UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION REPORT
Pathfinder Reactor Vessel
Certificate of Compliance No. 9238
Revision No. 0

SUMMARY

By application dated October 26, 1989, as supplemented, Northern States Power Company requested approval of the Pathfinder Reactor Vessel as a transportation package for low specific activity radioactive materials exceeding a Type A quantity. Based upon the statements and representations contained in the application and the conditions listed below, we have concluded that the Pathfinder Reactor Vessel package meets the requirements of 10 CFR Part 71.

REFERENCES

Northern States Power Company application dated October 26, 1989.
Supplements dated May 30, June 5, and August 20, 1990.

DESCRIPTION

A carbon steel reactor vessel, containing irradiated metal components meeting the requirements of low specific activity radioactive material. The vessel has approximate dimensions of 138 inches in diameter and 408.5 inches in length. The reactor vessel wall is ASTM A212 Grade B carbon steel approximately 3 inches thick. The reactor vessel head is bolted on the vessel with 48, 3-inch diameter studs. Steel cover plates are welded on the reactor vessel nozzles. The reactor vessel contains irradiated steel reactor internals and pea gravel, and is filled with solidified grout. A 1-3/4-inch thick steel radiation shield is welded to the outer surface of the vessel at the core region. An aluminum honeycomb impact limiter surrounds the vessel and the radiation shield. A 1/4-inch thick steel skin covers the aluminum honeycomb material. The overall dimensions of the package, including impact limiter, are approximately 158 inches in diameter and 408.5 inches in length. The weight of the package is approximately 580,000 pounds.

DRAWINGS

The package is constructed and assembled in accordance with Northern States Power Company Drawing Nos. 15377-050-S6005, Rev. B, 15377-050-S6001, Rev. 1, and 15377-050-S6002 Rev. 2, and Figures 1.1 and 1.2 of Northern States Power Company application dated October 26, 1989.

CONTENTS

A. Type and Form of Material

Irradiated steel reactor components meeting the requirements of low specific activity radioactive material.

B. Maximum Quantity of Material per Package

Greater than Type A quantity of byproduct material contained in irradiated reactor components and associated corrosion product layers.

STRUCTURAL EVALUATION

A. General Standards for All Packages (10 CFR §71.43)

Minimum Package Size

The overall length of the package is 408.5 inches and the overall diameter is 158.0 inches. The package meets the minimum size requirements specified in 10 CFR §71.43(a).

Tamper Proof Feature and Positive Closure

The containment boundary consists of the reactor vessel, the reactor vessel head, which is bolted to the vessel by 48 head closure studs, and closure plates welded on all vessel nozzles. The vessel is covered with an aluminum honeycomb impact limiter. A 1/4-inch thick steel skin surrounds the impact limiter and is welded at all joints. The package could not be unintentionally opened, and would show evidence of opening by unauthorized personnel.

Chemical and Galvanic Reactions

The vessel is fabricated from 3-inch thick, ASTM A212 Grade B carbon steel, with integrally bonded 304 L stainless steel cladding. The contents are stainless steel reactor internal components, pea gravel and solidified grout. There will be no significant galvanic or chemical reactions between the components of the packaging or the packaging and the contents.

B. Lifting and Tie-Down Standards for All Packages (10 CFR §71.45)

Lifting Devices and Excessive Load Failure

Temporary lifting points are provided by lugs, which are attached to six modified high strength head studs inserted in existing head stud holes. The applicant has shown by analysis that the lifting lugs and bolts meet the requirements of 10 CFR §71.45(a). Prior to installing the impact limiter, the lugs will be burned off so they cannot be inadvertently used for package tie-down.

Tie-Down Devices

There are no tie-down devices which are a structural part of the package. The package is positioned in a saddle during transport, and two bands tie the package down to the saddle. Neither the bands nor the saddle are a structural part of the package.

C. Normal Conditions of Transport (10 CFR §71.71)

Heat

The package internal heat source is negligible (less than 5 watts decay heat distributed in the 580,000 pounds of the package).

Cold

The reactor vessel is made from ASTM A212 Grade B carbon steel. The applicant performed a brittle fracture analysis to demonstrate that, for an ambient temperature of -40 ^{o}F, the reactor vessel meets the fracture toughness criteria specified in NUREG/CR-1815, "Recommendations for Protecting Against Failure by Brittle Fracture in Ferritic Steel Shipping Container Up to Four Inches Thick."

To evaluate thermal stresses, the applicant calculated stresses which would result if the package were instantaneously cooled from a uniform temperature of 100 °F to -40 °F. For this conservative case, the thermal stresses were calculated as 23.9 ksi. This is well within acceptable limits for thermal stresses.

Reduced External Pressure and Increased External Pressure

The reactor vessel was assessed for a reduced external pressure of 3.5 psia and an increased external pressure of 20 psia. The stresses produced by these conditions were low and would have no effect on the vessel.

Vibration

The effects of shock and vibration loads on the vessel were shown to be negligible.

Water Spray

The water spray conditions will not have a significant effect on the package.

One Foot Free Drop

Because of the large size, weight, and bulk of the package, the package will always be oriented in the horizontal position during transport. Therefore, two horizontal drop orientations were evaluated by the applicant: (1) side drop and, (2) oblique drop at an angle of 4.2°. For the second drop orientation, it was assumed that 50 percent of the package weight was transmitted to the impact edge. The second drop orientation was considered more severe, and it was used to analyze the impact limiter and stresses in the vessel section. The g-load determined for one foot oblique drop was 2.35 g. The maximum stresses in the vessel wall were calculated by combining impact loads with other loads as specified in Regulatory Guide 7.8. The maximum membrane stress intensity was 5,900 psi, the maximum membrane plus bending stress was 22,830 psi. These values are less than the allowable stresses specified in Regulatory Guide 7.6, which are 17,880 psi for membrane stress intensity, and 26,800 psi for membrane plus bending stress intensity. The maximum tensile stress in the closure head bolts was 510 psi, and the maximum shear stress in the bolts was 430 psi. These stress levels are considerably below the yield stress of the bolts.

Penetration

The applicant demonstrated that a 13 pound steel cylinder dropped through a distance of 40 inches onto the 3-inch thick vessel wall or onto the 3/4-inch thick nozzle closure plates would have no significant effect on the integrity of the containment system.

D. Hypothetical Accident Conditions

Since the Pathfinder Reactor Vessel contains only low specific activity radioactive material and will be transported as exclusive use, the package need not be evaluated for hypothetical accident conditions.

THERMAL EVALUATION

The decay heat present in the package is less than 5 watts distributed throughout the 580,000 pound package. This heat load is insignificant.

CONTAINMENT EVALUATION

The containment boundary for the package consists of the reactor vessel, the reactor vessel head, and the nozzle cover plates. The radioactive materials are primarily present as activation products in the reactor vessel and the reactor internal components. The activation products are fixed in the matrix of the component material and are not available for release from the package. The reactor internals are large components which are structurally fixed in the vessel, as they were when the reactor was operating. These components can only be removed from the vessel by way of the reactor vessel head. In addition to the activation products, a small amount of radioactivity is present in the corrosion product layer which adheres to the surfaces of the internal components. To ensure that even this small amount of activity is immobilized in the package, the vessel will be completely filled with grout prior to shipment. The grout will fill void spaces in the vessel and will solidify into a monolithic solid. The grout will provide additional radiation shielding, as well as assuring that the radioactivity present in component corrosion layers is fixed in the package.

The structural analysis of the package showed that the package would retain the radioactive contents during normal conditions of transport. The vessel head is bolted on the vessel by 48, 3-inch diameter head studs. Calculations demonstrated that the bolts would not yield and that the reactor vessel would not rupture during normal conditions of transport.

EVALUATION OF RADIOACTIVE CONTENTS AND SHIELDING

A. Activation Analysis

The applicant performed a neutron activation analysis, using the ORIGEN2 point activation code, to determine the radioactivity present in the reactor vessel and internal components, and to show that the contents meet the requirements for low specific activity radioactive materials.

In the activation model, the reactor vessel and internals were divided into four concentric cylinders representing: (1) the core region, (2) the steam separator region, (3) the vessel cladding region, and (4) the vessel wall region. Each cylindrical region was then divided into seven levels: the core level and three 30-cm thick sections above and below the core. Materials more than 90 cm above and below the core were not included in the model, since activation in these materials would be insignificant.

Neutron fluxes, based on the plant's operating records, were assigned to the core level of each cylindrical region. The analysis used a factor of ten reduction in neutron flux for each 30-cm section above and below the core. This is considered reasonable and is consistent with the radial flux reduction which is based on operating records. The maximum total neutron flux for specific components is shown below.

Radial Location	Total Flux (n/cm^2-s)
Core Boundary	3.50E+13
Boiler Baffle	1.16E+13
Steam Separator	1.07E+12
Reactor Vessel Inner Wall	3.30E+10
Reactor Vessel Outer Wall	2.50E+09

The ORIGEN2 BWR-U cross-section library was used for the materials in the cylindrical region of the reactor core (all levels). This library contains one-group energy spectrum averaged cross-sections. The cross sections are based on a typical boiling water reactor neutron energy spectrum. The ORIGEN2 thermal cross-section library was used for materials outs de the core region. This library contains thermal cross-section data fcr room temperature (68 ^{o}F). A temperature correction was used to account for the higher reactor operating temperatures.

For the purpose of the activation analysis, the applicant conservatively assumed the reactor operated for 87.6 days at full power, just prior to being shut down on September 16, 1967. This assumption corresponds to 17,485 MWD core burnup, which exceeds the 16,635 MWD burnup indicated by plant records. The ORIGEN2 results were then decayed to June 1, 1990.

Independent confirmatory analyses were performed by the NRC staff. The staff used the ORIGEN-S point activation code, which is the ORNL SCALE version of ORIGEN. The neutron flux distribution and activation model provided by the applicant was used in the confirmatory analysis. The staff assumed 90 full power days of operation and a decay period of 23 years. The staff used the BWR spectrum within the reactor core region only. A totally thermal flux was assumed for all other regions, including levels directly above and below the core.

B. Activation Analysis Results

The total radioactivity and the specific radioactivity in the package were determined using the results of the activation analysis. The applicant estimated the total package activity to be 562 curies. The NRC staff estimated the total package activity to be 1,232 curies. The difference in these values is primarily due to the different assumptions used for neutron spectrum for regions directly above and below the core. NRC staff used a more conservative assumption of a totally thermal flux.

The tables below summarize the package nuclide inventory and activities per region.

PATHFINDER PACKAGE RADIONUCLIDE INVENTORY

Nuclide	Applicant Activity (Ci)	% Total	NRC Staff Activity (Ci)	%Total
H-3	0.74	0.13	1.52	0.12
C-14	0.31	0.05	0.66	0.05
Fe-55	44.52	7.92	91.80	7.45
Co-60	299.95	53.35	753.00	61.12
Ni-59	1.78	0.32	3.68	0.30
Ni-63	213.44	37.96	380.00	30.84
Others	< 1.60	< 0.30	< 1.50	< 0.20
TOTAL	562		1232	

ESTIMATED ACTIVITY PER CYLINDRICAL REGION (CURIES)

Cylindrical Region	Applicant	NRC
Reactor Core	387.8	1000.1
Steam Separator	172.9	220.2
Vessel Cladding	1.1	1.5*
Vessel Wall	0.7	3.2*
TOTAL PACKAGE ACTIVITY	562.2	1232.0

*Inner vessel wall neutron flux was used.

The applicant estimated 95 mCi of surface contamination within the Pathfinder reactor vessel, based on samples from the feedwater system. The surface contamination activity is negligible, relative to the activation products. The surface contamination is fixed in place by the solidified grout.

C. Radioactivity Concentration

The applicant calculated the specific activity of radioactive materials in each component to show that the contents meet the concentration limits for low specific activity material. The results of the applicant's analyses and confirmatory analyses performed by the staff are summarized below.

MAXIMUM SPECIFIC RADIOACTIVITIES FOR REACTOR COMPONENTS

Region	Material	Specific Activity (mCi/g) (Applicant)	(NRC)
Reactor Core	304 SS	2.01E-1	1.94E-1
Reactor Core	Zircaloy-2	1.74E-3	1.52E-3
Boiler Shroud	304 SS	3.80E-1	3.86E-1
Steam Separator	304 SS	5.66E-3	3.56E-2
Vessel Cladding	304 SS	1.09E-3	1.47E-3*
Vessel Wall	A212 CS	7.54E-5	2.62E-4*

*Inner vessel wall neutron flux was used.

The maximum specific activity calculated in any region was less than 0.3 mCi/g for all components except the boiler shroud. The maximum specific activity for the core level region of the boiler shroud was calculated to be 0.380 mCi/g by the applicant and 0.386 by the staff. These concentrations are conservative estimates since the maximum flux at the core midplane was used for the whole core region, and the flux was assumed to be totally thermal. When averaged over the activated portions of the boiler shroud, the specific activity was 0.248 mCi/g in the applicant's analysis and 0.252 in the staff's analysis. When activity in the core level region of the boiler shroud was averaged over the other stainless steel components in the core region, the specific activity was calculated by the staff to be 0.24 mCi/g. The stainless steel components in the core region have a specific activity of 0.194 mCi/g. This range of specific activities show that the activity is essentially uniformly distributed in the activated materials in and near the core region. The estimated average radioactivity concentrations meet the requirements for low specific activity material, as defined in 10 CFR §71.4.

D. Shielding Evaluation

The results of the activation analysis were used to determine external radiation dose rates for the package. The components of the Pathfinder Reactor Vessel package included in the shielding analysis model were the 66 cubic yards of gravel within the vessel, the vessel internals, the vessel wall and the two inches of steel surrounding the outer vessel wall. No shielding credit was taken for the solidified grout. The sources of gamma radiation were the activation products present in the reactor vessel and its internals.

In the shielding analysis model, the vessel and internals were represented by four concentric homogenized cylinders of the component materials and the gravel. Concrete at 50 percent of full density was used to represent the gravel. The source term for each cylinder was taken from the activation analysis. The applicant used MICROSHIELD, a point kernel computer code, to calculate the dose rates at the package surface and at 2 meters from the package surface. Results of the applicant's shielding analysis show that the external dose rates were within the limits of 10 CFR §71.47.

The NRC staff performed independent shielding analyses to confirm the applicant's results, using a model similar to the applicant's. The source terms were derived from the staff's activation analysis. The staff used QADS, a multidimensional point kernel computer code in the SCALE system, to calculate the radial dose rates (sides of the package). MICROSHIELD was used to estimate the maximum axial dose rates (top and bottom of the package).

The table below summarizes the results of the applicant's and staff's shielding analyses.

EXTERNAL DOSE RATES (MREM/HOUR)

Location	Applicant	NRC
Package Surface		
Side	28	16.4
Top/Bottom	--	20
2 Meters from the Surface		
Side	8.4	9.1
Top/Bottom	3.2	6

The region which contributes the most to the external radiation levels is the steam separator region, since it is less shielded than the core region. Therefore, even though the staff analysis showed a much larger radioactive inventory for the reactor core region, the calculated dose rates were similar since the calculated inventories for the steam separator regions were similar. The analyses show that the external radiation dose rates are within the limits specified in 10 CFR Part 71.

CRITICALITY EVALUATION

All nuclear fuel has been removed from the reactor vessel, and there are no other fissile materials present in the package. Therefore nuclear criticality is not a concern.

FABRICATION EVALUATION

The reactor vessel and head are fabricated from 3-inch thick carbon steel ASTM A212 Grade B plate. Closure plates, fabricated from steel plate, are welded to the reactor vessel nozzles. The types, sizes and locations of the closure welds are shown on Northern States Power Company Drawing 15377-050-S6005.

The recommendations of NUREG/CR-3019 will be followed except that fillet welds will be used on the cover plates instead of full penetration welds, and the closure work will be performed using Northern States Power Company procedures which have been qualified in accordance with ASME Code Section IX, rather than under an ASME Certificate of Authority. Since the radioactive contents are not dispersible and would not be released even if the cover plates fail, these exceptions are acceptable.

The welders performing closure work will be qualified per ASME Code Section IX. The closure welds will be 100 percent visually examined. In addition, a minimum of 10 percent of the welds will be examined by magnetic particle or liquid penetrant techniques. The examination procedures and acceptance standards will be consistent with requirements of ASME Code Section V and VIII.

OPERATING PROCEDURES, ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

The Pathfinder Reactor Vessel will be prepared for a single shipment. Chapter 8 of the application addresses the preparation of the package. The preparations will include: visual inspection of the nozzle cover plate weld integrity, shield weld integrity, head closure bolt condition, impact limiter condition and installation; radiation surveys to show that the package meets the requirements for external radiation dose rates and surface contamination; package identification as described in 10 CFR §71.85(c).

CONDITIONS

1. The package authorized by this certificate must be transported on a motor vehicle, railroad car, inland water craft, or hold of a deck of a seagoing vessel assigned for sole use of the licensee.

2. The lifting points must be rendered inoperable for tie-down of the package.

3. In addition to the requirements of Subpart G of 10 CFR Part 71, the package must be prepared for shipment and operated in accordance with Chapter 8 of the application.

CONCLUSIONS

Based on the review of the statements and representations contained in the application and the conditions listed above, we have concluded that the Pathfinder Reactor Vessel package meets the requirements of 10 CFR Part 71.

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Date OCT 1 1 1990