

LARGE COMPONENT REMOVAL/DISPOSAL

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ABSTRACT

This paper describes the removal and disposal of the large components from Maine Yankee Atomic Power Plant. The large components discussed include the three steam generators, pressurizer, and reactor pressure vessel. Two separate Exemption Requests, which included radiological characterizations, shielding evaluations, structural evaluations and transportation plans, were prepared and issued to the DOT for approval to ship these components; the first was for the three steam generators and one pressurizer, the second was for the reactor pressure vessel. Both Exemption Requests were submitted to the DOT in November 1999. The DOT approved the Exemption Requests in May and July of 2000, respectively. The steam generators and pressurizer have been removed from Maine Yankee and shipped to the processing facility. They were removed from Maine Yankee's Containment Building, loaded onto specially designed skid assemblies, transported onto two separate barges, tied down to the barges, then shipped 2750 miles to Memphis, Tennessee for processing. The Reactor Pressure Vessel Removal Project is currently under way and scheduled to be completed by Fall of 2002. The planning, preparation and removal of these large components has required extensive efforts in planning and implementation on the part of all parties involved.

INTRODUCTION

The Maine Yankee Nuclear Power Plant was a 3-loop Pressurized Water Reactor rated at 870 Mwe net and 2700 Mwt. The NSSS supplier was Combustion Engineering (CE) and Stone & Webster (S&W) was the Architect/Engineer. Maine Yankee began commercial operation on December 28, 1972. It completed 15 cycles of operation with the last cycle of operation ending on December 6, 1996 when the plant ceased operation. The decision to go to a permanent shutdown was announced on August 7, 1997. Plant decommissioning began on August 4, 1998 with S&W as the decommissioning operations contractor (DOC). Maine Yankee Atomic Power Company took over as the DOC on July 1, 2000 when a decision was made to "self perform" the decommissioning.

At the time of this writing the three SG's and the Pzr have been removed, packaged, and transported to Duratek's processing facility in Memphis, Tennessee. Most of the preparations are completed for the removal, packaging, transport, and disposal of the RPV. The RPV internals segmentation has been completed and approximately 2/3's of the internals have been re-packaged in the RPV. The greater than Class C (GTCC) internals packaging is in process in the Refueling Cavity, in preparation for storage in the Independent Spent Fuel Storage Installation (ISFSI). Once the GTCC internals have all been packaged and removed from the Refueling Cavity, the water will be drained from the cavity and the RPV. The Refueling Cavity will then be decontaminated so the RPV Removal Project may begin.

COMPONENT DESCRIPTIONS

The large components for which this paper pertains include three Steam Generators (SG's), the Pressurizer (Pzr) and the Reactor Pressure Vessel (RPV).

The SG's were designed and fabricated in accordance with the ASME Boiler & Pressure Vessel Code, Section III (Reference 1). The primary coolant side of the SG's was designed for an internal pressure of 2500 psi at a design temperature of 650 degrees F. The secondary (steam) side of the SG's was designed for an internal pressure of 1000 psi at a design temperature of 550 degrees F. The SG's, which were designed and fabricated by CE, were approximately 59 feet long, with an upper shell outside diameter of

approximately 15 feet 7-1/2 inches, and a lower shell outside diameter of 11 feet 8-3/4 inches. The shell wall varied from a bottom bowl thickness of 7 inches to a cylindrical wall of 3-3/4 inches, and a top head of 2-1/2 inches. The dry weight of each SG was approximately 356 Tons.

The Pzr was designed for an internal pressure of 2500 psi at a design temperature of 700 degrees F to the requirements of Section III of the ASME Boiler and Pressure Vessel Code. It was about 37 feet long with a cylinder wall thickness of 4-7/8 inches. The Pzr had an outside diameter of approximately 10-1/2 feet. The dry weight of the Pzr was about 100 Tons.

The RPV was designed for an internal pressure of 2500 psi at a design temperature of 700 degrees F to meet the requirements of Section III of the ASME Boiler and Pressure Vessel Code. The RPV, which was designed and fabricated by CE, has an outside diameter of 15'-9" at the belt line with a wall thickness of 8-5/8". The total length (excluding the head) is 34 feet. The dry weight of the RPV is approximately 340 Tons, but when it is lifted from the Refueling Cavity it will weigh approximately 591 Tons because it includes 2/3's of the internals, grout, and a specially designed two part cover.

SITE ENGINEERING AND PREPARATION

Site engineering and preparation activities began in early 1999 to pave the way for the removal of the large components. The SG & Pzr Removal Project and RPV Removal Project both shared common site engineering and preparation needs. The site engineering activities included all of the technical evaluations, planning, and work order preparations necessary for the removal of these components. Site preparation included work that was implemented as a result of engineering efforts performed both inside and outside the Containment Building.

Containment interior work began with source term reduction in the loop areas and outer annulus of the Containment Building. Special "Gamma Cam" radiation inspections were performed to locate "hot spots" within the piping and qualitatively determine the best removal sequence. This effort was performed in conjunction with commodity removals to effectively remove the "hottest" (i.e., most radioactive) components and piping first, thereby reducing the worker exposure levels significantly.

The SG ("loop") areas and Pzr cubicle were selectively cleared out to remove all items that could interfere with the removal of these associated large components. Scaffolding platforms were erected in these areas to aid with the interference removals, and to provide safe work areas for removal of the base plate attachment bolts. These platforms were later utilized during the lifts of each of the large components for final verification that adequate clearance existed for the lifts. The Reactor Coolant Pumps and Loop Isolation Valves were also removed from the loop areas under the commodity removal plan, prior to large component removals. Special shield covers were fabricated and installed on each of the six - reactor coolant loop pipe ends to seal the pipe ends so that the RPV and Refueling Cavity could be flooded for RPV internals segmentation.

Engineering analyses were performed to verify the structural adequacy of the Polar Crane and support structure to lift and down-end the SG's. The Polar Crane wall had to be notched to provide sufficient space for down-ending the SG's and RPV. The Equipment Hatch opening was enlarged to approximately 32 feet high by 40 feet wide to allow for down ending and removal of the SG's and RPV.

The inside load paths for the SG's and RPV were evaluated to ensure that the floors and ground were sufficiently strong to take the loads. As a result of these evaluations, floor columns had to be designed, fabricated, and installed below the Equipment Hatch floor to give it sufficient strength to withstand the loads imparted upon it by the SG's and RPV. Special A-frame columns were also designed, fabricated, and installed so that one of the down-ending rails used for the RPV, which will overhang the Head Lay-down Area, will be supported.

All yard areas and transport paths were evaluated to confirm that the ground could sustain the loads imparted by the SG's, Pzr, and RPV. A lay-down area was created for the SG's and Pzr. This area was

leveled, back-filled with processed gravel, and compacted to establish a level and supportive surface for temporary storage of these components. Steel plates were placed over all storm drain and fire protection piping to protect it during transport of the components. A 175' long by 13'-6" high shield wall was erected on the outside edge of the lay-down area to shield the public from exposure to the SG's and Pzr. The barge slip access road was also straightened, widened, back-filled, and compacted for better accessibility to and from the barge slip.

STEAM GENERATOR AND PRESSURIZER REMOVAL EXEMPTION REQUEST

Maine Yankee submitted the Steam Generator (SG) and Pressurizer (Pzr) Exemption Request to the Department of Transportation (DOT) on November 16, 1999 (Reference 2). This exemption request identified that two SG's would be shipped on one barge and one SG and the Pzr on another barge for a total of two shipments. It was issued in accordance with the guidance provided in USNRC Generic Letter 96-07¹ to allow the shipment of three SG's and one Pzr package from the Maine Yankee Plant to the Duratek processing facility in Memphis, Tennessee. Requests were made for exemptions from the packaging requirements of 49CFR173.427(b)(1), to allow shipment of the three SG's and Pzr as unpackaged surface contaminated objects (SCO); and from the SCO contamination limits of 49CFR173.403 because it was not practical to determine the contamination level on each 300 cm² of the SG internals.

The Exemption Request package that was issued to the DOT included a Compliance Matrix, outline drawings of the SG's and Pzr, the SG and Pzr Waste Characterization, the Transportation and Emergency Response Plan, an evaluation of residual water in the plugged SG tubes, and the SG One Foot Drop Evaluation. The compliance matrix identified that the proposed packages met all DOT shipping requirements, with the exception of the two requirements for which exemptions were requested. It justified how the intent of all shipping requirements was met by the package design.

SG and Pzr Waste Characterization

The Waste Characterization determined the waste classification for disposal of the SG's and Pzr in accordance with 10CFR61 (Reference 3), and the transportation classification in accordance with 49CFR173 (Reference 4). The nuclide distribution for the SG's and Pzr were based upon smear sample analysis. This analysis data was decay adjusted to account for the time lapse between survey dates and data analysis. Dose-to-curie conversion factors were established by calculation. Radiation surveys were utilized to establish average dose rates for each SG and Pzr. The Cobalt-60 activity levels were established by multiplying the dose-to-curie conversion factors by the average dose rates. The 10CFR61 waste classification was established by determining the nuclide concentrations (i.e., by volume and mass of waste), then comparing these concentrations to the respective waste class limits in 10CFR61, Tables 1 and 2. All three SG's were determined to be **Class C waste**. The Pzr was determined to be Class A waste.

The 49CFR173 transportation classification was established by comparing the surface activity of each SG and the Pzr to the SCO limits, and the nuclide activities to the corresponding A₂ values in 49CFR173. All three SG's were determined to be **>A SCO-II**. The Pzr was determined to contain <A quantity of radioactivity. The bowl region was the only region that failed to meet the SCO-II limits, but overall the Pzr met the SCO-II limits.

Additional requirements were evaluated to confirm that the packages met conveyance limits, 3-meter unshielded dose rate limit, fissile material limits, and reportable quantities (RQ) limits. All packages met the 49CFR173 transportation criterion. Each of the SG's contained a reportable quantity of radioactive material.

Transportation and Emergency Response Plan

The Transportation and Emergency Response Plan established the minimum requirements for overall management, coordination and control for the safe shipment of the three SG's and Pzr from Maine Yankee

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to the Duratek site in Memphis, TN. This plan contained the prerequisites and restrictions for transport; a procedural checklist and signoff; a description of the routes to be taken; and directions for response to emergencies that could occur during transit from Maine Yankee to the Duratek site. The plan established the responsibilities for all parties involved with the transportation of the SG's and Pzr. The parties involved included the owner (Maine Yankee Atomic Power Co.), the decommissioning operations contractor (Stone & Webster), the barge contractor (Canal Barge Co.), the marine inspection service (Marine Safety Consultants, Inc.), the rigging and land transport contractor (Barnhart Crane & Rigging Co.), radiation protection contractor (Radiological Services, Inc.), and the processing facility contractor (Duratek).

Residual Water Evaluation

An evaluation of residual water in the plugged SG tubes was performed and included with the Exemption Request. Its purpose was to estimate the amount of gamma emitting radionuclides in the water trapped within the plugged tubes. The total activity of gamma-emitting nuclides in the trapped water in SG #1 (i.e., the worst case SG) was ~0.94 mCi. Since SG #1 had the greatest number of plugged tubes, 0.94 mCi was the bounding estimate for the other SG's.

Steam Generator One-Foot Drop Analysis

The final attachment that was included with the Exemption Request was the Steam Generator One-Foot Drop Analysis. This analysis was required to comply with the regulatory requirements of 49CFR173, which requires prevention of loss or dispersal of radioactive contents or significant increase in the external radiation dose rates when subjected to the tests prescribed in 49CFR173.465 (c) and (d), namely one-foot drop and compression tests. The one-foot drop requires that the package be evaluated for a drop in the most vulnerable orientation so that the maximum damage is caused. This evaluation demonstrated that the Maine Yankee SG's were capable of safely withstanding a free drop from a height of one foot on a flat non-yielding target, while maintaining their leak tightness and structural integrity, in accordance with the requirements of 49CFR173.

Steam Generator and Pzr Exemption Request Approval

The Exemption Request approval for the SG's and Pzr (DOT-E 12385) was received May 8, 2000, approximately 6 months after it was submitted. The regulations from which exemptions were granted by the DOT² included: "a. 49 CFR § 173.403 in so far as the fixed and non-fixed contamination levels for surface contaminated objects, class II, (SCO-II) are waived for the components specified in the exemption application" and; "b. 49 CFR § 173.427(a) in so far as requirement that SCO-II, must be transported in authorized packaging is waived. The components specified in the exemption application are authorized to be transported in non-specification packages under the specified transport plan."

SG AND PZR REMOVAL SUMMARY

The outside yard area preparations (i.e., the lay-down area, barge slip, enlargement of the Equipment Hatch, and construction of the shield wall) were completed in March 2000. Closure covers were installed on the components prior to removal of the components from the Containment Building (with the exception of the stay tube covers at the bottom of the SG's which were installed after they were down-ended). The component removals began in April 2000 with the removal of the Pzr first. The sequence for removal was the Pzr, SG #1, SG #3 then SG #2. Each component was lifted from its respective location via the Polar Crane. The Pzr weighed approximately 100 Tons and each of the SG's weighed approximately 356 Tons. Each component was lifted off its base up to above the Charging Floor elevation, trolleyed and bridged over to the Equipment Hatch area, then lowered down on a specially designed and fabricated down-ending base plate, which was pre-positioned in the Equipment Hatch. Each component was then bolted to its down-ending base plate, and down-ended via the Polar Crane and Barnhart's slide track system (Figure 1).

Once in a nearly horizontal position, the base plate was linked to a lift beam, which rested on top of two hydraulic jacks. The hydraulic jacks and lift beam formed a rolling gantry that lifted the bottom end of each component to a horizontal orientation. The respective shipping carriage assembly was loaded on a transporter, and the transporter/carriage assembly was then backed under the component. The component was lowered onto the carriage assembly via the Polar Crane and rolling gantry. Radiation Protection personnel then surveyed the component and the rigging was disconnected. Following the completion of the survey and rigging disconnection, the component was driven out of the Equipment Hatch. The Pzr was driven to its pre-determined location in the lay-down area. Each of the SG's were driven out of the Equipment Hatch, then stopped so that the stay tube plugs could be welded onto the bottom (i.e., underneath the "tea cup" shaped base). After the stay tube plugs were tack-welded in place, the SG's were driven to the lay down area and parked in their respective pre-determined positions. The second phase of the component removal process was preparation for transport. Each of the components was welded to the saddles on its respective carriage assembly. The shield frames and panels were then installed on the carriage assemblies. This work took place in April and May of 2000. The first barge, Canal Barge Companies CBD 4501, arrived at Maine Yankee the end of May. SG's #2 and #3 were loaded onto the first barge via a 9-line, double-wide, transporter.



Fig. 1. Steam Generator #1 being down-ended

Once the components were off-loaded from the transporter onto the barge, they had to be welded and tied down to the barge deck. This evolution took approximately 2 weeks. At the end of this evolution the barge compartments were pumped out (i.e., the barges were flooded to seat them on the river bottom) and then inspected by the Marine Inspectors.

Barge CBC-4501 left Maine Yankee loaded with SG's #2 and #3 on June 12, 2000 (Figure 2.) and arrived in Memphis, TN 21 days later on July 3, 2000. Canal Barge Companies CBC-4506 left Maine Yankee loaded with SG #1 and the Pzr on June 28, 2000 and arrived in Memphis 15 days later on July 13, 2000. The barges were off-loaded at the Port of Memphis by Barnhart Crane & Rigging via Barnhart's 1250 Ton stiff-legged derrick crane ("Icabod Crane"), and transported ~2.5 miles to Duratek's waste processing facility on President's Island.



Fig. 2. Steam Generators #2 and #3 leaving Maine Yankee

RPV REMOVAL EXEMPTION REQUEST

Maine Yankee submitted the RPV Exemption Request to the Department of Transportation (DOT) on November 10, 1999 (Reference 5). Two exemptions were requested from the requirements of 49CFR173 for the shipment of the RPV and internals from the Maine Yankee plant to Chem-Nuclear Systems (CNS) Low Level Radioactive Waste Management Facility in Barnwell, SC. Exemptions were requested from the packaging requirements of 49CFR173.427(a)(1), to allow the radiation dose limit of 1 Rem/hour at 3 meters from the unshielded radioactive material for the RPV internal components to be exceeded; and from the requirements of the 49CFR173.465(c) free drop test, to allow testing of the package for the one-foot drop in only a horizontal orientation.

The Exemption Request package that was issued to the DOT included a Compliance Matrix and Summary Report. The Summary Report included a Package Description, Characterization, Structural Evaluation, Thermal Evaluation, Shielding Evaluation, Package Handling instructions, and Transportation and Emergency Response Plan. The Compliance Matrix identifies that the proposed package meets all DOT shipping requirements, with the exception of the two requirements for which exemptions were requested. It justifies how the intent of all shipping requirements are met by the package design.

RPV Package Description

The components that make up the RPV package are the RPV (including portions of the shell insulation), the RPV internals, cellular concrete (interior and annulus) and the RPV container (see Figure 3). A brief description of each of these components, and their associated weights, are provided below.

The Maine Yankee RPV is approximately 34 feet long without the closure head. At the belt line, the approximate outside diameter and wall thickness are 15 feet 9 inches and 8-5/8 inches, respectively. The weight of the packaged portion of the RPV is 340 Tons.

WM'02 Conference, February 24-28, 2002, Tucson, AZ

The insulation that will be included with the package is the “hour glass” shaped sections between each nozzle and the cylindrical shaped section, which is installed on the outer shell below the RPV nozzles down to approximately the bottom head attachment weld. This insulation is made up of prefabricated stainless steel panels. Each insulation panel contains multiple layers of stainless steel foil, which are sandwiched between three sheets of heavier gage stainless steel, creating two layers of foil. The weight of the insulation that will be included with the package is approximately 3 Tons.

The insulation will be removed from the around the nozzles (i.e., the “doughnut” rings), nozzle piping, portions of the “hour glass” shaped sections (i.e., to make room for the nozzle cutting equipment), above the nozzles, and below the RPV cylinder wall-to-lower head attachment weld. This removed insulation will be disposed of as low level rad waste.

The RPV internals that will be returned to the RPV are itemized in Table I below. The total weight for these internals is 127 Tons.

The concrete that will be installed in the RPV includes a combination of interior and annulus concrete. The interior concrete will be placed inside the RPV to lock down any contamination and fix the internals in place. The annulus concrete will be placed around the outside of the RPV to fill the void between the outside of the RPV and the inside surface of the container. A combination of low density concrete (i.e., 40 +/- 10 lbs/cu ft) and normal density concrete (145 +/- 5 lbs/cu ft) are used in both the interior and annulus placements (see Figure 3.). The normal density concrete is used in the RPV nozzles for shielding, and at the bottom and top sections of the annulus for structural support. Low density concrete is used for all other placements. The interior concrete weighs 96 Tons and the annulus concrete weighs 108 Tons, yielding a total of 204 Tons of concrete (i.e., normal and low density combined).

The RPV shipping container, which will house the package components, measures 19 feet 1 inch in diameter and 35 feet 8 inches high. It consists of a three-inch thick steel cylinder with four-inch thick closed ends. Supplemental radiation shielding is provided by a two-inch thick steel cylinder located opposite the RPV core region and a two-inch thick steel plate located on top of the RPV. The 4-inch thick top plate and 2-inch thick supplemental top shield plate replace the reactor vessel closure head. The total weight for the shipping container is 243 Tons.

The total RPV shipping package weight is:

RPV	340 Tons
Insulation	3 Tons
Internals	127 Tons
Concrete	
Low density	96 Tons
Normal density	108 Tons
<u>Container</u>	<u>243 Tons</u>
Total Weight	917 Tons

Once assembled, the RPV package has no operational features. Fill and vent ports in the DOT approved transport container and its cover will be seal-welded and rendered inoperable prior to off-site transport. There are no valves, connections, piping, accessible openings, or seals associated with the RPV package.

The RPV package tie-down system consists of saddle assemblies, wire ropes, and end stops designed in accordance with ANSI N14.24 (Reference 6). The container was designed using the requirements of Section VIII of the ASME Boiler and Pressure Vessel (B&PV) Code (Reference 7). Criteria from ASME Section III (Reference 8) were used to supplement ASME Section VIII for test cases. The use of the ASME code for the design and fabrication of the RPV shipping container provided reasonable assurance that the 49CFR173 requirements for package integrity are met

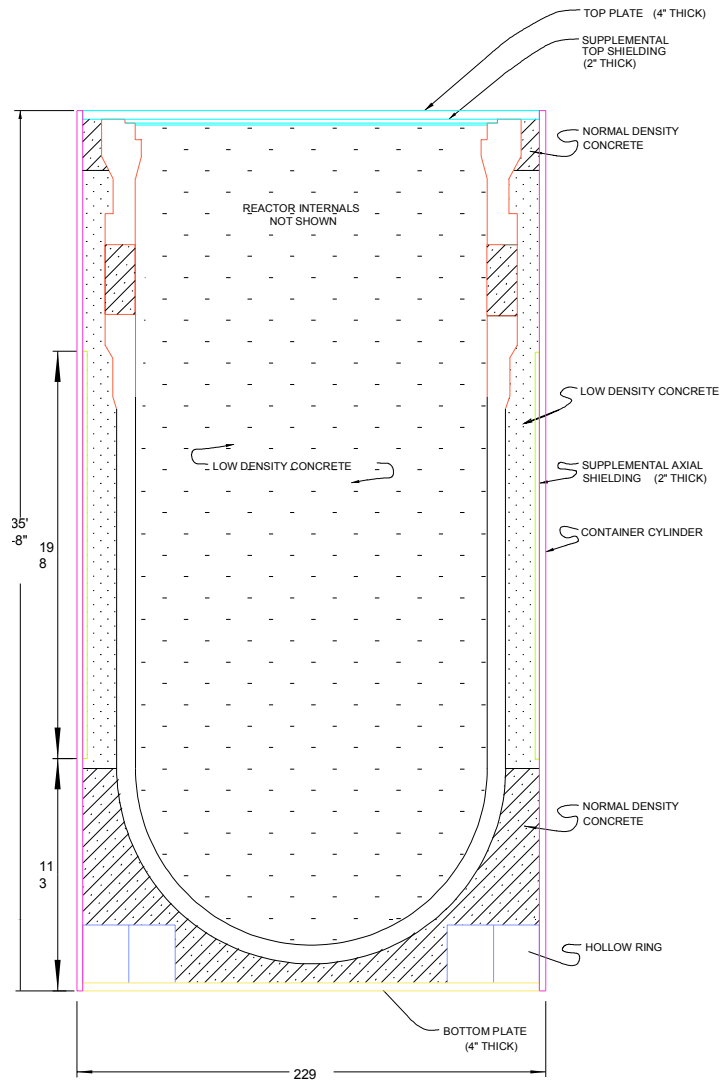


Fig. 3. Cross-Section of the RPV Package

RPV Package Waste Characterization

Characterization was used to determine the waste classification for disposal in accordance with 10CFR61 (Reference 3) and 10CFR71 (Reference 9), and the transportation classification in accordance with 49CFR173 (Reference 4). The characterization of the RPV Package is the comparison of the calculated radioactivity within the vessel to the regulatory limits. It is based on the activity and mass of the activated metal components within the RPV Package. The activity of each component within the package was determined via the performance of an activation analysis of the RPV and internals. The activation analysis is the calculation of radioactivity in the RPV and surrounding structures specific to the reactor geometry, material composition, and irradiation history. The calculated activities were projected for the originally scheduled ship date of March 1, 2001. Maine Yankee operated the 2,700 MWth Combustion Engineering pressurized water reactor reliably starting in 1972. The unit underwent 15 cycles of operation (the first cycle was split into cycle 1 and 1a) with the last cycle of operation ending on December 6, 1996. The activation analysis encompassed this entire period of operation with the results decayed to March 1, 2001. The results of these analyses for each component, and for the total package, were compared to the DOT

transportation limits for "Type" and "Sub-Type" for the purposes of characterization. Current packaging plans result in a package with an estimated total activity of 49,100 curies and a specific activity of 1.09E-04 curies/gram. The specific activity of the most activated component is 5.2 percent of the LSA III limit.

The 10CFR61 waste classification was established by determining the nuclide concentrations (i.e., by volume and mass of waste), then comparing these concentrations to the respective waste class limits in 10CFR61, Tables 1 and 2. The RPV Package was determined to be **Class C waste**.

The 49CFR173 transportation classification was established by comparing the surface activity of the RPV Package to the SCO limits, and the nuclide activities to the corresponding A_2 values in 49CFR173. The RPV Package was determined to be **LSA III**.

The most activated component within the RPV Package is the thermal shield. It is characterized as containing a greater than Type A quantity of LSA III radioactive material. All other components contain either a Type A quantity or greater than a Type A quantity of LSA II or LSA III material. The total package DOT determinations are represented in the Table I.

The activated metal components within the reactor vessel and the reactor vessel internals are also surface contaminated. Laboratory results from primary system samples were used to estimate component surface area activity. The calculated surface area from each component was used in conjunction with the sample results to determine activity. These surface contamination activities per component were added to the activation activities per component to estimate total activity.

The surface contamination on the reactor vessel interior surface and reactor internals is uniformly distributed on all wetted surfaces. The activity contribution from surface contamination is approximately 91 Ci, which is less than 1 percent of the total package activity. Surface contamination on the reactor vessel external surfaces and insulation are indicative of general area contamination and are deemed insignificant to this package.

The leach testing requirements under 49CFR173.468 assume a loss of packaging and mobility of exposed activity. The components within the reactor vessel package are activated metal components with some surface area contamination. Since the activated metal itself is considered immobile, and all components are grouted in place within the reactor vessel, only the surface area contamination on the outside of the reactor vessel is considered available for leaching. The activity resulting from surface area contamination on the outside of the reactor vessel and insulation are representative of general area contamination and are considered insignificant to the package.

Table I: Maine Yankee Reactor Vessel - Component Classification Summary

ID#	Component Name	Activity (Ci) 3/1/2001	Class (10CFR61)	DOT Type (49CFR173)	DOT LSA (49CFR173)
1	Extension Shaft Guide Assembly	6.89E+00	B	A	LSA II
2	Upper Guide Support Plate	9.99E+00	A	A	LSA II
3	Upper Guide Support Flange Segment	1.37E+00	A	A	LSA II
4	Expansion Compensating Ring	1.56E+00	A	A	LSA II
5	Control Element Shroud Assembly	6.39E+02	B	>A	LSA II
6	Fuel Assembly Alignment Plate	6.36E+03	C	>A	LSA III
7	Thermal Shield Remnants	2.09E+04	C	>A	LSA III
8	Core Support Barrel Remnants	1.65E+04	B	>A	LSA II
9	Core Support Assembly Skirt	3.36E+02	B	>A	LSA II
10	Bottom Support Plate	2.92E+00	A	A	LSA II
11	Support Beams and Flanges	1.21E+01	A	A	LSA II
12	Support Columns	1.90E+03	C	>A	LSA II
13	Instrumentation Support Assembly	7.51E+00	B	A	LSA II
14	Reactor Pressure Vessel and Clad	2.39E+03	A	>A	LSA II
15	Reactor Pressure Vessel Insulation	3.86E+01	A	>A	LSA II
	Reactor Vessel Assembly with Remaining Internals	4.91E+04	C	>A	LSA III

Structural Evaluation

The Maine Yankee RPV package contains only LSA-II and -III materials and will be transported as exclusive use. Therefore, in accordance with 49CFR173.427, Table 8, the package was designed as an IP-2 package per the requirements of 49CFR173.410 and 49CFR173.411. The shipping container was designed and fabricated using the guidance of Section VIII of the ASME Boiler and Pressure Vessel (B&PV) Code and was evaluated for all normal and test conditions required by 49CFR173 for IP-2 packaging. Based on the controls instituted by the transportation plan, the package is evaluated for one-foot drops from a horizontal orientation only. An exemption is requested from the test requirements of 49CFR173.465 to allow this. Criteria from ASME Section III were used to supplement ASME Section VIII for test cases. The use of the ASME code for the design and fabrication of the RPV shipping container provides reasonable assurance that the 49CFR173 requirements for package integrity are met.

The structural design loads for the normal conditions of transport included thermal, pressure, mechanical, and test loads. The thermal loading that was considered included stresses produced from temperature gradients through the shipping container wall or along the container length, and stresses that arise when there is differential expansion between the contents of the container and the container itself. These temperature gradients are a result of insulation and decay heat. Internal pressure for the package was also evaluated for the effects of insulation and decay heat.

Mechanical loading of the package was evaluated per 49CFR173.410(f), which states that the packaging must be capable of withstanding the effects of any acceleration, vibration, or vibration resonance that may arise under normal conditions of transport. Stresses are produced in the container due to interaction of the shipping package with its tie down system during the shock and vibration normally incident to transport. The package was evaluated for the wave acceleration loads defined in ANSI N14.24. The wave accelerations from ANSI N14.24 are considered to envelope the accelerations the RPV shipping package will experience during its short overland transport because of the slow speed of the transporter and the controls that will be exercised by the transport plan during highway shipment. The wave accelerations are

also considered to envelop any other vibrations or vibration resonances that may occur during package transport. The package was not evaluated for the ANSI N14.24 collision accelerations since a collision is not considered a normal condition of transport.

In accordance with 49CFR173.411(b)(2), the packaging must prevent loss or dispersal of its radioactive contents and must not allow a significant increase in the radiation levels recorded on its external surfaces when subject to the tests specified in 49CFR173.465(c) and (d). The package is assumed to be in equilibrium at an ambient temperature of 100°F (49CFR173.461). The specified tests included: a) 49CFR173.465(c): One-foot drop onto an essentially unyielding surface. Three-drop scenarios were evaluated. All drop scenarios started with the package in a horizontal position since the RPV package will only be transported in a horizontal orientation and the transport is controlled by the transport plan; and b) 49CFR173.465(d): Stacking test. The package was subjected to a compressive load equal to the greater of (a) a pressure of 1.9 psi times the vertically projected area of the package, or (b) five times the mass of the package.

This structural evaluation shows that the Maine Yankee RPV Package design is sufficient to meet the regulatory design requirements. This package design provides for a high level of assurance that there will be no loss or dispersal of its radioactive contents, and no significant increase in the radiation levels at its external surface, when subjected to the normal conditions of transport and the test conditions for an IP-2 package described in 49CFR173, Subpart I.

Shielding Evaluation

The components of the RPV package include the activated RPV (without the closure head), the activated metal external insulation and the activated reactor internals. The reactor internals have been segmented such that no greater-than-Class-C (GTCC) material is contained in the package. The reactor internals have been placed in the RPV and will be grouted so they will remain fixed within the RPV.

49 CFR 173.441 provides the external radiation level limitations for packages for the transport of radioactive materials. Per 49CFR173.441, "each package of Class 7 (radioactive) materials offered for transportation must be designed and prepared for shipment, so that under conditions normally incident to transportation, the radiation level does not exceed 2 milliSieverts (mSv)/hr (200 mrem/hr) at any point on the external surface of the package, and the transport index (TI) does not exceed 10". The RPV package TI (i.e., the maximum radiation level at one meter from the external surface of the package times 100) is greater than 10, therefore, the following 49CFR173.441(b) limits cannot be exceeded for the transportation of the RPV package:

- 200 millirem/hr on the external surface of the package (49CFR173.441(b)(1))
- 200 millirem/hr at any point on the outer surfaces of the transport vehicle, including the top and underside of the vehicle; (49CFR173.441(b)(2))
- 10 millirem/hr at any point two meters from, in the case of an open vehicle, the vertical planes projected from the outer edges of the conveyance (49CFR173.441(b)(3))
- The requirements for 49CFR173.441(b)(4) are met by the Transportation Plan, therefore, the requirements for this section do not apply to the shielding evaluations

The calculated dose rates at 0 and 2 meters from the surface of the RPV package are given in Table II. These dose rates are well below the requirements of 49CFR173.441(b). These calculated values will be verified prior to shipment.

The dose rates at 0 and 3 meters from the unshielded reactor vessel and insulation were calculated using source terms from the activation analysis and normalized to the measured contact dose rate of 5 R/hr. The dose rate measurements were taken with a probe in contact with the RPV exterior insulation. At the time of the measurements, all reactor internals were in-place and the vessel filled with water. Even though measured on contact with the insulation, contact dose rate measurements are considered conservative since the activated neutron shield tank located approximately 6 inches away will increase the recorded dose rate

readings. The influence of the reactor internals and water is negligible due to the shielding provided by the 8.625-inch thick, steel reactor vessel wall. The worst case dose rate at 3 meters from the vessel insulation for the vessel and insulation only is 5 mSv/hr (0.5 rem/hr).

As an additional assurance of the safety of the intended shipment, a dose rate calculation was performed for the vessel as prepared for shipment, but without the outer packaging. In the packaged configuration, the internals are securely grouted within the reactor vessel, all reactor vessel openings are sealed with grout and/or metal plates, all additional grout is in place and the top cover and supplemental 2-inch thick shield plate installed.

As noted above, the inclusion of the internals has no impact on the dose rates from the vessel sidewall - that dose rate is less than 1 rem/hr at 3 meters. Since the bottom head is thinner and the bottom head insulation will be removed in preparation for shipping, the dose rate at 3 meters from the bottom head was calculated using the characterization activities as the source term with the grout included. The calculated dose rate at 3 meters from the vessel bottom head, with the internals in their packaged configuration, is less than 1 rem/hr. Thus, at all locations around the vessel as prepared for shipment, the dose rate at 3 meters is less than 1 rem/hr.

The activated RPV components are the RPV cladding, the RPV wall, and the RPV insulation. Approximately 90% of the activity within the reactor vessel is located adjacent to the active core region and 99% of the activity is contained within +/- 2 feet of the active core region. For conservatism, all activity was assumed to be located in the active core region. The calculated dose rates from these components were normalized to the measured dose rate of 5 R/hr in contact with the vessel sidewall insulation. The final calculated dose rates were used to evaluate the package shielding and the dose rates at 3 meters for the unshielded reactor vessel with internals packaged inside.

The dose rate at each dose point was obtained by summing the contributions from all sources. At the side of the RPV package, the dose rate contribution from the activated reactor internals is negligible due to the shielding from the 8.625-inch thick RPV wall. The dose rates at 0 and 2 meters from the sidewall of the shipping container were calculated to be 4.4 and 2.3 mrem/hr, respectively. At 0 and 2 from the bottom surface of the shipping container, the sum of the dose rate contributions from the RPV and the reactor internals were calculated to be less than 1 mrem/hr. These dose rates are well below the 49CFR173.441 limits.

Each Primary coolant nozzle is filled with normal density (145 pcf) concrete. This is equivalent to approximately 7-inches of steel. Since the nozzle elevation is more than 4 feet above the top of the core region, the source strength near the nozzle is at least 10 times less than the source strength of the RPV wall around the core region. Thus, at the same distance from the side surface of the package, even with slightly less shielding (7-inches vs. 8.625-inches), the dose rate at the nozzle elevation will be equal to or less than that at the core centerline elevation.

The principal sources of the reactor internals remaining in the reactor vessel are placed near the bottom of the vessel. Since the vessel is filled with low density concrete, the dose points above the top of the container are shielded by a much thicker layer of concrete than the dose points below the bottom of the container. Thus, the dose points below the bottom of the container are more limiting than the corresponding dose points above the top of the package.

Table II: Summary of Calculated Dose Rates for RPV Package
(Based on 3/1/01 Source Terms)

	Surface Dose Rate ⁽¹⁾ (mrem/hr)	2 meter Dose Rate ⁽²⁾ (mrem/hr)
Side of RPV Package	4.4	2.3
Bottom of RPV Package	<1.0	<1.0

NOTES:

- (1) Maximum allowable dose rate is 200 mrem/hr (§173.441(b)(1)).
- (2) Maximum allowable dose rate is 10 mrem/hr (§173.441(b)(3)).

Table III: Summary of Calculated Unshielded 3-meter Dose Rates for RPV Package
(Based on 3/1/01 Source Terms)

	Side	Bottom
RPV plus Insulation	.5	(1)
RPV plus Insulation and Internals	.5	<1.0E-03

NOTE:

- (1) Dose rate at the bottom will be much less than side because of the distance from the core region and lack of activated insulation.

Package Handling

This section of the Exemption Request describes the preparation of the RPV and internals for shipment, the loading of the package content into the shipping container, the preparation of the container for shipment and handling of the RPV package from the containment to the transport barge. All of the activities described will take place within the Maine Yankee site. Handling of the package in the public domain from the onset of the barge leaving the Maine Yankee site to entry to the disposal facility in Barnwell, South Carolina is detailed in the Transportation Plan.

Transportation and Emergency Response Plan

Transportation and Emergency Response Plan is the last chapter of the Exemption Request. It contains the minimum requirements for overall management, coordination and control for the safe shipment of the RPV from the Maine Yankee barge slip to the burial site in Barnwell, South Carolina. This plan contains all of the prerequisites and restrictions for transport including a procedural checklist and signoff; a description of the routes to be taken; and directions for response to emergencies that could occur during transit from Maine Yankee to Barnwell, South Carolina. The plan contains the responsibilities for all parties involved with the transportation of the RPV. The parties involved include the owner (Maine Yankee Atomic Power Co.), the barge transport contractor (Lockwood Brothers, Inc.), the land transport contractor (Mammoet, Inc.), and the waste disposal contractor (Duratek – previously Chem-Nuclear Systems, Inc.).

Exemption Request Approval

The Exemption Request approval for the RPV transportation (DOT-E 12386) was received on July 3, 2000, approximately 8 months after it was submitted. The regulations from which exemptions were granted by the DOT³ include: “a. 173.427(a)(1) in that the dose rate at 3 meters from the unshielded radioactive material in this shipment may exceed 10 mSv/hr (1 rem/hr)”; and “ b. Section 173.411 (including

173.465(c)) in that packaging containing the reactor vessel and its internals components may be certified as an IP-2 package even though compliance with all listed provisions has not been demonstrated.”

RPV REMOVAL ENGINEERING AND PREPARATION

Additional engineering evaluations and modifications were necessary for the RPV removal project. First among these was the evaluation of the Polar Crane and its support structure. The RPV, with the re-packaged internals and grout, will weigh approximately 591 Tons when it is lifted out of the Refueling Cavity. Barnhart Crane & Rigging (the RPV rigging contractor) will install a “strand jack” system on top of the Polar Crane girders to make this lift (i.e., only the Polar Crane girders and support structure will be utilized). The Polar Crane is only rated for 360 Tons so the girders and support structure were evaluated to determine their capability to handle this load. The engineering evaluation revealed that the Polar Crane girders needed to be modified to provide additional strength. The girders have been modified, via the installation of horizontal stiffeners, which were welded to the girders to provide the additional strength.

A girder deflection calculation was also performed to provide a girder deflection control limit during the actual RPV lift. A laser beam will be affixed to the Polar Crane girder at mid-span and directed to point to a scale that will be attached to the Polar Crane wall. The laser beam will be shone on the scale with the girder in the unloaded condition, then the calculated deflection will be marked on the scale. If the laser light indication reaches the calculated deflection mark during while the RPV is being lifted, the lift will be terminated, and an engineering evaluation will be performed to reassess the situation.

The Polar Crane support structure and strand jack system will be load tested prior to the RPV lift to verify the adequacy of the system. This load test will be performed with the strand jack system installed on the Polar Crane girders. A load beam will be inserted through a hole in a 3-foot thick concrete wall, then rigged to the strand jack system with the rigging that will be utilized for the RPV lift. Load cells will be installed between the load beam and the top of the hole. The crane/strand jack system will be load tested to 680 Tons prior to the RPV lift to simulate the conditions of the lift and verify that the crane/strand jack system is adequate for lifting the RPV. A pressure vs. load curve will be generated during the load test. This curve will be used during the RPV lift to determine the actual weight of the RPV.

Barnhart has developed the Rigging Procedure that will be utilized for their work activities. These work activities include the static load test, movement of the empty RPV shipping container from the transport skid assembly into the Containment Building, the RPV lift, the RPV package down-ending, and the RPV package movement from the Containment Building back to the transport skid assembly. The Rigging Procedure establishes the plan for implementation of these activities. It will be used in conjunction with a Maine Yankee work order to ensure that all of the engineering and administrative controls are properly implemented. Pre- and post-inspections of the Polar Crane will also be performed for the static load test and the RPV lift to assess the condition of the Polar Crane.

Mock-up testing has been utilized as a tool to flush out potential problems and gain familiarization before the actual work begins. Mock-up testing has been performed to simulate grouting, nozzle cutting, and strand jack system installation. It will also be performed to simulate container-to-cover welding. These mock-up tests have proved to be very helpful and will result in considerable savings in exposure.

FOOTNOTES

¹ USNRC Generic Letter 96-07, Interim Guidance on Transportation of SG's, dated December 5, 1996.

² DOT-E 12385, dated May 8, 2000 from U.S. Department of Transportation, Research and Special Programs Administration

³ DOT-E 12386, dated July 3, 2000 from U.S. Department of Transportation, Research and Special Programs Administration

REFERENCES

1. ASME Boiler & Pressure Vessel Code, Section III, (latest edition in effect on 5/1/67)
2. MYAPCO No, GAZ-99-35, "Request for Exemption for Shipment of the Maine Yankee Plant Steam Generators and Pressurizer", dated November 16, 1999
3. 10CFR61, Licensing Requirements for Land Disposal of Radioactive Waste
4. 49CFR173, Subpart I, Class 7 (Radioactive) Materials, Revised October 1, 1998
5. MYAPCO No, GAZ-99-34, "Request for Exemptions for Shipment of the Maine Yankee Atomic Power Station Reactor Pressure Vessel to the CNS Barnwell, S.C. Burial Site", dated November 10, 1999
6. ANSI N14.24 – 1985 (R1993), American National Standard for Highway Route Controlled Quantities of Radioactive Materials – Domestic Barge Transport
7. ASME Boiler & Pressure Vessel Code, Section VIII, Division 1, Rules for Construction of Pressure Vessels, 1995 edition
8. ASME Boiler & Pressure Vessel Code, Section III, Division 2, Rules for Construction of Nuclear Power Plant Components, 1995 edition
9. 10CFR71, Packaging and Transportation of Radioactive Material