April 9, 2001

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

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION, LICENSE AMENDMENT REQUEST TO INCREASE POWER LEVEL BY 1.4 PERCENT, HOPE CREEK GENERATING STATION (TAC NO. MB0644)

Dear Mr. Keiser:

In a letter dated December 1, 2000, as supplemented on February 12, 2001, PSEG Nuclear LLC (PSEG) submitted a license amendment request to increase the Hope Creek Generating Station (HCGS) power level by 1.4 percent. In the letter dated December 1, 2000, PSEG also requested that the staff approve two exemptions to allow PSEG to use American Society of Mechanical Engineers (ASME) Code Cases N-588 and N-640 as part of the basis for generating the new pressure-temperature (P-T) limit curves for the HCGS pressure vessel and reactor coolant pressure boundary.

The U.S. Nuclear Regulatory Commission staff is reviewing your submittal and has determined that additional information is required to complete the review. The specific information requested is addressed in the enclosure. We request that the additional information be provided within 30 days of receipt of this letter. The 30-day response timeframe was discussed with Mr. John Nagle of your staff on April 4, 2001. If circumstances result in the need to revise your response date, or if you have any questions, please contact me at (301) 415-1420.

Sincerely,

/RA/

Richard B. Ennis, Project Manager, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-354

Enclosure: Request for Additional Information

cc w/encl: See next page

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REQUEST FOR ADDITIONAL INFORMATION

LICENSE AMENDMENT REQUEST TO INCREASE POWER LEVEL BY 1.4 PERCENT

HOPE CREEK GENERATING STATION

The following questions pertain to the PSEG Nuclear LLC (PSEG or the licensee) submittal dated December 1, 2000, for Hope Creek Generating Station (HCGS).

1. Attachment 1, Section 9.1, of the submittal provides the justification for the requested power uprate with respect to the design of the fuel pool cooling and cleanup system (FPCCS). The FPCCS is designed to remove heat and impurities from the spent fuel pool. The licensee has indicated that the FPCCS heat removal function will not be affected by the power uprate, but its cleaning function was not addressed. Describe how the removal of impurities from the water in the spent fuel pool will be affected by the power uprate.

   The regulatory basis for this question is that the cleanup portion of the FPCCS conforms to the requirements of General Design Criteria (GDC) 61 of Appendix A to Part 50 of Title 10 of the Code of Federal Regulations (10 CFR Part 50) as it relates to appropriate filtering systems for fuel storage.

2. Attachment 1, Section 9.3, of the submittal provides the justification for the requested power uprate with respect to the design of the Standby Liquid Control System (SLCS). Provide justification for why the concentration of sodium pentaborate in the SLCS is not changed after the power uprate.

   The regulatory basis for this question is that the SLCS conforms to the reactivity control requirements of 10 CFR 50.62(c)(4).

3. Attachment 1, Section 10, of the submittal provides the justification for the requested power uprate with respect to the design of the Steam and Power Conversion Systems. The submittal states that the power conversion systems and their support systems were designed for 105 percent of rated steam flow and that the proposed 1.4 percent power uprate will increase the rated steam and feedwater flow by about 1.8 percent. Therefore, the proposed power uprate has no impact on the power conversion systems since the increased flow is bounded by the design conditions. Does the design analysis also bound the turbine overspeed and associated missile production for the 1.8 percent increase in steam flow?

   The regulatory basis for this question is that the turbine generator system conforms to the requirements of GDC 4 of Appendix A to 10 CFR Part 50 as it relates to the protection of structures, systems, and components important to safety from the effects of turbine missiles.

Enclosure

4. Attachment 5 of the submittal provides PSEG's justification for an exemption request associated with the use of American Society of Mechanical Engineers (ASME) Code Case N-588. In a telephone conversation on March 30, 2001, the NRC staff questioned if the exemption was needed for HCGS. Specifically, the staff stated that Code Case N-588 does not appear to provide any benefit since the HCGS reactor pressure vessel is not limited by circumferential weld material in the vessel. The NRC staff requested that PSEG either withdraw the exemption request or provide additional information that demonstrates the need for the exemption. Your staff indicated that the exemption was needed with respect to procedures for determining stress intensity factors and stated that additional information would be provided to justify the exemption request.

5. In order to assist in the evaluation of the effects of the proposed change on the Updated Final Safety Analysis Report Chapter 15 analyses, please provide a copy of the fuel vendor's supplemental reload analysis report (or similar documentation as discussed in a telephone conversation on March 28, 2001) for the current fuel cycle. This information is required to assure that proposed changes conform to the requirements of:

    a) GDC 10 of Appendix A to 10 CFR Part 50 as it relates to the reactor coolant system being designed with appropriate margin to ensure that specified fuel design limits are not exceeded during normal operations including anticipated operational occurrences;

    b) GDC 15 of Appendix A to 10 CFR Part 50 as it relates to the reactor coolant system and its associated auxiliaries being designed with appropriate margin to ensure that the pressure boundary will not be breached during normal operations including anticipated operational occurrences;

    c) GDC 20 of Appendix A to 10 CFR Part 50 as it relates the reactor protection system being designed to initiate automatically the operation of appropriate systems, including the reactivity control systems, to ensure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including anticipated operational occurrences; and

    d) GDC 26 of Appendix A to 10 CFR Part 50 as it relates to the reliable control of reactivity changes to ensure that specified acceptable fuel design limits are not exceeded, including anticipated operational occurrences.

6. Attachment 1, Section 5.5, of the submittal provides the justification for the requested power uprate with respect to the design of the reactor coolant and balance-of-plant (BOP) piping. List the most critical BOP piping systems that were evaluated for the power uprate. Provide a summary of the evaluation used for BOP piping, components, and pipe supports, nozzles, penetrations, guides, valves, pumps, heat exchangers, and anchorage for pipe supports.

   The regulatory basis for this question is that the BOP piping systems conform to the requirements of GDCs 1, 2, 4, 14, and 15 of Appendix A to 10 CFR Part 50 as they relate to maintaining structural integrity of pressure-retaining components and their supports (reference Standard Review Plan (SRP) Section 3.9.3).

7. Attachment 1, Section 5.11, of the submittal provides the justification for the requested power uprate with respect to the design of the control rod drive hydraulic system. Provide a summary of evaluation for the effects of the 1.4 percent power uprate on the design basis analysis of the control rod drive mechanism (CRDM). Confirm that the CRDMs structural integrity will be adequate for the 1.4 percent power uprate.

   The regulatory basis for this question is that the CRDMs conform to the requirements of GDC 14 of Appendix A to 10 CFR Part 50 as it relates to maintaining the reactor coolant pressure boundary.

8. Discuss the functionality of safety-related mechanical components (i.e., all safety-related valves and pumps, including air-operated valves (AOV) and power-operated relief valves) affected by the power uprate to demonstrate that the performance specifications and technical specification requirements (e.g., flow rate, close and open times) will be met for the proposed power uprate. Confirm that safety-related motor-operated valves (MOVs) in your Generic Letter (GL) 89-10 MOV program at HCGS will be capable of performing their intended function(s) following the power uprate including such affected parameters as fluid flow, temperature, pressure and differential pressure, and ambient temperature conditions. Please discuss effects of the proposed power uprate on the pressure locking and thermal binding of safety-related power-operated gate valves for GL 95-07 and on the evaluation of overpressurization of isolated piping segments for GL 96-06.

   The regulatory basis for this question is that the assumptions, analyses, and conclusions of the HCGS programs associated with GL 89-10, GL 95-07, and GL 96-06 remain valid (i.e., consistent with the current licensing basis).

9. 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 HCGS 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.