UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

March 3, 1999

Mr. A. Joseph Nardi
License Administrator
Westinghouse Electric Company LLC
P.O. Box 355
Pittsburgh, PA 15230-0355

SUBJECT: MCDEL NO. 6400 PACKAGE

Dear Mr. Nardi:

As requested by your application dated September 28, 1998, as supplemented on February 22, 1999, enclosed is Certificate of Compliance No. 6400, Revision No. 25, for the Model No. 6400 package. This certificate supersedes, in its entirety, Certificate of Compliance No. 6400, Revision No. 24, dated February 17, 1998.

Changes made to the enclosed certificate are indicated by vertical lines in the margin.

Those on the attached list have been registered as users of the package under the general license provisions of 10 CFR §71.12 or 49 CFR §173.471.

Notwithstanding the date the enclosed Certificate of Compliance was signed, the change in Certificate holder becomes effective on the date of the closing of ownership transfer of assets, so long as that transfer occurs not later than 30 days from the date of the letter transmitting this Certificate of Compliance to the former Certificate holder. The Certificate holder shall notify the Director of the Spent Fuel Project Office, by letter or facsimile, not later than 30 days after the date of this letter, of the date the transfer occurred.

The approval constitutes authority to use the package for shipment of radioactive material and for the package to be shipped in accordance with the provisions of 49 CFR §173.471.

Sincerely,

Cass R. Chappell

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

Docket No.: 71-6400

Enclosures: 1. Certificate of Compliance
No. 6400, Rev. No. 25
2. Approval Record

cc w/encl: J. K. O'Steen, Department of Transportation
Registered Users

NRC FORM 618
(3-96)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIALS PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. PACKAGE IDENTIFICATION NUMBER	d. PAGE NUMBER	e. TOTAL NUMBER PAGES
6400	25	USA/6400/B( )F	1	9

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

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

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Westinghouse Electric Corporation application dated August 7, 1981, as supplemented.

71-6400

c. DOCKET NUMBER

4. CONDITIONS

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

5.

(a) Packaging

(1) Model No.: 6400

(2) Description

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

(3) Drawings

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

9903180024 11pp.

5.(b) Contents

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

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

T[illegible] large equipment waste must be enclosed in a tight-fitting, 1-inch thick plywood box [illegible]structed in accordance with Westinghouse Electric Corporation's Drawing No. [illegible]20E43, Sheets 1, 2, 3, and 4, Rev. 3; a tight fitting 3/16" thick corrugated steel box constructed in accordance with Rockwell Hanford Operations' Drawing No. H-2-91888, Sheet 1, Rev. 0 (modified or unmodified); or enclosed in a tight fitting box constructed in accordance with General Electric Company Drawing Nos. 908E614, Rev. 1, and 908E619, Rev. 2 or 908E648, Rev. 0 or 908E649, Rev. 0; or enclosed in a tight fitting box constructed in accordance with Babcock and Wilcox Company Drawing No. LRC-70019 H, Rev. 2. The space between the equipment and the box must be filled with foam (1" minimum foam thickness) and between equipment (1/2" minimum foam thickness). Alternatively, gloveboxes contaminated and fixed as described above may be broken down as follows:

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

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

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

5.(b) Contents (continued)

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

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

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

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

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

5.(b) Contents (continued)

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

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

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

5.(b) Contents (continued)

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

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

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

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

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

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

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

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

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

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

5.(b) Contents (continued)

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

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

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

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

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

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

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

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

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

5.(b) Contents (continued)

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

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

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

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

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

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

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

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

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

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

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

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

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

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

5.(c) Transport Index for Criticality Control

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

6. The polyurethane foam must be Instapak 200, or equivalent.

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

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

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

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

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

12. Contents packaged under the conditions of this certificate of compliance are exempt from the requirements of 10 CFR §71.63. Condition 5(c) of this certificate of compliance is not applicable where the fissile material is excluded as provided by 10 CFR §71.53.

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

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

15. Expiration date: July 31, 2002.

REFERENCES

Westinghouse Electric Corporation application dated August 7, 1981.

General Electric Company supplement dated: October 1, 1981.

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

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

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

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

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

Westinghouse Electric Company, Division of CBS Corporation supplement dated: December 22, 1997, September 28, 1998 and February 22, 1999.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Cass R. Chappell

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

Date: 3/3/99

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

APPROVAL RECORD
Model No. 6400 Package
Certificate of Compliance No. 6400
Revision No. 25

By application dated September 28, 1998, as supplemented on February 22, 1999, CBS Corporation, requested a name change to Certificate of Compliance No. 6400 for the Model No. 6400 Package. The applicant requested that the name be changed from Westinghouse Electric Company, a Division of CBS Corporation, to Westinghouse Electric Company LLC (WELCO). No design changes were requested to the package.

Cass R. Chappell

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

Date: 3/3/99

ATTACHMENT 12

License Number, COC-9239

New Fuel Shipping Container, MCC Series

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

March 3, 1999

Mr. A. Joseph Nardi
License Administrator
Westinghouse Electric Company LLC
P.O. Box 355
Pittsburgh, PA 15230-0355

SUBJECT: MODEL NOS. MCC-3, MCC-4 and MCC-5 PACKAGES

Dear Mr. Nardi:

As requested by your application dated September 28, 1998, as supplemented on February 22, 1999, enclosed is Certificate of Compliance No. 9239, Revision No. 8, for the Model Nos. MCC-3, MCC-4, and MCC-5 packages. This certificate supersedes, in its entirety, Certificate of Compliance No. 9239, Revision No. 7, dated February 22, 1999.

Changes made to the enclosed certificate are indicated by vertical lines in the margin.

Those on the attached list have been registered as users of the package under the general license provisions of 10 CFR §71.12 or 49 CFR §173.471.

Notwithstanding the date the enclosed Certificate of Compliance was signed, the change in Certificate holder becomes effective on the date of the closing of ownership transfer of assets, so long as that transfer occurs not later than 30 days from the date of the letter transmitting this Certificate of Compliance to the former Certificate holder. The Certificate holder shall notify the Director of the Spent Fuel Project Office, by letter or facsimile, not later than 30 days after the date of this letter, of the date the transfer occurred.

This approval constitutes authority to use the package for shipment of radioactive material and for the package to be shipped in accordance with the provisions of 49 CFR §173.471.

Sincerely,

Cass R. Chappell

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

Docket No.: 71-9239

Enclosures:
1. Certificate of Compliance No. 9239, Rev. No. 8
2. Approval Record

cc w/encl: J. K. O'Steen, Department of Transportation
Registered Users

9903180020 1P-

NRC FORM 618
(3-96)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIALS PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. PACKAGE IDENTIFICATION NUMBER	d. PAGE NUMBER	e. TOTAL NUMBER PAGES
9239	8	USA/9239/AF	1	4

2. PREAMBLE

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

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

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

a. ISSUED TO (Name and Address)

Westinghouse Electric Company
LLC (WELCO)
P.O. Box 355
Pittsburgh, PA 15230

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Westinghouse Electric Corporation application dated January 31, 1991, as supplemented.

c. DOCKET NUMBER 71-9239

4. CONDITIONS

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

5.

(a) Packaging

(1) Model Nos.: MCC-3, MCC-4, and MCC-5

(2) Description

The MCC packages are shipping containers for unirradiated uranium oxide fuel assemblies. The packagings consist of a steel fuel element cradle assembly equipped with a strongback and an adjustable fuel element clamping assembly. The cradle assembly is shock mounted to a 13-gauge carbon steel outer container by shear mounts. The MCC-3 container is closed with thirty 1/2-inch T-bolts. The MCC-4 and MCC-5 containers are closed with fifty 1/2-inch T-bolts.

The MCC-3 and MCC-4 containers are permanently equipped with vertical Gd_2O_3 neutron absorber plates that are mounted on the center wall of the strongback. Additional horizontal Gd_2O_3 neutron absorber plates, mounted on the underside of the strongback, are required for the contents as specified.

The MCC-5 container is permanently equipped with both the vertical and horizontal Gd_2O_3 neutron absorber plates. Additional vee-shaped, guided Gd_2O_3 neutron absorber plates are required for the contents as specified.

Approximate dimensions of the MCC-3 packaging are 44-1/2 inches O.D. by 194-1/2 inches long. The gross weight of the packaging and contents is 7,544 pounds. The maximum weight of the contents is 3,300 pounds.

Approximate dimensions of the MCC-4 packaging are 44-1/2 inches O.D. by 226 inches long. The gross weight of the packaging and contents is 10,533 pounds. The maximum weight of the contents is 3,870 pounds.

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5. (a) Packaging (continued)

Approximate dimensions of the MCC-5 packaging are 44-1/2 inches O.D. by 226 inches long. The gross weight of the packaging and contents is 10,533 pounds. The maximum weight of the contents is 3,700 pounds.

(3) Drawings

The MCC-3 packaging is constructed in accordance with Westinghouse Electric Corporation Drawing No. MCCL301, Sheets 1, 2 and 3, Rev. 4.

The MCC-4 packaging is constructed in accordance with Westinghouse Electric Corporation Drawing No. MCCL401, Sheets 1, 2, 3, and 4, Rev. 6.

The MCC-5 packaging is constructed in accordance with Westinghouse Electric Corporation Drawing No. MCCL501, Sheets 1 through 9, Rev. 3.

(b) Contents

(1) Type and form of material

Unirradiated PWR uranium dioxide fuel assemblies with a maximum uranium-235 enrichment of 5.0 weight percent.

The fuel assemblies shall meet the specifications given in Westinghouse Drawing No. 6481E15, Rev. 3, and in the following tables of Appendix 1-4 of the application, as supplemented:

Table 1-4.1, Rev. 6, dated July 26, 1994	Fuel Assembly Parameters 14x14 Type Fuel Assemblies
Table 1-4.2, Rev. 6, dated July 26, 1994	Fuel Assembly Parameters 15x15 Type Fuel Assemblies
Table 1-4.3, Rev. 6, dated July 26, 1994	Fuel Assembly Parameters 16x16 Type Fuel Assemblies*
Table 1-4.4, Rev. 7, dated February 19, 1999	Fuel Assembly Parameters 17x17 Type Fuel Assemblies*
Table 1-4.5, Rev. 4, dated January 14, 1994	Fuel Assembly Parameters VVER-1000 Type Fuel Assembly**

* 16x16 CE fuel assemblies and the 17x17 W-STD/XL fuel assemblies may be shipped only in the Model No. MCC-4 package.

** VVER-1000 fuel assemblies may be shipped only in the Model No. MCC-5 package.

5. (b) Contents (continued)

(2) Maximum quantity of material per package

Two (2) fuel assemblies

(c) Transport Index for Criticality Control

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

6. For shipments of 14x14, 15x15, 16x16, and 17x17 fuel assemblies with U-235 enrichments of over 4.65 wt% and up to 5.0 wt%, horizontal Gd_2O_3 neutron absorber plates shall be positioned underneath each assembly. The horizontal absorber plates shall be placed horizontally on the underside of the strongback, as shown on Westinghouse Electric Corporation Drawing No. MCCL301, Sheet 1, Rev. 4, or Westinghouse Electric Corporation Drawing No. MCCL401, Sheet 1, Rev. 6.

7. For shipments of VVER-1000 fuel assemblies with U-235 enrichments of over 4.80 wt% and up to 5.0 wt%, a guided Gd_2O_3 neutron absorber plate shall be positioned underneath each assembly. The guided absorber plates shall be placed horizontally on the topside of the strongback, as shown on Westinghouse Electric Corporation Drawing No. MCCL501, Sheet 5, Rev. 3.

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

9. The dimensions, minimum Gd_2O_3 loading and coating specifications, and acceptance testing of the neutron absorber plates shall be in accordance with the "Gd_2O_3 Neutron Absorber Plates Specifications," Appendix 1-6, Rev. 2, dated January 14, 1994, of the application. The minimum Gd_2O_3 coating areal density on the vertical and horizontal neutron absorber plates shall be 0.054 g-Gd_2O_3/cm^2. The minimum Gd_2O_3 coating areal density on guided neutron absorber plates shall be 0.027 g-Gd_2O_3/cm^2.

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

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

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

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

Page 4 - Certificate No. 9239 - Revision No. 8 - Docket No. 71-9239

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

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

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

12. Expiration date: March 31, 2002.

REFERENCES

Westinghouse Electric Corporation application dated January 31, 1991.

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

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Cass R. Chappell

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

Date: 3/3/99

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

APPROVAL RECORD
Model Nos. MCC-3, MCC-4, and MCC-5 Packages
Certificate of Compliance No. 9239
Revision No. 8

By application dated September 28, 1998, as supplemented on February 22, 1999, CBS Corporation, requested a name change to Certificate of Compliance No. 9239 for the Model Nos. MCC-3, MCC-4 and MCC-5 Packages. The applicant requested that the name be changed from Westinghouse Electric Company, a Division of CBS Corporation, to Westinghouse Electric Company LLC (WELCO). No design changes were requested to the package.

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

Date: 3/3/99

ATTACHMENT 13

License Number, COC-1001

Irradiated Fuel Storage Cask - MC-10

Docket No. 72-1001 (Pr'iect No. M-41)
Package Identification . . USA/72-1001

Westinghouse Electric Corp.
ATTN: William J. Johnson, Manager
Nuclear Safety Department
P. O. Box 355
Pittsburgh, Pennsylvania 15230-0355

AUG 17 1990

Gentlemen:

Pursuant to Title 10, Code of Federal Regulations, Part 72 (55 FR 29181), we are enclosing is Certificate of Compliance No. 1001, issued on the basis of the safety analysis report of the cask design, Model No. MC-10, identified in "Topical Safety Analysis Report for the Westinghouse MC-10 Cask for an Independent Spent Fuel Storage Installation (Dry Storage) (TSAR)."

This Certificate of Compliance constitutes authorization for a twenty-year term. Casks of the Model No. MC-10 are approved for general use by holders of 10 CFR Part 50 licenses for use at civilian power reactor sites under the general license issued pursuant to §72.210, 10 CFR Part 72, subject to the conditions specified by §72.212 and Conditions for Cask Use.

If you have any questions regarding this issuance of Certificate of Compliance No. 1001, please contact me or John P. Roberts of my staff (301-492-0608).

Sincerely,

Original Signed by

Charles J. Haughney, Chief
Fuel Cycle Safety Branch
Division of Industrial and
Medical Nuclear Safety
Office of Nuclear Material Safety
and Safeguards

Enclosures:
Certificate of Compliance
No. 1001

cc: Mr. W. L. Stewart, VEPCO
Mr. M. Smith, VEPCO

Distribution:
Docket 72-1001 (Project M-41) PDR NRC File Center

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NAME: FBrown	:FSturz	:RFonner	:CHaughney	:	:	:
DATE: 08/10/90	:08/10/90	:08/10/90	:08/10/90	:	:	:

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M-41 PDC

NF09

Certificate of Compliance
FOR DRY SPENT FUEL STORAGE CASKS

10 CFR 72 AUG 17 1990

1. a. **CERTIFICATE NUMBER:** 1001
   b. **REVISION NUMBER:** 0
   c. **PACKAGE IDENTIFICATION NUMBER:** USA/72-1001
   d. **PAGE NUMBER:** 1
   e. **TOTAL NUMBER OF PAGES:** 3

2. **Preamble** This certificate is issued to certify that the cask and contents, described in item 5 below, meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste."

3. ***THIS CERTIFICATE*** is issued on the basis of a safety analysis report of the cask design.

a. PREPARED BY (Name and Address)

Westinghouse Electric Corp.
Power Systems
P.O. Box 355
Pittsburgh, PA 15230-0355 (USA)

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Topical Safety Analysis Report for the Westinghouse MC-10 Cask for an Independent Spent Fuel Storage Installation (Dry Storage) (TSAR)

c. DOCKET NUMBER 72-1001

4. **CONDITIONS** This certificate is conditional upon fulfilling the requirements of 10 CFR 72, as applicable, and the conditions specified below.

5. **Cask**
   a. **Model No:** MC-10
   b. **Description**
   The MC-10 cask is designed for the storage and shipment of irradiated spent fuel assemblies. This certificate addresses spent fuel handling, transfer, and storage on an NRC-licensed nuclear reactor site but does not address any use or certification of this cask design for offsite transport of spent fuel.

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The MC-10 cask consists of a thick-walled forged steel cylinder and weighs approximately 85.2 tonne (94 ton). The cask has a cylindrical cask cavity which holds a fuel basket and is designed to accommodate 24 PWR fuel assemblies. The loaded weight of the cask is about 103 tonne (113.3 ton).

The overall length is 4775 mm (188 in), and the side wall thickness including neutron absorber and without fins is 333.8 mm(13.1 in). The cross-sectional diameter of the cask including neutron absorber is 2394 mm (94.3 in). The overall diameter including fins is 2725 mm (107.28 in). The cask cavity has a diameter of 1727 mm (68 in) and a length of 4130 mm (162.6 in). The cask body is low alloy steel approximately 2235 mm (88 in) in diameter and 4699 mm (185 in) long. The forged steel walls and bottom are approximately 254 mm (10 in) thick to provide radiation (gamma) shielding and structural integrity. Three covers seal the top end of the cask cylinder. A low alloy steel cover, approximately 127 mm (5 in) thick with metallic O-rings provides initial seal and shielding following fuel loading. A carbon steel cover approximately 89 mm (3.5 in) thick with a metallic O-ring provides the primary seal.

The cask contains a basket assembly which consists of 24 storage locations utilizing a honeycomb-type basket structure. The stainless steel basket structure maintains the subcritical array of storage locations, provides lateral structural integrity, and conducts fuel assembly decay heat to the cask wall.

Each of the 24 removable cell storage locations consists of an enclosure, neutron poison material, and wrappers. The enclosure is a stainless steel sheet, 2 mm (.75 in) thick by 890 mm (35.06 in) basket structure. The upper ends of the enclosure walls are flared to facilitate fuel loading. Neutron absorbing material is attached to the enclosure walls and held in place with a stainless steel wrapper welded to the panel.

c. Drawing

The Model No. MC-10 dry spent fuel storage cask is described by drawings in Figures 4.2-1 thru 4.2-10 of the TSAR.

d. Basic Components

The Basic Components of the Model No. MC-10 storage cask that are important to safety are listed on page 3.4-1 of the TSAR.

6. Cask fabrication activities shall be conducted in accordance with the reviewed and approved quality assurance program submitted with the TSAR.

7. Notification of cask fabrication schedules shall be made in accordance with the requirements of §72.232(c), 10 CFR Part 72.

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8. Casks of the Model No. MC-10 authorized by this certificate are hereby approved for general use by holders of 10 CFR Part 50 licenses for nuclear reactors at reactor sites under the general license issued pursuant to §72.210, 10 CFR Part 72, subject to the conditions specified by §72.212 and the attached Conditions for Cask Use.

9. Expiration Date:

August 31, 2010

FOR THE NUCLEAR REGULATORY COMMISSION

Charles J. Haughney

SIGNATURE

Chief, Fuel Cycle Safety Branch
Division of Industrial and
Medical Nuclear Safety
Office of Nuclear Material Safety
and Safeguards

CONDITIONS FOR CASK USE

CERTIFICATE OF COMPLIANCE

72-1001

TABLE OF CONTENTS

1.0 INTRODUCTION

These Conditions for Cask Use govern the safety of the receipt, possession, and storage of irradiated nuclear fuel at an Independent Spent Fuel Storage Installation (ISFSI) and the transfer of such irradiated nuclear fuel to and from a Nuclear Power Station and its ISFSI.

1.1 GENERAL CONDITIONS

1.1.1 Operating Procedures

Written operating procedures shall be prepared for cask handling, movement, emplacement, surveillance, and maintenance.

1.1.2 Quality Assurance

Activities at the ISFSI shall be conducted in accordance with the requirements of Appendix B, 10 CFR Part 50.

1.2 PREOPERATIONAL CONDITIONS

The user shall not allow the initial loading of spent nuclear fuel in the Model No. MC-10 cask until such time as the following preoperational license conditions are satisfied:

(1) A training module shall be developed for the Station Training Program establishing an ISFSI Training and Certification Program which will cover the following:

a. Cask Design (overview)

b. ISFSI Facility Design (overview)

c. ISFSI Safety Analysis (overview)

d. Fuel loading and cask handling procedures and abnormal procedures

(2) A training exercise (Dry Run) of cask loading and handling activities shall be held which shall include but not be limited to:

a. Moving cask in and out of spent fuel pool area.

b. Loading a fuel assembly (using dummy assembly).

c. Cask sealing and cover gas backfilling operations.

d. Moving cask to and placing it on the storage pad.

e. Returning the cask to the reactor.

f. Unloading the cask assuming fuel cladding failure.

g. Cask decontamination.

2.0 FUNCTIONAL AND OPERATING LIMITS

2 1 FUEL TO BE STORED AT ISFSI

2.1.1 Specification

The spent nuclear fuel to be received and stored at the ISFSI in MC-10 casks shall meet the following requirements:

(1) Only irradiated 14 x 14, 15 x 15, and 17 x 17 PWR fuel assemblies with Zircaloy fuel rod cladding may be used. Total assemblies per cask ≤ 24.

(2) Maximum initial enrichment shall not exceed 3.7 weight percent U-235 for fuel stored in the stainless steel basket (with boral plates attached to each of the 24 cell enclosure walls) reviewed and found acceptable.

(3) Maximum assembly average burnup shall not exceed 35,000 megawatt-days per metric ton uranium and specific power shall not exceed 35 kW/kg.

(4) Maximum heat generation rate shall not exceed 0.5625 kilowatt per fuel assembly.

(5) Fuel shall be intact unconsolidated fuel. Partial fuel assemblies, that is, fuel assemblies from which fuel pins are missing must not be stored unless dummy fuel pins are used to displace an amount of water equal to that displaced by the original pins.

(6) Fuel assemblies known or suspected to have structural defects sufficiently severe to adversely affect fuel handling and transfer capability unless canned shall not be loaded into the cask for storage.

(7) A procedure shall be developed for the documentation of the characterizations performed to select spent fuel to be stored in the casks. Such procedure shall include independent verification of fuel assembly selection by an individual other than the original individual making the selection.

(8) Immediately prior to insertion of a spent fuel assembly into a cask, the identity of the assembly shall be independently verified by two individuals.

2.1.2 Basis

The design criteria and subsequent safety analysis assumed certain characteristics and limitations for the fuels that are to be received and stored. Specification 2.1.1 assures that these bases remain valid by defining the type of spent fuel, maximum initial enrichment, irradiation history, and maximum thermal heat generation.

2.2 MC-10 DRY STORAGE CASK

2.2.1 Specification

The MC-10 Dry Storage Casks used to store spent nuclear fuel at an ISFSI shall have the operating limits shown in Table 2-1.

2.2.2 Basis

The design criteria and subsequent safety analysis of the MC-10 assumed certain characteristics and operating limits for the use of the casks. This specification assures that those design criteria are not exceeded.

Table 2-1 MC-10 OPERATING LIMITS

	Operating Limit
Max. Lifting Height with Non-Redundant Lifting Device	5 feet
Dose Rate	
. 2 m Distance	≦ 10 mrem/hr
. Surface	≦ 200 mrem/hr
Cask Tightness (at closure):	
(Standard He-Leak Rate)	
. Primary Cover Seal	≦ 10^{-6} std cc/s
. Primary Cover, Vent, Drain and Pressure Sensing Element Penetrations	≦ 10^{-6} std cc/s
Optional Seal Cover Weld	≦ 2×10^{-4} std cc/s
Max. Specific Power of One Fuel Assembly	0.5625 kW
Initial Helium Pressure Limit (Cask Cavity)	≦ 1.5 atmospheres

2.3 LIMITING CONDITION - HANDLING HEIGHT

2.3.1. Specification

This specification applies to handling of a cask being used for spent fuel storage outside of the Fuel Building and Crane Enclosure Building.

The MC-10 dry storage cask shall not be handled at a height of greater than 5 feet.

2.3.2 Basis

The drop analysis performed for the MC-10 dry storage casks for postulated cask drop incidents on the ISFSI storage pad indicates that the material of the fuel basket and cask body has sufficient ductility and toughness to sustain a drop of 5 feet or less without sustaining unacceptable damage to the casks and fuel basket. This limiting condition ensures that the handling height limits will not be exceeded at the storage pad or in transit to and from the reactor.

2.4 DRY STORAGE CASK SURFACE CONTAMINATION

2.4.1 Specification

Initial removable contamination on the dry storage cask shall not exceed 2200 dis/min/100 cm^2 from beta-gamma sources, and 220 dis/min/100 cm^2 from alpha sources.

2.4.2 Basis

Compliance with this limit ensures that the decontamination requirements of 49 CFR 173.443, will be met over the lifetime of the cask in storage.

2.5 DRY STORAGE CASK INTERNAL COVER GAS

2.5.1 Specification

The dry storage casks shall be backfilled with helium.

2.5.2 Basis

The thermal analysis performed for the dry storage casks assumes the use of helium as a cover gas. On addition, the use of an inert gas (helium) is to ensure long-term maintenance of fuel clad integrity.

2.6 LIMITING CONDITION - PRESSURE MONITORING DEVICE

2.6.1 Specification

The pressure monitoring device used to monitor the leak tightness of MC-10 dry storage cask or fuel rod integrity shall have the performance characteristics shown in Figure 5.1-1 of the TSAR.

3.0 SURVEILLANCE REQUIREMENTS

Requirements for surveillance of various radiation levels, cask internal pressure, contamination levels, cask seal leak rates, and fuel related parameters are contained in this section. These requirements are summarized in Table 3-1 from details contained in Section 3.1 through 3.6. Specified time intervals may be adjusted plus or minus 25 percent to accommodate normal test schedules.

Table 3-1 SURVEILLANCE REQUIREMENTS SUMMARY

Section	Quantity or Item	Period
3.1.1	Cask Loading Measurements	P
3.2.1	Cask Seal Testing	L
3.3.1	Cask Contamination	L
3.4.1	Dose Rates (Cask surface or up to 2 meters from cask surface)	L
	Dose Rates (Fence)	Q
3.5.1	Safety Status Surveillance	Q
3.6.1	Pressure Monitoring Device Parameters	P & L
3.7.1	Alarm System	A

P - Prior to cask loading

L - During loading operations

Q - Quarterly

A - Annually

3.1 CASK LOADING MEASUREMENTS

3.1.1 Specification

For the first loading of a cask model, cask side-wall surface dose rate shall be measured upon cask draining. Prior to moving the cask to the storage pad, cask surface temperature shall be measured after the cask has been sealed for an appropriate period, which should not be less than that expected for the cask surface temperature to come into approximate equilibrium. These dose rate and temperature measurements shall be made at the cask side-wall mid-line at three locations 120° apart around the cask circumference and shall be recorded to establish a baseline of comparison for all subsequent loadings of this model of cask.

For all subsequent loadings of casks of this model, measure and record cask side-wall surface dose rates and temperatures at the cask side-wall mid-line at three locations 120° apart and compare these to the baseline established during first cask use. Do not transfer the cask to the storage pad if unexplained variations (which can not be resolved through known differences in spent fuel assemblies loaded) are found.

3.1.2 Basis

These measurements are to assure that casks have been properly loaded.

3.2 CASK SEAL TESTING

3.2.1 Specification

Prior to storage, the cask must be properly sealed by testing as specified in Section 10.2.6 of the TSAR to an initial leak rate of 10^{-6} std cc/sec.

3.2.2 Basis

The safety analysis of leak tightness of the cask as discussed in the topical report is based on the seals after 20 years being leak tight to 10^{-4} std cc/s. This check is done to ensure compliance with this design criteria.

3.3 CASK CONTAMINATION

3.3.1 Specification

After cask loading and prior to moving the cask to the storage pad, the cask shall be swiped to ensure that removable surface contamination levels are less than 2200 dis/min/100 cm^2 from beta-gamma emitting sources, and 220 dis/min/100 cm^2 from alpha emitting sources.

3.3.2 Basis

This surveillance requirement will ensure compliance with the decontamination requirements of 49 CFR 173.443 during storage in the ISFSI.

3.4 DOSE RATES

3.4.1 Specification

The following dose rate measurements shall be made for the ISFSI:

(1) Cask Surface Gamma and Neutron Dose Rates: After completion of cask loading, gamma and neutron measurements shall be taken on the outside surface (or within 2 meters of the cask surface). The combined gamma and neutron dose rates shall be less than the surface dose rate stated in Table 2-1 (or the specified rate at a distance of up to 2 meters from the cask surface).

(2) Dry Cask ISFSI Boundary: Doses shall be determined by measurement at the Dry Cask ISFSI site fence and shall be evaluated on a quarterly basis to demonstrate compliance with §20.105(b)(2), 10 CFR Part 20.

3.4.2 Basis

These measurements are necessary to assure compliance with the cask specifications and that the dose rates at the security fence meet Part 20 limits as additional casks are placed in storage.

3.5 SAFETY STATUS SURVEILLANCE

3.5.1 Specification

A visual surveillance shall be performed on a quarterly basis of the ISFSI to determine that no significant damage or deterioration of the exterior of the explaced casks has occurred. Surveillance shall also include observation to determine that no significant accumulation of debris on cask surfaces has occurred.

3.5.2 Basis

The surveillance requirements shall ensure cask maintenance.

3.6 CASK CONFINEMENT INTEGRITY (MC-10)

3.6.1 Specification

The cask confinement integrity shall be monitored by use of a pressure monitoring device to verify the leak tightness of the cask. A functional test shall be performed during cask preparation.

3.6.2 Basis

This specification requires the cask cavity atmosphere be maintained and monitored to detect any possible leakage of cask seals.

3.7 ALARM SYSTEM

3.7.1 Specification

An alarm system to which all of the pressure monitoring devices are connected shall be installed at the storage site and functionally tested annually to ensure proper operation of the system.

3.7.2 Basis

The alarm system must be capable of alerting surveillance personnel of possible cask seal failure and must permit identification of the specific cask indicating a seal failure.