March 8, 2001

Mr. Harold W. Keiser
Chief Nuclear Officer & President
PSEG Nuclear LLC - X04
Post Office Box 236
Hancocks Bridge, NJ 08038

SUBJECT: SALEM NUCLEAR GENERATING STATION, UNIT NOS. 1 AND 2, REQUEST FOR ADDITIONAL INFORMATION RE: INCREASE LICENSED POWER LEVELS BY APPROXIMATELY 1.4 PERCENT (TAC NOS. MB0521 AND MB0522)

Dear Mr. Keiser:

By application dated November 10, 2000, PSEG Nuclear LLC requested amendments to Facility Operating License Nos. DPR-70 and DPR-75 and the Technical Specifications, to increase the licensed power levels at the Salem Nuclear Generating Station, Unit Nos. 1 and 2, by approximately 1.4%.

The U.S. Nuclear Regulatory Commission staff is reviewing your amendment application and requires additional information in order to complete its evaluation. The enclosed request for additional information was discussed with Mr. Brian Thomas during a conference call on January 31, 2001. During the call, we agreed to establish a target date of 30 days from the date of this letter to receive your response. If circumstances result in the need to revise the target date, please contact me at (301) 415-1324.

Sincerely,

/RA/

Robert J. Fretz, Project Manager, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-272 and 50-311

Enclosure: Request for Additional Information

cc w/encl: See next page

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OFFICE	PDI-2/PM	PDI-2/LA	PDI-2/SC
NAME	RFretz	TLClark	JClifford
DATE	03/07/01	3/7/01	3/8/01

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REQUEST FOR ADDITIONAL INFORMATION

POWER UPRATE AMENDMENT REQUEST

SALEM NUCLEAR GENERATING STATION, UNIT NOS. 1 AND 2

By application dated November 10, 2000, PSEG Nuclear LLC submitted a request to increase licensed power levels for Salem Nuclear Generating Station, Unit Nos. 1 and 2 by 1.4 percent. By letter dated December 5, 2000, PSEG Nuclear provided additional information (Westinghouse Topical Reports WCAP-15565, Revision 0 and WCAP-15566, Revision 0) to support its November 10, 2000, submittal. The U.S. Nuclear Regulatory Commission (NRC) staff has reviewed the submittals and finds that additional information in the following areas is needed to complete its review.

Section 4.2.3 Steam Generator Blowdown System

1. The submittal contains a statement that the rate of addition of dissolved solids to the secondary system, in addition to being a function of condenser leakage and the quality of secondary makeup water, also depends on the rate of erosion-corrosion within the secondary system. Although the first two sources of dissolved solids do not change with power uprate, generation of particulates by erosion-corrosion may be affected by power uprate due to a change in velocities which may occur in the secondary systems. Please provide justification that the power uprate will not significantly alter generation of particulates by erosion-corrosion.

Section 5.2 Reactor Vessel Integrity - Neutron Irradiation

2. Regarding the information in Section 5.2, the Technical Specification changes in Attachment 4, and the exemption request in Attachment 6 to the November 12, 2000, submittal, the NRC staff has not approved the application of WCAP-15315 to remove reactor pressure vessel (RPV) head flange requirements from the Salem licensing basis. The staff was petitioned (as published in *Federal Register* notice 65 FR 6044) to undertake rulemaking to modify the requirements of Appendix G to 10 CFR Part 50 as they relate to RPV flange material property issues. The staff is in the process of acting on this petition and will follow the rulemaking process. Therefore, the staff has determined that, since we have not determined the contents of the final rule, it would be inappropriate to grant plant-specific exemptions during the rulemaking process.

   We request that you submit revised P-T limit curves that do not include the elimination of the flange requirements for Salem, Unit Nos. 1 and 2, to replace those submitted in Attachment 4 to the November 10, 2000, submittal.

3. Regarding the information submitted in Table 4-1 of WCAP-15565, Revision 0, the staff has compared the cited surface (which, based on other information in the WCAP, is apparently at the clad-to-base metal interface) fluence values to the values previously reported by the licensee and contained in the NRC staff's Reactor Vessel Integrity Database (RVID). The staff noted that while most of the fluence values calculated in WCAP-15565 for post-power uprated conditions did go up, the values cited for all of the longitudinal weld seams (2-042 A, B, and C and 3-042 A, B, and C) decreased slightly.

ENCLOSURE

Please explain how these numbers decreased as a result of the most recent fluence recalculations.

4. Regarding the information submitted in Table 4-1 of WCAP-15566, Revision 0, the staff has compared the cited surface (which, based on other information in the WCAP, is apparently at the clad-to-base metal interface) fluence values to the values previously reported by the licensee and contained in the NRC staff's Reactor Vessel Integrity Database (RVID). The staff noted that while most of the fluence values calculated in WCAP-15566 for post-power uprated conditions did go up, the values cited for intermediate shell longitudinal weld seam 2-442 A and lower shell longitudinal weld seam 3-442 B decreased slightly. Explain how these numbers decreased as a result of the most recent fluence recalculations.

5. Explain whether or not a change to the Salem Unit No. 1 or Unit No. 2 low-temperature overpressure protection (LTOP) system (or Pressurizer Overpressure Protection System) pressure setpoint or enable temperature is required as a result of the recalculation of RPV material properties for 32 effective full power years (EFPY) of operation.

Section 5.9 Steam Generators

6. In Section 5.9.5 of the power uprate submittal, PSEG Nuclear stated, without many details, that power uprate will have a negligible impact on the existing and potential tube degradation mechanisms. The NRC staff understands that the Unit No. 2 steam generators are experiencing the following active degradation: primary stress corrosion cracking in hot leg top of tubesheet transition zones, at hot leg dented tube support plate intersections, in low row U-bends, and in tube plugs; outside stress corrosion cracking in the hot leg freespan regions.

   In addition, the following degradation mechanisms have previously occurred in the Unit 2 steam generators: anti-vibration bar wear; thinning at cold leg tube support plate intersections; intergranular attack/stress corrosion cracking at hot leg top of tubesheet (sludge pile); outside diameter stress corrosion cracking at hot leg top of tubesheet and at tube support plate intersections.

   Therefore, in order to verify that General Design Criterion (GDC) No. 14, "Reactor Coolant Pressure Boundary," will continue to be met during future operating cycles at uprated conditions, please address the following:

   (a) Confirm whether our understanding is correct, and that the above potential degradation mechanisms are currently active. Provide a brief discussion describing the impact that power uprate will have on each of these degradation mechanisms;

   (b) Also, discuss whether the 40% thoughwall plugging limit for the steam generator tubes in the technical specifications under the power uprate condition satisfies NRC Regulatory Guide 1.121.

   (c) Will the power uprate impact future tube inspection and inspection frequencies?

(d) In Section 5.9.4, U-Bend Fatigue Evaluation, it states that an evaluation found that some steam generator tubes would be susceptible to high cycle fatigue at the uprated conditions with the plant operating at lower steam pressures. Therefore, according to your evaluation for Unit Nos. 1 and 2, which steam generator tubes did you find to be susceptible to U-bend fatigue? Also, where along the tubes are the critical positions? Do these differ between Unit Nos. 1 and 2? If so, why? What are the relevant parameters, with regard to fatigue, at those positions?

Furthermore, in order to independently evaluate the impact that uprated power levels have on certain limiting conditions when comparing current licensed power levels with the proposed uprated levels, please provide the following information described in the table below:

	Unit No. 1		**Unit No. 2**	
Parameter	**Current Power Level**	**1.4 % Increased**	**Current Power Level**	**1.4 % Increased**
Steam Flow				
Circulation Ratio				
Steam Pressure				
Primary System Temperature				
Amplitude and Direction of the Cyclic Deformation at the Limiting Point along the Tube				
Frequency of Deformation at the Limiting Point along the Tube				
Limiting Number of Cycles				
Expected Number of Cycles to End-of-Service				

Section 1.4.6 Instrumentation and Controls - Uncertainty Determination

7. In order to confirm that licensed power levels will not be exceeded at uprated conditions, the NRC staff needs additional information concerning how instrument uncertainty was calculated. Therefore, the following needs to be addressed:

   (a) Attachment 1, Section 1.4.6, states that CENP has completed the Salem, Unit Nos. 1 and 2, CENP Crossflow uncertainty calculations A-SA1-PS-0001, Revision 0, and A-SA2-PS-0001, Revision 0. Therefore, using a copy of one of these calculations, please provide a further explanation of how the estimated uncertainty of the net heat input from the reactor coolant pump (RCP) to the reactor coolant system (RCS) resulted in the values for total heat input and core power uncertainties stated on page 8 of WCAP-15553.

   (b) Section 5.10 of CENPD-397-P-A stated that licensees desiring to lower the total feedwater flow measurement uncertainty can do so by simply improving the accuracy of the feedwater temperature instrumentation. Westinghouse Topical Report WCAP-15553, Table 1, shows a value for the feedwater temperature instrumentation uncertainty. How was this value for the uncertainty determined? Was this value based on actual plant data or was it provided by the instrument supplier?

   (c) Westinghouse Topical Report WCAP-15553, Table 1, shows instrumentation uncertainties of [x] pounds per square inch (psi), [y]% flow span, and [z] psi for feedwater pressure (percent span), Steam Generator Blowdown (percent differential pressure (dP) span), and steam pressure (percent span), respectively. Explain how the values for [x], [y], and [z] were calculated.

cc:

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