NRC FORM 618
(3-96)
10 CFR 71

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIALS PACKAGES

U.S. NUCLEAR REGULATORY COMMISSION

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

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

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

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION:

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

c. DOCKET NUMBER 71-9255

4. CONDITIONS

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

5.

5.a. Packaging:

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

(2) Description:

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

Cask

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

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

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

Dry Shielded Canisters (DSCs)

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

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

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

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

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

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

- Failed Fuel Dry Shielded Canister (FF-DSC)

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

Impact Limiters

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

(3) Drawings

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

NUH-05-4000NP, Revision 7,
Sheets 1 through 2
MP187 Multi-Purpose Cask
General Arrangement

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

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

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

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

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

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

5.b Contents of Packaging

(1) Type and Form of Material:

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

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

(c) The fuel authorized for shipment in the NUHOMS®-MP187 package is B&W 15X15 uranium oxide PWR fuel assemblies with a maximum initial pellet enrichment of 3.43% by weight of U235, and a total uranium content not to exceed 466 Kg per assembly

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

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

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

5.b Contents of Packaging:

(1) Type and Form of Material Continued:

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

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

(2) Maximum quantity of material per package:

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

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

Table 1

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

5.c. Transport Index for Criticality Control

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

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

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

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

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

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

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

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

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

      (2) procedures requiring that before shipment the licensee shall:

         (a) perform a measured radiation survey to assure compliance with 49 CFR 173.441 and 10 CFR 71.47 and assure that the neutron measurement instruments are calibrated for the energy spectrum of neutrons being emitted from the package;

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

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

            1) before first use of each package,

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

            3) after seal replacement.

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

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

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

(1) With the exception of the weld between the inner shell and top forging, all longitudinal and circumferential inner shell welds, which form the containment boundary of the cask, shall be radiographically inspected (RT) with acceptance standards in accordance with the ASME Code, Section III, Division 1, NB-5320. The weld between the inner shell and top forging shall be verified by RT or ultrasonically inspected (UT). The substitution of UT for the examination of the completed weld may be made provided the examination is performed using detailed written procedures, proven by actual demonstration to the satisfaction of the inspector as capable of detecting and locating defects described in ASME Code, Section III, Division 1 Subsection NB-5000.

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

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

(4) The cask containment boundary shall be pressure tested to 150% of the design pressure per 10 CFR 71.85(b). The minimum test pressure shall be 75 psig.

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

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

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

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

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

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

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

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

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

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

11. Expiration Date: September 10, 2003.

REFERENCES

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

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

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

M. Wayne Hodges
for

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

Date: March 30, 2000