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June 29, 1981

In reply, please
refer to LAC-7634

DOCKET NO. 50-409

U. S. Nuclear Regulatory Commission
ATTN: Mr. Darrell G. Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulation
Division of Operating Reactors
Washington, D. C. 20555

SUBJECT: DAIRYLAND POWER COOPERATIVE
LA CROSSE BOILING WATER REACTOR (LACBWR)
PROVISIONAL OPERATING LICENSE NO. DPR-45
SEP TOPIC III-7.D
CONTAINMENT STRUCTURAL INTEGRITY TESTS

REFERENCE: (1) DPC Letter, LAC-7387, Linder to Eisenhut, dated February 27, 1981.

Gentlemen:

Enclosed find Safety Evaluation Report (SER) for Containment Structural Integrity Tests (SEP-III-7.D) which we have prepared for the La Crosse Boiling Water Reactor.

Our letter, Reference 1, identified topics for DPC to submit for NRC evaluation. The subject topics were listed in the schedule submitted with Reference 1.

If there are any questions regarding this letter, please contact us.

Very truly yours,

DAIRYLAND POWER COOPERATIVE

Frank Linder/ee

Frank Linder, General Manager

FL:HT:ee

cc: J. G. Keppler, Reg. Dir., NRC-DRO III
NRC Resident Inspectors

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LA CROSSE BOILING WATER REACTOR

SYSTEMATIC EVALUATION PROGRAM
SAFETY EVALUATION REPORT

TOPIC III.7.D

CONTAINMENT STRUCTURAL INTEGRITY TESTS

Introduction

In order to assure that a steel containment structure will respond satisfactorily to the postulated design pressure loads, a program of measurements, under the Containment Structural Integrity Test, is required to demonstrate the adequacy of the structure with respect to the quality of construction and material. The scope of this safety topic evaluation is to review the adequacy of the structural integrity testing procedures used by the licensee and, using current review criteria as a basis, to evaluate the measurements taken during the testing.

Current Review Criteria

The current review criteria for this specific safety topic are:

1. Standard Review Plan, Section 3.8.2;
2. Regulatory Guide; 1.57
3. ASME BPV Code Section III, Subsection NE-6000

Evaluation

Description of Containment Structure

The containment building is a right circular cylinder with a hemispherical dome and semi-ellipsoidal bottom. It has an overall internal height of 144 feet and an inside diameter of 60 feet, and it extends 26 feet 6 inches below grade level. The steel shell thickness is 1.16 inches, except for the upper hemispherical dome, which is 0.60 inch thick.

The containment building is designed to withstand the instantaneous release of all the energy of the primary system to the containment atmosphere at an initial temperature of $80^{o}F$., neglecting the heat losses from the building and heat absorption by internal structures.

The interior of the shell is lined with a 9 inch thick layer of concrete, to an elevation of 727 feet 10 inches, to limit direct radiation doses in the event of fission-product release within the containment building.

The containment building is supported on a foundation consisting of concrete-steel piles and a pile capping of concrete approximately 3 ft thick. This support runs from the bottom of the semi-ellipsoidal head at about el 612 ft 4 in. to an elevation of 621 ft 6 inches. The 232 piles that support the containment structure are driven deep enough to support over 50 tons per pile. (Reference 1)

The containment bottom head above el 621 ft 6 in. and the shell cylinder from the bottom head to approximately 9 in. above grade elevation (639 ft 9 in.) are enveloped by reinforced concrete laid over a ½ in. thickness of premolded expansion joint filler. The reinforced concrete consists of a lower ring, mating with the pile capping concrete. The ring is approximately 4½ ft thick at its bottom and 2½ ft thick at a point 1½ ft below its top (owing to inner surface concavity). The ring then tapers externally to a thickness of 9 in. at the top (el 627 ft 6 in.), and the 9 in. thickness of concrete extends up the wall of the shell cylinder to el 639 ft 9 in. The filler and concrete are not used, however, where cavities containing piping and process equipment are immediately adjacent to the shell.

Except for areas of the shell adjacent to other enclosures, the exterior surface of the shell above el 639 ft 9 in. is covered with 1½ in. thick siliceous filter insulation, faced with aluminum. The insulation of the dome is Johns-Manville Spintex of 9 $lb/ft.^3$ density, faced with embossed aluminum sheet approximately 0.032 in. thick. The insulation of the vertical walls is Johns-Manville Spintex of 6 $lb/ft.^3$ density, faced with corrugated embossed aluminum sheet approximately 0.016 in. thick. The insulation minimizes heat losses from the building and maintains the required metal temperature during cold weather, and reduces the summer air-conditioning load.

The shell includes two airlocks. The principal access to the shell will be through the personnel airlock that connects the containment building to the turbine building. The airlock is 21 ft 6 in. long between its two doors, which are 5 ft 6 in. by 7 ft and are large enough to permit passage of a spent fuel element shipping cask. The containment building can also be evacuated, if necessary, through the emergency airlock, which is 7 ft long and 5 ft in diameter, with two circular doors of 32½ in. diameter (with a 30-in. opening). Both airlocks are at el 642 ft 9 in. and lead to platform structures from which descent to grade level can be made. When the doors are closed, a clamp exerts a positive force, which is transmitted through the doors to live-rubber gaskets around the door frames to ensure gas tightness. The airlocks and doors are designed to remain gastight under a pressure of 52 psig from inside the containment shell and 1 psig from outside of the shell. The airlock doors are manually operated.

An 8 ft-by-10 ft freight door opening in the containment building accomodates large pieces of equipment. It will be used only when the reactor is shut down and only if large pieces of equipment must be removed. During operation, 9-in.-thick concrete blocks are placed on the outside of the door for shielding. The door is bolted internally to the door frame in the shell. Two rubber gaskets in parallel between the door and door frame ensure a pressure-tight seal.

Approximately 300 MI cables and 75 bulkhead conductors penetrate the containment shell.

Two 30-ton air-conditioning units keep the containment building air temperature at or below 80°F. The units also have steam heating coils to provide heat to the building, in case a prolonged shutdown is necessary during the winter, and provide air circulation throughout the building.

Air is exhausted from the building by means of an exhaust blower, which discharges approximately 5000 cfm through a pair of isolation dampers and through the containment shell penetration into the suction plenum side of the stack blowers. The exhaust air is pulled through two filters located just upstream of the exhaust blower. The first filter is a coarse filter to remove large particles, and the last is a high-efficiency type filter. An automatic bypass permits the exhaust blower to continue to circulate air whenever the containment isolation dampers are closed.

Ventilation air is admitted into the building through the containment penetration and a pair of isolation dampers, and then through duct work to the suction side of the blower of each air-conditioning unit.

Excessive external pressure on the building is prevented by two vacuum breakers which start to open when the negative pressure within the building exceeds 0.2 psig.

A 42,000-gal storage tank in the dome of the containment building supplies water for the emergency core spray system and the building spray system.

A 50-ton traveling bridge crane with a 5-ton auxiliary hoist is located in the upper part of the containment building. The bridge completely spans the building and travels on circular tracks supported by a ring of concrete around the inside of the building just below the hemispherical upper head. The crane is operated from a pushbutton station suspended from the trolley.

DESIGN DATA AND STRUCTURAL DESIGN

General

The design and construction of the containment vessel conforms to the applicable requirements of the 1962 edition of the ASME Boiler and Pressure Vessel Code, Section VIII, Unfired Vessels, and applicable code cases 1270N, 1271N and 1272N. The containment vessel has the ASME code stamp.

Design Conditions for Code Calculations

The design conditions for the code calculations are as follows:

(1) maximum internal pressure . . . . 52 psig

(2) maximum negative pressure . . . . 0.5 psig

(3) maximum temperature . . . . 280°F

(4) minimum temperature . . . . -20°F

(5) welded joint efficiency 100%

(6) Basic wind pressure . . . . 20 lb/ft^2

Materials

Forgings conform to ASTM A350, LF1, and piping conforms to ASTM A333. Containment vessel plates and reinforcements and their welded attachments, conform to the applicable requirements of ASTM A201 ("Standard Specification for Carbon-Silicon Steel Plates of Intermediate Tensile Ranges for Fusion-Welded Boilers and Other Pressure Vessels") for Grade B steel. Plates also meet the applicable test requirements for ASTM A300 ("Standard Specifications for Steel Plates for Pressure Vessels for Service at Low Temperatures").

Drop weight tests on the 31 heats of ASTM A Grade B materials used in the containment vessel indicate a nil ductility temperature range of -30°F to -60°F. Thermocouples are provided on the vessel external surface at locations given in Table 6-1 in order to ensure that the shell temperature is always above 0°F (NDT + 30).

Construction

The containment vessel is airtight, with a maximum leakage of less than 0.1 percent of the contained volume per 24 hr at design pressure (52 psig). The out-of-roundness does not exceed 0.5 percent.

Radiography

All butt welds were fully radiographed in accordance with Paragraph UW-51, ASME Boiler and Pressure Vessel Code. All welds of doors, nozzles, and opening frames, and all welds that could not be radiographed, were examined for cracks by the magnetic-particle or fluid-penetrant methods of inspection.

Testing

Requirements for the initial testing of the containment vessel included the following:

(1) visual inspection of welding

(2) pressurizing the containment vessel to 5.0 psig and performing soap bubble tests on all welds

(3) pressurizing the shell to 59.80 psig, then reducing the pressure psig for final soap bubble tests on all welds at a vessel pressure of 52 psig. The 52 psig pressure was maintained for a 3-day period, with leakage measurements taken hourly. The inner chamber method was used to measure leakage.

After completion of the vessel penetrations, a final leak-rate test was performed at 52 psig for two days (preoperational leak-rate test).

Vacuum Breakers

Vacuum breakers are set to start opening at a 0.2 psig-vacuum and to be fully open at a 0.5-psig vacuum. They will remain tight at the 52 psig design pressure without damage to the seat, gasket, diaphragm, or casing. Each vacuum breaker is mounted inside a test chamber, which permits periodic testing of the vacuum breaker for leak-tightness. During normal operation, the upper half of the test chamber is jacked up from the lower half to permit air to flow into the vessel whenever the negative pressure exceeds 0.2 psig.

Contained Energy

The containment building can contain all the steam and water released from the primary system in the event of a major system rupture.

The free-air volume of the containment building is 264,160 ft^3. If all the energy of the primary system is assumed to be instantaneously released to the containment atmosphere at an initial temperature of 80^oF, and if there are no heat losses from the building or heat absorption by internal structures, the resulting pressure and temperature would be 48.5 psig and 273^oF. These values compare favorably to the design values of 52 psig and 280^oF.

LEAK TEST PROCEDURES AND ASSESSMENT OF RESULTS

The overload test procedure fulfilled the requirements of the ASME Code, Section VIII, as modified by Code Case 1272 N.

The method used for the leakage test consisted basically of comparing the pressure in the containment vessel with that in an airtight inner chamber. The inner chamber was proved tight by thorough preliminary inspection methods, and any relative decrease in containment vessel pressure after temperature equilibrium was regarded as from external leakage. A high degree of sensitivity to this pressure differential was achieved by use of a water manometer to measure the pressure differential between the two air volumes.

Preliminary Check

Preliminary testing was performed in the shop and field before the overload and leakage-rate tests at the site. All airlocks were shop tested for tightness and for operation of door mechanism, including the equalizing valves, and were found to be adequate. All shop-welded manholes and nozzles were magnafluxed inside and outside after shop stress relief; there was no indication of cracks or other defects.

Overload Test

Following construction the overload test was performed by the constructor (Reference 2). The vessel was pressurized to 5 psig and all vessel connections and welds were checked by a soapsuds inspection. The air pressure was then increased in increments to the test pressure of 59.8 psig. After a 1-hr holding period, vessel pressure was reduced to the design 52 psig. The second and final soapsuds inspection was conducted at design pressure. The outer door of each lock was left open. The soapsuds inspection at 5 psig found several minor leaks in connections but no leaks in welded seams or in gaskets. All leaks were repaired and were then found to be tight under soapsuds inspection. The overload test to 59.8 psig on the vessel and the airlocks was successfully completed. The final soapsuds inspection at 52 psig found no leaks in welded seams.

Initial Leakage Rate Test

Following the overload test, the constructor commenced the initial leakage rate test. The pressure was adjusted so that maximum containment pressure would be approximately 52 psig during the holding period for the leakage rate test. Water was introduced into the differential manometer to approximately the mid-height of the scale. Air was then pumped into the containment vessel until th[illegible]ter manometer indicated a pressure about 10 inches higher than that i[illegible]nner chamber system.

The pressure and temperature readings were recorded hourly for four successive nights and the average leakage per 24-hr period is calculated to be:

$$\frac{\%\ \text{leakage}}{24\text{-hr}} = 1/3 \times \frac{0.11}{129.80 \times 13.6} \times 100 = 0.002\%$$

The calculated leakage is well within the specification value of 0.1 percent per 24-hr period. The measured tightness of the vessel is consistent with the results of the soapsuds inspection.

Pre-Operational Leakage Rate Test

After installation of all building penetrations, the preoperational leakage rate test was performed by Allis Chalmers (Reference 3). Individual penetrations were tested to establish their leak rate. The reactor building leak rate was again measured, using the same reference vessel method. The test was performed similarly to the initial leakage rate test. The containment building leakage rate was found to be 0.038 percent of contained volume per 24-hr period as measured over a two-day interval. This is well within the allowable limit. A detailed description of the structure integrity test for the La Crosse Boiling Water Reactor Containment Building is contained in Reference 2. Pre-operational test information is contained in Reference 3.

In-Service Leakage Tests

There have been eight in-service integrated leakage rate tests performed on the La Crosse Boiling Water Reactor Containment Building (Reference 3 through 10). The first 6 were performed using the reference vessel method. The last two were done using the mass balance method.

Based on the results from all the in-service tests it is apparent that no deterioration of the integrity of the containment vessel has occurred. The tests results also show that the integrity of the containment vessel penetrations is acceptable.

Comparison to Current Review Criteria

The criteria that was applied during the overload test on the La Crosse Boiling Water Reactor Containment Vessel conformed to all the requirements of Article NE-6000 of Reference 11.

The first two in-service leakage rate tests were conducted by the internal reference vessel method per the requirements of Reference 12. The acceptance criteria was more stringent than present day acceptance values e.g. 0.055%/day of contained volume vs. 0.07[illegible] /day. The next four in-service leakage rate tests were conducted by the internal reference vessel method per the requirements of Reference 12 with acceptance criteria established by LACBWR Technical Specifications as $\leq$ 0.08%/day of contained volume, less than the present day acceptance criteria.

The leakage rate tests of Reference 7 and 8 were conducted to the applicable requirements of Title 10 of the Code of Federal Regulations, Section 50, Appendix J. LACBWR Technical specifications had been revised to comply with the acceptance criteria of 10 CFR 50 Appendix J. The test method still used the internal reference vessel method. Leakage rates were determined inaccordance with Reference 13.

The leakage rate tests of References 9 and 10 were conducted to the applicable requirements of 10 CFR 50, Appendix J and LACBWR Technical specifications, which meet the acceptance criteria of 10 CFR 50, Appendix J. The test method used the absolute method of leakage determination, specifically, the mass plot analysis technique. Leakage rates were determined inaccordance with Reference 14.

Based on our review of the above reports, no evidence of unusual response of the containment building deterioration showed up during any of the tests. The leakage rates were always within the acceptable limit prior to reactor operation after every test.

Conclusion

Based on the information provided in the test reports and Reference 15 and the evaluation stated above, we conclude that the test procedures used are adequate and the test results provide a basis to assure that the containment structure will safely perform its intended functions and will continue to withstand the design pressure load of 52 psig.

REFERENCES

(1) Sargent & Lundy Report SL-2003, dated February 25, 1963. "Containment vessel Pile Driving Operations for 50 MWe Boiling Water Reactor at Genoa, WI."

(2) Chicago Bridge & Iron Report, "Final Leakage Rate Determination of Reactor Containment Vessel", dated May 1967.

(3) ACNP-68511 La Crosse Boiling Water Reactor Containment Leakage Rate Test, 1968.

(4) LACBWR Containment Vessel Class-A Integrated Leak Rate Test of October 1969, DPC-851-27.

(5) LACBWR Containment Vessel Class-A Integrated Leak Rate Test of November 1970, DPC-851-34.

(6) Containment Building Leak Rate Test, September 1971, DPC-851-35.

(7) 1975 Containment Building Integrated Leak Rate Test, LAC-TR-032, September 1975.

(8) 1978 Reactor Containment Building Integrated Leak Rate Test, LAC-TR-066, December 1978.

(9) 1979 Reactor Containment Building Integrated Leak Rate Test, LAC-TR-071, May, 1979.

(10) 1980 Reactor Containment Building Integrated Leak Rate Test, LAC-TR-093, December 1980.

(11) ASME Boiler and Pressure Vessel Code, Section III, Division I, Subsection NE, "Class MC Components", American Society of Mechanical Engineers.

(12) "Proposed Standards for Leakage Rate Testing of Containment Structures for Nuclear Reactors", ANS 7.60, Approved for publication for comments by ANS Standards Committee, June 15, 1964.

(13) ANSI N45.4 - 1972, "Leakage-Rate Testing of Containment Structures for Nuclear Reactors."

(14) ANS-N274, "Containment System Leakage Testing Requirements, " Revision 2, May 15, 1978.

(15) ACNP - 65544 La Crosse Boiling Water Reactor (LACBWR) Safeguards Report for Operating Authorization, Revised, August 1967.