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PSEG
Nuclear LLC

United States Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Gentlemen:

**RESPONSE TO APRIL 12, 2001 REQUEST FOR ADDITIONAL INFORMATION IN REGARDS TO REQUEST FOR LICENSE AMENDMENT INCREASED LICENSED POWER LEVEL SALEM GENERATING STATION, UNIT NOS. 1 AND 2 FACILITY OPERATING LICENSE DPR-70 AND DPR-75 DOCKET NOS. 50-272 AND 50-311**

On April 12, 2001, the NRC issued a request for additional information (RAI) to support the staff's review of the request for license amendment submitted by PSEG Nuclear LLC on November 10, 2000 requesting an increase in licensed power levels for Salem Generating Station Unit Nos. 1 and 2. The response to the request for additional information is contained in Attachment 1.

Should you have any questions regarding this request, please contact Mr. Brian Thomas at (856)339-2022.

Sincerely,

G. Salamon
Manager – Nuclear Safety and Licensing

Attachments (2)

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**ATTACHMENT 1**
**SALEM GENERATING STATION**
**UNIT NOS. 1 AND 2**
**FACILITY OPERATING LICENSE DPR-70 AND DPR-75**
**DOCKET NOS. 50-272 AND 50-311**
**RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION**
**INCREASED LICENSED POWER LEVEL**

On April 12, 2001, the NRC issued a request for additional information (RAI) concerning PSEG Nuclear's request for amendment to increase the licensed power level for Salem Unit Nos. 1 and 2. This attachment provides the response to the RAI question.

**NRC Question:**

Nuclear power plants are licensed to operate at a specified power, which, at operating power levels, is indicated in the control room by neutron flux instrumentation that has been calibrated to correspond to core thermal power. Core thermal power is determined by a calculation of the energy balance of the plant nuclear steam supply system. The accuracy of this calculation depends primarily upon the accuracy of feedwater flow, temperature, and pressure measurements, which are not safety grade and are not included in the plant technical specifications.

The uncertainty of calculating values of core thermal power determines the probability of exceeding the power levels assumed in the design basis transient and accident analyses. In this regard, to allow for uncertainties in determining thermal power (e.g., instrument measurement uncertainties), Appendix K to 10 CFR Part 50, requires loss of coolant accident (LOCA) and emergency core cooling system (ECCS) analyses to assume that the reactor had operated continuously at a power level at least 102 percent of the licensed thermal power. The 2 percent power margin uncertainty value was intended to address uncertainties related to heat sources in addition to instrument measurement uncertainties. Later, the NRC concluded that, at the time of the original ECCS rulemaking, the 2 percent power margin requirement appeared to be based solely on considerations associated with power measurement uncertainty.

Appendix K to 10 CFR Part 50 did not require demonstration of the power measurement uncertainty and mandated a 2 percent margin, notwithstanding that the instruments used to calibrate the neutron flux instrumentation may be more accurate than originally assumed in the ECCS rulemaking. In the June 1, 2000, *Federal Register* (Volume 65, Number 106, Rules and Regulations, pages 34913-34921) the Commission published a final rule to reduce an unnecessarily burdensome regulatory requirement by allowing licensees to justify a smaller margin for power measurement uncertainty by using more accurate instrumentation to calculate the reactor thermal power and thereby calibrate the neutron flux instrumentation.

The purpose of the proposed changes is to obtain a power uprate on the basis of plant modifications that would result in improved accuracy of feedwater flow rate measurement, which is used in the calculation of reactor thermal power. The improved instrumentation (Crossflow ultrasonic flow measurement system) would allow the licensee to operate Salem with a reduced margin between the actual power level and the 102 percent margin used in the licensing basis ECCS analyses.

To complete its review of the proposed license changes, the staff requests a description of the programs and procedures that will control calibration of the non-safety-grade instrumentation that affect the total power uncertainty described in the licensee's proposed power uprate license amendment. The licensee has provided this information for the Crossflow system. For the remaining instrumentation the description should include a discussion of the procedures for:

a. Maintaining calibration;
b. Controlling software and hardware configuration;
c. Performing corrective actions;
d. Reporting deficiencies to the manufacturer; and
e. Receiving and addressing manufacturer deficiency reports.

The regulatory basis for this question is to verify that programs and procedures are in place to demonstrate that the actual power measurement uncertainty will not exceed the 0.6 percent uncertainty assumed in the licensee's analyses. This will provide assurance that the 1.4 percent power uprate is justified given the 2 percent margin required by Appendix K to 10 CFR Part 50.

**PSEG Nuclear response:**

**Maintaining Calibration**

Preventive maintenance (PM) is performed on the feedwater measurement instruments as well as the instruments listed below that affect the power uncertainty. The PMs listed and intervals are current practice but may be revised in the future based on the PM program requirements. The PM program is currently controlled by procedure NC.WM-AP.ZZ-0003, "Regular Maintenance Process."

Feedwater Flow

The feedwater flow instruments S1(2)CN -1(2)FL8924Z, 1(2)FL8925Z, 1(2)FL8926 and 1(2)FL8927Z are calibrated every eighteen months. Calibration of these devices is however not critical to maintaining feedwater flow accuracy as the Crossflow system will provide a correction factor to ensure feedwater mass flow is maintained at 0.5% mass flow uncertainty.

Procedures S1(2)IC-LC.CN-0025, S1(2)IC-LC.CN-0026 S1(2)IC-LC.CN-0027 and S1(2)IC-LC.CN-0028 currently perform the calibration of the above listed devices.

Feedwater Pressure

Feedwater pressure instrument S1(2)CN –1PT508 will be used to provide feedwater pressure input used by Crossflow to calculate feedwater mass flow. This instrument is a new parameter for power uncertainty to support the installation of the Crossflow instrumentation. A PM will be established as part of the implementation of the power uprate to calibrate this instrument on a nominal 24-month cycle. Procedures S1(2).IC-SC.CN-0110 are used to calibrate these instruments.

Steam Generator (SG) Blowdown Flow

Steam Generator Blowdown Flow instrumentation S1(2)GBD-1(2)FA3178, 3180, 3182, 3184 are calibrated by procedures S1(2).IC-LC.GBD-0001. These calibrations are performed every 18 months.

Feedwater Temperature

To support the installation of the Crossflow instrumentation, additional feedwater temperature computer points (T2402A, T2403A, T2404A and T2405A) were added. As part of the implementation of the uprate, the feedwater temperature computer points will be checked on a monthly basis and recalibrated if found out of tolerance. An annual calibration check will also be performed concurrent with the calibration of redundant local temperature indicator S1(2)CN -1(2)TL8885. For the monthly and annual calibrations, new computer coefficients for the feedwater RTDs will be provided if plant computer indications are out of tolerance. The calibrations will be included under procedure SC.IC-CC.CN-0011(Q).

Steam Pressure

Steam Pressure transmitters S1(2)CN-1(2)PT514, PT515, PT 516, PT524, PT525, PT526, PT534, PT535, PT536, PT544, PT545, and PT 546 are calibrated every 18 months. These devices are calibrated in accordance with procedures S1(2).IC-SC.RCP-0030, S1(2).IC-SC.RCP-0031, S1(2).IC-SC.RCP-0032, S1(2).IC-SC.RCP-0040, S1(2).IC-SC.RCP-0041, S1(2).IC-SC.RCP-0042, S1(2).IC-SC.RCP-0050, S1(2).IC-SC.RCP-0051, S1(2).IC-SC.RCP-0052, S1(2).IC-SC.RCP-0060, S1(2).IC-SC.RCP-0061, S1(2).IC-SC.RCP-0062.

### Controlling Software and Hardware Configuration

The software and hardware configuration of digital plant instrumentation (e.g., Crossflow and plant computer) are controlled by procedure NC.NA-AP.ZZ-0064(Q), "Software Quality Assurance" and the associated implementing procedures. These procedures ensure that the appropriate quality level classifications are identified for the equipment. The quality level classification in turn determines the appropriate software quality assurance program elements that are applied to the equipment.

### Performing Corrective Actions

Maintenance and corrective action items are generated through PSEG Nuclear's notification process that is governed by procedure NC.WM-AP.ZZ-0000, "Notification Process." This program is constructed to ensure conditions adverse to quality are dispositioned and corrected in accordance with 10CFR50, Appendix B, Criterion XV, Nonconforming Materials, Parts, or Components, and Criterion XVI, Corrective action.

### Reporting Deficiencies to the Manufacturer

Vendors are contacted to assist in the determination of Part 21 reporting for equipment deficiencies that cross the threshold of requiring reporting under 10 CFR Part 21. During the course of maintenance, vendors are routinely contacted to assist in the repair of station equipment, however, there is no formal process for reporting every equipment deficiency to the manufacturer,

### Receiving and Addressing Manufacturer Deficiency Reports

Manufacturer deficiency reports are handled through PSEG Nuclear's vendor information process which is governed by procedure NC.NA-AP.ZZ-0043, "Vendor Information Program." When vendor documents are received they are routed to the responsible group for evaluation and disposition. External 10 CFR Part 21 deficiencies submitted by vendors are processed as prescribed in procedure NC.PM-AP.ZZ-0603(Q), "Specification Review, Approval and Processing of Supplier Part 21 Data." Vendor Part 21 items are tracked under PSEG Nuclear's corrective action program for proper disposition.