SAFETY EVALUATION REPORT

Docket No. 71-9300
Model No. BRP RVP SAR-5339 Package
Certificate of Compliance No. 9300
Revision No. 0

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SUMMARY

By application dated June 19, 2001, as supplemented August 21 and December 19, 2001, BNFL, Inc., requested that the Nuclear Regulatory Commission approve the Model No. BRP RVP SAR-5339 package as a Type B(U)-85 package for the transport of the irradiated reactor vessel and internals from the Big Rock Point plant to a disposal facility.

The package consists of the decommissioned reactor vessel and internals, contained within an outer steel packaging that is welded closed and filled with low density concrete. The overall package dimensions are 25 feet in length and 13 feet in diameter. The package weighs approximately 565,000 pounds, as prepared for transport. The package is designed for a single use trip from the Big Rock Point plant to the low level radioactive waste disposal facility near Barnwell, South Carolina. The package will be shipped primarily by rail.

The package was evaluated against the regulatory standards in 10 CFR Part 71, including the general standards for all packages and the performance standards under normal conditions of transport and hypothetical accident conditions. The structural, thermal, containment and shielding evaluations in the safety analysis report showed that under normal conditions of transport and hypothetical accident conditions, the package would maintain external radiation levels and the release of radioactive materials within applicable regulatory limits.

NRC staff reviewed the application using the guidance in NUREG-1609, "Standard Review Plan for Transportation Packages for Radioactive Material." Based on the statements and representations in the application, as supplemented, and the conditions listed below, the staff concluded that the package meets the requirements of 10 CFR Part 71.

References

BNFL, Inc., application dated June 19, 2001.

Supplements dated August 21 and December 19, 2001.

1.0 GENERAL INFORMATION

1.1 Package Description

The package consists of a reactor pressure vessel and internals within a cylindrical steel outer packaging which is filled with low density concrete and welded closed. The reactor vessel is an open, cylindrical, 5-1/4-inch thick carbon steel vessel with a hemispherical bottom head. The inner diameter of the reactor vessel is approximately 8 feet, 10-7/8 inches, and the length of the reactor vessel is approximately 24 feet. The reactor vessel contains internal components, including the top guide plate, grid bar end pieces, steam baffle, emergency cooling sparger, seal housing, thermal shield, thermal shield retainer, seal weights, core support plate, inlet diffuser and inlet baffle. The internal components are constructed of stainless steel, and the internal surfaces of the reactor vessel are clad with stainless steel. The reactor vessel is partially surrounded with 3-inch thick stainless steel mirror insulation.

The outer packaging is composed of a 3-inch thick cylindrical shell welded to a 4-inch thick bottom plate, and is constructed of ASME SA-516, Grade 70, carbon steel. Supplemental steel shielding approximately 4 inches thick is welded to the inside of the cylindrical shell at the core beltline. The 4-inch thick top plate is attached to the reactor vessel flange by 14 studs and nuts, and is welded to the packaging outer shell. The reactor vessel cavity and the region between the reactor vessel and the outer packaging are filled with low density concrete. The overall dimensions of the package are approximately 25 feet in length and 13 feet in diameter. The package weighs approximately 565,000 pounds.

1.2 Contents

1.2.1 Type and form of material

Irradiated steel reactor pressure vessel, mirror insulation, and internal components.

1.2.2 Maximum quantity of material per package

Greater than a Type A quantity of radioactive material contained in an irradiated reactor vessel, mirror insulation, and internal components and associated surface contamination. Fissile material may be present provided the fissile material meets the exemption standards in 10 CFR 71.53.

1.3 Drawings

The package is constructed and assembled in accordance with BNFL, Inc., Figure 2-1, Sheets 1 through 4, Rev. 0; Sheet 5, Rev. 1; and Sheets 6 and 7, Rev. 0; included in the application dated June 19, 2001, as supplemented December 19, 2001.

2.0 STRUCTURAL

The objective of the structural review is to ensure that the information presented in the application regarding the structural performance of the package is acceptable, complete, and demonstrates that the package structural design meets the requirement of 10 CFR Part 71. The information includes: description of the package, design and fabrication criteria, structural material properties, and structural performance of the package under normal conditions of transport and hypothetical accident conditions.

2.1 Description of Structural Design

The packaging is composed of a cylindrical steel shell, approximately 13 feet in diameter and 25 feet long, with welded steel flat top and bottom plates. The material of construction of the packaging is ASME SA-516, Grade 70, carbon steel. The package will be used for transportation and disposal of the Big Rock Point plant reactor vessel (RV) and some of its internals as described in the Section 1.2.5 of the application. The void space in the RV is filled with low density cellular concrete (LDCC) having a density range of 30 to 36 pounds per cubic foot, and the annulus between the RV and the outer packaging is filled with LDCC having a density range of 50 to 60 pounds per cubic foot. The LDCC prevents surface contamination migration, component shifting, and dose rate changes during transportation. The LDCC has a negligible contribution to the package stiffness. Thus, except for its weight, the LDCC is not included in the structural model.

The 4-inch thick package bottom plate is shop welded to the shell with a full penetration weld. A donut shape support structure at the bottom of the package is designed to support the bottom of the RV when the package is in both horizontal and vertical positions. The package top cover is a 4-inch thick circular plate. As part of the loading operation, the top plate is attached to the RV flange by 14 existing SA-193 B7 RV head studs with new cap nuts. After the RV is loaded into the outer packaging, the top plate is field welded to the package shell with a full penetration weld.

The package shell thickness varies along the length of the package to satisfy radiological shielding and structural design requirements. The cylindrical package shell is 3 inches thick with an additional 4-inch thick plate welded to its interior surface for a length of approximately 8 feet in the former core region of the RV. The outer packaging shell therefore provides 7 inches of steel for radiation shielding at the RV region of highest radioactivity.

Stress intensity values for Class 1 of the ASME Boiler & Pressure Vessel Code, Section III, were used for the design of the package. These values contain assumptions about materials, fabrication, and examination. Consequently, the package pressure retaining materials, fabrication, and examination will be in accordance with Subsection NB of the ASME Code. The design of the package under normal conditions of transport is based on linear-elastic structural analysis. For the hypothetical accident conditions, the package is evaluated using nonlinear elastic-plastic dynamic finite element analyses. The nonlinear elastic-plastic dynamic analyses were used to predict the permanent deformation and the extent of local structural failure of the package caused by the 30-foot free drop. The nonlinear elastic-plastic dynamic analysis approach was acceptable because the package can experience severe structural damage or

failure and still meet the regulatory requirements for external radiation and radioactivity release limits specified in 10 CFR 71.51(a)(2). To demonstrate that the package meets the acceptance criteria under hypothetical accident conditions, the applicant considered a set of worst case damage scenarios. These non-mechanistic scenarios were considered conservative, since they bound the structural damage that was predicted based on the structural analyses. The increase in external radiation and the possible release of radioactivity were evaluated for the non-mechanistic scenarios and were shown to be acceptable. The non-mechanistic scenarios evaluated were:

- A 1-inch wide circumferential gap forming in the 7-inch thick package shell, exposing a 1-inch wide band of the RV and its insulation along the package beltline, in the core region of the RV.

- A 1-inch wide by 48-inch long longitudinal gap forming in the package, exposing a 1-inch by 48-inch longitudinal band of the RV and its insulation, centered on the package beltline, in the core region of the RV.

- A 6-inch diameter hole through the 7-inch thick package shell, exposing a 6-inch diameter area of the RV and its insulation, centered on the package beltline, in the core region of the RV.

- The top plate completely separated from the package.

The structural analyses predicted that the total damaged area of the package due to the hypothetical accident condition free drop and puncture would be less than any of the above scenarios. Thus, compliance with the radiation and radioactivity release limits specified in 10 CFR 71.51(a)(2) was assured. The containment evaluation was described in Chapter 4 of the application.

The staff reviewed the package description presented in the General Information and Structural Evaluation sections of the application and found that the regulatory requirements of 10 CFR 71.33 for package description were adequately addressed.

2.2 Mechanical Properties of Material

The temperature-dependent mechanical properties presented in Tables 2-3 and 2-4 of the application are based on the values given in ASME Code, Section II, Part D. Similarly, the temperature-dependent design stress intensity and yield strength for the SA-193 B7 bolting material was presented in Table 2-5 of the application. For the fire accident analysis, the material properties at elevated temperatures were derived from the temperature trend curve given in ASCE Manual No. 78, "Structural Fire protection" and presented in Tables 2-6 and 2-7 of the application. Fracture toughness considerations were based on the provisions of Regulatory Guide 7.11 and NUREG/CR-1815. The staff found that the mechanical properties of materials were based on acceptable industry standards.

2.3 General Standards for All Packages

2.3.1 Minimum Package Size

The smallest overall dimension of the package is the 13-foot outside diameter; therefore, the minimum package size requirement of 10 CFR 71.43(a) is met.

2.3.2 Tamper Indicating Device

The package has welded closures, seal welds on all threaded plugs, and seal welds on the RV attachment cap nuts. When the welds are intact, it is an indication that the package has not been opened. Thus, the requirement of 10 CFR 71.43(b) is satisfied.

2.3.3 Positive Closure

The package has a welded containment system. All threaded plugs and the RV attachment cap nuts are seal welded. Unintentional opening will be precluded, thereby satisfying the positive closure requirement of 10 CFR 71.43(c).

2.3.4 Chemical and Galvanic Reactions

The containment boundary and internal support structure are constructed of welded carbon steel. The package contents consist of carbon steel, stainless steel, and low density cellular concrete (LDCC). The LDCC is a Portland cement-based mixture that acts as an insulator between the dissimilar metals and eliminates galvanic or chemical reactions through direct contact or coupling of stainless and carbon steels. There are no chemical reactions between Portland cement mixtures and ferrous metals. The LDCC will ensure an essentially dry and benign environment inside the containment. Thus, the requirements of 10 CFR 71.43(d) are met.

2.3.5 Valves

The package does not contain valves or other pressure relief devices. Containment penetrations will be closed and seal welded, thus precluding escape of the contents. The design of the package meets the requirements of 10 CFR 71.43(e).

2.3.6 Package Design

The package is constructed in accordance with the requirements of the ASME Code, Section III, Division 1, Subsection NB. The evaluation of the package under normal conditions of transport showed that the requirements of 10 CFR 71.43(f) are satisfied (see Section 2.5 of this Safety Evaluation Report).

2.3.7 External Temperature

Compliance with the accessible surface temperature limits of 10 CFR 71.43(g) was demonstrated by the thermal evaluation of the package in Section 3.1.3.1 of the application.

2.3.8 Venting

The package is a welded container with no feature for continuous venting during transport. Thus, the package design meets the requirement of 10 CFR 71.43(h).

2.4 Lifting and Tie-Down Devices

The lifting and tie-down devices for the BRP RV package are designed such that they are not structural parts of the package and will be removed before transport. Therefore, their design was not evaluated in the application for package approval. The criteria of 10 CFR 71.45 do not apply to the package.

2.5 Normal Conditions of Transport

The package evaluation was performed to demonstrate that the package is structurally adequate to meet the performance requirements of Subpart E of 10 CFR Part 71. The structural evaluation was performed using finite element analyses. The finite element computer code ANSYS, Version 5.6, was used to analyze the package for the loading conditions of normal conditions of transport (10 CFR 71.71).

2.5.1 Heat

The package was evaluated for an ambient temperature of 100°F with a total insolation of 400 g-cal/cm^2 on the horizontal curved surfaces, and 200 g-cal/cm^2 on the vertical flat surfaces for duration of 12 hours, as specified in 10 CFR 71.71(c)(1). The dead weight (1 g inertial loads in the vertical direction) and the maximum internal design pressure (20 psig) were applied on the package, which was assumed to be positioned horizontally on saddles during transport. The analysis results showed a minimum safety margin of stress intensity of 6.4 (Table 2-8 of the application). Thus, differential thermal expansion of components and the design internal pressure under the heat condition would not adversely affect the package performance.

2.5.2 Cold

The ambient temperature was taken as -40°F with no insolation or decay heat for the evaluation of the package for the cold temperature condition. Dead weight (1 g inertial loads in the vertical direction) and the minimum internal pressure (-3.3 psig) were applied on the package, which was assumed to be positioned horizontally on saddles during transport. The finite element analysis results showed a minimum safety margin of stress intensity of 5.4 (Table 2-8 of the application). Thus, differential thermal contraction of components and the minimum internal pressure under the cold temperature condition would not adversely affect the package performance.

2.5.3 Increased External Pressure

An increased external pressure of 20 psia was combined with the package dead weight and the minimum internal pressure. The analysis results for this loading condition showed a minimum safety margin of stress intensity of 5.6 (Table 2-8 of the application). Thus, the increased external pressure requirement of 10 CFR 71.71(c)(4) would not adversely affect the package performance.

2.5.4 Reduced External Pressure

A reduced external pressure of 3.5 psia was combined with the package dead weight and the maximum internal pressure (20 psig). The analysis results for this loading condition showed a minimum safety margin of stress intensity of 4.6 (Table 2-8 of the application). Thus, the reduced external pressure requirement of 10 CFR 71.71(c)(3) would not adversely affect the package performance.

2.5.5 Vibration

The package was evaluated for vibration incident to transport. The tie-down design criteria of the Association of American Railroads, "Open Top Loading Rules Manual," Part 1, dated March 1, 2000, and American National Standards Institute, ANSI N14.2 (draft) "Tie-Down for Truck Transport of Radioactive Materials," were used to evaluate the effects of vibration on the package. The enveloping vertical, transverse, and longitudinal accelerations considered in the evaluation were:

- Vertical direction: 2.0 g
- Transverse direction: 2.0 g
- Longitudinal direction: 3.0 g

Each shock load was analyzed independently. Shock acceleration was applied to the package, which is secured on saddles with cable ties and axial bumpers. The results are shown in Table 2-8 of the application. The minimum safety margin against the shock loads was 1.9 (Table 2-8, Load Case N7). The normal condition shock fatigue evaluation of the package was performed based on the enveloped shock load stress intensity range. The allowable number of cycles was determined to be 1.38E+04 from the fatigue curve in ASME Code, Section III, Appendix I. This number of cycles was far above the estimated total number of full magnitude shocks of 1500 which was based on the assumption that the package may experience one full magnitude shock per one mile of travel. Thus, it was concluded that vibration incident to normal transport would not adversely affect the package performance, as required by 10 CFR 71.71(c)(5).

2.5.6 Water Spray

The RV package is fabricated from thick welded steel plate. All joints and openings are welded. Therefore, the water spray test specified in 10 CFR 71.71(c)(6) will have no effect on the package.

2.5.7 Free Drop

During transport, the package rests horizontally in cradles and is secured by tie-down cables and axial bumpers. The package is not lifted during transportation. Considering the package size and weight, the governing drop orientation for the 1-foot drop is horizontal. The flat, unyielding, horizontal surface in the 1-foot drop is considered to be a flat surface of an infinite thick concrete slab. ANSYS dynamic transient large deflection analysis was used to calculate the responses of the lumped parameter model of the package. The calculated peak impact acceleration was 12 g. The calculated peak g-load may then be applied to the three-dimensional finite element model of the package for stress recovery and buckling analysis. The application performed stress recovery for a postulated 5-foot drop due to a handling accident based on a calculated peak impact acceleration of 27 g. Thus, the 1-foot drop stresses were obtained by applying a linear scale factor to the handling drop stresses. The scale factor was 12 g / 27 g = 0.45. The resulting primary stress intensities for the 1-foot drop plus internal pressure loads are shown in Table 2-8 of the application. In addition, the buckling analysis showed that the package remains stable and no buckling occurs as shown by the convergence of the solution under the 1-foot drop impact load.

2.5.8 Corner Drop

The corner drop in 10 CFR 71.71(c)(8) is not applicable to the package because the package is not constructed of either fiberboard or wood and does not contain fissile material.

2.5.9 Compression

The compression test in 10 CFR 71.71(c)(9) is not applicable to the package because the package weighs more than 11,000 pounds.

2.5.10 Penetration

The package was evaluated for the penetration condition using the analytical method developed by the Ballistics Research Laboratories (BRL) and presented in ASCE Manual No. 58, "Structural Analysis and Design of Nuclear Plant Facilities." The recommended minimum design thickness calculated from the BRL formulation was 0.55 inch. Since the minimum thickness of the package is 3 inches, which is well over the recommended design thickness, the package shell thickness is adequate for resisting the required penetration loading.

2.6 Hypothetical Accident Conditions

2.6.1 Free Drop

The assessment of the package damage resulting from the 30-foot free drop was performed using nonlinear elastic-plastic dynamic impact analyses. The ANSYS element type SHELL181, a three-dimensional, four-node shell element, with both bending and membrane capabilities, was used to model the package cylindrical shell, top and bottom plates, RV, donut support, internal shielding plate, and ring plate. During the analysis, the package stresses were

monitored for element failure. Failure was considered to occur at any element for which the maximum principal stress exceeded the material tensile strength. The failed elements were removed from the FEA model, and the analysis process was continued until the package displacements reached their maximum values. The following drop orientations were considered:

- Vertical drop
- Horizontal drop
- Corner drop through the package center of gravity
- Oblique drop (slap down)

The results of the 30-foot drop dynamic impact analysis were verified using the principle of energy conservation. The total kinetic energy (KE) and the cumulative plastic work done must be approximately equal to the initial impact energy, $KE = 1/2\ MV^2$, where M is the package mass and V is the impact velocity. The verification showed that the total impact energy of $1/2\ MV^2$ was transformed into the total plastic work done at the final load step of the analysis, thereby demonstrating that the impact energy was fully dissipated through plastic deformation of the package.

During vertical and horizontal drop impact, the analysis results showed the peak value of the maximum principal stress was less than the material tensile strength of 70 ksi and the maximum stud stress intensity was less than the material tensile strength of 100 ksi. Thus, the package will remain intact and there will be no containment failure or gross buckling failure.

The maximum potential damage to the package during a corner drop through the package center-of-gravity occurred at the impact corner due to local crushing effects. The high stress location occurred at the shell junction with the donut support plate. During the impact, the peak value of the maximum principal stress reached 71 ksi, exceeding the material tensile strength of 70 ksi for a short duration before reducing to a value below the material tensile strength. A circumferential crack may be initiated at the outer fiber of the package shell near the junction of the donut support plate, but breach of containment is unlikely due to the short duration of this high stress and the thickness of the package shell. The package gross structure remained intact after the impact, and no buckling failure occurred in the package.

The damage caused by the oblique (slap down) drop was the most critical among the drop orientations considered due to a higher vertical impact velocity combined with a horizontal impact velocity at the package top plate. The analysis showed that during the impact, the peak value of the maximum principal stress reached the material tensile strength of 70 ksi at several locations along the top plate rim. The element stiffness at these locations was removed; the interactive analysis continued, and the adjacent elements continued to take loads without additional failure until the end of the impact duration. The analysis showed that the package gross structure remained intact after the impact. The maximum stud stress intensity was 97.73 ksi, which was less than the material tensile strength of 100 ksi. Thus, it would be unlikely that the top plate would separate completely from the package.

2.6.2 Puncture

The applicant considered two puncture locations: (1) near the mid-shell, between the top plate and the internal shield plate, and (2) on the top plate next to the damaged section of the rim. The analysis showed that the package deformed at the puncture location of the steel bar but that the maximum principal stresses in the package, except at the puncture location, remained within the material tensile strength of 70 ksi. Therefore, the 40-inch puncture onto a steel bar would not cause any gross structural failure of the package.

2.6.3 Thermal

Under fire test conditions, differential thermal expansion occurs between the package and the RV in both longitudinal and radial directions. Because the RV is enclosed inside the package shell, and the package shell expansion is greater than the RV, there is no binding due to thermal expansion. The thermal response of the package when exposed to the fire test was evaluated in Chapter 3 of the application. The resulting temperature distribution of the package was then used for the structural analysis. The analysis included the maximum internal pressure of 95 psig under hypothetical accident conditions. The analysis showed that the maximum stress intensity in the package was less than the material tensile strength and therefore there would be no additional damage to the package structure.

2.6.4 Immersion - All Packages

The package was subjected to an external pressure equivalent to immersion under 50 feet of water (21.7 psig) as specified in 10 CFR 71.73(c)(6). The analysis was performed on an undamaged package as allowed by 10 CFR 71.73(a). Combining the external pressure of 21.7 psig with the minimum internal pressure of -3.0 psig (at a cold ambient temperature of -20°F), the equivalent external pressure was 24.7 psig. The analysis showed an allowable external pressure of 385 psig for the package. Thus, the package meets the 10 CFR 71.73(c)(6) requirements for the water immersion condition with a large safety factor.

2.6.5 Fracture Toughness Considerations

The provisions of NRC Regulatory Guide 7.11 and NUREG/CR-1815, "Recommendations for Protecting Against Failure by Brittle Fracture in Ferritic Steel Shipping Containers Up to Four Inches Thick," were used to determine the nil ductility transition (NDT) temperature for the package plate material. The NDT temperature for SA-516, Grade 70, steel is approximately -20°F. However, based on a lowest service temperature of -20°F, the maximum NDT temperature for the 4-inch thick top and bottom plates was only -35°F. To raise the NDT temperature from -35°F to -20°F, the lowest service temperature must be increased from -20°F to -5°F. The applicant has proposed that operational controls be included that assure that the package is not transported unless the minimum ambient temperature predicted during the shipment is at least 0°F. This includes an added safety margin against brittle fracture. The package approval is conditioned to limit transportation based on this consideration.

2.7 Appendix

Appendix 2-1 summarized the calculations performed to document the structural evaluation of the package.

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 design has adequate structural integrity to meet the requirements of 10 CFR Part 71.

3.0 THERMAL

3.1 Package Description

The Big Rock Point reactor vessel package consists primarily of the reactor vessel (RV), some remaining internals, an outer steel cylindrical shell with top and bottom closure plates, and low density cellular concrete (LDCC) inside the RV and between the package shell and the RV. Since the package is filled with LDCC and no significant air gaps are present, convection and radiation heat transfer inside the package are negligible and the primary heat transfer mechanism is conduction through the LDCC, RV, internals, and the package shell.

3.2 Decay Heat and Thermal Analysis

The decay heat of the package was calculated to be 485 Btu per hour, largely contained in the core region of the thermal shield and RV. The derivation of the radioactive source term is described below in Section 4.2 of this Safety Evaluation Report. The applicant performed a thermal analysis of the package using the HEATING 7.2f finite difference thermal analysis code. Under normal conditions of transport, the package was assumed to have the emissivity of rolled steel, $\varepsilon = 0.66$. In still air at 100°F with no solar insolation, the steady state package surface temperature was determined to be 103.8°F, meeting the requirement of 10 CFR 71.43(g) for exclusive use shipments.

3.3 Thermal Evaluation Under Normal Conditions of Transport

For the maximum package temperature under normal conditions of transport, the applicant modeled the package in still air at 100°F with 12 hours of insolation, 400 g-cal/cm^2 on the curved surfaces and 200 g-cal/cm^2 on the vertical end plates, as required by 10 CFR 71.71(c)(1). The maximum temperatures calculated by the applicant under these conditions were approximately 191°F which occurred both at the point in the package corresponding to the center of the active fuel region and on the package surface. For the cold conditions under normal conditions of transport, the applicant modeled the package in an ambient temperature of -40°F, as required by 10 CFR 71.71(c)(2), and calculated a minimum temperature of -39.4°F. The maximum normal operating pressure calculated under normal conditions of transport was 17.9 psig, taking into account the vapor pressure due to residual water, the buildup of gases generated through radiolysis, and elevation changes along the

transportation route. The structural analysis in Chapter 2 of the application showed that the temperatures and pressures in the package under the normal conditions of transport heat and cold requirements would not result in a loss or dispersal of radioactive contents, increase in external radiation levels, or reduction in the effectiveness of the package.

3.4 Thermal Evaluation Under Hypothetical Accident Conditions

Under hypothetical accident conditions, the applicant's model was identical to that described above for normal conditions of transport, except the boundary conditions were changed to match those prescribed in 10 CFR 71.73(c)(4). Potential damage resulting from the free drop and puncture tests prescribed in 10 CFR 71.73(c)(1) and (3) was considered minimal and of no significant effect in the thermal analysis. The package emissivity was changed to 0.9 during the 30-minute fire transient, and then returned to 0.66 at the end of the 30-minute period. A convection heat transfer coefficient of 2.6 Btu/hr-ft^2-°F was applied to all surfaces during the fire transient. This coefficient was twice that calculated by the applicant for forced convection by hot combustion gases during the fire. The applicant used initial conditions resulting from the analyses under normal conditions of transport at ambient temperatures of 100°F and -20°F, without solar insolation. These same initial conditions were applied after the 30-minute fire transient. The resulting maximum temperatures are given in Table 3-3 of the application for specific package materials and locations. The maximum temperature of the LDCC was 1355°F in the higher density LDCC near the outer shell of the package, and the maximum temperature of the steel was 1458°F on the package surface. The maximum pressure calculated during the hypothetical accident condition fire test was 95.9 psig, taking into account the vapor pressure due to residual water, the buildup of gases generated through radiolysis, and elevation changes along the transportation route. The applicant's structural analysis in Chapter 2 of the application showed that the temperatures and pressures in the package under the hypothetical accident condition fire test do not cause any additional damage to the package or result in a release of radioactive material or a significant increase in dose rates from the package.

3.5 Conclusion

The applicant has shown and the staff agrees that the package meets the accessible package surface temperature requirement of 10 CFR 71.43(g), and has adequate thermal performance to meet the requirements of 10 CFR 71.51(a) and (c).

4.0 CONTAINMENT

4.1 Containment System

The containment boundary of the package consists of the 3-inch thick cylindrical steel shell with 4-inch thick steel top and bottom covers. The shell and cover plates are fabricated from ASME SA-516, Grade 70 steel. The bottom cover is welded during fabrication and the top cover is welded after the RV is loaded at the Big Rock Point plant. The package has penetrations consisting of bolt holes for attaching the RV flange to the top plate, bolt holes for attaching the lifting lugs, and openings for filling the package with

LDCC. The RV attachment studs are covered by cap nuts and sealed. The holes for lifting lug attachment bolts and LDCC filling are plugged and seal welded prior to shipment.

4.2 Radioactive Contents

The radioactive material in the package is present primarily as large activated metal components and is not available for release from the package. The quantity of radioactivity that is present in the form of surface contamination was estimated as 2.9 curies. The predominant radionuclides were cobalt-60 (2.49 curies), iron-55 (0.2 curie), and nickel-63 (0.14 curie). Based on the mixture of radionuclides, the applicant calculated the effective A_2 value (3.62 curies) for the surface contamination. The effective A_2 value was larger than the quantity of surface contamination, therefore the package contains less than an A_2 quantity of releasable radioactivity. The radioactivity was based on decay to September 1, 2002.

The quantity and distribution of the radionuclides in the reactor vessel package was determined by performing an activation analysis on the components of the package. The ANISN one-dimensional discrete ordinates neutron transport code was used to determine the neutron flux levels in the various RV components, and then ORIGEN2 was used to determine the activation of RV components over the 29 operating cycles experienced at the Big Rock Point plant. The total radioactivity in the RV components was calculated as less than 13,100 curies. The surface contamination of the RV and its components was determined by measuring a metal sample from the core spray nozzle. Since the core spray nozzle was in a location of relatively low neutron fluence, the activity associated with the sample was assumed to be from surface contamination. The activity per unit surface area determined from this measurement was then multiplied by the total surface area of the internals and the internal surface area of the reactor vessel. The total surface contamination contained in the package was calculated to be 2.9 curies.

4.3 Containment System Evaluation

The containment system of the package is completely welded closed. The applicant evaluated the containment boundary under normal conditions of transport and hypothetical accident conditions. The applicant concluded that the design of the welds was adequate to ensure that they would remain intact and that the containment system would not be breached under normal conditions of transport. For accident conditions the applicant evaluated a set of non-mechanistic damage scenarios that included breaching the outer packaging. The evaluation included the potential release of available surface contamination and possible leaching of radioactivity into water, should the package become submerged during an accident. The results of this analysis showed that the release rate from the package would be less than an A_2 per week, as required by 10 CFR 71.51(a)(2). In addition, the void spaces within the reactor vessel are filled with solidified low density concrete that is intended to fix the radioactive material in place.

4.4 Flammable Gas Generation

The applicant provided an evaluation of the generation of flammable gases within the package during transport. The applicant used guidance provided in Electric Power Research Institute (EPRI) Publication NP-4938, "Methodology for Calculating Combustible Gas Concentration in Radwaste Containers," to calculate the hydrogen generation rate within the package. Hydrogen is generated by radiolysis of the water that is present in the concrete, and the rate of generation is based on the characteristics of the material and the amount of energy absorbed in the material. The quantity of hydrogen generated from the LDCC per unit of absorbed energy from the radiation was based on experiments performed for gamma irradiation of grout mixtures. The concentration of the hydrogen was then calculated based on the ability of the hydrogen to migrate throughout the vessel, and based on the estimated void volume within the concrete (approximately 50 percent). A five percent concentration of hydrogen was used as the lower limit for flammability. The applicant calculated that the hydrogen concentration within the reactor vessel would be no more than 3.7 percent by volume within one year following closure of the package.

4.5 Conclusion

Since the containment system has a welded closure with penetrations that are completely welded closed, and since the radioactivity present as surface contamination is fixed by solidified low density concrete, the staff agrees with the applicant’s conclusion that the containment provided by the package meets the requirements of 10 CFR 71.51.

5.0 SHIELDING

5.1 Package Description and Contents

The reactor vessel (RV) package is a 3-inch thick cylindrical steel shell welded to 4-inch thick top and bottom plates. The outer dimensions of the package are approximately 13 feet in diameter and 25 feet long. The package shell is 3 inches thick with an additional 4-inch thick shield plate welded to the interior surface of the cylinder shell. The interior shield plate is approximately 8 feet long and covers the reactor core region of the RV.

The package contents are normal form radioactive material. The package contains the RV, without the vessel head, some of the RV stainless steel mirror insulation, and stainless steel internal components. The list of internal components remaining in the RV package are listed in Section 1.2.5 of the application. Low density cellular concrete (LDCC) is used to fill the void space in the RV and the annulus between the RV and the outer packaging. The LDCC prevents migration of any remaining surface contamination, component shifting, and dose rate changes during transportation.

5.2 Source Specification

Surface contamination and activation of the large steel components in the RV are the two components of the source term. The total activity in the package is estimated to be less than 13,100 curies as of September 1, 2002 (the proposed shipping date of package). The total activity is based on measured dose rates, isotopic analyses of samples, and conservative assumptions. There are no gaseous or liquid radioactive components to the source term.

The inner wall surfaces and internals of the RV were chemically decontaminated in 1998 in an effort to remove as much removable contamination as possible. After the contamination process, a sample was taken from the wetted surface of the core spray nozzle. An isotopic analysis of the sample was performed and the isotopic mix from the analysis was increased by 50 percent to account for any variations in tritium and transuranic radionuclides. The final isotopic mix was multiplied by the total wetted surface area of the RV to obtain the total amount of remaining removable surface contamination. The total surface contamination remaining on the surfaces in the RV was estimated to be approximately 2.90 curies.

The activation source term was determined using the ANISN and ORIGEN2 computer programs. The ANISN code, using the SAILOR cross section library, was used to estimate neutron fluxes at various locations in the RV. The resulting fluxes were used as inputs to the ORIGEN2 code to determine the activation levels on individual RV components.

The principal gamma-emitting radionuclide in the package was cobalt-60, which accounted for approximately 99 percent of the total dose rate. No spent nuclear fuel or other neutron source material is present in the package. Therefore, there is no significant neutron source component to the source term or dose rate.

The source term evaluations are discussed in detail in Appendices 1-2 and 5-2 of the application. Based upon a review of the information provided in the application, as supplemented, including the Appendices, staff determined there is reasonable assurance the source term has been adequately developed and identified.

5.3 Shielding Evaluation

The ISOSHLD-PC computer code was used to evaluate dose rates from the package. Each shield transition and source component was explicitly modeled. Dose rates were calculated near shield boundary transitions and along axial and radial traverses so that dose rates would be calculated at the locations of highest values. Additionally, 1 curie source was added to the bottom of the RV to include any residual dross from component cutting and removal operations.

Potential streaming pathways were from the 14-inch diameter steam nozzle in the upper section of the RV and from the 20-inch diameter inlet nozzle in the lower section of the RV. However, all RV openings are filled with LDCC and capped with steel, thereby eliminating significant streaming pathways.

Gamma dose rates were calculated using the ANSI/ANS-6.1.1-1977 gamma flux-to-dose-rate conversion factors. Dose rates calculated for the RV package are listed in the following table.

Peak Dose Rates for Normal Conditions of Transport (mrem/hr)						
	Package Surface			2 Meters from Package Surface		
	Side	Top	Bottom	Side	Top	Bottom
Gamma*	52.8	16.2	1.16	7.31	8.67	0.44
10 CFR 71.47(b) Limit	200	200	200	10	10	10
* There is no neutron component to the dose rate.						

Staff reviewed the information in the application, as supplemented. The staff performed confirmatory analyses using the Microshield computer program. Staff's results were consistent with the dose rates in the application. Staff has reasonable assurance that the calculated dose rates are appropriate for this package and meet the requirements of 10 CFR 71.47.

5.4 Hypothetical Accident Conditions

The RV package was evaluated for hypothetical accident conditions as specified in 10 CFR 71.73. The structural evaluation provided in Section 2.7 of the application describes the computer program used to perform the analyses for the various accident scenarios. The results of the these analyses indicate that the potential damage to the package structure are acceptable. The evaluation of the structural review can be found above in Section 2.0 of this Safety Evaluation Report.

The potential external radiation dose rates due to damage to the package boundary from accident conditions tests were evaluated based on the structural evaluation in Section 2.7 of the application. Four non-mechanistic package damage scenarios, which were considered bounding, were evaluated. The scenarios were described in Section 4.3 of the application.

The results of these evaluations demonstrate that the requirements in 10 CFR 71.51(a)(2) have been met. Because there is no gaseous component to the source term, no krypton-85 will be released from the package. The maximum allowable release rate under hypothetical accident conditions is A_2 in one week, and the effective A_2 value for the mix of radionuclides present was calculated as 3.59 curies. Because the total quantity of radioactive material available for release was 2.90 curies, the package satisfies the allowable release rate criterion of an A_2 quantity per week. For the scenario of the package completely losing the top plate, the dose rate was estimated to be 0.79 rem per hour at 1 meter which is below the regulatory limit of 1 rem per hour at 1 meter.

5.5 Conclusion

Based upon the staff's review of the shielding evaluation, the staff has reasonable assurance the requirements in 10 CFR 71.47 and 71.51(a) are met.

6.0 CRITICALITY

The package contents are limited to fissile material quantities that meet the exemption standards in 10 CFR 71.53. The applicant estimated that the package contains less than 1 gram of fissile material. Therefore accidental criticality in transport is not a concern.

7.0 OPERATING PROCEDURES

Chapter 7 of the application describes how the package is loaded, closed and prepared for transport. The RV will be prepared by removing free standing water, cutting and capping the RV nozzles, bolting the package top plate to the RV flange using head studs and cap nuts, and removing RV stabilizers and supports. The RV will be lifted by lugs attached to the top plate and loaded into the outer package. After the top plate has been welded, LDCC will be injected through holes in the top plate. After placement of the LDCC is complete and it has cured, the injection holes will be plugged and seal welded. Lifting lugs and trunnions that are used for moving and up-ending the RV package will be removed prior to transport. The package will be marked and labeled, and radiation and contamination surveys will be performed.

The package will initially be placed on a road transporter designed for heavy loads and transported to the rail siding where the package will be transferred to a rail car. The package will be transferred in the horizontal orientation by use of hydraulic jacks and a slide beam system, and will not be lifted by crane. After rail transport, the package may then again be transferred to a road transporter by the same transfer process for final delivery to the disposal facility.

The applicant has proposed an operational control that will assure protection against brittle failure of the package due to impact in cold temperature conditions. Prior to transport, weather reports along the transport route will be reviewed and transportation will not begin if an ambient temperature below 0°F is predicted.

8.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

Chapter 8 of the application describes the acceptance tests that will be performed prior to transport. Welding of the outer package will be performed and inspected in accordance with ASME Code, Section III, Subsection NB. Section 8.1.2 of the application specifies non-destructive examinations to be performed for the various welds. The package will be pressure tested to 1.5 times the maximum normal operating pressure (26.85 psig) as required by 10 CFR 71.85(b). No leak testing of the package is specified prior to transport for the following reasons: (1) the containment system is fully welded, (2) welded closures are examined prior to transport, (3) the releasable radioactivity is fixed in place by the LDCC, (4) the containment system will not be breached under normal conditions of transport, and (5) the total inventory of releasable radioactivity does not exceed a Type A quantity. No maintenance procedures were proposed since this is a single use package that will also be used for final disposal of the RV.

CONDITIONS

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

   (a) The package must be prepared for shipment and transported in accordance with Chapter 7 of the application, as supplemented.

   (b) The package must be acceptance tested in accordance with Chapter 8 of the application, as supplemented.

2. Transport of the package may only be initiated if (1) the ambient temperature at the Big Rock Point plant is greater than 0°F, and (2) the minimum predicted temperature along the transport route for the anticipated transport period is greater than 0°F.

3. The package must be transported on a motor vehicle and on a railroad car assigned for the sole use of the shipper.

CONCLUSION

Based on the statements and representations contained in the application, as supplemented, and the conditions listed above, the staff has concluded that the
Model No. BRP RVP SAR-5339 package meets the requirements of 10 CFR Part 71.

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