December 18, 1996

Mr. Leon R. Eliason
Chief Nuclear Officer & President-
Nuclear Business Unit
Public Service Electric and Gas
Company
Post Office Box 236
Hancocks Bridge, NJ 08038

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION (RAI) REGARDING MARGIN RECOVERY AMENDMENT REQUEST, SALEM NUCLEAR GENERATING STATION, UNITS 1 AND 2 (TAC NOS. M95383 AND M95384)

Dear Mr. Eliason:

The staff is reviewing your May 10, 1996, amendment request for the margin recovery program for Salem Nuclear Generating Station, Units 1 and 2. Enclosed is an RAI that we need to complete our review. Please provide a response within 30 days of receipt of this letter.

Sincerely,

/S/

Leonard N. Olshan, Senior Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-272/311

Enclosure: As stated

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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REQUEST FOR ADDITIONAL INFORMATION
SALEM NUCLEAR GENERATING STATION UNITS, 1 AND 2
PROPOSED CHANGE OF TECHNICAL SPECIFICATIONS
(MARGIN RECOVERY PROGRAM)

1) Section 4.0, "Accident Analysis" - Please provide discussion on any computer code used in the transient and accident analyses which are not approved by NRC staff.

2) Section 4.1.1, "Uncontrolled RCCA Bank Withdrawal From a Subcritical Condition" - Please provide transient DNBR curve to demonstrate that the criterion of the MDNBR is met during this transient.

3) Section 4.1.3, "Rod Cluster Control Assembly Misalignment" - Please provide transient DNBR curve to demonstrate that the criterion of the MDNBR is met during this transient.

4) Section 4.1.4, "Uncontrolled Boron Dilution" - Please provide the results of an analysis to demonstrate sufficient times are available between the time of the alarm and the time of lost shutdown margin for all modes of plant operation per the SRP 15.4.6.

5) Section 4.1.5.3, "Single Reactor Coolant Pump Locked Rotor and Reactor Coolant Pump Shaft Break" - It is indicated that less than 5% of the total fuel rods experience DNB during a lock rotor event. Please confirm that in your evaluation, all fuel rods with a transient DNBR less than 1.34 are assumed experiencing DNB and fuel failure. Using the amount of fuel failure determined above, provide the results of an analysis to demonstrate that the radiological consequences are within 10 CFR Part 100 guidelines.

6) Section 4.1.8, "Loss of Normal Feedwater" and Section 4.1.9, "Loss of Offsite Power" - The PORVs were assumed operable during these transients. However, the technical specification allows power operation with PORVs isolated. Please provide the results of analyses considering PORVs inoperable.

7) Section 4.1.10, "Excessive Heat Removal Due to Feedwater System Malfunctions" - In the assessment of this section, it is indicated that this transient is less limiting than the excessive load increase evaluated in Section 4.1.11. However, the results of an excessive load increase is not presented in Section 4.1.11. Please provide the needed analysis results.

8) Section 4.1.13, "Main Steam System Failures" - Please provide transient DNBR curves for the accidental depressurization of main steam system and main steam line break events to demonstrate that the acceptance criteria of these events are met.

9) Section 4.1.14, "Spurious Operation of the SIS at Power" - Please address the effect of this event regarding potential solid pressurizer. (concern raised in Westinghouse NSAL-93-013)

ENCLOSURE

10) Section 4.1.15, "Single Rod Cluster Control Assemble Withdrawal at Full Power" - It is indicated that the results of this transient may cause fuel failure. However, this is a condition II event (SRP 15.4.2) and no DNB is allowed for this transient. Please discuss the acceptability of this analysis.

11) Section 4.1.16, "Major Rupture of a Main Feedwater Line" - Please provide the results of an analysis assuming PORVs inoperable. This is because the technical specification allows power operation with PORVs isolated.

12) Section 4.1.17, "RCCA Ejection" - The acceptance criteria of this event are specified in SRP 15.4.8. Specifically, the transient peak system pressure is below 110% of design pressure and the radiological consequences are well within the 10 CFR Part 100 guidelines. Please discuss the results of the analysis with respect to these acceptance criteria.

13) Section 4.3, "Steam Generator Tube Rupture" - Please describe the limiting single failure assumed in this analysis.

14) Page 7, 8 lines from the top. A reference is made to one T-hot RTD. Should the correct number be three or two depending on methodology for a failed T-hot RTD? One RTD would appear to be using the bypass manifolds not RTD bypass. If only one RTD is used then the CSA for the electronics may be ambitious. Table 2 states RTD used as three.

15) Page 8, Table 2. RMTE is assumed to be 0. Do plant procedures and available test equipment support this assumption?

16) Page 8, Table 2. Is hot leg streaming included in the uncertainty for T-average? Is it included in hot leg enthalpy, Table 5, page 18?

17) Page 15, second paragraph. Under what conditions <u>is</u> a systematic temperature error allowance included as a cross calibration systematic error.