April 20, 2001

Mr. James Scarola, Vice President
Shearon Harris Nuclear Power Plant
Carolina Power & Light Company
Post Office Box 165, Mail Code: Zone 1
New Hill, North Carolina 27562-0165

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION REGARDING AMENDMENT REQUEST FOR STEAM GENERATOR REPLACEMENT/POWER UPRATE - SHEARON HARRIS NUCLEAR POWER PLANT (TAC NOS. MB0199 AND MB0782)

Dear Mr. Scarola:

By letters dated October 4, and December 14, 2000, you requested license amendments to revise the Shearon Harris Nuclear Power Plant Facility Operating License and Technical Specifications to support steam generator replacement and to allow operation at an uprated core power level of 2900 MWt.

During the course of our review of these requests, the NRC staff has determined that additional information is necessary to complete our review. The enclosed request for additional information was e-mailed to your licensing staff on April 5, 2001, and discussed during a telephone call on April 19, 2001. A mutually agreeable target date of May 31, 2001, for your response was established. If circumstances result in the need to revise the target date, please call me at the earliest opportunity.

Sincerely,

/RA/

Richard J. Laufer, Project Manager, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-400

Enclosure: As stated

cc w/encl: See next page

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Request for Additional Information
Request for License Amendment: Steam Generator Replacement/Power Uprate
Shearon Harris Nuclear Power Plant
Docket No. 50-400

1. Please clarify if an assessment has been performed to confirm that the current plugging limit of 40 percent in the Technical Specifications (TS) will continue to be applicable for the replacement steam generators (SGs).

2. Although not required by the TS, please clarify whether the licensee will be following the Electric Power Research Institute (EPRI) Pressurized-Water Reactor (PWR) Steam Generator Examination Guidelines for their pre-service and in-service inspection for the replacement SGs. If exceptions are taken to the EPRI Guidelines, please provide a summary of the scope for the pre-service inspection and the first in-service inspection.

3. Please clarify the discussion of sleeving and F* criteria starting on page 23 of Enclosure 1, "Basis for Proposed Changes," of the October 4, 2000, submittal. Since sleeving and F* will be deleted, why are they included as a basis for the proposed changes?

4. Section 5.7.1.3 of Enclosure 6, "NSSS [Nuclear Steam Supply System] Licensing Report," of the October 4, 2000, submittal, states that the primary side and the secondary side components were evaluated for the effects of changes to the thermal transients due to the power uprate. Section 5.7.1.4.1 also states, on page 5.7-4, that "LOCA [Loss-of-Coolant Accident] effects, which would show some impact due to uprate, were evaluated for the effects of a loop pipe break. Since leak-before-break has been applied to HNP [Harris Nuclear Plant], only the large auxiliary line breaks need to be considered. Therefore, the current analysis is conservative and will continue to envelop the LOCA effects in the uprate condition."

    1. Please clarify whether the thermal transient effects due to large-bore reactor coolant system (RCS) pipe break LOCAs were considered in the current licensing basis for the design of the HNP SGs. If not, explain why they were not considered (Note that the approved leak-before-break condition applies only to dynamic effects).

    2. Please provide the stress analysis results for the primary side components of the SGs, including the SG tubes, to demonstrate the adequacy of the HNP Model Delta 75 replacement SGs for the effects of thermal transients arising from postulated large-bore RCS pipe break LOCA conditions during the power uprate.

5. As a result of the SG replacement and power uprate, the feedwater system flow and pressure have to increase from those required for the current SGs at both the current and uprated power levels. Section 5.7.3 of the NSSS Licensing Report provides some discussion on the flow-induced vibration and wear. In general, the report states that the structural evaluation of the tubes addresses the effects of pressure and temperature resulting from the transients. However, no discussion is provided regarding the effects of increased flows on the flow-induced vibration of tubes.

Please discuss the potential for flow-induced vibration of the SG tubes due to various mechanisms, including, in particular, the fluid-elastic instability in the current SG at the current power level. Provide an evaluation of the flow-induced vibration of the SG tubes in the replacement SGs at the uprated power condition describing the analysis methodology, damping value of the tubes and the computer code used in the analysis. Provide the results of the predicted vibration levels during the normal operating condition and the worst-case transient condition, and the calculated fluid-elastic instability ratios. If the details of the analysis and the results are documented in a report, submit the report for staff review. Explain whether or not the analysis results are applicable to the degraded SG condition and why.

cc:

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