Proceedings of the 15th International Symposium on the Packaging and Transportation of Radioactive Materials
PATRAM 2007
October 21-26, 2007, Miami, Florida, USA

Packaging and Transportation of the Dairyland LACBWR Reactor Vessel

Mark S. Lewis
EnergySolutions

Abstract
The packaging and transportation of large, retired reactor components poses unique challenges from a technical as well as regulatory compliance standpoint. The current regulatory framework, domestically in Titles 49 and 10 and internationally in TS-R-1, does not lend itself to the transport of these large radioactively contaminated components, such as reactor vessels, steam generators, reactor pressure vessel heads, and pressurizers, without application for a special permit or arrangement.

This paper addresses the methods of overcoming the technical and regulatory challenges associated with the packaging and transportation of the Dairyland LaCrosse Boiling Water Reactor's (LACBWR) reactor pressure vessel (RPV). After the LACBWR reactor vessel was packaged and prepared for transportation, multiple domestic transport modes were utilized under a special package authorization from the U.S. Nuclear Regulatory Commission, 10 CFR 71.41, and special transport permit from the U.S. Department of Transportation, 49 CFR 107.105, to facilitate the disposal of the 312 ton package containing 9,790 curies.

Introduction
The La Crosse Boiling Water Reactor (LACBWR) is owned and was operated by Dairyland Power Cooperative (DPC) of La Crosse, Wisconsin.

LACBWR was a nuclear power plant of nominal 50 Mw electrical output, which utilized a forced-circulation, direct-cycle boiling-water reactor as its heat source. The plant is located on the east bank of the Mississippi River in Vernon County, Wisconsin, approximately 1 mile south of the village of Genoa, Wisconsin, and approximately 19 miles south of the city of La Crosse, Wisconsin.

LACBWR achieved initial criticality on July 11, 1967, and the low power testing program was completed by September 1967. In November 1967, the power testing program began. The power testing program culminated in a 28-day power run between August 14 and September 13, 1969.

DPC operated the facility as a base-load plant on its system since November 1, 1969, when the Atomic Energy Commission (AEC) accepted the facility from Allis-Chalmers, until LACBWR was permanently shut down on April 30, 1987. During this time the reactor was critical for a total of 103,287.5 hours

Packaging and Contents Description
The LACBWR reactor vessel packaging consists of a steel canister surrounding the reactor pressure vessel, with the annulus between the vessel and the canister filled with concrete, as shown in the drawings in Figure 1. The canister is formed of a 1.5" steel cylindrical shell with end plates of 4" steel plate. The completed package is 39' 7" long with an outer diameter of 10' 6". The total weight of the package is 624,500 lbs. All joints in 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 reactor core. The lower section of the canister has a raised flat ring, on which the eight (8) RPV support legs rest when the RPV is placed inside the lower section of the canister. There are no tie-down devices that are a structural part of the packaging and, at the time of shipment, there are no operable lifting attachments that are a structural part of the packaging. The packaging will be fabricated and assembled in accordance with an NRC approved Part 71 Quality Assurance program.

The contents of the LACBWR RPVP are the irradiated reactor pressure vessel and the reactor internals. 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 has an overall height of 37', an inside diameter of 99", and a nominal wall thickness of 4" (including 3/16" of integrally bonded stainless steel cladding). The reactor vessel is ferritic steel (ASTM A-302-Gr-B) plate with integrally bonded Type 304L stainless steel cladding. The reactor internals consist of the following: 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 steam dryer with support structure, an emergency core spray tube bundle structure combined with fuel hold-down mechanism, control rods, and reactor core support structures. The internals are made of AISI Type 304 stainless steel. The voids in the reactor vessel will be filled with low-density cellular concrete (LDCC) prior to cutting the nozzles and lifting the vessel to remove it from the reactor building. The total weight of the filled vessel is 185 tons.

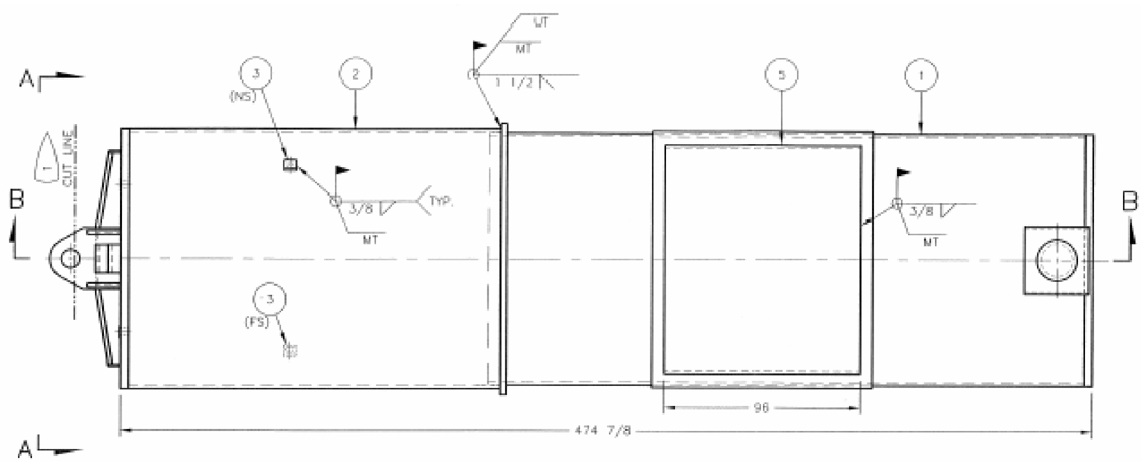

Figure 1 – LACBWR Reactor Vessel Package

Radionuclide Content

In 2003, the results of an analytical determination of nuclide activation levels in the LACBWR reactor pressure vessel (RPV), reactor internals, and subcomponents was documented in a characterization report (Ref. 1). A dual approach was used – manual calculation using a simplified reactor model for isotopic irradiation/decay, and a detailed irradiation analysis using the ORIGEN-ARP 2.00 computer code. The resulting activity was decayed from the time of shutdown (April 1987) to January 2003. The manual calculation results were nearly a factor of two higher than the ORIGEN-ARP results, 15,631 versus 8,130 total curies. In the activation calculation, the upper bound material percentage for niobium was used from NUREG/CR-6567 (Ref. 2). The listed range for niobium in stainless steel (the predominate material in the reactor internals) is 5-300 ppm, so 300 ppm was used in the activation calculation. Thus, the results for Nb-94 are extremely conservative.

A surface coating evaluation, based on removable contamination samples from the Shutdown Condenser, was performed for the internal surfaces of the RPV and internals. The measured activity per unit area was distributed over the area of the vessel and internals, 1047.47 m^2, to determine the total activity of surface contamination in the vessel. This activity is a small fraction of the total activity.

Three fuel assembly designs (Type I, Type II, and Type III) were used in the LACBWR reactor (Ref. 3). All assemblies were stainless steel clad. Visible fuel rod clad failures were evident in many spent Type I and Type II fuel assemblies. There has been no evidence of any fuel rod clad failures in Type III fuel. During refueling in 1977 and in 1979, after the grossly failed fuel assemblies were moved from the reactor to the spent fuel pool (SFP), several pieces of fuel rod and fuel debris were recovered from the tops of other fuel assemblies and control rods in the reactor and placed in the SFP. During 1977, a significant fraction of the reactor internals, including other fuel assemblies, tops of control rods, below the core, unfueled positions, steam separator down-comer region, etc, was examined and searched for identifiable fuel debris. Very little other than a few small pieces of fuel clad was found, and all were recovered and placed in the SFP. Cladding failures decreased after Cycle 5 (Mar. 1978- Mar. 1979).

After detailed examination of the failed fuel rods, an estimate of the amount of uranium displaced from the failed rods was made. After including the collected debris and the uranium identified in waste shipments sent offsite for disposal, a residual of 58.2 grams of uranium remains (Ref. 4). It is assumed that this material is distributed throughout the primary system and has plated out on the reactor vessel, internals, primary system piping, and other primary system components outside the vessel. The calculated TRU generated from this distributed uranium is shown below (decayed to January 2003), based on the burnup and power levels characteristic of the LACBWR reactor. The activity per unit area conservatively assumes 100% of the activity is distributed only on the reactor vessel and internals. This gives a TRU activity per unit area approximately 100 times the measured contamination values included in the characterization report.

Table 1-1 Activity from Residual Uranium

Radionuclide	Grams	decayed Ci	μCi/cm2
U-234	1.76E-02	1.09E-04	1.04E-05
U-235	1.13E+00	2.49E-06	2.38E-07
U-236	1.98E-01	1.29E-05	1.23E-06
U-238	5.69E+01	1.93E-05	1.85E-06
Np-237	7.82E-03	5.55E-06	5.30E-07
Pu-238	1.81E-03	2.72E-02	2.59E-03
Pu-239	2.66E-01	1.65E-02	1.57E-03
Pu-240	7.15E-02	1.64E-02	1.57E-03
Pu-241	3.64E-02	1.71E+00	1.63E-01
Pu-242	6.84E-03	2.67E-05	2.54E-06
Am-241	1.26E-03	4.19E-03	4.00E-04
Am-243	9.25E-04	1.85E-04	1.76E-05
Cm-242	3.01E-04	2.52E-11	2.40E-12
Cm-244	1.46E-04	6.48E-03	6.19E-04
Total	5.86E+01	1.78E-01	1.70E-01

The results were updated by substituting the conservative estimate (Table 1-1) of the contamination levels from uranium and TRU that could be present due to the fuel failures that occurred during operation. Finally, the activities were decayed to the expected date of shipment, i.e., June 1, 2007. The resulting activity is shown in Tables 1-2 and 1-3.

Table 1-2 Activity in Activated Metal

Radionuclide	Ci	TBq	A_2	**Fraction A_2**
C-14	1.28E+01	4.73E-01	3	1.58E-01

Fe-55	9.20E+02	3.40E+01	40	8.51E-01
Co-60	4.32E+03	1.60E+02	0.4	4.00E+02
Ni-59	5.10E+01	1.89E+00	Unlimited	0.00E+00
Ni-63	4.81E+03	1.78E+02	30	5.94E+00
Nb-94	5.60E-01	2.07E-02	0.7	2.96E-02
Total	1.01E+04	3.74E+02		4.07E+02

Table 1-3 Activity in Surface Contamination

Radionuclide	Ci	TBq	A_2	**Fraction A_2**
H-3	8.35E-06	3.09E-07	40	7.72E-09
C-14	1.82E-05	6.73E-07	3	2.24E-07
Fe-55	1.40E-02	5.17E-04	40	1.29E-05
Co-57	8.56E-08	3.17E-09	10	3.17E-10
Co-60	7.61E-02	2.82E-03	0.4	7.04E-03
Ni-59	6.87E-04	2.54E-05	Unlimited	0.00E+00
Ni-63	7.05E-02	2.61E-03	30	8.70E-05
Sr-90	7.29E-05	2.70E-06	0.3	8.99E-06
Cs-137	2.96E-04	1.10E-05	0.6	1.83E-05
U-233/234	1.09E-04	4.03E-06	0.006	6.71E-04
U-235	2.49E-06	9.21E-08	Unlimited	0.00E+00
U-238	1.93E-05	7.15E-07	Unlimited	0.00E+00
Pu-238	2.62E-02	9.71E-04	0.001	9.71E-01
Pu-239/240	3.29E-02	1.22E-03	0.001	1.22E+00
Pu-241	1.38E+00	5.12E-02	0.06	8.53E-01
Cm-242	2.63E-14	9.74E-16	0.01	9.74E-14
Cm-243	5.82E-03	2.15E-04	0.001	2.15E-01
Pu-242	2.67E-05	9.86E-07	0.001	9.86E-04
Am-241	4.16E-03	1.54E-04	0.001	1.54E-01
Am-243	1.85E-04	6.83E-06	0.001	6.83E-03
Total	1.61E+00	5.97E-02		3.43E+00

Of the total activity in the vessel only 1.61 Ci, with an A_2 value of 3.43, is potentially dispersible. The rest of the activity is in the activated metal components. The total quantity of fissile material is 1.7 g, which qualifies as "fissile exempt" material. The total decay heat is less than 70 watts.

Regulatory Authority

The packaging and transportation of large, retired reactor components poses unique challenges from a technical as well as regulatory compliance standpoint. The current regulatory framework, domestically in Titles 49 and 10 and internationally in TS-R-1, does not lend itself to the transport of these large radioactively contaminated components, such as reactor vessels, steam generators, reactor pressure vessel heads, and pressurizers, without application for a special permit or arrangement.

Internationally, TR-R-1 Paragraph 312, Special Arrangement, was promulgated to provide a regulatory approval mechanism for consignments where it is impractical to conform to the standard provisions. Competent Authority approval can be granted under these situations when the required standards of safety can be demonstrated through alternative means. The overall level of safety in transport must be at least equivalent to the standard in order to justify a deviation from the standard. Similar provisions exist in the domestic regulations via a dual approval process.

Domestically, a memorandum of understanding (MOU) was signed on June 8, 1979 that delineated the roles and responsibilities of the Department of Transportation (DOT) and the Nuclear Regulatory Commission (NRC) in the regulation of the transportation of radioactive materials.

Generally, the DOT is responsible for regulating safety in transportation of all hazardous materials, including radioactive materials as defined in 49 CFR 173.403, and the NRC is responsible for regulating safety in receipt, possession, use, and transfer of byproducts, source, and special nuclear materials. The NRC reviews and approves or denies approval of package designs for fissile materials and for other radioactive materials (other than low specific activity materials) in quantities exceeding Type A limits, as defined in 10 CFR Part 71.

In the case of the LACBWR RPV, the NRC had approval authority for the package design, since the total activity exceeded the Type A limit under the Special Package Authorization provisions of 10 CFR 71.41(d), yet the package did not meet the Type B package performance criteria. The DOT had approval authority for the use of the non-conforming package in commerce under the provisions of 49 CFR 107.105.

NRC Special Package Authorization

Although a Special Package Authorization was requested per 10 CFR 71.41(d), a Safety Analysis Report (SAR) was prepared in accordance with Regulatory Guide 7.9, "Standard Format and Content of Part 71 Applications for Approval of Packages for Radioactive Material." Based on the contents form and amount of radioactivity (normal form, radioactive contents between $300A_2$ and $3000A_2$ and not greater than 30,000 Ci), the LACBWR RPV package is categorized as Type-B, Category II package (Ref. 5). Based on the recommendations of Ref. 6 the fabrication, examination, and inspection of the containment boundary components of a Type II package should be per ASME B&PV Code Section III, Subsection ND. Therefore, the design of the containment boundary is also based on the ASME Code requirements as much as practicable.

Figure 2 – RPV Packaging

In addition to the standard quality assurance (QA) 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 0°F. The ambient temperature will be monitored throughout the transport process and transportation will be stopped should the temperature fall to 0°F.
(4) The package authorized by this approval must be transported by motor vehicle or 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. 1 ; C-068-163041-003, Rev. 0; C-068- 163041 -004, Rev. 0. The 1 314" supplemental shielding plates identified as optional on Drawing C-068-163041-002, Rev. 1, may be omitted or may be replaced with shield plates less than 1 314" thick, provided that the plates are of the same configuration, material of construction, and type of attachment as those specified.
(6) The package is a one-time only, exclusive use shipment.
(7) The package must be marked with the following:
Model Number - LACBWR RPV Package;
Package Identification Number - 71 -9322;
Gross weight; and Serial number.

DOT Special Transport Permit
DOT's special permit, per the provisions of 49 CFR 107.105, authorized the one-way, one-time transportation in commerce of one RPV in alternative packaging, containing Class 7 radioactive material, to be classified as radioactive material. The RPV will be transported under a special arrangement, and transported in a non-specification package from the LACBWR site in Genoa, WI to the Barnwell Disposal Facility in Barnwell, SC. The special permit provided no relief from the any other Hazardous Materials Regulations (HMR).

The DOT regulations effected by this special permit is 49 CFR 173.416(a) insofar as the reactor pressure vessel may be transported under a special package authorization granted by the U.S. Nuclear Regulatory Commission in accordance with 10 CFR 71.41(d).

Performance Analysis
The package was designed and analyzed to satisfy the requirements of 10CFR71.71 under the normal conditions of transport (NCT), compliance with the "General Standards for All Packages" specified in 10 CFR 71.43, and the "Lifting and Tie-Down Standards" specified in 10 CFR 71.45. Analysis for satisfaction with the hypothetical accident conditions (HAC) was also performed to determine if compliance was achievable and found, with only some leniency, to also satisfy the requirements. Since satisfaction with the HAC criteria could not strictly be achieved, a special package authorization request was necessary and justifiable.

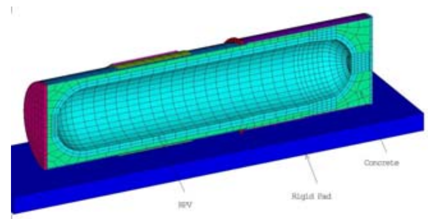

Figure 3 – Finite Element Modeling

The demonstration of compliance with the regulatory requirements of the free drop test is accomplished by analytical evaluation as permitted by the regulations (10 CFR 71.41). Finite element analysis methods, using the ANSYS/LS-DYNA (Ref. 7) explicit dynamics computer code, have been employed to simulate the regulatory drop tests. Inelastic behavior of the package components – RPV, concrete, and the canister material is incorporated into the models. Under each drop condition, the finite element model of the package is dropped freely from the specified height on a rigid unyielding surface. The models are analyzed over a sufficiently large time period so that the kinetic energy of the package has been transformed into the internal energy and/or external work. The state of stresses and strain in the canister is observed throughout this period. The failure of the containment material is assumed to occur when the maximum tensile strain reaches the maximum specified elongation at the ultimate tensile strength of that material.

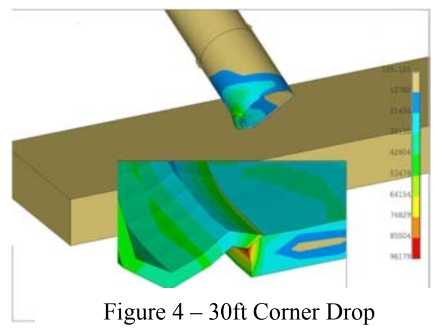
Figure 4 – 30ft Corner Drop

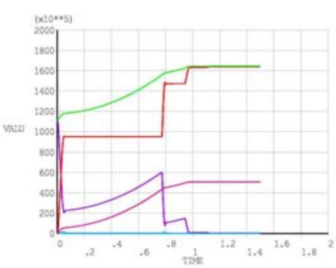

Figure 5 – Time-History Plot 30ft Drop

Transportation
Multi-model transportation from LACBWR site in Genoa, WI to the Barnwell Disposal Facility in Barnwell, SC was achieved via rail and road. The distance by rail was approximately 1,200 miles and was interlined between two different rail companies. The RPV was transferred from rail to road at the closest, suitable rail siding to its disposal destination. Road transportation was less than one (1) mile. During both modes of transportation multi-axle


Figure 6 – Rail Transport Configuration

conveyances were necessary to spread the weight of the load to comply with axle weight regulatory limitations.

Figure 7 – Road Transport Configuration

Conclusion

Although the packaging and transportation of large, retired reactor components poses unique challenges, the technical and regulatory hurdles can be overcome in order to effectively conduct the projects without risk to the public health and safety. In general, early packaging, transport and disposal of large components:

(1) Has significant economic advantages over postponing until decommissioning;
(2) Foregoes the cost of a temporary storage facility;
(3) Guaranteed disposal availability and a known price;
(4) Results in a reduction in the overall cost to decommission the facility and ensure that more funds will be available to decommission the reactor at the time the reactor ceases operation;
(5) Eliminates a radioactive source term from the site;
(6) Site workers will be exposed to less radiation (ALARA); and
(7) Eliminates an unnecessary regulatory burden associated with storing and maintaining the large components on site

Specific to DPC, early packaging, transportation, and disposal of their RPV:

(1) Afforded them the opportunity to dispose while they had access to the Barnwell Waste Management Facility, scheduled to restrict access on July 1, 2008;
(2) Accelerated their decommissioning schedule;
(3) Reduced the site radiological inventory (excluding the stored spent fuel) by 98%; and
(4) Afforded them the opportunity to maintain and utilize an experienced, but aging work force.

References

1. LACBWR RPV Activated Materials Report, DPC.0101.02.01, Jan. 2003
2. Low-Level Radioactive Waste Classification, and Assessment: Waste Streams and Neutron Activated Metals, NUREG/CR-6567, August 2000
3. Response to NRC re: Accounting at Reactors and Wet Spent Fuel Storage Facilities, LAC-13866, March 2005
4. Summary Report – Research of Material Displaced from LACBWR Spent Fuel Assemblies, LAC-TR-141, Feb. 2003
5. U.S. NRC Regulatory Guide 7.11, 1991.
6. *Fabrication Criteria for Shipping Containers*, NUREG/CR-3854, 1985.
7. ANSYS Release 9.0 (including the LS-DYNA module), ANSYS Inc., Canonsburg, PA.