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

SAFETY EVALUATION REPORT

Model No. NAC-STC Package
Certificate of Compliance No. 9235
Revision No. 0

SUMMARY

By application dated September 27, 1990, as supplemented, NAC Services Inc. (NAC), formerly Nuclear Assurance Corporation, requested approval of the Model No. NAC-STC package as a Type B(U)F package. Based on the statements and representations in the application, as supplemented, and the conditions listed below, we have concluded that the Model No. NAC-STC package meets the requirements of 10 CFR Part 71.

REFERENCES

Nuclear Assurance Corporation application dated September 27, 1990.

Nuclear Assurance Corporation supplements dated December 23, 1991; August 20, 1992; and August 19, 1993.

NAC Services Inc. supplements dated February 1 and 15, May 18, June 24, July 19, August 10, and September 30, 1994.

PACKAGE DESCRIPTION

A steel, lead and polymer (NS4FR) shielded shipping cask for irradiated pressurized water reactor (PWR) fuel assemblies. The cask body is a right circular cylinder with an impact limiter at each end. The package has approximate dimensions as follows:

Cavity diameter	71 inches
Cavity length	165 inches
Cask body outer diameter	87 inches
Neutron shield outer diameter	99 inches
Lead shield thickness	3.7 inches
Neutron shield thickness	5.5 inches
Impact limiter diameter	124 inches
Package length:	
without impact limiters	193 inches
with impact limiters	257 inches

The approximate weight of the packaging and contents are as follows: cask body and lids 175,970 pounds; fuel basket 16,350 pounds; impact limiters 17,730 pounds; and 26 PWR fuel assemblies 39,650 pounds. The maximum gross weight of the package is 250,000 pounds.

The cask body is made of two concentric shells of Type 304 stainless steel. The inner shell is 1.5 inches thick and has an inside diameter of 71 inches. The outer shell is 2.65 inches thick and has an outside diameter of 86.7 inches. The annulus between the inner and outer shell is filled with lead.

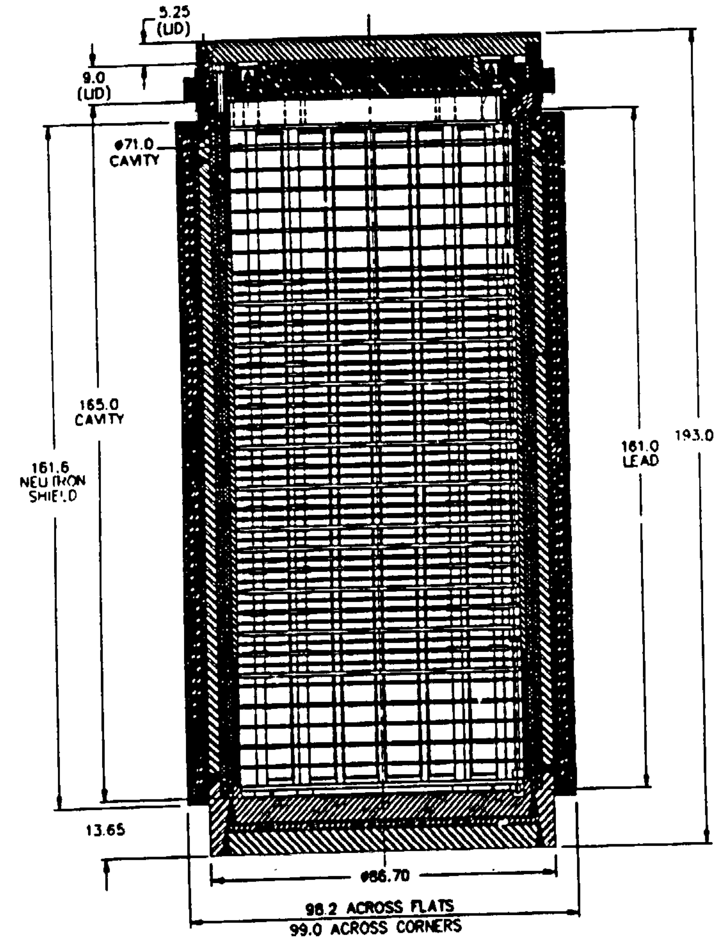
The inner and outer shells are welded to steel forgings at the top and bottom ends of the cask. The cask is closed at the bottom end by two circular plates which are made of Type 304 stainless steel and welded to the bottom end forging. The inner bottom plate is 6.2 inches thick and the outer bottom plate is 5.45 inches thick. The space between the two bottom plates is filled with a 2-inch thick disk of a synthetic polymer (NS4FR) neutron shielding material.

At the top end, the cask is closed by two steel plates (lids) which are bolted to the upper end forging. The inner lid (containment boundary) is 9 inches thick and is made of Type 304 stainless steel. The outer lid is 5.25 inches thick and is made of SA-705 Type 630 stainless steel. The inner lid is fastened to the cask by 42, 1-1/2-inch diameter bolts and the outer lid is fastened by 36, 1-inch diameter bolts. The inner lid is sealed by two metallic O-rings. The outer lid is equipped with a single metallic O-ring. The inner lid is fitted with a vent and drain port. The ports are sealed by metallic O-rings and cover plates. Two non-containment ports are recessed into the top forging. These are used for draining, leak testing, and pressure monitoring during storage.

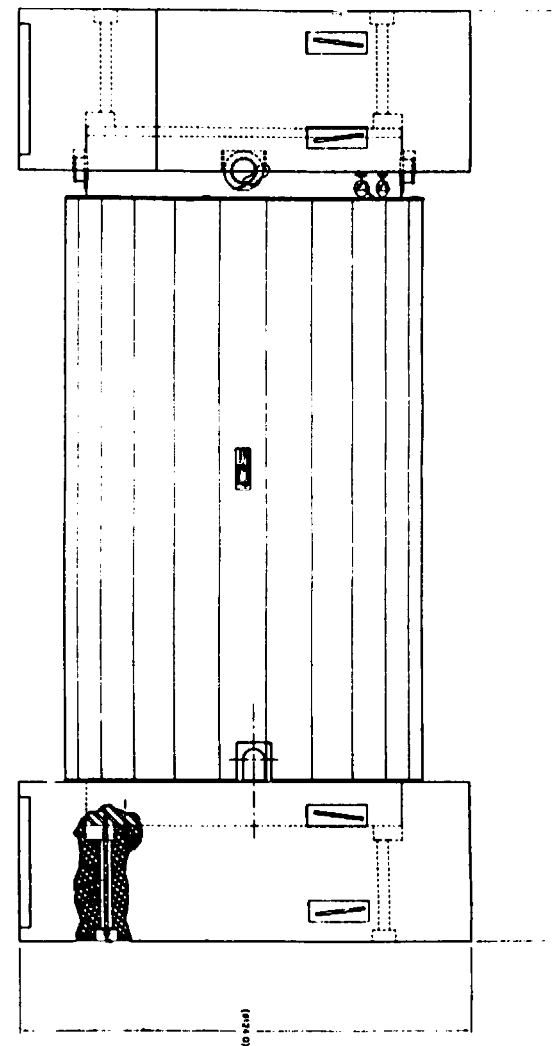
The cask body is surrounded by a 1/4-inch thick jacket shell constructed of 24 stainless steel plates. The outside diameter of the jacket shell is approximately 99 inches. The jacket shell is supported by 24 longitudinal stainless steel fins which are connected to the outer shell of the cask body. Copper plates are bonded to the fins. The space between the fins is filled with NS4FR shielding material.

The package is equipped at each end with impact limiters made of redwood and balsa. Each impact limiter is 124 inches outside diameter by 44 inches deep, and extends 12 inches along the side of the cask body.

The fuel basket within the cask cavity can accommodate up to 26 PWR fuel assemblies. The fuel assemblies are positioned within square sleeves made of stainless steel. BORAL sheets are encased within the walls of the sleeves. The sleeves are laterally supported by 31, 1/2-inch thick, 70.86-inch diameter stainless steel disks. The basket also includes 20 fins made of Type 6061-T6



Internal View



External View

NAC-STC Shipping Package

aluminum alloy. The support disks and fins are connected by six, 1-5/8-inch diameter by 161-inch long threaded rods made of Type 17-4 PH stainless steel.

Four lifting trunnions are welded to the top end forging. The package is shipped in a horizontal position and is supported by a cradle under the top forging and by two trunnion sockets located near the bottom end of the cask.

The package also has a configuration for long term storage of spent PWR fuel. The storage configuration is the same as the transport configuration, except that the impact limiters are not attached and the cask sits in a vertical position.

DRAWINGS

The package is constructed and assembled in accordance with the following Nuclear Assurance Corporation Drawing Nos.:

423-800,	sheets 1-2,	Rev. 3	423-811,	sheets 1-2,	Rev. 4
423-802,	sheets 1-6,	Rev. 6	423-812,		Rev. 0
423-803,		Rev. 1	423-870,		Rev. 2
423-804,	sheets 1-3,	Rev. 2	423-871,		Rev. 1
423-805,		Rev. 1	423-872,		Rev. 3
423-806,		Rev. 1	423-873,		Rev. 1
423-807,	sheets 1-2,	Rev. 0	423-874,		Rev. 1
423-809,	sheets 1-2,	Rev. 1	423-875,		Rev. 1
423-810,	sheets 1-2,	Rev. 1	423-900,		Rev. 2

CONTENTS AND FISSILE CLASS

(1) Type and Form of Material

Irradiated PWR fuel assemblies with solid UO_2 pellets. Each fuel assembly may have a maximum burnup of 40,000 MWD/MTU when cooled for at least 6.5 years, or 45,000 MWD/MTU when cooled for at least 10 years. The maximum heat load per package is 22.1 kilowatts. The maximum heat load per assembly is 850 watts. Prior to irradiation, the fuel assemblies must be within the dimensions and specifications shown in Table 1, below.

(2) Maximum Quantity of Material per Package

Twenty six (26) PWR fuel assemblies

(3) Fissile Class I

TABLE 1
Fuel Assembly Dimensions and Specifications

Assembly Type	14x14	15x15	16x16	17x17	17x17 (OFA)
Cladding Material	Zirc-4	Zirc-4	Zirc-4	Zirc-4	Zirc-4
Maximum Initial Uranium Content (kg/assembly)	407	469	426	464	426
Maximum Initial Enrichment (wt% ^{235}U)	4.2	4.2	4.2	4.2	4.1
Assembly Cross-Section (in)	7.76 to 8.11	8.20 to 8.54	8.10 to 8.14	8.43 to 8.54	8.43
Number of Fuel Rods per Assembly	176 to 179	204 to 216	236	264	264
Fuel Rod OD (in)	0.422 to 0.440	0.418 to 0.430	0.382	0.374 to 0.379	0.360
Minimum Cladding Thickness (in)	0.023	0.024	0.025	0.023	0.023
Pellet Diameter	0.344 to 0.377	0.358 to 0.390	0.325	0.3225 to 0.3232	0.3088
Maximum Active Fuel Length (in)	145.20	144	150	144	144

STRUCTURAL EVALUATION

The applicant performed various structural analyses, engineering evaluations, and physical tests to demonstrate that the package has adequate structural integrity to meet the requirements of 10 CFR Part 71. The analyses considered load combinations from Regulatory Guide 7.8, "Load Combinations for the Structural Analysis of Shipping Casks." The analyses show 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 also performed engineering evaluations to show that the package is not subject to brittle fracture or buckling under the test conditions specified in 10 CFR Part 71.

The applicant conducted a series of physical tests to support and confirm the results of the engineering analyses. A quarter-scale model of the packaging was subjected to 30-foot drop and 40-inch puncture tests. Crush tests were performed on eighth-scale models of the impact limiters to measure their force-deflection characteristics. The results of the physical tests were consistent with the engineering analyses of the package design.

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.

General Standards for All Packages

Minimum Package Size

No dimension of the package is less than four inches. The package meets the requirement of 10 CFR §71.43(a) for minimum size.

Tamper-Indicating Feature

It is necessary to remove the upper impact limiter to gain access to the package closure assembly. As a tamper indicator, a numbered, crimped wire seal is looped through a hole in the flange of one of the lifting trunnions and a hole in an adjacent corner of the upper impact limiter. An intact seal is evidence that the package has not been opened by unauthorized persons. The non-containment ports in the top forging are protected by port covers. The bolts for the port covers are drilled for installation of a numbered wire seal. These seals satisfy the tamper-indication requirement of 10 CFR §71.43(b).

Positive Closure

The package and its containment system are positively closed by bolted lids and port cover plates, and can not be opened unintentionally.

Chemical and Galvanic Reactions

The materials of construction are such that there will be no significant chemical, galvanic or other reaction between package components or between package components and package contents.

Valves or Other Devices

All ports are protected against unauthorized operation and are sealed by a cover plate and O-ring seals to retain any leakage. Thus, the package design meets the requirements of 10 CFR §71.43(e).

Continuous Venting

The package does not incorporate any feature which would allow venting during transport.

Lifting and Tie-Down Standards for All Packages

Lifting Devices

Four lifting trunnions are spaced at 90-degree intervals on the top end forging of the cask. Two lifting trunnions can support three times the weight of the package without yielding, as required by 10 CFR §71.45(a). Also, the lifting trunnions were analyzed for an overload condition to show that failure of the trunnions would not impair the ability of the cask to meet its other safety requirements.

Tie-Down Devices

The package is secured to the transport vehicle by two rear supports and two trunnion sockets near the bottom of the cask, and by two saddle supports, a hold-down strap, and a shear ring near the top of the cask. The two rear supports at the trunnion sockets resist vertical and lateral tie-down forces and the longitudinal forces which act toward the bottom end of the package. At the front of the package, the shear ring is welded to the outer shell of the cask and resists longitudinal forces which act toward the top end. The saddle supports resist the lateral force, and the hold-down strap resists the vertical force.

The tie-down devices that are a structural part of the package are the shear ring and the two trunnion sockets. The applicant performed an analysis to show that the package meets the tie-down device requirements in 10 CFR §71.45(b). Each tie-down device which is a structural part of a package is designed so that the failure of the device under excessive load would not impair the ability of the package to meet the other requirements of 10 CFR Part 71. Both the shear ring and the trunnion sockets are welded to the outer shell of the cask. Under excessive overload, the welds will fail in shear before the cask body suffers any appreciable damage. Therefore, failure of the shear ring or the trunnion sockets would not impair the ability of the package to meet the other requirements of 10 CFR Part 71.

For rail transport, the Certificate of Compliance has been conditioned to require approval by the Association of American Railroads of the railcar and the components of the tie-down system that are not a structural part of the package. For marine or barge transport, the National Cargo Bureau, Inc., must certify that package stowage is in accordance with U.S. Coast Guard regulations.

Normal Conditions of Transport

Heat

The ANSYS computer program was used to calculate the stresses that would result from the normal condition heat test (i.e., 100 °F ambient temperature, maximum heat load, solar insolation, and an internal pressure of 50 psig). The analysis considered differential thermal expansion of the cask components. The closure bolts were modeled as beam elements and the bolt pre-loads were applied to the model by imposing an initial strain to the beam elements. The analysis shows that the stresses in the cask are below the allowable values specified in Regulatory Guide 7.6, and that the normal condition heat test will not adversely affect the package.

Cold

The stresses in the package were calculated for an ambient temperature of -40 °F in still air and shade, and without decay heat. The analysis was performed using the ANSYS computer code and was similar to the analysis performed for the heat test above, except that an internal pressure of 12 psig was used for the cold condition. The analysis considered thermal contraction of the lead. The analysis shows that the stresses in the cask are within allowable values, and the cold condition test will not adversely affect the structural performance of the package.

Reduced and Increased External Pressure

A decrease in external pressure to 3.5 psig, or an increase in external pressure to 20 psig, will have no significant effect on the package. Small changes in internal or external pressure would not cause significant stresses in either the inner or outer shell, or in the end closures of the cask.

Vibration

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

Water Spray

All exterior surfaces of the package are stainless steel. The water spray test will not affect the package.

One-Foot 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 normal condition 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 top end, top corner, side, bottom corner and bottom end. The maximum g-loads and impact limiter deformations for the one-foot drop test were determined to be as follows:

TABLE 2

One-Foot Drop Test

Orientation	Maximum Load (g's)	Maximum Impact Limiter Deformation (in.)
Top End	19.6	2.1
Top Corner	5.3	9.7
Side	18.1	1.7
Bottom Corner	5.3	9.6
Bottom End	19.6	2.1

The applicant performed a linear elastic, quasi-static analysis of the package using the ANSYS computer code. Both two-dimensional and three-dimensional finite element models were used to represent the cask. The impact load was considered to be 20 g's for each package orientation. The impact stresses were combined with other individual loads such as bolt pre-load, internal pressure loads, and normal condition heat loads. The results show that the stresses are within the allowable limits specified in Regulatory Guide 7.6 for normal conditions of transport.

Corner Drop

The corner drop test is not applicable because the package weight exceeds 220 pounds and neither wood nor fiberboard are used as materials of construction.

Compression

The compression test is not applicable because the weight of the package exceeds 11,000 pounds.

Penetration

The exterior shells and 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 impacted by the 13 pound steel bar used in the test.

Hypothetical Accident Conditions

Scale Model Tests

The applicant's evaluation of the package under hypothetical accident conditions included both engineering analyses and physical testing of a quarter-scale model of the design. The quarter-scale model was an exact replica of a full-scale packaging with the following exceptions: (1) the O-rings used in the inner and outer lids were not scaled, (2) the outer lid was bolted to the inner lid while in the final package design each lid bolts directly to the top forging, (3) the shells were made entirely of Type 304 stainless steel and did not include the Type XM-19 stainless steel transitions, (4) the neutron shield tank was not modeled, but its weight was represented by steel blocks welded to the outer shell, and (5) the fuel basket was similar, but not identical, to the final basket design. The weight of the 26 fuel assemblies was represented by steel bars. The overall weight of the cask model, and the weight of its basket and simulated fuel assemblies, were appropriate for a quarter-scale model.

Multiple 30-foot drop and 40-inch puncture tests were conducted using a single scale model specimen. During the course of testing, the impact limiters failed to perform as intended during the first tests conducted in the corner and side drop orientations. In the case of the side drop test, there was spalling of the impact limiter, and the weights attached to the cask model made contact with the test pad. This caused distortion of the closure and the cask did not maintain internal pressure. Also, the inner and outer shells of the model cask were indented in the region where the weights were attached. Following this result, the impact limiters were re-designed, the model was refurbished, and additional drop tests were performed.

A total of nine 30-foot drop tests were conducted as follows: end (1 test), corner (2 tests), side (3 tests), and oblique (3 tests). Because design changes were made to the impact limiters during the course of the test program, only four of the tests are fully applicable for evaluating the performance of the final impact limiter design. The results of these four tests are summarized in Table 3, below.

The results of the applicable quarter-scale model drop tests confirmed that the impact limiters will remain attached to the cask and that they will perform their intended function to absorb impact energy during the 30-foot drop test. Except for the first side drop test, discussed above, the quarter-scale cask model experienced no significant damage during the 30-foot drop tests. The cask shells did not buckle and no damage was observed in the cask lids, lid bolts, or other components. Measurements before and after the tests indicate that there was no loss of pressure and the closure lid system performed satisfactorily.

The applicant also performed 40-inch puncture tests on the scale model packaging. A puncture test was conducted with the cask in a horizontal position so that the pin impinged on the outer shell. A second puncture test was conducted with the cask in a vertical position so that the pin impinged on the outer closure lid. The vertical test was conservatively performed with the impact limiters removed so that the pin would strike directly against the closure plate. In both of these initial tests, the test pin was too long and failed by bending. Both puncture tests were repeated using a shorter length pin. The results of this second set of tests showed that the cask was adequately designed for the puncture test. The cask shell was not punctured, the outer closure lid did not collapse, and the cask maintained internal pressure. The only damage was a small indentation on the outer shell where the pin had impinged against the cask.

The applicant performed quasi-static crush tests on one-eighth scale models of the impact limiters to measure their force-deformation relationships. Tests were performed for the end, side, and corner impact orientations. The results of the one-eighth scale tests are summarized in Table 3, below.

30-Foot Drop

The package is equipped with crushable impact limiters made of redwood and balsa. The applicant performed a number of crush tests on redwood samples to determine the stress-strain relationship of the material. The wood crush strengths were adjusted to account for hot (+230 °F) and cold (-20 °F) temperatures, and to allow for construction variations. The adjusted wood crush strengths were then used in the RBCUBED computer program to calculate the force-deformation characteristics of the impact limiters, and the g-loads that would be applied to the package. For the purpose of designing the impact limiters, the applicant conservatively assumed that all the available kinetic energy would be dissipated by impact limiter deformation. The applicant conducted static crush tests on one-eighth scale models of the impact limiters to verify the calculated force-deformation characteristics. The applicant also measured the impact decelerations experienced by the quarter-scale models in the 30-foot drop tests. The results of the calculations and tests were as follows:

TABLE 3

30-Foot Drop Decelerations (g's)

Drop Orientation (angle from vertical)	Calculated (Hot)	Calculated (Cold)	Quasi-Static Test (1/8-Scale Model)	Drop Test (1/4-Scale Model)	Cask Design Value
End (0°)	44.6	56.1	54.8	55.6	56.1
Corner (24°)	44.0	49.3	32.6	29.2	55.0
Oblique (75°)	29.9	24.0	-	53.8*	55.0
Side (90°)	51.7	51.3	45.6	51.3	55.0

* Measured value for secondary impact ("slapdown").

To determine stresses in the package, the applicant performed a linear-elastic, quasi-static analysis using the ANSYS computer code. For end drop orientations, the cask was represented by a two dimensional, axisymmetric, finite element model. A three dimensional model was used for other impact orientations. The package was evaluated for end, corner, side and oblique orientations. The g-loads due to secondary impact (i.e., "slapdown") are less than the g-loads for which the cask is designed. Impact stresses were combined with stresses produced by other loads such as internal pressure, temperature, and closure bolt pre-load stresses. The combined stresses were within the allowable values specified in Regulatory Guide 7.6. Stresses in the containment vessel closure bolts were less than the yield strength of the material. The cask shells were evaluated using ASME Code Case N-284, and shown not to buckle under 30-foot drop test conditions. Calculations were performed to show that the impact limiters would remain attached to the cask body following the drop test. The applicant performed a separate analysis to show that lead slump would be less than the 1.73 inch value used in the shielding evaluation.

The results of the ANSYS analysis are consistent with the results of the quarter-scale model tests. In the model tests, no permanent deformation of the cask shells was observed, the shells did not buckle, the impact limiters remained attached, and there was no loss of internal pressure.

The applicant's finite element model used a modulus of elasticity for lead which the staff considered to be too high. Also, the applicant's evaluation of combined stresses due to impact and cold temperature (-20 °F) included the maximum decay heat, and did not consider the case where the temperature would be -20 °F throughout the cask. The staff performed a bounding type calculation to evaluate the package for these conditions. In the analysis, the weight of the lead was considered to be entirely supported by the inner and outer shells for both end and side impact orientations (i.e., the strength of the lead was neglected). The impact stresses were enveloped by combining the maximum axial and hoop stresses from the 30-foot end drop with the maximum bending stress from the 30-foot side drop (even though these would not occur at the same axial location). The analysis considered the stresses due to lateral pressure from slumping of the lead. The impact stress was then combined with the maximum stress due to a cold temperature of -40 °F. The resulting stress was within the allowable values specified in Regulatory Guide 7.6 for accident conditions, and in ASME Code Case N-284 for shell buckling.

The fuel basket has 31 circular stainless steel disks which position and support 26 square fuel tubes. The circular steel disks are equally spaced along the length of the inner cavity. The disks are connected by means of six threaded stainless steel rods and spacer nuts. The basket also includes 20 circular heat transfer fins. The heat transfer fins are made of aluminum and are not considered to be structural members.

To evaluate the circular steel disks, the applicant performed linear-elastic, quasi-static analyses using the ANSYS computer code. The stainless steel connecting rods were analyzed separately as beam-columns. Analyses were performed for end and side drop orientations of the cask. For the side drop case, the analysis considered nine different angular orientations of the basket about its longitudinal axis. To envelop cask drop orientations other than end and side, stresses from the end drop analysis were combined with the maximum stresses from the side drop analyses (considering all nine angular orientations about the longitudinal axis of the basket). The stresses in the circular disks, and in the connecting rods, were less than the allowable values specified in Subsection NG of the ASME Code, and met the buckling criteria in Subsection NF of the ASME Code.

The fuel tubes are constructed of thin gauge stainless steel plates which enclose BORAL plates. The BORAL is not a structural material. The clear span distance between stainless steel support disks is 4.37 inches (neglecting the intervening aluminum heat transfer fins). The applicant performed a large deflection, plastic analysis of the fuel tubes using the ANSYS computer code. The analysis considered two cases for the fuel load distribution: (1) a uniformly distributed load, and (2) a concentrated load applied through a fuel grid spacer located at mid-span (conservatively neglecting possible support from the intervening aluminum heat transfer fins). The analysis showed that the stainless steel components of the fuel sleeve are adequate to support the inertial load of the fuel across the 4.37 inch span.

Puncture

The package was analyzed for the 40-inch puncture test. Four orientations were evaluated; the puncture pin was considered to impinge directly on : (1) the mid-section of the cask side, (2) the center of the lid, (3) the center of the bottom end plate, and (4) the port covers.

The Nelms equation was used to show that the outer shell of the cask is sufficiently thick to prevent puncture if the steel pin impinges directly against the side of the package. The evaluation conservatively neglected the neutron shield jacket shell, the longitudinal fins within the neutron shield tank, and the NS4FR neutron shield material. Stresses due to overall bending of the cask were evaluated and shown to be bounded by the 30-ft side drop.

The cask has inner and outer plates at both the top and bottom ends. The applicant performed a finite element analysis to evaluate puncture pin impingement on the top closure lid. Closed-form calculations were performed to analyze pin impingement on the bottom end of the cask. A separate, two-dimensional, axisymmetric model was used to represent the inner and outer plates of the top closure. The puncture pin was considered to apply a pressure of 47,000 psi over a 6-inch diameter region at the center of the outer plates. The analysis at each end assumed that loads would be transferred through the NS4FR shielding material, and that the total load would be resisted by both the inner and outer plates. The presence of the impact limiters was neglected. The results showed that the maximum stresses were less than allowable values.

The staff does not believe that NS4FR should be relied upon as a structural material to transfer loads. The staff performed a separate bounding type analysis to evaluate puncture on the ends of the cask. The staff's analysis only considered the outer plate at each end of the cask (i.e., loads were not assumed to be transferred from the outer plate to the inner plate). The presence of the impact limiter was neglected. An elastic analysis was performed for both ends of the cask. The puncture pin was considered to apply a pressure of 47,000 psi over a 6-inch diameter region at the center of each end plate. The results showed that the stresses in the plates would be within the allowable limits specified in Regulatory Guide 7.6.

Two ports which provide access to the interlid region are located on the top forging. The valves for these non-containment ports are recessed within 2.93-inch diameter openings in the top forging, and are not subject to being struck by the 6-inch diameter puncture pin.

Puncture tests were performed on the quarter-scale model (see discussion, above). In the tests, the cask shell was not punctured, the outer closure lid did not collapse, and the cask maintained internal pressure. The only damage was a small indentation on the outer shell where the pin had impinged against the cask.

Fire

The applicant demonstrated that the package has adequate structural integrity to withstand the half-hour fire test. The lead shielding is not bonded to the inner and outer shells, and the temperature difference between the inner and outer shells does not exceed 181 °F. Consequently, differential thermal expansion will not produce large stresses. The maximum calculated internal pressure during the fire test was 65.5 psig. This pressure would produce relatively small stresses in the containment vessel shell.

The applicant preformed an ANSYS finite element analysis to evaluate the stresses in the package under the half-hour fire test. The analysis considered cask temperatures, differential thermal expansion and internal pressure. The results showed that the stresses in the packaging would be within allowable values. The staff agrees with the applicant's conclusion that the package has adequate structural integrity to withstand the 30-minute fire test.

Immersion-Fissile Material

The criticality analysis considered an infinite array of undamaged packages with optimum internal and external moderation (i.e., optimally flooded). The staff concluded that there would be no significant difference in the reactivity of a damaged array and an undamaged array of packages when the packages are flooded with water. Since the criticality safety of an array of damaged packages does not depend upon exclusion of water from the cask cavity, evaluation of the package for the three-foot immersion test was not necessary.

Immersion-All Packages

An external pressure of 21 psig, equivalent to immersion under 50 feet of water, would have no significant effect on the package. The stresses in the cask shells would be within acceptable limits, and the shells would not buckle.

THERMAL EVALUATION

The package is designed for 26 PWR spent fuel assemblies, each with a maximum decay heat load of 0.85 kilowatts, and a maximum package decay heat load of 22.1 kilowatts. The package uses a passive cooling system in which the primary coolant is helium gas. The heat transfer performance of the basket is enhanced by 20 aluminum fins located between the stainless steel disks that support the basket. The stainless steel disks and aluminum fins were sized to facilitate heat transfer and to preclude interference due to differential thermal expansion. To provide a path for heat transfer across the neutron shield tank, 24 longitudinal stainless steel fins, with copper backing, are used to connect the outer shell to the jacket shell.

The applicant performed a thermal analysis to show that the package is adequately designed for normal conditions of transport and hypothetical accident conditions. The thermal evaluations for both normal and hypothetical accident conditions considered the inner cavity to be filled with air, and do not rely on the presence of helium for heat transfer. The thermal design of the package conservatively does not rely on the neutron shield material (NS4FR) to transfer heat.

<u>Normal Conditions</u>

The applicant performed three steady-state calculations under normal conditions of transport. The steady-state calculations combined the maximum decay heat load together with:

An ambient temperature of 100 °F, with solar insolation.
An ambient temperature of -40 °F, without solar insolation.
An ambient temperature of -20 °F, without solar insolation.

Three models were used to determine normal condition temperatures. The first was a three-dimensional, quarter-symmetry model that was used to calculate temperatures at the ends of the package. The top and bottom surfaces of the package were modeled as adiabatic surfaces to represent the impact limiters. The neutron shield material and longitudinal fins were represented by an effective heat transfer coefficient. The fuel tubes were modeled as a single material with an effective heat transfer coefficient. The fuel assemblies were modeled as volumetric heat sources. Convection in the cask cavity was conservatively ignored.

The second thermal model was a three-dimensional half-symmetry representation of the package that was used to calculate temperatures in the cask walls. The model consisted of an axial segment of the package sufficient in length to include a stainless steel support disk and an aluminum heat transfer fin. The top and bottom surfaces of the package were modeled as adiabatic.

The third model, a two-dimensional representation of a fuel assembly, was used to calculate the effective heat transfer characteristics of the fuel assemblies and the maximum cladding temperatures.

Hypothetical Accident Conditions

The applicant performed a transient thermal analysis to evaluate the package under hypothetical accident conditions. The model represented a cylindrical sector of the package which subtended one radian of arc. The analysis considered a total decay heat load of 26 kilowatts and assumed the impact limiters remain in place following the drop tests. The applicant's thermal analysis conservatively assumed that the neutron shield material would be consumed during the fire. Following the fire, conduction through the neutron shield was represented by conduction through air and the fins.

The transient thermal analysis considered a nine-hour period, beginning at the start of the fire test, and ending when cask temperatures had reached their peak and began to decline. The analysis determined the temperatures in the cask wall and neutron shield tank. The temperatures of the basket and fuel were approximated by using the temperature differentials from the normal condition analysis.

Maximum Temperatures

The maximum temperatures calculated by the applicant are given below. For both normal and accident conditions, the inner cavity was considered to be filled with air.

TABLE 4

Maximum Calculated Temperature (°F)

Location	Normal Conditions* (steady state)	Accident Conditions (peak transient)
Outer Surface	241	1344
Neutron Shield (NS4FR)	284	n/a
Inner Lid Metallic O-rings	190	293
Outer Shell	292	680
Lead Shielding	314	483
Inner Shell	331	499
Aluminum Fin	491	660
Steel Support Disk	498	667
Fuel	588	756

* The normal condition temperatures are for the neutron shield cavity filled with NS4FR material.

The temperatures under both normal and hypothetical accident conditions do not exceed the allowable service temperature of package components. The lead reaches a maximum temperature of 483 °F, which is below the 620 °F melting point of lead. The maximum temperature of the containment boundary O-ring seals is 293 °F, which is below the allowable service temperature of 500 °F, for metallic O-rings.

The package is shipped on an exclusive use vehicle. A personnel barrier excludes access to the cask surface between the impact limiters. The maximum temperature at the personnel barrier, or on the impact limiter surface, is less than 180 °F.

The NRC staff performed confirmatory calculations using the HEATING Version 7.2 computer code, which is part of the SCALE system. The temperatures calculated by the staff agree with those calculated by the applicant. The staff's calculations confirm the applicant's conclusion that the temperature of safety-related components remain within safe operating ranges.

The results of the applicant's analyses demonstrate that the package is adequately designed for the thermal conditions specified in 10 CFR Part 71.

Minimum Temperatures

Cask components, including the containment system seals, would not be adversely affected by temperatures of -40 °F.

Maximum Pressure

The applicant calculated the maximum normal operating pressure (MNOP) assuming that 100% of the fuel rods fail, and 30% of the gaseous fission products are available for release. The total gas volume considered the gaseous fission products, the helium fill gas in the rods and the cavity backfill gas. The gaseous fission products were based on a fuel burnup of 45,000 MWD/MTU.

The average gas temperature was calculated to be 411 °F. A gas temperature of 450 °F was used to calculate the MNOP of 45.3 psig. The maximum pressure under hypothetical accident conditions is 65.5 psig, based on the maximum fuel cladding temperature of 756 °F.

Thermal Acceptance Tests and Maintenance Program

Prior to first use, each packaging will undergo a thermal acceptance test to verify that its heat rejection capabilities are consistent with the thermal analysis. The thermal acceptance test will compare measured temperatures and gradients with the values calculated for normal conditions of transport. The applicant established specific acceptance criteria for measured temperatures and gradients within the package.

Each packaging will also undergo a periodic thermal test at five year intervals. Temperature will be measured on the upper and lower forging, and at three axial locations on eight of the longitudinal fins. Specific acceptance criteria were established for the ratio of various measured temperatures.

CONTAINMENT EVALUATION

The containment boundary of the package consists of the following components: (1) the inner shell; (2) the inner bottom plate; (3) the upper end forging; (4) the inner lid; (5) the vent port cover plate; (6) the drain port cover plate; (7) the outer, metallic O-ring on the inner lid and on the vent and drain port cover plates; and (8) the interseal test port plug on the inner lid and on the vent and drain port cover plates. The test port plugs are threaded and equipped with a metallic O-ring. The application specifies provisions for leak testing the outer O-rings and the test port plugs.

The inner lid and the vent and drain port cover plates are equipped with dual O-ring seals. The outer O-rings, rather than the inner O-rings, are the containment boundary because the inner O-rings cannot be easily leak tested after long-term storage. By designating the outer O-rings as the containment boundary, the presence of helium inside the cask cavity is not required to perform an acceptable leak test.

The maximum package contents are 26 irradiated PWR fuel assemblies. The sources of radioactive release are fission gases (primarily ^{85}Kr) and crud (primarily ^{60}Co). The bounding case for the containment analysis was the Westinghouse 17x17 fuel assembly with a burnup of 40,000 MWD/MTU, a cool time of 6.5 years, and an enrichment of 3.7 wt%. To determine the fission gas inventory, the applicant used the SAS2H computer program in the ORNL SCALE system. For the crud activity, the applicant assumed that the ^{60}Co concentration on the fuel assembly surface was 140 $\mu Ci/cm^2$. Available data indicate that this is a conservative estimate for crud activity on PWR assemblies at initial discharge.

The maximum allowable leak rate for the package was calculated using the method described in ANSI N14.5-1987. To determine the maximum allowable leak rate under normal conditions of transport, the applicant assumed that 3% of the fuel rods fail, 30% of the fission gas is releasable, and 15% of the crud is releasable. The maximum allowable leak rate for normal conditions of transport (450 °F, 1.79 atm) was calculated to be 1.34×10^{-4} cm^3/sec. Correlated to standard conditions (77 °F, 1 atm), this leak rate is equivalent to 5.79×10^{-5} std-cm^3/sec, assuming that air is the tracer gas, or to 2.20×10^{-5} std-cm^3/sec, assuming that helium is the tracer gas.

Results of the structural and thermal evaluations show that the hypothetical accident conditions do not affect the integrity of the containment boundary. To determine the maximum allowable leak rate under hypothetical accident conditions, the applicant assumed that 100% of the fuel rods fail, 30% of the fission gas is releasable, and 100% of the crud is releasable. The maximum allowable leak rate for hypothetical accident conditions was calculated to be 0.059 cm^3/sec, which is greater than the allowable leak rate for normal conditions of transport. Therefore, the leak test acceptance criteria are based on the standard leak rates calculated for normal conditions of transport.

A fabrication verification leak test will be performed before the first use (shipment) of the package. Periodic leak tests will be performed immediately after each loading, and again within 12 months before shipment if the time between loading and shipment exceeds 12 months. Also, a leak test will be performed on any containment component that has been replaced. Seal replacement is required after each use. All leak tests will have the same sensitivity and acceptance criteria. The outer O-rings will be helium leak tested to 2.20×10^{-5} std-cm^3/sec. The test port plugs will be leak tested to 5.79×10^{-5} std-cm^3/sec, using a test bell and the vacuum air pressure rise method. Leak tests procedures are specified in the operating procedures, acceptance tests, and maintenance program.

The staff reviewed the applicant's containment evaluation and performed an independent confirmatory analysis. The staff agrees with the applicant's assumptions, and has verified the applicant's calculations. The staff concludes that the leak test methods and acceptance criteria specified in the application are adequate, and that the package meets the containment requirements of 10 CFR §71.51.

SHIELDING EVALUATION

The applicant performed an analysis to show that the package meets the shielding requirements of 10 CFR Part 71.

The applicant used the SAS2H sequence and the ORIGEN-S computer code to determine the radiation source terms. Source term calculations were performed for PWR assemblies with a burnup of 40,000 MWD/MTU and decay times of both 5 and 6.5 years, and a burnup of 45,000 MWD/MTU with a 10-year decay time. A conservative design-basis source term was developed as a composite of the Westinghouse 17x17 and 15x15 fuel assemblies with an enrichment of 3.7 w/o and a maximum burnup of 40,000 MWD/MTU. The source term used in the shielding analysis was conservatively based on a 5-year cooling time.

The XSDRNPM/XSDOSE, QAD-CGGP and MORSE computer codes, from the SCALE system, were used to calculate the dose rate at the surface of the package and at two meters from the vehicle. The XSDRNPM and XSDOSE codes were used to calculate radial dose rates along the midplane of the fuel. The flux peaking and heterogeneous basket correction factors used in the XSDRNPM and XSDOSE calculations were calculated by QAD-CGGP. The applicant used MORSE to perform a three-dimensional calculation of the dose rates in the transition regions, including the areas around the port cover plates and trunnion cutouts.

The dose rate calculations were based on a basket with 27 aluminum support disks and 26 borated aluminum fuel tubes. The applicant showed that this is a conservative representation of the basket which contains 33 stainless steel support disks, 20 aluminum heat transfer fins and 26 fuel tubes constructed of BORAL sheet enclosed in stainless steel tubes.

The one-dimensional radial model consisted of infinitely long concentric cylinders representing the package at the fuel midplane. The model considered the fuel and basket tubes to be homogenized. The basket region between the fuel and the inner shell was also homogenized, with the aluminum mass conserved. Since the neutron shield is not exactly cylindrical, it was modeled such that the volume of the neutron shield is conserved. This model was used to calculate radial dose rates at fuel midplane, under both normal and hypothetical accident conditions.

Three-dimensional models of the package were used to determine the dose rates at locations above and below the fuel midplane and to determine the heterogeneous-to-homogeneous ratios to correct the one-dimensional calculations for the basket. The aluminum disks between the fuel and the inner shell were modeled explicitly, but the basket webbing between the fuel assemblies was homogenized.

The applicant used two axial models to perform the shielding analysis for the package. They are one-dimensional models of the top and bottom of the package through the centerline.

The NRC staff performed calculations to confirm the applicant's results. The staff used the SAS2H sequence in the SCALE system to generate the radiation source terms. The shielding calculations utilized this source term with the SAS1 sequence from SCALE. Dose rates were calculated by the staff for both normal and hypothetical accident conditions.

The staff used two one-dimensional models to assess the dose rates. The first is an axial model used to estimate dose rates at the top of the package. The model represented the fuel as a homogenized region and included the spacing, but not the mass, of the impact limiters. The second model was a radial representation used to calculate the dose rates at the personnel barrier and at two meters from the edge of the vehicle. The staff modeled the fuel as a homogenized region, ignoring the effects of the basket disks and the BORAL tubes.

The staff performed shielding calculations to consider loss of neutron shielding under hypothetical accident conditions. The applicant's dose rate calculations are summarized in Table 5, below.

TABLE 5

Maximum Dose Rates (millirem/hr)

Location	Calculated	Limit
Normal Conditions -		
Surface at:		
Side	341.6	1000
Top End	0.5	200
Bottom End	3.2	200
Two Meters from:		
Vehicle Side	8.4	10
Top End	0.2	10
Bottom End	1.3	10
Surface of Personnel Barrier	43.8	200
Hypothetical Accident Conditions -		
One Meter from Cask Surface	320	1000

The dose rates calculated by the staff were consistent with those reported in the application. The dose rates for the package will be within the limits specified in 10 CFR §§71.47 and 71.51. The applicant's shielding analyses, as confirmed by the staff, demonstrate that the package design meets the requirements of 10 CFR Part 71.

CRITICALITY EVALUATION

The applicant performed a criticality analysis which shows that the package remains subcritical under normal conditions of transport and under hypothetical accident conditions. The analysis shows that the package meets the requirements of 10 CFR Part 71 for a Fissile Class I package.

The package is used to transport irradiated, PWR fuel assemblies. The assemblies consist of solid UO_2 pellets in fuel rods cladded with zircaloy. The uranium has a maximum initial enrichment of 4.2 wt% ^{235}U. The zircaloy cladding may not contain defects other than pin holes and hairline cracks. The parameters for the authorized fuel assemblies are specified in Table 1, above.

Each package may contain up to 26 fuel assemblies. The assemblies are placed in a steel fuel basket that is made of square sleeves and support disks. The basket provides the spacing and neutron poison needed to maintain subcriticality. Each sleeve is surrounded by four BORAL sheets that are held in place by stainless steel cladding. The BORAL sheets have a minimum loading of 0.01 g/cm^2 of ^{10}B. The acceptance tests for the package (Chapter 8 of the application) include procedures for verifying the minimum ^{10}B loading in the BORAL. In the criticality calculations, the applicant considered only 75% of the minimum ^{10}B loading. The support disks in the fuel basket maintain 1.47-inch and 3.27-inch gaps between the fuel tubes. These gaps act as flux traps when water enters the cask cavity. The basket is designed to allow the regions inside and outside the fuel tubes to drain or flood evenly. The applicant has shown that the configuration of the fuel basket does not change under normal conditions of transport or under hypothetical accident conditions.

The applicant used the SCALE CSAS25 code sequence, which includes the KENO-Va code, and the 27 group cross-section library to perform the criticality analysis. This analytical method was benchmarked with thirty critical experiments. The benchmarking calculations yielded a bias of 0.0040 and a method error of ± 0.00644. This bias and error were applied to all calculational results.

The criticality analysis was based on the Westinghouse 15x15 assembly, with an enrichment of 4.2 wt%. The applicant determined that this assembly is the most reactive of the requested contents. (Note: The Westinghouse 17x17 OFA is typically the most reactive when compared to other PWR assemblies of the same enrichment. However, for this package, the applicant requested a maximum enrichment of only 4.1 wt% for the Westinghouse 17x17 OFA.)

The models used in the criticality analysis considered only the central region of the cask, which is the most reactive region because of the number of disks displacing water in the flux trap. This central region was assumed to be infinitely long. The applicant considered models of both an undamaged and a damaged package. Because the geometry of the basket does not change under hypothetical accident conditions, the only difference between the model of an undamaged and a damaged package is that, in the latter, interspersed moderation replaces the neutron shield material. The applicant's models are conservative and are consistent with the engineering drawings and the results of the structural and thermal evaluations.

Using the models described above, the applicant performed criticality calculations to show that the package remains subcritical under normal conditions of transport and under hypothetical accident conditions. To show that the package meets the single package requirements of 10 CFR §71.55(b), the applicant considered an infinite array of undamaged packages. The packages were assumed to be optimally moderated internally and externally.

Optimum moderation occurred when the packages were flooded with full-density water. Considering an array of undamaged packages for the single package evaluation is acceptable because the package is a large cask and its contents are surrounded by a thick lead shell, which is a more effective reflector than water. Under flooded conditions, the contents of the package are isolated and the materials or conditions beyond the lead shield (e.g., other packages, water reflection, and presence or absence of neutron shielding) will have an insignificant effect on reactivity. Also, the geometry of the basket and contents is the same for normal conditions of transport and hypothetical accident conditions.

To show that the package meets the requirements of 10 CFR §71.57, the applicant considered an infinite array of dry, undamaged packages and an infinite array of dry, damaged packages. Both cases assumed optimum interspersed moderation.

The applicant's criticality analysis shows that the package meets the requirements of 10 CFR §§71.55 and 71.57. The results of the analysis are summarized as follows:

TABLE 6

Applicant's Criticality Safety Calculations

Package Condition (Infinite Array, Infinite Length)	Maximum K_{eff} (Including Bias and Uncertainty)
Undamaged - Optimum Moderation Inside and Outside	0.9457
Undamaged - Dry Inside, Optimum Moderation Outside	0.4662
Damaged - Dry Inside, Optimum Moderation Outside	0.5219

The NRC staff performed an independent criticality analysis. The staff used the CSAS/KENO-Va codes and the 27GROUPNDF4 cross-section set in the SCALE 4.1 system. For a single package fully reflected and with optimum internal moderation, the staff calculated a maximum k_{eff} of 0.936. The staff's analysis supports the applicant's conclusion that the package remains subcritical under normal conditions of transport and under hypothetical accident conditions.

The applicant did not consider fuel rod gap flooding in its criticality analysis. To determine the effect of gap flooding, the NRC staff performed one-dimensional, scoping calculations with the SCALE CSAS1X code and the 27GROUPNDF4 cross-section set. The results of the staff's calculations show that the maximum reactivity increase due to gap flooding is less than 0.33%. The staff also considered gap flooding in its confirmatory analysis of the package.

The staff agrees with the applicant's conclusion that the package design meets the requirements for a Fissile Class I package.

OPERATING PROCEDURES

Chapter 7 of the application specifies operating procedures for the package. The chapter includes sections on package receipt, loading, preparation for transport, unloading, and leakage testing. Since the cask may be used for long term storage of spent PWR fuel, as well as for shipment, the procedures for preparing the package for transport consider shipment immediately after loading and after long term storage.

The operating procedures specify various tests and determinations that are to be made before each shipment. These include: (1) visual inspection of the package, including bolts, and seals, (2) leak testing the package within 12 months prior to shipment, (3) monitoring dose rates and contamination levels, and (4) application of a tamper-indicating seal. The metallic seals must be changed after each use. When a seal is replaced, the new seals must be leak tested in accordance with Section 7.5 of the application.

The Certificate of Compliance has been conditioned to specify that the package be operated and prepared for shipment in accordance with Chapter 7 of the application. For rail transport, the Certificate of Compliance has been conditioned to require approval by the American Association of Railroads of the railcar and the components of the tie-down system that are not a structural part of the package. For marine or barge transport, the National Cargo Bureau, Inc., must certify that package stowage is in accordance with U.S. Coast Guard regulations.

ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

Section 8.1 of the application specifies fabrication criteria and acceptance tests for the package. Welding will be in accordance with designated sections of the ASME Boiler and Pressure Vessel Code. Various acceptance tests will be performed prior to the first use of each packaging. These tests include: (1) examination of all welds, (2) a hydrostatic pressure test at 68 psig, (3) a fabrication verification leak test, (4) a trunnion load test, (5) a neutron shield tank leak test, (6) verification of the impact limiter wood density, (7) testing the shell surrounding each impact limiter for water tightness,

(8) testing the performance of the gamma-ray and neutron shielding, (9) testing the thermal performance of the package, (10) verifying the amount of boron in the neutron absorber plates, and (11) verification measurements of the cylindricity of the inner cavity and the diameter of the disks and fins in the basket.

Nuclear Assurance Corporation Drawing No. 423-802, Rev. 6, specifies standards for performing, examining and accepting welds. Welds are to be accepted per designated sections and sub-sections of the ASME Boiler and Pressure Vessel Code. The welds in the containment vessel of the packaging will be radiographed and accepted per Section III, Sub-section NB.

Section 8.2 of the application specifies a maintenance program for the package. The maintenance program includes: (1) a verification leak test of the package after the third use and annually thereafter, (2) thermal testing at five year intervals, (3) replacement of metallic O-rings after each use, and (4) visual inspection of various package components prior to loading and shipment. The periodic thermal test was previously discussed as part of the Thermal Evaluation, above.

Section 8.4 of the application specifies detailed procedures for fabricating the cask body and pouring the lead shield.

CONDITIONS

The Certificate of Compliance includes the following conditions of approval:

1. The maximum heat load within the packaging at any time (transport, storage or testing) shall not exceed 850 watts per assembly nor 22.1 kilowatts per package.

2. Known or suspected failed fuel and fuel with cladding defects greater than pin holes and hairline cracks are not authorized.

3. Water and residual moisture shall be removed from the containment vessel in accordance with the procedures in Section 7.1 of the application.

4. Containment vessel seals must be tested to a sensitivity of at least 2.9×10^{-5} std-cm^3/sec, and shown to have a leak rate no greater than 5.79×10^{-5} std-cm^3/sec:

    (a) Before first use of each packaging;

    (b) Within the 12-month period prior to each shipment; and

    (c) After seal replacement.

5. All containment vessel O-rings shall be replaced with new O-rings after each use.

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

   (a) Each packaging must be fabricated in accordance with the fabrication specifications in Chapter 8 of the application;

   (b) Each package shall be prepared for shipment and operated in accordance with the Operating Procedures in Chapter 7 of the application; and

   (c) Each package must meet the acceptance tests and be maintained in accordance with the Acceptance Tests and Maintenance Program in Chapter 8 of the application.

7. Prior to transport by rail, the Association of American Railroads must have evaluated and approved the railcar and the system used to support and secure the package during transport.

8. Prior to marine or barge transport, the National Cargo Bureau, Inc., must have evaluated and approved the system used to support and secure the package to the barge or vessel, and must have certified that package stowage is in accordance with the regulations of the Commandant, United States Coast Guard.

CONCLUSIONS

Based upon the statements and representations contained in the application, as supplemented, and the conditions listed above, we have concluded that the Model No. NAC-STC package meets the requirements of 10 CFR Part 71.

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