ROCHESTER GAS AND ELECTRIC CORPORATION • 89 EAST AVENUE, ROCHESTER, N.Y. 14649

July 7, 1981

RECEIVED
JUL 14 1981
U.S. NUCLEAR REGULATORY COMMISSION

Director of Nuclear Reactor Regulation
Attention: Mr. Dennis M. Crutchfield, Chief
Operating Projects Branch #5
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: SEP Topic III-7.C, "Delamination of Prestressed Concrete Containment Structures"

Dear Mr. Crutchfield:

Rochester Gas and Electric has reviewed the NRC's draft evaluation of this SEP topic, transmitted by letter dated June 24, 1981.

We concur in the factual information presented, and agree with the NRC's conclusion that the containment at Ginna would not experience delamination.

Very truly yours,

John E. Maier

John E. Maier

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555
June 24, 1981

Docket No. 50-244
LS05-81-06-093

Mr. John E. Maier
Vice President
Electric and Steam Production
Rochester Gas & Electric Corporation
89 East Avenue
Rochester, New York 14649

Dear Mr. Maier:

SUBJECT: SYSTEMATIC EVALUATION PROGRAM TOPIC III-7.C, DELAMINATION OF PRESTRESSED CONCRETE CONTAINMENT STRUCTURES - GINNA

Enclosed is a copy of our draft evaluation of Systematic Evaluation Program Topic III-7.C.

You are requested to examine the facts upon which the staff has based its evaluation and respond either by confirming that the facts are correct, or by identifying errors and supplying the corrected information. We encourage you to supply any other material that might affect the staff's evaluation of this topic or be significant in the integrated assessment of your facility.

Your response is requested within 30 days of receipt of this letter. If no response is received within that time, we will assume that you have no comments or corrections.

In future correspondence regarding Systematic Evaluation Program topics, please refer to the topic numbers in your cover letter.

Sincerely,

Dennis M. Crutchfield

Dennis M. Crutchfield, Chief
Operating Reactors Branch No. 5
Division of Licensing

Enclosure:
As stated

cc w/enclosure:
See next page

Mr. John E. Maier

cc

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ENCLOSURE

GINNA PLANT

DOCKET NO. 50-244

SEP TOPIC III-7.C

DELAMINATION OF PRESTRESSED CONCRETE CONTAINMENT STRUCTURE

I. INTRODUCTION

Delaminations of concrete have occurred in the domes of two prestressed concrete containments, Crystal River and Turkey Point. The safety objective of this review is to assure that the containment will maintain its structural integrity in order that it may perform its intended safety function.

II. REVIEW CRITERIA

REFERENCES

a. SER Turkey Point No. 3, Docket No. 50-250

b. Containment Dome Report, Turkey Point No. 3, dated February, 1972.

c. SER Crystal River No. 3, Docket No. 50-302

d. Reactor Building Dome Delamination Final Report, Crystal River No. 3 dated December 10, 1976.

Problem history, analyses and repair procedures are described in the above references for the plants where dome delaminations occurred. The containment at Ginna, as described in the FSAR was compared with the containments referenced above in order to determine if such a failure could occur at Ginna.

III. RELATED SAFETY TOPICS AND INTERFACES

1. Containment structural integrity tests are reviewed under SEP Topic III-7.d.

2. Containment tendon inservice inspection program is reviewed under SEP Topic III-7.A.

IV. REVIEW GUIDELINES

The containment design and configuration are reviewed in order to assess the possibility that delamination might occur. Recommendations, based on that assessment are noted below.

V. EVALUATION

Delamination (cracks in planes parallel to inner and outer concrete surfaces) is caused by radial tension developed in the concrete by the forces from curved prestressing tendons. The curved prestressing tendons attempt to relieve the stresses in them and as a result may cause the concrete delaminate.

It appears that the two most significant factors which led to the delamination of the Turkey Point #3 and Crystal River #3 domes were radial tension in the concrete above the prestressing tendons and the use of a marginal strength coarse aggregate in the concrete.

The containment at Ginna is substantially different from Turkey Point #3 and Crystal River #3 in that the Ginna containment only contains straight, vertical prestressing tendons in the containment wall. There is no prestressing in the hoop direction of the containment wall or in the dome. Since there is no curvature in the prestressing tendons at Ginna, there would be no mechanism to cause radial tension in the concrete due to prestressing forces.

VI. CONCLUSION

The containment at Ginna would not experience delamination because the containment has no curved prestressing tendons to cause radial tension and delamination in the concrete due to prestressing forces.

Topic

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

May 8, 1980

Docket No. 50-244

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,

Dennis M. Crutchfield

Dennis M. Crutchfield, Chief
Operating Reactors Branch #5
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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. Curent 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.