA	15 Yr. CEA	20 Yr. CEA
1.9 ≤ E < 2.1	8	8	9	8	8	8
2.1≤ E < 2.3	7	7	9	8	8	8
2.3≤ E < 2.5	7	7	8	7	7	7
2.5 ≤ E < 2.7	7	7	8	7	7	7
2.7≤ E < 2.9	6	7	7	7	7	7
2.9≤ E < 3.1	6	6	7	7	6	6
3.1≤ E < 3.3	6	6	7	6	6	6
3.3 ≤ E < 3.5	6	6	7	6	6	6
3.5 ≤ E < 3.7	6	6	6	6	6	6
3.7 ≤ E≤ 4.2	6	6	6	6	6	6

Table 5.(b)(1)(iii)(A)-1, continued , Loading Table for Maine Yankee CE 14x14 Fuel with and without CEA Cooled to Indicated Time						
Burnup 40 GWD/MTU		Minimum Cool Time (Years) for				
Enrichment	No CEA (Class 1)	No CEA (Class 2)	5 Yr CEA	10 Yr CEA	15 Yr. CEA	20 Yr. CEA
1.9 ≤ E < 2.1	11	12	14	13	12	12
2.1≤ E < 2.3	10	10	13	11	11	11
2.3≤ E < 2.5	9	9	12	10	10	10
2.5 ≤ E < 2.7	9	9	10	9	9	9
2.7≤ E < 2.9	8	8	10	9	8	8
2.9≤ E < 3.1	8	8	9	8	8	8
3.1≤ E < 3.3	7	7	8	8	8	8
3.3 ≤ E < 3.5	7	7	8	7	7	7
3.5 ≤ E < 3.7	7	7	8	7	7	7
3.7 ≤ E≤ 4.2	7	7	7	7	7	7

Loading Table for Maine Yankee CE 14x14 Fuel with and without CEA Cooled to Indicated Time						
Burnup 45 GWD/MTU		Minimum Cool Time (Years) for				
Enrichment	No CEA (Class 1)	No CEA (Class 2)	5 Yr CEA	10 Yr CEA	15 Yr. CEA	20 Yr. CEA
1.9 ≤ E < 2.1	18	18	21	19	18	18
2.1≤ E < 2.3	15	16	19	17	17	16
2.3≤ E < 2.5	14	14	18	16	15	15
2.5 ≤ E < 2.7	12	13	16	14	14	13
2.7≤ E < 2.9	11	12	14	13	12	12
2.9≤ E < 3.1	10	11	13	12	11	11
3.1≤ E < 3.3	10	10	12	11	10	10
3.3 ≤ E < 3.5	9	9	11	10	10	10
3.5 ≤ E < 3.7	9	9	10	10	10	10
3.7 ≤ E≤ 4.2	9	9	10	10	10	10

Table 5.(b)(1)(iii)(A)-1, continued, Loading Table for Maine Yankee CE 14x14 Fuel with and without CEA Cooled to Indicated Time

Burnup 50 GWD/MTU		Minimum Cool Time (Years) for				
Enrichment	No CEA (Class 1)	No CEA (Class 2)	5 Yr CEA	10 Yr CEA	15 Yr. CEA	20 Yr. CEA
1.9 ≤ E < 2.1	27	27	29	27	27	27
2.1≤ E < 2.3	24	24	27	25	24	24
2.3≤ E < 2.5	22	22	25	23	22	22
2.5 ≤ E < 2.7	19	19	23	21	20	20
2.7≤ E < 2.9	17	17	21	19	18	18
2.9≤ E < 3.1	15	16	19	18	18	18
3.1≤ E < 3.3	15	15	18	17	17	17
3.3 ≤ E < 3.5	15	15	17	17	17	17
3.5 ≤ E < 3.7	14	14	15	15	15	15
3.7 ≤ E≤ 4.2	14	14	15	15	15	15

Table 5.(b)(1)(iii)(A)-2, Loading Table (Years) for Maine Yankee CE 14x14 fuel containing ICI Thimbles

Minimum Initial Enrichment wt% ^{235}U (E)	Burnup ≤ 30 GWD/MTU	30 < Burnup ≤ 35 GWD/MTU	35 < Burnup ≤ 40 GWD/MTU	40 < Burnup ≤ 45 GWD/MTU	45 < Burnup ≤ 50 GWD/MTU
1.9 ≤ E < 2.1	6	8	11	18	27
2.1≤ E < 2.3	6	7	10	16	24
2.3≤ E < 2.5	6	7	9	14	22
2.5 ≤ E < 2.7	6	7	9	13	19
2.7≤ E < 2.9	6	6	8	11	17
2.9≤ E < 3.1	5	6	8	10	15
3.1≤ E < 3.3	5	6	7	10	15
3.3 ≤ E < 3.5	5	6	7	9	15
3.5 ≤ E < 3.7	5	6	7	9	14
3.7 ≤ E≤ 4.2	5	6	7	9	14

Table 5.(b)(1)(iii)(A)-3, Required Cool Time for Maine Yankee Fuel Assemblies with Activated Stainless Steel Replacement Rods

Assy Number	Burnup (GWD/MTU)	Enrichment (wt %)	SSR Source (g/s/assy)	Cool Time (years)	Earliest Transportable
N420	45	3.3	2.1602E+13	10	Jan 2001
N842	35	3.3	3.1396E+12	6	Jan 2001
N868	40	3.3	5.2444E+12	7	Jan 2001
R032	45	3.5	1.4550E+13	9	Jan 2005
R439	50	3.5	1.3998E+13	14	Jan 2010
R444	50	3.5	5.5993E+13	19	Jan 2015

Table 5.(b)(1)(iii)(A)-4, Cool time (years) for Maine Yankee CE 14x14 damaged fuel

Minimum Initial Enrichment wt% ^{235}U (E)	Burnup ≤ 30 GWD/MTU	30 < Burnup ≤ 35 GWD/MTU	35 < Burnup ≤ 40 GWD/MTU	40 < Burnup ≤ 45 GWD/MTU	45 < Burnup ≤ 50 GWD/MTU
1.9 ≤ E < 2.1	7	11	19	28	37
2.1 ≤ E < 2.3	6	9	16	26	34
2.3 ≤ E < 2.5	6	8	14	23	32
2.5 ≤ E < 2.7	6	8	12	21	30
2.7 ≤ E < 2.9	6	7	11	19	27
2.9 ≤ E < 3.1	6	7	10	17	25
3.1 ≤ E < 3.3	5	7	9	15	23
3.3 ≤ E < 3.5	5	6	8	13	21
3.5 ≤ E < 3.7	5	6	8	12	19
3.7 ≤ E ≤ 4.2	5	6	7	11	17

5.(b)(1)(iv) Greater Than Class C Waste from Maine Yankee

The package is designed to transport Maine Yankee Greater Than Class C Waste within a TSC. Maine Yankee GTCC waste consists of solid, irradiated, and contaminated hardware and solid, particulate debris or filter media, provided the quantity of fissile material does not exceed a Type A quantity and does not exceed the mass limits of 10 CFR 71.15. The maximum curie inventory shall not exceed the values shown in Table 5.(b)(1)(iv)-1.

Table 5.(b)(1)(iv)-1, Maine Yankee GTCC Curie Inventory Limits per TSC

Radionuclide	Curie Inventory (Ci)/ TSC
H-3	3.00E+02
C-14	1.50E+02
Mn-54	3.50E+02
Fe-55	2.00E+05
Co-58	1.00E+01
Co-60	2.90E+05
Ni-59	8.20E+02
Ni-63	9.00E+04
Nb-94	1.00E+01
Tc-99	1.00E+01

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

The maximum weight of the contents shall not exceed 77,500 pounds.

(i) For the contents described in 5.(b)(1)(i) and 5.(b)(1)(iii): 24 PWR fuel assemblies, including standard inserts such as burnable poison rods or guides or guide tube thimble plugs, with a maximum weight of 38,500 pounds and a maximum decay heat limit per package not to exceed the values in Table 5.(b)(2)-1. The individual PWR assembly decay heat is limited to 0.83 kW.

Table 5.(b)(2)-1, PWR Decay Heat Limits				
Cool Time (Years)	PWR Decay Heat Limit (kW) Burnup (MWD/MTU)			
	35,000	40,000	45,000	50,000[1]
5	20.0	20.0	19.9	19.3
6	19.5	19.3	19.2	18.7
7	17.8	17.8	17.7	17.2
10	17.4	17.3	17.2	16.8
15	16.8	16.8	16.7	16.5

[1]Maine Yankee PWR fuel assemblies

(ii) For the contents described in 5.(b)(1)(ii): 56 BWR assemblies with a maximum weight of 39,000 pounds and a maximum decay heat limit per package of 16 kW. The individual BWR assembly decay heat is limited to 0.29 kW.

(iii) For the contents described in 5.(b)(1)(iv): GTCC waste with a maximum weight per package of 20,000 pounds in total or 10,000 pounds per compartment. The maximum decay heat for the GTCC is 4.5 kW per package.

5.(c) Criticality Safety Index 0.0

6. The package must be transported as exclusive use in a closed transport vehicle as per 10 CFR 71.47(b).

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

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

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

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

9. Transport by air of fissile material is not authorized.

10. Expiration date: October 31, 2012.

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9270	3	71-9270	USA/9270/B(U)F-96	20	OF	20

REFERENCES

NAC International, Inc., Application dated April 30, 1997.

NAC International, Inc., Supplements dated June 18, 1999, May 31, June 29, August 8, and September 20, 2000; February 28, March 14, March 31, June 1, and November 16, 2001; January 31, March 13, August 12, September 27, and October 21, 2002; March 31, and September 28, 2004; May 4, and June 6, 2005; and September 25, 2007.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Robert A. Nelson, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: October 29, 2007.

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
9274	8	71-9274	USA/9274/AF	1	OF	3

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

Westinghouse Electric Company, LLC
P.O. Box 355
Pittsburgh, PA 15230-0355

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Westinghouse Electric Company application dated May 15, 2003, as supplemented

4. CONDITIONS

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

5.

(a) Packaging

(1) Model No.: ABB-2901

(2) Description

A shipping container for low-enriched uranium oxide pellets, composed of an inner container, surrounded by insulating material, and an outer drum. The inner container is 10.75 ± 1/4 inches square and approximately 30 inches long, constructed of minimum 14-gauge steel, with bolted and gasketed top flange closure and welded bottom sheet. The inner container is centered and supported in an 18-gauge steel drum by asbestos or ceramic sheet, plywood, hardboard, and insulating material. The drum has a 16-gauge closure lid. The drum lid is closed with a 12-gauge locking ring with drop forged lugs and a 5/8-inch diameter bolt. In addition to the locking ring, three lid clamps are installed to secure the drum lid. The drum has approximate dimensions of 22.5-inch ID by 36-inch height. The uranium oxide pellets are packaged in boxes positioned within a steel insert. The maximum gross weight of the package is 660 pounds.

(3) Drawings

The packaging is constructed and assembled in accordance with Westinghouse Electric Company, LLC, Drawing Nos.

10004E01, Rev. 2;
10004E02, Sheets 1 and 2, Rev. 2; and
10004E03, Rev. 2.

5. (b) Contents

(1) Type and form of material

Sintered uranium oxide pellets enriched to a maximum 5.0 w/o in the U-235 isotope. The maximum pellet diamefer 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) Criticality Safety Index (minimum index to be shown on label): 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 71.17.

10. Transport by air of fissile material is not authorized.

11. Revision No. 7 of this certificate may be used until August 31, 2008.

12. Expiration date: September 30, 2012.

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
9274	8	71-9274	USA/9274/AF	3	OF	3

REFERENCES

Westinghouse Electric Company application dated May 15, 2003.

Supplements dated November 21, 2003, and July 23, 2007.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Robert A. Nelson, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: August 23, 2007

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
9276	3	71-9276	USA/9276/B(U)F-85	1	OF	26

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

EnergySolutions
2105 S. Bascom Ave., Suite 160
Campbell, CA 95008

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

BNFL Fuel Solutions application dated April 20, 2001, as supplemented.

4. CONDITIONS

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

5.

(a) Packaging

(1) Model No. FuelSolutions™ TS125 Transportation Package

(2) Description

The FuelSolutions™ TS125 Transportation Package consists of a TS125 Transportation Cask and impact limiters, together with a FuelSolutions™ W21 or W74 canister and its payload. The FuelSolutions™ canister and its payload are contained inside the TS125 Transportation Cask cavity. The TS125 Transportation Cask cavity is sized to accommodate one FuelSolutions™ long canister, or alternatively, one FuelSolutions™ short canister with a cask cavity spacer. The approximate dimensions and weights of the package are as follows:

Package Length:	342.4 inches
Package Outside Diameter:	143.5 inches
Cask Length (w/o impact limiters):	210.4 inches
Cask Outside Diameter (w/o impact limiters):	94.2 inches
Cask Cavity Length:	193.0 inches
Cask Cavity Diameter (section at rails):	66.88 inches
Canister Outside Diameter:	66.0 inches
Maximum Long Canister Length:	192.25 inches
Maximum Short Canister Length:	182.25 inches
Cask Cavity Spacer Length:	10.0 inches
Max. Package Weight:	285,000.0 pounds
Max. Cask Payload Weight (incl. canister and cavity spacer):	85,000.0 pounds

The TS125 Transportation Cask body is an assembly composed of stainless steel components of an inner shell, an outer shell, a top ring forging, a closure lid with a seal test port and a cavity vent port, a bottom plate forging, and a cavity drain port. The inner and outer shells are welded to the bottom plate forging and the top ring forging. The cask body also includes an annular lead gamma shield; an annular neutron shield with cask tie-down rings, support angles, and jacket; a bottom end neutron shield with a support ring and jacket; a longitudinal shear block; and lifting trunnion mounting bosses. The inner and outer shells form the annular cavity for the lead gamma shield. The outer shell and the neutron shield jacket form the annular cavity for the solid neutron shield. The neutron shield support angles facilitate heat rejection through the solid neutron shielding material to the outer surface of the cask body. The cask closure lid includes a thick recessed plate with two concentric "Helicoflex" silver-jacketed metallic o-ring seals, the cavity vent port, and the seal test port. The closure lid is secured to the cask body during transport with 60 – 2 inch diameter closure bolts. The vent and drain ports are closed by a plug assembly to maintain containment integrity during transportation.

The Transportation Cask's containment boundary consists of: the inner cylindrical shell, the bottom plate forging (which forms the bottom closure of the cask), the top ring forging and sealing surfaces, the closure lid and sealing surfaces, the welds associated with the above components, the closure bolts, the innermost closure lid o-ring seal, the cavity vent port seal gland and o-ring seal, and the cavity drain port seal gland and o-ring seal. The package is designed to be "leaktight" as defined by ANSI N14.5 (leakage rate less than or equal to 1×10^{-7} ref-cm^3/s. The structural components of the Transportation Cask are made of high-strength austenitic stainless steel. The gamma shielding is made of lead and is completely enclosed within the annular region between the inner and outer steel shells. The neutron shielding is solid hydrogenous material that is completely enclosed within the annular region between the cask outer shell and neutron shield jacket with tie-down rings at each end.

The FuelSolutions™ TS125 Transportation Cask has identical energy-absorbing impact limiters at both ends. Each impact limiter assembly consists of crushable aluminum honeycomb energy-absorbing core segments that are encased in a sealed stainless steel shell. In addition to confining the aluminum honeycomb core segments in the event of a free drop, the impact limiter shell protects the aluminum honeycomb material from the weather. Both the top and bottom impact limiters are attached to the transportation cask body tie-down rings with 12, one inch diameter bolts. A tamper-indicating device is provided which connects each impact limiter to the transportation cask to assure that the package has not been opened by unauthorized personnel during transport.

A FuelSolutions™ canister consists of a steel shell assembly and an internal basket assembly. The shell assembly maintains a helium atmosphere for transport conditions. Credit is not taken for containment provided by the canister shell for transport conditions. The shell assembly also provides radiological shielding in both the radial and axial directions. The internal basket assembly provides geometric spacing, structural support, and criticality control for the spent nuclear fuel (SNF) assemblies for transport conditions.

There are two classes of W21 canisters (W21T and W21M), differing primarily in materials of construction. Each W21 canister class includes four different canister types, as follows. The W21T canister class includes a long canister with lead shield plugs (W21T-LL), a long canister with carbon steel shield plugs (W21T-LS), a short canister with lead shield plugs (W21T-SL), and a short canister with carbon steel shield plugs (W21T-SS). The W21M canister class includes a long canister with depleted uranium shield plugs (W21M-LD), a long canister with carbon steel shield plugs (W21M-LS), a short canister with depleted uranium shield plugs (W21M-SD), and a short canister with carbon steel shield plugs (W21M-SS). There are also two classes of W74 canisters (W74T and W74M), differing primarily in materials of construction. Both the W74T and W74M canister classes include only a long canister with carbon steel shield plugs.

A FuelSolutions™ canister shell assembly consists of a steel cylindrical shell, bottom end closure, bottom shield plug, bottom shell extension, bottom outer plate, top shield plug, top inner closure plate, and top outer closure plate. The closure plates at the top and bottom are welded to the cylindrical shell. All structural components of the canister shell assembly are constructed of austenitic stainless steel, with the exception of the shield plugs. The shield plug materials may be composed of lead, depleted uranium or carbon steel, depending upon the specific canister variant. To prevent any corrosion, galvanic, or chemical reactions between the shield plug materials and the cask environment or contents, the shield materials are isolated from the environment and cask interior. The lower shield plugs are encased within stainless steel. The upper shield plugs that are made of lead or depleted uranium are encased in stainless steel. The carbon steel upper shield plug is electroless nickel-plated.

A FuelSolutions™ W21 canister basket assembly consists of 21 guide tubes that are positioned and supported by a series of circular spacer plates, which are in turn positioned and supported by support rod assemblies. The W21 guide tubes include neutron absorber sheets on all four sides.

The W74 canister includes two stackable basket assemblies with a capacity to accommodate up to 64 Big Rock Point fuel assemblies. Each basket includes 37 cell locations, with the center five cell locations mechanically blocked to prevent fuel loading in these locations. The W74 basket assembly consists of a series of circular spacer plates that are positioned and supported by four support tubes that run through the spacer plates and support sleeves between the spacer plates. Each basket cell location, with the exception of the four support tubes and the five blocked-out center cells, contain a guide tube assembly. The W74 guide tube assemblies include borated stainless steel neutron absorber sheets on either one side or two opposite sides. The guide tubes are arranged in the basket to position at least one poison sheet between adjacent fuel assemblies, with the exception of intact fuel assemblies placed in the support tubes.

In the W74 basket, damaged fuel is placed in damaged fuel cans that are accommodated in the support tube cell locations. The W74 damaged fuel cans are similar to the W74 guide tubes, but include a screened bottom end, a screened removal lid, and borated stainless steel neutron absorber sheets on all four sides.

(3) Drawings

The FuelSolutions™ TS125 Transportation Package is constructed and assembled in accordance with the following drawings:

FS-200, Revision 1, Sheets 1 through 3
FS-205, Revision 2, Sheets 1 through 3
FS-210, Revision 2, Sheets 1 through 9
FS-220, Revision 1, Sheets 1 through 7
FS-230, Revision 1, Sheets 1 and 2
W21-110, Revision 4, Sheets 1 through 9

W21-120, Revision 5, Sheets 1 through 10
W21-121, Revision 5, Sheet 1
W21-122, Revision 3, Sheets 1 and 2
W21-130, Revision 4, Sheets 1 through 9
W21-131, Revision 3, Sheets 1 and 2
W21-140, Revision 5, Sheets 1 through 4
W21-150, Revision 4, Sheets 1 and 2
W21-190, Revision 4, Sheet 1
W74-110, Revision 5, Sheets 1 and 2
W74-120, Revision 5, Sheets 1 through 6
W74-121, Revision 7, Sheet 1
W74-122, Revision 6, Sheet 1
W74-130, Revision 6, Sheets 1 and 2
W74-140, Revision 5, Sheets 1 through 4
W74-150, Revision 5, Sheets 1 and 2
3319, Revision 6, Sheets 1 through 5

(b) Contents

(1) Type and Form of Material

Shipment of spent fuel, with plutonium in excess of 20 curies per package, in the form of debris, particles, loose pellets, and fragmented rods or assemblies, is not authorized.

(i) W21 Canister

The contents of the W21 canister are limited to 21 pressurized water reactor (PWR) SNF assemblies meeting the requirements of Table 1 and Table 2. Two different loading configurations, designated as W21-1 and W21-2, are permitted in the W21 canister. The W21-2 loading configuration, which accommodates SNF with higher initial ^{235}U enrichments, consists of up to 20 PWR SNF assemblies meeting the requirements of Table 1 and Table 2. The W21-2 loading configuration requires that the center guide tube be mechanically blocked to prevent inadvertent loading of a SNF assembly. If less than the maximum number of PWR assemblies are loaded, dummy assemblies having a width, length, and weight similar to that of the PWR assemblies they are replacing, must be loaded in the empty guide tubes.

The SNF assemblies that are permitted in the W21 canister must meet all of the parameter requirements of at least one criticality class. Table 2 lists the dimensional and initial enrichment limits for each criticality class of PWR fuel assembly. Table 2 provides separate assembly initial ^{235}U enrichment limits for the W21-1 and W21-2 canister loading configurations. The initial enrichment limits presented in Table 2 are bounding for assemblies containing any type of control insert, including assemblies with fuel rods replaced with any type of rod of equal or greater diameter and height.

Table 3 lists minimum required cooling times, as a function of burnup, for PWR assemblies loaded into the W21 canister. For a given fuel burnup level, assembly radiation sources increase with decreasing initial enrichment. Table 3 lists two minimum initial enrichment values for each assembly burnup level. Table 3 also lists two different minimum allowable cooling times, corresponding to the two minimum initial enrichment levels. An assembly must have an initial enrichment level equal to or greater than the value shown in Table 3, to qualify for the corresponding minimum allowable cooling time also shown in Table 3. Assemblies with initial enrichment levels lower than the lowest values shown (for the assembly's burnup level) in Table 3 are not qualified for transportation in the W21 canister.

Table 3 also gives limits on the total amount of initial (pre-irradiation) cobalt that may be present in the assembly active fuel zone (including both assembly and control insert hardware). For assemblies with less than 11 grams of cobalt in the fuel zone, the shorter cooling times shown in Table 3 may be used (provided that the minimum initial enrichment requirement is also met). The longer cooling times shown in Table 3 must be used for assemblies with over 11 grams of cobalt in the fuel zone. Cobalt present in control components that do not extend into the assembly fuel zone (such as thimble plug assemblies) or that do not reside in the core during operation (such as control rod assemblies) do not need to be included in the total fuel zone cobalt content.

All PWR SNF assembly control inserts placed in the W21 canister must be intact, and may contain B_4C, borosilicate glass, silver-indium-cadmium, hafnium, or Gd_2O_3 poison materials. Control insert rod cladding, and other insert hardware may consist of any type of zircaloy, stainless steel, or inconel. Any PWR assembly control insert that meets these material requirements may be loaded into the W21 canister. Control inserts that employ solid inconel rods that reside in the core, such as the B&W Grey APSRA, are not qualified for transportation in the W21 canister. Any insert that contains significant quantities of inconel (such as inconel rod cladding) requires an evaluation of total assembly fuel zone cobalt quantity. Fuel rods may also be replaced with solid steel or Inconel rods, or rods containing any of the above poison materials, provided that the fuel zone cobalt requirements are met. UO_2 fuel rods containing Gd_2O_3 poison material are also permissible, although the poison is not relied upon to increase allowable ^{235}U initial enrichment levels for the fuel rod or assembly in question.

(ii) W74 Canister

The W74 canister contents are limited to 64 Big Rock Point (BRP) SNF assemblies without channels, including intact, partial, and damaged UO_2 and mixed oxide (MOX) fuel assemblies meeting the applicable acceptance criteria specified in Table 4 through Table 9. Specifications W74-1 and W74-2 for intact UO_2 and MOX fuel assemblies are provided in Table 4 and Table 5, respectively. Specifications W74-3 and W74-4 for partial UO_2 and MOX fuel assemblies are provided in Table 6 and Table 7, respectively. Lastly, specifications W74-5 and W74-6 for damaged UO_2 and MOX fuel assemblies are provided in Table 8 and Table 9, respectively. All UO_2 rods may contain any quantity of Gd_2O_3 poison material, provided that the specified ^{235}U initial enrichment limits are satisfied. BRP assemblies containing any amount of plutonium fuel (before irradiation) must meet the requirements of the MOX fuel specifications given in Table 5, Table 7, or Table 9. If less than the maximum number of BRP assemblies are loaded, dummy assemblies having a width, length, and weight similar to that of the BRP assemblies they are replacing, must be loaded in the empty guide tubes or support tubes.

The BRP UO_2 fuel assembly types permitted in the W74 canister are identified in Table 10. Any BRP fuel assemblies that do not meet all of the parameter requirements given for any fuel assembly class in Table 10 may only be loaded into the W74 canister damaged fuel can, as long as the requirements given in the applicable damaged fuel loading specification (W74-5 or W74-6) are still met. Any BRP fuel assemblies that meet all of the parameter requirements shown in Table 10, except for the requirement for the number of non-corner water holes, are classified as partial assemblies. The lower initial enrichment limits given in Specification W74-3 apply for those assemblies.

The specific BRP intact MOX fuel assembly types accommodated in the W74 canister are shown in Figure 1 through Figure 4. The specific BRP partial MOX fuel assembly types accommodated in the W74 canister are shown in Figure 5 through Figure 8. These figures show the maximum initial ^{235}U enrichment levels for the uranium present in all UO_2 and MOX fuel rods in each MOX assembly array. The figures also show the maximum overall weight percentage of PuO_2 in the initial MOX fuel rod (metal-oxide) material composition, with one exception. For the two MOX rods shown in Figure 4, the maximum total plutonium (metal) content, rather than the maximum overall weight percent of PuO_2, is specified. The limits on maximum burnup, maximum heavy metal loading, and minimum cooling time for each BRP MOX fuel type are shown in Table 11.

Table 1 - Generic Requirements for All W21 Canister PWR SNF Contents

Fuel Assembly Parameter	Requirement
Fuel Rod Cladding Material	Zircaloy 2, 4
Assembly Condition	Intact (1)
Maximum Assembly Width (inch)	8.54
Maximum Burnup Level (MWd/MTU)	60,000 (2)
Maximum Uranium Loading (MTU/assy)	0.471
Axial Uranium Loading (kg/assy-inch)	3.27
Maximum Fuel Zone Height (inch)	150
Maximum Fuel Pellet Stack Density	96.5% (3)
Minimum Bottom Nozzle Height (inch)	1.97 (4)

Notes:

(1) Intact assemblies have no known or suspected fuel rod cladding defects greater than pinhole leaks and hairline cracks. Intact fuel also has no detectable grid spacer damage, or axial shifting in grid spacer location. Fuel assemblies with missing fuel rods (from the standard rod array configuration) may be loaded if all missing fuel rods are replaced with dummy rods that have a height and diameter at least as great as that of a standard fuel rod (i.e., by rods that displace an equal or greater volume of water).

(2) For assembly burnups exceeding 45,000 MWd/MTU, it is necessary to verify that the cladding oxide layer thickness does not exceed 70 μm, by measurement of a statistical sample of limiting fuel assemblies. The exposure (burnup) of any inserted control component must not exceed that of the host fuel assembly.

(3) Defined as the average material density within the cylindrical envelope volume covered by the fuel pellets, relative to the theoretical UO_2 density of 10.97 g/cc. Thus, "smearing" over fuel pellet dishes and chamfers to determine the "stack" density is acceptable.

(4) The bottom nozzle height is defined as the distance between the assembly bottom and the bottom of the active fuel.

Table 2 - W21 Canister SNF Assembly Dimensional and Enrichment Limits

Fuel Assembly Class[1]	Criticality Class[1]	Max. Initial Enrichment (w/o ^{235}U)[2]		Number of Fuel Rods	Min. Clad O.D. (in.)	Min. Clad Thickness (in.)	Min. Pellet Diameter (in.)	Fuel Rod Pitch (in.)	No. Guide / Instrument Tube Locations[5]
		W21-1[3]	W21-2[4]						
B&W 15x15	B&W 15x15	4.70	5.00	208	0.4300	0.0265	0.3675	0.568	17
B&W 17x17	B&W 17x17	4.60	4.90	264	0.3770	0.0220	0.3232	0.502	25
CE 14x14	CE 14x14	5.00	5.00	176	0.4400	0.0260	0.3700	0.580	5[6]
	CE 14x14 A	5.00	5.00	176	0.4400	0.0260	0.3795	0.568	5[6]
Palisades	CE 15x15 P	5.00	5.00	208 - 216	0.4135	0.0240	0.3500	0.550	1-9
Yankee Rowe	15x16	5.00	5.00	231	0.3650	0.0240	0.3105	0.472	1
	15x16 A	5.00	5.00	237	0.3650	0.0240	0.3105	0.468	1
CE 16x16 CE System 80 St. Lucie 2	CE 16x16	5.00	5.00	236	0.3820	0.0250	0.3250	0.506	5[6]
WE 14x14	WE 14x14	5.00	5.00	179	0.4000	0.0243	0.3444	0.556	17
WE 15x15	WE 15x15	4.70	5.00	204	0.4200	0.0240	0.3569	0.563	21
	WE 15x15 A	4.90	5.00	204	0.4240	0.0300	0.3565	0.563	21
WE 17x17	WE 17x17	4.70	5.00	264	0.3740	0.0225	0.3195	0.496	25
	WE 17x17 A	4.60	4.90	264	0.3600	0.0225	0.3088	0.496	25
	WE 17x17 B	4.60	4.90	264	0.3600	0.0250	0.3030	0.496	25

Notes:

(1) Assembly class defined per Energy Information Administration, *Spent Nuclear Fuel Discharges from U.S. Reactors 1993*, U.S. Department of Energy, 1995. The fuel assembly criticality classes are arbitrary designations given to each set of assembly parameters that are evaluated for criticality.

(2) The maximum allowable enrichments apply for all assemblies that meet the specified physical parameter requirements for the defined assembly class. The maximum allowable enrichments are defined as the maximum planar average enrichment at any axial assembly location. An exception is the CE 15x15 P assembly class, for which the maximum allowable enrichment applies to each individual fuel pin within the assembly.

(3) This enrichment limit applies for up to 21 SNF assemblies, in any W21 canister guide tube.

(4) This enrichment limit applies for up to 20 SNF assemblies, with the center guide tube empty.

(5) Whereas the number of guide tube locations is a specified parameter, the materials and dimensions of the guide tubes are not specified, since any quantity of steel or zircaloy in the guide tube locations will reduce assembly reactivity. Guide tube locations may contain nothing, hollow zircaloy or stainless rods (or rod clusters), solid zircaloy or stainless rods (or rod clusters), or poison rods (or rod clusters).

(6) The CE 14x14 and CE 16x16 assembly guide tubes occupy four fuel rod locations within the assembly array.

Table 3 - W21 Canister Minimum PWR Assembly Cooling Time Requirements

Assembly Burnup Level (GWd/MTU)[(1)]	Assembly Initial Enrichment (w/o ^{235}U)[(1)]	Assembly Fuel Zone Cobalt Qty (g/assy)[(2)]	Required Cooling Time (years)
≤35	≥2.8 %	≤ 11	≥ 6
≤40	≥3.0 %	≤11	≥ 8
≤45	≥3.3 %	≤11	≥ 10
≤50	≥3.5 %	≤11	≥ 12
≤55	≥3.8 %	≤11	≥ 15
≤ 60	≥4.0 %	≤11	≥18
≤35	≥1.5 %	≤50	≥15
≤40	≥1.5 %	≤50	≥20
≤45	≥1.5 %	≤50	≥25
≤50	≥ 2.5 %	≤50	≥25
≤55	≥3.0 %	≤ 50	≥25
≤60	≥3.5 %	≤50	≥25

Notes:

(1) Assembly average values.
(2) Defined as the total initial (pre-irradiation) cobalt mass within the assembly fuel zone, including any cobalt present in inserted control components.

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Table 4 - W74 Canister Contents Specification W74-1
Intact UO_2 Fuel Assemblies

SNF Parameter	Loading/Acceptance Criteria
Payload Description	≤64 Big Rock Point BWR intact UO_2 fuel assemblies. (1,2,3) Any remaining empty canister basket guide tubes and/or support tubes may be loaded with fuel assemblies meeting any of the acceptable payload specifications W74-2 through W74-6, subject to the limitations of those specifications.
Cladding Material/Condition	Zircaloy 2,4 cladding with no known or suspected cladding defects greater than hairline cracks or pinhole leaks.
Maximum Uranium Loading	≤142.1 kg/assembly.
Maximum Initial Enrichment(4)	≤4.10 w/o ^{235}U.
Minimum Assembly Average Initial Enrichment	≥3.0 w/o ^{235}U.
Maximum Burnup	≤32,000 MWd/MTU.
Minimum Cooling Time	≥6.0 years.(5)

W74-1 Notes:

(1) Loaded assemblies must meet all of the assembly geometry requirements specified in Table 10, for any one of the defined assembly classes.

(2) Intact fuel assemblies include those BRP fuel assemblies with 1 to 4 corner rods missing, and BRP 9x9 fuel assemblies with 1 rod missing from a non-corner location. This includes assemblies with partial length rods, or rod fragments inside stainless tubes, in any of the array corner locations. It also includes 9x9 assemblies with 11x11 assembly rods in corner locations.

(3) Intact UO_2 assemblies may have any number of fuel rods replaced with solid zircaloy or stainless steel rods, or with poison rods, given that the length and diameter of the replacement rod are at least as great as that of the fuel rod. The empty array or guide tube locations may contain nothing, hollow zircaloy or stainless steel rods, neutron source rods, or any similar non-fissile fuel assembly component.

(4) Defined as the maximum array-average enrichment, which is the peak planar average initial enrichment considering all elevations along the assembly axis.

(5) If an intact UO_2 assembly has been further irradiated after having fuel rods replaced by dummy stainless rods, an evaluation must be performed that shows that the active fuel region non-fuel gamma source strength is bounded by that described in Section 5.2.2.1 of the WSNF-123 SAR. A similar evaluation is required for any assembly containing over 2.9 grams of initial cobalt in the assembly fuel zone.

Table 5 - W74 Canister Contents Specification W74-2
Intact MOX Fuel Assemblies

SNF Parameter	Loading/Acceptance Criteria
Payload Description	≤64 Big Rock Point BWR intact MOX fuel assemblies. (1,2,3) Any remaining empty canister basket guide tubes and/or support tubes may be loaded with fuel assemblies meeting any of the acceptable payload specifications W74-1 and W74-3 through W74-6, subject to the limitations of those specifications.
Cladding Material/Condition	Zircaloy 2,4 cladding with no known or suspected cladding defects greater than hairline cracks or pinhole leaks.
Maximum Heavy Metal Loading	The heavy metal loading varies by MOX assembly type and must not exceed the maximum values defined in Table 11.
Allowable Fuel Composition	Maximum initial ^{235}U enrichment and maximum PuO_2 weight percentage is shown for every fuel rod location in the MOX assembly array in Figure 1 through Figure 4. (4,5)
Maximum Burnup	The burnup varies by MOX assembly type and must not exceed the maximum values defined in Table 11.
Minimum Cooling Time	The cooling time varies by MOX assembly type and must not be less than the minimum values defined in Table 11. (6)

W74-2 Notes:

(1) Intact MOX assemblies may have any number of fuel rods replaced with solid zircaloy or stainless steel rods, or with poison rods, given that the length and diameter of the replacement rod are at least as great as that of the fuel rod. They may also have hollow zircaloy or stainless steel rods, neutron source rods, or any similar non-fissile fuel assembly component placed in the empty array or guide tube locations, including all forms of inserts or control components.

(2) J2 (Figure 1) MOX assemblies must meet all of the assembly geometry requirements shown for Siemens 9x9 fuel in Table 10. DA and G-Pu (Figure 2 and Figure 3, respectively) MOX assemblies must meet all of the assembly geometry requirements shown for Siemens 11x11 fuel in Table 10. One exception is that J2 MOX assemblies with a cladding thickness of 0.05 inches and a fuel pellet diameter of 0.4515 inches are also acceptable. UO_2 9x9 assemblies with 2 inserted MOX rods (shown in Figure 4) must meet all of the assembly geometry requirements shown for Siemens 9x9 in Table 10.

(3) Intact G-Pu MOX assemblies may have 0 to 4 fuel rods in the array corner locations. G-Pu MOX assemblies may also have partial length rods, or rod fragments inside stainless tubes, in any of the array corner locations.

(4) The maximum ^{235}U enrichment shown in Figure 1 through Figure 4 is defined as the weight percentage of ^{235}U in any uranium that is present in the rod. The PuO_2 weight percentage is the overall mass of PuO_2 in the rod divided by the overall metal-oxide ($UO_2 + PuO_2$) mass in the rod. Fuel rods in candidate assemblies may have 235U enrichment levels and PuO_2 weight percentages that are equal to or less than the values shown in Figure 1 through Figure 4 for that fuel rod array location.

(5) Figure 4 specifies a maximum total MOX fuel rod plutonium metal mass as opposed to a maximum PuO_2 weight percentage.

(6) If an intact MOX assembly has been further irradiated after having fuel rods replaced by dummy stainless rods, an evaluation must be performed that shows that the active fuel region non-fuel gamma source strength is bounded by that described in Section 5.2.2.1 of the WSNF-123 SAR. A similar evaluation is required for any assembly containing over 2.9 grams of initial cobalt in the assembly fuel zone.

Table 6 - W74 Canister Contents Specification W74-3
Partial UO_2 Fuel Assemblies

SNF Parameter	Limit/Specification
Payload Description	≤64 Big Rock Point BWR partial UO_2 fuel assemblies.[1,2] Partial fuel assemblies are defined as those assemblies having one or more full-length fuel rods missing from the intact fuel assembly array (except as permitted by W74-1 Notes 2 and 3). The affected array locations may contain nothing, partial length rods, hollow zircaloy or stainless steel rods, neutron source rods, or any other non-fissile fuel assembly component with a lower length or diameter than a full-length fuel rod. Any remaining empty canister basket guide tubes and/or support tubes may be loaded with fuel assemblies meeting any of the acceptable loading specifications W74-1, W74-2, and W74-4 through W74-6, subject to the limitations of those specifications.
Cladding Material/Condition	Zircaloy 2,4 cladding with no known or suspected cladding defects greater than hairline cracks or pinhole leaks.
Maximum Uranium Loading	≤142.1 kg/assembly
Maximum Initial Enrichment[3]	≤3.55 w/o ^{235}U (9x9) ≤3.6 w/o ^{235}U (11x11)
Minimum Assembly Average Initial Enrichment	≥3.0 w/o ^{235}U
Maximum Burnup	≤32,000 MWd/MTU
Minimum Cooling Time	≥6.0 years[4]

W74-3 Notes:

(1) Partial UO_2 assemblies may have any number of fuel rods replaced with solid zircaloy or stainless steel rods, or with poison rods.

(2) Loaded partial assemblies must meet all of the geometry requirements shown (for any of the defined assembly classes) in Table 10, except for the "maximum number of non-corner water holes."

(3) Defined as the maximum array average initial enrichment, which is the peak planar average initial enrichment considering all elevations along the fuel assembly axis. The averaging is applied only to those fuel rods that are present in the partial array.

(4) If a partial UO_2 assembly has been further irradiated after having fuel rods replaced by dummy stainless rods, an evaluation must be performed that shows that the active fuel region non-fuel gamma source strength is bounded by that described in Section 5.2.2.1 of the WSNF-123 SAR. A similar evaluation is required for any assembly containing over 2.9 grams of initial cobalt in the assembly fuel zone.

Table 7 - W74 Canister Contents Specification W74-4
Partial MOX Fuel Assemblies

SNF Parameter	Loading/Acceptance Criteria
Payload Description	≤64 Big Rock Point BWR partial MOX fuel assemblies.[1,2,3] Partial MOX assemblies must conform exactly to one of the four partial assembly array configurations shown in Figure 5 through Figure 8, with respect to the number and location of missing fuel rods within the assembly array. The missing fuel rod array locations may contain nothing, hollow zircaloy or stainless steel rods, neutron source rods, or any other non-fissile fuel assembly component. Any remaining empty canister basket guide tubes and/or support tubes may be loaded with fuel assemblies meeting any of the acceptable loading specifications W74-1 through W74-3, W74-5, and W74-6, subject to the limitations of those specifications.
Cladding Material/Condition	Zircaloy 2,4 cladding with no known or suspected cladding defects greater than hairline cracks or pinhole leaks.
Maximum Heavy Metal Loading	The heavy metal loading varies by fuel assembly type and must not exceed the maximum values defined in Table 11.
Allowable Fuel Composition	Maximum initial ^{235}U enrichment and maximum PuO_2 weight percentage is shown for every fuel rod location (in each of the four allowable partial MOX assembly array configurations) in Figure 5 through Figure 8.[4]
Maximum Burnup	The burnup varies by MOX assembly type and must not exceed the maximum values defined in Table 11.
Minimum Cooling Time	The cooling time varies by MOX assembly type and must not be less than the minimum values defined in Table 11.

<u>W74-4 Notes:</u>

(1) Partial MOX assemblies may have any number of fuel rods replaced with solid zircaloy or stainless steel rods, or with poison rods, given that the length and diameter of the replacement rod are at least as great as that of the fuel rod.

(2) If a partial MOX assembly has been further irradiated after having fuel rods replaced by dummy stainless rods, an evaluation must be performed that shows that the active fuel region non-fuel gamma source strength is bounded by that described in Section 5.2.2.1 of the WSNF-123 SAR. A similar evaluation is required for any assembly containing over 2.9 grams of initial cobalt in the assembly fuel zone.

(3) Loaded partial assemblies must meet all of the geometry requirements shown (for any of the defined assembly classes) in Table 10, except for the "maximum number of non-corner water holes."

(4) The maximum ^{235}U enrichment shown in Figure 5 through Figure 8 is defined as the weight percentage of ^{235}U in any uranium that is present in the rod. The PuO_2 weight percentage is the overall mass of PuO_2 in the rod divided by the overall metal-oxide (UO_2 + PuO_2) mass in the rod. Fuel rods in candidate assemblies may have ^{235}U enrichment levels and PuO_2 weight percentages that are equal to or less than the values shown in Figure 5 through Figure 8 for that fuel rod array location.

Table 8 - W74 Canister Contents Specification W74-5
Damaged UO_2 Fuel Assemblies

SNF Parameter	Limit/Specification
Payload Description	≤8 Big Rock Point BWR damaged UO_2 fuel assemblies. Damaged fuel assemblies are defined as those with fuel cladding damage in excess of hairline cracks or pinhole leaks. Fuel assemblies with damaged grid spacers (defined as damaged to a degree where fuel rod structural integrity cannot be assured, or where grid spacers have moved from their design position) are also considered to be damaged fuel assemblies. Each fuel assembly designated as damaged must be placed within a damaged fuel can and loaded into a basket support tube in the upper or lower basket. The remaining empty canister basket guide tubes and support tubes may be loaded with fuel assemblies meeting any of the acceptable loading specifications W74-1 through W74-4 and W74-6, subject to the limitations of those specifications, for a total of ≤64 Big Rock Point BWR fuel assemblies. Any intact or partial UO_2 fuel assembly that does not meet all of the assembly geometry requirements shown in Table 10 (other than the number of water holes) must also be loaded into a damaged fuel can.
Cladding Material/Condition	Zircaloy 2,4 cladding with fuel rod damage in excess of hairline cracks or pinhole leaks.
Maximum Uranium Loading	≤142.1 kg/assembly.
Maximum Initial Enrichment	≤4.61 w/o ^{235}U peak fuel pellet initial enrichment.
Maximum Pellet Density	≤96.5% (as defined in Table 10, Note 1).
Minimum Assembly Average Initial Enrichment	≥3.0 w/o ^{235}U
Maximum Burnup	≤32,000 MWd/MTU.
Minimum Cooling Time	≥6.0 years.[(1)]

W74-5 Note:

(1) If a damaged UO_2 assembly has been further irradiated after having fuel rods replaced by dummy stainless rods, an evaluation must be performed that shows that the active fuel region non-fuel gamma source strength is bounded by that described in Section 5.2.2.1 of the WSNF-123 SAR. A similar evaluation is required for any assembly containing over 2.9 grams of initial cobalt in the assembly fuel zone.

Table 9 - W74 Canister Contents Specification W74-6 Damaged MOX Fuel Assemblies

SNF Parameter	Limit/Specification
Payload Description	≤ 8 Big Rock Point BWR damaged MOX fuel assemblies. Damaged fuel assemblies are defined as those with fuel cladding damage in excess of hairline cracks or pinhole leaks. Fuel assemblies with damaged grid spacers (defined as damaged to a degree where the fuel rod structural integrity cannot be assured, or where the grid spacers have shifted vertically from their design position) are also considered to be damaged fuel assemblies. Each fuel assembly designated as damaged must be placed within a damaged fuel can and loaded into a support tube locations in the upper and lower basket. The remaining empty canister basket guide tubes and support tubes may be loaded with fuel assemblies meeting any of the acceptable loading specifications W74-1 through W74-5, subject to the limitations of those specifications, for a total of ≤64 Big Rock Point BWR fuel assemblies. Any intact or partial MOX assembly that does not meet all of the assembly geometry requirements shown in Table 10 (other than the number of water holes) must also be loaded into a damaged fuel can.[1]
Cladding Material/ Condition	Zircaloy 2,4 cladding with fuel rod damage in excess of hairline cracks or pinhole leaks.
Maximum Pellet Density	96.5% (as defined in Table 10, Note 1)
Maximum Heavy Metal Loading	The heavy metal loading varies by MOX assembly type and must not exceed the maximum values defined in Table 11.
Allowable Fuel Composition	≤4.61 w/o ^{235}U for all UO_2 fuel pellets. All MOX fuel pellets must meet the maximum ^{235}U enrichment and PuO_2 weight percentage requirements for one of the four MOX fuel material compositions described in Figure 1 through Figure 3.
Maximum Burnup	The burnup varies by MOX assembly type and must not exceed the maximum values defined in Table 11.
Minimum Cooling Time	The cooling time varies by MOX assembly type and must not be less than the minimum values defined in Table 11.[2]

W74-6 Notes:

(1) The UO_2 9x9 assemblies with 2 inserted MOX rods (shown in Figure 4) may not be loaded into the W74 damaged fuel can.

(2) If a damaged MOX assembly has been further irradiated after having fuel rods replaced by dummy stainless rods, an evaluation must be performed that shows that the active fuel region non-fuel gamma source strength is bounded by that described in Section 5.2.2.1 of the WSNF-123 SAR. A similar evaluation is required for any assembly containing over 2.9 grams of initial cobalt in the assembly fuel zone.

Table 10 - W74 Canister Fuel Geometry Specifications

Fuel Assembly Parameter	Fuel Assembly Class			
	GE 9x9	Siemens 9x9	Siemens 11x11	Siemens 11x11A
Fuel Pellet Stack Density(1)	≤96.5%	≤96.5%	≤96.5%	≤96.5%
Number of Fuel Rods	≤ 81	≤ 81	≤121	≤ 121
Clad O.D. (in)	0.5625	0.5625	0.449	0.449
Clad Thickness (in)	0.040	0.040	0.034	0.034
Pellet Diameter (in)	0.471	0.4715(2)	0.3715	0.3735
Fuel Rod Pitch (in)	0.707	0.707	0.577	0.577
Active Fuel Length (in)	≤70	≤70	≤70	≤ 70
Number of Array Corner Rods(3)	0-4	0-4	0-4	0-4
Number of Non-Corner Water Holes(3)	≤ 1	0	0	0
Number of Inert Rods(3)	≥0	≥ 0	≥0	≥0
Bottom Tie Plate Height (in)(4)	≥1.25	≥1.25	≥1.25	≥1.25

Notes:

(1) The fuel pellet stack density is defined as the average density of the fuel pellet material (within the cylindrical envelope volume covered by the pellet stack) divided by the theoretical UO_2 density of 10.97 g/cc. Thus, smearing the fuel material over the dishing and chamfer voids in the pellet stack is acceptable for determining the stack density.

(2) Assemblies E65 and E72 may each contain two MOX fuel rods with either solid pellets or annular pellets with a 0.1 inch or 0.2 inch inside diameter. In any given MOX fuel rod, the entire pellet stack must contain the same pellet type (i.e., solid, 0.1-inch annular, or 0.2-inch annular).

(3) The definitions of corner rods, non-corner rods, and inert rods are given in the W74-1 and W74-3 assembly loading specifications.

(4) Defined as the distance from the bottom of the assembly to the bottom of the active fuel.

Table 11 - W74 Canister Assembly Specific Requirements for Big Rock Point MOX Fuel

BRP MOX Assembly Type	Maximum Heavy Metal Loading (kg)	Maximum Burnup (MWd/MTIHM)[1]	Minimum Cooling Time (years)
J2 (9x9)	124	22,820	22
DA (11x11)	126	21,850	22
G-Pu (11x11)	127	34,220	15
UO_2 9x9 with 2 inserted MOX rods	142.1	32,000	6

Note:

(1) The exposure (burnup) of any inserted control component must not exceed that of the host fuel assembly.

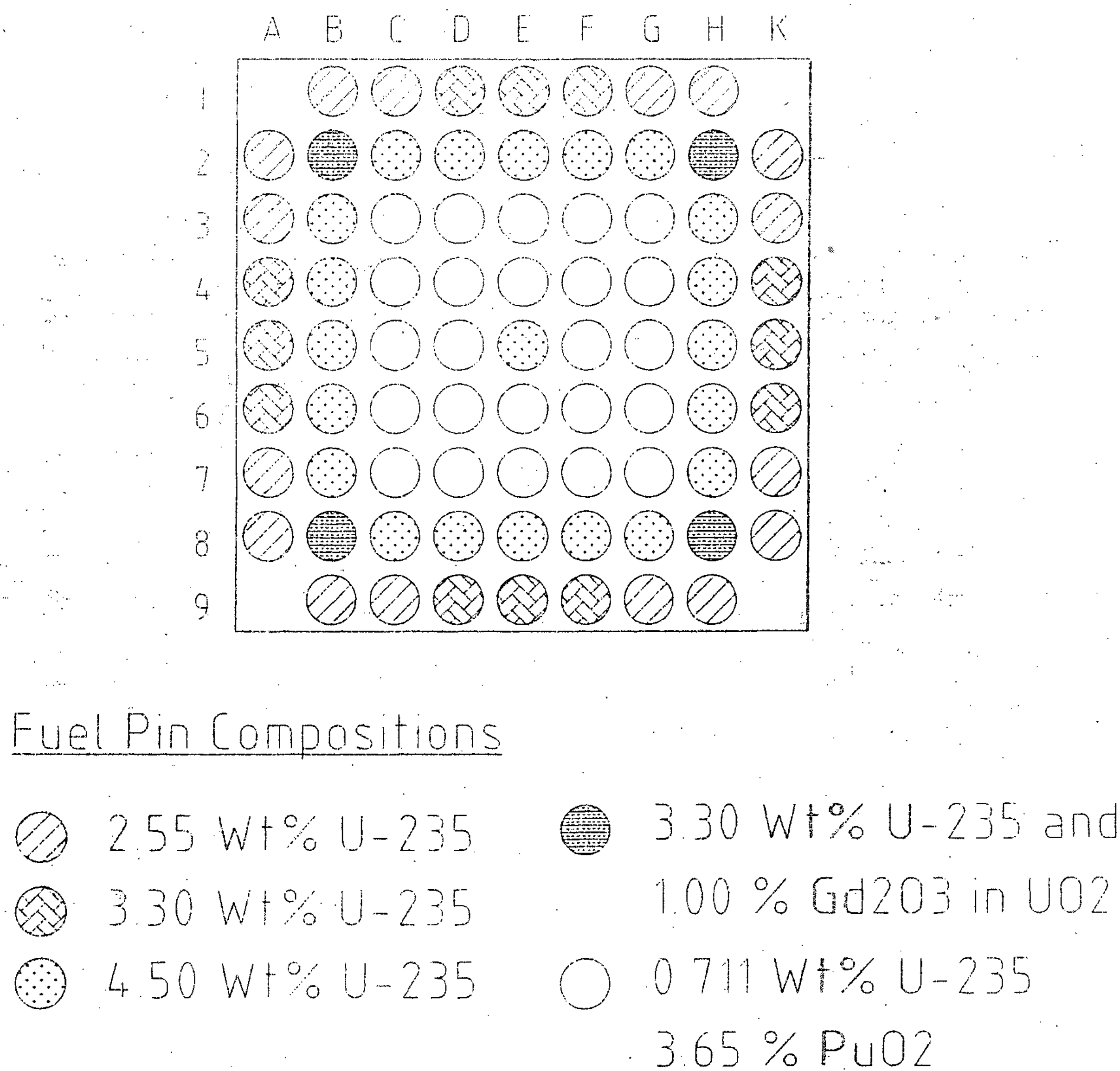
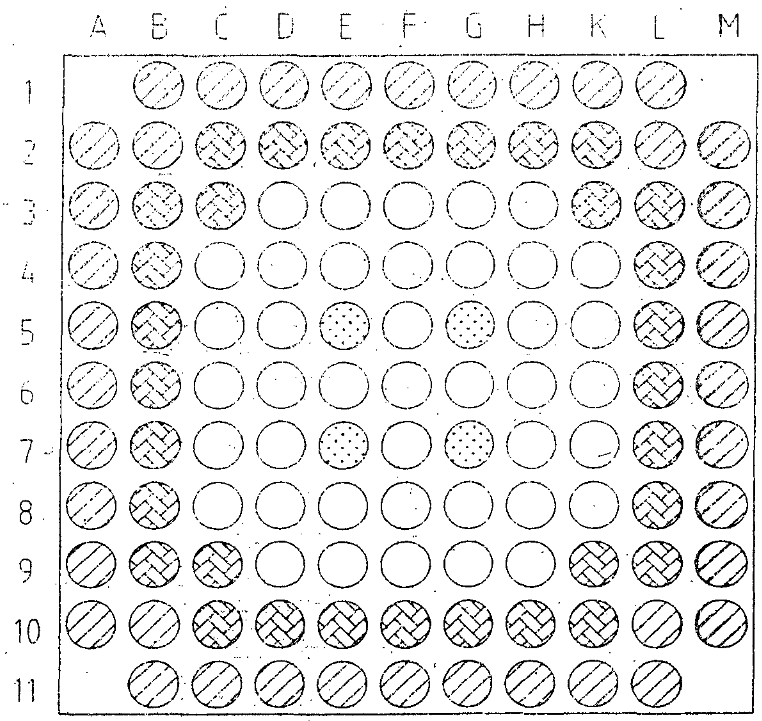


Figure 1 - J2 (9x9) BRP MOX Assembly Array



Fuel Pin Compositions

2.40 Wt% U-235

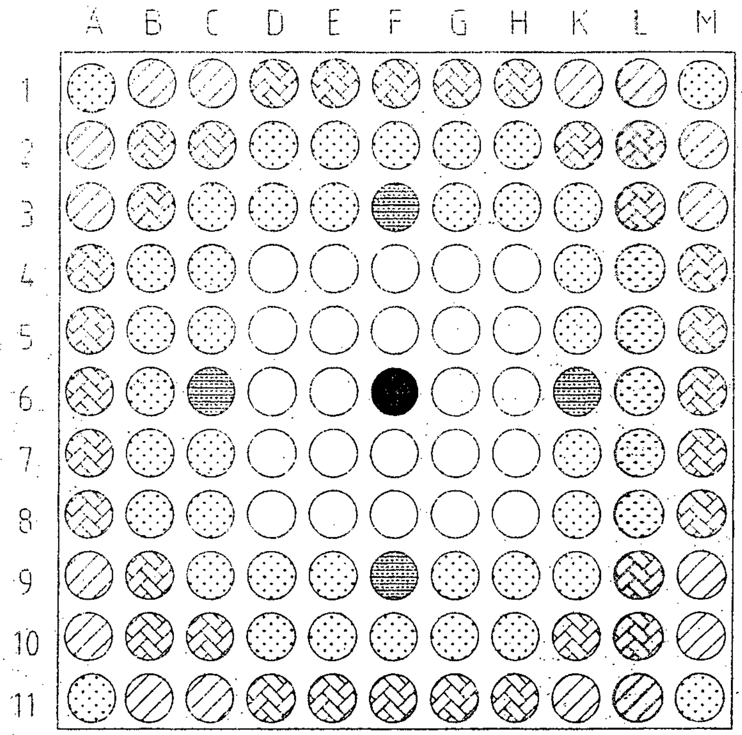
1.56 Wt% U-235
1.03 Wt% PuO2

2.40 Wt% U-235
2.45 Wt% PuO2

Water Rods

Note: Water rods are identical to the fuel rods (same diameter and cladding thickness), except that they contain no fuel pellets.

Figure 2 - DA (11x11) BRP MOX Assembly Array

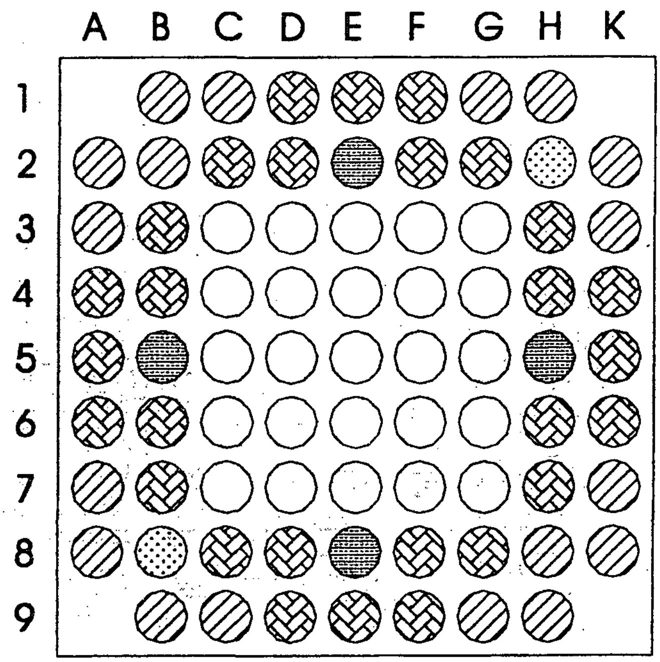


Fuel Pin Compositions

2.30 Wt% U-235
3.20 Wt% U-235
4.60 Wt% U-235
Solid Zirc Rod
4.60 Wt% U-235
1.20 Wt% Gd2O3
0.711 Wt% U-235
5.45 Wt% PuO2

Note: G-Pu assemblies may have any number of fuel rods missing (or present) in the four array corner locations

Figure 3 - G-Pu (11x11) BRP MOX Assembly Array



Fuel Pin Compositions

2.50 Wt% U-235

3.40 Wt% U-235

0.711 Wt% U-235
25.4 g/rod Pu

3.40 Wt% U-235
2.00 Wt% Gd203 in UO2

4.5 Wt% U-235

Figure 4 - UO2 9x9 BRP Assembly with Two Inserted MOX Rods

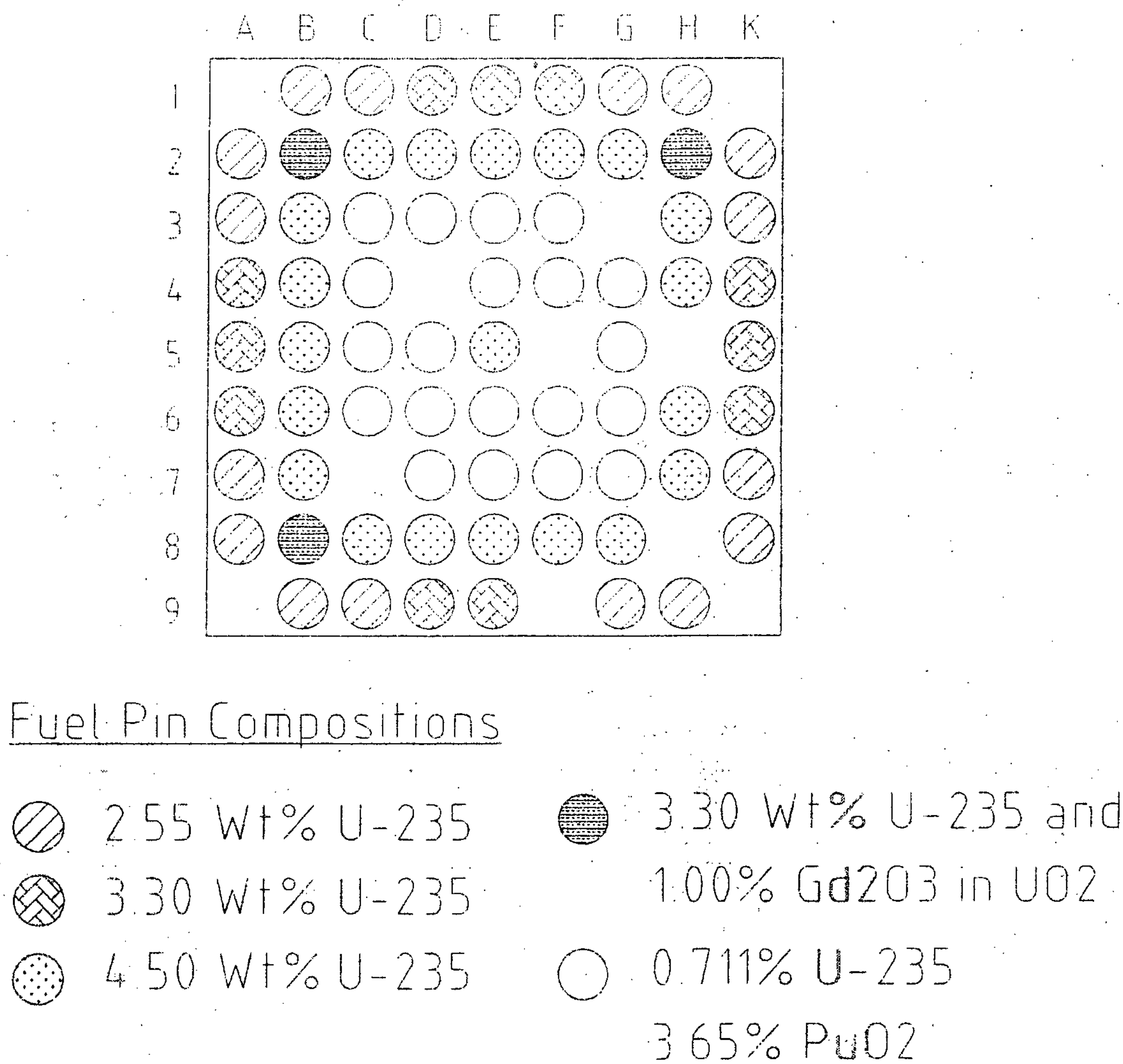


Figure 5 - J2 Partial MOX Assembly Array #1

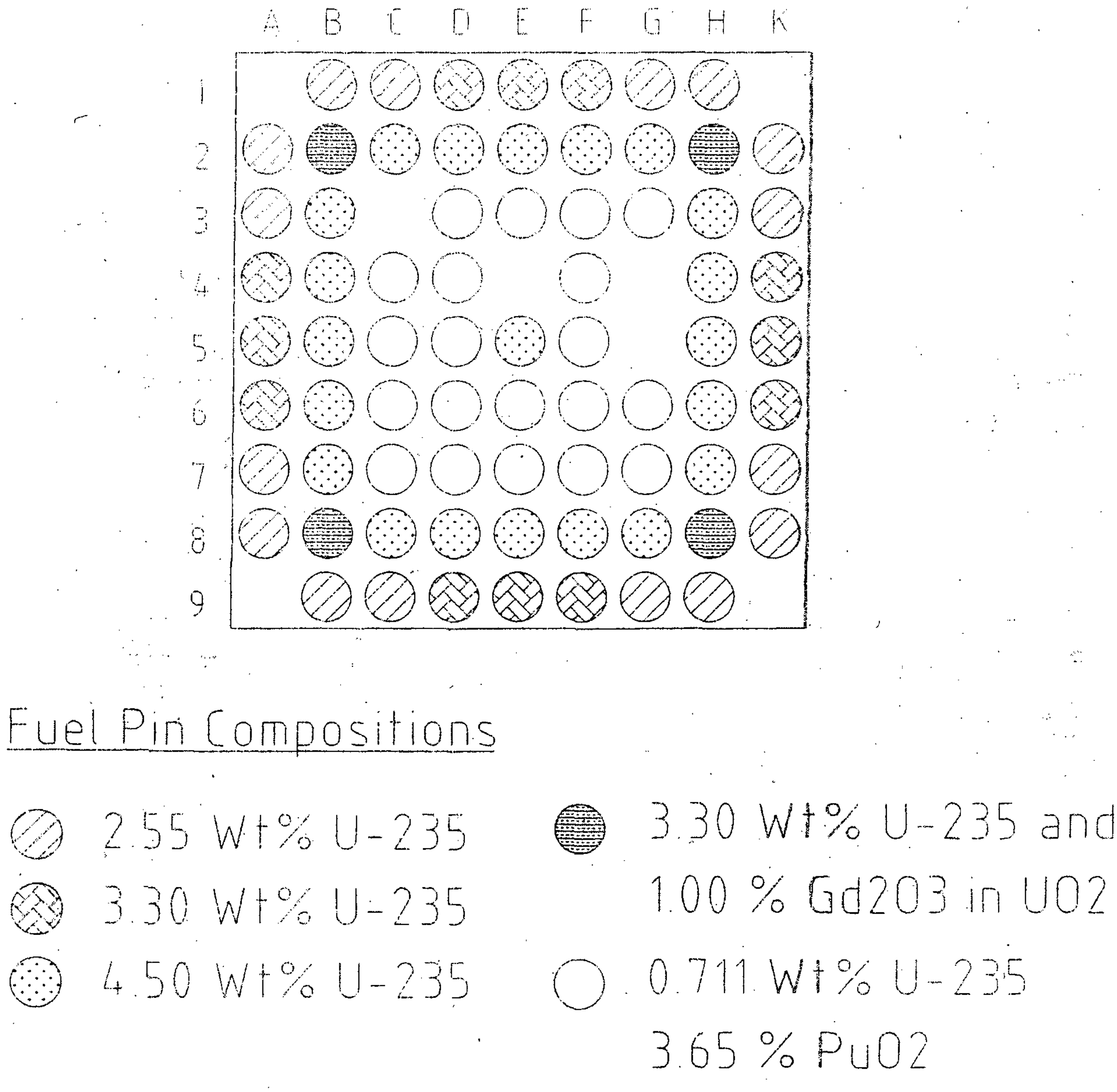


Figure 6 - J2 Partial MOX Assembly Array #2

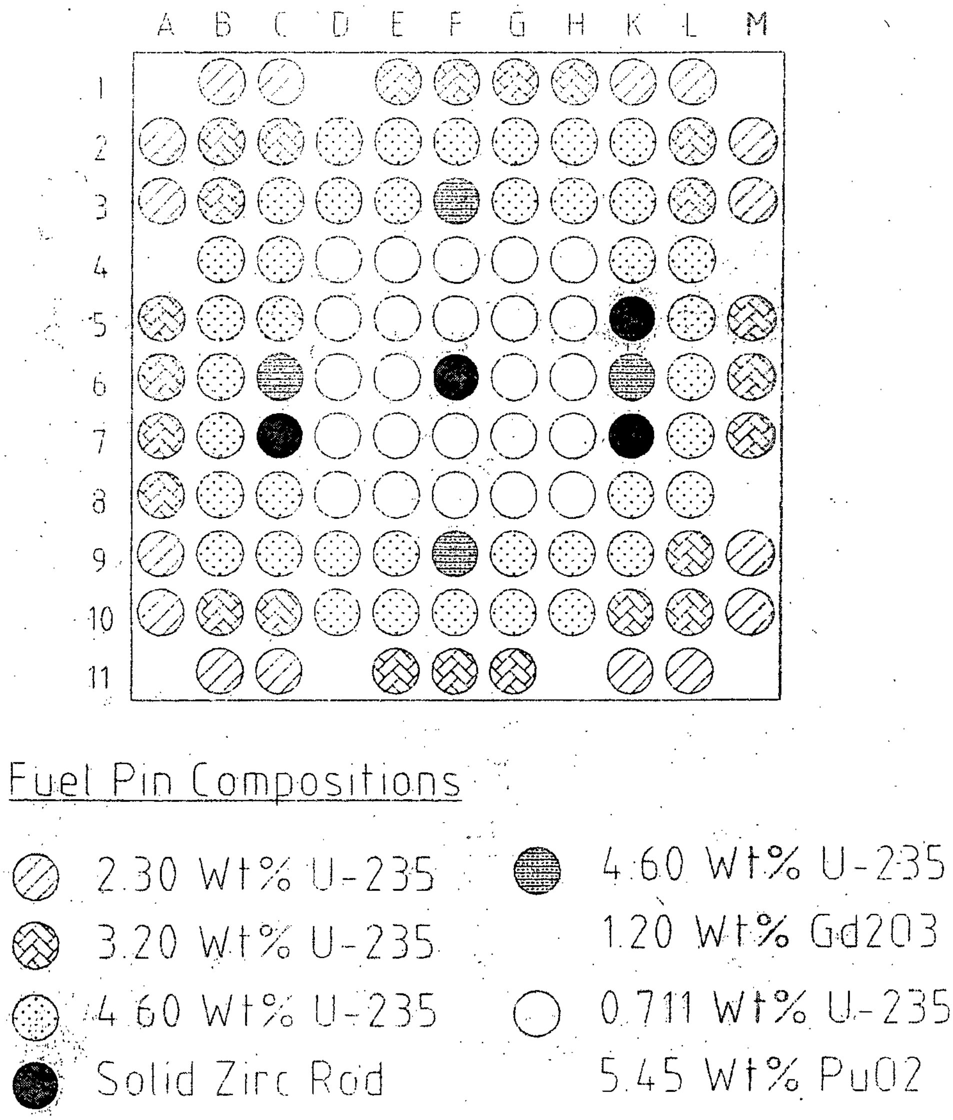
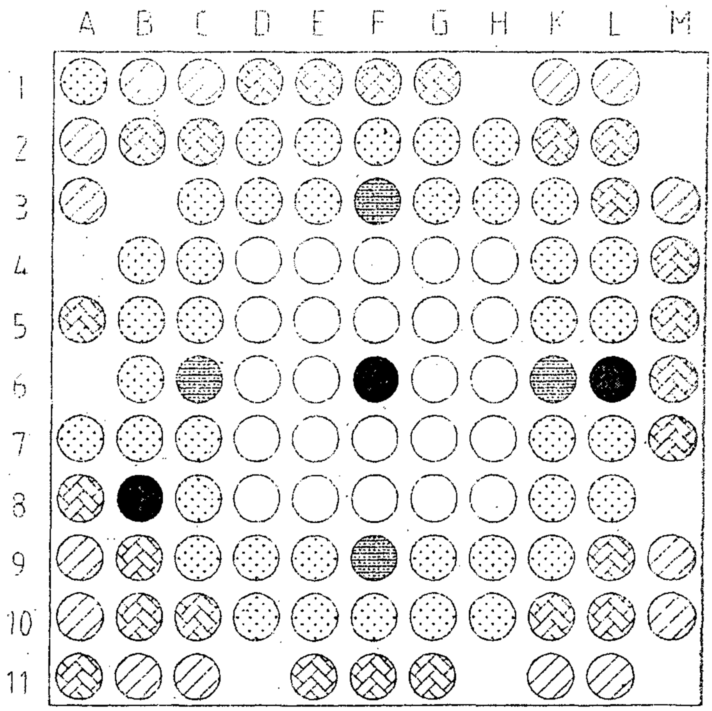


Figure 7 - G-Pu Partial MOX Assembly Array #1



Fuel Pin Compositions

2.30 Wt% U-235
3.20 Wt% U-235
4.60 Wt% U-235
Solid Zirc Rod
4.60 Wt% U-235
1.20 Wt% Gd2O3
0.711 Wt% U-235
5.45 Wt% PuO2

Figure 8 - G-Pu Partial MOX Assembly Array #2

(2) Maximum Quantity of Material Per Package

The maximum payload weight of the TS125 Transportation cask is 85,000 pounds. The payload weight includes the weight of the FuelSolutions™ canister and its SNF payload, plus the weight of the cask cavity spacer for short canisters.

(3) Decay Heat Limit

The W74 canister loading criteria can be described as follows:

A Big Rock Point spent fuel assembly is allowed to be shipped in the canister if Q (heat generation per assembly) $\leq$ 0.275 kW.

No decay heat limit is specified for the W21 canister. The PWR assembly fuel parameters requirements given in Table 3 ensure that assembly heat generation levels will not exceed the heat generation level that was analyzed in the thermal licensing evaluations (1.05 kW/assembly).

(c) Criticality Safety Index

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

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

(1) The package shall meet the Acceptance Tests and Maintenance Program of Chapter 8 of the application, as supplemented.

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

7. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17, provided the fabrication of the packagin was satisfactorily completed by December 31, 2006.

8. Transport by air of fissile material is not authorized.

9. Revision No. 2 of this certificate may be used until November 31, 2008.

10. Expiration date: October 31, 2012

REFERENCES

BNFL Fuel Solutions Corporation, application dated April 20, 2001.

Supplements dated June 7, 2001; January 22, February 5, February 28, April 11, and April 30,2002; January 17, August 7, and November 26, 2003; and April 20, April 28, April 29, May 7, May 12, 2004, and August 27, 2007.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Robert A. Nelson, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: October 31, 2007

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE	PAGES
9277	3	71-9277	USA/9277/B( )F	1 OF	5

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*
General Atomics
3550 General Atomics Court
San Diego, CA 92121

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION
Public Service Company of Colorado
application dated March 28, 1996, as supplemented

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

5.
(a) Packaging

(1) Model No.: FSV-1 Unit 3

(2) Description

The FSV-1 Unit 3 is a stainless steel-encased, depleted uranium-shielded cask. The cask body is a cylinder 208-inches long and 28-inches in diameter, except for the top flange area, which is 31-inches in diameter. The cavity is approximately 17.7-inches in diameter and 187.6-inches long.

The cask may be used in one of seven configurations (A through G) depending on contents. Configurations A, B, C, and D are used to ship solid, non-fissile irradiated hardware. These configurations use an outer lid consisting of a 3.75-inch thick stainless steel plate and a 2.25-inch thick depleted uranium shield. The lid is bolted to the cask body by 24 1.25-inch diameter fasteners. The primary seal is a silicone elastomeric seal ring between the outer lid and cask body. Configuration B does not require an inner container. Configuration C uses a supplemental stainless steel shield ring and cover plate. Configuration D uses a supplemental carbon steel shield ring and cover plate.

Configuration E is used to ship Fort St. Vrain (FSV) high temperature gas reactor (HTGR) fuel elements. This configuration uses the stainless steel inner container (as shown in General Atomic Drawing Nos. GADR 55-2-1, Rev. C, and GADR 55-2-2, Rev. A) as the containment vessel. The inner container lid is a stainless steel shell containing depleted uranium 4.15-inches thick. The inner lid is secured to the inner container body by 12 0.5-inch diameter fasteners. The primary seal is a silicone elastomeric seal ring between the inner lid and inner container body. Configuration E is equipped with an impact limiter on the upper end.

Configuration F and G are used to shop solid non-fissile irradiated and contaminated hardware from the FSV HGTR. These configurations use a 4.75-inch thick steel outer lid. The lid is secured to the cask body by 24 1.25-inch diameter fasteners. The primary seal is a molded silicone elastomeric seal ring between the outer lid and the cask body. Configurations F and G both use an impact limiter on the upper end. Configurations F and G also use a burial canister with a 12-inch thick carbon steel plug. The shielded spacer in the burial canister is used only in Configuration G.

The overall weight for the FSV-1 Unit 3 package is 46,025 pounds for Configurations A, B, C, and D and 47,600 pounds for Configurations E, F, and G.

(3) Drawings

The FSV-1 Unit 3 package is constructed in accordance with the following drawings:

Configuration A

National Lead Company Drawing Nos. : 70086F, Rev. 7; 70296F, Rev. 2; and General Atomics Drawing No. 1501-003, Rev. C

Configuration B

Same as for Configuration A except that an inner container is not required.

Configuration C and D

In addition to the drawings for Configuration A, General Atomics Drawing Nos. GADR 55-2-10, Issue D, and GADR 55-2-14, Issue N/C (optional). Configuration C uses a supplemental stainless steel shield ring and cover plate constructed in accordance with Drawing No. GADR 55-2-11, Issue B. Configuration D uses a supplemental carbon steel shield ring and cover plate constructed in accordance with Drawing No. GADR 55-2-11, Issue A.

Configuration E

In addition to the drawings for Configuration A, General Atomics Drawings Nos. GADR 55-2-1, Issue C; GADR 55-2-2, Issue A; and GADR 55-2-3, Issue B.

Configuration F and G

In addition to the drawings for Configuration A, General Atomic Drawings Nos. GADR 55-2-1, Issue C; GADR 55-2-2, Issue A; GADR 55-2-12, Issue C; and GADR 55-2-13, Issue A.

5. (b) Contents

(1) Type and form of material

(i) Irradiated fuel elements consisting of graphite body, hexagonal in horizontal cross section, approximately 31.2-inches high and 14.2-inches across the flats. Prior to irradiation, each fuel element contains thorium and uranium enriched to a maximum of 93.5 w/o in the U-235 isotope, or

(ii) Solid, irradiated, and contaminated hardware, which may include fissile material, provided the quantity of fissile material does not exceed a Type A quantity and does not exceed the mass limits of 10 CFR 71.53 until October 1, 2004, and 10 CFR 71.15, thereafter and neutron source components, or

(iii) Solid, nonfissile, irradiated and contaminated hardware which has been removed from the Fort St. Vrain High Temperature Gas Cooled Reactor and the surface contamination does not exceed 51 millicuries per package.

(2) Maximum quantity of material per package

Decay heat not to exceed 4.1 kw and:

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

Six fuel elements each containing a maximum of 1.4 kg of enriched uranium, having a thorium/uranium ratio greater than 8.1:1 and weighing approximately 300 pounds. The gross weight of the cask cavity contents, including the component spacers, inner container, and irradiated fuel elements shall not exceed 4,430 pounds. Contents must be shipped in Configuration E.

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

The gross weight of the cask cavity contents, including appropriate component spacers, liners, inner containers, shield rings and solid, non-fissile, irradiated and contaminated hardware shall not exceed 3,720 pounds. Contents must be shipped in Configurations A, B, C, or D.

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

The gross weight of all of the cask cavity contents, including burial canister and spacers, with or without supplemental shielding shall not exceed 4,430 pounds. Contents must be shipped in Configurations F or G.

5. (c) Criticality Safety Index

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

6. As needed, appropriate component spacers must be used in the cask cavity when shipping the contents described in paragraph 5(b) to limit movement of contents during shipment.

7. For transport of the contents of Item 5(b)(1)(ii) in Configuration D, the dose rate measured on the surface of the package must not exceed 200 mr/hr. For the purpose of this requirement, the surface of any personnel barrier may not be considered the surface of the package.

8. The Model No. FSV-1 Unit 3 cask may be wrapped with reinforced plastic when shipping the contents described in Item 5(b)(1)(ii) or (iii) provided the heat generation rate does not exceed 500 watts. The applicable requirements of 10 CFR 71.87 must be satisfied prior to wrapping the cask.

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

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

(a) Configurations A, B, C, and D of the Model FSV-1 Unit 3 shipping cask shall be prepared for shipment and operated in accordance with the Operating Procedures of Section 7.0, Volume I, of the application, as supplemented. The package shall be maintained in accordance with the Maintenance Program in Section 8.0, Volume I, of the application, as supplemented.

(b) Configurations E, F, and G of the Model FSV-1 Unit 3 shipping cask shall be prepared for shipment and operated in accordance with the Operating Procedures of Section 7.0, Volume II, of the application, as supplemented. The packages shall be maintained in accordance with the Maintenance Program in Section 8.0, Volume II, of the application, as supplemented.

(c) The main flange seals must be replaced within twelve (12) months prior to any use of the packaging and must be replaced if inspection shows any defect.

(d) The silicone O-ring on the inner container primary plug in Configuration E must be replaced within the twelve (12) months prior to any use of the packaging and must be replaced if inspection shows any defect.

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

12. Revision 2 of this certificate may be used until May 31, 2007.

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

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9277	3	71-9277	USA/9277/B( )F	5	OF	5

REFERENCE

Public Service Company of Colorado application dated March 28, 1996, as supplemented by Chem-Nuclear Systems, L.L.C., letter dated May 19, 1997, and General Atomics letter dated June 6, 1997, as supplemented April 11, 2001, June 7, 2001, and May 5, 2006

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

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

Date: 5/31/06

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE	PAGES
9279	4	71-9279	USA/9279/B(U)F-85	1 OF	4

2. PREAMBLE

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

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

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

   a. ISSUED TO *(Name and Address)*
   Department of Energy
   Washington, DC 20586

   b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION
   Washington TRU Solutions LLC application dated October 4, 2004, as supplemented.

4. CONDITIONS

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

5.

Packaging

(1) Model No.: HalfPACT Waste Shipping Container

(2) Description

A stainless steel and polyurethane foam insulated shipping container designed to provide double containment for shipment of contact-handled transuranic waste. The packaging consists of an unvented, 1/4-inch thick stainless steel inner containment vessel (ICV), positioned within an outer containment assembly (OCA) consisting of an unvented 1/4-inch thick stainless steel outer containment vessel (OCV), an approximate 8-inch thick layer of polyurethane foam, a 1/4-inch thick layer of ceramic fiber paper and a 1/4 to 3/8-inch thick outer stainless steel shell. The package is a right circular cylinder with outside dimensions of approximately 94 inches diameter and 92 inches height. The package weighs not more than 18,100 pounds when loaded with the maximum allowable contents of 7,600 pounds.

The OCA has a domed lid which is secured to the OCA body with a locking ring. The OCV containment seal is provided by a butyl rubber O-ring. The OCV is equipped with a seal test port and a vent port.

The ICV is a right circular cylinder with domed ends. The outside dimensions of the ICV are approximately 74 inches diameter and 69 inches height. The ICV lid is secured to the ICV body with a locking ring. The ICV containment seal is provided by a butyl rubber O-ring. The ICV is equipped with a seal test port and vent port. Aluminum spacers are placed in the top and bottom domed ends of the ICV during shipping. The cavity available for the contents is a cylinder of approximately 73 inches diameter and 45 inches height.

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5.(a)(3) Drawings

The package is constructed and assembled in accordance with Packaging Technology, Inc., Drawing 707-SAR Sheets 1-12, Rev. 6. The standard pipe overpack is constructed and assembled in accordance with Packaging Technology, Inc. Drawing No. 163-001, Sheets 1-3, Rev. 6. The S100 pipe overpack is constructed and assembled in accordance with Packaging Technology, Inc., Drawing No. 163-002, Sheets 1 and 2, Rev. 4. The S200 pipe overpack is constructed and assembled in accordance with Packaging Technology, Inc., Drawing No. 163-003, Sheets 1 and 2, Rev. 3. The S300 pipe overpack is constructed and assembled in accordance with Packaging Technology, Inc., Drawing No. 163-004, Sheet 1, Rev. 1. The compacted puck drum spacers needed for the purpose of maintaining subcriticality in 55-, 85-, and 100-gallon drums are constructed and assembled in accordance with Drawing No. 163-006, Rev. 0.

(b) Contents

(1) Type and form of material

Byproduct, source, and special nuclear material in the form of dewatered, solid or solidified materials and wastes. Materials must be packaged in one of the following payload containers: a 55-gallon drum, standard waste box (SWB), 85-gallon drum, standard pipe overpack, S100 pipe overpack, S200 pipe overpack, S300 pipe overpack, or 100-gallon drum. The payload containers are described in Section 2.9, "Payload Container/Assembly Configuration Specifications," of the CH-TRAMPAC, Rev. 2. Explosives, corrosives (pH less than 2 or greater than 12.5), nonradioactive pyrophorics, and compressed gases are prohibited. Within a payload container radioactive pyrophorics must not exceed 1 weight percent by weight and free liquids must not exceed 1 percent by volume. Flammable organics and methane are limited along with hydrogen to ensure the absence of flammable gas mixtures in TRU waste payloads as described in Chapter 5.0 of the CH-TRAMPAC, Rev. 2. For payload configurations with an unvented heat-sealed bag layer, the absence of flammable gas mixtures is ensured as described in Appendix 6.13 of the CH-TRU Payload Appendices, Rev. 1.

(2) Maximum quantity of material per package

The package contents are limited to 7,600 pounds, including the weight of the payload containers and any other components of the payload assembly. The maximum gross weight for a payload container not to exceed the following:

(i) 328 pounds per 6-inch standard pipe overpack,
(ii) 547 pounds per 12-inch standard pipe overpack,
(iii) 550 pounds per S100 pipe overpack,
(iv) 547 pounds per S200 pipe overpack,
(v) 547 pounds per S300 pipe overpack,
(vi) 1,000 pounds per 100-gallon drum,
(vii) 1,000 pounds per 55-gallon drum,
(viii) 1,000 pounds per 85-gallon drum, or
(ix) 4,000 pounds per SWB.

5.(b)(2) Maximum number of payload containers per package and authorized packaging configurations as follows:

(i) 7 55-gallon drums,
(ii) 7 standard pipe overpacks,
(iii) 7 S100 pipe overpacks,
(iv) 7 S200 pipe overpacks,
(v) 7 S300 pipe overpacks,
(vi) 4 85-gallon drums,
(vii) 3 100-gallon drums, or
(viii) 1 SWB.

Fissile material not to exceed the limits specified in CH-TRAMPAC, Rev. 2, Section 3.1, "Nuclear Criticality."

The S100, S200, and S300 pipe overpack payloads shall meet the curie limits specified in CH-TRAMPAC, Rev. 2, Section 3.3, "Activity Limits."

Maximum decay heat per package not to exceed 30 watts. Decay heat per payload container not to exceed the values in Table 5.2-1 of the CH-TRAMPAC, Rev. 2, "List of Approved Alphanumeric Shipping Categories, Maximum Allowable Hydrogen Gas Generation Rates, and Maximum Allowable Wattages," or calculated for approved shipping categories in accordance with the methodology specified in Section 5.2.3 of the CH-TRAMPAC, Rev. 2."

5. (c) Criticality Safety Index: 0.0

6. Physical form, chemical properties, chemical compatibility, configuration of waste containers and contents, isotopic inventory, fissile content, decay heat, weight and center of gravity; and radiation dose rate must be determined and limited in accordance with CH-TRAMPAC, Rev. 2.

7. Each payload container must be assigned to a shipping category in accordance with Section 5.1, "Payload Shipping Category" of CH-TRAMPAC, Rev. 2. Each payload container and payload assembly must not exceed the allowable wattage in accordance with Section 5.2.3, "Hydrogen Gas Generation Rate and Decay Heat Limits for Analytical Category," or must be tested for gas generation in accordance with Section 5.2.5, "Unified Flammable Gas Test Procedure," of CH-TRAMPAC, Rev. 2. For a payload made up of payload containers with different (nonequivalent) shipping categories, the flammability index of each payload container must not exceed 50,000 in accordance with CH-TRAMPAC, Rev. 2, Section 6.2.4, "Mixing of Shipping Categories," and Appendix 2.4 of the CH-TRU Payload Appendices, "Mixing of Shipping Categories and Determination of the Flammability Index."

8. Payload containers within a package shall be selected in accordance with Section 6.0, "Payload Assembly Requirements" of CH-TRAMPAC, Rev. 2.

9. Each payload container must be vented in accordance with Section 2.5, "Filter Vents" of CH-TRAMPAC, Rev. 2. Drums which were not equipped with filtered vents during storage must be aspirated in accordance with Section 5.3, "Venting and Aspiration" of CH-TRAMPAC, Rev. 2.

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE	PAGES
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10. For close-proximity and controlled shipments meeting the conditions specified in Appendices 3.5 and 3.6, respectively, of CH-TRU Payload Appendices, shipping periods of 20 days and 10 days may be applicable. The shipping period for any mode of transport is not to exceed 60 days. The content code LA 154 and SQ 154 are not authorized for loading and shipment in the HalfPACT packagings.

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

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

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

    (c) All free standing water must be removed from the inner containment vessel cavity and the outer containment vessel cavity before shipment.

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

13. Expiration date: October 31, 2010.

14. Revision No. 3 of this certificate may be used until October 31, 2006.

REFERENCES

Washington TRU Solutions, LLC, October 4, 2004 and March 4, June 8, and August 19, 2005.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

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

Date: 10/19/05

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9280	3	71-9280	USA/9280/AF-85	1	OF	3

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

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

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

BWX Technologies, Inc., application dated January 25, 2008.

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.

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5.(b) Contents (continued)

(2) Maximum quantity of material per package

550 pounds. The uranium may be of any enrichment, and the beryllium may be present in any concentration. The maximum fissile mass is 100 grams U-235 per package, and the maximum average fissile mass density in the package is 0.5 gram U-235 per liter. Fission and activation products may be present, provided that the total quantity is less than 1×10^{-3} A_2 per package.

(c) Criticality Safety Index to be shown on label for nuclear criticality control:

Maximum Fissile Mass Per Package (grams U-235 per package)	Minimum Criticality Safety Index
2.0	0.5
5.0	1.0
6.0	1.2
10.0	2.0
20.0	4.0
25.0	5.0
50.0	10.0
100.0	20.0

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

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

(b) Each packaging must be acceptance tested in accordance with the Acceptance Tests in Section 8 of the application.

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

8. Transport by air of fissile material is not authorized.

9. Revision No. 2 of this certificate may be used until April 30, 2009.

10. Expiration date: May 31, 2013.

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

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REFERENCES

BWX Technologies, Inc., application dated January 25, 2008.

Supplement dated March 20, 2008.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Meraj Rahimi, Acting Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date May 2, 2005

NRC FORM 618 (8-2000) 10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

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9281	5	71-9281	USA/9281/AF-85	1	OF	3

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

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

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

BWX Technologies, Inc., application dated April 10, 2008.

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.

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

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5.
(b) Contents

(1) Type and form of material

Uranium and uranium-beryllium waste of solid form. The waste may be contained within compacted 55-gallon drums.

(2) Maximum quantity of material per package:

950 pounds, including compacted secondary containers. The uranium may be of any enrichment, and the beryllium may be present in any concentration. The maximum fissile mass is 100 grams U-235 per package, and the maximum average fissile mass density in the package is 0.5 gram U-235 per liter. Fission and activation products may be present, provided that the total quantity is less than 1×10^{-3} A_2 per package.

(c) Criticality Safety Index

Maximum Fissile Mass Per Package (grams U-235 per package)	Criticality Safety Index
2.0	0.5
5.0	1.0
6.0	1.2
10.0	2.0
20.0	4.0
25.0	5.0
50.0	10.0
100.0	20.0

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

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

(b) Each packaging must be acceptance tested in accordance with the Acceptance Tests in Section 8 of the application.

7. Transport by air of fissile material is not authorized.

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

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE	PAGES
9281	5	71-9281	USA/9281/AF-85	3 OF	3

9. Revision No. 4 of this certificate may be used until August 31, 2009.

10. Expiration date: August 31, 2013.

REFERENCES

BWX Technologies, Inc., application dated April 10, 2008.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Eric J. Benner, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: August 22, 2008

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

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9282	1	71-9282	USA/9282/B(U)-96	1 OF	3

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

Source Production
and Equipment Company, Inc.
113 Teal Street
St. Rose, LA 70087-9691

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

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

4. CONDITIONS

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

5.

(a) Packaging

(1) Model No.: SPEC-300

(2) Description

The SPEC-300 is a radiographic device that consists of a source assembly, a depleted uranium shield, and a stainless steel enclosure. The radioactive source assembly is housed in a zircaloy or titanium "S" tube that is surrounded by the depleted uranium shield. The depleted uranium shield is secured in the stainless steel enclosure. The void space between the depleted uranium shield and the enclosure is filled with high density polyurethane foam. The package is approximately 26 inches long, 14 inches wide, and 15 inches high. The maximum gross weight of the package is 780 pounds.

(3) Drawings

The packaging is constructed and assembled in accordance with Source Production and Equipment Co., Inc. General Arrangement drawings: 19B000 sheets 1-8, Rev. 4 and B190700 sheet 1, Rev. 3.

(b) Contents

(1) Type and form of material

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

5.(b) Contents (continued)

(2) Maximum quantity of material per package

300 Curies (output)

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

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

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

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

(a) The package shall be prepared for shipment in accordance with the Operating Procedures in Chapter 7.0 of the application, as supplemented; and

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

9. Packagings may be marked with Package Identification Number USA/9282/B(U)-85 until April 30, 2006, and must be marked with Package Identification Number USA/9282/B(U)-96 after April 30, 2006.

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

11. Expiration date: April 30, 2010.

REFERENCES

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

Supplements dated: October 6, November 4, November 22, and December 15, 1999; February 29 and March 27, 2000; and March 14, 2005.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

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

Date: 28 April 2005

UNITED STATES NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

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

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

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

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

a. ISSUED TO *(Name and Address)*

QSA Global, Inc.
40 North Avenue
Burlington, MA 01803

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

AEA Technology/QSA Inc. application dated May 21, 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.: OPL-660 and OP-660

(2) Description

The Model Nos. OPL-660 and OP-660 consist of a radiography camera within a protective container. The protective container is a 20 mm Cartridge Shipping and Storage box fabricated according to military specification MIL-S-23389B. The protective container is of welded steel construction and is approximately 18½ inches long, 14½ inches high, and 8¼ inches wide. The protective container is fitted with foam and wood inserts and a lid that is secured by latches. The Model 660 series projector fits snugly in the center of the foam inserts within the protective container. The Model No. OPL-660 container has thin lead sheets to provide extra shielding at the ends and bottom. The maximum weight of the package is 88 pounds.

The Model 660 series projector is a radiography device. The projector's overall dimensions are approximately 12⅞ inches long, 5¼ inches wide, and 9⅝ inches high. The projector weighs a maximum of 56 pounds. The principal components of the 660 series projectors include an outer steel shell, polyurethane foam, a depleted uranium shield, an "S" tube, and end plugs. The sealed source contents are securely positioned in the "S" tube by a source cable locking device and shipping plug.

(3) Drawings

The packaging is constructed in accordance with the following AEA Technology QSA, Inc., Drawings:

R66050, Rev. C, Sheets 1 & 2, and R66060, Rev. A, Sheets 1-3.

5. (b) Contents

(1) Type and form of material

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

(2) Maximum quantity of material per package

(i) 140 Curies (output) for the Model No. 660B or 660BE projectors.

(ii) 120 Curies (output) for the Model No. 660, 660E, 660A or 660AE projectors.

Output curies are determined by measuring the source output at 1 meter and expressing its activity in curies derived from the following: 0.48 R/h-Ci Iridium-192 at 1 meter. (Ref: American National Standard N432-1980, "Radiological Safety for the Design and Construction of Apparatus for Gamma Radiography.")

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.

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

10. Revision No. 2 of this certificate may be used until June 30, 2008.

12. Expiration date: June 30, 2013.

REFERENCES

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

Supplements dated: June 15, 1998; March 6, 2003; May 30, 2006; and November 6, 2007.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Robert A. Nelson, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: December 7, 2007.

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

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2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*
Columbiana Hi Tech, LLC
1802 Fairfax Road
Greensboro, NC 27407

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION
Eco-Pak Specialty Packaging application dated June 19, 1998, as supplemented.

4. CONDITIONS

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

5.

(a) Packaging

(1) Model No.: ESP-30X Protective Shipping Package for 30-inch UF_6 Cylinders

(2) Description

An overpack for the transport of 30-inch enriched uranium hexafluoride (UF_6) cylinders. The shape of the overpack is a right circular cylinder constructed of two 11 gauge carbon steel shells. The area between the shells is filled with fire retardant, phenolic foam per ESP specification ESP-PF-1. The volume between the 1/2" inch thick end plates of the two shells is also filled with phenolic foam. A stepped horizontal joint permits the top half of the overpack to be removed from the base. The horizontal joint of each half of the overpack is covered with steel and a 5/8" thick silicone gasket seals the joint. The overpack halves are secured with ten 3/4" diameter steel bolts and nuts.

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

Outer shell inside diameter	43"
Outer shell length	96"
Inner shell inside diameter	30 7/8"
Inner shell length	82 5/8"
Overpack weight	2,955 pounds
30B Cylinder weight	1,390 pounds
UF_6 maximum load	5,020 pounds
Maximum package gross weight (including contents)	9,365 pounds

(3) Drawings

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

30X-1 SAR, Rev. 2, Sheets 1-4

5.(b) Contents

(1) Type and form of material

The UF_6 must be packaged in Model 30B UF_6 cylinders which have been fabricated, inspected, tested and maintained in accordance with the requirements of ANSI N14.1. The UF_6, which may contain either virgin or recycled uranium, must not contain more than the following maximum quantities of radionuclides and impurities:

U^{232}	5.0E-09 g/gU
U^{234}	2.0E-03 g/gU
U^{235}	5.0E-02 g/gU
U^{236}	2.5E-02 g/gU
U^{238}	balance of total uranium content
Pu and Np	Alpha activity not exceed 3.3 Bq/gU
Tc^{99}	5.0E-06 g/gU
Th^{228}	1.17E-09 g/gU

Fission Products 4.4 X 10^5 Mev Bq/d kgU (total contribution from gamma emitting fission products); this results in the following individual maximum activities:

Ru^{106}/Rh^{106}	2095 Bq/gU
Ru^{103}/Rh^{103}	885 Bq/gU
$Ce^{144}/Pr^{144}/Pr^{144}$	8349 Bq/gU
Sb^{125}	1030 Bq/gU
Cs^{134}	283 Bq/gU
Cs^{137}/Ba^{137}	778 Bq/gU
Zr^{95}	598 Bq/gU
Nb^{95}	574 Bq/gU

The total concentration of elements that form non-volatile fluorides (including Al, Ba, Bi, Cd, Co, Cr, Cu, Fe, Pb, Li, Mg, Mn, Ni, K, Ag, Na, Sr, Th, Sn, Zn, and Zr) must not exceed 3.0E-03 g/gU.

The contents of other elements must not exceed the following concentrations in g/gU.

Sb<1	As<3	B<1	Bi<5	Cl<100
Cr<10	Nb<1	P<50	Ru<1	Si<100
Ta<1	Ti<1	Mo<1.4	W<1.4	V<1.4

Additionally, for reprocessed UF_6, the maximum total activity present in the package is limited to 957 mixture A_2 values.

(2) Maximum quantity of material per package

The package contents are limited to a maximum of 5,020 pounds of UF_6 enriched to not more than 5 wt%U^{235}. The maximum H/U atomic ratio for the UF_6 is 0.088.

5. (c) Criticality Safety Index 5.0

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

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

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

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

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

9. The leak tightness of the 30B UF_6 cylinder shall be verified using a test having a sensitivity of at least 1 x 10^{-3} std-cc/sec per ANSI Standard N14.5-1997 prior to loading into the ESP-30X overpack.

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

11. Expiration date: May 31, 2010.

REFERENCES

ESP application dated June 19, 1998.

Supplements dated: August 27, 1999; March 22, May 12, and May 18, 2000; April 11, 2002; January 28, and April 12, 2005.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

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

Date 20 April 2005

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSIO

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGE
9285	3	71-9285	USA/9285/AF-85	1	OF	2

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*
Global Nuclear Fuel - Americas, L.L.C.
Mail Code K-84
3901 Castle Hayne Road
Wilmington, NC 28401

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION
General Electric Company application dated August 4, 1998, as supplemented.

4. CONDITIONS

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

5.

(a) Packaging

(1) Model No.: SRP-1

(2) Description

A steel drum for the transport of solid uranium contaminated residues. The packaging is a 55-gallon, open-head steel drum with a minimum 18-gauge shell and bottom head, and a minimum 16-gauge closure lid. The lid is closed by a 12-gauge bolted locking ring with drop forged lugs, one of which is threaded, having a 5/8 inch bolt and nut. The closure includes a gasket. The gross weight of the package, including the maximum weight of contents, is 825 pounds.

(3) Drawings

The packaging is constructed and assembled in accordance with General Electric Company Drawing. No. 0025E98, Rev. 1.

(b) Contents

(1) Type and form of material

Uranium-contaminated solid residues.

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE	PAGE
9285	3	71-9285	USA/9285/AF-85	2 OF	2

5. (b) Contents (Continued)

(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) Criticality Safety Index (CSI): 0.6

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

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

(b) Each packaging must be acceptance tested in accordance with the Acceptance Tests in Section 8 of the application.

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

8. Air transport of fissile material is not authorized.

9. Revision No. 2 of this certificate may be used until October 31, 2009.

10. Expiration date: October 31, 2013.

REFERENCES

General Electric Company application dated August 4, 1998.

Supplements dated: October 2, 1998; October 14, 1999; August 6, 2003; and October 8, 2008.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Eric J. Benner, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: Oct 24, 2008

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
9287	2	71-9287	USA/9287/B(U)-85	1 OF 3

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

AREVA Federal Services LLC
1102 Broadway Plaza, Suite 300
Tacoma, WA 98402-3526

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

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

4. CONDITIONS

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

5.

(a) Packaging

(1) Model No.: SteriGenics Eagle

(2) Description

A stainless steel, lead shielded shipping cask for special form cobalt-60 sealed sources. The package consists of a cylindrical cask body with closure lid, and removable toroidal impact limiters, and a basket that carries and positions the cobolt-60 sealed source capsules. The packaging is constructed primarily of ASTM Type 304 stainless steel. The package is designed to transport up to 330,000 curies of cobalt-60.

The outside diameter of the cask body is approximately 37-11/16 inches. The diameter of the inner cavity is approximately 10-3/4 inches. The stainless steel inner shell has a minimum thickness of 1 inch and the stainless steel outer shell is 1 inch thick. The region between the two shells is filled with lead shielding. The closure lid and cask bottom end each consist of two stainless steel plates with lead between the two plates. The lead shielding thickness is approximately 10-3/8 inches on the side, 14-3/8 inches in the closure lid, and 11-7/8 inches on the cask bottom. The closure lid is secured by 12, 3/4-inch bolts. The closure lid is equipped with a Viton O-ring seal. The lid has a drain port and a vent port, and the cask body has a drain port. Each port is closed by a plug.

A double stainless steel thermal radiation shield is provided on the outside of the cask body in the region between the two impact limiters. The inner thermal shield is about 3/4-inches thick and is radially separated from the cask outer shell by 12 gauge spacers at each end. The outer shield is a sheet of 10 gauge material separated from the inner shield by a spiral wrap of 12 gauge wire.

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

The top and bottom impact limiters are toroidal stainless steel shells. They are attached to either end of the cask body using 12, 1-inch diameter ball-lock pins orientated radially around the cask body. One pin on each limiter is installed with a lockwire to provide a tamper-indicating device.

The cask lifting attachments thread into the upper cask body. The cask lid is also equipped with removable lid-lifting attachments. The cask rests on a steel pallet and is held down to the pallet by means of a steel frame placed on the top impact limiter. This steel frame is used to tie the cask to the conveyance. The maximum weight of the package, including contents is 20,000 lbs.

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

Cask Body Outer Diameter	37-11/16 inches
Cask Body Height	49-7/8 inches
Cask Cavity Inner Diameter	10-3/4 inches
Cask Cavity Inner Height	19 inches
Lead Shield Sidewall Thickness	10-3/8 inches
Overall Package Dimension	
Diameter at Impact Limiters	60 inches
Diameter at Body	37-11/16 inches
Height with Impact Limiters	76 inches
Maximum Contents Weight	50 pounds
Maximum Package Weight (Including Contents)	20,000 pounds

(3) Drawings

The packaging is constructed and assembled in accordance with Packaging Technology, Incorporated, Drawing No. 98003-SAR, Rev.1, Sheets 1 through 8.

(b) Contents

(1) Type and form of material

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

(2) Maximum quantity of material per package:

330,000 curies. Not to exceed 18,400 curies per special form source.

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

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

   (b) Each packaging must meet the Acceptance Tests and Maintenance Program of Chapter 8.0 of the application, as supplemented.

7. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR §71.17, provided the fabrication of the package was satisfactorily completed by December 31, 2006.

8. Revision No. 1 of this certificate may be used until December 31, 2008.

9. Expiration date: December 31, 2009.

REFERENCES

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

Supplements dated: August 20, 1999, November 29, 2004, and November 26, 2007.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Robert A. Nelson, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: January 1, 2008

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
9288	8	71-9288	USA/9288/B(U)F-96	1	OF	6

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

Columbiana Hi Tech, LLC
1802 Fairfax Road
Greensboro, NC 27407

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Columbiana Hi Tech, LLC, consolidated application dated February 27, 2006, as supplemented

4. CONDITIONS

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

5.

(a) Packaging

(1) Model No.: CHT-OP-TU

(2) Description

A shipping container for uranium oxide pellets, powder, and uranium-bearing materials. The package is roughly cubical and is approximately 45-inches x 45-inches x 62-inches high. The package has four internal sleeves in which Oxide Vessels are inserted.

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

The sleeves are constructed of 11-gauge mild or stainless steel with an inner diameter of 10-1/4 inches. The sleeves are closed with twelve 1/2-inch-diameter bolts using an outer lid assembly on a 1/16-inch-thick neoprene or silicone gasket. The outer lid assembly is filled with fire-retardant, closed cell phenolic or polyurethane foam.

The Oxide Vessel is constructed of series 300 stainless steel, with an inner diameter of either 6, 7.5, or 8 inches. The Oxide Vessel is closed by eight 1/2-inch-diameter bolts on a 5/8-inch-thick stainless steel lid with a double O-ring seal. The O-ring seal material is either silicon rubber, fluorosilicon or fluorocarbon (viton). A pellet shipping assembly is used within the Oxide Vessel for certain shipments.

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

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

Sleeve inside diameter	10 1/4-inches
Oxide Vessel inside diameter	6, 7.5, or 8 inches
Oxide Vessel inside height	40 3/4-inches
Overall package dimensions	
width	45 inches
length	45 inches
height	62 inches
Maximum contents weight per Oxide Vessel	402 pounds
Maximum empty transport weight including four empty Oxide Vessels	2576 pounds
Maximum loaded package weight (with four filled Oxide Vessels)	3757 pounds

(3) Drawings

The packaging is constructed and assembled in accordance with Columbiana Hi Tech Drawing Nos.:

OP-TU-SAR, Rev. 12, Sheets 1 of 2 and 2 of 2;
OP-TU-A2, Rev. 12, Sheet 1 of 1;
OP-TU-A3, Rev. 12, Sheet 1 of 1;
OP-TU-A4, Rev. 12, Sheet 1 of 1; and,
OPTU-V-AB1, Rev. 8, Sheets 1 of 2 and 2 of 2.

The Oxide Vessel Pellet Shipping Assembly is constructed and assembled in accordance with ARIVA NP, Inc., Drawing No. 9046816, Rev. 1.

5.(b) Contents

(1) Type and form of material

Uranium-bearing compounds in solid form, heterogenous or homogenous (i.e., pellets and powder). The contents may include up to 1000 grams of polyethylene or other plastics as packaging, waste or impurities per Oxide Vessel (4000 grams per package), provided that: (1), the total water equivalent of the plastic is less than 1307 grams per Oxide Vessel (5228 grams per package); and, (2) the decay heat is less than 0.068 W/m^3. Materials with a decay heat greater than 0.068 W/m^3 may not be packaged using hydrogen bearing plastics, and may only use non-hydrogen bearing plastics such as Teflon™ (polytetrafluoroethylene or PTFE) or metallic containers. In addition, the contents are limited to:

A. Unirradiated uranium oxide powder enriched to no more than 5.0 weight percent in the U-235 isotope.

B. Unirradiated uranium oxide pellets or a mixture of pellets and powder enriched to no more than 5.0 weight percent in the U-235 isotope.

C. Reprocessed uranium oxide powder enriched to no more than 5.0 weight percent in the U-235 isotope, with limits specified in Table 1.

Table 1: Allowable Content for Shipment of Reprocessed Uranium Oxide

	Maximum Content		
Isotope	Type A	Type B Level I	Type B Level II
U-232 (g/gU)	Mixtures of isotopes shall be evaluated and designated as a Type A quantity per 10 CFR Part 71 Appendix A. The maximum enrichment per package is 5 weight per cent ^{235}U.	2.00E-09	5.00E-09
U-234 (g/gU)		2.00E-03	2.00E-03
U-235 (g/gU)		5.00E-02	5.00E-02
U-236 (g/gU)		2.50E-02	2.50E-01
Np-237 (g/gU)		1.66E-06	5.00E-03
Pu-238 (g/gU)		6.20E-11	4.00E-08
Pu-239 (g/gU)		3.04E-09	3.04E-09
Pu-240 (g/gU)		3.04E-09	6.00E-09
Gamma Emitters (MeV-Bq/kgU)		6.38E+05	1.91E+06

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

D. Reprocessed uranium oxide pellets or a mixture of pellets and powder enriched to no more than 5.0 weight percent in the U-235 isotope, with the limits specified in Table 1.

E. Homogeneous (powder or crystalline form) uranium-bearing materials enriched to 5.0 weight percent in the U-235 isotope in the form of solids, or solidified or dewatered materials.

Uranium compounds must have a ratio of non-fissile atoms to uranium atoms greater than two (2) and the density of these compounds is less than 10.96 g/cm^3 (density of UO_2). Material such as U-metal, U-metal alloys, or uranium hydrides (e.g., UH_X) may not be shipped. Uranium-bearing materials may include oxides, carbides, silicates or other compounds of uranium. Uranium-bearing materials may be moderated by graphite to any degree. Compounds may be mixed with other non-fissile materials with the exception of beryllium or hydrogenous material enriched in deuterium. Materials with a hydrogen density greater than water must be excluded, except for the allowance provided by Condition No. 5.(b)(1).

F. Heterogeneous (pellets or previously pelletized materials) uranium-bearing materials enriched to 5.0 weight percent in the U-235 isotope in the form of solids, or solidified or dewatered materials.

Uranium compounds must have a ratio of non-fissile atoms to uranium atoms greater than two (2) and the density of these compounds is less than 10.96 g/cm^3 (density of UO_2). Material such as U-metal, U-metal alloys, or uranium hydrides (e.g., UH_X) may not be shipped. Uranium-bearing materials may include oxides, carbides, silicates or other compounds of uranium. Uranium-bearing materials may be moderated by graphite to any degree. Compounds may be mixed with other non-fissile materials with the exception of beryllium or hydrogenous material enriched in deuterium. Materials with a hydrogen density greater than water must be excluded, except for the allowance provided by Condition No. 5.(b)(1).

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

The maximum allowable contents heat generation rate is 1.0 BTU/hr/ft^3 (10.3 W/m^3). The maximum weight of contents, including the uranium compounds and all packaging materials within the Oxide Vessel, is 402 pounds per 8-inch, 7.5-inch, or 6-inch diameter Oxide Vessel, and a maximum of 1608 pounds per package.

For contents described in Condition Nos. 5(b)(1)(B), 5.(b)(1)(D), and 5.(b)(1)(F), the Oxide Vessel Pellet Shipping Assembly, as described in Condition No. 5(a)(3), must be used within the 8-inch diameter Oxide Vessel. The Oxide Vessel Pellet Shipping Assembly is not required when using the 7.5-inch, or 6-inch diameter Oxide Vessel.

(c) Criticality Safety Index 2.0

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

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

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

7. Transport by air of fissile material is not authorized.

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

9. Revision No. 7 of this certificate may be used until August 31, 2008.

10. Expiration date: March 31, 2010.

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9288	8	71-9288	USA/9288/B(U)F-96	6	OF	6

REFERENCES

Columbiana Hi Tech, LLC, consolidated application dated February 27, 2006.

Supplements dated: April 10, 2006; and July 17 and August 29, 2007.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Robert A. Nelson, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: September 26, 2007

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9289	3	71-9289	USA/9289/B(U)F-85	1	OF	5

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

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

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Framatome Cogema Fuels application dated May 1, 2002, as supplemented.

4. CONDITIONS

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

5.

(a) Packaging

(1) Model No.: WE-1

(2) Description

A fresh fuel assembly shipping container. The package has two shipping configurations: one for shipping a single BW 17x17 fuel assembly composed of uranium dioxide pellets within zircalloy cladding; and the other for shipping up to 48 Pathfinder fuel assemblies within a steel canister which functions as a secondary containment vessel. The package consists of a cylindrical outer container and a rectangular inner container bolted to a strongback. The outer container is constructed of 11 gauge carbon steel and opens into two semi-cylindrical halves. The inner container is comprised of 1-inch thick carbon steel plates that are bolted together. The inner container is secured to the strongback by bolts and clamp arms. Wood blocks surround the region between the inner container and the strongback. The strongback is supported by 14 shock mounts attached to the outer container.

<u>For BW 17x17 Fuel Shipment Configuration:</u>

The BW 17x17 fuel assembly shipment configuration consists of the fuel assembly placed into the inner container. The fuel assembly is surrounded by thermal insulation and secured inside the inner container with nine integral clamp frames.

5. (2) Description (Continued)

For Pathfinder Fuel Shipment Configuration:

Pathfinder Fuel shipment configuration consists of the Pathfinder fuel in the Pathfinder Canister, which is placed into the inner container. The Pathfinder Canister is a sealed cylindrical canister which houses up to 48 Pathfinder fuel assemblies. Wood dunnage or empty sheaths may be used to fill empty spaces in the canister. The canister is made of austenitic stainless steel and has a welded body and a bolted closure lid. The Pathfinder Canister is surrounded by thermal insulation, and secured inside the inner container with five integral clamp frames. The clamp frames, which consist of bolted clamp arms, are bolted to the inner rectangular container. Wood blocks surround both ends of the Pathfinder Canister. A stainless steel spacer tube is used to fill the space between the Pathfinder Canister and the inner container.

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

Inner container length	165 inches
Inner container width (sq)	16 ½ inches
Outer container length	216 inches
Outer container diameter	44 inches
Maximum content weight	1610 pounds
Maximum package weight (including contents)	9090 pounds

(3) Drawings

The packaging is constructed in accordance with the following Framatome Cogema Fuels Drawing Nos.:

1273964, Rev. 0
1273965, Rev. 1
1273966, Rev. 0
1273967, Rev. 0
1273968, Rev. 0

The Pathfinder Canister Configuration is constructed in accordance with the following Framatome ANP Drawing Nos.:

5016270, Rev. 1
5021426, Sheets 1 and 2, Rev. 0.

(b) Contents

(1) Type and form of material

(i) For BW 17x17 Fuel Shipment Configuration:

A fuel assembly composed of uranium dioxide pellets within zircalloy cladding. The fuel assembly has the following specifications:

Assembly type	BW 17x17
No. fuel rods	264
No. non-fuel tubes	25
Nominal fuel rod pitch, in.	0.496
Maximum fuel pellet OD, in.	0.3232
Nominal clad OD, in.	0.374
Nominal clad thickness, in.	0.022
Nominal guide and instrument tube OD, in.	0.48
Nominal guide and instrument tube ID, in.	0.452
Nominal active fuel length, in.	144
Maximum uranium enrichment, weight percent U-235	4.6
Maximum U-235 mass, kg	22.14

(ii) For Pathfinder Fuel Shipment Configuration:

An unirradiated fuel assembly composed of six fuel pins clustered around a center absorber pin in a hexagonal pattern. The fuel pins consist of uranium dioxide pellets inside Incoloy 800 cladding. The absorber pin consists of Incoloy 800 cladding with or without poison material. Fuel pins and absorber pins are separated by spacer wires and enclosed in a cylindrical sheath made of stainless steel, incoloy or incoloy alloy. The fuel assembly has the following specifications:

Assembly type	Pathfinder
No. fuel pins per assembly	6
No. non-fuel pins per assembly	1
Maximum uranium enrichment, weight percent U-235	7.51
Maximum uranium mass per assembly, kg U	2.2281

5. (b) Contents (continued)

Maximum UO_2 density, g/cm^3	10.61
Fuel pellet outer diameter (OD), in.	0.207 ± 0.0005
Nominal active fuel length, in.	72.0
Minimum clad OD, in.	0.246
Maximum clad inner diameter (ID), in.	0.212
Nominal center-to-center pin pitch, in.	0.289
Nominal sheath ID, in.	0.945
Nominal sheath OD, in.	1.00

(2) Maximum quantity of material per package

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

One BW 17x17 fuel assembly contents, not to exceed 1610 pounds. The radioactive material may not exceed any of the following limits:

U-232	0.01 microgram per gram of uranium
U-234	0.001 gram per gram of uranium
U-236	0.013 gram per gram of uranium
Tc-99	5 micrograms per gram of uranium
Fission Products	4.4 x 10^5 MeV-Becquerel per kilogram of uranium
Np and Pu	35 Becquerels per gram of uranium

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

Up to 48 unirradiated Pathfinder fuel assembles inside a Pathfinder Canister. The weight of the fully loaded canister not to exceed 800 pounds.

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

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

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

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

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

NRC FORM 618 (8-2000) 10 CFR 71	CERTIFICATE OF COMPLIANCE FOR RADIOACTIVE MATERIAL PACKAGES			U.S. NUCLEAR REGULATORY COMMISSION
a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE / PAGES
9289	3	71-9289	USA/9289/B(U)F-85	5 OF 5

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

8. Expiration date: February 28, 2009.

REFERENCES

Framatome ANP, Inc. application dated: May 1, 2002.

Supplement dated: November 12, 2002, and January 8, 2004.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

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

Date: January 26, 2004

UNITED STATES NUCLEAR REGULATORY COMMISSION

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9290	4	71-9290	USA/9290/B(U)-85	1	OF	3

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

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

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

MDS Nordion application dated February 20, 2003, as supplemented.

4. CONDITIONS

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

5.

(a) Packaging

(1) Model No. F-430/GC-40 Transport Package

(2) Description

The Model No. F-430/GC-40 Transport package is designed to transport MDS Nordion's Gammacell-40 (GC-40) Irradiator containing cesium-137 sealed sources in special form. The F-430 overpack provides impact and thermal protection for the radioactive contents. Containment is provided by the special form sealed source and shielding is provided by the GC-40 irradiator body.

The F-430 is stainless steel cylindrical package with a 50" diameter and a height of 50" that is placed on a removable mild steel skid. The maximum weight of the package is 7000 pounds. The maximum weight of the GC-40 contents is 4000 pounds.

The overpack consists of nested cylindrical shells. The shells are made from stainless steel and the volume between the shells is filled with rigid foam. This foam provides insulation during an accidental fire. Vent holes, plugged with material designed to melt in a fire, are provided between the shells to prevent pressure buildup and allow a pathway for escape of gases from foam during an accidental fire.

The package contents consists of a Cesium-137 sealed source contained within an MDS Nordion GC-40 irradiator (upper or lower heads). The GC-40 is a research irradiator with lead shielding and a lead filled source drawer.

5.(a)(2) (continued)

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

Package outside diameter	50 inches
Package height	50 inches
Cavity diameter	36 inches
Cavity height	35.25 inches
Removable skid	50 inches x 50 inches x 8 inches (height)
Overpack weight	2640 pounds
Contents weight	4000 pounds
Maximum package weight	7000 pounds

(3) Drawings

The packaging is constructed in accordance with the MDS Nordion drawings F643001-001, Rev. K, Sheet 1 of 2, and F643001-001, Rev. D, Sheet 2 of 2.

(b) Contents

(1) Type and form of material

Cesium-137 as a sealed source which meet the requirements of special form radioactive material.

(2) Maximum quantity of material per package

2,000 Curies.

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

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

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

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9290	4	71-9290	USA/9290/B(U)-85	3	OF	3

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

8. Revision No. 3 of this certificate may be used until November 30, 2007.

9. Expiration date: February 28, 2012.

REFERENCES

MDS Nordion application dated February 20, 2003.

Supplements dated: July 21, August 25, and December 18, 2003; January 16, July 16, July 21, and July 23, 2004; April 21, and October 30, 2006.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Robert A. Nelson, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: November 22, 2006

NRC FORM 618 (8-2000) 10 CFR 71	**CERTIFICATE OF COMPLIANCE FOR RADIOACTIVE MATERIAL PACKAGES**			U.S. NUCLEAR REGULATORY COMMISSION
a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE / PAGES
9291	6	71-9291	USA/9291/B(U)F-96	1 OF 3

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*
Columbiana Hi Tech, LLC
1802 Fairfax Road
Greensboro, NC 27407

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION
Columbiana Hi Tech, LLC, consolidated application dated February 17, 2006, as supplemented.

4. CONDITIONS

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

5.

(a) Packaging

(1) Model No.: Liqui-Rad (LR) Transport Unit Package

(2) Description

The LR Package is designed to transport Type B quantities of fissile uranyl nitrate solutions. The package uses thermal and impact limiting systems to protect the containment vessel and prevent the contents from being released. The primary structural components of the LR packaging consist of a stainless steel containment vessel, a carbon steel outer vessel and a carbon steel framing system. The containment vessel is built in accordance with ASME Pressure Vessel Code (Section VIII, Division 1) but does not require an ASME stamp. Double O-ring seals on the containment vessel's primary and secondary lids provide a leak tight seal which is leak testable. A closed-cell phenolic foam or polyurethane foam surrounds the top and bottom head area of the containment vessel and ceramic fiber blanket and board insulation are used in the sidewalls and outer lid for thermal insulation and impact absorption. The maximum volume of the contents is limited to 230 gallons which maintains a minimum ullage of 33 gallons.

The LR is a cylindrical package set in a rectangular angle frame. The dimensions of the package are approximately 56"(l) x 56"(w) x 73"(h). The maximum weight of the package is 5692 pounds. The outer vessel is constructed of 10 gauge carbon steel. The containment vessel is constructed of 1/4 inch stainless steel with 1/4 inch thick flanged and dished heads. The containment vessel is rated at 50 psig pressure. Closed-cell phenolic or polyurethane foam and ceramic fiber insulation are sandwiched between the containment vessel and the package's outer shell.

The package is designed to be leak-tight (maximum allowable leakrate of 1×10^{-7} ref-cm^3/sec). The containment vessel is closed using a double O-ring and is secured by sixteen 5/8 inch stainless steel studs. The outer lid is closed with four 5/8 inch

5.(a)(2) Continued

stainless steel bolts and nuts. The package is also equipped with plastic plugs to vent any gases that may be generated by the insulation during a fire event. All valves and fittings are provided within sealed enclosures to contain any leakage during valve failure.

(3) Drawings

The packaging is constructed and assembled in accordance with Columbiana Hi Tech Drawing Nos. LR-SAR, Sheets 1 through 4, Rev. 7.

5.(b) Contents

(1) Type and form of material

Low enriched Uranyl Nitrate solutions with the specifications shown in Table 1 below. The uranium concentration must be less than or equal to 125 gU/liter with an enrichment less than or equal to 5.0 wt% U-235. Non-fissile chemical impurities may be present up to the chemical impurity specification in Table 1. Additionally, fissile isotopes are also limited to the quantities in Table 1.

(2) Maximum quantity of material per package

230 gallons of Uranyl Nitrate solution with limits as shown in table 1.

Table 1

ITEM	SPECIFICATION
Solution Density	≤1.17 g/cc
Chemical Impurities	≤1500 μg/gU
Nitric Acid Normality	0.1 - 0.7
Uranium Concentration	≤125 gU/l
U-232	≤2.0E-03 μg/gU
U-234	≤2.0E+03 μg/gU
U-235	≤0.05 g/gU (12 pounds maximum quantity of U-235 per LR)
U-236	≤2.5E+04 μg/gU
U-238	remainder of uranium
Pu/Np Alpha Activity	≤93 Bq/gU
Gamma Emitters	0.515E-01 Ci

5. (c) Criticality Safety Index 0.0

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

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

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

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

8. Packagings may be marked with Package Identification Number USA/9291/B(U)F-85 until March 31, 2007, and must be marked with Package Identification Number USA/9291/B(U)F-96 after March 31, 2007.

9. Transport by air of fissile material is not authorized.

10. Revision No. 5 of this certificate may be used until August 31, 2007.

11. Expiration date: October 31, 2011.

REFERENCES

Columbiana Hi Tech, LLC, consolidated application dated February 17, 2006.

Supplement dated: July 25, 2006.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

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

Date: August 3, 2006

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9292	5	71-9292	USA/9292/AF-85	1	OF	5

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

Westinghouse Electric Company, LLC
P.O. Box 355
Pittsburgh, PA 15230-0355

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Westinghouse Electric Company, LLC application dated September 16, 2004, as supplemented.

4. CONDITIONS

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

5.

(a) Packaging

(1) Model No.: PATRIOT

(2) Description

A shipping container for unirradiated fuel assemblies. The package consists of a right rectangular metal inner container and a wooden outer container, with cushioning material between the inner and outer containers.

There are two versions of the metal inner container. Both versions measure approximately 11-1/4 inches high by 18-1/8 inches wide by 182 inches long. There are two channel sections within the inner container, and each channel section holds one BWR fuel assembly. The inner container is equipped with a lid and an end cap that are closed by 18 bolts and fastening lugs. The overall dimensions of the wooden outer container are approximately 30-1/4 inches wide by 31-1/4 inches high by 207-3/4 inches long. The cushioning material between the inner and outer containers is phenolic impregnated honeycomb and ethafoam. The inner container may be positioned on a series of vibration dampers mounted on the inside bottom of the wooden outer container.

The maximum weight of the package, including contents, is 2,988 pounds with the version #1 inner container and 2,964 pounds with the version #2 (optional) inner container.

5.(a)(3) Drawings

The packaging is constructed and assembled in accordance with Westinghouse Electric Company, LLC Drawing Nos.:

10014E27, Rev. 1,
10014E28, Sheets 1 and 2, Rev. 2,
10015E58, Sheets 1 and 2, Rev. 2

5.(b) Contents

(1) Type and form of material

The package is designed to hold two unirradiated BWR fuel assemblies, comprised of UO_2 fuel rods in a 10 x 10 square array. The fuel cross-sectional area is 25 square inches.

(i) Description of Assembly Type #1

Each assembly is made up of 96 full-length fuel rods having a maximum active fuel length of 150 inches. The fuel pellet diameter is 0.819 ± 0.002 cm, encapsulated in 0.063 cm zirconium alloy cladding. There is a 0.0085 cm gap between the pellets and the cladding. The maximum U-235 enrichment of any fuel rod is 5.0 weight percent. Each assembly contains water holes in the four center rod positions of the assembly. Three different fuel package loadings have the following specifications:

(A) Maximum average U-235 enrichment is 4.0 weight percent within any axial zone of the assembly; Maximum U-235 content is 3.25 weight percent of any gadolinia-urania rod or axial zone of any gadolinia-urania fuel rod; Maximum number of fuel rods per assembly containing 5.0 weight percent U-235 enriched pellets is 36; Maximum U-235 enrichment is 4.0 weight percent for all edge rods, and 3.5 weight percent for all corner rods; Each assembly must include at least eight fuel rods with a minimum gadolinia content of 2.5 weight percent in all axial regions with enriched pellets. The eight gadolinia rods are arranged with two rods in each quadrant of the fuel assembly. The two gadolinia rods within each quadrant must be symmetric about the geometric diagonal of the fuel assembly, and must not be in an edge or corner rod location. Other fuel rods containing gadolinia may be present.

(B) Maximum average U-235 enrichment is 4.725 weight percent within any axial zone of the assembly; Maximum U-235 content is 4.2 weight percent of any gadolinia-urania rod or axial zone of any gadolinia-urania fuel rod; Maximum number of fuel rods per assembly containing 5.0 weight percent U-235 enriched pellets is 52; Maximum U-235 enrichment is 4.5 weight percent for all edge rods, and 4.0 weight percent for all corner rods; Each assembly must include at least eight fuel rods with a minimum gadolinia content of 5.3 weight percent in all axial regions with enriched pellets. The eight gadolinia rods are arranged with two rods in each quadrant of the fuel assembly. The two gadolinia rods within each quadrant must be symmetric about the geometric diagonal of the fuel assembly, and must not be in an edge or corner rod location. Other fuel rods containing gadolinia may be present.

5.(b) Contents (continued)

(C) Maximum average U-235 enrichment is 4.858 weight percent within any axial zone of the assembly; Maximum U-235 content is 4.2 weight percent of any gadolinia-urania rod or axial zone of any gadolinia-urania fuel rod; Maximum number of fuel rods per assembly containing 5.0 weight percent U-235 enriched pellets is 80; Maximum U-235 enrichment is 4.0 weight percent for all corner rods; Each assembly must include at least twelve fuel rods with a minimum gadolinia content of 2.4 weight percent in all axial regions with enriched pellets. The twelve gadolinia rods are arranged with three rods in each quadrant of the fuel assembly. The three gadolinia rods within each quadrant must be symmetric about the geometric diagonal of the fuel assembly, and must not be in an edge or corner rod location. Other fuel rods containing gadolinia may be present.

(ii) Description of Assembly Type #2

Each assembly is made up of 96 fuel rods having a maximum active fuel length of 150 inches. Each assembly contains four one-third length fuel rods and eight two-thirds length fuel rods. The four one-third length fuel rods are located on the outside corners of the assembly. The eight two-thirds length fuel rods, arranged as two rods in each quadrant of the assembly, are located symmetric to the geometric diagonal, toward the center of the assembly. The fuel pellet diameter is 0.848 cm nominal, encapsulated in 0.061 cm nominal zirconium alloy cladding. There is a 0.0075 cm gap between the pellets and the cladding. The maximum U-235 enrichment of any fuel rod is 5.0 weight percent. Each assembly contains water holes in the four center rod positions of the assembly. The fuel assembly must be transported in channels. The specifications for each one-third length axial section of the fuel assembly are as follows:

(A) Upper section must contain 84 fuel rods, arranged as 21 rods per quadrant. Maximum U-235 enrichment of any rod is 5.0 weight percent. This section of the assembly must include at least eight fuel rods with a minimum gadolinia content of 4.0 weight percent in all axial regions with enriched pellets. The eight gadolinia rods are arranged with two rods in each quadrant of the fuel assembly, arranged symmetrically along the geometric diagonal of the assembly, and must not be in an edge or corner rod location. The section must contain 12 water holes, arranged as three water holes in each quadrant of the assembly. One of the three water holes within each quadrant must be located on the outside corner location of the assembly, and the other two water holes must be located on the geometric diagonal of the fuel assembly. Other fuel rods containing gadolinia may be present.

(B) Middle section must contain 92 fuel rods, arranged as 23 rods per quadrant. Maximum U-235 enrichment of any rod is 5.0 weight percent. This section of the assembly must include at least ten fuel rods with a minimum gadolinia content of 4.0 weight percent in all axial regions with enriched pellets. The ten gadolinia rods must be arranged symmetrically along the geometric diagonal of the assembly, and must not be in an edge or corner rod location. The section must contain four water holes, arranged as one water hole in each quadrant of the assembly. Each water hole within each quadrant must be located on the outside corner location of the assembly. Other fuel rods containing gadolinia may be present.

(C) Lower section must contain 96 fuel rods, arranged as 24 rods per quadrant. Maximum U-235 enrichment of any rod is 5.0 weight percent. This section of the assembly must include at least twelve fuel rods with a minimum gadolinia content of 4.0 weight percent in all axial regions with enriched pellets. The twelve gadolinia rods must be arranged symmetrically along the geometric diagonal of the assembly, and must not be in an edge or corner rod location. Other fuel rods containing gadolinia may be present.

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9292	5	71-9292	USA/9292/AF-85	4	OF	5

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

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

5.(c) Criticality Safety Index: 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. For the contents described in 5.(b)(1)(i), polyethylene inserts may be positioned between rods within the fuel assemblies. The quantity of polyethylene must not exceed 18.33 g polyethylene per centimeter length of the fuel assembly, and must not exceed a total of 6.99 kg per fuel assembly. The polyethylene may be borated. No polyethylene inserts may be used for the contents described in 5.(b)(1)(ii).

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. For packagings fabricated in accordance with Drawing No. 10015E58, Rev. 1 (referred to as version #2 inner containers), only Serial Nos. 001 through 039, inclusive, are authorized for use.

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

11. Revision No. 4 of this certificate may be used until August 31, 2007. Revision No. 3 of this certificate may be used until January 31, 2007.

12. Expiration date: August 31, 2010.

NRC FORM 618 (8-2000) 10 CFR 71	CERTIFICATE OF COMPLIANCE FOR RADIOACTIVE MATERIAL PACKAGES		U.S. NUCLEAR REGULATORY COMMISSION	
a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE / PAGES
9292	5	71-9292	USA/9292/AF-85	5 OF 5

REFERENCES

Vestinghouse Electric Company, LLC consolidated application dated: September 16, 2004.

upplements dated: April 14, June 14, August 9, and September 22, 2005; January 6, and May 13, 2006.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

for

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

ate: August 10, 2006

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9293	2	71-9293	USA/9293/B(U)F-85	1	OF	9

2. PREAMBLE

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

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

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

   a. ISSUED TO *(Name and Address)*
   Transnuclear, Inc.
   7135 Minstrel Way
   Columbia, MD 21045

   b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION
   Transnuclear, Inc., application dated May 19, 1999, as supplemented.

4. CONDITIONS

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

5. Packaging

   (1) Model No. TN-68 Transport Package

   (2) Description

   The TN-68 is predominantly a steel package that is used to transport up to 68 intact BWR fuel assemblies with or without channels. The overall dimensions of the package are 271 inches long and 144 inches in diameter with the impact limiters installed.

   The package generally consists of four components, the fuel basket assembly, a containment vessel within a forged steel cask body, a radial neutron shield, and impact limiters.

   The basket assembly locates and supports the fuel assemblies, transfers heat to the cask body wall and provides neutron absorption to satisfy sub-criticality requirements. The basket structure consists of an assembly of stainless steel cells, joined by fusion welding of 1.75 inch wide stainless steel plates. Above and below the plates are slotted borated aluminum (or boron carbide/aluminum) metal matrix composite neutron poison plates which form an egg-crate structure. This construction forms a honey-comb like structure of cell liners which provides compartments for 68 fuel assemblies. The nominal dimensions of each cell is 6.0 inches x 6.0 inches.

   A thick-walled (6.0 inch), forged steel cask body for gamma shielding surrounds the containment vessel, by an independent shell and bottom plate of carbon steel. The gamma shield completely surrounds the containment vessel inner shell and bottom closure. The thickness of the bottom of the cask body is 8.25 inches. A 4.5 inch thick steel gamma shield is also welded to the inside of the containment lid.

5.(a)(2) continued

The containment boundary consists of the inner shell and bottom plate, shell flange, lid outer plates, lid bolts, penetration cover plate and bolts and the inner metallic O-rings of the lid seal and the two lid penetrations (vent and drain). The containment vessel length is approximately 189 inches with a wall thickness of 1.5 inches. The cylindrical cask cavity has a nominal diameter of 69.5 inches and a length of 178 inches. The containment lid is 5 inches thick and is fastened to the cask body with 48 bolts. Double metallic O-ring seals are provided for lid closure. To preclude air in-leakage, the cask cavity is pressurized with helium to above atmospheric pressure. There are two penetrations through the containment vessel which are located in the lid. These penetrations are for draining and venting. Double metallic seals are also used on these two lid penetrations. The OP port provides access to the interspace lid seals for leak testing purposes. The OP transport cover is not part of the containment boundary.

Neutron shielding is provided by a borated polyester resin compound surrounding the gamma shield. The resin compound is cast into long, slender aluminum containers. The total thickness of the resin and aluminum is approximately 6 inches. The array of resin-filled containers is enclosed within a smooth 0.75 inch outer steel shell constructed of two half cylinders.

The package has impact limiters at each end of the cask body. The impact limiters consist of balsa wood and redwood blocks, encased in sealed stainless steel shells that maintain the wood in a dry atmosphere and provide wood confinement when crushed during a free drop. The impact limiters have internal radial gussets for added strength and confinement. The impact limiters are attaching to each other using 13 tie rods and to the cask by eight bolts attaching to brackets welded to the outer shell in eight locations (four bolting locations per impact limiter).

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

Overall length (with impact limiters, in)	271
Overall length (without impact limiters, in)	197
Impact Limiter Outside diameter, (in)	144
Outside diameter (without impact limiters, in)	98
Cavity diameter (in)	69.5
Cavity length (in)	178
Containment shell thickness (in)	1.5
Containment vessel length (in)	184
Body wall thickness (in)	7.5
Containment lid thickness (in)	5
Overall lid thickness (in)	9.5
Bottom thickness (in)	9.75
Resin and aluminum box thickness (in)	6
Outer shell thickness (in)	0.75
Overall basket length (in)	164
Maximum weight of package (pounds)	272,000
Maximum weight of BWR fuel contents (pounds)	47,900
Maximum weight of impact limiters and attachments (pounds)	32,000

5.(a)(3) Drawings

The package is constructed and assembled in accordance with TN drawings:

972-71-1, Revision 1
972-71-2, Revision 2
972-71-3, Revision 4
972-71-4, Revision 2
972-71-5, Revision 1
972-71-6, Revision 1
972-71-7, Revision 3
972-71-8, Revision 2
972-71-9, Revision 2
972-71-10, Revision 1
972-71-11, Revision 1
972-71-12, Revision 0
972-71-13, Revision 0
972-71-14, Revision 1

Contents

(1) Type and form of material

Contents are limited to 68 unconsolidated intact irradiated GE BWR fuel assemblies with zircalloy cladding. An intact fuel assembly is a spent nuclear fuel assembly without known or suspected cladding defects greater than pinhole leaks or hairline cracks. Partial fuel assemblies (i.e. spent fuel assemblies from which fuel rods are missing), shall not be classified as intact fuel assemblies unless dummy fuel rods are used to displace an amount of water equal to that displaced by the original rod(s).

Spent nuclear fuel may be transported with or without channels. Any fuel channel thickness up to 0.120 is acceptable on any of the fuel designs shown below. The maximum initial rod pressurization is 155 psig. The maximum fuel assembly length is 176.2 inches and the maximum fuel assembly width is 5.44 inches.

Permissible fuel assemblies are limited as stated in table 1 (fuel types may be C, D, or S lattice):

5.(b)(1) continued

Table 1, Fuel characteristics

GE fuel generation	model	array	rod pitch	fuel rods	rod od	clad thick	pellet dia.	water rods	water rod od	water rod id	U content (MTU/ Assembly)	Max active fuel length
2A	2a	7x7	0.738	49	0.570	0.036	0.488	0	x	x	0.1977	144
2, 2B	2	7x7	0.738	49	0.563	0.032	0.487	0	x	x	0.1977	144
3, 3A, 3B	3	7x7	0.738	49	0.563	0.037	0.477	0	x	x	0.1896	144
4, 4A,4B	4	8x8	0.640	63	0.493	0.034	0.416	1	0.493	0.425	0.1880	146
5	5	8x8	0.640	62	0.483	0.032	0.410	2	0.591	0.531	0.1876	150
6, 6B	5	8x8	0.640	62	0.483	0.032	0.410	2	0.591	0.531	0.1876	150
7, 7B	5	8x8	0.640	62	0.483	0.032	0.410	2	0.591	0.531	0.1876	150
8, 8B -2w	82	8x8	0.640	62	0.483	0.032	0.411	2	0.591	0.531	0.1885	150
[illegible]B-4W*	84	8x8	0.640	60	0.483	0.032	0.411	4	0.591	0.531	0.1824	150
[illegible]B-4W**	84	8x8	0.640	60	0.483	0.032	0.411	4	0.483	0.431	0.1824	150
9, 9B	9	8x8	0.640	60	0.483	0.032	0.411	1	1.34	1.26	0.1824	150
10	9	8x8	0.640	60	0.483	0.032	0.411	1	1.34	1.26	0.1824	150
11	11	9x9	0.566	74	0.440	0.028	0.376	2	0.98	0.92	0.1757	146 full, 90 partial
13	11	9x9	0.566	74	0.440	0.028	0.376	2	0.98	0.92	0.1757	146 full, 90 partial
12	12	10x10	0.510	92	0.404	0.026	0.345	2	0.98	0.92	0.1857	150 full, 93 partial

*2 large water rods
**2 small water rods

Notes on table 1:

1. All dimensions in inches.
2. All fuel channels 5.278 inches inside, and from 0.065 to 0.120 inches thick.
3. All fuels are evaluated with 96.5% theoretical density and 3.7 wt% U-235 average enrichment.
4. The fuel pitch is for C and D lattice designs. The S lattice fuels have a smaller pitch, which is less reactive.
5. The fuel designs designated by GE as 6, 6B, 7, and 7B are sometimes referred to as "P" (pressurized) and "B" (barrier).

Provided all of the requirements of this section are met, the bounding fuel characteristics are: a) maximum initial lattice-average enrichment is 3.7%; b) the mnimum initial bundle average enrichment is 3.3%; c) the maximum assembly average burnup is 40,000

5.(b)(1) continued

MWD/MTU; d) the minimum cool time is 10 years; and e) the maximum heat load per assembly is 0.313 Kw.

Fuel assemblies are categorized into three types, Type I, Type II and Type III. There are two basic loading configurations for the package. The first configuration is a mixture of Type I and Type II fuel assemblies. The second configuration is Type III fuel assemblies. The maximum burnup, minimum initial enrichments and cooling times for each of the three fuel assembly types is contained in the tables below.

In the mixed Type I and Type II configuration, Type I assemblies shall be placed only into the interior compartments of the fuel basket as shown in figure 5.3-3 of the application. Type II fuel assemblies may be placed in any basket fuel compartment.

In the second configuration, Type III fuel assemblies may be placed in any basket fuel compartment.

Acceptable cooling time as a function of maximum burnup and minimum initial enrichment and BWR Cooling times (years)
TYPE I BWR Fuel

	Burnup (GWd/MTU)											
Initial Enrichment (bundle ave %w)	**15**	**20**	**30**	**32**	**33**	**34**	**35**	**36**	**37**	**38**	**39**	**40**
1.0	10	10										
1.1	10	10										
1.2	10	10										
1.3	10	10										
1.4	10	10										
1.5	10	10	10	10	11	11	11					
1.6	10	10	10	10	10	11	11	11				
1.7	10	10	10	10	10	11	11	11	12			
1.8	10	10	10	10	10	11	11	11	11	12		
1.9	10	10	10	10	10	11	11	11	11	12		
2.0	10	10	10	10	10	10	11	11	11	12	12	
2.1	10	10	10	10	10	10	11	11	11	12	12	12
2.2	10	10	10	10	10	10	11	11	11	12	12	12
2.3	10	10	10	10	10	10	11	11	11	11	12	12
2.4	10	10	10	10	10	10	10	11	11	11	12	12
2.5	10	10	10	10	10	10	10	11	11	11	12	12
2.6	10	10	10	10	10	10	10	11	11	11	12	12
2.7	10	10	10	10	10	10	10	10	11	11	11	12
2.8	10	10	10	10	10	10	10	10	10	11	11	12
2.9	10	10	10	10	10	10	10	10	10	11	11	12
3.0	10	10	10	10	10	10	10	10	10	10	11	12
3.1	10	10	10	10	10	10	10	10	10	10	11	12
3.2	10	10	10	10	10	10	10	10	10	10	10	11
3.3	10	10	10	10	10	10	10	10	10	10	10	10
3.4	10	10	10	10	10	10	10	10	10	10	10	10
3.5	10	10	10	10	10	10	10	10	10	10	10	10
3.6	10	10	10	10	10	10	10	10	10	10	10	10
3.7	10	10	10	10	10	10	10	10	10	10	10	10

Acceptable cooling time as a function of maximum burnup and minimum initial enrichment and BWR Cooling times (years)

TYPE II BWR Fuel

Burnup (GWd/MTU)

Initial Enrichment (bundle ave %w)	15	20	30	32	33	34	35	36	37	38	39	40
1.0	18	21										
1.1	17	20										
1.2	17	20										
1.3	17	20										
1.4	17	20										
1.5	16	19	25	26	26							
1.6	16	19	25	26	26							
1.7	16	19	25	25	26	26	27					
1.8	16	19	24	25	26	26	27	27				
1.9	16	19	24	25	25	26	27	27				
2.0	16	18	24	25	25	26	26	27	28			
2.1	15	18	23	25	25	26	26	27	27			
2.2	15	18	23	25	25	25	26	27	27			
2.3	15	18	23	24	25	25	26	26	27	27		
2.4	15	18	22	24	24	25	26	26	27	27		
2.5	15	17	22	24	24	25	25	26	26	27		
2.6	15	17	22	24	24	24	25	26	26	27		
2.7	15	17	22	24	24	24	25	26	26	26	27	27
2.8	14	17	22	23	24	24	25	25	26	26	27	27
2.9	14	17	22	23	23	24	24	25	26	26	27	27
3.0	14	17	21	23	23	23	24	25	25	26	27	27
3.1	14	17	21	23	23	23	24	25	25	26	27	27
3.2	13	16	21	23	23	23	24	24	25	25	26	27
3.3	13	16	21	23	22	23	23	24	25	25	26	26
3.4	13	16	21	23	22	23	23	24	25	25	26	26
3.5	13	16	21	22	22	23	23	24	25	25	26	26
3.6	13	16	21	21	22	22	23	24	25	25	26	26
3.7	12	15	20	21	22	22	23	24	25	25	25	26

Acceptable cooling time as a function of maximum burnup and minimum initial enrichment and BWR Cooling times (years)
TYPE III BWR Fuel

Burnup (GWd/MTU)

Initial Enrichment (bundle ave %w)	15	20	30	32	33	34	35	36	37	38	39	40
1.0	10	11	X	X	X	X	X	X	X	X	X	X
1.1	10	11	X	X	X	X	X	X	X	X	X	X
1.2	10	10	X	X	X	X	X	X	X	X	X	X
1.3	10	10	X	X	X	X	X	X	X	X	X	X
1.4	10	10	X	X	X	X	X	X	X	X	X	X
1.5	10	10	15	16	16	17	17	X	X	X	X	X
1.6	10	10	14	16	16	17	17	17	X	X	X	X
1.7	10	10	14	15	16	16	17	17	17	X	X	X
1.8	10	10	14	15	15	16	16	17	17	18	X	X
1.9	10	10	14	15	15	16	16	17	17	18	X	X
2.0	10	10	14	15	15	16	16	16	17	17	18	X
2.1	10	10	14	15	15	15	16	16	16	17	18	18
2.2	10	10	13	14	15	15	16	16	16	17	17	18
2.3	10	10	13	14	15	15	16	16	16	17	17	18
2.4	10	10	13	14	15	15	15	16	16	17	17	18
2.5	10	10	13	14	14	15	15	16	16	16	17	18
2.6	10	10	13	14	14	15	15	16	16	16	17	17
2.7	10	10	13	14	14	15	15	15	16	16	17	17
2.8	10	10	13	13	14	14	15	15	16	16	17	17
2.9	10	10	13	13	14	14	15	15	15	16	16	17
3.0	10	10	12	13	14	14	14	15	15	16	16	17
3.1	10	10	12	13	14	14	14	15	15	15	16	16
3.2	10	10	12	13	14	14	14	15	15	15	16	16
3.3	10	10	12	13	13	14	14	14	15	15	16	16
3.4	10	10	12	13	13	13	14	14	15	15	16	16
3.5	10	10	12	13	13	13	14	14	14	15	15	16
3.6	10	10	12	12	13	13	14	14	14	15	15	15
3.7	10	10	12	12	13	13	14	14	14	15	15	15

5.(b) continued

(2) Maximum quantity of material per package

The maximum contents weight is 75,600 pounds. The maximum weight of the irradiated fuel contents is 47,900 pounds.

(3) Decay Heat Limit

Maximum decay heat per package not to exceed 21.2kW. The maximum heat load per assembly is 0.313 kW/assembly.

(c) Criticality Safety Index 0.0

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

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

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

7. Known or suspected fuel assemblies with cladding defects greater than pin hole leaks and or hairline cracks are not authorized.

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

9. Revision No. 1 of this certificate may be used until February 28, 2007.

10. Expiration date: February 28, 2011.

REFERENCES

Transnuclear, Inc., application dated May 19, 1999.

Supplements dated March 2, October 18, and November 13, 2000, January 12, 2001, and January 20, 2006.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

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

Date: February 10, 2006

NRC FORM 618
(8-2000)
10 CFR 71

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
9294	5	71-9294	USA/9294/AF-96	1	OF	3

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

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

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Global Nuclear Fuel - Americas, LLC, application dated January 29, 2001.

4. CONDITIONS

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

5.

(a) Packaging

(1) Model No.: NPC

(2) Description

A cubic stainless steel and foam outer packaging with nine cylindrical containment vessels for the transport of type A quantities of low-enriched uranium oxide powder, pellets, and compounds of uranium as defined in 5(b). The overall package dimensions are approximately 45 inches wide, 45 inches deep, and 44 inches high.

The outer packaging consists of a 10-gage stainless steel outer shell with a ceramic fiber board liner and rigid polyurethane foam filler. The foam filler has a three-by-three array of vertical cylindrical cutouts that accommodate stainless steel sleeves for placement of the containment vessels. The outer packaging is equipped with a top cover that is secured to the outer packaging body by a combination of 16 closure cap screws and four closure strips secured by 24 bolts.

The containment vessel is a maximum 8.515 inches in inner diameter and approximately 32 inches in overall length. The containment vessel is constructed of 18-gage stainless steel, surrounded by a cadmium sheet and polyethylene wrap within a 24-gage stainless steel jacket. The containment vessel is closed by a 16-gage closure lid, a silicone rubber gasket, and a band clamp assembly, which is composed of a 0.063-inch thick strap and retainer, a T-bolt, and a nut.

The gross weight of the package (packaging and contents) is 1,302 kg (2,870 pounds). The maximum weight of the contents is 540 kg (1,190 pounds).

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE	PAGES
9294	5	71-9294	USA/9294/AF-96	2 OF	3

5.(a) (3) Drawings

The packaging is fabricated and assembled in accordance with the following Global Nuclear Fuel - Americas, LLC, Drawing Nos.:

177D4970, Sheet 1, Revision 1
177D4970, Sheet 2, Revision 0
177D4970, Sheet 3, Revision 0
177D4970, Sheet 4, Revision 0
177D4970, Sheet 5, Revision 0
177D4970, Sheet 6, Revision 0
177D4970, Sheet 7, Revision 0
177D4970, Sheet 8, Revision 1
SK105E4037, Sheet 2, Revision 2

(b) Contents

Type, Form, and Maximum Quantity of Material Per Package

Material Forms[1] (≤5.00 wt.% U-235)	Particle Size Restriction: Minimum OD (Inches)	Maximum Loading per ICCA (kgs)		Maximum Loading per NPC (kgs)	
		Net[4]	Uranium	Net[4]	Uranium
Homogenous Uranium Oxide/Compounds[2]	N/A	60.0	52.89	540.0	476.1
Heterogenous UO_2 Pellets (BWR)	0.342	60.0	40.54	540.0	364.8
Heterogenous UO_2 Pellets(PWR)	0.300	60.0	40.54	540.0	364.8
Heterogenous Uranium Compounds[3]	Unrestricted particle size	60.0	40.54	540.0	364.8

[1]No solutions or liquids are authorized and there shall be no free liquid present. The Material Form within any NPC must be the same.
[2]Homogenous compounds limited to UO_2, U_3O_8, $UO_{x,\,x>2}$, dried calcium-containing sludges, $UO_2(NO3)_2{\cdot}6H_2O$, and uranium oxide bearing ash.
[3]Heterogenous compounds limited to UO_2, U_3O_8, and $UO_{x,\,x>2}$.
[4]Maximum content weight of any ICCA including plastic or secondary packaging (i.e., dunnage). Materials with a hydrogen atom density greater than that of water are limited to a mass of 3.7 kg per ICCA.

(c) Criticality Safety Index 0.7

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

   (a) The package must be prepared for shipment and operated in accordance with the Operating Procedures in Chapter 7 of the application, as supplemented. Within each ICCA, the contents and secondary packaging (i.e., dunnage) must provide a snug fit. The payload may be enclosed in plastic receptacles (e.g., bags, bottles, etc.). For payloads in plastic bottles, empty bottles may be used to minimize movement of the bottles within the ICCA.

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

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

8. Transport by air of fissile material is not authorized.

9. Revision No. 4 of this certificate may be used until November 30, 2008.

10. Expiration date: November 30, 2010.

REFERENCES

Global Nuclear Fuel - Americas, LLC, application dated January 29, 2001.

Supplements dated: August 1, 23 and 27, 2001; March 4 and September 30, 2002; June 30 and October 3, 2005; October 3, 2006; April 27, July 31, and October 3, 2007.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Robert A. Nelson, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: November 9, 2007

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE	PAGES
9295	2	71-9295	USA/9295/B(U)F-96	1 OF	8

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

AREVA Federal Services LLC
1102 Broadway Plaza, Suite 300
Tacoma, WA 98402-3526

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Packaging Technology, Inc., application dated June 25, 2004, as supplemented.

4. CONDITIONS

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

(a) Packaging

(1) Model No.: MFFP

(2) Description

The MFFP is designed to transport unirradiated mixed oxide (MOX) fuel assemblies and individual MOX fuel rods contained in rod boxes.

The MFFP body is made of a 9/16-inch thick XM-19 austenitic stainless steel cylindrical shell with the flange section and a 1-1/2 inch bottom end plate welded to it. A circumferentially continuous doubler plate, constructed of Type XM-19 austenitic stainless steel, is welded to each end of the shell, near the end of each impact limiter. Welded to the doubler plate are the impact limiter attachment lugs, six per impact limiter. The doubler plate also serves to provide a tiedown interface with the transportation skid.

The seal flange is located at the open end of the body, and consists of a locally thicker wall section to accommodate the closure lid sealing area and the closure bolt threaded holes. The transition between the shell and the seal flange section is a 3:1 taper. Polyurethane foam is used to build the outer diameter of the body out to the full diameter of the sealing flange and closure lid.

5.(a)(2) continued

The closure lid is a weldment constructed of Type XM-19 3/4-inch outer plate and 5/8-inch thick inner plate, stiffened with eight 1/2-inch thick radial ribs that are three inches deep. A 1/2-inch thick, 6 inch inner diameter cylinder forms a hub at the inner end of the radial ribs. The ribs are welded on all four edges to the adjacent structure. Each rib has a projection that passes through a slot in the outer plate, and the ribs and outer plate are welded together.

The closure lid inner plate is welded to the outer ring. The seal flange of the closure lid has a minimum thickness of one inch, and provides location for three O-ring bore seals with the middle seal providing the containment seal. The seals are 3/8-inch diameter butyl rubber O-ring.

Up to three unirradiated fuel assemblies are held in place inside the overpack by a strongback assembly which is constructed from 1/4-inch thick Type 304 stainless steel weldment, a series of clamp arm assemblies, a top, and a bottom plate assemblies. For shipping less than three fuel assemblies, non-fuel dummy assemblies are used in the strongback locations not occupied by the fuel assemblies. The physical size and weight of the non-fuel dummy assemblies are nominally the same as the MK-BW/MOX1 17 x 17 design. Neutron poison plates are placed inside the weldment. A series of fuel control structure (FCS) limits lateral expansion of fuel rods during vertical and near vertical hypothetical accident condition (HAC) free drops and also hold neutron poison plates.

A pair of conical-shaped impact limiters filled with polyurethane foam provide thermal and impact protections. The closure lid end impact limiter has 1/4-inch thick shells to resist perforation from the HAC puncture drop, and to protect the closure lid and sealing area from puncture and HAC fire damage. Shock indicators are attached to the outside of the MFFP shell.

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

Overall package outside dimensions (inches)		
Without Impact Limiters		
Diameter	30	
Length	171	
With Impact Limiters		
Diameter	60	
Length	201	
Maximum content weight		4,740 lbs
Maximum package weight (Including contents)		14,260 lbs

(3) Drawings

The packaging shall be constructed and assembled in accordance with Packaging Technology, Inc., drawing numbers:

(a) Shipping Package	99008-10, Rev. 4, Sheet 1
(b) Body Assembly	99008-20, Rev. 3, Sheets 1 through 6
(c) Strongback Assembly	99008-30, Rev. 5, Sheets 1 through 7
(d) Top Plate Assembly	99008-31, Rev. 1, Sheets 1 through 3
(e) Bottom Plate Assembly	99008-32, Rev. 1, Sheets 1 and 2
(f) Clamp Arm Assembly	99008-33, Rev. 3, Sheets 1 through 4
(g) Fuel Control Structure Assembly	99008-34, Rev. 4, Sheets 1 and 2
(h) Impact Limiter	99008-40, Rev. 2, Sheets 1 through 3
(i) AFS-B Assembly	99008-60, Rev. 1, Sheets 1 and 2
(j) AFS-C Assembly	99008-61, Rev. 1, Sheets 1 and 2

(b) Contents

(1) Type and Form of Material

Unirradiated 17 x 17 fuel assemblies with solid PuO_2+UO_2 pellets in zirconium based alloy (M5) tubes. The fuel assemblies are based on the MK-BW/MOX1 17 x 17 PWR design. The fuel assemblies may contain Burnable Poison Rod Assemblies (BPRA). The physical specifications for the unirradiated fuel assemblies and the burnable poison rod assemblies are provided in Tables 1 and 2. For shipping less than three fuel assemblies, non-fuel dummy assemblies are used in the strongback locations not occupied by the fuel assemblies. The physical size and weight of the non-fuel dummy assemblies are nominally the same as the MK-BW/MOX1 17 x 17 design.

5.(b)(1) continued

The ARB-17 is a rod container designed to transport up to 17 MOX fuel rods. The rods type is identical to the rods comprising the standard MOX fuel assembly. The rods may be either undamaged, damaged, or a combination of both (e.g., 9 undamaged and 8 damaged). Damaged fuel rods may be bent, scratched, or dented, but under no circumstances may exhibit cladding breaches. A 2-inch Schedule 40 pipe mounted with pipe clamps against one wall of the ARB-17 is used to transport undamaged or slightly damaged fuel rods. Damaged fuel rods may be transported within this pipe only if the bending in the fuel rod is minor. The ARB-17 MOX fuel rod container has been designed with outer dimensions consistent with a standard fuel assembly so that it will interface with the strongback and clamp arms.

The AFS-B Rod Container is designed to contain up to 175 MOX fuel rods. The container has outer cross sectional dimensions of 8.4 inches square, a length from bottom to top of 159.9 inches, and an overall length (to the lift ring bolt head) of 161.2 inches. The primary material of construction of the container is ASTM 6061-T651 aluminum alloy.

The AFS-C Rod Container is designed to contain up to 116 Exxon rods, up to 69 Pacific Northwest Laboratory (PNL) rods, or both quantities together. The container is the same as the AFS-B Rod Container except the AFS-C container has two internal 2-inch thick aluminum plates which form rod cavities to accommodate both types of rods the AFS-C Rod Container may hold.

The EMA is similar to MOX fuel assemblies with the exceptions that the OD of the fuel pellets may be out of tolerance (nominal pellet diameter = 0.323 inch), and the weight percent Pu-238 exceeds the 0.05 wt.% limit specified in Table 1.2-2 of the SAR (EMA fuel rods have Pu-238/Pu as high as 0.19 wt.%).

5.(b)(1) continued

Table 1 - Fuel Assembly Physical Parameters
(nominal values unless stated otherwise)

Parameter	Values
Fuel Rod Cladding Material	M5
Fuel Rod Array	17 x 17
Fuel Rods per Fuel Assembly	264
Guide Tubes per Fuel Assembly	24
Instrument Tubes per Fuel Assembly	1
Guide/Instrument Tube Thickness (inches)	0.016
Fuel Assembly Length (inches)	161.61
Fuel Assembly Maximum Width (inches)	8.565
Fuel Rod Pitch (inches)	0.496
Fuel Rod Length (inches)	152.4
Fuel Rod Outside Diameter (inches)	0.374
Fuel Rod Clad Thickness (inches)	0.023
Active Fuel Length (inches)	144.0
PuO_2 + UO_2 Weight (pounds)	1,157
Heavy Metal Weight (pounds)	1,020
Maximum Fuel Assembly Weight including Burnable Poison Rod Assembly (pounds)	1,580
Maximum Initial Pu Loading (weight% of heavy metal)	6.0

Table 2 - Burnable Poison Rod Assembly Parameters

Parameter	Value
Poison Rod Cladding Material	Zircaloy-4
Poison/Thimble Plug Rod Array	Up to 24 rods
Burnable Poison Material	Al_2O_3-B_4C

5. (b) (2) Maximum Quantity of Material per Package

Three unirradiated fuel assemblies with specifications on fuel pellets and enrichment are provided in Table 3. Three Areva Rod Box 17 (ARB-17) containers may contain up to 17 standard MOX fuel rods. One AFS-B rod container may contain up to 175 standard MOX fuel rods and one Excess Material Assembly. Three AFS-C rod containers may contain up to 116 Exxon rods and 69 PNL rods. The permissible configurations of contents are summarized in Table 4.

Table 3 - Nuclear Design Parameters for Fuel Assemblies

Parameter	Value
Nominal Pellet Diameter (inches)	0.323
Maximum Effective Pellet Density (gram/cm^3)	10.85
Maximum Total Plutonium (Pu) Content	0.06 g Pu/g Heavy Metal (Pu+U)
Plutonium Isotopic Contents	Pu-238: Up to 0.0005 g/g Pu Pu-239: 0.90 to 0.95 g/g Pu Pu-240: 0.05 to 0.09 g/g Pu Pu-241: Up to 0.01 g/g Pu Pu-242: Up to 0.001 g/g Pu
Minimum Total Uranium (U) Content	0.94 g U/g Heavy Metal (Pu+U)
Uranium Isotopic Contents	U-234: Up to 0.0005 g/g U U-235: Up to 0.003 g/g U U-238: Remainder of U content

Table 4 – Payload Table

Payload Type	Strongback Positions (3)		
MOX Fuel	MOX FA	MOX FA or dummy FA	MOX FA or dummy FA
MOX Fuel and ARB-17 Rod Container	MOX FA or ARB-17	MOX FA, ARB-17, or dummy FA	MOX FA, ARB-17, or dummy FA
EMA	EMA	dummy FA	dummy FA
AFS-B and EMA	AFS-B	EMA or dummy FA	dummy FA
AFS-C	AFS-C	AFS-C or dummy FA	AFS-C or dummy FA

(c) Criticality Safety Index 0.0

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

(a) The package shall be prepared for shipment and operated in accordance with the Package Operations of Chapters 7, 7A, 7B, and 7C of the application, as applicable, as supplemented.

(b) The packaging must meet the Acceptance Tests and Maintenance Program of Chapters 8, 8A, 8B, and 8C of the application, as applicable, as supplemented.

(c) The boron-10 areal density within each of the internal neutron poison plates shall be verified as described in Section 8.1.5.2 of the application, as supplemented.

(d) Wrapping shall not be used on the unirradiated fuel assemblies.

(e) Non-fuel dummy assemblies with the same nominal size and weight as the MK-BW/MOX1 17 x 17 design shall be used in the case of loading less than three fuel assemblies in a MFFP packaging.

7. Transport by air of fissile material is not authorized.

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

9. Revision No. 1 of this certificate may be used until June 30, 2009.

10. Expiration date: June 30, 2010.

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9295	2	71-9295	USA/9295/B(U)F-96	8	OF	8

REFERENCES

Packaging Technology, Inc., application dated June 25, 2004.

Supplement dated: February 4 and 10, April 8, June 3, 2005, and January 19, August 15, and November 26, 2007, and April 4 and July 25, 2008.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Eric J. Benner, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: August 4, 2008

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9296	6	71-9296	USA/9296/B(U)-96	1	OF	3

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

QSA Global, Inc.
40 North Avenue
Burlington, MA 01803

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

QSA Global, Inc., consolidated application dated October 20, 2005.

4. CONDITIONS

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

5.

(a) Packaging

(1) Model No. 880 Series Packages

(2) Description

The Model No. 880 series packages are designed for use as a radiography exposure device and a transport package for Type B quantities of radioactive material in special form. The Model No. 880 has three versions called the 880 Delta, 880 Sigma and the 880 Elite. The 880 Delta has a maximum capacity of 150 Curies of Iridium-192 or 150 Curies of Selenium-75, the 880 Sigma has a maximum capacity of 130 Curies of Iridium-192 or 150 Curies of Selenium-75, and the 880 Elite has a maximum capacity of 50 Curies of Iridium-192 or 150 Curies of Selenium-75. The Delta and Sigma versions are identical and the Elite has a lighter weight depleted uranium shield. An optional jacket can be placed on the packages when they are used as an industrial radiography exposure device or a transport package.

All versions of the package, without the jacket, are cylindrical in shape with a diameter of 5 inches and a length of 13 5/16 inches. With the jacket, the shape of the packages is an extruded triangle 9 inches high, 7 ½ inches wide, and 13 5/16 inches long. The weight of the Delta and Sigma versions is 46 pounds (52 pounds with the jacket) and the Elite version is 37 pounds (42 pounds with the jacket).

The major components of the packages consist of a welded stainless steel cylindrical body, a depleted uranium shield, a stainless steel rear plate with a locking assembly, a stainless steel front plate with a shielded port, and an optional jacket.

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

The welded cylindrical body consists of a five inch diameter, 0.06 inch wall tube shell with 0.12 inch end-plates. A U-bracket is welded to each end-plate and is located on the inside cavity of the shell tube. The depleted uranium shield is centrally located within the welded body between the end-plate and is fastened to each U-bracket by a 0.37 inch diameter titanium shield pin. A U-shaped copper spacer fills the gap between the shield and the U-bracket. An S-shaped titanium source tube is cast into the center of the shield to provide a cavity for the source wire assembly to travel through during use.

The front and rear plates are attached to the welded body with four tamperproof screws through rivnuts assembled into end-plates. The rear plate assembly consists of a source locking mechanism fastened to the rear plate. The front plate assembly consists of a shielded port mechanism contained within the front plate.

An optional polyurethane jacket covers the package cylinder and provides a handle and a stable base. The jacket handle contains a wire molded in for additional reinforcement.

(3) Drawings

The packaging is constructed in accordance with the AEA Technology/QSA, Inc., drawings R88000, Rev. J, Sheets 1-5.

(b) Contents

(1) Type and form of material

Iridium-192 as a sealed source which meets the requirements of special form radioactive material.

Selenium-75 as a sealed source which meets the requirements of special form radioactive material.

(2) Maximum quantity of material per package

150 Curies (5.55 TBq) (output) Ir-192 for the Model No. 880 Delta.
150 Curies (5.55 TBq) (output) Se-75 for the Model No. 880 Delta.

130 Curies (4.81 TBq) (output) Ir-192 for the Model No. 880 Sigma.
150 Curies (5.55 TBq) (output) Se-75 for the Model No. 880 Sigma.

50 Curies (1.85 TBq) (output) Ir-192 for the Model No. 880 Elite.
150 Curies (5.55 TBq) (output) Se-75 for the Model No. 880 Elite.

5. (b) (2) Contents (continued)

Output curies are determined by measuring the source output at 1 meter and expressing its activity in curies derived from the following: 0.48 R/hr - Ci Iridium-192 at 1 meter and 0.20 R/hr - Ci Selenium-75 at 1 meter. (Ref: Radiological Health Handbook, rev. ed., U.S. Public Health Service, Bureau of Radiological Health, Rockville, MD, 1970.)

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

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

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

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

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

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

10. Revision No. 5 of this certificate may be used until August 31, 2007.

11. Expiration date: March 31, 2011.

REFERENCES

QSA Global, Inc., consolidated Safety Analysis Report dated October 20, 2005.

Supplement dated July 19, 2006.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

[signature] for

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

Date: September 1, 2006

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
9297	2	71-9297	USA/9297/AF-96	1	OF	6

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

Westinghouse Electric Company
P.O. Drawer R
Columbia, SC 29250

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Westinghouse Electric Company application dated April 1, 2004, as supplemented.

4. CONDITIONS

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

5.

(a) Packaging

(1) Model Nos.: Traveller STD and Traveller XL

(2) Description

The Traveller package is designed to transport non-irradiated uranium fuel assemblies or rods with enrichment up to 5.0 weight percent. The package is designed to carry one fuel assembly or one container for loose rods. The package consists of three components: 1) an outerpack, 2) a clamshell, and 3) a fuel assembly or rod container.

The outerpack is a structural component that serves as the primary impact and thermal protection for the fuel assembly or rod container. The outerpack has a long horizontal tubular design consisting of a top and bottom half. At each end of the package are thick limiters consisting of two sections of foam of different densities sandwiched between three layers of sheet metal. The impact limiters are integral parts of the outerpack and reduce damage to the contents during an end, or high-angle drop. The outerpack also provides for lifting, stacking, and tie down during transportation.

The clamshell is a horizontal structural component that serves to protect the contents during routine handling and in the event of an accident. The clamshell consists of an aluminum "V" extrusion, two aluminum door extrusions, and a small access door. Each extruded aluminum door is connected to the "V" extrusion with piano-type hinges (continuous hinges). These doors are held closed with a latching mechanism and quarter-turn bolts. Neutron absorber plates are installed in each leg of the "V" extrusion and in each of the doors. The "V" extrusion and the bottom plate are lined with a cork rubber pad to cushion and protect the contents during normal handling and transport conditions.

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

The Traveller package is designed to carry loose rods using either of two types of rod containers: a rod box or rod pipe. The rod box is an ASTM, Type 304 stainless steel container of rectangular cross section with stiffening ribs located approximately every 60 centimeters (cm) (23.6 inches (in.)) along its length. It is secured by fastening a removable top cover to the container body using socket head cap screws. The rod pipe consists of a 15.2 cm (6 in.) standard 304 stainless steel, Schedule 40 pipe, and standard 304 stainless steel closures at each end. The closure is a 0.635 cm (0.25 in.) thick cover secured with Type 304 stainless steel hardware to a flange fabricated from 0.635 cm (0.25 in.) thick plate.

There are two models of the Traveller packaging, the Traveller STD and the Traveller XL.

Traveller STD:

Package gross weight	2,041 kilograms (kg) (4,500 pounds (lbs))
Packaging gross weight	1,293 kg (2,850 lbs)
Contents gross weight	748 kg (1,650 lbs)
Outer dimensions	
Length	500 cm (197 in.)
Width	68.6 cm (27.1 in.)
Height	100 cm (39.3 in.)

Traveller XL:

Package gross weight	2,313 kg (5,100 lbs)
Packaging gross weight	1,419 kg (3,129 lbs)
Contents gross weight	894 kg (1,971 lbs)
Outer dimensions	
Length	574 cm (226.1 in.)
Width	68.6 cm (27.1 in.)
Height	100 cm (39.3 in.)

(3) Drawings

The packagings are fabricated and assembled in accordance with the following Westinghouse Electric Company's Drawing Nos.:

10004E58, Rev. 4 (Sheets 1-8)
10006E58, Rev. 5
10006E59, Rev. 1 (Sheets 1-2)

(b) Contents (Type and Form of Material)

(1) Fuel Assembly

(i) Unirradiated PWR uranium dioxide fuel assemblies with a maximum uranium-235 enrichment of 5.0 weight percent. The parameters of the fuel assemblies that are permitted are as follows:

5.(b)(1)(i) Fuel Assembly (Continued)

Parameters for 14 x 14 Fuel Assemblies

Fuel Assembly Description	14 x 14	14 x 14	14 x 14
Fuel Assembly Type	W-STD	W-OFA	CE-1/CE-2
No. of Fuel Rods per Assembly	179	179	176
No. of Non-Fuel Rods	17	17	20
Nominal Guide Tube Wall Thickness	0.043 cm (0.017 in.)	0.043 cm (0.017 in.)	0.097 cm (0.038 in.)
Nominal Guide Tube Outer Diameter	1.369 cm (0.539 in.)	1.336 cm (0.526 in.)	2.822 cm (1.111 in.)
Nominal Pellet Diameter	0.929 cm (0.366 in.)	0.875 cm (0.344 in.)	0.956/0.966 cm (0.376/0.381 in.)
Nominal Clad Outer Diameter	1.072 cm (0.422 in.)	1.016 cm (0.400 in.)	1.118 cm (0.440 in.)
Nominal Clad Thickness	0.062 cm (0.024 in.)	0.062 cm (0.024 in.)	0.071/0.066 cm (0.028/0.026 in.)
Clad Material	Zirconium alloy	Zirconium alloy	Zirconium alloy
Nominal Assembly Envelope	19.70 cm (7.76 in.)	19.70 cm (7.76 in.)	20.60 cm (8.11 in.)
Nominal Lattice Pitch	1.412 cm (0.556 in.)	1.412 cm (0.556 in.)	1.473 cm (0.580 in.)

Parameters for 15 x 15 Fuel Assemblies

Fuel Assembly Description	15 x 15	15 x 15
Fuel Assembly Type	STD/OFA	B&W
No. of Fuel Rods per Assembly	205	208
No. of Non-Fuel Rods	20	17
Nominal Guide Tube Wall Thickness	0.043/0.043 cm (0.017/0.017 in.)	0.043 cm (0.017 in.)
Nominal Guide Tube Outer Diameter	1.387/1.354 cm (0.546/0.533 in.)	1.354 cm (0.533 in.)
Nominal Pellet Diameter	0.929 cm (0.366 in.)	0.929 cm (0.366 in.)
Nominal Clad Outer Diameter	1.072 cm (0.422 in.)	1.072 cm (0.422 in.)
Nominal Clad Thickness	0.062 cm (0.024 in.)	0.062 cm (0.024 in.)
Clad Material	Zirconium alloy	Zirconium alloy
Nominal Assembly Envelope	21.39 cm (8.42 in.)	21.66 cm (8.53 in.)
Nominal Lattice Pitch	1.430 cm (0.563 in.)	1.443 cm (0.568 in.)

5.(b)(1)(i) Fuel Assembly (Continued)

Parameters for 16 x 16 Fuel Assemblies

Fuel Assembly Description	16 x 16	16 x 16	16 x 16	16 x 16
Fuel Assembly Type	W-STD	CE	NGF	ATOM
No. of Fuel Rods per Assembly	235	236	235	236
No. of Non-Fuel Rods	21	20	21	20
Nominal Guide Tube Wall Thickness	0.046 cm (0.018 in.)	0.102 cm (0.040 in.)	0.041 cm (0.016 in.)	0.057 cm (0.023 in.)
Nominal Guide Tube Outer Diameter	1.196 cm (0.471 in.)	2.489 cm (0.980 in.)	1.204 cm (0.474 in.)	1.354 cm (0.533 in.)
Nominal Pellet Diameter	0.819 cm (0.323 in.)	0.826 cm (0.325 in.)	0.784 cm (0.309 in.)	0.914 cm (0.360 in.)
Nominal Clad Outer Diameter	0.950 cm (0.374 in.)	0.970 cm (0.382 in.)	0.914 cm (0.360 in.)	1.075 cm (0.423 in.)
Nominal Clad Thickness	0.057 cm (0.023 in.)	0.064 cm (0.025 in.)	0.057 cm (0.023 in.)	0.072 cm (0.029 in.)
Clad Material	Zirconium alloy	Zirconium alloy	Zirconium alloy	Zirconium alloy
Nominal Assembly Envelope	19.72 cm (7.76 in.)	20.63 cm (8.12 in.)	19.72 cm (7.76 in.)	22.95 cm (9.03 in.)
Nominal Lattice Pitch	1.232 cm (0.485 in.)	1.285 cm (0.506 in.)	1.232 cm (0.485 in.)	1.430 cm (0.563 in.)

Parameters for 17 x 17 and 18 x 18 Fuel Assemblies

Fuel Assembly Description	17 x 17	17 x 17	18 x 18
Fuel Assembly Type	W-STD/XL	W-OFA	ATOM
No. of Fuel Rods per Assembly	264	264	300
No. of Non-Fuel Rods	25	25	24
Nominal Guide Tube Wall Thickness	0.041/0.051 cm (0.016 /0.020 in.)	0.041 cm (0.016 in.)	0.065 cm (0.026 in.)
Nominal Guide Tube Outer Diameter	1.204/1.224/1.24 cm (0.474/0.482/0.488 in.)	1.204 cm (0.474 in.)	1.240 cm (0.488 in.)
Nominal Pellet Diameter	0.819 cm (0.323 in.)	0.784 cm (0.309 in.)	0.805 cm (0.317 in.)
Nominal Clad Outer Diameter	0.950 cm (0.374 in.)	0.914 cm (0.360 in.)	0.950 cm (0.374 in.)
Nominal Clad Thickness	0.057 cm (0.023 in.)	0.057 cm (0.023 in.)	0.064 cm (0.025 in.)
Clad Material	Zirconium alloy	Zirconium alloy	Zirconium alloy
Nominal Assembly Envelope	21.39 cm (8.42 in.)	21.39 cm (8.42 in.)	22.94 cm (9.03 in.)
Nominal Lattice Pitch	1.260 cm (0.496 in.)	1.260 cm (0.496 in.)	1.270 cm (0.500 in.)

5.(b)(1) Fuel Assembly (Continued)

(ii) Non-fissile base-plate mounted core components and spider-body core components are permitted.

(iii) Neutron sources or other radioactive material are not permitted.

(iv) Materials with moderating effectiveness greater than full density water are not permitted.

(v) There is no restriction on the length of top and bottom annular blankets.

(2) Loose Fuel Rods

Unirradiated uranium-dioxide fuel rods with a maximum uranium-235 enrichment of 5.0 weight percent. Fuel rods shall be transported in the Traveller package inside either a rod pipe or rod box as specified in License Drawings 10006E58 or 10006E59, specified in Section 5(a)(3). The fuel rods shall meet the parametric requirements given below:

Parameter	Limit
Maximum Enrichment	5.0 weight percent uranium-235
Pellet diameter	0.508 – 1.524 cm (0.20 – 0.60 in.)
Maximum stack length	Up to rod container length
Cladding	Zirconium alloy
Integral absorber	Gadolinia, erbia, and boron
Wrapping or sleeving	Plastic or other material with moderating effectiveness no greater than full density water
Maximum number of rods per container	Up to rod container capacity

5.(c) Criticality Safety Index

(1) When transporting fuel assemblies: 0.7

(2) When transporting loose rods in a rod container: 0.0

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
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6. In addition to the requirements of Subpart G of 10 CFR Part 71:

   (a) The package must be prepared for shipment and operated in accordance with the Operating Procedures in Chapter 7 of the Traveller License Application, Revision 4.

   (b) Each packaging must be acceptance tested and maintained in accordance with the Acceptance Tests and Maintenance Program in Chapter 8 of the Traveller License Application, Revision 4.

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

8. The package is not authorized by this certificate for air transport.

9. Revision No. 1 of this certificate may be used until December 31, 2007.

10. Expiration date: March 15, 2010.

REFERENCES

Westinghouse Electric Company application dated April 1, 2004.

Supplements dated: October 15 and November 16, 2004, and February 16, March 4, and March 10, 2005, and March 17 and April 12, 2006, September 26 and December 12, 2006.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Christopher M. Regan, Acting Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: December 21 2006

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9299	2	71-9299	USA/9299/B(U)-85	1	OF	3

2. PREAMBLE

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

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

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

   a. ISSUED TO *(Name and Address)*

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

   b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

   MDS Nordion application dated November 29, 2006, as supplemented.

4. CONDITIONS

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

5.

(a) Packaging

(1) Model No.: F-423

(2) Description

A double-walled welded stainless steel overpack for shipping sealed sources within the Gammacell 220 (GC220) gamma irradiator. The packaging consists of concentric box-like stainless steel shells separated by an annulus of rigid polyurethane foam. The overall overpack wall thickness is eight inches on the sides, twelve inches on the front and rear, and four inches on the base. The overpack lid is constructed of a sheet of 1/2-inch thick stainless steel on top, a sheet of 1/4-inch thick cold-rolled steel on the bottom, and 4-inches of polyurethane foam in between. The package is closed by bolting the lid to the body with 40 one-inch diameter bolts.

The GC220 irradiator is positioned inside the cavity formed by the inner stainless steel shell, along with an inner steel frame and a rigid polyurethane foam bonnet and lower crush pad. Shielding is provided by the GC220 irradiator, which is a welded steel lead-filled device. The GC220 is a lead-filled shielding head mounted on a steel stand. The GC220 shielding head consists of inner and outer steel shells with lead in between. The nominal lead thickness is 10 inches. The GC220 has an irregular shape, however, the base is 60-inches long by 40-inches wide. In its shipping configuration, the GC220 is 58-inches high. The GC220 shielding plug is welded from 304 stainless steel and lead filled. The GC220 drawer is welded from 304 stainless steel and is lead filled.

5(a) (2) (continued)

The maximum package weight (including contents) is 21,000 lbs (9,524 kgs). The approximate package component dimensions and weights are as follows:

Component	Weight (lbs / kg)	Nominal Dimensions (L x W x H inches)
Overpack Lid	1,036 / 470	67.50 x 55.00 x 4.75
Inner Frame	1,257 / 570	60.50 x 48.00 x 54.13
Bonnet	871 / 395	52.00 x 41.50 x 36.75
GC220	8,576 / 3,890	60.00 x 40.00 x 58.00
Overpack Body	8,708 / 3,950	86.50 x 66.00 x 80.37
Lower Crush Pad	386 / 175	47.00 x 31.00 x 7.00

(3) Drawings

The packaging is constructed in accordance with MDS Nordion Drawing No. F642301-001, Sheet 1, Revision G, and Sheet 2, Revision D.

(b) Contents

(1) Type and form of material

i. Cobalt-60 as sealed sources that meet the requirements of special form radioactive material.
ii. Cobalt-60 as sealed sources described in Condition No. 6 below.

(2) Maximum quantity of material per package

26,000 curies, a maximum of 48 sources per package, and a maximum of 5,000 curies per source.

6. Sealed sources limited to MDS Nordion sealed source capsules manufactured before February 19, 1973: C-166, C-167, and C-185. In addition, these sources must meet the following:

   (a) Sources must conform to the specifications identified in the application in Figure 4.2 for the C-166 source, Figure 4.3 for the C-167 source, and Figure 4.4 for the C-185 source;

   (b) Sources must be shown to not be leaking within six months prior to shipment; and

   (c) Sources must not have been damaged during their service life.

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

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

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

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

9. Revision No. 1 of this certificate may be used until March 31, 2008.

10. Expiration date: March 31, 2012.

REFERENCES

MDS Nordion application dated November 29, 2006.

Supplement dated: February 8, 2007.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Robert A. Nelson, Chief
Licensing Branch
Division of Spent Fuel Storage
and Transportation
Office of Nuclear Material Safety
and Safeguards

Date 3/15/07

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
9301	4	71-9301	USA/9301/AF-85	1	OF	4

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*
Transnuclear, Inc.
7135 Minstrel Way
Columbia, MD 21045

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION
Packaging Technology, Inc., application dated July 24, 2002, as supplemented.

4. CONDITIONS

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

5.
(a) Packaging

(1) Model No.: TNF-XI

(2) Description

A shipping container for unirradiated enriched forms of homogenous and heterogeneous uranium oxides. The packaging body is a parallelepiped and is approximately 44 inches x 44 inches x 37 inches. The package contents are enclosed in pails which each have a borated stainless steel ring. Three pails are stacked inside four inner wells of the packaging body. Each inner well is closed by a primary lid and an upper plug.

The packaging body is constructed of an outer stainless steel envelope which is 0.08 inches thick. The space between the outer shell and the inner wells is filled with fire-retardant, open cell phenolic foam.

The four inner wells each have an inside diameter of 14 inches and height of 27 inches. The inner wells are constructed of (1) and outer shell of stainless steel sheet 0.04 inches thick, with a diameter of 17 inches, (2) and inner shell of stainless steel sheet 0.04 inches thick with a diameter of 14 inches, and (3) a flat bottom of 0.04 inch thick stainless steel sheet with a 0.08 inch thick borated stainless steel plate glued to it. A molded annular layer of neutron-poison BORA resin is inserted between the inner and outer steel shells of the inner well.

Each upper plug consists of two thermal insulating disks of phenolic foam, with an internal stiffener disk made of aluminum alloy. The upper plug assembly is encapsulated inside a 0.03 inch thick stainless steel envelope.

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

The four primary lids closing off the inner wells are stainless steel circular plates 0.2 inches thick on the center part, and 0.4 inches thick on the periphery. Four bayonet teeth are welded to the primary lid to lock in the well flanges. A primary lid locker is located between the well flange and the primary lid to prevent the rotation of the primary lid during transport. The primary lid and the inner well are sealed by an elastomer gasket set in a rectangular groove machined on the inner face of the primary lid.

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

Inner well inside diameter	14 inches
Overall package dimensions	
Width	44 inches
Length	44 inches
Height	41 inches
Maximum weight of contents in any pail	25 kg
Maximum content weight	300 kg
Maximum package weight (including contents)	1050 kg

(3) Drawings

The packaging is constructed in accordance with the Packaging Technology, Inc., Drawing No. 10799-SAR, Rev. 3, Sheets 1 through 7.

(b) Contents

(1) Type and form of material

Uranium oxide pellets, powder, and scrap meeting the requirements of Enriched Commercial Grade Uranium, as defined in ASTM C996-96. U_3O_8 or $UO_{x,\ x>2}$ are authorized provided that the equivalent UO_2 mass is less than the limits specified below:

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

Max ^{235}U Enrichment (weight percent)	Homogenous UO_2 Powder Maximum Loading (kg)	Heterogeneous UO_2 Pellet Maximum Loading (kg)
≤4.05	300	300
4.1	300	293
4.15	300	287
4.25	300	271
4.35	300	259
4.45	300	247
4.55	294	238
4.65	281	228
4.75	265	219
4.85	255	208
4.95	244	202
5.0	239	197

(2) Maximum quantity of material per package

No more than 25 kg of contents per pail. No more than 300 kg of contents per package.

(c) Criticality Safety Index: 0.5

6. Transport by air is not authorized.

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

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

(b) The package must by acceptance tested and maintained in accordance with the Acceptance Tests and Maintenance Program in Chapter 8 of the application, as supplemented; and,

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9301	4	71-9031	USA/9301/AF-85	4	OF	4

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

8. The packaging authorized by this certificate is hereby approved for use under the general license provision of 10 CFR 71.71.

9. Expiration date: August 31, 2013.

REFERENCES

Packaging Technology, Inc., application dated July 24, 2002.

Supplements dated: October 29, 2002; March 7, April 3, May 6, June 26, July 21, 2003; November 26, 2007; and August 6, 2008.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Kimberly J. Hardin

Eric J. Benner, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: September 12, 2008.

NRC FORM 618 (8-2000) 10 CFR 71	CERTIFICATE OF COMPLIANCE FOR RADIOACTIVE MATERIAL PACKAGES		U.S. NUCLEAR REGULATORY COMMISSION	
a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE / PAGES
9302	2	71-9302	USA/9302/B(U)F-85	1 OF 8

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

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

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Transnuclear Inc., application dated May 2, 2001.

4. CONDITIONS

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

5.

(a) Packaging

(1) Model No.: NUHOMS®-MP197

(2) Description

The NUHOMS®-MP197 package consists of an outer cask, into which a NUHOMS®-61BT transportable dry shielded canister (DSC) is placed. During shipment, energy-absorbing impact limiters are utilized for additional package protection. Additionally, a personnel barrier is mounted to the transportation frame to prevent access to the cask body.

Cask

The NUHOMS®-MP197 transport cask is fabricated primarily of stainless steel. Non-stainless steel members include the cask lead shielding between the containment boundary inner shell and the structural shell, the o-ring seals, the neutron shield, and carbon steel closure bolts. The body of the cask consists of a 1.25 inch thick, 68 inch inside diameter, stainless steel inner (containment) shell and a 2.5 inch thick, 82 inch outside diameter stainless steel structural shell, without impact limiters, which sandwich the 3.25 inch thick cast lead shielding. The overall external dimensions of the cask are 208 inches long and 91.5 inches in outer diameter. The weight of cask body is 148,840 pounds, including about 10,000 pounds of neutron shield and 60,000 pounds of cast lead.

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

The containment system of the NUHOMS®-MP197 transportation cask consists of the inner shell, a 6.50 inch thick bottom plate, 2.5 inch thick RAM access closure with a diameter of approximately 24 inches, a top closure flange, a 4.5 inch thick top closure lid with closure bolts, drain port closures and bolts, and double o-ring seals for each penetration. The containment vessel prevents leakage of radioactive material from the cask cavity. The cask cavity is pressurized to above atmospheric pressure with an inert gas (helium). Helium assists in the heat removal. Shielding is provided by about 4 inches of stainless steel, 3.25 inches of lead, and about 4.5 inches of neutron shielding. Four removable trunnions are provided for handling and lifting of the cask.

Dry Shielded Canister (DSC)

The purpose of the DSC, which is placed within the transport cask, is to permit the transfer of spent fuel assemblies, into or out of a storage module, a dry transfer facility, or a pool as a unit. The DSC also provides additional axial biological shielding during handling and transport. The DSC consists of a stainless steel shell and a basket assembly. The shell has an outside diameter of about 67 inches and an external length of about 200 inches. The DSC basket assembly provides criticality control and contains a storage position for each fuel assembly. No credit is given to the DSC as a containment boundary. The basket is designed to accommodate 61 intact BWR fuel assemblies with or without fuel channels. The basket structure consists of a welded assembly of stainless steel tubes (fuel compartments) separated by poison plates and surrounded by larger stainless steel boxes and support rails.

The poison plates are constructed from borated aluminum, and provide a heat conduction path from the fuel assemblies to the canister wall, as well as the necessary criticality control.

Impact Limiters

The impact limiter shells are fabricated from stainless steel. Within that shell is a laminate of balsa wood and redwood. Each impact limiter is attached to the cask top (front) and bottom (rear) by 12 bolts. The impact limiters are provided with seven fusible plugs that are designed to melt during a fire accident, thereby relieving excessive internal pressure. Each impact limiter has two hoist rings for handling. The hoist rings are threaded into the impact limiter shell. During transportation, the impact limiter hoist rings are removed. An aluminum thermal shield is added to the bottom impact limiter to reduce the impact limiter wood temperature. The weight of the impact limiters, the thermal shield, and attachment bolts, is approximately 28,000 lbs.

(3) Drawings

The package shall be constructed and assembled in accordance with the following Transnuclear Inc., drawing numbers:

1093-71-1, Revision 0,
NUHOMS®-197 Packaging
Transport Configuration

1093-71-2, Revision 1,
NUHOMS®-197 Packaging
General Arrangement

1093-71-3, Revision 1,
NUHOMS®-MP197 Packaging
Parts List

1093-71-4, Revision 1,
NUHOMS®-MP197 Packaging
Cask Body Assembly

1093-71-5, Revision 0,
NUHOMS®-MP197 Packaging
Cask Body Details

1093-71-6, Revision 0,
NUHOMS®-MP 197 Packaging
Cask Body Details

1093-71-7, Revision 0,
NUHOMS®-MP197 Packaging
Lid Assembly & Details

1093-71-8, Revision 0,
NUHOMS®-MP197 Packaging
Impact Limiter Assembly

1093-71-9, Revision 0,
NUHOMS®-MP197 Packaging
Impact Limiter Details

1093-71-10, Revision 0,
NUHOMS®-61BT Transportable
Canister for BWR Fuel Basket
Assembly

1093-71-11, Revision 1,
NUHOMS®-61BT Transportable
Canister for BWR Fuel Basket Details

1093-71-12, Revision 0,
NUHOMS®-61BT Transportable
Canister for BWR Fuel Basket Details

1093-71-13, Revision 1,
NUHOMS®-61BT Transportable
Canister for BWR Fuel General
Assembly

1093-71-14, Revision 1,
NUHOMS®-61BT Transportable
Canister for BWR Fuel General
Assembly

1093-71-15, Revision 2,
NUHOMS®-61BT Transportable
Canister for BWR Fuel Shell Assembly

1093-71-16, Revision 0,
NUHOMS®-61BT Transportable
Canister for BWR Fuel Shell Assembly

1093-71-17, Revision 2,
NUHOMS®-61BT Transportable
Canister for BWR Fuel Canister Details

1093-71-18, Revision 1,
NUHOMS®-61BT Transportable
Canister for BWR Fuel Canister Details

1093-71-20, Revision 0,
NUHOMS®-MP197 Packaging
Regulatory Plate

1093-71-21, Revision 0,
NUHOMS®-MP197 Packaging
on Transport Skids

5. (b) Contents of Packaging

(1) Type and Form of Material

(a) Intact irradiated BWR fuel assemblies, with or without fuel channels, with uranium oxide pellets and zircaloy cladding. Channel thickness is limited to 0.065 to 0.120 inches. Prior to irradiation, the fuel assemblies must meet the dimensions and specifications of Table 1. Assemblies containing fuel rods with no known or suspected cladding defects greater than hairline cracks or pinhole leaks are authorized when contained in the NUHOMS®-61BT DSC.

(b) The maximum burn-up and minimum cooling times for the individual assemblies shall meet the requirements of Table 2. In addition, the fuel shall have been decayed for a time sufficient to meet the thermal criteria of 5(b)(1)(c). The maximum total allowable cask heat load is 15.86 kW.

(c) The maximum assembly decay heat of an individual assembly is 260 watts.

(d) BWR fuel assembly poison material shall meet the design requirements of Table 3.

TABLE 1[1]

Assembly Type	7x7 49/0	8x8 63/1	8x8 62/2	8x8 60/4	8x8 60/1	9x9 74/2	10x10 92/2
Maximum Initial Enrichment (wt% ^{235}U)	See Table 3	See Table 3	See Table 3	See Table 3	See Table 3	See Table 3	See Table 3
Rod Pitch (in)	0.738	0.640	0.640	0.640	0.640	0.566	0.510
Number of Fuel Rods per Assembly	49	63	62	60	60	66-full 8-partial	78-full 14-partial
Fuel Rod OD (in)	0.563	0.493	0.483	0.483	0.483	0.440	0.404
Minimum Cladding Thickness (in)	0.032	0.034	0.032	0.032	0.032	0.028	0.026
Pellet Diameter	0.487	0.416	0.410	0.410	0.411	0.376	0.345
Maximum Active Fuel Length (in)	144	146	150	150	150	146-full 90-partial	150-full 93-partial

[1] Maximum Co-59 content in the Top End Fitting region is 4.5 gm per assembly
Maximum Co-59 content in the Plenum region is 0.9 gm per assembly
Maximum Co-59 content in the In-Core region (including the whole fuel channel) is 4.5 gm per assembly
Maximum Co-59 content in the Bottom region is 4.1 gm per assembly

TABLE 2

Intact BWR Fuel Assembly Characteristics	
Physical Parameters:	
Fuel Design	7x7, 8x8, 9x9, or 10x10 BWR fuel assemblies manufactured by General Electric or equivalent reload fuel
Cladding Material	Zircaloy
Fuel Damage	Cladding damage in excess of pinhole leaks or hairline cracks is not authorized to be stored as "Intact BWR fuel"
Channels	Fuel may be stored with or without fuel channels
Maximum assembly weight	705lbs
Radiological Parameters:	
Group 1:	
Maximum Burnup:	27,000 MWd/MTU
Minimum Cooling Time:	6-Years
Maximum Initial Enrichment:	See Table 3
Minimum Initial Bundle Average Enrichment:	2.0 wt. % U-235
Maximum Initial Uranium Content:	198 kg/assembly
Maximum Decay Heat:	260 W/assembly
Group 2:	
Maximum Burnup:	35,000 MWd/MTU
Minimum Cooling Time:	12-Years
Maximum Initial Enrichment:	See Table 3
Minimum Initial Bundle Average Enrichment:	2.65 wt. % U-235
Maximum Initial Uranium Content:	198 kg/assembly
Maximum Decay Heat:	260 W/assembly

Intact BWR Fuel Assembly Characteristics	
Radiological Parameters:	
Group 3:	
Maximum Burnup:	37,200 MWd/MTU
Minimum Cooling Time:	12-Years
Maximum Initial Enrichment:	See Table 3
Minimum Initial Bundle Average Enrichment:	3.38 wt. % U-235
Maximum Initial Uranium Content:	198 kg/assembly
Maximum Decay Heat:	260 W/assembly
Group 4:	
Maximum Burnup:	40,000 MWd/MTU
Minimum Cooling Time:	15-Years
Maximum Initial Enrichment:	See Table 3
Minimum Initial Bundle Average Enrichment:	3.4 wt. % U-235
Maximum Initial Uranium Content:	198 kg/assembly
Maximum Decay Heat:	260 W/assembly

TABLE 3
Minimum Boron-10 Areal Density as a Function of Maximum Fuel Assembly Lattice Average Enrichment

NUHOMS® -61BT DSC Basket Type	Maximum Fuel Assembly Lattice Average Enrichment(wt.% U-235)	Minimum Boron-10 Areal Density for Boral® (g/cm^2)	Minimum Boron-10 Areal Density for Borated Aluminum, Metamic®, and Boralyn® (g/cm^2)	Areal Density Used in the Criticality Evaluation [75% Credit for Boral®] (g/cm^2)
	Intact Fuel Assemblies			
A	3.7	0.025	0.021	0.019
B	4.1	0.038	0.032	0.029
C	4.4	0.048	0.040	0.036

5. (b) Contents of Packaging (continued)

(2) Maximum quantity of material per package

(a) The quantity of material authorized for transport is 61 intact standard BWR fuel assemblies with or without fuel channels. Where a DSC is to be loaded with fewer fuel assemblies than the DSC capacity, dummy fuel assemblies with the same nominal weight as a standard fuel assembly shall be installed in the unoccupied spaces.

(b) For material described in 5(b)(1) the approximate maximum payload is 21,500 lbs.

(c) Criticality Safety Index "0"

6. Fuel assemblies with missing fuel rods shall not be shipped unless the missing fuel rods are replaced by dummy rods that displace an equal or greater amount of water.

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

(a) Each package shall be both prepared for shipment and operated in accordance with the Operating Procedures in Chapter 7 of the application, as supplemented. In addition this will include:

(1) verification of the basket type A, B, or C, by inspection of the last digit of the serial number on the grapple ring at the bottom of the DSC,

(2) verification that the fuel assemblies to be placed in the DSC meet the maximum burnup, maximum initial enrichment, minimum cooling time, and maximum decay heat limits for fuel assemblies as specified in Tables 2 and 3. The enrichment limit must correspond to the basket type determined in 7(a)(1) above.

(b) All fabrication acceptance tests and maintenance shall be performed in accordance with Acceptance Tests and Maintenance Program in Chapter 8 of the application, as supplemented. In addition this will include replacement of the cask lid bolts after 85, or fewer, round trip shipments to ensure that the allowable fatigue damage factor will not be exceeded during normal conditions of transport.

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

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

10. Revision No. 1 of this certificate may be used until August 31, 2008.

11. Expiration Date: August 31, 2012.

REFERENCES

Transnuclear Inc., Safety Analysis Report for the NUHOMS®-MP197 Transport Packaging, dated May 2, 2001.

Transnuclear Inc., letters dated January 29, 2002, January 31, 2002, March 1, 2002, March 20, 2002, April 29, 2002, May 16, 2002, and June 19, 2007.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Robert A. Nelson, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: August 30, 2007

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE	PAGES
9309	7	71-9309	USA/9309/B(U)F-96	1 OF	8

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

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

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Global Nuclear Fuel - Americas, LLC, application dated March 31, 2004, as supplemented.

4. CONDITIONS

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

5.

(a) Packaging

(1) Model No.: RAJ-II

(2) Description

The RAJ-II package is a rectangular box that is 742 mm (29.21 in) high by 720 mm (28.35 in) wide by 5,068 mm (199.53 in) long to transport a maximum of two Boiling Water Reactor (BWR) fuel assemblies or individual rods that meet the ASTM C996-96 standard of enriched commercial grade uranium, enriched reprocessed uranium, uranium oxide generic pressurized water reactor (PWR) or uranium carbide loose fuel rods in a 5 inch diameter stainless steel pipe.

It is comprised of one inner container and one outer container both made of stainless steel. The inner container is comprised of a double-wall stainless steel sheet structure with alumina silicate thermal insulator filling the gap between the two walls to reduce the flow of the heat into the contents in the event of a fire. Foam polyethylene cushioning material is placed on the inside of the inner container for protection of the fuel assembly. The outer container is comprised of a stainless steel angular framework covered with stainless steel plates. Inner container clamps are installed inside the outer container with a vibro-isolating device between to alleviate vibration occurring during transportation. Wood and honeycomb resin impregnated kraft paper are placed as shock absorbers to reduce shock in the event of a drop of the package. The fuel rod clad and ceramic nature of the fuel pellets provide primary containment of the radioactive material.

5.(a)(2) continued

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

Maximum gross shipping weight	1,614 kg (3,558 lbs)
Maximum weight of inner container	308 kg (679 lbs)
Maximum weight of outer container	622 kg (1,371 lbs)
Maximum weight of packaging	930 kg (2,050 lbs)
Dimensions of inner container	
Length	4,686 mm (184.49 in)
Width	459 mm (18.07 in)
Height	286 mm (11.26 in)
Dimensions of outer container	
Length	5,068 mm (199.53 in)
Width	720 mm (28.35 in)
Height	742 mm (29.21 in)

(3) Drawings

This packaging is constructed in accordance with the Global Nuclear Fuel (GNF) Drawing Nos.:

Outer Container Drawings	Inner Container Drawings	Contents Containers
105E3737, Rev. 6	105E3745, Rev. 8	105E3773, Rev. 1
105E3738, Rev. 7	105E3746, Rev. 1	0028B98, Rev. 1
105E3739, Rev. 4	105E3747, Rev. 4	
105E3740, Rev. 4	105E3748, Rev. 2	
105E3741, Rev. 1	105E3749, Rev. 6	
105E3742, Rev. 3		
105E3743, Rev. 4		
105E3744, Rev. 5		

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9309	7	71-9309	USA/9309/B(U)F-96	3	OF	8

5. continued

(b) Contents

(1) Type and form of material

Enriched commercial grade uranium or enriched reprocessed uranium, as defined in ASTM C996-96, uranium oxide or uranium carbide fuel rods enriched to no more than 5.0 weight percent in the U-235 isotope, with limits specified in Table 1 and Table 2 below.

Table 1: Maximum weight of uranium dioxide pellets per fuel assembly

Type 8x8 fuel assembly	Type 9x9 fuel assembly	Type 10x10 fuel assembly
235 kg	240 kg	275 kg

Table 2: Maximum Authorized Concentrations

Isotope	Maximum content
U-232	2.00×10^{-9} g/gU
U-234	2.00×10^{-3} g/gU
U-235	5.00×10^{-2} g/gU
U-236	2.50×10^{-2} g/gU
Np-237	1.66×10^{-6} g/gU
Pu-238	6.20×10^{-11} g/gU
Pu-239	3.04×10^{-9} g/gU
Pu-240	3.04×10^{-9} g/gU
Gamma Emitters	5.18×10^{5} MeV - Bq/kgU

5.(b)(1) continued

(i) 8 x 8 fuel assemblies comprised of 60 to 64 rods in a square array with a maximum active fuel rod length of 381 cm. The maximum pellet diameter, minimum clad thickness, rod pitch, water rod specifications, and poison rod specification are in accordance with Table 3 below.

(ii) 9 x 9 fuel assemblies comprised of 72 to 81 rods in a square array with a maximum active fuel rod length of 381 cm. The maximum pellet diameter, minimum clad thickness, rod pitch, water rod specifications, and poison rod specification are in accordance with Table 3 below.

(iii) 10 x 10 fuel assemblies comprised of 91 to 100 rods in a square array with a maximum active fuel rod length of 385 cm. The maximum pellet diameter, minimum clad thickness, rod pitch, water rod specifications, and poison rod specification are in accordance with Table 3 below.

(iv) Oxide fuel rods configured loose, in a 5 inch diameter schedule 40 stainless steel pipe/protective case or strapped together. When fuel rods are placed in polyethylene sleeves, each polyethylene sleeve shall not exceed 0.0152 cm in thickness. The maximum pellet diameter, minimum clad thickness, and rod specifications are in accordance with Table 4 below.

(v) Uranium carbide or generic PWR uranium oxide fuel rods configured loose, in a 5 inch diameter schedule 40 stainless steel pipe. When fuel rods are placed in polyethylene sleeves, each polyethylene sleeve shall not exceed 0.0152 cm in thickness. The maximum pellet diameter, minimum clad thickness, and rod specifications are in accordance with Table 4 below.

5.(b)(1) continued

Table 3: Fuel Assembly Parameters

Parameter	**Units**	**Type**	**Type**	**Type**	**Type**
Fuel Assembly Type	Rods	8x8	9x9	FANP 10x10	GNF 10x10
UO_2 Density		≤98% Theoretical	≤98% Theoretical	≤98% Theoretical	≤98% Theoretical
Number of water rods (See Condition 8)	#	0, 2x2	0, 2-2x2 off-center diagonal, 3x3	0, 2-2x2 off-center diagonal, 3x3	0, 2-2x2 off-center diagonal, 3x3
Number of fuel rods	#	60 - 64	72 - 81	91 - 100	91 - 100
Fuel Rod OD	cm	≥1.176	≥1.093	≥1.000	≥1.010
Fuel Pellet OD	cm	≤1.05	≤0.96	≤0.895	≤0.895
Cladding Type		Zirconium Alloy	Zirconium Alloy	Zirconium Alloy	Zirconium Alloy
Cladding ID	cm	≤1.10	≤1.02	≤0.933	≤0.934
Cladding Thickness	cm	≥0.038	≥0.036	≥0.033	≥0.038
Active fuel length	cm	≤381	≤381	≤385	≤385
Fuel Rod Pitch	cm	≤1.692	≤1.51	≤1.350	≤1.350
U-235 Pellet Enrichment	wt%	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0
Maximum Lattice Average Enrichment	wt%	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0
Channel Thickness[a]	cm	0.17 – 0.3048	0.17 – 0.3048	0.17 – 0.3048	0.17 – 0.3048
Partial Length Fuel Rods (1/3 through 2/3 normal length)	Max #	None	12	14	14
Gadolinia Requirements Lattice Average Enrichment[b]	# @ wt% Gd_2O_3				
≤ 5.0 wt % U-235		7 @ 2 wt %	10 @ 2 wt %	12 @ 2 wt %	12 @ 2 wt %
≤ 4.7 wt % U-235		6 @ 2 wt %	8 @ 2 wt %	12 @ 2 wt %	12 @ 2 wt %
≤ 4.6 wt % U-235		6 @ 2 wt %	8 @ 2 wt %	10 @ 2 wt %	10 @ 2 wt %
≤ 4.3 wt % U-235		6 @ 2 wt %	8 @ 2 wt %	9 @ 2 wt %	9 @ 2 wt %
≤ 4.2 wt % U-235		6 @ 2 wt %	6 @ 2 wt %	8 @ 2 wt %	8 @ 2 wt %
≤ 4.1 wt % U-235		4 @ 2 wt %	6 @ 2 wt %	8 @ 2 wt %	8 @ 2 wt %
≤ 3.9 wt % U-235		4 @ 2 wt %	6 @ 2 wt %	6 @ 2 wt %	6 @ 2 wt %
≤ 3.8 wt % U-235		4 @ 2 wt %	4 @ 2 wt %	6 @ 2 wt %	6 @ 2 wt %
≤ 3.7 wt % U-235		2 @ 2 wt %	4 @ 2 wt %	6 @ 2 wt %	6 @ 2 wt %
≤ 3.6 wt % U-235		2 @ 2 wt %	4 @ 2 wt %	4 @ 2 wt %	4 @ 2 wt %
≤ 3.5 wt % U-235		2 @ 2 wt %	2 @ 2 wt %	4 @ 2 wt %	4 @ 2 wt %
≤ 3.3 wt % U-235		2 @ 2 wt %	2 @ 2 wt %	2 @ 2 wt %	2 @ 2 wt %
≤ 3.1 wt % U-235		None	2 @ 2 wt %	2 @ 2 wt %	2 @ 2 wt %
≤ 3.0 wt % U-235		None	None	2 @ 2 wt %	2 @ 2 wt %
≤ 2.9 wt % U-235		None	None	None	None
Polyethylene Equivalent Mass (Maximum per Assembly)[c]	kg	11	11	10.2	10.2

a. Transport with or without channels is acceptable
b. Required gadolinia rods must be distributed symmetrically about the major diagonal
c. Polyethylene equivalent mass calculation (refer to 6.3.2.2 of the application)

5.(b)(1) continued

Table 4: Fuel Rod Parameters

Parameter	Units	Type					
Fuel Assembly Type		8x8 [1] (UO_2)	9x9 [1] (UO_2)	10x10 [1] (UO_2)	CANDU-14 (UC)	CANDU-25 (UC)	Generic PWR (UO_2)
UO_2 or UC Fuel Density		<98% theoretical	<98% theoretical	<98% theoretical	<98% theoretical	<98% theoretical	<98% theoretical
Fuel rod OD	cm	≥1.10	≥1.02	≥1.00	≥1.340	≥0.996	≥1.118
Fuel Pellet OD	cm	≤1.05	≤0.96	≤0.90	≤1.254	≤0.950	≤0.98
Cladding Type		Zirc. Alloy	Zirc. Alloy	Zirc. Alloy	Zirc. Alloy or SS	Zirc. Alloy or SS	Zirc. Alloy or SS
Cladding ID	cm	≤1.10	≤1.02	≤1.00	≤1.267	≤0.951	≤1.004
Cladding Thickness	cm	≥0.038	≥0.036	≥0.038	≥0.033	≥0.033	≥0.033
Active fuel Length	cm.	≤381	≤381	≤385	≤47.752	≤40.013	≤450
Maximum U-235 Pellet Enrichment	wt.%	≤5.0	≤5.0	≤5.0	≤5.0	≤5.0	≤5.0
Maximum Average fuel rod Enrichment	wt.%	≤5.0	≤5.0	≤5.0	≤5.0	≤5.0	≤5.0
Loose Rod Configuration							
Freely Loose		≤25	≤25	≤25	N/A	N/A	N/A
Packed in 5" SS Pipe or Protective Case [3]		≤22	≤26	≤30	≤74 [2]	≤130 [2]	≤105 [2]
Strapped Together		≤25	≤25	≤25	N/A	N/A	N/A

[1] Previous analysis (Ref. 1) based on most conservative loose rod configuration (i.e., no credit taken for 5" SS pipe)
[2] Including partial rods (in reality, apply dense packing of congruent rods in the pipe) and only in 5" SS pipes
[3] Protective case consists of SS box with lid

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

Total weight of payload contents (fuel assemblies, or fuel rods and rod shipping containers) not to exceed 684 kg (1508 pounds).

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

(ii) For the contents described in 5(b)(1)(iv) and 5(b)(1)(v): allowable number of fuel rods per compartment (2 compartments per package).

(c) Criticality Safety Index, except for contents described in 5(b)(1)(v) and limited in 5(b)(2)(ii) 1.0

Criticality Safety Index for contents described in 5(b)(1)(v) and limited in 5(b)(2)(ii) 2.1

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

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

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

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

(d) If wrapping is used on the unirradiated fuel assemblies, the ends must be assured to be open during the shipment in the package.

7. Cluster separators are optional and may be comprised of polyethylene or other plastics. Polyethylene or plastic mass limits shall be determined in accordance with Section 6.3.2.2 (Material Specifications) of the application, as supplemented.

8. Water rods are limited as shown in Table 3 above.

For 8 x 8 fuel assembly designs, there can be either 0 or 1 water rod, and the water rod location occupies a space equivalent to 2 x 2 fuel rods. This is designated as 0, 2 x 2 in the table.

For 9 x 9 and 10 x 10 fuel assembly designs, there can be either 0, 1, or 2 water rods in the assembly, and the water rod location occupies a space equivalent to (a) two 2 x 2 fuel rod equivalent spaces on a diagonal at the center of the assembly, or (b) one 3 x 3 fuel rod equivalent space (9 fuel rods space) in the center of the assembly. These configurations are designated as 0, 2 - 2x2 off-center diagonal, 3x3 in the table.

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

10. Transport by air of fissile material is not authorized.

11. Revision No. 6 of this certificate may be used until May 31, 2009.

12. Expiration date: November 30, 2009.

REFERENCES

Global Nuclear Fuel - Americas, LLC, application dated March 31, 2004.

Supplement dated: April 22, September 3, September 16, October 28, November 8 and 29, 2004; and April 8, May 25, June 6, August 3, 2005; and January 27, 2006; and February 16 and April 21, 2006; and June 12, July 11, November 8, 2007, February 29, March 14, and March 20, 2008.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Meraj Rahimi, Acting Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: May 28, 2008

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE	PAGES
9310	2	71-9310	USA/9310/B(U)-96	1 OF	3

2. PREAMBLE

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

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

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

   a. ISSUED TO *(Name and Address)*

   MDS Nordion
   447 March Road
   Ottawa, ON K2K 1X8
   Canada

   b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

   MDS Nordion application dated May 27, 2003, as supplemented.

4. CONDITIONS

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

5.

(a) Packaging

(1) Model No. F-431 Transport Package

(2) Description

The Model No. F-431 Transport Package is designed to transport Cesium-137 in either special form or RAMCO-50 non-special form sealed sources. The F-431 Transport Package consist of: (1) the overpack which provides impact and thermal protection; (2) either the MDS Nordion Gammacell-1000 irradiator (GC-1000), or the MDS Nordion Gammacell-3000 irradiator (GC-3000) which provides shielding protection; and (3) the radioactive contents in either special form or RAMCO-50 non-special form sealed sources which provides containment.

The F-431 Transport Package is a stainless steel cylindrical package with a 1,067-millimeter (mm) (42-inch (in.)) outside diameter and a height of 1,283 mm (50.5 in.) that is placed on a removable mild steel skid. The maximum weight of the package is 2,270 kilograms (kg) (5000 pounds (lb)).

The overpack consists of nested cylindrical shells. The shells are made from stainless steel and the volume between the shells is filled with rigid foam. This foam provides insulation during an accidental fire. Vent holes, plugged with material designed to melt in a fire, are provided between the shells to prevent pressure buildup and allow a pathway for escape of gases from foam during an accidental fire.

The GC-1000 and the GC-3000 are lead-shielding casks each with a source cavity. The package contents may consists of up to eight cesium-137 special form sealed sources or RAMCO-50 non-special form sealed sources (provided Condition 5.(b)(1)(ii) is met) inside a source holder, within the source cavity. The maximum total activity of cesium-137 is 113

5.(a)(2) continued

tera-Becquerels (TBq)(3,050 Curies (Ci)). The following are the features of the GC-1000 and GC-3000:

Irradiator Model	Rated Capacity	Diameter*	Height*	Lead Thickness*	Steel Shell Thickness*	Weight*
GC-1000	113 TBq (3,050 Ci)	457 mm (18 in.)	610 mm (24 in.)	150 mm (6 in.)	9.5 mm (0.375 in.)	1,035 kg (2,280 lb)
GC-3000	113 TBq (3,050 Ci)	457 mm (18 in.)	610 mm (24 in.)	110 mm (4.3 in.)	9.5 mm (0.375 in.)	1,035 kg (2,280 lb)

* Nominal Values

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

Package outside diameter	1,067mm (42 inches)
Package height	1,283 mm (50.5 inches)
Cavity diameter	559 mm (22 inches)
Cavity height	813 mm (32 inches)
Removable skid	1,118 mm (44 inches) x 1,003 mm (39.5 inches) x 203 mm (8 inches)
Overpack weight	1044 kg (2300 lbs)
Contents weight (max.)	1226 kg (2700 lbs)
Maximum package weight	2,270 kg (5000 lbs)

(3) Drawings

The packaging is constructed in accordance with the MDS Nordion drawing F643101-001, Sheet 1, Revision F and Sheet 2, Revision B.

(b) Contents

(1) Type and form of material

(i) Cesium-137 as a sealed source which meets the requirements of special form radioactive material. The sealed sources consist of the following special form sources: C-378, C-1000, C-1001, C-3000, C-3001, or ISO-1000.

(ii) Cesium-137 as the RAMCO-50 non-special form sealed source, provided the following conditions are met:

- Source must conform to the specifications given in Figure 4.8 of the Safety Analysis Report and sealed source registry Certificate No. NR-0880-S-804-S.
- Source must have been shown to not be leaking within six months prior to shipment.

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

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5.(b) continued

- Source must not have been damaged during its service in the GC-1000.

(2) Maximum quantity of material per package

113 TBq (3,050) Curies.

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

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

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

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

Expiration date: June 30, 2009.

REFERENCES

MDS Nordion application dated May 27, 2003.

Supplements dated: April 16, July 16, July 21, and July 23, 2004.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

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

Date July 27, 2004

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
9314	3	71-9314	USA/9314/B(U)-96	1 OF	6

2. PREAMBLE

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

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

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

   a. ISSUED TO *(Name and Address)*

   QSA Global, Inc.
   40 North Avenue
   Burlington, MA 01803

   b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

   QSA Global, Inc., consolidated application dated December 6, 2005, as supplemented.

4. CONDITIONS

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

5.

(a) Packaging

(1) Model No.: 976 Series

(2) Description

The Model No. 976 Series packages are designed for use as transport packages for Type B quantities of radioactive material in special form. The Model No. 976 has six versions called the 976A, 976B, 976C, 976D, 976E and 976F. The Model 976A package contains a 855 shield container. The Model 976B package contains a 3015 shield container. The Model 976C package contains a 3056 shield container. The Model 976D package contains a 3018 shield container. The Model 976E package contains a 3078 shield container. The Model 976F package contains a 1911 shield container. All versions of the package include a 16 gauge stainless steel 20 gallon drum, four 3/8" - 16 UNC x ¾" long stainless steel lid closure bolts, a clamp band with M8 stainless steel bolt, and cork inserts to position and support the individual shield containers within the package. All Model No. 976 Series packages measure 19 ¾" in diameter by 21 ¼" tall.

The shield containers are described as follows:

855 - An outer carbon steel shell, rigid polyurethane potting material, uranium shield, eight titanium "J" tubes, source stop, top and bottom support plates, and a gasketed lid which is secured with eight 3/8" - 16 UNC x 5/8" long stainless steel hex head bolts. Approximately 11 ¼" in diameter at the base by 11 ¾" tall (without the eyebolt).

5(a) (2) (Description continued)

3015 - A lead shield container surrounded on the sides and partially on the top by an outer stainless steel jacket. The steel jacket incorporates two stainless steel lifting handles. The container includes a lower depleted uranium shielding insert encased in stainless steel, a tungsten capsule holder, an upper lead insert, a lead top shield plug with a stainless steel extension, and a gasketed shield lid which secures to the shield container body by two M10 stainless steel screws and washers. Measures approximately 7 ½" in diameter (including the handle bosses) by 10.1" tall.

3056 - A lead shield container which incorporates stainless steel strapping, handle bosses and lifting handles along with a combination lower depleted uranium insert and upper lead insert with ten stainless steel "J" tubes. The lead insert is partially enclosed by stainless steel. The "J" tubes are covered with tube caps and the tube caps are further covered by a stainless steel "top hat" or lid secured to the container by an M12 steel rod and retaining nut. Measures approximately 7.7" in diameter (including the handle bosses) by 10.4" tall.

3018 - A lead shield container surrounded on the sides and partially on the top by an outer stainless steel jacket. The steel jacket incorporates two stainless steel lifting handles. The container includes a lower depleted uranium shielding insert encased in stainless steel and upper lead insert with four stainless steel "J" tubes. The "J" tubes are covered with tube caps. The shield inserts are secured to the shield body by means of a stainless steel bracket and two M10 stainless steel bolts and washers. The metal bracket also incorporates a stainless steel disk above the "J" tubes which further protects the tube caps during shipment. Measures approximately 7 ½" in diameter (including the handle bosses) by 11" tall.

3078 - A stainless steel encased, depleted uranium shield container which includes two stainless steel lifting handles. The shield container incorporates a stainless steel encased depleted uranium upper shield plug that is inserted into the shield body over an optional stainless steel or aluminum source holder can. The upper shield insert is secured to the shield body by a stainless steel cover bolted above the shield insert by four M8 stainless steel screws. Measures approximately 6.1" in diameter by 8.4" tall.

1911 - A stainless steel encased, lead shield container which includes a bolted shield lid and an M10 stainless steel lifting eyebolt. The shield lid is secured to the shield container body by four stainless steel M8 bolts and washers. The inner shield cavity incorporates either a depleted uranium upper and lower shield insert, a tungsten upper and lower shield insert or a lead upper and lower shield insert. Additional handling source stainless steel, aluminum or tungsten capsule holders or cans may be used in the shield insert cavities. Measures approximately 8" in diameter by 8 ¾" tall (without the eyebolt).

5(a) (2) (Description continued)

The following table gives the maximum package weight.

Model No.	Maximum Package Weight (lbs)
976A	300
976B	190
976C	190
976D	190
976E	226
976F	263

(3) Drawings

This packaging is constructed in accordance with the following AEA Technology or QSA Global Drawings.:

R97608, Rev. E, Sheet 1	20 Gallon Drum Model 976
RCLM009, Rev. B, Sheets 1-2	Clamp, Band
R97637, Rev. A, Sheet 1	Cork Spacer Top Inner
R97623, Rev. B, Sheet 1	Bottom Inner Cork Insert
R97623A, Rev. B, Sheet 1	Bottom Inner Cork Insert
R97615, Rev. C, Sheet 1	Top Outer Cork Insert
R97615-1, Rev. B, Sheet 1	Top Outer Cork Insert
R97615-2, Rev. A, Sheet 1	Bottom Cork Insert
R97616, Rev. B, Sheet 1	Bottom Outer Cork Insert
R976A, Rev. D, Sheet 1	976A Type B Package with 855 Shield Container
R85590, Rev. E, Sheets 1-6	Model 855 Source Changer
R976B, Rev. E, Sheet 1	976B Type B Package with 3015 Shield Container
R3015, Rev. C, Sheets 1-3	3015 Shield Container
R976C, Rev. F, Sheet 1	976C Type B Package with 3056 Shield Container
R3056, Rev. D, Sheets 1-4	Model 3056 Shield Container Top Level Assy
R976D, Rev. E, Sheet 1	976D Type B Package with 3018 Shield Container
R3018, Rev. D, Sheets 1-4	3018 Shield Container
R976E, Rev. E, Sheet 1	976E Type B Package with 3078 Shield Container
R3078, Rev. D, Sheets 1-4	Model 3078 Shield Container Top Level Assembly
R976F, Rev. C, Sheet 1	976F Type B Package with 1911 Shield Container
R1911, Rev. D, Sheets 1-8	Model 1911 Shield

(b) Contents

(1) *Type and form of material*

Iridium-192, Selenium-75, and Ytterbium-169 as sealed sources that meet the requirements of special form radioactive material.

(2) Maximum quantity of material per package

Model No.	Inner Shield	Nuclide	Maximum Capacity[1] (Ci)	Maximum Capacity (TBq)
976A	855	Ir-192	1,000	37
		Se-75	1,000	37
		Yb-169	865	32
976B	3015	Ir-192	350	12.95
		Se-75	350	12.95
		Yb-169	350	12.95
976C	3056	Ir-192	1,250	46.25
		Se-75	1,250	46.25
		Yb-169	1,000	37
976D	3018	Ir-192	500	18.5
		Se-75	500	18.5
		Yb-169	500	18.5
976E	3078	Ir-192	1,000	37
		Se-75	1,000	37
		Yb-169	1,000	37
976F	1911	Ir-192	1,000	37
		Se-75	1,000	37
		Yb-169	1,000	37

[1] Output curies are determined by measuring the source output at 1 meter and *expressing its activity in curies derived from the following: 0.48 R/h-Ci Iridium-*192 at 1 meter. (Ref: American National Standard N432-1980, "Radiological Safety for the Design and Construction of Apparatus for Gamma Radiography.")

6. Tensile and yield strength for the materials of construction must comply with the following values:

Material	Tensile Strength	Yield Strength
Depleted Uranium	65 ksi	30 ksi
Copper	25 ksi	9 ksi
Steel (nominal)	53 ksi	36 ksi
Stainless Steel	75 ksi	30 ksi
Tungsten	142 ksi	109 ksi
Cork (minimum)	80 psi	NA
Lead (^{96}Pb/^{4}Sb)	3,990 psi	NA

7. The sources shall be secured in the shielded positions of the packaging in accordance with the Package Loading requirements contained in Section 7 of the application, as supplemented. For "J" tube style shield containers, the flexible cable of the source assembly and source cap must be of sufficient length and diameter to provide positive positioning of the source in the shielded position.

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

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

   (a) The package shall be prepared for shipment in accordance with the Package Operations in Section 7 of the application, as supplemented, and,

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

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

11. Revision No. 2 of this certificate may be used until June 30, 2008.

12. Expiration date: June 30, 2010.

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

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

QSA Global, Inc., consolidated application dated December 6, 2005.

Supplements dated December 13 and December 15, 2005; October 31, 2006; February 27, and May 31, 2007. |

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Robert A. Nelson, Chief
Licensing Branch
Division of Spent Fuel Transportation and Storage
Office of Nuclear Material Safety
and Safeguards

Date: June 21, 2007

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE	PAGES
9315	8	71-9315	USA/9315/B(U)F-96	1 OF	7

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*
U.S. Department of Energy
Washington, DC 20585

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION
BWXT Y-12, L.L.C., application dated February 25, 2005, as supplemented

4. CONDITIONS

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

5.
(a) Packaging

(1) Model No.: ES-3100

(2) Description

The ES-3100 package is a cylindrical container that is approximately 110 cm (43 in) in overall height and 49 cm (19 in) in overall diameter and is composed of an outer drum assembly and an inner containment vessel. The containment vessel is placed inside the drum and surrounded by a cement based borated neutron absorber, Catalog 277-4. The purpose of the ES-3100 is to transport bulk high enriched uranium in oxide form, uranium metal and alloy, and uranyl nitrate crystals.

The outer drum assembly consists of a reinforced stainless steel, standard mil spec 30-gal drum with an increased length. The volume formed between the drum and the attached inner liner is filled with an inorganic, castable refractory material, Kaolite 1600™, which is comprised of concrete and vermiculite. The Kaolite 1600™ acts as both a thermal insulating and an impact limiting material.

The containment vessel is approximately 82 cm (32 in) in overall height and 13 cm (5 in) in overall diameter and is constructed of 304L stainless steel. The containment boundary consists of the 0.1 in thick containment vessel body and the lid assembly. The lid assembly consists of a sealing lid, a closure nut, and external retaining ring, which holds both the assembly and closure nut together. The double ethylene-propylene elastomer O-rings in the top flange of the containment vessel permit leak testing of the containment vessel. The maximum gross weight of the package, including contents, is 190.5 kg (420 lb).

5.(a) Packaging (continued)

(3) Drawings

The Model No. ES-3100 package is constructed and assembled in accordance with:

(i) BWXT Y-12, L.L.C., Drawing No. M2E801580A037, Sheets 1 through 6, Rev. B, "Consolidated Assembly Drawing."

(ii) BWXT Y-12, L.L.C., Drawing No. M2E801580A026, Rev. C, "Heavy Can Spacer Assembly."

(iii) Equipment Specification JS-YMN3-801580-A001, Rev. E, "ES-3100 Containment Vessel."

(iv) Equipment Specification JS-YMN3-801580-A002, Rev. D, "ES-3100 Drum Assembly."

(v) Equipment Specification JS-YMN3-801580-A003, Rev. C, "Manufacturing Process Specification for Casting Kaolite 1600™ into the ES-3100 Shipping Package."

(vi) Equipment Specification JS-YMN3-801580-A005, Rev. F, "Casting Catalog No. 277-4 Neutron Absorber for the ES-3100 Shipping Package."

5.(b) Contents (Type and form of material, maximum quantity of material per package, and Criticality Safety Index (CSI)).

The weight of the radioactive contents, convenience containers, can lift attachments, polyethylene bags, spacers, and other material in the containment vessel shall not exceed 90 lb. The maximum mass of hydrogenous packaging materials in the containment vessel (e.g., polyethylene containers or bagging, silicone rubber pads, etc.) shall not exceed 500 grams. The maximum content decay heat load shall not exceed 0.4 watts.

5.(b) Contents (continued)

The concentration limits of uranium and transuranic constituents shall be the following:

Isotope	Maximum Concentration
U-232	0.040 μg/gU[a]
U-233	0.006 g/gU[b]
U-234	0.02 g/gU
U-235	1.00 g/gU
U-236	0.40 g/gU
Transuranics (except Np)	40.0 μg/gU
Np-237	0.003 g/gU

[a] μg/gU = 10^{-6} grams per gram of total uranium
[b] g/gU = grams per gram of total uranium

(1) Uranium as solid metal or alloy, packaged in stainless-steel or tin-plated carbon steel convenience cans.

The maximum uranium enrichment is 100 weight percent U-235.

For contents that must be shipped with spacers, the spacers must be in accordance with BWXT Y-12, L.L.C., Drawing No. M2E801580A026 and Equipment Specification JS-YMN3-801580-A005, as specified in Condition No. 5.(a)(3). The quantity of fissile material in any convenience can shall not exceed one third of the mass loading limit per package for that content. Spacers must be positioned between every two convenience cans.

(i) For metal and alloy in the form of solid geometric shapes, meeting the following restrictions, mass limits are listed in Table 1. Contents not meeting the following restrictions must be shipped as broken metal (see Condition No. 5.(b)(1)(ii)).

(A) Spheres having a diameter no larger than 3.24 in (maximum of two spheres per convenience can)

(B) Cylinders having a diameter no larger than 3.24 in (maximum of one cylinder per convenience can)

(C) Square bars having a cross section no larger than 2.29 in × 2.29 in (maximum of one bar per convenience can)

(D) Slugs having dimensions of 1.5 in diameter × 2 in tall (maximum of 10 slugs per convenience can)

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

Table 1: Loading Limits for Metal and Alloy in Solid Geometric Shapes

Solid uranium metal or alloy (specified geometric shapes)	Uranium Enrichment (weight percent U-235)	CSI	With Spacers Maximum Mass U-235 (kg)		No Spacers Maximum Mass U-235 Per Package (kg)
			Per Convenience Can	Per Package	
Spheres	≤ 100	0.0	0.000	0.0	0.0
Cylinders	≤ 100	0.0	6.000	18.000	12.000
Sq. Bars	≤ 100	0.0	10.000	30.000	18.000
Slugs	> 80	0.0	5.447	16.342	Spacer req'd
Slugs	≤ 80	0.0	8.738	26.213	Spacer req'd

(ii) For metal and alloy defined as broken metal, mass limits are specified in Table 2. Uranium metal and alloy pieces must have a surface-area-to-mass ratio of not greater than 1.00 cm^2/g or must not pass freely through a 3/8-inch (0.0095m) mesh sieve. The uranium metal must also have had no more than a limited contact with water and been subsequently dried. Particles and small shapes that do not pass this size restriction, as well as powders, foils, turnings, and wires, are not permitted, unless they are in a sealed container under an inert cover gas. Uranium material or alloy which has been stored in water or is visibly wet at the time of packaging is not authorized to be shipped in this package.

Table 2: Loading Limits for Solid Metal or Alloy in the Form Defined as Broken Metal

Uranium Enrichment (weight percent U-235)	CSI	With Spacers Maximum Mass U-235 (kg)[a]		No Spacers Maximum Mass U-235 Per Package (kg)[a]
		Per Convenience Can	Per Package	
> 95 and ≤ 100	0.0	0.925	2.774	Spacer req'd
	0.4	1.849	5.548	Spacer req'd
	0.8	2.774	8.323	Spacer req'd
	2.0	3.699	11.097	Spacer req'd

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

Table 2: Loading Limits for Solid Metal or Alloy in the Form Defined as Broken Metal (Continued)

Uranium Enrichment (weight percent U-235)	CSI	With Spacers Maximum Mass U-235 (kg)[a]		No Spacers Maximum Mass U-235 Per Package (kg)[a] Spacer req'd
		Per Convenience Can	Per Package	
> 90 and ≤ 95	0.0	0.879	2.637	Spacer req'd
	0.4	1.758	5.274	Spacer req'd
	0.8	3.516	10.549	Spacer req'd
	2.0	5.568	16.703	Spacer req'd
> 80 and ≤ 90	0.0	0.833	2.500	Spacer req'd
	0.4	2.500	7.500	Spacer req'd
	0.8	3.333	10.000	Spacer req'd
	2.0	5.278	15.834	Spacer req'd
> 70 and ≤ 80	00	0.742	2.225	Spacer req'd
	0.4	2.967	8.900	Spacer req'd
	0.8	0.000	0.0	Spacer req'd
	2.0	7.911	23.734	Spacer req'd
> 60 and ≤ 70	0.0	0.000	0.0	1.949
	0.4	4.115	12.346	0.0
	0.8	6.931	20.793	0.0
	2.0	8.231	24.692	0.0
60	0.0	3.718 kgU	11.153 kgU	5.576 kgU
	0.4	0.0 kgU	0.0 kgU	0.0 kgU
	0.8	11.773 kgU	35.320 kgU	0.0 kgU
	2.0	11.773 kgU	35.320 kgU	35.320 kgU

[a] All limits are expressed in kg U-235 unless specified as kgU, which means kilograms of total uranium.

5.(b) Contents (continued)

(2) Uranium as oxide, which may include UO_2, UO_3, and U_3O_8, packaged in stainless-steel, tin-plated carbon steel, or nickel-alloy convenience cans, or polyethylene bottles. The physical form of all contents is dense, loose powder which may contain clumps and pellets. Moisture content in oxide is limited to 3 weight percent water. Carbide compounds are not authorized. The mass limit shall be 24.0 kg of oxide, with a maximum mass of 21.124 kg U-235, with a CSI of 0.0. The maximum uranium enrichment is 100 weight percent U-235. No spacers are required in the containment vessel.

(3) Solid uranyl nitrate in the form of uranyl nitrate crystals, [$UO_2(NO_3)_2 \cdot xH_2O$, where x is ≤ 6]. Uranyl nitrate crystals must be contained in a non-metallic convenience container (such as polyethylene bottles). The mass limit shall be 0.0 kg of uranyl nitrate crystals, with a maximum mass of 0.0 kg U-235, with a CSI of 0.0. The maximum uranium enrichment is 100 weight percent U-235. No spacers are required in the containment vessel.

(4) Unirradiated TRIGA fuel elements and pellets (sections). The fuel is composed of uranium zirconium hydride (UZrH). The uranium concentration in the fuel is a nominal 8.5 weight percent, and the maximum H to Zr ratio in the fuel is 2.0. The maximum uranium enrichment is 70 weight percent U-235. The fuel sections may be from any of three types of fuel elements: standard fuel elements, instrumented standard fuel elements, and fuel follower control rods. The U-235 mass for standard and instrumented fuel elements is a nominal 136 grams per element, and the U-235 mass for fuel follower control rods is a nominal 112 grams per element. Each fuel element contains three fuel sections, either stainless steel or aluminum clad or unclad. The fuel elements are approximately 15 inches in length, with sections approximately 5 inches in length; the approximate diameter of the fuel is 1.44 inches for the standard and instrumented fuel elements, and 1.31 inches for the fuel follower control rods. The fuel elements and sections are packaged within stainless steel or tin-plated carbon steel convenience cans. Disassembled fuel elements are to be packaged with a maximum of three fuel sections, or three fuel elements, per convenience can. Fuel sections from different fuel elements may not be mixed within a single convenience can. A maximum of three convenience cans with disassembled fuel elements may be loaded into a single package. Three stainless steel or aluminum clad elements with crimped ends are to be packaged in a single convenience can with a maximum of one can per package. No spacers are required. The maximum quantity of fissile material per package is 408 grams U-235. The CSI is 0.0.

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

7. Content forms may not be mixed in a single ES-3100 containment vessel.

8. *Any combination of convenience can sizes is allowed in a single package, as long as the total height of the can stack (including silicone rubber pads and spacers, if required) does not exceed the inside working height of the containment vessel (31 in). Any closure on the convenience can is allowed.*

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

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9. Empty convenience cans, spacers, silicone rubber pads, and/or stainless-steel scrubbers (i.e., stainless steel trimmings that act as dunnage) may be used to fill the void space in the containment vessel. Empty convenience cans must have a minimum 0.125 in diameter hole through the lid.

10. The contents and the convenience cans may be bagged or wrapped in polyethylene for contamination control provided the limits of Condition No. 5.(b) are met.

11. Transport by air is not authorized, except for shipment of unirradiated TRIGA fuel pellets, as described and limited in Condition No. 5(b)(4).

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

    (a) The package shall be prepared for shipment and operated in accordance with the Package Operations in Section 7 of the application, as supplemented.

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

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

14. Revision 7 of this certificate may be used until July 31, 2009.

15. Expiration date: April 30, 2011.

REFERENCES

BWXT Y-12, L.L.C., application dated February 25, 2005, as supplemented.

BWXT Y-12, L.L.C., supplements dated April 27, May 26, August 15, 2005; and January 9, February 6, March 20, May 8, June 6, July 18, August 21 and 24, and October 26, 2006; and January 19, January 31, February 22, April 11, April 26, May 30, June 27 (2 supplements on this date), August 8, 28, 30, October 11, and November 5, 2007; and March 25 and May 30, 2008.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Eric J. Benner, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: July 7, 2008

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9319	1	71-9319	USA/9319/B(U)F-96	1	OF	6

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

AREVA NP, Inc.
1724 Mt. Athos Rd
Lynchburg, VA 24504

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

AREVA NP, Inc., application dated March 13, 2007, as supplemented.

UNITED STATES NUCLEAR REGULATORY COMMISSION

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.: MAP-12 and MAP-13

(2) Description

The MAP package is designed to transport unirradiated uranium fuel assemblies with enrichment up to 5.0 weight percent. The package is designed to carry two fuel assemblies with core components. The package consists of two components, a base and lid. The containment system of the MAP package is the fuel rod cladding.

The base consists of a fixed stainless steel strong-back which supports the fuel assemblies. A series of inner stiffeners are secured to the underside of the strong-back to support the fuel assemblies. A neutron moderator and absorber are positioned directly beneath the strong-back between each inner stiffener. The base inner stiffeners are retained by a stainless steel cover. Exterior to the cover is a layer of rigid polyurethane foam and an outer shell of 11 gauge stainless steel. A 12-gauge stainless steel sheet is provided between the two middle stiffeners. Four stainless steel outer stiffeners support the package base. The payload rests on the "W" shaped strong-back (referred to as a W-plate) and is held in place with hinged and latched aluminum doors. The lid is very similar to that of the base – a "W" shaped stainless steel inner shell is fitted with a series of inner stiffeners, neutron moderator and absorbers, and a stainless steel cover is fitted over the stiffeners. The lid is fitted with trapezoidal impact limiters at each end. The impact limiters are constructed from rigid polyurethane foam encased by the package outer stainless shell skin. The base and lid include end plates with interlocking, interfacing angles.

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

There are two models of the MAP package, the MAP-12 and MAP-13. The weights and dimensions of the package are as follows:

<u>MAP-12 (for 144-in Maximum Nominal Active Fuel Length)</u>:

Maximum Gross Weight	8,630 lbs
Maximum Payload Weight	3,400 lbs
Outer Dimensions	
Length	208 in
Width	45 in
Height	31 in

<u>MAP-13 (for 150-in Maximum Nominal Active Fuel Length)</u>:

Maximum Gross Weight	8,630 lbs
Maximum Payload Weight	3,400 lbs
Outer Dimensions	
Length	221 in
Width	45 in
Height	31 in

(3) Drawings

The MAP-12 and MAP-13 packages are fabricated and assembled in accordance with the following AREVA NP, Inc., Drawing Nos.: 9045393, Rev. 2; 9045397, Rev. 0; 9045399, Rev. 0; 9045401, Rev. 0; 9045402, Rev. 0; 9045403, Rev. 0; 9045404, Rev. 0; 9045405, Rev. 0.

(b) Contents

(1) Type and Form of Material

Enriched commercial grade uranium or enriched reprocessed uranium, as defined in ASTM C996-04, oxide fuel rods enriched to no more than 5.0 weight percent in the U-235 isotope, with limits specified in Table 1 below.

5.(b) Contents (continued)

(2) Maximum Quantity of Material per Package

Table 1: Maximum Authorized Concentrations

Isotope	**Maximum Content**
U-232	2.00×10^{-9} g/g U
U-234	2.00×10^{-3} g/g U
U-235	5.00×10^{-2} g/g U
U-236	2.50×10^{-2} g/g U
U-238	Balance of Uranium
Np-237	1.66×10^{-6} g/g U
Pu-238	6.20×10^{-11} g/g U
Pu-239	[illegible].04 $\times 10^{-9}$ g/g U
Pu-240	3.04×10^{-9} g/g U
Gamma Emitters	5.18×10^{5} MeV-Bq/kg U

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5.(b) Contents (continued)

(3) Fuel Assembly

(i) The parameters of the fuel assemblies that are permitted are specified in the table below.

Fuel Rod Array	14x14		15x15					16x16	17x17	
Assembly Type	1	2	1			2	3	1	1	2
No. of Fuel Rods	176	179	208			216	204	236	264	264
No. of Non-Fuel Cells	20	17	[illegible]			9	21	20	25	25
Nominal Fuel Rod Pitch (in)	0.580	[illegible]	0.568			[illegible]	0.563	0.506	0.502	0.496
Maximum Pellet Outer Diameter (in)	0.3812	0.3682	0.3622	0.3707	0.3742	0.3617	[illegible]	0.3282	0.3252	0.3232
Minimum Fuel Rod Outer Diameter (in)	[illegible]	[illegible]	0.414	0.428	0.428	[illegible]	0.422	0.380	0.377	0.372
Minimum Clad Wall Thickness (in)	[illegible]	0.0230	[illegible]	[illegible]	[illegible]	[illegible]	0.0230	0.0220	0.0220	0.0205
Minimum Guide Tube Wall Thickness (in)	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	N/A	N/A	N/A	N/A
Minimum Guide Tube Outer Diameter (in)	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	N/A	N/A	N/A
Number of Guide Tubes	N/A	N/A	16	[illegible]	N/A	N/A	N/A	N/A	N/A	N/A
Minimum Instrument Tube Wall Thickness (in)	N/A	N/A	[illegible]	N/A	N/A	N/A	N/A	N/A	N/A	N/A
Minimum Instrument Tube Outer Diameter (in)	N/A	N/A	0.491	N/A	N/A	N/A	N/A	N/A	N/A	N/A
Number of Instrument Tubes	N/A	N/A	1	N/A	N/A	N/A	N/A	N/A	N/A	N/A
Clad/Tube Material Type	Zr Alloy	Zr Alloy	Zr Alloy			Zr Alloy	Zr Alloy	Zr Alloy	Zr Alloy	Zr Alloy
Maximum Active Fuel Length (in)	160	160	160			160	160	160	160	160

5.(b) Contents (continued)

(3) Fuel Assembly (continued)

(ii) Non-fissile base-plate mounted and spider body core components are permitted.

(iii) Fuel rods assembled into the fuel assemblies are those loaded with sintered pellets of uranium oxides and/or with sintered pellets of uranium oxides mixed with various additives (e.g., Chromium, Boron, Gadolinium, and Europium).

(c) Criticality Safety Index: 2.8

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

(a) The package shall be prepared for shipment and operated in accordance with the Package Operations in Section 7 of the application, as supplemented.

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

(c) Each fuel assembly must be unsheathed or must be enclosed in an unsealed, polyethylene or polypropylene 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.

(d) The fuel rods must be leak tested after fabrication to ensure that the leakage rate of the containment boundary is less than 1×10^{-7} ref cc/sec.

7. Transport by air of fissile material is not authorized.

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

9. Expiration date: January 31, 2013.

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
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REFERENCES

AREVA NP, Inc., application dated March 13, 2007.

Supplements dated: October 24, December 6 and 14, 2007; and April 11, October 13 and 31, 2008.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Eric J. Benner, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: November 6, 2008.

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
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9320	1	71-9320	USA/9320/B(U)-96	1	OF	3

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

EnergySolutions Spent Fuel Division
2105 South Bascom Ave., Suite 160
Campbell, CA 95008

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

EnergySolutions Spent Fuel Division application dated June 20, 2008.

4. CONDITIONS

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

(a) Packaging

(1) Model No.: MIDUS

(2) Description

A depleted-uranium shielded package for the transport of medical isotopes. The package has two primary components: (1) an inner cask assembly that provides containment of the radioactive material and radiation shielding, and (2) an overpack that provides impact and thermal protection.

The cask assembly consists of the cask body, closure lid, shield plug, and shield lid. The cask body is a monolithic, machined 2.5-mm thick stainless steel containment vessel, surrounded by approximately 62 mm of depleted uranium gamma shielding, and a 4-mm thick stainless steel outer shell. The containment system closure lid is a 19-mm thick stainless steel plate which is attached to the cask body by 8, M10 X 1.5 X 30 socket head cap screws. The containment system is sealed by two concentric ethylene propylene O-rings, and the lid is equipped with a leak test port. A stainless steel clad depleted uranium shield plug in the cask cavity and a shield lid that is installed over the closure lid provide shielding at the top end of the package. The overpack base and lid are constructed of thin stainless steel shells filled with rigid polyurethane foam. The overpack lid is attached to the base by eight recessed alloy steel bolts. The overpack lid is equipped with four stainless steel lugs for lifting and tie-down, and the overpack base has a bottom flange with four lugs that may also be used for tie-down.

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

The approximate dimensions and weight of the package are:

Overall package outer diameter	520 mm
Overall package height	551 mm
Cask assembly diameter	225 mm
Cask assembly height	347 mm
Cask cavity inner diameter	85 mm
Cask cavity inner height	134 mm
Maximum package weight	330 kg

(3) Drawings

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

TYC01-1601, Sheets 1 and 2, Rev. 0	General Arrangement of Packaging and Contents
TYC01-1602, Sheets 1 through 4, Rev. 1	General Arrangement of Cask Assembly
TYC01-1603, Sheets 1 through 3, Rev. 1	General Arrangement of Overpack Assembly
TYC01-1604, Sheets 1 through 3, Rev. 1	Containment System
TYC01-1605, Sheets 1 and 2, Rev. 0	Closure Devices
TYC01-1606, Sheets 1 through 3, Rev. 0	Gamma Shielding
TYC01-1607, Sheets 1 and 2, Rev. 0	Heat Transfer Features
TYC01-1608, Sheet 1, Rev. 0	Energy Absorbing Features
TYC01-1609, Sheets 1 and 2, Rev. 0	Lifting and Tie-Down Devices

(b) Contents

(1) Type and form of material

Molybdenum-99 with its daughter products as natrium molybdate ($NaNO_3$ 1M / NaOH 0.2M) in liquid form.

The liquid may be contained within product bottles, consisting of stainless steel flasks with stainless steel caps, with or without elastomeric seals. Various stainless steel components may be used as dunnage. The total volume of the payload hardware may not exceed 125 ml (as indicated by a maximum mass of 1.0 kg).

(2) Maximum quantity of material per package

4,400 Ci molybdenum-99. The maximum specific activity is 60 Ci/ml Mo-99. The product volume may vary from 0 to 150 ml.

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

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

   (a) The package shall be prepared for shipment and operated in accordance with the Package Operations in Section 7.0 of the application. Optional polymeric dunnage may be placed in the space between the cask assembly and the overpack.

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

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

8. Revision No. 0 of this certificate may be used until August 31, 2009.

9. Expiration date: May 31, 2012.

REFERENCES

EnergySolutions Spent Fuel Division application dated June 20, 2008.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Eric J. Benner, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: August 28, 2008

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
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2. PREAMBLE

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

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

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)*

AREVA Federal Services LLC
1102 Broadway Plaza, Suite 300
Tacoma, WA 98402-3526

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Packaging Technology, Inc., application dated August 28, 2006, as supplemented.

4. CONDITIONS

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

5.

a. Packaging

(1) Model No.: TN-55

(2) Description

The Model No. TN-55 packaging is designed to transport uranium oxide powder. The packaging consists of a 55-gallon drum within an overpack. The uranium oxide powder is transported inside the drum. The 55-gallon drum is a DOT specification 7A Type A drum with a reinforced closure system and ceramic gasket. The overpack for the drum has an outer surface of nominal 18-gauge galvanized carbon steel, and a reinforced fiberglass liner. The void space between the liner and outer surface is filled with a polyurethane foam. Forklift pockets, made of 12-gauge galvanized carbon steel, are located on the bottom of the overpack. The nominal outside dimensions of the overpack are 32 inch diameter and 51-5/8 inch height. The maximum gross weight of the package is 1,010 pounds.

(3) Drawing

The packaging shall be constructed and assembled in accordance with Areva Drawing No. 60699-SAR, Revision No. 1, sheets 1 through 4.

b. Contents (continued)

(1) Type and Form of Material

Unirradiated uranium oxide powder with a maximum moisture content of 2 percent. The maximum enrichment is 1.2 weight percent u-235. The content may be packaged in plastic packaging or with other plastic material within the 55-gallon drum, provided the total plastic per drum does not exceed 1,307 grams water-hydrogen equivalent or (1,000 grams of polyethylene). Materials with a hydrogen density greater than water are not authorized, with the exception of polyethylene.

(2) Maximum Quantity of Material per Package

Up to 650 pounds (295 kilograms) of unirradiated uranium oxide powder per package.

c. Criticality Safety Index: 1.7

6. Transport by air is not authorized.

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

a. Each package shall be prepared for shipment and operated in accordance with the "Package Operations," in Chapter 7 of the application.

b. Each packaging shall be tested and maintained in accordance with the "Acceptance Tests and Maintenance Program," in Chapter 8 of the application.

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

9. Revision No. 2 of this certificate may be used until December 31, 2008.

10. Expiration date: April 30, 2012.

REFERENCES

Packaging Technology, Inc., application dated August 28, 2006.

Supplement dated: January 31, 2007, and November 26, 2007.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Robert A. Nelson, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: January 1, 2008

UNITED STATES NUCLEAR REGULATORY COMMISSION

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
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2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*
U.S. Department of Energy
Washington, D.C. 20585

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION
U.S. Department of Energy
application dated August 23, 2006,
as supplemented.

4. CONDITIONS

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

5.

a. Packaging

(1) Model No.: S300

(2) Description

The Model No. S300 package is a cylindrical container that is approximately 89 centimeters (35 inches) in overall height and 60 centimeters (23 inches) in overall diameter. The Model No. S300 is comprised of an overpack, pipe component, and a shielding insert. The Model No. S300 is designed to transport a single plutonium-beryllium (PuBe) special form capsule (SFC). The maximum gross weight of the package is 217.7 kilograms (480 lbs).

The overpack design utilizes a standard 55-gallon drum as the outer container. A standard bolted clamping ring secures the drum lid to the drum body. Within the drum body is a rigid polyethylene liner (body and lid). Lid liner and lid are pierced and the drum lid is fitted with a filter vent. Within the liner is cane fiberboard dunnage and a sheet of plywood to provided shock absorption for the pipe component.

The pipe component consists of a stainless steel cylindrical pipe welded to a stainless steel flat cap at the bottom end and a bolted pipe flange at the other end. The pipe component is closed with a stainless steel flat lid attached to the flange with 12 stainless steel bolts. A filter vent is installed in the lid. The flange-to-lid seal is either a butyl or ethylene propylene elastomeric o-ring.

The shielding insert is located within the pipe component. The shielding insert is made from solid high density polyethylene plastic. Within the shielding insert is a cylindrical opening sized to accommodate the SFC.

(3) Drawings

The packaging is constructed in accordance with Areva drawing No. 60999-SAR, sheets 1 through 3, Revision 0, S300 Packaging SAR Drawing.

b. Contents

(1) Type and form material

Plutonium-239 (Pu-239) as PuBe sources meeting the requirements of special form sources and limited to:

(a) The SFC Model II SFC - IAEA Certificate of Competent Authority Special Form Radioactive Materials Certificate Number USA/0696/S–96, Revision 2, issued by the U.S. Department of Transportation (DOT). |

(b) The Model III SFC - IAEA Certificate of Competent Authority Special Form Radioactive Materials Certificate Number USA/0695/S–96, Revision 2, issued by the DOT. |

(2) Maximum quantity of material per package:

One SFC, containing a maximum quantity of Pu-239 as shown below.

Non-Exclusive Use Shipment		Exclusive Use Shipment	
Model II SFC	Model III SFC	Model II SFC	Model III SFC
206 grams Pu-239	160 grams Pu-239	350 grams Pu-239	160 grams Pu-239

c. Criticality Safety Index 0.0

6. Transport by air is not authorized.

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

a. Each package shall be prepared for shipment and operated in accordance with the "Package Operations," in Chapter 7 of the application.

b. Each package shall be tested and maintained in accordance with the "Acceptance Tests and Maintenance Program," in Chapter 8 of the application.

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

9. Expiration date: November 30, 2011.

REFERENCES

U.S. Department of Energy application dated August 23, 2006.

Supplement dated: November 8, 2006, and April 19, 2007.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Robert A. Nelson, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: April 26, 2007

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9330	0	71-9330	USA/9330/AF-96	1	OF	4

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

U.S. Department of Energy
Washington, DC 20585

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

U.S Department of Energy application dated April 1, 2008, as supplemented.

4. CONDITIONS

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

5.
(a) Packaging

(1) Model No.: ATR FFSC

(2) Description

An insulated stainless steel package for the transport of unirradiated research reactor fuel, including intact fuel elements or fuel plates. The packaging consists of: (1) a body; (2) a closure lid; and (3) inner packaging internals. The approximate dimensions and weights of the package are:

Overall package outer width and height	8 inches
Overall package length (including handle)	72 inches
Cavity diameter	5-3/4 inches
Cavity length	68 inches
Packaging weight (without internals)	240 pounds
Maximum package weight (including internals and contents)	290 pounds

The body is composed of two thin-walled, stainless steel shells. The outer shell is a square tube with an 8-inch cross section, a 72-inch length, and a 3/16 inch wall thickness. The inner shell is a round tube with a 6-inch diameter and a 0.120-inch wall thickness. The inner tube is wrapped with ceramic fiber thermal insulation, overlaid with a stainless steel sheet. At the bottom end, the shells are welded to a 0.88-inch thick stainless steel base plate. At the top end (closure end), the shells are welded to a 1.5-inch thick stainless steel flange.

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

The closure is composed of circular stainless steel plates with ceramic fiber insulation. The closure engages the top end flange by way of four bayonets that are rotated and secured by two spring pins. The closure is equipped with a handle, which may be removed during transport. The closure does not have a gasket or seal.

The package internals consist of either a Fuel Handling Enclosure for intact fuel elements, or a Loose Plate Basket Assembly for loose fuel plates.

(3) Drawings

The packaging is constructed and assembled in accordance with the following Areva Federal Services, LLC, or Packaging Technology, Inc., Drawing Nos.:

60501-10, Sheets 1-5, Rev. 2	ATR Fresh Fuel Shipping Container SAR Drawing
60501-20, Rev. 1	Loose Plate Basket Assembly
60501-30, Rev. 1	Fuel Handling Enclosure

(b) Contents

(1) Type and form of material

Unirradiated Mark VII Advanced Test Reactor (ATR) fuel. The fuel material is composed of uranium aluminide (UAl_x). The uranium is enriched to a maximum 94 weight percent U-235; the maximum U-234 content is 1.2 weight percent; and the maximum U-236 content is 0.7 weight percent. The fuel is contained in aluminum-clad fuel plates. The fuel meat thickness is a nominal 0.02 inch, and the fuel meat width ranges from approximately 1.5 inches to 3.44 inches. The active fuel length is approximately 48 inches.

For intact fuel elements: Elements are composed of 19 curved plates fitted within aluminum side plates, and the maximum channel thickness between fuel plates is 0.085 inch. The maximum mass of U-235 per intact fuel element is 1200 grams. The fuel element must be contained within the Fuel Handling Enclosure, as specified in 5.(a)(3).

For loose fuel plates: Loose plates may be flat or curved and may be banded or wire-tied in a bundle, and dunnage plates may be included. The fuel plates must be contained within a Loose Plate Basket Assembly, as specified in 5.(a)(3).

5.(b) Contents (Continued)

(2) Maximum quantity of material per package

The maximum total weight of contents and internals, including dunnage and other secondary packaging, is 50 lbs. Radioactive contents not to exceed a Type A quantity

For intact fuel elements: One fuel element.

For loose fuel plates: A maximum of 600 grams U-235.

(c) Criticality Safety Index (CSI) 4.0

6. Fuel elements and fuel plates may be bagged or wrapped in polyethylene.

7. Air transport of fissile material is not authorized.

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

(a) The package must be loaded and prepared for shipment in accordance with the Package Operations in Section 7 of the application.

(b) The package must be tested and maintained in accordance with the Acceptance Tests and Maintenance Program in Section 8 of the application.

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

10. Expiration date: July 31, 2013.

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9330	0	71-9330	USA/9330/AF-96	4	OF	4

REFERENCES

U.S. Department of Energy application dated April 1, 2008.

Supplement dated: June 24, 2008.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Eric J. Benner, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
And Safeguards

Date: 7/22/08

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9786	6	71-9786	USA/9786/B(U)	1	OF	5

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

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

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

S3G Core Basket Disposal Container Safety Analysis Report for Packaging dated June 1980, as supplemented

4. CONDITIONS

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

5.

(a) Packaging

(1) Model No.: S3G Core Basket Disposal Container Assembly

(2) Description

The package consists of either one irradiated S3G, S1C or S7G core basket packaged in an inner, lead-filled container (S5W Core Basket Removal Container (CBRC)) which is placed inside an outer container (S3G Core Basket Disposal Container (CBDC)). The package weighs approximately 172,000 pounds.

The S3G CBDC is a 4-inch thick steel cylinder, 89 inches in outside diameter, 131 inches long, with an 8-inch thick top end plate and a 5-inch thick bottom end plate. Both end plates are welded to the cylinder with full penetration welds.

The S5W CBRC, which will be disposed of along with the outer S3G CBDC and inner core basket, is basically a cylindrical shaped container comprised of lead shielding sandwiched between two 304 stainless steel shells. The 1-inch thick inner shell is 60 inches O.D. and 107.5 inches long. The outer shell is made up of two geometries, a 72.5-inch O.D., 0.5-inch thick cylindrical shell that measures 66 inches long and joins a truncated conical shell which has a 64-inch O.D. at the small end. The two shells are joined by a full thickness penetration weld and a weld backup strap on the inside shell surface. Full penetration welds are also made on both ends of the shells to the top canning and shield ring.

The S5W CBRC will contain either an S3G, S1C or S7G core basket. The irradiated S3G core basket is an Inconel 600 cylindrical shell. Three, 3-inch thick 304 stainless steel plates are positioned in the core basket prior to removal to provide overhead radiation shielding. The lower plate is 46.2 inches in diameter. The upper plates have the same diameter but contain six extensions that fit inside recessed cutouts within the core basket. The total core basket weight is approximately 9,650 pounds.

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9786	6	71-9786	USA/9786/B(U)	2	OF	5

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

The S1C core basket is a 304 stainless steel cylindrical shell positioned inside a 304 stainless steel thermal shield. The overhead shielding consists of a set of 2-inch thick 304 stainless steel plates attached to the S1C core basket to provide radiation shielding during handling. The core basket weight is approximately 8,523 pounds.

The S7G core basket is an Inconel 600 cylindrical shell. A 304 stainless steel laminated plate (8-inches thick) with lifting attachments is attached to the top of the S7G core basket to provide radiation shielding during handling. The core basket weight is approximately 8,873 pounds.

The package may alternatively consist of S8G irradiated components positioned within an irradiated components discharge rack (ICDR) which is placed in an S3G CBDC. The ICDR is a steel rack approximately 128 inches high and 80 inches in diameter, and is designed to fit inside the S3G CBDC. The ICDR consists of a center cylinder assembly surrounded by 23 storage tubes, a top plate and a cylinder support base. The center cylinder is HY-80 steel, has a 36-inch outer diameter and a 4.5-inch wall thickness, and is 117 inches high. There are 9 storage tubes positioned inside the center cylinder. The total weight of the irradiated components, the ICDR, and the S3G CBDC is approximately 125,000 pounds.

(3) Drawings

The packaging is constructed in accordance with Bettis Drawing No. 1527E40 for the S3G Core Basket Assembly and KAPL Drawing No. 152D7009 for the S1C Core Basket Assembly and KAPL Drawing No. 232B4874 for the S7G Core Basket Assembly and KAPL Drawing No. 978E644 for the S8G Irradiated Components.

(b) Contents

(1) Type and form of material

(i) An irradiated core basket either the S3G, S1C or S7G and S5W CBRC. The shipment may include surface contamination in the form of activated corrosion products and for the S3G core basket approximately 8 gallons of residual water.

(ii) S8G irradiated components within an ICDR. The shipment may include surface contamination in the form of activated corrosion products.

(2) Quantity of material per package

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

One irradiated core basket and S5W CBRC as described in 5(b)(1). Surface contamination not to exceed 20.6 curies for the S3G core basket, 7.45 curies for the S1C core basket or 1.2 curies for the S7G core basket. The activation level of the irradiated S3G core basket is not to exceed 131,000 curies; the irradiated S1C core basket not to exceed 20,000 curies; and the activation level of the irradiated S7G core basket is not to exceed 140,000 curies.

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

Irradiated components, including 141 instrument lines, 18 lower control drive mechanism assemblies, 4 filled sleeves, and 1 instrumentation stalk. Surface contamination not to exceed 65.5 curies. Activation level of the irradiated components not to exceed 2,440 curies.

6. Shipment of an irradiated S3G core basket must be made no earlier than 75 days after reactor shutdown.

7. Shipment of an irradiated S1C core basket must be made no earlier than 60 days after reactor shutdown.

8. Shipment of an irradiated S7G core basket must be made no earlier than 180 days after reactor shutdown.

9. Shipment of S8G irradiated components must be made no earlier than 100 days after reactor shutdown.

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

(a) Each packaging must meet the following Acceptance Tests and Maintenance Program:

<u>S3G Core Basket</u>

Section 8.0 of application dated June 1980

<u>S1C Core Basket</u>

Section 8.0 of application dated August 1983

<u>S7G Core Basket</u>

Section 8.0 of application dated May 1987

<u>S8G Irradiated Components</u>

Section 8.0 of application dated September 1991

(b) The package shall be prepared for shipment and operated in accordance with the following operating procedures:

<u>S3G Core Basket</u>

Section 7.0 of application dated June 1980

<u>S1C Core Basket</u>

Section 7.0 of application dated August 1983

<u>S7G Core Basket</u>

Section 7.0 of application dated May 1987

<u>S8G Irradiated Components</u>

Section 7.0 of application dated September 1991

11. Revision No. 5 of this certificate may be used until April 30, 2007.

12. Expiration date: August 31, 2011.

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9786	6	71-9786	USA/9786/B(U)	5	OF	5

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; G#01-03414 dated January 31, 2001; and G#06-01024 dated March 7, 2006.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

James R. Hall for

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

Date: May 12, 2006

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9788	14	71-9788	USA/9788/B(U)-96	1	OF	4

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

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

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Deactivated S5W Reactor Compartment Safety Analysis Report for packaging, dated July 1981, as supplemented.

4. CONDITIONS

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

5.

(a) Packaging

(1) Model Nos.: S5W Reactor Compartment and SSN 688 Class Reactor Compartment

(2) Description

The package consists of a deactivated and defueled S5W or SSN 688 Class reactor compartment which has been separated from the remainder of the submarine hull and prepared for shipment by sealing all openings and attaching handling fixtures. For each package model, the reactor compartment itself is between two containment bulkheads which are added to the package before shipping. The ship's hull and the containment bulkheads define the package containment boundaries. The containment bulkheads are either installed at the ends of the package or recessed. The strength of all package boundary closures is at least equivalent to the strength of the bulkheads. The deactivated reactor plant remains in place within the reactor compartment during shipment. The plant is defueled and drained except for small inaccessible pockets of liquid, primarily water. Potentially radioactively contaminated components and piping from other locations in the ship may be placed within the package and secured.

5. (a) Packaging (Continued)

(2) Description (Continued)

The S5W Reactor Compartment package is between 35 and 45 feet long and approximately cylindrical with a maximum diameter of approximately 33 feet. The containment bulkheads are made of HS steel. The bulkheads may be installed at the ends of the package or may be recessed. The forward containment bulkhead may include existing ship structure which has been sealed to form a watertight bulkhead. The hull is constructed of HY-80 steel. The maximum weight of the S5W package is 2,160,000 pounds for the 598 and 585 classes and is 2,262,400 pounds for all other classes.

The SSN 688 Class Reactor Compartment package is approximately 46 feet long and approximately cylindrical with a maximum diameter of approximately 33 feet. The containment bulkheads are made of HS steel. The bulkheads may be installed at the ends of the package or may be recessed. The hull is constructed of HY-80 steel. The maximum weight of the package is 3,360,000 pounds.

(3) Drawings

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

(b) Contents

Activated structural components associated with the S5W and SSN 688 Class reactor vessel complex, plant piping, ion exchanger resin, purification filter media (SSN 688 Class only), residual liquid and other miscellaneous components and materials contaminated with radioactive corrosion products (crud).

6. Residual liquids contained within plant systems must be removed prior to transport to the maximum extent practical, in accordance with established procedures, methods, and controls, as described in submittal dated April 5, 1996, or in the Safety Analysis Report for the individual submarine class reactor compartment packages. Not more than 660 gallons of residual liquids remain in the S5W Reactor Compartment package, and not more than 1,200 gallons of residual liquids remain in the SSN 688 Class Reactor Compartment package.

7. For packages with recessed containment bulkheads, the aft containment bulkheads and stiffeners, horizontal divider plate, and any structure between the pressure hull and the outer non-pressure hull must be recessed at least 7 inches from the aft end of the S5W package. The forward containment bulkhead and stiffeners, existing tank stiffeners, deck structure, and horizontal girders must be recessed at least 15 inches from the forward end of the S5W package. For SSN 688 Class packages with recessed containment bulkheads, both the aft and forward containment bulkheads, stiffeners and horizontal girders must be recessed at least 15 inches from the end of the package.

8. The Lowest Service Temperature (LST) must be determined for each package. The package shall not be shipped unless its LST is less than or equal to the normal daily minimum temperature expected during the shipment of the package as determined on the basis of weather forecasts.

9. Ion exchanger resin with up to 3.1 curies (1.1×10^{11} becquerels) of Co-60 may be shipped in the S5W package. Shipment of the S5W packages shall not occur before 180 days after final reactor shutdown.

10. For SSN 688 Class packages, the Co-60 curie content of the ion exchanger resin and purification filter media shall be limited as described in Table 4-2 of the supplement dated March 15, 2002. For packages that have supplemental shielding as described in Appendix 2.10.14 of the supplement dated March 15, 2002, alternative radioactivity limits for the ion exchanger resin and purification filter media may apply, as specified in Table 2.10.14-1 of Appendix 2.10.14.

11. Shipment of the SSN 688 Class packages shall not occur before 365 days after final reactor shutdown.

12. Additional shielding may be provided on the exterior of the package by steel plates securely welded to the package surface so as to remain in place under the hypothetical accident conditions in 10 CFR Part 71.

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

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

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

14. The hydrogen concentration within the package must be less than 5 percent by volume during the shipment period, as demonstrated by test or analysis.

15. Expiration date: September 30, 2013.

REFERENCES

Deactivated S5W Reactor Compartment Safety Analysis Report for Packaging, WAPD-REO(C)-250, dated July 1981.

Supplements: Naval Reactors Memoranda Nos. Z#C90-14416 dated March 29, 1990, and supplement dated July 6, 1990; Z#C90-14456 dated August 30, 1990; Z#C92-14438 dated August 3, 1992; Z#C93-00069 dated October 14, 1993; Z#C95-00113 dated March 16, 1995; Z#96-14430 dated April 5, 1996; Z#96-14434 dated April 10, 1996; Z#C95-00191 dated December 14, 1995; Z#96-14457 dated June 20, 1996; Z#C96-14520 dated November 22, 1996; Z#C96-14549 dated December 19, 1996; Z#C97-14698 dated October 31, 1997; Z#C98-00021 dated February 27, 1998; Z#C02-03057 dated March 15, 2002; Z#C07-02023 dated September 19, 2007; and Z#08-03540 dated September 11, 2008.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Kimberly J. Hardin for

Eric J. Benner, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: September 16, 2008

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
9791	7	71-9791	USA/9791/B(U)-85	1 OF	3

2. PREAMBLE

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

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

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

   a. ISSUED TO *(Name and Address)*
   U.S. Department of Energy
   Division of Naval Reactors
   Washington, DC 20585

   b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION
   PWR-2 Lower Core Barrel Safety Analysis Report for Packaging dated January 1982, as supplemented

4. CONDITIONS

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

5.
(a) Packaging

(1) Model No.: PWR-2 Lower Core Barrel Shipping and Disposal Container

(2) Description

The PWR-2 Lower Core Barrel Shipping and Disposal Container package consists of an inner burial container and a reusable outer container. The inner container is loaded with a D1G prototype pressure vessel assembly. The package weighs approximately 400,000 pounds.

The outer container is a 4-inch thick steel cylinder, 127 inches in outside diameter, 212 inches long, with two 6-inch thick end plates. The bottom end plate is welded to the cylinder with a full penetration weld and the top end plate is bolted with 107, 2-inch diameter fasteners.

The package is equipped with two 2.5-inch thick by 10-inch long circumferential impact limiter rings on the side, two concentric impact limiter rings on the ends, and aluminum honeycomb crush blocks in the top and bottom spaces between the inner and outer containers.

The container is supported horizontally on the railroad car by eight gussets attached to two horizontal plates. Each plate is bolted to the top flange of an I-beam. The bottom flange of the I-beam is bolted to a 300-ton railroad car.

5. (a) Packaging (continued)

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

The D1G prototype pressure vessel assembly has an inner burial container that consists of two cylinders constructed of HY-80 steel connected by a transition ring that is welded to the two cylinders. The maximum outer diameter of the cylinder is approximately 118 inches at the upper flange. The overall length of the inner container is 184.5 inches. The container wall is 3.12 inches in the upper cylinder and 4 inches in the bottom cylinder. The bottom plate varies in thickness from 6 to 2.4 inches and is attached to the container by 12, 4.5-inch thick gussets. The cover plate is approximately 10 inch thick and is attached to the container by a 3.25-inch thick closure weld. The container is axially positioned within the outer container by aluminum honeycomb energy absorbers.

(3) Drawings

The packaging is constructed in accordance with Westinghouse Drawing Nos. 1575E12, 1574E96, and KAPL, Inc., Drawing Nos. 108E6847 and 108E6846.

(b) Contents

(1) Type and form of material

An irradiated D1G prototype pressure vessel assembly, including pressure vessel, core barrel, thermal shields, and two surveillance train assemblies. In addition, the contents may include surface contamination in the form of activated corrosion products and 119 gallons of residual water.

(2) Quantity of material in package

One irradiated D1G prototype pressure vessel assembly. Surface contamination not to exceed 4.61 curies. Displaced material from cutting operations not to exceed 10.6 curies. The irradiated components not to exceed 60,000 curies.

6. The package will be operated in accordance with the procedures described in Chapter 7 of the application and in accordance with Naval Reactors letter G#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#C98-10723 dated February 13, 1998.

7. Revision No. 6 of this certificate may be used until July 31, 2008.

8. Expiration date: July 31, 2012.

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
9791	7	71-9791	USA/9791/B(U)-85	3	OF	3

REFERENCES

PWR-2 Lower Core Barrel Safety Analysis Report for Packaging, WAPD-LP(CES)CS-670 dated January 1982.

Supplements: Naval Reactors letters G#7241 dated December 2, 1982; G#84-452 dated March 28, 1984; G#C92-03331 dated January 29, 1992; G#92-03546 dated June 5, 1992; G#92-03589 dated July 2, 1992; G#97-053513 dated June 11, 1997; G#C97-03596 dated August 28, 1997; G#C98-10723 dated February 13, 1998; G#98-10801 dated May 5, 1998; G#02-0688 dated January 16, 2002; and G#07-00297.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Robert A. Nelson, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: March 28, 2007

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
9792	8	71-9792	USA/9792/B(U)	1 OF	3

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*
U.S. Department of Energy
Division of Naval Reactors
Washington, DC 20585

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION
Department of Energy application dated April 22, 1991, as supplemented.

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

5.

(a) Packaging

(1) Model No.: Model 1 D1G Core Basket-Thermal Shield Shipping and Storage Container

(2) Description

The Model 1 D1G Core Basket-Thermal Shield (CB-TS) Shipping and Storage Container is a right circular cylinder approximately 115 inches in diameter and either 209 inches long (D1G design including impact limiter assembly) or 216 inches long (D2W design including impact limiter assembly). Access for loading is provided by a removable closure head. The container, consisting of the cylindrical side walls and the bottom end, has a three layer construction with a steel inner vessel approximately eight inches thick covered with approximately nine inches of reinforced concrete which is encased by a 3/8-inch thick outer shell. The CB-TS is secured in place inside the container with an 8-inch thick steel preload ring which is bolted to the inner vessel with 72 high strength bolts.

Closure of the containment vessel is provided by the 6-inch thick steel closure head which is fastened to the inner vessel with 72 high strength bolts. A steel closure ring is welded over the bolts and provides containment. A carbon steel inner impact limiter is welded to the top end of the closure ring. A wood outer impact limiter is bolted to the top plate of the container outer shell.

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

For land transport, the shipping container is transported with its axis horizontal and is supported by a shipping skid. For sea transport, the shipping container is transported with its axis vertical and is supported by a shipping frame assembly. The loaded container weighs up to 185 tons.

(3) Drawings

Packagings for which fabrication was begun before March, 1991, are constructed in accordance with the General Electric Company Drawings contained in Appendix 2.10.4 of the application, and packagings for which fabrication was begun after March, 1991, are constructed in accordance with the KAPL Drawings for the redesign configuration in Appendix 2.10.4 of the application.

(b) Contents

One irradiated D1G core basket-thermal shield assembly, and not more than one core's worth of irradiated D1G support assemblies, D1G lower control rod drive mechanisms, and D1G upper support assemblies; surface contamination in the form of activated corrosion products; and not more than 3.5 gallons of residual water.

6. (a) Preloading of the preload plate and the closure head and sealing the container must be done with a temperature at or above +40 °F.

(b) Shipment of containers S/N 0000001 through 0000007 shall be made only when the average daily temperature is expected to be above +40 °F. Shipment of containers S/N 000008 through 000019 and S/N N00020 through N00031 shall be made only when the average daily temperature is expected to be above +10 °F, subject to the following exception: shipment of any container with the closure head identified as 04241-171D6617 P5, SER N00031 (Forging S/N BG-7140) shall be made only when the average daily temperature is expected to be above +30 °F.

(c) The D1G CB-TS Shipment shall be made no earlier than 150 days after shutdown of the reactor.

7. The package shall be prepared for shipment and operated in accordance with the Operating Procedures in Chapter 7.0 of the application, and each packaging shall be tested and maintained in accordance with the Acceptance Tests and Maintenance Program in Chapter 8.0 of the application.

8. For sea transport, the supplemental operating procedures and acceptance tests in Sections 11.0 and 12.0 of the submittal dated April 5, 2002, shall be used.

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9792	8	71-9792	USA/9792/B(U)	3	OF	3

9. Expiration Date: January 31, 2013.

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; G#C96-03576, dated November 1, 1996; G#C02-0751, dated April 5, 2002; and G#07-01492, dated April 17, 2007.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Robert A. Nelson, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: 1/3/09

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
9793	12	71-9793	USA/9793/B(U)F-85	1	OF	7

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

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

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

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

4. CONDITIONS

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

5.

a) Packaging

(1) Model No.: M-140

(2) Description

The M-140 is a stainless steel cask for transporting spent fuel. The cask is a right circular cylinder and is transported in the upright position. The package's approximate dimensions and weights are as follows:

Cavity diameter	70 inches
Cavity height	46 inches
Body outer diameter	98 inches
Body steel wall thickness	14 inches
Package overall outer diameter	126 inches
Package overall height	194 inches
Packaging weight, including standard internals	315,000 pounds
Maximum package weight, including contents	375,000 pounds

The cask body is made from 304 stainless steel forgings. The cask walls are 14-inches thick and the bottom plate is 12-inches thick. The cask body flange provides a seating surface for the closure head and its protective dome. The flange contains 36 wedge assemblies located radially around the inside diameter. Retention of the closure head is achieved by engaging the wedges in a tapered groove in the circumferential edge of the closure head. The cask body has 180 external cooling fins welded to the exterior wall. A support ring is welded to the external cooling fins at a point above the center of gravity. The support ring seats on, and is

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

bolted to, the railcar mounting ring during transport. For sea transport, the center section of the railcar is shipped with the package and serves as the package support structure. The cask bottom is equipped with an energy absorber which is composed of five concentric stainless steel rings varying in thickness and height.

The closure head is made from forged 304 stainless steel and is approximately 13-inches thick and 81.7 inches in diameter. The closure head is equipped with an access port, which is approximately 24 inches in diameter, and is offset from the center of the closure head. The access port plug is a stepped design with a maximum diameter of approximately 31 inches and is attached to the closure head by 24 bolts. The closure head and access port are sealed with double ethylene propylene O-ring seals. Seal test ports are provided for the closure head and access port seals. A stainless steel protective dome is positioned over the closure head and is secured to the cask body flange by 12, 1.38-inch diameter, 38.5-inch long studs installed in a vertical direction and 6, 2.5-inch diameter, 9-inch long shear bolts installed in the radial direction.

The containment system is composed of the cask body, the closure head, and the closure head access port plug. There are seven penetrations in the standard containment system: a closure head, a drain port, a vent port, and an access port in the closure head, a thermocouple penetration, a water inlet penetration, and a water outlet penetration in the cask body. Each penetration is sealed with a plug and a double ethylene propylene O-ring seal and is equipped with a leak test port. For some shipping configurations, two additional penetrations may be present in the closure head: a removable fuel assembly (RFA) access port and another vent penetration.

The spent fuel modules are positioned in an internals assembly. The internals assembly is composed of stacked internal spacer plates which have openings for the spent fuel modules. The internals assembly has a top plate or top plate subassembly which is preloaded by springs against a retaining ring fitted in a groove in the cask cavity wall. The internals assembly may be a standard, Type 1, Type 2, or Type 3 internals assembly.

(3) Drawings

The packaging is constructed and assembled in accordance with the Westinghouse Electric Corporation Drawings in Appendix 1.3.2 of the application. Internals assemblies and fuel modules are constructed and assembled in accordance with drawings in Chapter 1 of the applicable Safety Analysis Reports for Packaging.

5.(b) Contents

(1) Type and form of material

Spent fuel, limited to the following types, including associated activated corrosion products:

(i) S3G-3 spent fuel.

(ii) S8G spent fuel.

(iii) D1G Core 2 spent fuel.

(iv) D2W spent fuel.

(v) A1G spent fuel.

(vi) S6W spent fuel.

(vii) S9G spent fuel.

(2) Maximum quantity of material per package

Total package weight, including spent fuel and internals assembly, not to exceed 375,000 pounds; and

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

S3G-3 spent fuel modules, not to exceed 62,300 Btu/hr decay heat per package.

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

S8G spent fuel, not to exceed 51,609 Btu/hr decay heat per package (prototype spent fuel modules), or 45,713 Btu/hr decay heat per package (shipboard modules).

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

D1G-2 spent fuel with thermal limits as determined either by calculation of the wet hold time using Curve B from Figure 3-5 of the Safety Analysis Report for Packaging or by use of a shielding hold time from 8(b) below, whichever hold time is greater.

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

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

D2W spent fuel modules, not to exceed 63,000 Btu/hr decay heat per package for prototype spent fuel, 53,000 Btu/hr decay heat per package for shipboard Type 3 spent fuel modules, or 45,900 Btu/hr decay heat per package for shipboard Type 5 spent fuel modules.

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

A1G spent fuel with thermal limits as determined either by calculation of the wet hold time using Curve C from Figure 3-5 of the Safety Analysis Report for Packaging or an administrative hold time of 50 days, whichever hold time is greater.

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

S6W spent fuel modules, not to exceed 46,011 Btu/hr decay heat per package for a shipboard core or 47,160 Btu/hr for a prototype core at the time of container draining.

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

S9G spent fuel modules, not to exceed 55,002 BTU/hr decay heat per package at the time of container draining.

(c) *Criticality Safety Index*

Spent fuel module	Criticality Safety Index
S3G-3	100
S8G	100
D1G Core 2	100
D2W	100
A1G	0
S6W	100
S9G	0

6. For S3G-3 spent fuel shipments:

   (a) Authorized fuel loadings, internals assembly, and other loading restrictions are specified in Section 1.2 of the Safety Analysis Report for Packaging.

   (b) Minimum fuel cooling time is 130 days after shutdown.

   (c) Core age must be at least 4,000 logging corrected full power hours.

   (d) Control rod hold-down devices must be installed on cells which have control rods. Module grapple adapters serve as poison shipping rod holddown devices for refueling shipments.

7. For S8G spent fuel shipments:

   (a) Authorized fuel loadings, internals assembly, and other loading restrictions are specified in Section 1.2 of the Safety Analysis Report for Packaging.

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

   (c) Full and partial fuel modules may be shipped in any combination, but all modules must be shipped with control rods.

   (d) Control rod holddown devices must be installed on the cells. Module grapple adapters serve as control rod holddown devices.

8. For D1G Core 2 spent fuel shipments:

   (a) Authorized fuel loadings, internals assembly, and other loading restrictions are specified in Section 1.2 of the Safety Analysis Report for Packaging.

   (b) The minimum cooling time shall be the greater of 90 days for rail transport, 105 days for ship transport, or that calculated from Curve B of Figure 3-5 of the Safety Analysis Report for Packaging.

   (c) Control rod holddown devices must be installed on rodded modules. The universal grapple adapters serve as the control rod holddown devices.

9. For D2W spent fuel shipments:

   (a) Authorized fuel loadings, internals assembly, and other loading restrictions are specified in Section 1.2 of the Safety Analysis Report for Packaging.

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

   (c) Control rod holddown devices must be installed on all rodded modules. The universal grapple adapters serve as the rod holddown devices.

10. For A1G spent fuel shipments:

    (a) Authorized fuel loadings, internals assembly, and other loading restrictions are specified in Section 1.2 of the Safety Analysis Report for Packaging.

    (b) All fuel clusters must be shipped with either control rods or poison shipping rods, with rod holddown devices installed.

    (c) Minimum fuel cooling time shall be the greater of 50 days after shutdown or that calculated using Curve C from Figure 3-5 of the Safety Analysis Report for Packaging.

11. For S6W spent fuel shipments:

    (a) Authorized fuel loadings, internals assembly, and other loading restrictions are specified in Section 1.2 of the Safety Analysis Report for Packaging.

    (b) The minimum fuel cooling time before container draining shall be 300 days after shutdown for a shipboard core or 450 days after shutdown for a prototype core.

    (c) All fuel modules must be shipped with control rods, control rod restraints, and grapple adapters installed. A lower pedestal must be installed in each module holder port.

12. For S9G spent fuel shipments:

    (a) Authorized fuel loadings, internals assembly, and other loading restrictions are specified in Section 1.2 of the Safety Analysis Report for Packaging.

    (b) The minimum fuel cooling time is 100 days.

    (c) All S9G spent fuel modules must have control rods, control rod holddown devices, and grapple adapters installed.

13. The package must contain no more than 6 gallons of residual water, except that shipments of D2W recoverable irradiated fuel may contain up to 11 gallons of residual water.

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

15. Each packaging must meet the Acceptance Tests and Maintenance Program of Chapter 8 of the application, except:

    All containment seals, including the main closure head seal, must be replaced with new seals within the 12-month period prior to each shipment, or earlier if inspection shows any defect.

16. The package must be prepared for transport and operated in accordance with Chapter 7 of the application, except:

    The containment seals, excluding the main closure head seal, must pass a leak test after final closure prior to each shipment. The leak test must have a sensitivity of at least 1×10^{-3} std-cm^3/sec.

17. Prior to first use, and within the 12-month period prior to each shipment, all containment seals, including the main closure head seal, must be leak tested to show a leak rate no greater than 1×10^{-4} std-cm^3/sec. The leak test must have a sensitivity of at least 5×10^{-5} std-cm^3/sec.

18. Revision No. 11 of this certificate may be used until May 31, 2007.

19. Expiration date: October 31, 2011.

REFERENCES

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

Supplements dated: May 28, June 21, and July 17, 1991; February 4 and 7, August 17, and December 2, 1992; October 14, 1994; September 1, and November 16, 1995; May 13, August 7, September 26, and November 26, 1996; February 10, 1997; June 11, 1998; April 11, 2001; March 5 and November 27, 2002; and April 18, 2006.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

James R. Hall for

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

Date: May 12, 2006

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9794	4	71-9794	USA/9794/B(U)-96	1	OF	3

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

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

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Safety Analysis Report for Packaging for CGN Reactor Compartment Disposal, dated July 12, 1994, as supplemented.

4. CONDITIONS

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

5.

(a) Packaging

(1) Model No.: CGN Reactor Compartment Disposal Package

(2) Description

The package consists of one deactivated and defueled CGN 36, 37, 38, 39, or 40 (36-40) Reactor Compartment that has been separated from the remainder of the cruiser hull and prepared for shipment by enclosing the entire reactor compartment within a welded steel container. The package is approximately cylindrical, about 40 feet high and about 32 feet in diameter. The entire package is a sixteen-sided polyhedron with an enlarged base containing support fixtures, which extend approximately 10 feet beyond the diameter of the package and provide lift points for the package. The container is constructed of high strength steel (MIL-S-22698). The reactor compartment decks, inner-bottom tank structure, secondary shield, and primary shield tank provide internal support and are fastened to the container by welding. The reactor compartment components are drained of water, except for small inaccessible pockets. The maximum weight of the CGN 36-40 package is 5,000,000 pounds. Potentially radioactive contaminated components and piping from areas outside the reactor compartment may be secured within the package.

(3) Drawings

The packaging is constructed in accordance with the drawings in Chapter 1 of the application.

5.(b) Contents

(1) Type and form of material

Activated structural components associated with the CGN 36-40 reactor, plant piping, ion exchanger resin, purification filter media (which may be solidified), and other components contaminated with radioactive corrosion products (crud). Residual liquid, primarily water, some of which contains low level radioactivity, may be present in quantities up to 850 gallons in the CGN 36-40 package.

(2) Maximum quantity of material per package

The maximum quantity of radioactive material contents (crud and activation) shall not exceed the quantities specified in Section 1.2.3.1 of the application.

6. (a) The shipment of a CGN 36-37 package shall be no earlier than 639 days after shutdown.

(b) The shipment of a CGN 38-40 package shall be no earlier than 365 days after shutdown.

7. The Lowest Service Temperature (LST) must be determined for each package. The package shall not be shipped unless its LST is less than or equal to the daily minimum temperature expected during shipment of the package, as determined on the basis of weather forecasts.

8. (a) For CGN 36-37 packages, the Co-60 curie content of ion exchanger resin shall be less than 6.8 curies. The Co-60 curie content of purification filter media (which has not been solidified) shall be less than 4.1 curies. The combined Co-60 curie content of ion exchanger resin and unsolidified purification filter media shall be less than 10.6 curies.

(b) For CGN 38-40 packages, the Co-60 curie content of ion exchanger resin shall be less than 6.8 curies. The Co-60 curie content of purification filter media (which has not been solidified) shall be less than 5.3 curies. The combined Co-60 curie content of ion exchanger resin and unsolidified purification filter media shall be less than 9.58 curies.

9. (a) CGN 36-37 reactor vessels shall have been operated for less than 18,683 effective full power hours.

(b) CGN 38-40 reactor vessel shall have been operated for less than 14,300 effective full power hours.

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

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

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

11. Revision No. 3 of this certificate may be used until February 28, 2007.

12. Expiration date: February 28, 2011

NRC FORM 618 (8-2000) 10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9794	4	71-9794	USA/9794/B(U)-96	3	OF	3

REFERENCES

"Safety Analysis Report for Packaging for CGN Reactor Compartment Disposal," dated July 12, 1994.

Supplements Dated: November 10, 1994; July 14, 1995; November 22, 1996; June 16 and July 17, 1998; December 22, 1999; and August 30, 2005.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

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

Date: February 8, 2006

NRC FORM 618 (8-2000) 10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1 a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9795	3	71-9795	USA/9795/B(U)-85	1	OF	3

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

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

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Irradiated Component Disposal Container
Safety Analysis Report for Packaging
dated July 10, 1997, as supplemented.

4. CONDITIONS

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

5.

(a) Packaging

(1) Model No.: Irradiated Component Disposal Container (ICDC)

(2) Description

The Model No. ICDC is stainless steel cask with an impact limiter at the upper end. The cask body is cylindrical in shape with overall dimensions of approximately 134.6 inches long by 122 inches diameter at the container body flange. The cask cavity is approximately 134.6 inches long by 91 inches diameter. The wall of the cask is 304 stainless steel, 10 inches thick at the bottom and 5 inches thick at the top. The bottom of the cask is an 11 inch thick circular steel plate. The cask lid is closed by a full penetration weld. The upper impact limiter is a stainless steel ring attached with 21 studs to the cask body. A centering plate and pedestals, welded to the bottom end plate, are used to position the contents within the package. The maximum weight of the package is 200,000 pounds. The maximum weight of the contents is approximately 36,300 pounds.

(3) Drawings

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

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1 a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
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5. (b) Contents

The contents of the package are cell support housings and other miscellaneous core components from a spent reactor core. The maximum number of these components per package is specified in Section 1.1 of the application. The other contents of the package include potential residual water not greater than 6 gallons, diatomaceous earth desiccant to absorb the residual water and a stainless steel pumpdown lance which may be left in the package. The maximum radioactivity of the contents is 5,600 curies. The total radioactivity is based on transport no earlier than 50 days after core shutdown.

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

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

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

(c) The package may contain no more than 6 gallons of residual water.

(d) The ICDC shall be shipped no earlier than 50 days after core shutdown.

(e) The total number of cluster joint stud remnants loaded into each ICDC must not exceed 25.

(f) The gross weight of the package shall not exceed 200,000 pounds.

7. Expiration date: April 30, 2013.

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
9795	3	71-9795	USA/9795/B(U)-85	3	OF	3

REFERENCES

Irradiated Component Disposal Container Safety Analysis Report for Packaging dated July 10, 1997.

Supplements dated: U.S. Department of Energy, Division of Naval Reactors letters G#C98-11009, dated December 2, 1998; G#99-03507, dated May 3, 1999; G#C02-4083,dated October 23, 2002; and G#07-04227, dated November 5, 2007.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Robert A. Nelson, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: January 3, 2008,

Type of Packaging: BYPROD. NORM. FORM

Model	Package ID #	Expiration Date
CI-20WC-2A	USA/9098/B( )	10/01/2008
MIDUS	USA/9320/B(U)-96	05/31/2012
PAS-1	USA/9184/B(U)	07/31/2009

U.S. Nuclear Regulatory Commission
List of Packages by Package Type
09/30/2008

Type of Packaging: BYPROD. SPEC. FORM

Model	Package ID #	Expiration Date
A-0109	USA/6280/B( )	02/28/2005
C-1	USA/9036/B(U)-96	10/31/2011
EAGLE	USA/9287/B(U)-85	12/31/2009
F-294	USA/9258/B(U)-96	12/31/2013
F-423	USA/9299/B(U)-85	03/31/2012
F-430/GC-40	USA/9290/B(U)-85	02/28/2012
F-431	USA/9310/B(U)-96	06/30/2009
IR-100	USA/9157/B(U)-85	09/30/2009
LCG-25A	USA/4888/B( )	10/01/2008
LCG-25B	USA/4888/B( )	10/01/2008
LCG-25C	USA/4888/B( )	10/01/2008
MW-3000	USA/9030/B( )	10/01/2008
NPI-20WC-6	USA/9102/B( )	10/01/2008
NPI-20WC-6 MKII	USA/9215/B(U)	05/31/2013
OP-100	USA/9185/B(U)-85	12/31/2008
OP-660	USA/9283/B(U)-96	06/30/2013
OPL-660, OP-660	USA/9283/B(U)-96	06/30/2013
SENTINEL-100F	USA/5862/B( )	10/01/2008
SENTINEL-25A	USA/4888/B( )	10/01/2008
SENTINEL-25B	USA/4888/B( )	10/01/2008
SENTINEL-25C	USA/4888/B( )	10/01/2008
SENTINEL-25C3	USA/4888/B( )	10/01/2008
SENTINEL-25D	USA/4888/B( )	10/01/2008
SENTINEL-25E	USA/4888/B( )	10/01/2008

U.S. Nuclear Regulatory Commission
List of Packages by Package Type
09/30/2008

Type of Packaging: BYPROD. SPEC. FORM

Model	Package ID #	Expiration Date
SENTINEL-25F	USA/4888/B( )	10/01/2008
SENTINEL-8	USA/9030/B( )	10/01/2008
SNAP-21	USA/5830/B( )	10/01/2008
SPEC 2-T	USA/9056/B(U)	04/30/2010
SPEC-150	USA/9263/B(U)-96	06/30/2010
SPEC-300	USA/9282/B(U)-96	04/30/2010
URIPS-8A & -8B	USA/6786/B( )	10/01/2008
URIPS-8B	USA/6786/B( )	10/01/2008
1500	USA/5939/B()F	10/01/2008
181375	USA/5796/B(U)	08/31/2007
5979	USA/5979/B( )	10/01/2008
5984	USA/5984/B( )	08/31/2007
650L	USA/9269/B(U)-96	11/30/2010
680-OP	USA/9035/B(U)-96	06/30/2010
702	USA/6613/B(U)-96	06/30/2013
741-OP	USA/9027/B(U)-96	08/31/2011
770	USA/9148/B(U)-85	03/31/2013
865	USA/9187/B(U)-96	12/31/2008
880 SERIES PKG	USA/9296/B(U)-85	03/31/2011
976 SERIES	USA/9314/B(U)-96	06/30/2010

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List of Packages by Package Type
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Type of Packaging: FISSILE URANIUM

Model	Package ID #	Expiration Date
ABB-2901	USA/9274/AF	09/30/2012
ANF-250	USA/9217/AF	06/30/2010
ATR	USA/9099/B(U)F-85	01/31/2009
ATR FFSC	USA/9330/AF-96	07/31/2013
BW-2901	USA/9251/AF	01/31/2013
CHT-OP-TU	USA/9288/B(U)F-96	03/31/2010
DHTF	USA/9203/AF	02/28/2011
D2G POWER UNIT	USA/6441/B( )F	09/30/2008
ES-3100	USA/9315/B(U)F-96	04/30/2011
ESP-30X	USA/9284/B(U)F-85	05/31/2010
FSV-3	USA/6347/AF	10/01/2008
INNER HFIR UN	USA/5797/B(U)F	09/30/2012
LIQUI-RAD	USA/9291/B(U)F-85	10/31/2011
MAP-12, MAP-13	USA/9319/B(U)F-96	01/31/2013
MCC-3 -4 & -5	USA/9239/AF	03/31/2012
MCC-4	USA/9239/AF	03/31/2012
MCC-5	USA/9239/AF	03/31/2012
MFFP	USA/9295/B(U)F-96	06/30/2010
MODEL B	USA/6206/AF	10/01/2008
MODEL 1 S-6213	USA/9186/B(U)F	03/31/2012
MODEL 2 S-6213	USA/9186/B(U)F	03/31/2012
NCI-21PF-1	USA/9234/B(U)F	12/31/2008
NONE SPECIFIED	USA/6406/AF	03/31/2008
NPC	USA/9294/AF-85	11/30/2010

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Type of Packaging: FISSILE URANIUM

Model	Package ID #	Expiration Date
OUTER HFIR UN	USA/5797/B(U)F	09/30/2012
PADUCAH TIGER	USA/6553/AF	10/01/2008
PATRIOT	USA/9292/AF-85	08/31/2010
RA-3	USA/4986/AF	10/01/2008
RAJ-II	USA/9309/B(U)F-96	11/30/2009
SP-1 SP-2 SP-3	USA/9248/AF	02/28/2009
SP-2	USA/9248/AF	02/28/2009
SP-3	USA/9248/AF	02/28/2009
SRP-1	USA/9285/AF-85	10/31/2013
ST	USA/9246/AF	11/30/2011
TN-55	USA/9328/AF-96	04/30/2012
TNF-XI	USA/9301/AF-85	08/31/2013
TRAVELLER STD	USA/9297/AF-96	03/15/2010
TRAVELLER XL	USA/9297/AF-96	03/15/2010
TRIGA-I	USA/9034/AF	12/31/2010
TRIGA-II	USA/9037/AF	12/31/2010
UBE-1	USA/9280/AF-85	05/31/2013
UBE-2	USA/9281/AF-85	08/31/2013
UNC-2600	USA/5086/B(U)F	02/28/2009
UX-30	USA/9196/AF-96	02/28/2011
WE-1	USA/9289/B(U)F-85	02/28/2009
235R001	USA/6386/B(U)F	04/30/2010
5X22	USA/9250/B(U)F-85	03/31/2008
51032-1	USA/6581/AF	10/01/2008
51032-2	USA/9252/AF	10/31/2008

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Type of Packaging: FISSILE URANIUM

Model	Package ID #	Expiration Date
927A1 & 927C1	USA/6078/AF	10/01/2008
927C1	USA/6078/AF	10/01/2008

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List of Packages by Package Type
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Type of Packaging: IRRADIATED FUEL

Model	Package ID #	Expiration Date
BMI-1	USA/5957/B( )F	10/01/2008
CNS 1-13G	USA/9216/B( )F	10/01/2008
FSV-1 UNIT 3	USA/9277/B( )F	10/01/2008
GA-4	USA/9226/B(U)F-85	10/31/2013
GE-100	USA/5926/B( )F	10/01/2008
HI-STAR 100	USA/9261/B(U)F-85	03/31/2009
IF-300	USA/9001/B( )F	10/01/2008
M-140	USA/9793/B(U)F-85	10/31/2011
NAC-LWT	USA/9225/B(U)F-96	02/28/2010
NAC-STC	USA/9235/B(U)F-96	03/31/2009
NAC-1	USA/9183/B( )F	09/30/2004
NLI-1/2	USA/9010/B( )F	10/01/2008
NUHOMS MP187	USA/9255/B(U)F-85	10/31/2008
NUHOMS-MP197	USA/9302/B(U)F-85	08/31/2012
T-2	USA/5607/B( )F	10/01/2008
TN-FSV	USA/9253/B(U)F-85	05/31/2009
TN-68	USA/9293/B(U)F-85	02/28/2011
TN-8	USA/9015/B( )F	10/01/2008
TN-8L	USA/9015/B( )F	10/01/2008
TN-9	USA/9016/B( )F	10/01/2008
TS125	USA/9276/B(U)F-85	10/31/2012
UMS UNIVERSAL	USA/9270/B(U)F-96	10/31/2012
125-B	USA/9200/B(M)F	06/30/2011
2000	USA/9228/B(U)F-96	05/31/2011

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Type of Packaging: PU AIR

Model	Package ID #	Expiration Date
PAT-1	USA/0361/B(U)F-96	03/31/2009

Type of Packaging: PU NORM. FORM

Model	Package ID #	Expiration Date
HALFPACT	USA/9279/B(U)F-85	10/31/2010
NRBK-41	USA/9221/B( )F	10/01/2008
RH-TRU 72-B	USA/9212/B(M)F-85	02/28/2010
TRUPACT-II	USA/9218/B(U)F-85	08/31/2009

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Type of Packaging: PU SPEC. FORM

Model	Package ID #	Expiration Date
NEUTRON SOURCE	USA/5757/B( )F	10/01/2008
S300	USA/9329/AF-96	11/30/2011

Type of Packaging: WASTE, B

Model	Package ID #	Expiration Date
CGN RCDP	USA/9794/B(U)-96	02/28/2011
CNS 1-13C	USA/9081/B( )	10/01/2008
CNS 1-13C II	USA/9152/B( )F	10/01/2008
CNS 10-160B	USA/9204/B(U)-85	10/31/2010
CNS 3-55	USA/5805/B( )	10/01/2008
CNS 8-120B	USA/9168/B(U)	06/30/2010
D1G CB-TS	USA/9792/B(U)	01/31/2013
ICDC	USA/9795/B(U)-85	04/30/2013
N-55	USA/9070/B(U)	01/31/2010
PWR-2 CORE BAR.	USA/9791/B(U)-85	07/31/2012
SSN 688	USA/9788/B(U)-85	09/30/2013
S3G CBDCA	USA/9786/B(U)	08/31/2011
S5W REC. COMPT.	USA/9788/B(U)-85	09/30/2013
TN-RAM	USA/9233/B(U)	04/30/2010
10-142	USA/9208/B( )	10/01/2008
3-82B	USA/6574/B( )	10/01/2008

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10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

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

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