NRC FORM 618
(8-85)
10 CFR 71

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIALS PACKAGES

NUCLEAR REGULATORY COMMISSION

1 a CERTIFICATE NUMBER	b REVISION NUMBER	c PACKAGE IDENTIFICATION NUMBER	d PAGE NUMBER	e TOTAL NUMBER PAGES
9200	0	USA/9200/B(M)F	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)*

Department of Energy
Idaho Operations Office
550 Second Street
Idaho Falls, ID 83401

b TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Nuclear Packaging, Inc. application dated June 14, 1985, as supplemented.

c DOCKET NUMBER 71-9200

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.: 125-B

(2) Description

A stainless steel and lead shielded shipping cask. The contents are shipped dewatered. The cask is a right circular cylinder, 65.5-inch outer diameter by 207.5-inch length. The cavity dimensions are 51.25-inch diameter by 192.5-inch length. A 1.0-inch thick stainless steel inner shell, 3.88-inch thick lead annulus and 2.0-inch thick stainless steel outer shell and 7.50-inch thick welded stainless steel bottom plate make up the cask body. A ten gauge stainless steel thermal shield surrounds the cask outer shell with standoff provided by a wire wrap on a 3.3-inch pitch spacing. The outer lid is 7.50-inch thick stainless steel equipped with a 300 psig rupture disc. The seal is provided by 2 Neoprene O-rings secured by 32, 1-1/2-6 UNC closure bolts. A test port is provided between the O-rings. The lid is also provided with a vent port. Protrusions from the outer cask external cylindrical surface include 2 lifting and 4 tie-down trunnions, 1 shear block for fitting to the shipping skid, and 16 impact limiter attachment lugs (8 at each end of the cask). The impact limiters are 120 inches in diameter by 75 inches long fabricated from 1/4-inch thick stainless steel and filled with closed-cell polyurethane foam. Each impact limiter is secured to the cask by 8, 1-1/4-7 UNC bolts necked down to 1 inch. Plastic pipe plugs are provided in each impact limiter. The overall dimensions of the cask with upper and lower impact limiters are 120-inch outer diameter by 279.5-inch length.

(a) (2) Description (continued)

A separate inner vessel (fuel/canister basket) is positioned within the cask cavity. The inner vessel consists of 7, 14.5-inch ID by 0.38-inch wall pipes with a welded bottom plate and top end fixture plate which provides a 151-inch long cavity for the canisters. The pipe assembly is positioned within a 50.25-inch OD by 1.0-inch thick steel shell with a 2.0-inch thick welded bottom plate. The space between the pipes and steel shell contain stainless steel structural members and solid neutron moderator and absorber. The top of each tube is shielded by a 10-inch thick stainless steel plug. The inner lid is 5.0-inch thick stainless steel equipped with 2, 300 psig rupture discs in series. The lid has 2 Neoprene O-rings and is secured to the inner vessel by 24, 3/4-10 UNC closure bolts. A test port is provided between the O-rings. The lid is also provided with a vent port.

A fuel, filter, or knockout canister is positioned within the inner vessel with canister impact limiters and a top 10.0-inch thick stainless steel shield plug. Each canister is 14.0-inch OD by 150.0-inch long by 0.25-inch wall and contains Boral sheets or B_4C rods. Canister containment is not required with closure provided by welded or bolted plate with 2 or 4 fittings.

The weight of the cask (100,500 pounds), impact limiters (11,700 pounds each), inner vessel (37,000 pounds), canisters (1,046 to 1,440 pounds each), and canister contents (1,500 to 1,894 pounds each) is approximately 181,500 pounds.

(3) Drawings

(i) The packaging is constructed in accordance with Nuclear Packaging, Inc. Drawing No. X-101-100, Sheets 1 through 6, Rev. H.

(ii) The canisters are constructed in accordance with Babcock and Wilcox Company Drawing Nos.: 1161299D, Rev. 1; 1161300D, Rev. 1; and 1161301D, Rev. 1.

(b) Contents

(1) Type and form of material

Byproduct and special nuclear material in the form of irradiated fuel particles, partial fuel rods, partial assemblies, and core debris. The maximum pre-irradiation U-235 enrichment must not exceed 2.98 weight percent. The average burnup of the fuel material must not exceed 3,165 MWD/MTU and must be cooled for at least 6.0 years.

(2) Maximum quantity of material per package

Seven fuel, knockout, or filter canisters or any combination thereof within the inner vessel. The radioactive decay heat load must not exceed 100 watts in each canister. The gross weight of each canister must not exceed 2,940 pounds.

(c) Fissile Class	III
Maximum number of packages per shipment	One

6. The cask cavity and inner vessel must be dry (no free water) when delivered to a carrier for transport. The canisters must be loaded and dewatered in accordance with Section 7.1.1 of the application which includes approximately two atmospheres of argon, nitrogen, or helium cover gas. The cask cavity and inner vessel must be filled with argon, nitrogen, or helium at 1.0 atm pressure.

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

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

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

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

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

8. (Continued)

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

9. Bolt torque:

The outer cask lid must be secured by 32, ASTM A320, Grade L43, 1-1/2-6 UNC-2A x 5.5 long bolts torqued to 1,200-1,450 ft-lbs (lubricated).

The inner vessel lid must be secured by 24, ASTM A320, Grade L43, 3/4-10 UNC-2A x 2.25 long bolts torqued to 200-240 ft-lbs (lubricated).

The upper and lower overpack limiters must each be secured by 8, ASTM A320, Grade L43, 1-1/4-7 UNC-2A x 41.75 long bolts torqued to 350-415 ft-lbs (lubricated).

10. Prior to each shipment, the licensee must confirm that the cask and inner vessel are properly sealed by tests as specified in Appendix 7.4.

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

12. Expiration date: April 30, 1991.

REFERENCES

Nuclear Packaging, Inc. application dated June 14, 1985.

Supplements dated: October 31 and November 22, 1985; and February 11, 1986.

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: APR 1 1 1986

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

Safety Evaluation Report
Department of Energy
Model No. 125-B Package
Docket No. 71-9200

SUMMARY

By application dated June 14, 1985, as supplemented, Nuclear Packaging, Inc. (NUPAC) requested design approval on behalf of the Department of Energy of the Model No. 125-B shipping package (cask). The Department of Energy is the owner of the cask design.

Based on the statements and representations in the application and the conditions listed below, we have concluded that the Model No. 125-B cask design meets the requirements of 10 CFR Part 71.

REFERENCES

1. NUPAC application dated June 14, 1985.

2. Supplement dated October 31, 1985.

3. Supplement dated November 22, 1985.

4. Supplement dated February 11, 1986.

DRAWINGS

1. The packaging is constructed in accordance with Nuclear Packaging, Inc. Drawing No. X-101-100, Sheets 1 through 6, Rev. H.

   The drawings provide information pertaining to materials of construction, component dimensions and tolerances, and the location and size of all weld joints. The drawings identify the weld joints to be nondestructively examined, the method to be used, and the code or standard for the examination procedure. Welders and welding procedures must be qualified in accordance with AWS D1.1 or the ASME Code Section IX. Nondestructive testing requirements are in accordance with ASME Code, Section III, Subsection NB.

2. The canisters are constructed in accordance with Babcock and Wilcox Company Drawing Nos.: 1161299D, Rev. 1; 1161300D, Rev. 1; and 1161301D, Rev. 1.

DESCRIPTION

A stainless steel and lead shielded shipping cask. The contents are shipped dewatered The cask is a right circular cylinder, 65.5-inch outer diameter by 207.5-inch length. The cavity dimensions are 51.25-inch diameter by 192.5-inch length. A 1.0-inch thick stainless steel inner shell, 3.88-inch thick lead annulus and 2.0-inch thick stainless steel outer shell and 7.50-inch thick welded stainless steel bottom plate make up the cask body. A ten gauge stainless steel thermal shield surrounds the cask outer shell with standoff provided by a wire wrap on a 3.3-inch pitch spacing. The outer lid is 7.50-inch thick stainless steel equipped with a 300 psig rupture disc. The seal is provided by 2 Neoprene O-rings secured by 32, 1-1/2-6 UNC closure bolts. A test port is provided between the O-rings. The lid is also provided with a vent port. Protrusions from the outer cask external cylindrical surface include 2 lifting and 4 tie-down trunnions, 1 shear block for fitting to the shipping skid, and 16 impact limiter attachment lugs (8 at each end of the cask). The impact limiters are 120 inches in diameter by 75 inches long fabricated from 1/4-inch thick stainless steel and filled with closed-cell polyurethane foam. Each impact limiter is secured to the cask by 8, 1-1/4-7 UNC bolts necked down to 1 inch. Plastic pipe plugs are provided in each impact limiter. The overall dimensions of the cask with upper and lower impact limiters are 120-inch outer diameter by 279.5-inch length.

A separate inner vessel (fuel/canister basket) is positioned within the cask cavity. The inner vessel consists of 7, 14.5-inch ID by 0.38-inch wall pipes with a welded bottom plate and top end fixture plate which provides a 151-inch long cavity for the canisters. The pipe assembly is positioned within a 50.25-inch OD by 1.0-inch thick steel shell with a 2.0-inch thick welded bottom plate. The space between the pipes and steel shell contain stainless steel structural members and solid neutron moderator and absorber. The top of each tube is shielded by a 10-inch stainless steel plug. The inner lid is 5.0-inch thick stainless steel equipped with 2, 300 psig rupture discs in series. The lid has 2 Neoprene O-rings and is secured to the inner vessel by 24, 3/4-10 UNC closure bolts. A test port is provided between the O-rings. The lid is also provided with a vent port.

A fuel, filter, or knockout canister is positioned within the inner vessel with canister impact limiters and a top 10.0-inch thick stainless steel (SS) shield plug. Each canister is 14.0-inch OD by 150.0-inch long by 0.25-inch wall and contains Boral sheets or B_4C rods. Canister containment is not required with closure provided by welded or bolted plate with 2 or 4 fittings.

The weight of the cask (100,500 pounds), impact limiters (11,700 pounds each), inner vessel (37,000 pounds), canisters (1,046 to 1,440 pounds each), and canister contents (1,500 to 1,894 pounds each) is approximately 181,500 pounds.

CONTENTS AND FISSILE CLASS

(1) Type and form of material

Byproduct and special nuclear material in the form of irradiated fuel particles, partial fuel rods, partial assemblies, and core debris. The maximum pre-irradiation U-235 enrichment must not exceed 2.98 weight percent. The average burnup of the fuel material must not exceed 3,165 MWD/MTU and must be cooled for at least 6.0 years.

(2) Maximum quantity of material per package

Seven fuel, knockout, or filter canisters or any combination thereof within the inner vessel. The radioactive decay heat load must not exceed 100 watts in each canister. The gross weight of each canister must not exceed 2,940 pounds.

(3) Fissile Class	III
Maximum number of packages per shipment	One

STRUCTURAL

The applicant has performed various structural analyses, engineering evaluations, and physical tests to satisfactorily demonstrate that the package has adequate structural integrity to meet the requirements of 10 CFR Part 71. The applicant's evaluation considered applicable load combinations from Regulatory Guide 7.8, "Load Combinations for the Structural Analysis of Shipping Casks." The applicant's analysis shows that stresses are within the limits specified in Regulatory Guide 7.6, "Design Criteria for the Structural Analysis of Shipping Cask Containment Vessels," The applicant has shown that the containment vessel is designed so that brittle fracture and buckling would not occur under the test conditions specified in 10 CFR Part 71. The applicant has conducted a series of physical tests using a quarter scale model of the shipping cask and full size payload canisters. The results of these tests support and confirm the applicant's engineering analyses and evaluations. The staff agrees with the applicant's conclusion that the package design has adequate structural integrity to meet the requirements of 10 CFR Part 71.

A. General Standards

Chemical and Galvanic Reactions

There is no contact between any materials of construction of the packaging that could produce a chemical or galvanic reaction.

Positive Closure

Inadvertent opening of the package is prevented by means of positive closure design. Primary access to the cask cavity is through a bolted closure. The other penetrations to the cask cavity are vent and rupture ports which are closed with a bolt or a rupture disk.

Lifting Devices

The applicant has shown that the trunnions used for lifting can support three times the weight of the package (with impact limiters removed) without yielding. Failure of these devices under excessive load would not impair the containment or shielding provided by the package.

Tie-Down Devices

The package is tied down through structural connections to two trunnions at each end of the cask and by engagement of a longitudinal shear block plate located near the middle of the cask. The shear block is designed to react loadings in the longitudinal direction. Clearance is provided at the trunnion connections to accommodate differential thermal expansion or contraction of the cask relative to the vehicle. The portions of the tie-down system which are a structural part of the package (i.e., trunnions and shear block) have been designed to resist, without yielding, a force applied to the center of gravity of the package having components of 2-g, 5-g, and 10-g in the vertical, transverse, and longitudinal directions respectively. The structural integrity of the cask is adequate to withstand the tie-down forces.

B. Normal Conditions of Transport

Heat

The package has adequate structural integrity for the normal condition heat test. The temperature of all cask components is less than 200°F. The maximum normal operating pressure is 125 psig. Differential thermal expansion of cask components produces only a relatively low level of stress in the shells because: (1) the thermal gradients through the walls are low, and (2) the lead shielding is not bonded to the steel shells. Stress analyses performed by the applicant show that the maximum stresses in the cask are substantially below allowable limits.

Cold

The inner and outer containment vessels are shipped dry; although a limited amount of water may be present in the canisters. Ample space is provided for expansion of the water in the event of freezing. The inner and outer containment vessels are constructed of stainless steel and are not subject to brittle failure at -40°F.

The applicant performed a stress analysis to evaluate differential thermal contraction of the lead shielding and the stainless steel shell. The results show that stresses in the shells will be well within allowable limits.

Reduced External Pressure

A reduced external pressure of 3.5 psia will not effect the structural integrity of the package.

Increased External Pressure

An increased internal pressure of 20 psia will not effect the structural integrity of the package.

Vibration

The applicant conservatively assumed normal vibration loads of 2-g's. With this loading, the applicant's analysis shows that the maximum stresses in the cask and tie-down trunnions is within the allowable alternating stress intensity for $(10)^{11}$ cycles of stress. The staff agrees with the applicant's conclusion that the cask is adequately designed for normal vibration.

One-Foot Free Drop

The applicant has analyzed the package to determine the effects of the one-foot drop test and has shown that the package has sufficient structural integrity to meet the acceptance standards in 10 CFR Part 71. Specifically, the applicant has shown that the stresses in the shells will be within allowable limits, the cask will provide containment of the contents, and the effectiveness of the packaging will not be reduced.

The applicant evaluated the g-loads and stresses that would result from a one-foot drop with the cask oriented to impact on its side, end, and at various oblique angles. Several cases were analyzed to consider variations in foam strength and impact limiter effectiveness. The maximum g-loads and impact limiter deformations for the one-foot drop test were estimated as follows:

Orientation	Maximum g-load (g's)	Maximum Impact Limiter Deformation (in.)
90° (End)	33.3	1.94
70°	6.8	11.63
50°	4.2	13.73
30°	2.0	10.71
0° (Side)	16.2	3.68

The applicant performed a stress analysis to show that stresses in the cask are within allowable limits. An analysis was also performed to estimate lead displacement ("slump") under the one-foot end drop test. The maximum lead displacement was calculated to be 0.91 inches.

Penetration

The exterior surfaces of the package are capable of withstanding the impact forces imposed by the normal condition penetration test. There are no valves or relief devices which could be contacted by the 13 pound steel bar used in the test.

C. Hypothetical Accident Conditions

Scale Model Tests

The applicant's evaluation of the package for hypothetical accident conditions included both engineering analysis of the design and physical testing of a quarter scale model. Although minor design differences exist between the quarter scale model and the full scale cask, the staff agrees with the applicant's conclusion that the structural performance of the test article is representative of the full scale cask.

The scale model cask was subjected to the following tests, in sequence:

- o 30-foot bottom end drop test at -20°F
- o 30-foot oblique angle test onto the lid end of the cask (with the cask axis oriented 62° from horizontal) at -20°F
- o 30-foot side drop test at ambient temperature
- o 40-inch side drop test onto puncture pin
- o 40-inch lid end drop onto puncture pin

Leak tests performed by a helium mass spectrometer indicated no leakage, before or after the tests, from either the primary or secondary containment vessels. Except for local indentation of the outer shell as a result of the side puncture test, permanent deformation was limited to the internal and external impact limiters. X-rays indicated no detectable displacement ("slump") of the lead shielding. The impact limiters remained attached and no buckling of the shells or internal structures was observed.

30-Foot Drop Test

The applicant has analyzed the package under 30-foot drop test conditions and has shown that the package has sufficient structural integrity to meet the acceptance standards in 10 CFR Part 71. The package was analyzed for side, end, corner, and various oblique angle orientations. For the purpose of evaluating the performance of the impact limiters, the applicant conservatively assumed that all the available kinetic energy would be dissipated by impact limiter deformation. Several cases were analyzed to consider variations in foam strength and impact limiter effectiveness. For the 30-foot drop test, g-loads and impact limiter deformations were estimated as follows:

Orientation	Maximum g-load (g's)	Maximum Impact Limiter Deformation (in.)
90° (End)	51.6	25.78
70°	37.1	35.28
50°	23.6	37.41
30°	16.5	31.22
0° (Side)*	56.6	19.32

The response of the package was determined by applying the peak g-loads as static forces to structural members of the cask. The calculated stresses were within the allowable limits specified in Regulatory Guide 7.6. The applicant performed an analysis which shows that the stresses produced by secondary impact ("slapdown") for oblique angle drop orientations are less than the stresses produced under side drop test conditions. An analysis was also performed to estimate lead displacement ("slump"). The maximum lead displacement was calculated to be 1.63 inches.

*A side drop g-load of 67.8 g's and a deformation of 22.65 inches were calculated for one case where the impact limiters were assumed to be only partially effective (i.e., energy would be dissipated in only a limited region) and the strength of the foam was assumed to be degraded. Based on the results of the quarter scale model test (45.0 g's for side drop), the applicant concluded that the analytical cases which considered the impact limiter to be fully effective were more representative of the actual performance of the impact limiters.

Puncture

An ORNL-NSIC-68 equation for lead backed steel cylinders was used by the applicant to demonstrate that the outer shell of the cask would not experience a local rupture near the point of impact as a result of the puncture test. The overall effect of the cask striking the pin in a horizontal position at mid-length was also evaluated. The applicant's analysis shows that the stresses in the shells would be within allowable limits.

The top and bottom ends of the cask are protected against puncture by the impact limiters. The applicant performed an inelastic analysis of the puncture test with the cask in a vertical position, striking the pin in the end region of the cask. The analysis considered the force required to cause the pin to deform plastically and conservatively neglected the beneficial effects of the impact limiters. Although the analysis concluded the cask has adequate structural integrity for this test, the applicant and the staff consider the quarter scale model test as the principal means of demonstrating the adequacy of the design.

Thermal Stresses

The applicant has demonstrated that the package has adequate structural integrity to safely withstand the half-hour fire test. The lead shielding is not bonded to the inner or outer shells and the thermal gradient across the cask wall does not exceed 105°F. Consequently, differential thermal expansion will not produce large stresses. The maximum internal pressure from the fire test is 125 psig. Both the inner and outer containment vessels are fitted with rupture disks which relieve at 300 psig. The applicant's stress analysis of the package under fire test conditions shows that the stresses in the shells are within allowable limits and that the containment vessels can withstand an internal pressure of 300 psi (relief pressure).

Payload Canisters

The contents of the package are loaded within payload canisters of three different types: fuel canister, knockout canister, and filter canister. The length, diameter, and lower head design of each canister is the same; although the internal components and upper head designs are different.

The geometry of the canisters and their internal components is used in the criticality evaluation of the package. The canisters are not intended to be leaktight in transport. They serve to provide adequate confinement of the radioactive debris and particles for the purpose of criticality control. The applicant conducted a series of physical tests to demonstrate the structural adequacy of the canisters and to support the criticality evaluation.

1. Fuel Canister - A full size fuel canister was dropped from a height of 18 feet onto an essentially unyielding surface, striking on its bottom end. (Although the drop height was 18 feet rather than the 30 feet specified for the cask, the canister was dropped bare and did not benefit from the protection that would be afforded by the cask and its impact limiters.) A shortened fuel canister was dropped on its side from a height of 30 feet. The results of these tests showed that the Boral shroud would remain in place and maintain its basic shape.

2. Knockout Canister - A full size knockout canister was subjected to the following tests, in sequence: 30-foot bottom end drop, 30-foot side drop, 30-foot top end drop, and 30-foot side drop. In conducting these tests, the canister was placed inside a steel pipe to simulate the receptacles inside the cask. The pipe was fitted with impact limiters to control the impact loads. The impact loads measured in the tests exceeded the impact loads that the canister would experience had it been dropped inside the cask. The results of the test show that the canister confined the simulated contents throughout the tests and the configuration of the canister internals remained within the bounds considered in the criticality analysis.

3. Filter Canister - Bench tests were conducted on individual filter elements to determine their axial and lateral crush strength. The criticality analysis is based upon the most reactive configuration of the filter bundle. In determining the most reactive configuration, the applicant conservatively assumed maximum filter element compaction per the bench tests, repositioning of the poison rod, filter elements and drain tube, and reduced diameter of the endcaps. The filter elements were assumed to overlap until the reduced diameter endcap touched the center drain tube, taking no spacing credit for intervening media. For additional conservatism, the B_4C pellets were assumed to be completely removed from the canister in the criticality analysis.

All canisters will be stamped per the ASME Code, Section VIII, Division I, prior to loading.

THERMAL

The applicant analyzed the thermal behavior of the Model No. 125-B cask design using a 208-node, three-dimensional heat transfer computer program (MITAS II).

The thermal model employed represented a 60 degree symmetrical segment of the cask. In modelling the cask, it was assumed that the heat load was uniformly distributed.

The internal structure and contents within each canister were modeled using single-lump nodes (four evenly distributed nodes per canister). The transfer of heat from the contents to the exterior wall of the canister was calculated using only radiation. This predicted a higher temperature for the canister internals than would be expected if convection was included. This assumption also results in a conservative estimate for internal pressure. In addition, a one-dimensional model was used to determine the effects of the overpacks on each end of the cask.

In order to determine the response of the Model No. 125-B cask design to the requirements of 10 CFR Part 71, the applicant performed four principal heat transfer analyses:

Normal Conditions of Transport:

1. Steady state at an ambient temperature of 100°F with insolation.

2. Steady state at an ambient temperature of 100°F without insolation.

Hypothetical Accident Conditions:

1. Transient thermal analysis with pin drop damage.

2. Transient thermal analysis with side drop damage to the overpacks.

The Model No. 125-B cask design was evaluated for normal transport and the hypothetical accident conditions using an internal heat load of 100 watts (340 BTU/hr) per canister. The following temperatures were predicted by the applicant for the thermal conditions specified in 10 CFR Part 71:

Location	NORMAL CONDITIONS (Steady State, °F) (w/ solar)	(w/o solar)	HYPOTHETICAL ACCIDENT CONDITIONS (Transient, °F) (Pin Drop Damage)	(Overpack Damage)
Canister Shell	161	141	180	180
Inner Vessel Tube	154	133	191	191
Inner Vessel Outer Shell	147	127	195	194
Cask Inner Shell	135	113	501	495
Lead Shield	135	113	526	517
Cask Outer Shell	135	113	606	600
Thermal Shield	134	113	1,090	1,090
Overpack Shell	135	100	1,423	1,423
Internal Shield Plug	143	122	128	133
Inner Vessel O-Ring Seal	139	118	123	134
Cask O-Ring Seal	135	113	133	228
BISCO NS-3 and NS-4	148	127	192	192
Cask End Plates	141	120	186	291

The results of the applicant's thermal analysis show that the maximum temperatures for both the normal transport and the hypothetical accident conditions do not exceed the maximum functional temperatures of the cask components. The cask's most thermally sensitive components, the Neoprene O-rings, the BISCO neutron moderators and the polyurethane foam overpacks do not exceed their allowable operating temperatures. The Neoprene O-rings, which have an allowable temperature range of -40°F to 250°F, do not exceed 230°F. The BISCO moderators can sustain prolonged exposures at temperatures of up to 220°F with short-term excursions to 300°F. The maximum temperature predicted for the moderators is 192°F. The recommended temperature range for the polyurethane foam is -30 to 150°F. However, in the hypothetical accident condition, the foam overpacks are assumed to char at approximately 400°F and be replaced by an air void of equivalent dimensions, maintaining thermal insulation. In addition, the lead shielding does not melt under hypothetical accident conditions. The thermal shield and foam overpacks provide adequate thermal protection to prevent critical cask components from exceeding their maximum functional temperatures.

The applicant calculated internal pressures for hypothetical accident conditions for the cask's three pressure retaining boundaries, inner vessel, cask cavity, and fuel canisters. Temperatures of 161°F and 191°F were calculated for the canisters and inner vessel, however, a temperature of 200°F was assumed for pressure calculations. Also considered in the pressure calculations were the effects of (1) radiolytic decomposition of water within the canisters, (2) off-gassing of the borated neutron moderators and absorbers, (3) vapor pressure of water within canister debris, and (4) volumetric expansion of the inert Argon purge gas filling each of the three pressure retaining boundaries. The maximum pressures were dependent on the canister and its contents:

	Canister Type			
	Fuel (psig)	Filter (psig)	Knockout (psig)	Design Limit (psig)
Canisters	165.5	104.4	119.7	---
Inner vessel	119.5	83.8	97.0	125
Cask cavity	92.0	70.9	81.3	125

The above results assume that radiolysis without recombination occurs over a one-year period and gases within the interior pressure boundaries are free to communicate with gases in the boundary being considered. This leads to a conservative (higher) prediction of pressure within the boundary being considered. The maximum internal pressure for any cask component (excluding the canisters) under hypothetical accident conditions was less than 125 psig. The pressures in the canisters ranged from 104-166 psig. The maximum external pressure acting on the inner vessel tubes (receptacles for the canisters) was approximately 33 psig - the difference between pressures calculated at the inner vessel outer shell at 349°F and the canister shell at 136°F. The temperatures are taken from the nodes in the MITAS II program which represent the average gas temperatures in the inner vessel and canister shell cavities. It was assumed in the calculation for the canisters that there was no communication across pressure boundaries.

In order to verify the results obtained by the applicant, an independent thermal evaluation using the HTAS1/HEATING6 computer programs, a part of the SCALE program (NUREG/CR-0200) was performed by the NRC staff. The thermal behavior predicted by the applicant (MITAS II) for each of above cases was found to be consistent with that predicted by the HTAS1/HEATING6 finite difference computer program. In most cases, the temperatures predicted by HTAS1/HEATING6 were within 20 degrees of those predicted by the applicant's model. In no case did the NRC staff calculate temperatures that would alter the conclusions reached by the applicant. Following the hypothetical accident conditions, the temperatures would be expected to return to those calculated by the applicant for normal conditions of transport.

The results of the applicant's thermal analysis, as confirmed by the NRC staff, demonstrate that the maximum temperatures and pressures for both normal and hypothetical accident conditions do not exceed the maximum functional temperature and pressures of any of the cask's components. The results indicated that the combination of foam overpacks and thermal shield was effective in preventing lead melting and thermal degradation of closure seals and BISCO moderators. The maximum internal pressures calculated were within the pressure ratings of the cask's components.

CONTAINMENT

The contents are dewatered solids and retained in canisters. The canisters are doubly contained. Each containment barrier uses a double Neoprene O-ring arrangement to provide a testable lid closure. Up to seven canisters can be held in the primary container (inner vessel) which is held within the secondary container (cask cavity). The seven canister compartments share a common lid. Both the primary and secondary lids have pressure relief devices in the event the pressure limits imposed by the normal and hypothetical accident conditions of transport are exceed.

The containment criteria specified for both the primary and secondary containment is leaktightness (i.e., leakage not to exceed 1×10^{-7} atm-cm^3/s dry air at 25°C and 1 atm leaking to 10^{-2} atm). Satisfying containment for 10 CFR Part 71 is based on the conclusions drawn from structural and thermal analyses of the cask that the normal and hypothetical accident conditions do not reduce containment capability.

The containment criteria of 10 CFR §71.51 and the special requirements for plutonium shipments in 10 CFR §71.63 have been demonstrated to be satisfied.

Assembly verification for both the primary and secondary containment prior to each shipment is accomplished by leaktesting to 1×10^{-3} atm-cm^3/s. Initial and annual testing is performed at a sensitivity sufficient to demonstrate leaktightness.

Each canister is fitted with a catalytic recombiner system. The recombiners serve to continually recombine H_2 and O_2 gases that may be formed by radiolysis. The product of recombination is H_2O. The recombination takes place at low concentrations of H_2 to O_2; thereby, precluding combustion of these gases and consequent heats of combustion that could evolve rapidly. The proper operation of the combiners, as necessary, to satisfy the approval condition pertaining to combustible gas generation must be determined prior to each shipment.

SHIELDING

The applicant has estimated dose rates via volumetric-cylindrical distributed-source (Rockwell-type) calculations using a realistic source term for fission and activation products in the loaded canisters based on ORIGEN-II computer program for the TMI-2 core. Due to low fuel burnup, no neutron sources exist in the fuel debris. Surface (side, top, bottom) and 2 meters from the cask surface dose rates were calculated for the normal condition of transport. In addition, dose rates were calculated at the trunnion location where the steel trunnion in effect replaces (locally) all but the 0.875 inches of the radial lead shield. The results are given in Table 1 below, all satisfying 10 CFR §71.47 and 49 CFR §173.441 limits.

TABLE 1
Normal Conditions, Dose Rates (mrem/hr)

Package Surface					2 Meters from Surface				
Side	Trunnion	Top	Bottom	Limit	Side	Trunnion	Top	Bottom	Limit
37	84	27	33	200	6.3	8.0	6.2	5.6	10

The structural analysis indicated that a small amount of lead slump may occur. The maximum normal condition lead slump was determined by analysis to be less than 1 inch with the slump occurring in a location at least 10 inches axially away from the ends of the canisters with no direct radiation paths developing. Thus, the shielding dose rates are not significantly affected by lead slump.

Only slight changes in shielding geometry occur as the result of hypothetical accident conditions. The flat end drop accident analysis resulted in a maximum lead slump of 1.63 inches. This occurs in a location at least 10 inches from the ends of the canister.

The thermal analysis indicates no appreciable melt of the lead to compromise the shielding effectiveness.

The hypothetical accident condition shielding dose rate was calculated for the localized puncture pin damage to the outer cask side wall determined by quarter-scale model testing. The puncture pin damage is modeled as a void located on the outside of the lead shield, half the thickness of the lead in depth and 6 inches in diameter. Using Rockwell formulation for a cylindrical source, the dose rates 1 meter from the package surface are given in Table 2, satisfying 10 CFR §71.51 limits.

TABLE 2
Hypothetical Accident Conditions, Dose Rate (mrem/hr)
1 Meter from Package Surface

Side	Top	Bottom	Limit
30	37	33	1,000

The source contents in the seven canisters was determined by the computer program ORIGEN-II burnup calculations, for 117 days at full thermal power of 2,186 MW with an average neutron flux of 2.47×10^{14} neut/cm^2-sec. A cooling time of 6 years was assumed. Each canister at maximum loading was calculated to contain 11,000 curies of mixed fission products (4,400 Ci Sr-90; 5000 Ci Cs-137) plus 300 curies of activation products, Co-60.

The radial shielding of the Model No. 125-B cask consists of 3.88 inches of lead sandwiched by a 2-inch outer cask steel cylinder and a 1-inch inner cask steel cylinder preceeded by a 1-inch steel inner vessel shell. For each canister there exists a 0.25-inch steel wall and a 0.375-inch steel inner vessel tube. The top shield in the shielding calculations is taken as 12 inches of stainless steel, the bottom shield is taken as approximately 13 inches stainless steel. In actuality, when the lid is in place, the top shield presents a thickness of over 22 inches of steel to the emergent photons. Similarly, the bottom shield has an additional 10 inches of steel not credited in the shielding analysis. Self absorption of canister contents and the BISCO regions was credited in all the shielding calculations.

The NRC staff has reviewed the shielding data used by the applicant and verified by independent calculations the side cask surface dose rate under normal conditions to be 35 (vs 37) mrem/hr.

CRITICALITY

1. Introduction

The contents of the Model No. 125-B cask are limited to the debris of the TMI-2 core, defueled into canisters by the approved NRC procedures. Three canisters - fuel, knockout, and filter canisters are to accommodate various sizes and forms of the core debris. Seven loaded canisters (any mix of the three types) will be positioned into the Model No. 125-B cask containing debris masses as per the following table:

TABLE 1

Canister type	Maximum payload (wt)	Maximum H_2O (wt) after dewatering
Fuel	1,710 lbs (775 kg)	8 lbs (3.6 kg)
Knockout	1,894 lbs (858 kg)	124 lbs (56 kg)
Filter	1,500 lbs (680 kg)	149 lbs (68 kg)

All criticality analyses have the following conservative assumptions regarding the canister contents. The contents are considered to consist of fresh $U(3)O_2$ fuel pellets mixed with unborated water at the most reactive fuel volume to water volume ratio (i.e., 30/70). The effects of cladding (Zr), structural (SS), core control poison rod material (Cd-In-Ag) or core fixed burnable poison rods (B_4C) are not considered in the analysis; nor is the boric acid in the core water (tech spec at time of defueling: 4,350 ppm natural boron) credited to the water associated with the debris. The B_4C poisons for the knockout and filter canisters were conservatively taken to be at a density of 1.35 gms/cc and the Boral plates of the fuel canister to have an areal density of 0.04 gm/cm^2 for B-10 in the Boral.

2. <u>Most Reactive Mixture of Fuel/Water</u>

Both the applicant and the NRC staff independently established through parametric cell-criticality calculations that the most reactive form of the fuel in water was that of the undamaged TMI-2 pellet with the optimum fuel/water volume mixture in unborated water to be approximately 30/70.

The most reactive form of the fuel in the borated core debris water with 4,350 ppm natural boron is again the undamaged pellet but the optimum V_F/V_W has shifted to approximately 60/40 since one needs more fuel (and pellets closer together) to affect reactivity in a strongly absorbing moderator. The k_{eff} for a pellet at this new optimum in borated water in the range between 2,500 to 4,500 ppm boron will be lower by about 0.20 to 0.30 in units of k_{eff} relative to the unborated case. Thus, the unborated mixture controls. All analyses used the unborated moderator.

Since the above ratios (30/70 and 60/40) represent optimum values, further increase of fuel into the system would decrease reactivity. Thus, small uranium slurry volumes and/or uranium fines in the moderator region give a crude first-approximation of reactivity reduction. This is not exactly correct since introducing fuel in the moderator region shifts the optimum value. This has been neglected and is considered a second order effect on the assumption the system spectrum remains constant and the shift is small.

3. Criticality Analyses and Results

Both the applicant and the NRC staff established the criticality safety of the Model No. 125-B cask for Fissile Class III shipment by calculational methods. Each used the KENO-IV Monte Carlo computer program (NUREG/CR-0200) with the 123-group GAM-Thermos neutron cross-section library using the NITAWL subroutine to adjust the resonance U-238 nuclide cross-sections via the Nordheim-Integral treatment. This calculational approach has successfully calculated many low-enriched UO_2-water criticals, with and without distributed and discrete poisons.

The applicant then performed a 1-D ANISN criticality 2 region cell calculation of the basic $U(3)O_2$-water cell (30/70) generating effective 123-group spatially flux weighted cross-sections for the homogenized fuel-water debris mixture. Using generalized geometry to describe the canister internals, the above homogenized debris mixture occupied all space inside the Boral plates of the fuel canister, all space inside the knockout canister not occupied by the 5 B_4C-SS clad poison rods and all space inside the 17 filter elements and outside the central B_4C-SS clad poison rod of the filter canister. All canisters in the cask have a debris height of 10.5 feet with an inner steel cylinder radius of 6.75 inches wall thickness of 0.625 inches (without the 0.25-inch gap) which includes the thickness of the fixed cask steel tubes into which the canisters reside. The 7 knockout and 7 fuel canisters are represented in quarter-symmetry using generalized geometry for the loaded cask including the BISCO regions, radial steel stiffeners, the 3.88-inch lead shield sandwiched by 1-inch inside and 2-inch outside steel shells. Proper reflective boundary conditions give a full cask. The internal fixed canister poisons are explicitly represented as 5 B_4C-SS clad cylinders in the knockout canister, 4 slabs for the Boral poisons in the fuel canister, and 1 B_4C-SS clad cylinder for the filter canister. The 7 filter canisters were represented in half-symmetry to accurately represent the unrealistic condition of the single B_4C-SS clad poison rod displaced to one side of the filter canister with optimum fuel/water mixture squeezed out of the filters and displaced to the opposite canister wall with the remaining filter steel in the intermediate location in the canister.

Since the filter canister contains by volume approximately 10 times the amount of internal steel as that of the knockout canister, and the contents of the fuel-canister are restricted spatially into a square geometry by the Boral plates, the knockout canister under normal conditions represents the most neutronically reactive canister. Under hypothetical accident conditions, the Boral plates were demonstrated to remain intact in the fuel canister and the central B_4C-SS poison rod of the knockout canister is estimated to displace at most 0.4 inches off center (1.0 inch is assumed in the analysis) while the filter canister internals are distributed as explained above in the previous paragraph.

The applicant's analysis of the single intact knockout canister and fuel canister fully flooded and reflected gave an average keff of 0.845 ± 0.004 for the knockout canister and 0.832 ± 0.004 for the fuel canister. Results are given in Table 2 compared with the staff's calculation. The maximum keff was calculated as 0.917 for the Model No. 125-B cask loaded with 7 knockout canisters under hypothetical accident conditions. This result is given in Table 3 compared with the NRC staff's calculation.

The NRC staff used the KENO-IV geometry option in its description of the single knockout and fuel canisters and in the Model No. 125-B cask loaded with 7 knockout canisters for the hypothetical accident conditions. The NRC staff verified and used the atomic number densities for all nuclides for various regions as those used by the applicant. The core debris (canister contents) was represented in discrete form - a $U(3)O_2$ pellet (0.47 cms radius) surrounded by the cell water (30/70) - the basic cuboid (sq. cross-sec: 1.52 cms x 1.52 cm; ht. 14 ft) debris. No flux weighting was used as in the applicant's approach to generate a homogeneous debris mixture.

372 cuboids out of a total of 400 (20x20) cuboids are fuel-water debris type and occupy the internal space of the knockout canister represented as a square cylinder of internal dimensions of 30.4 cms x 30.4 cms x 426.0 cms. The central 16 cuboids (4x4) - same dimensions as the fuel-water debris cuboid - represent the central B_4C-SS poison rod and its SS cladding. The inside 4 cuboids are B_4C, the 4 corner cuboids of the surrounding external 12 cuboids are all SS; the remaining 8 cuboids are half B_4C - half SS, with SS on the outside. Such a representation gives the exact mass of SS clad as built but the B_4C mass is 9.5% less the actual amount. Accordingly, the number densities for B_4C were increased by 9.5%. For a 14 foot high square cylinder, the mass loading due to the debris gives 1,143 kg UO_2 to which a mass of 255 kgs of water (for an approximate 30/70 mixture) can be added for the hypothetical accident. This mass reduces to 1,143 x 10.5/14.0 = 857 kg UO_2 or 1,886 lbs for payload compared to the maximum 1,894 lbs for a 10.5 ft height as specified in Table 1 - good agreement. The four peripheral B_4C-SS rods are represented by 3 cuboids each giving conservatively 16% less SS than actually surrounding the outside smaller poison rods and about 40% less the B_4C content existing in the rods.

Quarter symmetry was also used by the NRC staff in modeling the Model No. 125-B cask loaded with the above 7 square cylinders in a BISCO* region having the same BISCO mass as built (BISCO NS-3&4 have been taken as BISCO NS-3 only) with the actual SS-PB-SS shield regions surrounding the 7 knockout canisters. This modeling with the KENO-IV program and the 123-group set gave a maximum keff under hypothetical accident conditions and fully reflected to be 0.900 as compared to the applicant's maximum keff of 0.917.

The single knockout and fuel canisters flooded and reflected by water were calculated similarly as above and reported in Table 2.

The agreement between the NRC staff and the applicant keff's provides validation to the homogenization procedure used by the applicant for the fuel-water debris region as compared to the discrete procedure used by the NRC staff for this region.

The NRC staff did not calculate the normal conditions keff for the fully loaded casks based on the nuclear isolation of 2 normal dry casks in contact would experience from 6-inches of SS plus about 8-inches of lead. The applicant as shown in Table 3 calculated a maximum of 0.865 for the system keff for conservative conditions. The maximum permitted water following dewatering was concentrated with the fuel debris for this system reactivity.

The most significant concern for subcriticality is the status of the BISCO materials separating the 7 canisters in the cask. Table 4 gives some typical sensitivity results for various states of the BISCO regions for the cask with 7 knockout canisters under hypothetical accident conditions. Results of Table 4 show that the BISCO should not be diminished by more than 1/2 of its theoretical density.

In addition, the presence of the central B_4C with its associated SS cladding cannot be compromised for the 7 knockout canisters in the cask - regardless of the BISCO state.

The NRC staff has reviewed the applicant's nuclear and geometric data and modeling of the Model No. 125-B cask for the three types of canister loadings for normal and hypothetical accident conditions of transport. The NRC staff found them to be accurate, conservative, and representing of the cases intended. Independent NRC staff analysis and calculations confirmed the applicant's results. The applicant has demonstrated and the NRC staff has confirmed that the cask will be subcritical for normal and hypothetical conditions of 10 CFR Part 71.

*(a) Precast tiles of BISCO NS-4 (84.7 w/o) and B_4C (15.3 w/o).
(b) "Cast in place" BISCO NS-3 with increased hydrogen content and 11 w/o B_4C added.

TABLE 2

SINGLE CANISTER, AVG Keff's*
MOST REACTIVE FUEL/WATER RATIO WITH INF-WATER REFLECTOR

CANISTER	APPLICANT[a]	NRC[b]
Knockout (with 5 B_4C poison Rods	0.845	0.872
Fuel (with side Boral plates)	0.832	0.851
Filter (filters and B_4C rod in place)	0.838	---

*KENO-IV; 123-Gp; 30,000 histories; ±0.005 1 st. dev.
(a) applicant used a height of 10.5 feet for contents in cylinder
(b) NRC staff used a height of 14.0 feet for contents in cylinder

TABLE 3

MAXIMUM Keff's
FOR THE MODEL NO. 125-B CASK AS CALCULATED FOR
FISSILE CLASS III REQUIREMENTS

	APPLICANT	NRC
Normal Conditions		
Two touching casks H_2O reflected; each with 7 knockout canisters with max H_2O after dewatering	0.865*	---
Hypothetical Accident Conditions		
One cask; H_2O reflected; with 7 knockout canisters flooded** with optimum F/W ratio; all central B_4C poison rods shifted 1-inch off center	0.917	0.900

*If boron (from the boric acid in core) remaining in canisters after dewatering is credited to the water, the reported keff would drop to about 0.65.
**Inleakage assumed per 10 CFR §71.55.

TABLE 4

NRC STAFF CALCULATION OF MODEL NO. 125-B CASK keff*
(hypothetical accident conditions - 7 knockout canisters, optimum $V(UO_2)/V(H_2O)$)
AS A FUNCTION OF BISCO STATUS

BISCO Status	Avg Keff
(central B_4C+SS off center, 1-inch)	
Full density BISCO in place	0.888
Half density BISCO in place	0.924
BISCO replaced by water	0.929
BISCO replaced by void	1.072
(central B_4C+ Clad replaced by water)	
Full density BISCO in place	1.002

*KENO-IV; 123-group Gam-Thermos cross-section set; 30,000 neutron histories; all keff to ± 0.004 for 1 st. dev.

OPERATING PROCEDURES

General operating procedures for use of the cask are provided in Section 7.0 of the application. These procedures involve (1) loading, dewatering, and weight and pressure measurements for each canister; (2) dry-loading the prepared canisters into the cask; (3) closure and seal leak testing the inner vessel and cask cavity; (4) securing impact limiters to the cask; (5) and procedures for unloading the cask. Previously used, empty packages, are handled per the requirements of 49 CFR §173.427.

Assembly verification leak testing is called for as part of the Operating Procedures described in Section 7.0 of the application. Appendix 7.4 specifies the assembly verification leaktest. Both have been included as conditions of approval for this cask design.

The assembly verification test described in Section 7.4.1 is based on a pressure rise test with sensitivity of 1×10^{-3} atm-cm^3/s for both inner vessel and cask cavity containment systems. The acceptance criteria specifies that any detected leakage greater than 1×10^{-7} atm-cm^3/s is unacceptable.

ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

Section 8.1 describes the Acceptance Tests to be performed prior to first use of the cask. These tests include (1) visual inspection, (2) structural and pressure tests, (3) leak tests, (4) component tests, (5) tests for shielding integrity, (6) thermal acceptance tests, and (7) lead installation tests.

Section 8.2 describes the Maintenance Program used to ensure continued performance of the cask. This program includes (1) structural and pressure tests, (2) leak tests, (3) subsystem maintenance, and (4) valves, rupture discs, and gaskets on the inner vessel and cask cavity.

This section of the application includes leak tests to be performed before first use and annually on inner vessel and cask cavity (primary and secondary). The test requires sensitivities of $5x10^{-8}$ atm-cm^3/s to assure that the containment criteria is satisfied (i.e., leaktightness - 10^{-7} atm-cm^3/s).

The maintenance program includes annual seal replacement (or sooner if needed) for inner vessel and cask cavity seals.

Section 8.0 has been included as a condition of approval for this cask design.

CONDITIONS

1. The cask cavity and inner vessel must be dry (no free water) when delivered to a carrier for transport. The canisters must be loaded and dewatered in accordance with Section 7.1.1 of the application which includes approximately two atmospheres of argon, nitrogen, or helium cover gas. The cask cavity and inner vessel must be filled with argon, nitrogen, or helium at 1.0 atm pressure.

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

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

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

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

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

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

4. Bolt torque:

   The outer cask lid must be secured by 32, ASTM A320, Grade L43, 1-1/2-6 UNC-2A x 5.5 long bolts torqued to 1,200-1,450 ft-lbs (lubricated).

   The inner vessel lid must be secured by 24, ASTM A320, Grade L43, 3/4-10 UNC-2A x 2.25 long bolts torqued to 200-240 ft-lbs (lubricated).

   The upper and lower overpack limiters must each be secured by 8, ASTM A320, Grade L43, 1-1/4-7 UNC-2A x 41.75 long bolts torqued to 350-415 ft-lbs (lubricated).

5. Prior to each shipment, the licensee must confirm that the cask and inner vessel are properly sealed by tests as specified in Appendix 7.4.

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

CONCLUSION

Based on our review of the statements and representations contained in the application and the conditions listed above, we have concluded that the Model No. 125-B cask design meets the requirements of 10 CFR Part 71.

Charles E. MacDonald

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

Date: APR 1 1 1986

71-9200

Distribution: w/encls
Docket File
NRC PDR
IE HQ
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State Health Official
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CRChappell
CRMarotta
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NMSS R/F
FCTC R/F
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RETURN TO
A. Machlin
396-SS

APR 1 1 1986

FCTC:RHO
71-9200

Department of Energy
ATTN: Dr. Julio L. Torres
DP-4
Washington, DC 20545

Gentlemen:

Enclosed is Certificate of Compliance No. 9200, Revision No. 0, for the Model No. 125-B shipping container.

Department of Energy has been registered as a user of this package under the general license provisions of 49 CFR §173.471.

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

Sincerely,

Original Signed by
CHARLES E. MACDONALD

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

8604170702 860411
PDR ADOCK 07109200
C PDR

Enclosures:
1. Certificate of Compliance
No. 9200, Rev. 0
2. Safety Evaluation Report

cc w/encls
Mr. Wendell Carriker
Department of Transportation

Department of Energy
ATTN: Mr. W. W. Bixby
P.O. Box 88
Middletown, PA 17057

Nuclear Packaging, Inc
ATTN: Mr. Richard T. Haelsig
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OFFICE	FCTC	FCTC	FCTC	FCTC	FCTC	FCTC	FCTC
SURNAME	RHOdegaarden:alm	CRChappell	CRMarotta	WHLake	EPEaston	CEWilliams	CEMacDonald
DATE	04/10/86	04/11/86	04/10/86	04/10/86	04/10/86	04/10/86	04/11/86