Public Service
Electric and Gas
Company

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U. S. Nuclear Regulatory Commission
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Gentlemen:

GENERIC LETTER 88-20; MILESTONE REPORT
MECHANICAL STRESS ANALYSES FOR
SALEM AND HOPE CREEK GENERATING STATIONS
DOCKET NOS. 50-272, 50-311, AND 50-354

This submittal documents performance of respective Containment Mechanical Stress Analyses for Public Service Electric and Gas Company's (PSE&G's) Salem and Hope Creek generating stations. This effort is in accordance with the milestones and schedule for Generic Letter (GL) 88-20, which were conveyed in a letter dated September 18, 1991 (reference NLR-N91154).

Summaries of these analyses are provided for your information as Attachments A and B of this transmittal. Final PSE&G approval of these analyses is pending vendor resolution of PSE&G comments. It is not anticipated that resolution of these comments will result in any changes to the conclusions reached from the analyses. Furthermore, it should be noted that the development of the final IPE report is an iterative process and that these analyses may undergo revision as a result of this process. Please contact us if you have any questions regarding this transmittal.

Sincerely,

Steven E Miltenberger

Attachment

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C Mr. T. Martin, Administrator
Region I

Mr. T. Johnson
Senior Resident Inspector

Mr. J. Stone
Project Manager - Salem

Mr. S. Dembek
Project Manager - Hope Creek

Mr. K. Tosch, Chief
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ATTACHMENT A

Section 6.0 Summary and Tabulated Results

ABB Impell Report No. MV-0140-058-R001, Revision A
August 1991

6. SUMMARY

A probabilistic evaluation of the pressure capacity of the Salem Nuclear Power Generating Station Unit 1 containment structure is discussed in this report. Potential failure modes of the containment structure due to temperature and pressure loads well beyond the design basis conditions were considered in this evaluation. Failure was interpreted as leakage from the containment. As a result, the failure modes include both large structural failures as well as small leakage failures. The capacities of the various failure modes are reported as probabilistic quantities in terms of median failure pressures and their associated variabilities. The controlling failure modes were investigated for containment material temperatures ranging from 300° to 800°F. The potential failure modes examined included:

1. Membrane failures of the containment shell
2. Flexure and shear failures of the basemat
3. Local liner tearing
4. Failure at the containment wall - basemat junction
5. Failure at major hatches
6. Failure at pipe penetrations

In all cases, the failure modes were considered to be the result of a quasi-static pressure loading. The pressure rise times were assumed to be sufficiently long such that the dynamic transient response of the containment structure could be neglected. Also, the material temperatures were assumed to have reached a steady state.

Over the temperature range considered in this investigation, the critical failure mode, based on the median pressure capacity, was found to be associated with a membrane failure in the vicinity of the apex of the dome of the containment shell. The second most critical failure mode was found to be associated with a flexural failure of the basemat. Both of these failure modes correspond to large, gross structural failures. The third most critical failure mode was found to be due to local liner tearing adjacent to the thickened reinforcement plate around the equipment hatch and the personnel airlock. This failure mode corresponds to a leakage failure. The median failure pressures of the equipment hatch and the personnel airlock were found to be typically higher than the governing structural failure modes. A sampling of pipe penetrations were evaluated. The sampling was based on engineering judgment as to which penetrations would be expected to develop the largest penetration loads for a given radial deformation of the containment wall. For the pipe lines investigated, it was found that the pipes would develop plastic hinges prior to failing the penetrations. While pipe hinging can result in reduced fluid flow, it is not expected that the pressure boundary would be breached.

A review of the Salem Unit 2 containment structure was conducted. The purpose of the review was to look for differences in the structural configuration and details that could possibly lead to different results as compared to Unit 1. Based on the review of the available information, there are no differences between the Units 1 and 2 containment structures that impact the results of this study. Therefore, the pressure fragilities for the various failure modes and the leak areas evaluated for Unit 1 are applicable to Unit 2.

ATTACHMENT B

Section 4.0 Summary and Tabulation Results

ABB Impell Report No. MV-0140-058-R002, Revision A
September 1991

4. SUMMARY

The overpressure capacity of the containment structure at the Hope Creek Nuclear Power Generating Station has been discussed in this report. The capacities are reported as probabilistic quantities in terms of median failure pressures and their associated variabilities. In this investigation, failure was interpreted as leakage from the containment. Although several potential failure modes were investigated, the median capacities of only the controlling failure modes were reported. These controlling failure modes were investigated for containment metal temperatures ranging from 200° to 1000°F. The potential failure modes examined included:

1. Membrane failures of the drywell shell
2. Failure of the drywell head flange seal
3. Failure of the vent line from the drywell to the suppression chamber (torus)
4. Failure of the suppression chamber shell
5. Failure at penetrations

In all cases, the failure modes were considered to be the result of a quasi-static pressure loading. The pressure rise times were assumed to be sufficiently long such that the dynamic transient response of the containment structure could be neglected. Also, the material temperatures were assumed to have reached a steady state.

Over the range of temperatures considered in this investigation, leakage failure modes at hatches were found to be most critical. At temperatures less than 500°F, the silicone rubber o-rings are expected to have some rebound capability to maintain a seal with metal-to-metal flange separation. Thus, at low temperatures, leakage was found to be governed by the rebound of the o-ring seals. At the lower temperatures, the drywell head flange was found to be the critical failure mode. At temperatures greater than 500°F, the o-rings are expected to be completely degraded such that leakage occurs at pressure unseated flange connections with metal-to-metal separation. As a result, at the higher temperatures, the control rod drive removal hatch, the suppression chamber access hatches, and the drywell head access hatch also become controlling failure modes in addition to the drywell head flange.