April 27, 2006

Mr. Patrick Paquin
General Manager - Engineering & Licensing
Duratek, Inc.
140 Stoneridge Drive
Columbia, South Carolina 29210

SUBJECT: SPECIAL PACKAGE AUTHORIZATION FOR THE LA CROSSE REACTOR PRESSURE VESSEL TRANSPORTATION PACKAGE (TAC NO. L23931)

Dear Mr. Paquin:

By letter dated December 16, 2005, and supplemented January 5, March 5 and 16, 2006, Duratek, Inc., submitted an application for a Special Package Authorization for a transportation package for Dairyland Power Cooperative's La Crosse Boiling Water Reactor vessel. The reactor vessel has been decommissioned, all fuel elements have been removed, and the package will be transported from the La Crosse site in Wisconsin to a disposal site in Barnwell, South Carolina. The staff has reviewed the application, as supplemented, and determined that a Special Package Approval is acceptable for a one-time only shipment of the reactor vessel under the provisions of Title 10 of the Code of Federal Regulations (CFR) Part 71. The results of the staff's review are documented in the enclosed Safety Evaluation Report

The package, Model No. LACBWR RPV, is the reactor vessel consisting of a cylindrical shell section with a formed integral hemispherical bottom head and a removable hemispherical top head that is bolted to a mating flange on the vessel shell. The vessel wall has a nominal thickness of 4 inches. The vessel is ferritic steel plate with integrally bonded Type 304L stainless steel internal cladding. The reactor internals consist of a thermal shield, a core support skirt, a plenum separator plate, a bottom grid assembly, steam separators, a thermal shock shield, a baffle plate structure with a peripheral lip, a hold-down mechanism, control rods, fuel assembly shrouds, and reactor core support structures. The voids in the reactor vessel are filled with low-density cellular concrete. The nozzles and lifting devices are cut prior to lifting and removing it from the reactor building.

The annulus between the vessel and the reactor canister is filled with medium-density concrete. The canister is formed from 1.5-inch thick steel cylindrical shell with 4-inch end plates. The completed package is 39-feet 7-inches long with an outer diameter of 10-feet 6-inches. All joints of the canister are welded, forming the containment boundary and providing a tamper-resistant seal. Shielding is welded to the exterior of the canister at the location of the core. The lower portion of the canister will have eight support legs where the lower section of the reactor vessel will be placed. There are no lifting or tie-down devices that are a structural part of the package. The total design weight is 639,000 lbs.

The package, Model No. LACBWR PRV, is authorized to transport an irradiated steel reactor pressure vessel and internal components from the La Crosse Boiling Water Reactor. The maximum quantity of material per package is 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 71.15.

In addition to the requirements of Subpart G of 10 CFR Part 71, the following conditions apply to the Special Package authorization:

(1) The package must be prepared for shipment and transported in accordance with Chapter 7 of the application, as supplemented.
(2) The package must be acceptance tested in accordance with Chapter 8 of the application, as supplemented.
(3) Transport of the package may only be initiated if the ambient temperature at the site is greater than 0EF. The ambient temperature will be monitored throughout the transport process and transportation will be stopped should the temperature fall to 0EF.
(4) The package authorized by this certificate must be transported on a motor vehicle or on a railroad car assigned for the sole use of the shipper.
(5) The package is constructed and assembled in accordance with Duratek, Inc., Drawings: C-068-163041-002, Rev. 0; C-068-163041-003, Rev. 0; C-068-163041-004 Rev. 0.
(6) The package is a one-time only exclusive use shipment.
(7) Transport of the package will be in accordance with the referenced Transport Emergency Response Plan.

The package authorized by this Special Package Authorization is hereby approved for use under the general license provisions of 10 CFR 71.17.

The Special Package Authorization will expire on April 30, 2011.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

/RA/

Robert A. Nelson, Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Docket No. 71-9322
TAC No. L23931

Enclosure: Safety Evaluation Report

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NAME	JCaverly		MDeBose		JPiotter		CInterrante		CBajwa		SHelton	
DATE	04 / 10 /06		04 / 21 /06		04 / 24 /06		04 / 25 /06		04 /25 /06		04 / 10 /06	
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NAME	EThompson		Streby NLO		LCampbell		GBjorkman		RNelson			
DATE	04 / 10 /06		04 / 20 /06		04 /25 /06		04 / 26 /06		04 / 27 /06			

SAFETY EVALUATION REPORT

Docket No. 71-9322
Dairyland La Crosse Boiling Water Reactor
Pressure Vessel Transport Package
Revision No. 0

SUMMARY

By application dated December 16, 2005, as supplemented January 5, March 5 and 16, 2006, Duratek, Inc. (Duratek) requested that the U.S. Nuclear Regulatory Commission approve by Special Package Authorization, the design for the Dairyland Power Corporation's (DPC) La Crosse Boiling Water Reactor (LACBWR) Pressure Vessel transportation package for transporting the reactor vessel and internals from the La Crosse Reactor site in Genoa, WI to a disposal facility. Authorization is requested for a one-time only shipment.

The package consists of the decommissioned reactor vessel and internals, contained within an outer steel packaging that is welded closed and filled with medium density concrete. The overall package dimensions are 39-feet 7-inches in length and 10.5-feet in diameter. The package weighs approximately 639,000 pounds, as prepared for transport. The package is designed for a single-use trip from the La Crosse 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 Code of Federal Regulation Part 71, including the general standards for all packages and the performance standards under normal conditions of transport (NCT) and hypothetical accident conditions (HAC).

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, the staff concludes that a Special Package Authorization for the package is acceptable and meets the requirements of 10 CFR Part 71. Accordingly, the package is authorized for a single shipment for disposal.

References

Duratek, Inc., Application dated December 16, 2005.

Supplemented on January 5, March 5 and 16, 2006.

1.0 GENERAL INFORMATION

1.1 Special Package Authorization

The application was reviewed to meet the requirements of 10 CFR Part 71 and where applicable to meet the requirement of 10 CFR 71.41(d) - Special Package Authorization. This provision of the regulations became effective in 2004 and is intended to apply only in limited circumstances and only to one-time shipments.

The provision of 10 CFR 71.41(d) state that “packages for which compliance with other provisions of these regulations [i.e., 10 CFR Part 71] is impracticable shall not be transported except under special package authorization.” The provision states that a special package authorization may be issued if the applicant demonstrates the following: (1) compliance with the other provisions of the regulations is impracticable, (2) requisite standards of safety established by the regulations are demonstrated through means alternate to the other provisions, and (3) the overall level of safety in transport for these shipments is at least equivalent to that which would be provided if all the applicable requirements had been met.

The technical review of the package application considered the above stated requirements and noted where the provisions of 10 CFR 71.41(d) are applicable.

1.2 Package Description

The LACBWR Pressure Vessel packaging consists of a steel canister surrounding the reactor pressure vessel (RPV). The reactor vessel consists of a cylindrical shell section with a formed integral hemispherical bottom head and a removable hemispherical top head, which is bolted to a mating flange on the vessel shell. The vessel wall has a nominal thickness of 4". The vessel is ferritic steel plate with integrally bonded Type 304L stainless steel cladding. The reactor internals consist of a thermal shield, a core support skirt, a plenum separator plate, a bottom grid assembly, steam separators, a thermal shock shield, a baffle plate structure with a peripheral lip, a hold-down mechanism, control rods, fuel assembly shrouds and reactor core support structures. The voids in the reactor vessel will be filled with low-density cellular concrete prior to cutting the nozzles and lifting the vessel to remove it from the reactor building.

The annulus between the vessel and the reactor canister is filled with medium-density concrete. The canister is formed from 1.5-inch thick steel cylindrical shell with 4 inch end plates. The completed package is 39-feet 7-inches long with an outer diameter of 10-feet 6-inches. The total design weight is 639,000 lbs. All joints of the canister are welded, forming the containment boundary and providing a tamper-resistant seal. Shielding is welded to the exterior of the canister at the location of the core. The lower portion of the canister has a raised flat ring on which eight support legs will rest when the lower section of the reactor vessel is placed in the lower section of the canister. There are no tie-down devices that are structural at the time of shipping. The package will be fabricated and assembled in accordance with Duratek’s approved Quality Assurance program.

The type and form of material that is approved for the package is the Dairyland Power Cooperative irradiated steel reactor pressure vessel and internal components. The maximum quantity of material is greater than a Type A quantity of radioactive material contained in an irradiated reactor vessel 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. Activity per unit area was conservatively assumed at 100 percent of the activity distributed over reactor internals. The package is constructed and assembled in accordance with Duratek, Inc., Drawings: C-068-163041-002, Rev. 0; C-068-163041-003, Rev. 0; C-068-163041-004 Rev. 0.

2.0 STRUCTURAL

The staff reviewed the structural portion of the application for structural performance of the package and found that the design 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 NCT and HAC.

2.1 Description of Structural Design

The packaging is composed of a cylindrical steel shell, approximately 11-feet in diameter and 40-feet long, with welded steel flat top and bottom plates. The material of construction of the packaging is ASTM A-516, Grade 70, carbon steel. The package will be used for transportation of a RPV and some of its internals as described in the Chapter 1 of the application. The void space in the RPV is filled with low density cellular concrete (LDCC) having a density of 50 pounds per cubic foot, and the annulus between the RPV and the outer packaging is filled with medium density cellular concrete (MDCC) having a density of approximately 120 pounds per cubic foot.

The 4-inch thick package top and bottom plates are shop welded to the upper and lower canister sections with a full penetration weld. After the RPV is loaded into the lower section, the upper section is field welded to seal the package 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 1.5 inches thick with two additional plates welded to its exterior surface for a length of approximately 8 feet in the former core region of the RPV. The outer packaging shell provides 4.25 inches of steel for radiation shielding at the RPV region of highest radioactivity.

Stress intensity values for Division 1 of the ASME Boiler & Pressure Vessel Code, Section III, were used for the design of the package. The values contain assumptions about materials, fabrication, and examination. Consequently, the package pressure retaining materials, fabrication, and examination will be in accordance with Sub-section NB of the ASME Code. The design of the package under NCT is based on linear-elastic structural analysis. The NCT drop is evaluated using nonlinear elastic-plastic dynamic finite element analysis. For the HAC, 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).

Compliance with the radiation and radioactivity release limits specified in 10 CFR 71.51(a)(2) was assured as demonstrated by the containment evaluation 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 RPV is constructed from ASTM A-302 steel and the canister (containment shell) is a completely welded-shut enclosure covering the RPV with 1½ -inch-thick ASTM A516-Grade-70 steel with (even thicker) welded end plates that form the closure. Welds are full-penetration using E-70xx electrodes. Fabrication and inspection of the package should be conducted in accordance with ASME B&PV Code Section III, ND and this code will be followed as much as practical. Concrete (LDCC) fills the RPV which is placed inside a shell that contains injection ports to permit concrete (MDCC) to be placed between the RPV and the shell. The as-fabricated shipment will be checked for efficacy of shield materials (steel and concrete) in satisfying the requirements of 10 CFR 71.47. Materials selection, fabrication, testing and inspections are sufficient to meet regulatory requirements of 10 CFR Part 71.

2.3 General Standards for All Packages

2.3.1 Minimum Package Size

The smallest overall dimension of the package is the 11-feet outside diameter and meets the minimum package size requirement of 10 CFR 71.43(a).

2.3.2 Tamper Indicating Device

The package is designed with welded closures and seal welds on all plugs. Intact welds are an indication that the package has not been opened. This design characteristic satisfies the requirement of 10 CFR 71.43(b).

2.3.3 Positive Closure

The package has a welded containment system and all plugs are seal welded. Unintentional opening is precluded, and satisfies the positive closure requirement of 10 CFR 71.43(c).

2.3.4 Chemical and Galvanic Reactions

The applicant has demonstrated for this package no significant chemical or galvanic reactions will occur. The applicant analyzed the potential for hydrogen generation with the LACBWR RPV (see Section 4.2) and the only potential for hydrogen generation is radiolytic decomposition of water in the LDCC in the region of the activated core materials, which are poured in place using low- and medium-density cellular concrete materials. Staff concludes that no substantial hydrogen ignition could occur and any hypothetical ignition event would be quickly quenched because propagation would be precluded by the wide dispersion (distance) between the hydrogen-containing void spaces within the poured concrete. Thus, the staff concludes that there will be no adverse chemical, galvanic or other reactions for or among materials used in this transportation package and so it may be considered safe from these reactions. These findings are in compliance with 10 CFR 71.43 (d).

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 NCT 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.

2.3.8 Venting

The package is a welded container with no feature for continuous venting during transport. Therefore, the staff determined that 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 LACBWR RPV package are not structural elements 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 codes ANSYS - Version 9.0 and LS-DYNA - Version 970, were used to analyze the package for the loading conditions of NCT (10 CFR 71.71).

2.5.1 Heat

The package was evaluated for an ambient temperature of 100EF with and without solar insolation using a one dimensional analytical model. It was determined that the maximum accessible surface temperature in the shade was approximately 101EF and the maximum inside temperature was approximately 154EF. The analysis also demonstrated a very small temperature gradient through the wall thickness during all thermal loading cases and that the mean normal operating pressure was 7.24 psig that was conservatively assumed to be 7.5 psig. On the basis of these results, uniform expansion will occur resulting in minimal thermal stresses. The requirements of 10 CFR 71.71(c)(1) have been met based on these conclusions.

2.5.2 Cold

The ambient temperature was assumed at -20EF with no insolation and an internal decay heat of 70 Watts for the evaluation of the package for the cold temperature condition. A one dimensional analytical model was used to show that the canister will drop in temperature to approximately -19EF, with an insignificant temperature gradient of 1EF through the wall. As an operational measure, the lowest temperature for transport has been set at 0EF. This limit eliminates the need for an analysis at the regulatory lower limit of -40EF.

Brittle fracture is at the minimum operating temperature of 0°F by calculating the nil ductility transition (NDT) temperature for the materials of construction. The ASME Code Section-VIII, was used to evaluate whether the package was subject to impact testing. The analysis showed that 1½-inch-thick plate made of normalized A-516 material conforming to fine grain practice had a minimum service temperature of -13EF, which is less than the minimum operating temperature of 0EF.

Thus, differential thermal contraction of components and the minimum internal pressure under the cold temperature condition would not adversely affect the package performance including brittle fracture effects. The staff determined that the requirements of 10 CFR 71.71(c)(2) have been satisfied using the operational requirements as specified in 71.41(c). Therefore, the provision of a Special Package Authorization have been met.

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 factor of safety of 3.3 of the allowable stress. The applicant also performed a buckling calculation that conservatively predicted a factor of safety against buckling of 15.9. The increased external pressure requirement of 10 CFR 71.71(c)(4) would not adversely affect the package performance. Therefore, the staff finds that the design is acceptable and meets the Special Package Authorization requirements of 10 CFR 71.41(d).

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 (22 psig). The results for this loading condition showed a factor of safety of 3.5 of the allowable stress. Thus, the reduced external pressure requirement of 10 CFR 71.71(c)(3) would not adversely affect the package performance. Therefore, the staff determined that the package meets the requirements for a Special Package Authorization requirements of 10 CFR 71.41(d).

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2.5.5 Vibration

The applicant concluded that vibration incident to normal transport would not adversely affect the package performance. The calculated maximum alternating stress intensity of 24,654 psi for a vertical 2g loading resulted in an allowable number of cycles of 40,000. The analysis assumed one decoupling every mile results in 1500 cycles during the anticipated 1500 miles of travel. Therefore, fatigue due to vibration is negligible and the requirements of 10 CFR 71.71(c)(5) have been satisfied.

2.5.6 Water Spray

The LACBWR RPV 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 and the package can be shipped under the provisions of a Special Package Authorization of 10 CFR 71.41(d).

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. The application considered two likely drop orientations: a horizontal drop and a drop with the long axis oriented 5E to the horizontal plane.

The ANSYS/LSDYNA explicit dynamics computer code was used to simulate a half-symmetry drop on to a flat, essentially unyielding surface for each case. The results for each orientation showed maximum tensile strains of approximately 11 percent while the rupture strain was set at 21 percent. This indicates that for the 1-foot free drop, there is no breach of the outer shell. Staff noted several irregularities in the formulation of the stress strain input for the steel shell, although they are not a safety concern. Most notably, the use of elongation strain at rupture of 21 percent is very conservative as well as using engineering stress strain material properties for input values in LS-DYNA.

The application used a single through thickness element for the outer steel shell. A sensitivity study on the 30-foot drop case was performed to determine if the results would change appreciably with two elements through the shell thickness. The applicant demonstrated that the maximum total equivalent strain for two elements through the shell thickness was less than 15 percent different, thereby making a single element through the shell thickness a reasonable model simplification.

Based on these conclusions, the requirements of 10 CFR 71.71(c)(7) have been satisfied and the staff determined that the package meets the requirements for a Special Package Authorization under 10 CFR 71.41(d).

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 and presented in ASCE Manual No. 58, "Structural Analysis and Design of Nuclear Plant Facilities." Staff disagrees with the use of this analytical method because the approach is not sufficient in scope to yield reliable results. Specific objections are based on the statements made in the reference documents that state the targets are thin steel targets and that the test data parameters used and the range of applicability are both undefined. However, staff has determined that the applicant, in performing a containment calculation that bounds the penetration effects for HAC, has satisfied the intent of this requirement. Staff determined that the requirement of 10 CFR 71.71(c)(10) has been satisfied and that the provisions of a Special Package Authorization under the requirements of 10 CFR 71.41(d) have been met.

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 utilizing the ANSYS/LS-DYNA finite element software package. ANSYS element type SOLID164, a three-dimensional, fully-integrated, eight-node solid element was used to model the package upper and lower cylindrical shell, top and bottom plates, the medium density cellular concrete, and the reactor vessel. The application performed evaluations of three orientations that included horizontal, 5E above horizontal, and CG over corner. The applicant stated that these orientations envelope all other damaging scenarios such as an end drop or oblique orientations.

During the analysis, the package stresses were monitored for element failure. Failure was considered to occur at any element for which the maximum tensile strain exceeded the elongation at the material ultimate tensile strength.

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. The simulation was determined to be complete when the KE in the system was effectively zero indicating that all of the KE was converted to strain energy.

The results show that the supplementary shielding on the lower region of the canister may detach and that the region near the impact point may show severe deformation. In two cases, the 5Eoblique drop and the CG (Center of Gravity) over-corner case, the plastic strains in the steel exceeded the allowable strain of 21 percent. The application noted that these strains were confined to the outer regions of the shell/lid and concluded that the rupture would not completely penetrate the outer steel shell. In addition, the use of an elongation strain of 21 percent and input values of engineering stress and strain is extremely conservative in terms of material behavior. The applicant's analysis included additional conservatism by assuming a through-thickness rupture would occur and the consequences to such a rupture enveloping a region 10E in both circumferential directions at the point of impact. The consequences are evaluated in Chapter 4 of the SAR and the containment evaluation satisfies regulatory limits.

An issue of concern was the use of only a single element through the outer shell thickness. The staff requested that the applicant perform a sensitivity study on the 30-foot CG-over corner drop case to determine if the results would change appreciably with two elements through the shell thickness. The application demonstrated that the maximum total equivalent strain for two elements through the shell thickness was less than 15 percent different, thereby making a single element through the shell thickness a reasonable model simplification.

The staff has determined that the requirements of 10 CFR 71.73(c)(1) have been satisfied based on the requirements of Special Package Authorization under 10 CFR 71.41(d).

2.6.2 Puncture

The applicant considered a puncture event by utilizing Nelm's equation from the "Cask Designers Guide, ORNL 1970. Staff disagrees with the use of this analytical method because it was empirically derived specifically for a steel-lead-steel shell rather than a steel-concrete-steel shell. The methodology allows for no mechanism to assess the affects of a drop onto a steel pin with the specific material configuration present. Despite the flawed analytical approach, staff has determined that the applicant, in performing a containment calculation that bounds the penetration effects for HAC has satisfied the intent of this requirement.

Therefore, the staff has determined that the requirements of 10 CFR 71.73(c)(3) have been satisfied based on the requirements for a Special Package Approval of 10 CFR 71.41(d).

2.6.3 Thermal

Under fire test conditions, the entire package will rise to a temperature of 1475EF with very little through-thickness temperature gradient. This will result in uniform expansion of the container. The uniform expansion coupled with no thermal insulation and no dissimilar metal joints will result in no additional binding stresses caused by differential thermal expansion. In addition, the assumptions of the presence of weld cracks and a 6-inch diameter perforation due to the puncture test produce the results that show the package will hold no internal pressure resulting in no additional primary stresses. The requirements of 10 CFR 71.73(c)(4) have been satisfied.

2.6.4 Immersion - All Packages

The package design 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). The factor of safety for stress and shell buckling were both within acceptable limits. Therefore, the requirements of 10 CFR 71.73(c)(6) for the water immersion condition are adequately achieved.

2.7 Appendix

Attachment ST-517 of the application summarized the calculations performed to document the structural evaluation of the package.

2.8 Conclusion

Based on the review of the statements and representation in the application, the staff concludes that the structural design has been adequately described and evaluated and that the package design meets the requirements of the Special Package Authorization requirements of 10 CFR 71.41(d). Although the staff cannot conclude that certain equations are acceptable or apply for the penetration requirements of the NCT evaluation, the application provides sufficient assurance based on the size, material, and design of the package to conclude that requisite standards of safety will be maintained and that the level of safety for the package is maintained. Additionally, calculations for the penetration test for HAC were not found acceptable in this portion of the review. However, bounding assumptions used for other areas of the application provide a high level of assurance that the overall level of safety in transport will be equivalent.

3.0 THERMAL

3.1 Thermal Evaluation

The applicant provided an evaluation of the performance of the LACBWR RPV package design under the conditions specified in 10 CFR 71.71 for NCT and 10 CFR 71.73 for HAC.

To demonstrate that the package meets the NCT outlined in 10 CFR 71.71, the applicant submitted a steady-state thermal analysis of the design. The applicant's analysis indicated that the maximum temperature of the surface of the package would not exceed 155EF, which is below the limit of 185EF for an exclusive use shipment listed in 10 CFR 71.43(g).

The canister is a welded structure that does not have any thermal insulation or dissimilar metal joints. The applicant evaluated the maximum normal operating pressure (MNOP) of the package and determined that the package would not exceed 7.5 psig for normal conditions. Under the conditions of the fire test, the entire canister will rise to a temperature close to 1475EF with very little temperature gradient through its walls. Since the canister was assumed to develop cracks in the welds near the point of impact during the drop tests, and a puncture through its walls during the puncture test, no pressure will develop inside. Therefore, the package is assumed to be breached during HAC sequence of events, and no pressure will build up in the package due to the increase in temperature caused by the HAC fire transient. In the case where the package is not breached, this pressure increase would also be acceptable.

3.2 Conclusion

Although detailed analysis for Type B thermal conditions was not performed for this design, the staff concludes that the applicant has demonstrated the overall level of safety for this package will be equivalent to that which would have been provided if it met all the applicable requirements of 71.73. Therefore, the staff finds the package acceptable for thermal performance under the provisions a Special Package Authorization of 10 CFR 71.41(d).

4.0 CONTAINMENT

4.1 Description of the Containment System

The containment boundary of the LACBWR RPV package, shown on SAR drawing C-068-163041-002 in Appendix 1.3, consists of a fully welded canister made of ASTM A-516 Grade 70 steel. The canister is fabricated in two sections; the upper section consists of a 4-inch thick top plate and a 1½-inch thick cylindrical shell and the lower section consists of a 4-inch thick bottom plate and a 1½-inch thick cylindrical shell. The lower section has a steel ring welded to the outside circumference, approximately 6-inches from the open end, that is used for welding the upper section to the lower section.

The upper assembly will have at least four penetrations, called injection ports, for injecting medium-density concrete to fill the annulus between the canister and the RPV shell. Up to two additional injection ports (no more than a total of six injection ports overall) may be necessary to facilitate the pouring of the concrete into the annulus. Prior to transport, the 3½-inch diameter injection ports will be sealed by insertion of SA-516, Grade 70 plugs mounted on cover plates (the plug assemblies are shown on SAR drawing C-068-163041-004, Revision 0), and will be fillet welded to the upper section of the canister.

The containment boundary is fully welded; the package cannot be opened unintentionally. The structural analysis in the SAR, Chapter 2 demonstrates that no internal pressure may arise within the package that will result in unintentional opening of the package. The package does not have a device that allows continuous venting. The applicant demonstrated that no galvanic, chemical, or other reactions will occur between the packaging and its contents. The package design has no reliance on filters or a mechanical cooling system.

4.2 General Considerations

The LACBWR RPV package will be shipped under the provisions of a Special Package Authorization, containing an irradiated RPV and its reactor internals. All nuclear fuel has been previously removed. The internals will remain inside the pressure vessel for transport; the voids will be filled with LDCC prior to removal of the vessel from the reactor building. The RPV will be shipped inside a cylindrical, steel containment shell; the annulus between the RPV and the shell will be filled with MDCC. Once injection of the concrete is complete, the containment canister will be a fully welded structure.

The package contains a total of 1.01×10^4 Ci of radioactive material, or approximately 410 A_2 values. This includes the estimated 58.2 grams of residual uranium remaining in the RPV discussed in SAR Chapter 1.2.2. Of the total source term, 1.6 Ci (3.43 A_2 values) is loose contamination and potentially dispersible; the majority of the activity resides in the activated hardware, primarily in the former core region of the RPV. The LDCC serves to bind the loose contents to minimize the likelihood of dispersal in the event of a breach of containment.

The applicant analyzed the potential for hydrogen generation with the LACBWR pressure vessel package. The only potential for hydrogen generation is radiolytic decomposition of water in the LDCC in the region of the activated core materials. Any hydrogen accumulation would occur in the small amount of void space within the poured concrete. The applicant determined that over a one year period, the amount hydrogen gas generated in service is less than required to form a flammable mixture, and will ensure that the one-time only transportation of the package will be completed within one year. Hence, staff concludes that no substantial hydrogen ignition could occur and even if the calculations are in error and hydrogen did ignite, propagation would be precluded by the wide dispersion of and distance between the hydrogen-containing void spaces within the poured concrete. Additionally, the applicant will ensure that the chemical and physical properties of the LDCC and MDCC are consistent with the parameters used in the calculations that determined the amount of radiolytic hydrogen generation.

The structural analysis determined that the components comprising the containment boundary would not suffer brittle fracture, based on a lowest service temperature (LST) of 0EF. The operating procedures in SAR Chapter 7.4.1 ensure that the operating controls in the “Exclusive Use Instruction” will instruct the carrier not to begin the shipment if an ambient temperature below 0°F is predicted, and to stop the shipment if the temperature falls to 0°F during transport.

4.3 Containment under Normal Conditions of Transport

In the SAR Chapter 2.6, the applicant determined analytically that the package would survive the tests specified in 10 CFR 71.71 for NCT, and that containment boundary would not be breached. Therefore, no loss or dispersal of radioactive contents is expected to occur, and the maximum permitted leakage rates specified in 10 CFR 71.51(a)(1) will not be exceeded.

The staff agrees with the applicant’s assertions that the LACBWR pressure vessel package satisfies the containment requirements of 10 CFR 71.51 under NCT.

4.4 Containment under Hypothetical Accident Conditions

In SAR, Chapter 2.7 provides the applicant’s structural analysis of the package under the tests specified in 10 CFR 71.73 for HAC. The analysis for the 30-foot drop showed that the package would deform without breaching the containment boundary. However, the applicant conservatively assumed that the weld adjoining the upper and lower sections of the canister would partially breach (about 10E in the circumferential direction on either side of the plane of impact, or about 5 percent of the circumference of the endplate) as a result of the 30-foot drop. Additionally, the puncture accident could result in a localized breach of the canister and the MDCC in the annulus between the canister and the RPV. In either event, only a small portion of the containment boundary would be breached, and the release would be limited to the volume that could pass through the crushed region of the annulus concrete and leak out of the localized opening in the containment shell. The structural analysis shows that neither the 30-foot drop nor the puncture test would cause damage to the RPV shell or the LDCC inside the RPV shell, and that these would remain intact as barriers to dispersal of radioactive contents during HAC.

The applicant's bounding calculation for the 30-foot drop assumed a release of LDCC. The applicant assumed the LDCC volume that could be contaminated would result from a layer approximately 0.45 inches deep over the contaminated surface, corresponding to roughly 25 percent of the LDCC volume having mixed with the dispersable contents, and that 5 percent of the LDCC may be dispersed during HAC. Using these conservative assumptions, the applicant estimated that up to 0.32 Ci of radioactive material, or 0.7 A_2, could be released under HAC. The resulting release is below the requirement in 10 CFR 71.51(a)(2), which states that under HAC the release of radioactive material must not exceed a total amount A_2 in 1 week. The staff agrees with the applicant's assertions that the LACBWR reactor pressure vessel package satisfies the containment requirements of 10 CFR 71.51 under HAC.

4.5 Leakage Rate Tests for Packages

The containment boundary is a fully-welded canister. Weld examinations, described in SAR Chapter 8.1.2, will be performed in lieu of leak-testing to verify the integrity of the welds. All the shop welds will be visually inspected, magnetic particle (MT) examined, and volumetrically examined. The fillet welds used to seal the plug assemblies to the canister will be visually examined and MT examined. The weld adjoining the upper and lower containment shells will be subject to visual, MT, and UT examinations. All weld examinations will be performed with acceptance criteria being the ASME Code, Section III, Article ND-5300.

The staff finds this acceptable because this package will only be shipped once under the provisions of the Special Package Authorization, and because the applicant used conservative assumptions to determine package response under NCT and HAC.

4.6 Conclusion

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

5.0 SHIELDING

5.1 Package Description and Contents

The LACBWR RPV package is a 1½-inch thick two-piece cylindrical steel shell welded to 4-inch thick top and bottom plates. The package is approximately 10½ feet in diameter and 39½ feet long. The package shell is 1½ inches thick with additional shielding welded to the exterior surface of the cylindrical shell in the area where the core was located. This additional shielding consists of two parts. The first is a 10-foot long, 1¼-inch thick cylinder welded to the outside of the main RPV package and centered on the core midplane. The second part consists of two curved 8-foot long, 1¾-inch thick steel plates welded to the outer shield cylinder and centered on the core midplane. They are sized and located so that each one covers a 120E sector (circumferentially) on the side of the RPV package when it is placed on a conveyance for transportation.

The package contents are normal form radioactive material. The package contains the RPV and internals as listed in Section 1.2.2 of the application. LDCC is used to fill the void space in the RPV, and MDCC is used to fill the annulus between the RPV and the outer packaging. The concrete 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 internal components in the RPV are the two components of the source term. The total activity in the package was estimated to be about 10,000 curies as of June 1, 2007 (the proposed shipping date of package). The total activity is based on isotopic analyses of samples, activation calculations, and conservative assumptions. There are no gaseous or liquid radioactive components to the source term.

Samples were taken from internal piping surfaces of the Shutdown Condenser. An isotopic analysis of the sample was performed and the isotopic mix from the analysis was increased by the 2 sigma uncertainty reported for the analysis. The final isotopic mix was multiplied by the total surface area of the RPV (minus the RPV Head) plus the internals to obtain the total amount of potential removable surface contamination. The total surface contamination on the RPV and internals was estimated to be 1.61 curies.

The applicant's source term due to activation was determined using the ORIGEN-ARP computer program. This calculation was verified with a manual calculation and by verifying the ORIGEN-ARP installation by running a test case.

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

The source term evaluations are discussed in detail in Section 1.2.2 of the SAR (supplemented by SAR Reference 1-1). Based upon a review of the information provided in the application, as supplemented (including Reference 1-1 to Chapter 1 of the SAR), staff determined there is reasonable assurance the source term has been adequately developed and identified.

5.3 Shielding Evaluation

The shielding evaluation for this package was performed using surrogate source terms. Separate source terms were developed, as needed for the various calculational scenarios, that duplicated the dose rate measurements on the outside of the RPV at the core midline, and at the bottom, respectively. Each source term consisted of a quantity of cobalt-60 that resulted in a calculated dose rate that equaled the measured dose rate on the RPV surface at the specified location.

The applicant used the MicroShield computer code, which uses the ICRP-51 gamma flux-to-dose-rate conversion factors, to evaluate dose rates from the package. Dose rates were calculated at the points needed to determine compliance with the regulations for both NCT and HAC.

Potential streaming pathways were from various ports and nozzles in the RPV. However, all RPV openings are filled with LDCC and capped with steel, thereby eliminating significant streaming pathways.

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. Additionally, staff applied the ANSI/ANS-6.1.1-1977 gamma flux-to-dose conversion factors to the fluence rates output by MicroShield. The results were consistent with the applicant's results using the ICRP-51 conversion factors. Staff has reasonable assurance that the calculated dose rates are appropriate for this package and meet the requirements of 10 CFR 71.47 and the requirements of a Special Package Authorization of 10 CFR Part 71.

5.3.1 Normal Conditions of Transport

The LDCC filling the RPV is assumed to remain intact under NCT and thus keep the internals from shifting and changing the dose profile on the outside of the package. Dose rates calculated for the RPV package under NCT are listed in Table 5-1.

Table 5-1. Peak Dose Rates for Normal Conditions of Transport (mrem/hr)

	Package Surface			2 Meters from Package Surface*	
	Side	Top of load	Underside	Side	Ends
Gamma**	< 126	126	126	7.3	3.6
10 CFR 71.47(b) Limit	200	200	200	10	10

* This is less than or equal to the distance specified in the regulations (2 meters from the vertical planes projected by the outer lateral edges of the vehicle) and as thus it is conservative.
** There is no neutron component to the dose rate.

5.3.2 Hypothetical Accident Conditions

The RPV package was evaluated for HAC as specified in 10 CFR 71.73. The structural evaluation provided in Section 2.7 of the application describes the computer analyses for the various accident scenarios. The results of these analyses indicate that the potential damage to the package structure does not result in external dose rates or radioactive material releases that exceed regulatory limits. 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 from accident conditions tests were assessed based on the structural evaluation in Section 2.7 of the application. The results of these hypothetical accident condition evaluations were presented in Section 5.4 of the application. Specifically, for the free drop test, the maximum dose rate at 1 meter from the package was 226 mR/hr; and for the puncture test, the maximum dose rate was 541 mR/hr at 1 meter from the package.

These results demonstrate that the external dose rate requirements in 10 CFR 71.51(a)(2) have been met. Results of these evaluations relative to containment of the contents are addressed in Section 4.0 of this Safety Evaluation Report.

5.4 Operational Considerations

As specified in Section 7.1.3 of the SAR, radiation and contamination surveys will be performed prior to offsite transport to verify that there is no external contamination on the package and that none of the applicable limits in 10 CFR 71.47 is exceeded.

5.5 Conclusion

Based upon the staff’s review of the shielding evaluation, the staff has reasonable assurance that the shielding requirements in 10 CFR 71.47 and 71.51(a) are met and that the package design meets the requirements of a Special Package Authorization of 10 CFR Part 71 with respect to shielding.

6.0 CRITICALITY

The package contents are limited to fissile material quantities that meet the exemption standards in 10 CFR 71.15. The applicant estimated the amount of fissile material to be 1.7 grams. Therefore, criticality is not a concern.

7.0 PACKAGE OPERATIONS

Chapter 7 of the application describes the package loading, closure and preparation for transport. Prior to removal, the reactor vessel will be prepared by grouting the recirculation piping, grouting the reactor vessel cavity with low-density concrete, size reducing the vessel body flange outside diameter to 119-inches, cutting all nozzles, and unbolting the support legs from the reactor vessel support ring. The reactor vessel will then be lifted out of its cavity and loaded into the packaging.

The canister will be fabricated in two sections, an upper and a lower. The vessel will be loaded into the package lower section using a gantry system. Medium density concrete will then be injected in the annulus between the reactor vessel exterior and the package lower section interior. Once the annulus has been filled and the medium density concrete has cured, the gantry system will be disconnected. The upper package section will be placed on top of the exposed section of the reactor vessel and welded to the package lower section at the weld ring. Medium density concrete will be injected via the penetrations in the canister’s upper section to fill the annulus between the reactor vessel and the package upper section interior. The penetrations will be plugged and seal welded.

The complete welded package will then be down-ended into a horizontal position using a gantry system trolley crane, clevises, trunnions and an A-frame pivot device. Once in the horizontal position, the clevises and trunnions will be disabled or removed and the remaining shields will be welded to the shell. Radiation and contamination surveys of the package will be conducted prior to transport off-site.

The package will initially be moved by multi-axle hydraulic trailer to a rail spur where it will be transferred to rail conveyance. After rail transport, the package may then again be transferred to a multi-axle hydraulic trailer.

The applicant proposed an operational control that will assure protection against brittle failure of the package due to the impact in cold temperature conditions. Prior to leaving the La Crosse plant, weather along the transport route will be monitored and the transport will not commence if the ambient temperature is below 0°F. The ambient temperature will be monitored throughout the transport process. Should the temperature fall to 0°F then the transportation will be stopped. The criteria for resuming transport will be based on package surface temperature.

To organize and coordinate all of the transportation activities, a Transport Emergency Response Plan (TERP) will be developed by Duratek. The document will be utilized as the transportation operation controlling document throughout the entire shipment from the La Crosse site to the disposal site. Exclusive use instruction will be included in the TERP and will identify operating controls such as the minium required temperature requirement discussed above. The TERP will also include details such as the transportation route, mode of transportation and transfer locations, distances, processes, and equipment; and identifies responsibilities and interfaces for the next transportation activities. These activities include package transfer from one conveyance to the next, tie-down instructions and inspection, radiological controls, and package delivery to the disposal site. Additionally, guidance for interaction with appropriate local and federal agencies in the event of an accident will be provided.

8.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

Chapter 8 of the application describes the acceptance tests that will be performed prior to transport. The package is designed for exclusive use, one-time transportation and disposal of the RPV. Acceptance tests and inspections will be performed prior to the transportation of the package in compliance with 10 CFR 71.85. The fabrication program will be in accordance with Duratek's approved QA Program and the approved design. Visual examinations will identify any defects. Weld examination will be conducted under the ASME Code, Section III, Article ND-5300. The structural integrity of the package is analytically demonstrated. No pressure test will be performed.

No leak testing of the package is specified prior to transport because the package contains primarily solid radioactive material that is immobilized in concrete, the containment system is fully welded and the package integrity under NCT provides assurance that radioactive materials will remain contained in the package. Plate material will be certified for mechanical and chemical characteristics. All welding will be performed using procedures qualified for notch toughness requirements to match the base material.

The containment shell is a welded steel enclosure used for the both transportation and disposal of the RPV. All plate material shall be provided with certified mechanical and chemical test reports. These tests shall include determination of the nil-ductility transition temperature for materials equal to or greater that 3-inches in thickness. All welding will be performed using procedures qualified for notch toughness. Post weld heat treatment will not be performed unless the weld procedures qualification requires it. Verification of the material of construction will be completed under Duratek's approved QA program.

Shielding tests will not be required for final acceptance for shipment. The fabrication of the packaging will be performed in accordance with Duratek's QA program. The fabrication requirements and the pre-shipment dose rate surveys will confirm the adequacy of the shielding as required by the package design.

No maintenance procedures were proposed since this is a single use package that will also be used for final disposal of the reactor vessel.

CONDITIONS

In addition to the requirements of Subpart G of 10 CFR Part 71, the following conditions apply:

(1) The package must be prepared for shipment and transported in accordance with Chapter 7 of the application, as supplemented.
(2) The package must be acceptance tested in accordance with Chapter 8 of the application, as supplemented.
(3) Transport of the package may only be initiated if the ambient temperature at the site is greater than 0EF. The ambient temperature will be monitored throughout the transport process and transportation will be stopped should the temperature fall to 0EF.
(4) The package authorized by this certificate must be transported on a motor vehicle or on a railroad car assigned for the sole use of the shipper.
(5) The package is constructed and assembled in accordance with Duratek, Inc., Drawings: C-068-163041-002, Rev. 0; C-068-163041-003, Rev. 0; C-068-163041-004 Rev. 0.
(6) The package is a one-time only exclusive use shipment.
(7) Transport of the package will be in accordance with the referenced Transport Emergency Response Plan.

CONCLUSION

Based on the statements and representation contained in the application, as supplemented, and the conditions listed above, the staff concludes that the package for the LACBWR RPV package meets the requirements of 10 CFR Part 71.

Issued with a letter of approval
on April 27, 2006.