April 9, 2009

Mr. A. P. Cochran
Department of Energy
Naval Reactors
Washington, D.C. 20585

SUBJECT: CERTIFICATE OF COMPLIANCE NO. 9221 FOR THE MODEL NO. NRBK-41 PACKAGE

Dear Mr. Cochran:

As requested by your application dated November 5, 2007, as supplemented February 22, and November 14, 2008, enclosed is Certificate of Compliance No. 9221 for the Model No. NRBK-41 package. This certificate supersedes, in its entirety, Certificate of Compliance No. 9221, Revision No. 5, dated September 28, 2006. Changes made to the enclosed certificate are indicated by vertical lines in the margin. The staff's Safety Evaluation Report is also enclosed.

The U.S. Department of Energy is registered as user of the package under the general license provisions of 10 CFR 71.17. This approval constitutes authority to use the package for shipment of radioactive material and for the package to be shipped in accordance with the provisions of 49 CFR 173.471.

If you have any questions regarding this certificate, please contact me or Chris Staab of my staff at (301) 492-3321.

Sincerely,

/RA/

Eric J. Benner, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Docket No. 71-9221
TAC No. L24150

Enclosures:
1. Certificate of Compliance No. 9221, Rev. No. 6
2. Safety Evaluation Report
3. Registered Users

cc w/encl 1& 2: R. Boyle, Department of Transportation
J. Shuler, Department of Energy

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SAFETY EVALUATION REPORT
Docket No. 71-9221
Model No. NRBK-41 Package
Certificate of Compliance No. 9221
Revision No. 6

SUMMARY

By application dated November 5, 2007, supplemented February 22, and November 14, 2008, the U.S. Department of Energy (DOE) submitted an amendment request for Certificate of Compliance No. 9221 for the Model No. NRBK-41 package. Based on design changes to the inner containers, DOE requested removal of the restrictions for contents exceeding a Type A quantity, the radioactive material must be contained within a specimen with intact, undamaged cladding and the total quantity of radioactive material in the form of loose surface contamination within the package not exceed a Type A quantity. DOE also proposed resolution to 19 issues published by the NRC in Compatibility with International Atomic Energy Agency (IAEA) Transportation Safety Standards (TS-R-1) and Other Transportation Safety Amendments; Final Rule, Federal Register 69:16 (26 January 2004) p. 3698-33814 in order to comply with current regulations and obtain a designated certificate to "-96". The staff has evaluated the request, removed the restrictions, designated the certificate to "-96", and renewed the certificate up to April 30, 2013, with one additional condition. New condition No. 8 was included in the certificate to clarify that the package has not been evaluated under the provisions of 10 CFR 71.55(f) that became effective October 1, 2004. The staff has concluded that these changes will not affect the ability of the package to meet the requirements of 10 CFR Part 71 by reviewing against NUREG-1609, "Standard Review Plan for Transportation Packages for Radioactive Material."

1.0 GENERAL INFORMATION

The Model No. NRBK-41 package is designed for the transport of small quantities of test specimens between Naval Nuclear Propulsion Program (NNPP) sites and subcontractors.

1.1 <u>Packaging</u>

Top loading cylindrical lead shielded 304L stainless steel clad casks for the shipment of irradiated test specimens. The cask has an outside diameter of 27.16 inches and is 40 inches high. The outer shell is 1/2-inch thick stainless steel. The cask cavity is 5 inches in diameter by 16 inches deep and is provided with a bottom drain. The cavity shell is 1/4-inch thick stainless steel and is shielded by 10 inches of lead. The cask is closed by a lead-filled flanged plug fitted with an elastometer O-ring gasket and bolted closure. The cask has a seal-welded, 1/4-inch thick, stainless steel outer thermal shield which provides a 1/16-inch air gap between the outer surface of the cask outer shell and the inside surface of the thermal shield. A 1-inch thick stainless steel plate is welded to the bottom of cask. A second 1-inch thick stainless steel plate with a 1/8-inch deep, 25.5-inch diameter recess is welded to the first plate to provide a thermal shield for the bottom surface of the cask. The cask is bolted to a 48-inch square, all welded, "I" beam skid. Gross weight of the package is approximately 9,000 pounds.

1.2 Contents

1.2.1 Type and Form of Material

Byproduct and special nuclear material in solid form, contained within either the MIN-41 or the HIP-41 product containers. The MIN-41 container is constructed in accordance with Westinghouse Electric Corporation, Drawing No. 2D77456, Rev. F. The HIP-41 product container is constructed in accordance with Westinghouse Electric Corporation Drawing No. 5D06622, Rev. B.

1.2.2 Maximum quantity of material per package

The fissile contents of the package must be limited to a maximum of 350 equivalent grams of U-235. The number of equivalent grams of U-235 is determined by the equation: 1.0 x grams U-235 + 1.4 x grams U-233 + 1.6 x grams plutonium. The maximum decay heat load per package must not exceed 240 Btu/hr.

Plutonium in excess of twenty (20) curies per package must be in the form of metal, metal alloy or reactor fuel elements.

1.3 Criticality Safety Index 0.0

1.4 Drawings

The packaging is constructed in accordance with Battelle Memorial Institute Drawing No. 41-0001, sheet 1, Rev. D, and sheet 2, Rev. E, and Westinghouse Electric Corporation Drawing No. 1755E01, Rev. D.

1.5 No Evaluation Findings

2.0 STRUCTURAL

The NRBK-41 transport package is comprised of a cask body, a cover assembly, a skid assembly, and an inner container which provides containment for the radioactive material.

2.1 Structural Design Criteria

2.1.1 Design Criteria

The structural design criteria are developed to assure that the NRBK-41 has adequate structural strength to meet the requirements of normal conditions of transport as well as hypothetical accident conditions. These criteria are designated as those that affect the containment boundary and those that affect other package structures which contribute to the overall structural performance. The containment boundary is evaluated based on linear elastic analysis for Normal Conditions of Transport (NCT) and elastic-plastic analysis for Hypothetical Accident Conditions (HAC). Failure of the containment boundary or other structural components is evaluated by using a strain criterion to judge material acceptability.

2.1.1.1 Acceptance Criteria

The acceptance criteria for the NRBK-41 are established based on allowable stresses. The applicant makes no reference to any code requirements (such as the American Society of Mechanical Engineers (ASME) code) for the implementation of the acceptable allowable stresses. Staff inquired about the lack of code based acceptance criteria in a Request for Additional Information (RAI) and the applicant responded that no code or standard was used, however, the applied standards are "similar in many ways to the criteria found in the ASME Boiler and Pressure Vessel Code." The applicant stated that the acceptance criteria used were developed by the NNPP and have been used for several years and committed to adding discussion to the next revision of the Safety Analysis Report (SAR). The applicant provided justification for the following acceptance criteria in their response to a RAI and have committed to incorporating that justification in the next version of the SAR, as required by 10 CFR 71.31 (c):

1) For NCT and for the containment boundary portions during HAC
   - Maximum principal tensile and compressive stress must be less than the tensile yield strength of the material.

2) For combined loading under NCT
   - Maximum shear stress must be less than one half the minimum static tensile yield strength, or
   - Maximum effective von Mises stress must be less than the static uniaxial tensile yield strength.
   - For pure shear conditions, the maximum shear stress must be less than 0.577 times the minimum uniaxial tensile yield strength.

3) For HAC
   - Maximum stresses must be less than the minimum static ultimate tensile strength and the maximum shear stress must be less than one half the minimum static ultimate strength.
   - For Finite Element Analysis, the plastic strain must be less than the minimum uniform elongation tensile strain or the compressive strain limit.

The applicant provided a narrative and justification for the use of the minimum uniform elongation tensile strain criteria as well as the minimum compressive strain criteria. Staff evaluated the justification and found that the applicant is not relying on a substantial amount of reserve capacity in the material being evaluated. This approach takes into account geometric effects on material testing performance and removes uncertainty in the testing from being a factor in material performance and the staff finds that this approach is reasonably conservative.

2.1.2 Weights and Centers of Gravity

The SAR summarizes the weights and center of gravity locations for each of the inner containers. Staff finds this information complete and acceptable.

2.1.3 Codes and Standards

Codes and Standards for the outer cask are specified on the package drawings for requirements associated with testing, maintenance, and use as no additional casks are being fabricated. The inner containers will be fabricated/replaced on a regular basis and as such a more thorough presentation of applicable codes and standards are provided in Table 2.1-2 of the SAR.

2.2 Mechanical Properties of Materials

See Section 7.0.

2.3 Fabrication and Examination

Fabrication for the cask was based on engineering drawings and done in accordance with the ASME Boiler and Pressure Vessel Code (ASME B&PV), Sections VIII and IX including all welding. The two variations of inner containers were fabricated based on Naval Reactors requirements as well as the ASME B&PV code, Section IX.

Examinations for the outer cask consists of dimensional and surface finish examinations and main weld radiographic inspection prior to lead pouring. Acceptance standards were derived from ASME B&PV Section VIII, Part UW-51. All welds were Liquid Penetrant Tested on the root and final pass, lead integrity was verified by gamma probe testing, the cap casting was radiographically inspected and each NRBK-41 cask was hydrostatically tested to 150 psig which required that no leakage occurred at this pressure for a duration of one hour.

Examinations for both types of inner containers included dimensional and surface finish examinations to assure conformance with the engineering drawings, liquid penetrant testing of welds, and the HIP-41 is inspected externally via liquid penetrant testing.

2.4 General Standards for All Packages (10 CFR 71.43)

2.4.1 Minimum Package Size

The smallest overall dimension exceeds the specified requirement of 4 inches, therefore, the package meets the requirements of 10 CFR 71.43(a) for minimum size.

2.4.2 Tamper-Proof Feature

There are two tamper indicating features on this package. The first is a stainless steel wire which is passed through a hole which is drilled into each of six lid assembly attachment bolts. The presence of an intact seal passing between any two lid bolts assures that the content has not been tampered with. The second tamper indicating device is attached to the drain valve cover and the cask body when the assembly is prepared for shipment.

Thus, the requirements of 10 CFR 71.43(b) are satisfied.

2.4.3 Positive Closure

Positive closure is demonstrated by the use of a bolted closure lid, torqued to specified values and outfitted with devices which prevent bolt rotation once installed. Thus, the requirements of 10 CFR 71.43(c) are satisfied.

2.4.4 Chemical and Galvanic Reactions

See Section 7.0.

2.4.5 Package Valves or Other Devices

The applicant addresses the issue of the drain valve by installing a threaded hex head plug outfitted with an O-ring which is inserted into the valve coupled with a drain cap which limits the accessibility of the valve. The redundancy of this valve closure design precludes the escape of radioactive contents, therefore, 10 CFR 71.43(e) is satisfied.

2.4.6 Containment and Shielding Effectiveness

The applicant demonstrated that for Normal Conditions of Transport (NCT), there would be no damage that would cause the package to lose or disperse radioactive content. The intent of 10 CFR 71.43(f) is satisfied.

2.4.7 Accessible Surface Temperature

The applicant meets the accessible surface temperature limits by limiting the total decay heat of the contents, therefore, 10 CFR 71.43(g) is satisfied.

2.4.8 Continuous Venting

There is no feature of the package that is intended to allow continuous venting. Penetrations of the package are sealed to an extent which will prevent the inadvertent continuous venting of radioactive material. The intent of 10 CFR 71.43(h) is satisfied.

2.5 Lifting and Tie-Down Standards for All Packages (10 CFR 71.45)

2.5.1 Lifting Devices

The applicant performed classical stress calculations and determined that no stresses greater than one third the yield strength occurred in base material or structural welds. The staff reviewed these calculations and found them sufficient to demonstrate reasonable assurance that the lifting trunnions would not fail during lifting operations.

The requirements of 10 CFR 71.45(a)(1) for lifting devices are met.

2.5.2 Tie-Down Devices

The applicant evaluated the loads and stress state on associated structural components using classical engineering analysis and determined for the assumed tie down configuration, that all structural components resisting the 10G fore-aft, the 5G lateral, and the 2G vertical required loads, are adequately designed such that failure of the components will not occur. The applicant also developed a spreadsheet to evaluate scenarios that may not conform to the assumed tie down configurations. It was determined based on geometry and allowable stress values that the minimum and maximum spacing between lateral tie down points must fall between 82 inches and 90 inches respectively. Longitudinally, the distance from the centerline of the package to a tie down point must fall within the range of 33 inches to 51 inches.

Staff reviewed the calculation methodology, performed spot calculations, and performed confirmatory calculations and found that the analysis presented by the applicant is adequate for determining reasonable assurance of safety.

The requirements of 10 CFR 71.45(b)(1) for tie down devices are met.

2.6 Normal Conditions of Transport (10 CFR 71.71)

2.6.1 Heat

The applicant evaluated the package for thermal effects due to an ambient temperature of 100 degrees F in still air and direct sunlight. Internal pressure and differential thermal expansion were considered for the cask and both internal containers. The stresses due to thermal and pressure loadings were compared against the requisite allowable stresses for shear and tension for components important to safety. The applicant determined that all stresses were below established allowables, therefore the package successfully maintained containment. Staff reviewed the calculations and determined that sufficient margin of safety was demonstrated.

The requirements of 10 CFR 71.71(c)(1) are satisfied.

2.6.2 Cold

The applicant evaluated the package for thermal effects due to an ambient temperature of -40 degrees F in still air and shade. Differential thermal contraction and brittle fracture were considered for the cask and both internal containers.

Differential Thermal Contraction: The applicant evaluated interference fit due to thermal contraction and determined that the package function will not be adversely affected. One outcome of this evaluation showed that the bolt preload torque for the cask cover lid was lost, however, subsequent calculations demonstrated that the lid seals remained compressed to an extent that no leakage across the seal region was credible. Staff reviewed these calculations and found them acceptable.

Brittle Fracture: Materials subject to brittle fracture were identified as ESCALOY 45D stainless steel, carbon steel, and PH Condition H1100 stainless steel. The ESCALOY 45D Nil Ductility Transition Temperature (NDTT) was reported as -200 degrees F, and therefore remains ductile for NCT and HAC. Components made of carbon steel are subject to brittle fracture and staff disagrees with the rationale that compressive loads on thin web steel shapes will not initiate brittle fracture. The staff also disapproves of anecdotal evidence that brittle fracture is not a concern due to the lack of observation of such a phenomenon over the course of 35 years. Staff has determined that if in fact brittle fracture of the shipping skid were to occur, there would be no safety significance with respect to the cask and inner containers.

2.6.3 Reduced External Pressure

Under a reduced external pressure to 3.5 psia (25 kPa), the structural response is well below the pressure capacity of the cask seal would not be compromised. The staff reviewed the evaluation presented by the applicant and agrees that the requirements of 10 CFR 71.71(c)(3) are satisfied.

2.6.4 Increased External Pressure

Under an increased external pressure of 20 psia (25 kPa), the structural behavior is evaluated against the design strength and found to be acceptable. The staff reviewed the evaluation presented by the applicant and agrees that the requirements of 10 CFR 71.71(c)(4) are satisfied.

2.6.5a Vibration

The applicant concluded that based on the package design and 40 year operational history, vibration damage was not a concern for this package and the requirements of 10 CFR 71.71(c)(5) are satisfied for vibration.

2.6.5b Fatigue

The applicant stated in a response to a RAI, that operational history provided no evidence that fatigue damage due to vibration effects occurs in the package. The applicant did identify in their response, that differential thermal expansion can fatigue the cover attachment bolts. They provided an evaluation that showed the worst case for bolt material fatigue was 10^6 cycles at 20000 psi. Since the stress due to differential thermal expansion is approximately 16,500 psi and the number of thermal cycles due to differential thermal expansion is significantly less than 10^6 cycles, fatigue of the materials and components is not a concern. Staff reviewed the calculations and methodology and agrees with the applicant's conclusions.

2.6.6 Water Spray

The applicant demonstrated that water intrusion due to a water spray event is not credible therefore, the intent of 10 CFR 71.71(c)(6) is satisfied.

2.6.7 Free Drop

The applicant determined that the 30-foot drop for the Hypothetical Accident Conditions (HAC) produces significantly more damage to the package than what would occur during Normal Conditions of Transport (NCT) due to the higher values of deceleration associated with the HAC drops. The staff generally agrees that the HAC conditions bound the NCT conditions, however, as noted by the applicant, certain conditions must be evaluated as they have the potential to occur during the NCT 4-foot drop. The applicant identified two conditions that must be evaluated to determine whether the effectiveness of the package is reduced and they are as follows: (1) loss of containment and (2) excessive reduction of shielding material. The applicant demonstrated by way of bounding calculation that the HAC drop scenarios do not result in a containment breach. Because of similarity in materials and environmental conditions, an NCT 4-foot drop will not result in a loss of containment because the applied dynamic load is lower than in HAC drop scenarios. With respect to shielding reduction, staff reviewed calculations performed by the applicant to determine the relative axial and radial slump of the lead shielding and found the results satisfactory to support the assumptions made during the shielding and thermal reviews.

The requirements of CFR 71.71(c)(7) are satisfied.

2.6.8 Corner Drop

The corner drop test does not apply in accordance with 10 CFR71.71(c)(8).

2.6.9 Compression

The applicant performed a buckling analysis and determined that the compressive stresses developed were less than the critical buckling load for the outer cylindrical shell. Staff reviewed these calculations and found them credible. Therefore, the requirements of 10 CFR 71.71(c)(9) are satisfied.

2.6.10 Penetration

The applicant provided an empirical calculation based on a “shear plug model” to determine the penetration depth of a cylindrical penetration bar dropped from a height of 1 meter. Furthermore, in response to an RAI question requesting justification for the empirical approach used, the applicant provided a comparison of three different empirical calculation methods. The calculation that was used by the applicant required a wall thickness that was one order of magnitude greater than the other two methodologies indicating that the methodology used is conservative. The conservatism lies in the use of the yield strength, rather than the ultimate strength, when calculating the energy required to shear through the component wall. Staff reviewed the calculations and found the justification credible.

The intent of 10 CFR 71.71(c)(10) is satisfied.

2.7 Hypothetical Accident Conditions (10 CFR71.73)

2.7.1 9-meter Free Drop

The applicant presented evaluations for a 9-meter free drop in various orientations, utilizing analytical computer codes and classical stress analysis.

2.7.1.1 Flat Top Drop

The applicant used the analytical code Pronto2D to evaluate the effects of a 30-foot top drop considering both cold and hot conditions.

Deceleration of the package was determined using two separate bounding assumptions. The first was an evaluation that included a rigid circumferential corner geometry rather than a plate structure with an air gap. This assumption artificially increases the G load (1182 G’s) due to the lack of deformation in the idealized structure. Subsequently, the applicant performed an evaluation which simulated the opposite condition of an overly compliant circumferential corner geometry and determined that the G load would be 644 G's. The applicant state that the G load would actually fall somewhere in this range of G loads and determined that for conservatism, the upper bound G load would be used for remaining calculations.

The applicant evaluated maximum deformations due to the maximum G loads and found that air gaps due to lead slump are reduced significantly and were incorporated into the thermal and shielding calculations. Additionally, it was found that the cask cavity also deformed significantly such that the deformation imparted additional loading to the inner containers other than simply inertial loads do to self weight.

A summary of the governing allowable G loads as well as the corresponding G loads are presented in tabular form and all show Safety Factors exceeding 1.0.

2.7.1.2 Flat Bottom Drop

The applicant used the analytical codes CRUSHTAB and Pronto2D to evaluate the effects of a 30-foot bottom drop considering both cold and hot conditions.

Maximum deceleration for both hot and cold conditions was determined using the CRUSHTAB computer code which assumes all energy dissipation is achieved through crushing of the structural materials, rather than buckling or some other failure mechanism. It was unclear to staff whether this approach produced conservative results, so the staff requested that the applicant perform confirmatory calculations to demonstrate conservatism. The applicant provided an evaluation using the LS-Dyna computer code and demonstrated that the G loads produced by CRUSHTAB were approximately 2 times greater than those produced by LS-Dyna.

The applicant evaluated maximum deformations due to the maximum G loads and found that air gaps due to lead slump are reduced significantly and were incorporated into the thermal and shielding calculations. It was found for this case, that the cask cavity did not deform significantly such that the deformation imparted additional loading to the inner containers.

2.7.1.3 Top Corner Drop

The applicant used the analytical codes CORNRCR7 to evaluate the effects of a 30-foot top corner drop considering both cold and hot conditions. CORNRCR7 is in the same analytical software category as CRUSHTAB and uses material crushing as its failure mode for analysis.

The applicant extracted radial and axial components of this drop scenario and compared those with the corresponding maximum G loads for the side drop and end drop orientations. It was determined that the results of the axial or side drop evaluations bound the relevant components of the top corner drop with one exception; the hot condition top corner axial component exceeded the flat top drop axial G load by 12 G's. Given that the applicant demonstrated the inherent conservatism in the CRUSHTAB and the CORNRCR7 methodology, the applicant concluded that the flat top drop is the more limiting condition.

2.7.1.4 Bottom Corner Drop

The applicant used the analytical codes CORNRCR7 to evaluate the effects of a 30-foot bottom corner drop considering both cold and hot conditions. CORNRCR7 is in the same analytical software category as CrushTab and uses material crushing as its failure mode for analysis.

The applicant did not perform the same type of comparison as for the top corner drop, however, inspection of the respective G load component values determined by CORNRCR7 shows a similar trend, therefore the structural behavior and subsequent conclusions drawn for the side and end drops are equivalent or bound this case as well.

2.7.1.5 Side Drop

The applicant used the analytical code CRUSHTAB to evaluate the effects of a 30-foot bottom drop considering both cold and hot conditions.

Maximum deceleration for both hot and cold conditions was determined using the CRUSHTAB computer code which assumes all energy dissipation is achieved through crushing of the structural materials, rather than buckling or some other failure mechanism. As indicated for the flat bottom drop (Section 2.7.1.2), the use of CRUSHTAB produces conservative results with respect to calculation of maximum G load.

The applicant found that no additional loading was applied to the inner containers due to excessive deformation of the cavity tube. In addition, the applicant concluded that the deformation of components providing air gaps are bounded by deformations predicted for the flat top and flat bottom drops due to the uniformity of loading in the latter cases.

2.7.1.4 Secondary Impact Oblique Drop

The applicant used the analytical code Abaqus to evaluate the effects of a 30-foot oblique drop considering both cold and hot conditions.

The applicant demonstrated that this case is conservatively bounded by the side drop G loads for the cavity tube.

The staff reviewed the calculations and computational models and found the evaluations adequate to make a safety determination.

The analyses in aggregate satisfy the requirements of 10 CFR 71.73(c)(1).

2.7.2 Crush

This evaluation is not applicable due to the package mass exceeding 500 kg (1100 lbs) per 10 CFR 71.73(c)(2).

2.7.3 Puncture

The applicant evaluated puncture by using an empirical formula developed by Oak Ridge National Laboratory (ORNL) for lead backed steel shells. Upon review of the reference provided by the applicant, this analytical approach provides reasonable assurance that the package can withstand a 1 meter drop onto a puncture bar without breaching the outer shell. The requirements of 10 CFR 71.73(c)(3) are met.

2.7.4 Thermal

The applicant evaluated the package for thermal effects due to a fire temperature of 1425 degrees F for 30 minutes. Internal pressure and differential thermal expansion were considered for the cask and both internal containers. The stresses due to thermal and pressure loadings were compared against the requisite allowable stresses for shear and tension for components important to safety. The applicant determined that all stresses were below established allowables; therefore the package successfully maintained containment. Staff reviewed these calculations and agrees with the conclusions drawn by the applicant. The intent of 10 CFR 71.73(c)(4) is satisfied.

2.7.5 Immersion - Fissile

Water in-leakage is assumed in the criticality analysis, therefore, this requirement is not applicable. The intent of 10 CFR 71.73(c)(5) is satisfied.

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2.7.6 Immersion - All Packages

This requirement was evaluated using the same methodology used for evaluating increased external pressure presented in Section 2.6.4 of the SAR. The staff agrees that the requirements of 10 CFR 71.73(c)(6) are met.

2.7.7 Deep Immersion

The staff utilized ASME Code Case N-284 to perform confirmatory calculations of the buckling response due to deep immersion in water. The staff agrees that the package design is adequate for deep submersion and the requirements of 10 CFR 71.61 are satisfied with respect to stress limits and stability requirements.

2.8 Evaluation Findings

Based on review of the statements and representations in the application, the staff concludes that the structural design has been adequately described and evaluated and that the package has adequate structural integrity to meet the requirements of 10 CFR Part 71.

3.0 THERMAL

This section describes the results of staff review that verifies the thermal performance of the NRBK-41 radioactive material package design under normal conditions of transport (NCT), hypothetical accident conditions (HAC), and that verifies that the package meets the thermal performance requirements of 10 CFR Part 71.

3.1 Description of Thermal Design

The NRBK-41 radioactive materials cask is a right circular cylinder primarily of Type 304L SS with 10 inches of lead shielding in the radial and slightly more in the axial directions [1]. There are two inner containers currently designed for use with the NRBK-41 cask; the MIN-41 and HIP-41. The MIN-41 inner container is currently in service and was originally licensed for use in 1988. The HIP-41 inner container is designed to withstand higher internal pressures than the MIN-41 inner container. The inner containers provide the primary containment boundary of the cargo. The space between the outside of the inner cavity and inside of the outer shell is filled with lead shielding. A ¼-inch SS thermal shield is seal welded on the outside surface of the outer shell with an 1/16-inch air gap between the outer surface of the cask outer shell and the inside surface of the thermal shield. A thermal shield is also provided on the bottom of the cask in the form of a 1-inch thick SS plate with a 1/8-inch deep, 25.5-inch diameter recess that is welded to the cask bottom plate. The top access opening of the cask is sealed by a lead-filled cover assembly that is bolted to the top of the cask for shipment.

Two additional thermal features are incorporated into the cask design. One feature is the 40 copper fins, each 20 inches long by 0.06 inch thick, brazed on a 9-degree pitch to the inside diameter of the outer shell. The second feature is a pressure venting devise that is installed in the outer shell of the cask. The purpose of this venting device is to safely vent steam from the lead filled cavity and prevent water from entering the lead cavity without releasing the molten lead under HAC. To exhaust the steam or other gases from the lead cavity, an alloy with a low melting point is used to plug the vent, a stainless steel screen is used to restrict the lead flow, and the vent releases steam as the pressure increases. The vent design prevents water inflow through the vent due to the plug with a low melting point alloy. The inspections prior to each shipment are performed to ensure water is not entering the lead cavity through the venting device.

The MIN-41 container is housed in a seamless pipe that is 4.95 in diameter and 15.87 inches in length. The bottom of the container is formed by a 0.38 inch thick plate that is welded inside the end of the housing. The top closure consists of (1) a metallic C-ring, (2) a flat plate that compresses the C-ring, (3) a plug assembly that mates with the upper end of the housing by means of a breech lock arrangement, (4) eight screws that are threaded through the plug assembly and tightened against the cap to compress the C-ring, and (5) eight jam nuts to prevent loosening of the screws.

The housing of the HIP-41 inner container is a high strength SS casting of 4.95 inch diameter and 15.893 inch length. The inner cavity at the bottom of the container is hemispherical with a bottom thickness of 0285 inch. At the upper end of the container a SS plug is inserted until it seats on a tapered area in the body and is retained by an externally threaded SS ring that threads into the top of the container body. The plug also contains a pipe plug that seals off a leak test penetration and a larger plug that isolates the area where the leak fixture is inserted for testing.

The C-ring that is the primary boundary seal in the MIN-41 inner container is metallic, has an operating temperature capability in excess of 1000°F. The licensee assures that the C-ring seal is unaffected by the radiation exposure from any radioactive cargo. The pressure capability of MIN-41 is 200 psig under NCT and 200 psig under HAC.

The boundary seal on the closure of the HIP-41 is provided by double elastomer O-rings. Licensee performed tests to investigate the combined effects of radiation exposure during normal service, elevated temperature (during fire accident), and pressure (generated due to high cargo temperature due to fire accident) relative to the seal material's ability to maintain a seal. The licensee has concluded that radiation exposure, at least 3×10^8 R, has no significant affect on the sealing capability during and after exposure to the temperatures and pressures associated with HIP-41 O-rings during 10 CFR Part 71 hypothetical fire accident.

Contents Decay Heat

The NRBK-41 package is used to ship various fissile and non-fissile test specimens. The maximum weight of the MIN-41 inner container is 25 lbs and its maximum cargo weight is 58 lbs. The maximum weight of HIP-41 inner container is 34 lbs and the maximum cargo weight is 53 lbs. The maximum fissile content of the cargo that is shipped in the MIN-41 or HIP-41 inner container shall be limited to a maximum of 350 grams equivalent U-235.

The maximum decay heat load that is to be shipped in either inner container, MIN-41 or HIP-41, is 240 BTU/hr. The decay heat of 240 BTU/hr is calculated for a 122°F peak surface temperature for the worst-case shipping configuration and is selected as the maximum allowable decay heat load. The applicant has shown that this cargo decay heat load produces a maximum temperature on the outside surface of the cask of 122°F in still air at 100°F and in shade, in compliance with 10 CFR 71.43(g) for non-exclusive shipments.

At 240 BTU/hr, the cargo temperature is predicted to be 1323°F under NCT. Table 3.1.1 lists the peak temperatures for various components of NRBK-41 for NCT for a cargo decay heat of 240 BTU/hr. Table 3.1.2 lists temperatures for HAC with 30 minute fire for cargo decay heat of 240 BTU/hr. Table 3.1.3 lists pressures for NCT and HAC for the primary containment of NRBK-41.

Table 3.1.1 Summary of Temperatures at NCT

NRBK-41 Component	Peak Temperature (°F)
Thermal Shield	201
Thermal Shield Base Plate	193
Cask Base Plate	194
Cask Body Sidewall	198
Cavity Tube	203
Cap Casting	205
Cover Cone	204
Cover Top Plates	208
Lead in Cover	203
Lead in Cask Body	200
HIP-41 or MIN-41 Inner Container	221
HIP-41 O-ring	218
Cargo*	1323

*Bounding temperature for decay heat of 240 BTU/hr

Table 3.1.2 Summary of Temperatures at HAC

NRBK-41 Component	Peak Temperature (^{o}F)	Time after Start of Fire (hours)
Inner Container	602	1.9
Inner Container HIP-41 O ring	594	2.0
Cavity Tube	612	1.4
Cargo	1463	-
NRBK-41	915	0.5

Table 3.1.3 Summary of Maximum Pressures at NCT and HAC

Condition	NRBK-41 Component	Typical (psig)*	Maximum Permitted (psig)
Normal Conditions of Transport	HIP-41	17.7	1500
	MIN-41	17.7	180
	*Based on a cargo temperature of 600^{o} F		
Hypothetical Accident Conditions	**NRBK-41 Component**	**Maximum Calculated (psig)***	**Maximum Permitted (psig)**
	HIP-41	172.7	1600
	MIN-41	172.7	200
	*Based on limiting cargo for HAC (cargo = 1463^{o}F; Inner Container (HIP-41 or MIN-41 = 602^{o}F)		

3.2 Material Properties and Component Specifications

Table 3.2-1 of the Safety Analysis Report (SAR) lists density, temperature-dependent thermal conductivity, and heat capacity of the materials used in the calculation models. Table 3.2-2 lists properties of air as a function of temperature. Table 3.2-3 of SAR lists radiation properties of solids used in the calculation models.

NRBK-41 package components significant to safety and susceptible to heat input, elevated temperatures, or radiation, consists of the elastomer O-rings in the HIP-41, which is part of the primary boundary, and the pressure venting device. Technical specifications and results of high temperature and pressure of elastomer O-ring are presented in Appendix 4.5.2 of the SAR. The pressure venting device specifications and testing results are listed in Appendix 3.5.1 of the SAR.

3.3 Thermal Evaluation for Normal Conditions of Transport

NRBK-41 TRUMP Cask Model

The NRBK-41 heat transfer analysis for NCT is performed using the TRUMP computer program [2]. TRUMP solves a set of simultaneous partial differential equations with four independent variables of spatial coordinates and time, and three primary dependent variables of temperature and two reactant concentrations. The TRUMP program is capable of simulating heat transfer, by conduction, convection, or radiation at each node boundary.

TRUMP explicitly models the top, bottom, side structure, and the side and bottom thermal shields. Figure 3.3-1 of the SAR illustrates the schematic of the TRUMP model of the NRBK-41 cask. Figure 3.3-2 represents the schematic of the TRUMP model of NRBK-41 cask with HIP-41 inner container detail. The MIN-41 schematic closely resembles the HIP-41 in size, shell thickness and internal pressures.

The NCT model described in Section 3.3 of the SAR is determined to be an adequate representation for the cask condition resulting from the 4-foot free drop. The applicant has shown that the 4-foot free drop will result in minor lead slump causing formation of a slump void between the cask and the lead shielding opposite the impact surface. In the vicinity of the impact surface, this lead slump also reduces the gap between the cask body lead outboard surfaces and the cask. This gap exists as a result of differential thermal contraction between the lead and the stainless steel upon cooling during cask fabrication. Since these two effects tend to offset each other from a heat transfer perspective, the overall result is judged to have no significant effect on cask component temperatures.

Inner Container Cargo Model

Cargo is modeled to determine the average gas temperature in the inner container. The calculation for average gas temperature requires cargo temperature. For a given heat load, the cargo has a maximum decay heat power density, a minimum surface area, and is placed in the inner container in a way as to minimize heat transfer to the inner container. The cargo model assumes each of the three components (cargo, Zircaloy bag, and the inner container) is represented by a single surface, each having a single temperature for that surface. During normal operations, Zircaloy foil is used to wrap specimens to minimize spread of contamination between specimens and to the inner container. Zircaloy is used as a wrapping material since it has a low emissivity value which impedes radiation heat transfer. Appendix 3.5.4 of SAR gives the details of the inner container cargo model that yield a value of 220^{o}F.

3.3.1 Heat and Cold

Maximum Temperatures

Section 3.3.1.1 describes four possible shipment configurations for the NRBK-41 package. Of these shipping configurations, the maximum NRBK-41 cask temperatures for NCT occur for the shipping configuration of bare NRBK-41 casks shipped on an open truck with direct solar exposure. For the limiting shipping configuration, the 10 CFR Part 71 accessible surface temperature limit for nonexclusive use shipment of 122°F occurs at the decay heat load of 240 BTU/hr.

An upper bound for a cargo temperature of 1323°F is obtained by using the cargo model described in Section 3.3 of the SAR using the limiting cargos for maximum decay heat of 240 BTU/hr. The licensee assures that in actual practice, shipments are normally controlled through cargo content or hold time such that the cargo temperatures do not exceed 500 to 600°F in order to preserve the quality of the test specimens for metallographic examination purposes.

Evaluation of Package Performance for NCT

The most limiting thermal condition for the package under NCT occurs for maximum decay heat and full solar load and shipping configuration of casks on an open truck. For this configuration, the temperatures of the NRBK-41 cask and inner container are given in Figures 3.3-7 and 3.3-8 of the SAR and are listed in Table 3.3.1 of the safety evaluation. All NRBK-41 materials of this composition are well within the allowed limits.

The elastomer O-ring seal on the HIP-41, which is most thermally vulnerable package component, is at 218°F and is below the steady state recommended upper service temperature of 300°F.

Maximum Internal Pressure and Thermal Stresses

As a conservatism, a peak temperature of 300°F is assumed for all the components in determining the combined effects of differential thermal expansion of materials used throughout the cask and inner containers. Section 2.6.1.3 of the SAR evaluates the stresses induced by internal pressurization of the cask cavity and inner containers. Section 2.6.1.4 determines the combined effects of differential thermal expansion and pressure and compares calculated stresses to allowable stresses. With the boiling point of lead up to 1749°C (3180°F) and the thermal analysis for the worst-case scenario, the lead temperatures are much lower than the boiling point, and should not be vaporized. Therefore, the over-pressurization and the related thermal stress need not be considered in the thermal-related issues.

3.4 Hypothetical Accident Thermal Evaluation

3.4.1 Initial Conditions

The following conditions were utilized for the worst-case fire transient:

1) The analysis considers both convection and radiation heat transfer between the fire and the cask outer surfaces. The emissivity of the fire is 0.0, while the cask outer surface has the absorbtivity of 0.8. The outer surface convection coefficient during the fire has a value of 10 w/m^2-K. A 20% uncertainty factor is incorporated.

2) With the package in shade before fire, solar heating is not considered in pre-fire analysis, and is also neglected during hypothetical fire, when compared to fire heat source. After the termination of the fire, cool down is by both convection and radioactive heat transfer, with emissivity of the cask outer surface assumed

to be 0.5. The solar heating was considered during the cask cooldown phase (post-fire analysis). Solar heating of 800 g cal/cm^2 (246 BTU/hr-ft^2) was specified for flat horizontal surfaces, 200 g cal/cm^2 (62 BTU/hr-ft^2) was specified for flat surfaces not transported horizontally, and 400 g cal/cm^2 (123 BTU/hr-ft^2) was specified for curved surfaces.

3) The ambient temperature assumed is 100^{o}F prior to the fire, 1475^{o}F during the fire, and 100^{o}F during cool down.

4) Maximum decay heat rate for the package prior to, during, and after the hypothetical accident is 240 BTU/hr.

5) Based on structural analysis, the significant closing of the thermal shield air gap occurs as a result of the flat top drop and flat bottom drop accidents. For both drop orientations and material thermal expansion during fire, radial movement of the lead causes the expansion of the cask outer tube, leading to an almost complete collapse of the 1/16-inch air gap between the cask outer tube outside diameter and thermal shield inside diameter. Therefore, the 1/16 inch air gap is excluded in the thermal evaluation under a hypothetical fire accident.

Package Conditions

A fire transient analysis was performed for each of the 30-foot side, flat top, flat bottom drop, bottom corner, top corner, and secondary impact oblique drop accident conditions.

3.4.2 Hypothetical Fire Accident Analysis

Software Used

Hypothetical fire accident analysis was performed by the licensee using ABAQUS/Explicit version 6.6-1. ABAQUS is capable of modeling much greater levels of geometric detail and thus produces less conservative but more accurate results than TRUMP which was used for both normal and accident conditions previously. The ABAQUS inputs and modeling setup was reviewed by staff for the analysis with ambient temperature 800^{o}C (1475^{o}F) during fire and with no air gap existing between outer steel tube and steel thermal shield.

Model Geometry

The finite element model consists of two-dimensional, axisymmetric representation of the NRBK-41 cask and inner containers. Details of the models include representations of the cask upper and lower weldments, upper and lower cones, thermal shield, base plate, thermal shield base plate, cover assembly, HIP-41 inner container, and trunnion assemblies. Since HIP-41 resembles the MIN-41 closely in size, shell thickness, and material of construction (SS), temperatures and internal pressures calculated for HIP-41 are judged to be representative of the MIN-41 as well.

Cask and cover assembly lead fills are included in the model. Rearrangement of lead resulting from the drop accidents is modeled in the geometry for each damage condition considered in the model. Air gaps resulting from the lead rearrangement is also modeled.

Because of the differential expansion effects between lead and SS and the volume expansion of lead upon melting, all SS-lead interface air gaps, except those affected by lead slump, are assumed to be zero in the NRBK-41 cask finite element model. The 40 copper fins that bridge the air gap during normal conditions, between the SS outer tube inner diameter and the cask body lead outside diameter, are ignored for the HAC.

About 10,000, 4-node quadrilateral linear axisymmetric heat transfer elements make up the NBRK-41 cask finite element model.

Lead Melting

Based on the applicant's thermal analysis, a significant amount (~30%) of lead will melt under the hypothetical fire accident conditions when the air gap between lead and outer tube, as well as the air gap between outer tube and thermal shield, collapse due to the lead movement after drop tests and material thermal expansion. The lead melting phenomena in NRBK-41 was analyzed by the applicant using ABAQUS code. In order to confirm the applicant's findings, the staff performed some additional analyses using both the FLUENT and ANSYS codes. Fire conditions were taken from the applicant's analysis except for the convection heat transfer coefficient value. The staff used a more realistic value obtained from Sandia National Laboratory fire experiments, as described in Ref. 3. The staff's confirmatory analyses shows that about 31% of lead would melt under the hypothetical fire accident scenario with the air gap between steel outer tube and steel thermal shield closed due to lead movement after drop tests and material thermal expansion. NUREG-1609, "Standard Review Plan for Transportation Packages for Radioactive Material" [4], states, "Confirm that lead shielding does not reach melting temperature." While the applicant's analysis, as confirmed by the staff, shows ~30% lead melting, the NRBK-41 cask still meets the regulatory requirements based on the amount of lead predicted to remain solid during HAC. Both the applicant's and staff's analyses are conservative in nature and assume complete closure of the air gaps, and as such, are considered to bound the amount of lead that may become molten during HAC. Any amount of air gap remaining open would significantly reduce the maximum lead temperatures predicted by the staff's model.

Based on structural review SER 2.7.3 (Puncture), the staff agreed that the package can withstand a 1 meter drop onto a puncture bar without breaching the outer shell. Based on structural review 2.7.4 (Thermal), the staff is assured that for the package in a fire temperature of 1425^{o}F for 30 minutes, all stresses are still below established allowables. Therefore, the staff agrees the package successfully maintained containment and loss of potentially molten lead is not a concern.

Natural Convection Film Coefficient During Cooldown

Using the properties of air listed in Table 3.2-2 of SAR, the natural convection film coefficient as a function of the temperature difference between the cask outer surface and ambient environment is calculated.

Inner Container Cargo Model

The cargo temperature is required to determine the inner container pressurization. In the absence of a cargo-specific heat transfer analysis, a bounding temperature for the cargo is obtained by choosing a limiting cargo and the same calculation model used in the NCT with one exception. The exception is that the temperature for the inner container, used as a boundary condition in the model, is the maximum of the average inner container temperature occurring during the fire/cool down transient in lieu of the NCT inner container temperature. Use of the maximum inner container temperatures from the fire transient as a boundary condition adds conservatism to the calculation.

3.4.3 Package Temperatures and Pressures

Package Temperatures

The ABAQUS results for the 30-foot secondary impact oblique drop condition produced higher temperatures than the calculations for the side drop, flat top, and bottom drops accident conditions. Figure 3.4-2 of the SAR illustrates the cask component temperatures resulting from 30-foot secondary impact oblique drop fire accident. Figure 3.4-3 gives the temperatures of the inner container and the cavity tube which are most important to the structural considerations.

The containment components most vulnerable to elevated temperature are the HIP-41 elastomer O-rings. Figure 3.4-4 illustrates the O-ring temperature as a function of time during the hypothetical fire and post cool down period.

As a result of the hypothetical accident, significant melting of the lead shielding occurs when the air gap between steel outer tube and steel thermal shield does not exist due to thermal expansion of steel materials during a hypothetical fire.

An upper bound to the maximum cargo temperature reached during the fire/cool down transient is obtained by the application of the cargo model described in this section. Using a peak average inner container temperature of 602^{o}F as the boundary condition, the cargo temperature as a function of cargo decay heat is given in Figure 3.4-9. For the maximum decay heat of 240 BTU/hr, the corresponding maximum cargo temperature is given by 1463^{o}F, which is below the limit of 1475^{o}F specified in 10 CFR 71.73.

Maximum Internal Pressure

Maximum internal pressure is determined using the cargo temperature (1463^{o}F) and the inner container temperature (602^{o}F) which are the limiting fire accident temperatures in lieu of normal condition of transport temperatures. Detailed calculation for the maximum internal pressure is presented in Appendix 3.5.5. Using a cargo decay heat of 240 BTU/hr, the fire accident bounding peak pressure for the assumed cargo is calculated to be 187.4 psia.

For the HAC, the acceptability of this pressure is determined during the criterion:

$$P_{total} \leq P_{accident\ limit}$$

Where $P_{accident\ limit}$ = 214.7 psia (200 psig) for MIN-41 and

= 2,014.7 psia (2000 psig) for HIP-41.

Since P_{total} is less than $P_{accident\ limit}$ the cargo in this example calculation is acceptable with respect to inner container pressure under HAC.

3.4.4 Evaluation of Package Performance for the Hypothetical Accident Thermal Conditions

The most limiting thermal condition with respect to temperature is from the 30-foot secondary impact oblique drop accident. The temperatures of the inner container and the cavity tube for this limiting accident condition are given in Figure 3.4-3 of the SAR.

The peak O-ring temperature for this limiting accident is 594°F which is within the temperature range demonstrated as acceptable by the test results presented in Figure 3.4-4.

Significant melting of the lead is analyzed with ABAQUS code. The cask lead melt during the hypothetical fire and post fire cooldown is plotted in Figure 3.4-5 for the 30-foot flat top drop, Figure 3.4-6 for the 30-foot flat bottom drop, Figure 3.4-7 for the 30-foot flat side drop, Figure 3.4-8 for the most limiting 30-foot secondary impact oblique drop, and Figure 3.4-10 for all drop orientations. In average, around 33% of lead is melted under a 30-min hypothetical fire when the ambient temperature is set as 800°C (1475°F).

3.5 Test Descriptions

The thermal tests are not performed as part of the maintenance program because with the maximum cargo thermal load being only 240 BTU/hr, the degradation of the thermal characteristics of the cask due to normal conditions of transport is not expected. Instead of performing the thermal tests, other tests are performed periodically as supplements to provide indication of a change in the thermal characteristics of the cask. Once the periodic inspections indicate a change in cask's thermal characteristics, the thermal evaluations should be performed to ensure the heat transfer capability will not be affected.

3.6 Evaluation Findings

Based on review of the statements and representations in the application, the staff concludes the thermal design has been adequately described and evaluated, and that the thermal performance of the package meets the thermal requirements of 10 CFR Part 71.

3.7 References

1. B-REO (CD) -1211, "NRBK-41 Cask Safety Analysis Report for Packaging," Bettis Atomic Power Laboratory, a Report for US Department of Energy, Jan. 11, 2008.

2. UCRL-14754, Revision 3, "TRUMP A Computer Program for transient and Steady State Temperature Distributions in Multidimensional Systems," Arthur L Edwards, Lawrence Livermore Laboratory, September 1972.

3. "Thermal Measurements in a Series of Large Pool Fires," Sandia Report SAND85 –0196 TTC – 0659 UC 71, (August 1971).

4. NUREG-1609, "Standard Review Plan for Transportation Packages for Radioactive Material," Final Report of US Nuclear Regulatory Commission, May 1999.

4.0 CONTAINMENT

4.1 Description of Containment Design

This application requests two amendments. One amendment requests changes to the inner container's leak test penetration to make it leaktight following hypothetical accident conditions (HAC). Inner containments are designated as MIN-41 or HIP-41 and provide the primary containment boundary which is enclosed within the NRBK-41 transportation cask (the secondary containment boundary). The test penetration's closure mechanism has been redesigned to provide a metal to metal seal. The other amendment requests a "-96". Containment issues for the "-96" include revised A_2 values and elimination of the double containment provision for plutonium, where the latter is not applicable to this application.

The MIN-41 was licensed prior to 1996 and is designed for a maximum internal pressure of 180 psig and 200 psig for NCT (normal conditions of transport) and HAC, respectively. Whereas, the HIP-41 was licensed for use in 1996 and is designed for a maximum internal pressure of 1500 psig for NCT and 2000 psig, respectively.

4.2 Containment Evaluation

The applicant updated their analysis (included in Appendix 4.5.1 "Radioactive Material Within the NRBK-41 Cask Inner Cavity") for the permissible leak rate from the secondary containment, the NRBK-41 cask, which is designed to restrict loose crud from escaping the interior surface of the cask. This loose crud would be a result of exposure of the cask's interior or the inner container's exterior to contamination during the loading and unload operations. The source term was determined by swiping the surfaces of casks that had been immersed in a typical water pit. The staff checked the analysis for NCT and noted that the applicant demonstrated that their calculated leak rate for the most severe content is bounded by the pre-shipment leakage rate of 10E-3 ref-cc/s as recommended by American National Standards Institute (ANSI) 14.5-1997. The staff questioned the release fractions utilized for the crud (8.626 E-6) since an acceptable release fraction for crud is 0.15 as recommended in SRP-1607. The applicant explained that their basis for the aforementioned release fraction is based on testing documented in the SARP for the M-140, Rev. 14. The staff accepted the basis for this release fraction based on the conservatisms in the previously mentioned test that provide reasonable assurance of its validity. These conservatisms include, but are not limited to:

- The test differential pressure was approximately 9 times more than the operating pressure of NRBK-41 cask of 3.9 psig.
- The release path of test cask was a one inch diameter hole compared to a postulated hole diameter of approximately one micron (which is comparable to a leaktest to 10 E-3 ref cc/sec). This represents a leakage path diameter approximately four orders of magnitude larger than the postulated leakage path diameter.
- There is no spent fuel present in the NRBK-41, as there was in the tested configuration which would be more conducive for the presence of crud.

It should be noted that this secondary containment boundary analysis is only applicable for NCT and a similar analysis was not performed for HAC since the secondary containment is postulated to fail for HAC. Furthermore, the applicant stated that all the crud is assumed to be available for release to the environment in the event that the secondary containment is breached during HAC. The potential for release of all the crud content during a HAC is administratively controlled by limiting the activity from within the NRBK-41 cask but outside of the inner container to 0.2 Ci, which is 4 times less than the A_2 limit associated with the crud mixture from in side of a hot cell type environment and over 18 times less than the minimum composite value of a mixture from a water pit environment. These A_2 values are reviewed biennially and may change from the values shown in Table 4.5.1.1., "Radiaoactive Material Within Cask Cavity Due to Water Pit Exposure." Hence, the staff believes that reasonable administrative controls are in place to assure that the release of radioactive material would be limited to less than one A_2 in a week as required by 71.51(a)(2).

The primary containment, for both the MIN-41 and HIP-41, have had their leak test penetration reconfigured to add a metal to metal threaded seal which is leak tested to leaktight criteria prior to shipment to ensure the inner container's integrity. The connection of the leak test penetration and the plug is via threaded joint and it is sealed with adhesive.

4.3 Evaluation Findings

Since the applicant used A_2 values that were in general agreement with the latest regulations, the staff finds that a "-96" is warranted and that the containment meets the requirements of 10 CFR Part 71. Some Editorial Comments include:

- E4-1 In Section 4.1.1.1.1 the reference to Figure 4-1 should be to Figure 4.1-1.
- E4-2 Section 4.4, the page number is 4.3-5 but should be 4.4-1.
- E4-3 In Table 4.5.1-1, for NRBK-41-12, column Fi totals 0.9793 but it should total unity.

5.0 SHIELDING

5.1 Shielding Design

The NRBK-41 transportation cask is designed to allow transport of fissile material of varying ^{235}U ,^{233}U, or ^{239}Pu fuel in either the MIN-41 or HIP-41 inner container. Since the HIP-41 container is composed of significantly more structural material on both it's upper and lower ends, the MIN-41 container was used for all aspects of the shielding analysis since it's design is bounding for both inner containers.

5.2 Shielding Evaluation

The applicant provided an analysis to demonstrate that the package meets the external radiation standards of 10 CFR Part 71.47. The applicant continued to use the maximum decay heat load of 240 BTU/hr that was previously used to calculate source terms for fuel and non-fuel components. Based on these source terms, the applicant calculated both the maximum gamma and neutron dose rates external to the NRBK-41 package. In all instances, the maximum dose rates were below the limits for non-exclusive use shipments.

During hypothetical accident conditions it is possible for some of the lead shielding to melt due to high temperatures. Any volume of molten or partially molten lead is not taken credit for in the shielding analysis, however, the remaining solid lead after the fire accident is assumed to remain in the package and maintain its configuration around the inner cavity housing either the NIM-41 or HIP-41 inner containers. Based on the thermal analysis in Section 3.5 of their SAR, a minimum of approximately 3 inches of lead remains in the bottom corner near the inner cavity.

5.3 Evaluation Findings

NRC staff performed confirmatory calculations for the gamma dose rates using the MICROSHIELD version 5.05 computer code for both normal and hypothetical accident conditions. Staff's results agreed well with the applicant's calculations. Based on the verification of adequate shielding modeling by the applicant and supported by the analysis provided with the amendment request, staff finds that the NRBK-41 package, as amended continues to meet the shielding requirements of 10 CFR Part 71.

6.0 CRITICALITY

6.1 Criticality Design

The NRBK-41 transportation cask is designed to allow transport of fissile material of varying ^{235}U ,^{233}U, or ^{239}Pu fuel in either the MIN-41 or HIP-41 inner container. The cask is limited to a maximum loading of 350 grams of equivalent ^{235}U.

The applicant conservatively modeled a fuel loading of 375 grams of ^{235}U, 267.8 grams ^{233}U, or 234.7 grams ^{239}Pu or any combination that satisfied the following equation:

^{235}U grams/375 + ^{233}U grams/267.8 + ^{239}Pu grams/234.7 $\leq$ 1

This conserves the amount of fissile mass allowed in the NRBK-41 while allowing flexibility in overall fissile material composition that is loaded.

The applicant used the RCP01 Monte Carlo computer program version 899 and utilized cross-sectional data from the ENDF/B-IV data to model the NRBK-41 cask system for both normal and hypothetical accident conditions. The fuel was treated as a homogenous mixture of fuel and moderator and the interior stainless steel of the MIN-41 and HIP-41 inner container was omitted to provide additional conservatism. Additionally, the fuel mixture does not include any cladding, structural or packaging material that would likely be present in an actual shipment that would only serve to act as poisons in the calculations. Furthermore, the stated dimensions of both the MIN-41 and HIP-41 inner containers limit the H/X ratio to less than the optimum minimum critical mass/volume conditions.

Although the applicant did not specifically address a single package, a single package is bound by the hexagonal infinite array analysis the applicant did perform. In addition, for each potential loading configuration, the maximum allowable mass of material per package is always less than the minimum critical mass for each loaded fuel configuration.

The applicant evaluated a variety of scenarios in determining the maximum k_{eff} for the NRBK-41 cask, and in all instances the resultant effective multiplication factor was less than 0.95 for all NCT and HAC scenarios. The maximum calculated k_{eff} was 0.92 for an infinite, optimally moderated array of NRBK-41 casks each loaded with 350 grams of equivalent ^{235}U.

6.2 Criticality Evaluation

NRC staff performed confirmatory calculations using the SCALE 5.1 code package with v6-238 cross-sections. The results of these confirmatory calculations are consistent with those performed by the applicant.

6.3 Evaluation Findings

Based on the NRC staff verification of adequate system modeling by the applicant and supported by the analyses provided with this amendment, staff finds that the NRBK-41 package as amended continues to meet the requirements of 10 CFR Part 71.

7.0 MATERIALS

7.1 Materials Design

By application dated November 5, 2007, and supplemental information dated February 22, 2008, the U.S. Department of Energy – Naval Reactors submitted amendment requests to the U.S. Nuclear Regulatory Commission (NRC) for issue of a revised and renewed Certificate of Compliance (CoC) No. USA/9221/B(M)F-96 (DOE-NR). These amendments, proposed revision 7a to the NRBK-41 safety analysis report for packaging (SARP), proposed a design change to both inner containers MIN-41and HIP-41, addressed issues considered during the rule making process for transportation safety standards and updated format.

Revision 7A to the SARP for the NRBK-41 Radioactive Material Shipping Container requests two changes. The first request is a design change to the primary containment inner containers, the MIN-41 and HIP-41 leak penetration to make the inner containers leak tight following the Hypothetical Accident Condition (HAC) thermal tests.

The second request is a "-96," which involves addressing 19 issues, for applicability, that were considered during the rule making process for the current rules in Title 10 of the Code of Federal Regulations, Part 71. In addition, issues identified in previous Part 71 rulemakings would have to be considered as well. No material issues were identified or applicable to this request.

7.2 Materials Review

The NRC staff submitted a request for additional information (RAI) identifying additional information needed in connection with its material review of the amendments. Each individual RAI described information needed by the staff to complete its review of the application, proposed revision 7a to the final SARP and to determine whether the applicant demonstrated compliance with the regulatory requirements. NUREG-1609, "Standard Review Plan for Transportation Packages for Radioactive Material," was used by the staff in its review of the amendment application.

By letter dated November 14, 2008 (S#08-04186) the U.S. Department of Energy – Naval Reactors submitted response to the NRC RAI's on the SARP for the NRBK-41 radioactive material shipping container.

For shipments of the NRBK-41 container, the MIN-41 or HIP-41 inner container acts as the primary containment boundary for irradiated fuel, actinides, fission products, and crud adhering to test specimens and waste material. Both Inner containers have a leak test penetration that is sealed with a small pipe plug. This leak test penetration was unable to meet the leak-tight criteria after the HAC thermal test. Chapter 4, Appendix 4.5.2, discusses the elevated temperature testing performed on the new design.

The housing of the MIN-41 container is a seamless stainless steel pipe. The bottom of the container is formed by a plate welded inside the end of the housing. The top closure consists of a metallic C-ring which is installed on a step in the housing. A flat cap plate, used to compress the C-ring, incorporates a leak test fitting and leak test plug that seals off the leak test penetration. A plug assembly mates with the upper end of the housing by means of a breech-lock arrangement. Eight screws are threaded through the plug assembly then tightened against the cap to compress the C-ring and secure the plug assembly within the housing. Eight jam nuts are used to prevent loosening of the screws.

The housing of the HIP-41 inner container is essentially a hollow cylinder and is fabricated from a high strength steel casting. The inner cavity at the bottom of the container is hemispherical (concave). At the upper end of the container, a stainless steel closure plug is inserted until it seats on a tapered area in the body and is retained by an externally threaded stainless steel retaining ring that threads into the top of the container body. The closure plug is equipped with double ethylene-propylene O-rings which provide the containment boundary for the container assembly when the closure plug is installed in the container body. The closure plug also contains a leak test fitting and leak test plug that seals off a leak test penetration.

For the MIN-41 inner container a test penetration through the closure cap (standpipe) is provided for purging and pressurizing the container with helium after the closure cap has been installed for shipment. The test penetration is sealed by metal-to-metal contact between a welded on leak test fitting and mating plug. The stainless steel male connector leak test fitting is welded to the top of the standpipe. The leak test fitting plug is installed onto the fitting to provide the metal-to-metal seal for shipment. (See Figure 1.3-51, all sheets of the SAR.)

For the HIP-41 inner container, a test penetration, sealed by a metal-to-metal contact between a welded on leak test fitting and mating plug center to permit leak testing after the container is assembled for shipment. The penetration is provided in the containers closure plug for leak testing. The leak test penetration in the closure plug is sealed by a stainless steel male connector leak test fitting welded to the top of the closure plug. The leak test fitting plug is installed onto the fitting to provide the metal-to-metal seal for shipment. (See Figure 1.3-45 of the SAR.)

See Figures 1.3-43 and 1.3-45 in reference to the HIP-41 inner container and Figures 1.3-48 and 1.3-51 in reference to the MIN-41 inner container design change. Both inner containers utilize item CRES-316 male (pipe) connector (i.e., leak test fitting) threaded into and seal welded with CRES-316 filler material to their respective leak test penetrations. The leak test penetration item for the HIP-41 inner container is referred to as the closure plug and for the MIN-41 is referred to as the cap. Both male connectors require items Primer N and Adhesive to be applied to threads prior to installing the item plug (i.e., female plug used to blank pipe connector).

Following installation of the male (pipe) connector into its respective leak test penetration and before seal welding a minimum thread engagement is required followed by removal of excess threads above the top of the leak test penetration, required by machining.

The staff requested additional information concerning the basis for allowing localized loss of heat affected zone properties as acceptable for the seal weld stated in note 6 of Figure 1.3-51 (sheet 2 of 3) for the MIN-41, Inner Container and note 6 of Figure 1.3-45 for the HIP-41, Inner Container. When changes from the base metals exposure to heat are combined with the reduction in cross-section of any notched area, the mechanical strength is greatly reduced. This may be critical in applications that involve impact, low temperature or fatigue conditions. This information is needed to ensure compliance with 10 CFR 71.71 and 71.73.

DOE-NR responded that the weld is used to create a seal between the Swagelok SS-100-1-1 fitting and the MIN-41 cap and HIP-41 plug. A small heat affected zone does form in the region around the weld. The material in the heat affected zone can have material properties, such as the yield strength and the ultimate strength, that vary from the surrounding material. For example, heat treatments such as the precipitation hardening of the 17-4PH of the MIN-41 cap can be lost in the heat affected zone. However, this region is not highly loaded and mock-up welded pieces have been destructively examined to show that loss of the heat treatment properties in this region is minimal.

The region where the weld is deposited (MIN-41 cap as example) is on the end of the leak test “stand pipe.” The stand pipe is a 0.438-inch diameter cylindrical projection extending 0.910-inches from the flat top of the cap. When the weld is applied, the heat must travel through the stand pipe before reaching the main body of the cap where the material properties are required. The cap material properties are required in the flat portion of the MIN-41 cap to resist bending due to the internal pressure and the support configuration. During evaluation of the Swagelok fitting modification, mock-up pieces were produced of the arrangement. These mock-ups were sectioned and prepared for examination.

Knoop Hardness test results provide an indirect, approximate measure of the yield strength, the figure provided show that the hardness, and thus the yield strength, of the 17-4PH material located outside of the heat affected zone. See Enclosure 1 to DOE letter S#08-04186.

Additionally, the stand pipe does not need to have the same material properties to resist the hoop and axial stresses developed due to the internal pressure as are needed in the main body of the cap to resist bending moments. Shear stress in the fitting threads and shear stress in the leak test fitting body due to the internal pressure during the hypothetical fire accident are calculated in Section 2.10.5.4.2.6 and 2.10.5.4.2.7 of the SARP, respectively. These stresses are low (less than 3000 psi) and do not require full material properties of the fitting or the stand pipe. Therefore, the potential lower material properties surrounding the weld are acceptable.

The staff recognizes the Heat Affected Zones (HAZs) surrounding the seal weld area as affecting a relatively small amount of standpipe base material and that the standpipe does not resist the same hoop and axial stresses exerted on the MIN-41 cap/HIP-41 plug. Hardness testing of welds and their HAZs usually requires testing on a microscopic scale. Hardness testing of welds provides an indication of two parameters, strength and microstructure, significant to the determination of a successful weld joint. The tensile strength of steel is proportional to its hardness and for a given steel composition the hardness measured is related to its microstructure.

One factor that can influence the resultant hardness is weld heat input. Welding can impose a variety of thermal cycles on steel at various locations that produce undesirably hard microstructures susceptible to cracking and brittle fracture or excessively soft microstructures susceptible to plastic collapse under load. The hardness can therefore be a useful indicator to determine if the thermal cycle induced by welding has rendered the HAZ adjacent to the weld susceptible to cracking or plastic collapse. The hardness testing results provided appear to be consistent with Knoop hardness values for 17-4PH base material properties.

The staff accepts the decision for allowing localized loss of HAZ properties surrounding the seal weld based on the statements and testing provided within the DOE response to RAIs.

Prior to installing the leak test fitting plug onto the male connector leak test fitting for the MIN-41 and HIP-41 inner containers, a Primer N and Adhesive is applied to the male threads of the male connector. (See Figure 1.3-48 and Figure 1.3-43 of the SAR, respectively.)

Chapter 4, Appendix 4.5.4, evaluates the effects of elevated temperatures on the MIN-41 and HIP-41 leak test penetration seals. The external NRBK-41 cask body absorbs much of the energy from the regulatory 1475°F hypothetical fire accident, however, the remaining energy is transferred to the inner container such that the maximum temperature reaches 602°F during the thermal transient.

The staff requested additional information concerning what detrimental affect the degradation of the Primer N and Adhesive would have on the leak tightness, form, fit, or function of the mating plug to male connector post-hypothetical accident (fire) conditions should the maximum temperature reach in excess of 600 degrees F. Specifications for the Adhesive (Loctite 5772 Thread Sealant) states a temperature range of minus 65 degrees F to plus 300 degrees F. Furthermore, no temperature range specifications for Priner N (ASTM D 5363) were provided within the SARP. This information is needed to ensure compliance with 10 CFR 71.33 and 71.73(c)(4).

DOE-NR responded that the Primer N and the adhesive are not used as part of the containment boundary for the NRBK-41. Therefore, there is no adverse effect from degradation of these materials. Both the drawing and the equipment technical manual provide direction to apply the primer and adhesive only to the external threads of the fitting. Therefore, the rest of the male fitting will be clean and free of primer and adhesive.

It is the Swagelok Male Fitting and Plug that seal the leak test penetration of both the MIN-41 and HIP-41 containers. The primer and adhesive are not part of the containment boundary. Since a "metal-to-metal" seal is formed using the Swagelok Male Fitting and Plug, the Primer N and Adhesive items are only used to prevent unintentional removal of the mating Plug. As such, degradation of this material does not impact the leak tightness, form, fit, or function of the mating Plug. The only potential for impact from degradation of these materials is that the primer and thread locking compound may no longer prevent unintentional removal of the leak test plug.

It should be noted that breakdown of Primer N and/or the Adhesive in the past under normal conditions of transport has never resulted in any other detrimental effect, such as increased corrosion, on any of the other components in the NRBK-41. In addition, American Standard for Testing and Materials (ASTM) D 5363 does not clearly specify a maximum temperature for Primer N; however, this information is not needed to determine the safety posture of the NRBK-41.

The staff recognizes that the threaded connection does not form the critical "metal-to-metal" containment seal for both MIN-41/HIP-41 inner containers and acknowledges the service record of the Primer N and Adhesive under normal operating conditions. The "metal-to-metal" contact is formed between the Swagelok SS-100-1-1 Male Fitting and the Swagelok SS-100-P Plug. A junction is created when the upper tapered opening of the Male Fitting accepts the Ferrule of the Plug as the Plug is threaded onto the Male Fitting. This region also produces a primary containment system boundary of both the MIN-41 and HIP-41 inner containers intended to retain radioactive material during transportation. The staff accepts that no considerable detrimental affect will result from the degradation of the Primer N and Adhesive that would have an affect on the leak tightness, form, fit, and function of the Swagelok SS-100-1-1 Male Fitting to Swagelok SS-100-P Plug threaded connection post-hypothetical accident (fire) conditions.

A helium source is connected to the leak test penetration and a mass spectrometer leak detector attached to the leak test fitting and plug. Following successful leak test at room temperature, the cap is placed in a furnace and exposed to thermal profile simulating the hypothetical fire accident. Following cool down of the cap the cap is again leak tested as described above. The entire process is repeated two additional times to demonstrate repeatability. As summarized in Chapter 4, Appendix 4.5.4, Table 4.5.4-2, results of pre- and post-thermal exposure leak tests demonstrate that there is no significant deterioration in the leak test penetration seal capabilities following the hypothetical fire accident condition.

7.3 Evaluation Findings

Naval Reactors requested that the NRC review proposed Revision 7A of the NRBK-41 SARP. The NRC responded with RAIs. Satisfactory responses to NRC RAIs on Revision 7A of the NRBK-41 SARP (U) materials review are in Enclosure 1 to DOE letter S#08-04186.

The staff has reviewed Revision 7A design change to the SARP (materials) for the NRBK-41 shipping cask system. Based on the statements and representations contained in the SARP and the condition given in the Certificate of Compliance, the staff concludes that the NRBK-41 shipping cask system meets the requirements of 10 CFR Part 71.

8.0 PACKAGE OPERATIONS

Changes to this section include the addition of a secondary containment boundary pre-shipment leak test and inclusion of steps to ensure the package is operated consistent with maintaining occupational radiation exposures as low as reasonably achievable as required by 10 CFR 20.1101(b). The loaded or empty NRBK-41 cask is leak tested using the gas pressure rise test method per ANSI N14.5-1997, American National Standard for Radioactive Materials – Leakage Tests on Packages for Shipment to an acceptance leak rate of 1 x 10^{-3} ref-cc/sec or less.

9.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

Changes to this section include the NRBK-41 cask inner containers receiving complete dimensional and surface finish examinations following fabrication to ensure they meet dimensions and tolerances specified on design drawings.

CONDITIONS

The following conditions in CoC No. 9221, Revision No. 6, have been revised as follows:

Condition 5(b)(1), “Type and form of material,” was revised to delete that for any contents exceeding a Type A quantity, the radioactive material must be contained within a specimen with intact, undamaged cladding.

Condition 5(b)(2), “Maximum quantity of material per package,” was revised to delete that the total quantity of radioactive material in the form of loose surface contamination within the package must not exceed a Type A quantity.

Condition 8 was added to state transport by air of fissile material is not authorized.

Condition 9 was revised to include the new expiration date as April 30, 2013.

CONCLUSION

As requested by the application, CoC No. 9221 for the NRBK-41 package has been amended, designated to “-96”, and renewed to April 30, 2013, with one additional condition stating transport by air of fissile material is not authorized.

Based on the staff’s review, the statements and representations in the application, as supplemented, and for the reasons stated in this SER, and with the conditions listed above, we conclude that these changes will not affect the ability of the package to meet the requirements of 10 CFR Part 71.

Issued with Certificate of Compliance No. 9221, Revision No. 6,
on April 9, 2009.