NRC FORM 618
(6-83)
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
5874	3	USA/5874/B( )F	1	2

2. PREAMBLE

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

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. PREPARED BY (Name and Address):

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

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION:

Safety Analysis for Radioactive Material Shipping Cask No. WAPD-40 dated December 1984.

c. DOCKET NUMBER 71-5874

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.: WAPD-40

(2) Description

End or top loading cylindrical, 10 inch lead shielded, 304L stainless steel clad cask for the shipment of irradiated test specimens. Cask has integral skid welded to the body. Cylindrical cavity is 1/4 inch thick 304L stainless steel tube with 2 inches bore by 135.25 inches length. There are stainless steel clad 10 inch thick lead shielded plugs bolted to each end. Each lid weighs 100 pounds. Overall size of the cask is 24 inches in diameter x 168 inch skid length. Gross weight with skid is 27,100 pounds. Using the four lifting trunnions as tiedowns to a truck is forbidden; hence, a special holddown cradle is used during truck shipments. This cradle weighs approximately 5000 pounds.

(3) Drawings

The WAPD-40 cask was originally fabricated in accordance with Westinghouse Assembly Drawing No. 936F577, Rev. 4, and later modified in accordance with Battelle Memorial Institute Drawing No. 100-E, Rev. 0.

(4) Product Containers

The contents of the package must be packaged in inner product containers. The inner product containers are constructed in accordance with the following Westinghouse Electric Corporation Drawing Nos.:

Product Container	Drawing Nos.
IN-40	979C282, Rev. 1 and 971D362, Rev. 1
LRS	979C194, Rev. 2
LLR	979C277, Rev. 3

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5. (b) Contents

(1) Type and form of material

Byproduct and special nuclear material contained within product containers. The contents must be dry and unmoderated (H to X atomic ratio < 2).

(2) Maximum quantity of material per package

The fissile content of the cask must be limited to a maximum of 350 equivalent grams of U-235. The number of equivalent grams of U-235 is determined by the equation: 1.0 x grams U-235 + 1.4 x grams U-233 + 1.6 x grams plutonium.

(c) Fissile Class II

Minimum transport index to be shown on label 3.2

6. Maximum decay heat per package must not exceed 8,400 BTU/hr.

7. The acceptance tests and maintenance program must be in accordance with Chapter 8.0 to WAPD-REO(C)-270, Rev. 3.

8. Expiration date: May 31, 1990.

REFERENCE

Safety Analysis for Radioactive Material Shipping Cask No. NRBK-40 dated December 1984 (WAPD-REO(C)-270, through Rev. No. 4).

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Charles E. MacDonald

Charles E. MacDonald, Chief
Transportation Certification Branch
Division of Fuel Cycle and
Material Safety, NMSS

Date: MAY 3 1 1985

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

Transportation Certification Branch
Approval Record
Model No. WAPD-40 Package
Docket No. 71-5874

By application dated December 26, 1984, Department of Energy, Naval Reactors submitted a revised Safety Analysis Report (SAR) for the NRC Certificate of Compliance No. 5874. The revised Safety Analysis Report has taken into account the cumulative effects of the hypothetical accident conditions as stipulated in Part 71. In addition, a 1/2-inch thick, 9-inch diameter cover plate has been added to each of the cask end plugs. All other aspects of the cask are consistent with the original design. The package has been renewed for a 5-year period.

STRUCTURAL EVALUATION

Two lifting trunnions are provided at each end of the cask. They are shown by analysis to be capable of lifting three times the maximum package weight. Excessive forces on the trunnions would tear the 1/2-inch fillet weld between the trunnion base and the outer shell, the package containment and shielding properties would not be degraded in any way.

The WAPD-40 cask and skid (which are welded to each other) are held down in the transport vehicle by a tie-down cradle. The cask sits in the tie-down cradle. Two hold-down frames fit over each end of the cask. Tie-down of the cask was provided through overlapping fit. This tie-down arrangement was shown by analysis to be capable of withstanding the resultant force from the three specified component forces of 10 CFR §71.45(b). The lifting trunnions which could be used as tie-down devices will be covered during shipment.

The SAR used existing analytical techniques and reasoned argument to evaluate the package for the normal and accident tests specified in 10 CFR Part 71. The SAR adequately demonstrates that the package will remain intact and perform its intended safety function under the Normal Conditions of Transport.

The package was evaluated for a 30-foot drop test in the end, side, stable corner (COG), and oblique corner (20°, 40°, and 60° from the vertical) orientations. The results of the analyses indicate the following: (1) None of the end plug fasteners (at either end) are stressed above yield during any of 30-foot drop tests. Thus, the end plugs remain attached to the cask which will ensure that the inner container and cargo cannot excape from the cask; (2) the maximum load calculated for the fasteners is less than the preload applied to them. Thus, the sealing capability associated with the end plug will be maintained.

The package was also evaluated for the 40-inch puncture test. The results indicate the 6-inch diameter pin could puncture through the outer shell and into the lead a distance of 0.66 inches. The significance of puncture with respect to potential loss of containment integrity is evaluated in the application. The results of the analyses indicate that the package deformation associated with the puncture load and package inertia loads would not result in excessive stresses in either the cask inner cylinder or the inner container.

The cumulative effects of the 30-foot drops, the 40-inch puncture drop, and the 1475°F fire accident on the package are reported. Information presented in the structural section indicate that shielding remains adequate for the Accident Conditions of Transport (Chapter 5.0, Shielding Evaluation).

Due to the higher failure temperature of the metallic outer shell gaskets than that of the 1475°F fire, the outer shell gaskets would remain intact during the fire test. Thus, the package will not become flooded during the water immersion test.

CONTAINMENT

The containment provisions and associated tests for the Model No. WAPD-40 cask discussed in the SAR have been reviewed. The staff has concluded that the containment requirements of 10 CFR §71.51 are satisfied.

The containment criteria is leaktightness. Leaktightness is satisfied by demonstrating no leakage for test sensitivity of 1×10^{-7} atm-cm^3/s. Leaktightness is to be demonstrated before first use (8.1.3, Chapter 8.0 of SAR) and annually (8.?.2, Chapter 8.0 of SAR). In addition, an assembly verification leak test with sensitivity of 1×10^{-3} atm-cm^3/s is specified (8.2.2, Chapter 8.0 of SAR).

To assure adequate sealing of the containment system, both the metallic and silicon rubber O-rings of the containment system are replaced before each use (8.1.4.2, Chapter 8.0 of SAR).

THERMAL

An independent thermal analysis was completed for the Model No. WAPD-40 shipping cask using the HTAS1/HEATING6 computer code (NUREG/CR-0200). The initial conditions, such as ambient temperature, solar insolation, and the values used to define the fire accident condition, were taken directly from NRC regulations (Appendices A and B, 10 CFR Part 71). In evaluating the Model No. WAPD-40 cask, it was assumed that the molten lead shielding was completely retained by the inner and outer shells during the fire accident condition, i.e., it was not permitted to leak out forming insulating air gaps between the remaining shielding and the outer shell. The internal heat load was 2,000 BTU/HR.

The analysis showed that shield melting occurred under the prescribed fire accident conditions. Lead melting was most pronounced at a point approximately 45 minutes from the onset of the fire or 15 minutes after the fire was "extinguished." At that time, approximately 2.5 inches of lead were melted along the sides and one-half inch along the top and bottom. This represented approximately 40-45% of the original lead volume. Temperatures within the shielding ranged from 612°F at the inner wall interface to 621.5 at the outer wall interface; i.e., the whole shield was within 10 degrees of the lead melting point. However, for the whole shield to melt approximately twice as much heat would have to be provided than was experienced under the hypothetical fire conditions.

Under the worst case scenarios, the cask would rupture at the 45 minute mark. After the molten lead escaped, the remaining shielding would be about 7-7.5 inches thick. The applicant calculated that the lead thickness remaining would be about 5.4 inches thick. The difference between the calculations is due to more conservative assumptions by the applicant.

SHIELDING

The application demonstrates that after the thermal fire test, about 4.5 inches of lead (Pb) (originally 9.875" of Pb) melts along the side walls of the cask. Using the SPAN4 shielding computer program, the gamma dose rate was calculated at three feet from the package surface for irradiated steel contents to be 1,000 mrem/hr for the above lead (Pb) melt. Applicant's gamma dose rates for the two contents and both normal and accident conditions are given in the following table.

Contents	Gamma Dose Rate (mrem/hr) at 3' from surface of package Normal Condition	Accident Condition
Irradiated Steel	0.17	1000
Uranium Fuel Spent	0.81	963

The staff has calculated its own gamma dose rate for the irradiated steel case using a line source and gamma build-up in lead. Assuming $S_o=5.0\times10^{14}$ MEV/sec giving an $S_L=1.5\times10^{12}$ MEV/sec-cm, a gamma dose rate at 3 feet from the package surface was calculated to be 0.70 mrem/hr (versus applicant's value of 0.17) for normal conditions. For accident conditions (5.4"Pb) the result was 965 mrem/hr (versus applicant value of 1,000 mrem/hr). Since the applicant's dose rates include a safety factor of 1.3, the accident dose rates differ by about 25% in a conservative manner.

RENEWAL

The package has been renewed for a 5-year period.

Charles E. MacDonald

Charles E. MacDonald, Chief
Transportation Certification Branch
Division of Fuel Cycle and
Material Safety, NMSS

Date: MAY 3 1 1985