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Docket No. 50-244

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Mr. Leon D. White, Jr.
Vice President
Electric and Steam Production
Rochester Gas & Electric Corporation
89 East Avenue
Rochester, New York 14649

Dear Mr. White:

RE: SEP TOPIC III-7.D CONTAINMENT STRUCTURAL INTEGRITY TEST

Enclosed is a copy of our evaluation of Systematic Evaluation Program Topic III-7.D Containment Structural Integrity Test. This assessment compares your facility, as described in Docket No. 50-244 with the criteria currently used by the regulatory staff for licensing new facilities. Please inform us if your as-built facility differs from the licensing basis assumed in our assessment.

We have discussed this assessment with your staff and believe the facts concerning your plant are correct. Therefore, our review of this topic is complete and this evaluation will be a basic input to the integrated safety assessment for your facility unless you identify changes needed to reflect the as-built conditions at your facility. This topic assessment may be revised in the future if your facility design is changed or if NRC criteria relating to this topic are modified before the integrated assessment is completed.

Sincerely,

Original signed by

Dennis M. Crutchfield, Chief
Operating Reactors Branch #5
Division of Licensing

Enclosure:
Completed SEP
Topic III-7.D

cc w/enclosure:
See next page

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SEP SAFETY TOPIC EVALUATION

R. E. GINNA NUCLEAR POWER STATION, RG&E

Topic III-7.D Containment Structural Integrity Tests

Introduction

In order to assure that a concrete containment structure will respond satisfactorily to the postulated design pressure loads, a program of measurements, namely the Containment Structural Integrity Test Program, is required to demonstrate the correlation with theoretically predicted responses and to prove 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 procedure 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 3.8.1;

2. Regulatory Guide 1.18;

3. ACI 359 (ASME BPV-III-2) Code Art. 6000.

Related Safety Topics and Interfaces

The containment structure integrity test of Ginna nuclear station was performed based on the original calculated design pressure of 60 psig. Within the scope of the SEP safety Topic VI-3, "Containment Pressure and Heat Removal Capability", this original design pressure will be reviewed to assure it's adequacy. Thus, the validity of this safety evaluation is contingent upon whether or not a positive conclusion can be drawn in the review of Topic VI-3. A reevaluation of this topical review will be necessary if the original calculated design pressure is increased.

Evaluation

Description of Structure

The containment structure is a vertical prestressed concrete cylinder with a reinforced concrete flat base and a hemispherical dome. A welded steel liner (3/8" in thickness for the dome and cylinder and 1/4" for the base) is attached to the inside face of the concrete containment structure. The principle dimensions include an inside diameter of 105'-0" and a height (from top of base to spring line) of 99'-0". The nominal thickness dimensions of the reinforced concrete are 3'-6" for the wall and 2'-6" for the dome. The concrete base slab is 2 ft. thick, with an additional 2 ft. lean concrete fill over the bottom liner plate. A detailed description of the structure can be found in the "Final Facility Description and Safety Analysis Report" (Ref. 2).

Test Procedure and Assessment of Test Results

A detailed description of the structural integrity test for the Ginna containment is contained in GAI Report #1720, dated October 3, 1969 (Ref. 1). A number of different types of instruments (jig transit, invar tapes, LVDT strain gages, photoelastic discs, load cells, etc.) were utilized and are described in the test report. The containment vessel was pressurized to 69 psig (115 percent of the design pressure of 60 psig) in five pressure steps (increments) and then depressurized in three steps. At the maximum test pressure level (69 psig), the pressure was maintained for approximately four hours before the readings, measurements and observations were taken. Measurements and observations were also made at the other pressure step increments. At these steps, the vessel pressure was slightly increased above the level at which the measurements were taken and then the pressure was reduced to the specified level and observations made after at least ten minutes to permit an adjustment of strains within the structure. The detailed procedures can be found in the test report.

Based on our review of this report, no unusual response of the containment structure showed up during the process of pressurization and depressurization. The displacements (vertical and radial displacements) and the rebar and liner stresses calculated from measured strains were always within allowable limits, except for one displacement which was slightly higher than predicted. The observed concrete crack widths and the recovery after depressurization were also below the acceptable limits.

Significance of Deviations from Current Review Criteria

The test procedure and the assessments of measurements described in the report were compared with the requirements stated in the current review criteria. The following deviations have been identified:

1. Cur r ent criteria requires more measuring locations for global displacement and less for local displacement.

2. A larger surface area is required by current criteria for observing the concrete crack patterns.

3. Current criteria requires the measurements of strain near the base of the cylinder and under the prestressed tenden anchor point and vertical displacements on the dome.. No such measurements were described in the report.

4. Current criteria requires that the measurements to confirm the recovery of the structure should be taken 24 hours after depressurization. As stated in the report, these measurements were taken 3 to 6 hours after depressurization with a slightly lower recovery rate than that required by current criteria.

It is the staff's judgement that the deviations identified above are not significant and will not affect the assessments made in the section of the test report entitled "Test Procedure and Assessment of Test Results", since no unusual response of the structure was found during the test.

Conclusion

Based on the information provided in the test report and the FSAR and the evaluation stated above, we conclude that the test procedure used is adequate and the test results provide a basis to assure that the containment structure will safely perform its intended functions and will withstand the design pressure load of 60 psig.

References

1. "Structural Integrity Test of Reactor Containment Structure - R. E. Ginna Nuclear Power Station", GAI Report #1720, October 3, 1969.

2. "Final Facility Description and Safety Analysis Report", R. E. Ginna Nuclear Power Station Unit No. 1.