March 27, 2001

Mr. James Scarola, Vice President
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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 discussed with your licensing staff, and a mutually agreeable target date of May 18, 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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ADAMS ACCESSION NUMBER: ML010860456 * no major change to RAI

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

1. Provide a list of the methodologies and computer codes used in the loss-of-coolant accident (LOCA) and non-LOCA transient analysis for the steam generator replacement (SGR) and power uprate (PU) application, and reference the associated NRC acceptance letters to confirm the acceptance of the methodologies and codes used in the safety analysis for the Harris Nuclear Plant (HNP). Also, provide a discussion to address the compliance with each of applicable limitations and restrictions specified in the NRC safety evaluation reports (SERs) for use of the methodologies and codes applied to the HNP SGR/PU accident analysis.

2. Provide a list of the assumptions and ranges of initial conditions of the key plant parameters considered for each LOCA and non-LOCA transient analyzed. For example, the key parameters for the steamline break (SLB) analysis should include the initial core power level, initial core inlet temperature, initial reactor coolant system (RCS) flow rate, initial pressurizer pressure and water volume, radial peaking factor, control rod worth, initial steam generator (SG) liquid inventory, core burnup, blowdown fluid and blowdown area for each steamline. Also, confirm for each transient analyzed that the set of initial conditions used in the analysis is the limiting conditions that result in the worst case.

3. Provide a list of values for the input parameters that are used in the accident analysis to specifically reflect the changes of operating conditions and plant configurations for operation of HNP with the SGR and PU. Compare these values with that assumed in the existing accident analysis to identify the changes from the analysis of the record.

4. Provide a list of the systems or components that are non-safety-related and credited in the accident analysis. For each of these non-safety-related equipment, provide justification to show the acceptability of its use for consequence mitigation during a transient. Also, item (c)2(ii)(C) of Title10 of the *Code of Federal Regulations* (10 CFR) Section 50.36 requires a technical specification (TS) for the systems or components that are used for event mitigation. Accordingly, the licensee is requested to provide the required TS.

5. Provide a list of all the systems or components considered in determination of the single-failure events for the safety analyses. List the worst single-failure events assumed in the safety analysis for each event analyzed, and discuss the rationale of selecting the worst single-failure event for each event.

6. General Design Criterion (GDC) 17 in 10 CFR Part 50 , Appendix A requires, in part, that "An onsite electric power system and an offsite power system shall be provided to permit functioning of structures, systems, and components import to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of AOOs [anticipated operational occurrences] and (2) the core is cooled and containment and other vital function are maintained in the event of postulated accidents." In accordance with the GDC 17 requirements, a loss of offsite

Enclosure

power (LOOP) must not be considered as a single-failure event and must be assumed as part of the event initiation in the analysis for each AOO and accident without

changing the event category. The staff finds that the safety analyses in Section 6.2* do not address the LOOP effects for a majority of the transients. The licensee is requested to identify the events that do not assume an LOOP in the analyses as part of event initiation and analyze these events with an LOOP to comply with the GDC 17 requirements with respect to the LOOP assumption for the safety analysis. Submit the results of the requested analysis for the staff to review and approve.

7. The licensee submitted a new large-break (LB) LOCA analysis on December 14, 2000, to replace the original LBLOCA analysis submitted on October 4, 2000. The staff finds that in both the new and original analyses the same analytical methods were used and same ranges of power level, SG tube plugging and reactor vessel average coolant temperatures were assumed. The results of two sets of LBLOCA are different in that the limiting break and the peak clad temperature (PCT) are changed. The licensee is requested to compare the new and original LBLOCA analyses and identify any changes of the methods and assumptions that result in a different limiting case and PCT.

8. Section 6.1.2* presents the results of analysis for the small-break (SB) LOCA events. The events are assumed to be initiated from full-power (Mode 1) conditions. No discussion is provided to address the SBLOCA (a decreased RCS inventory event) effects for the plant operating at low-power modes or shutdown conditions. For the operations other than Mode 1, a decreased RCS inventory event may occur due to inadvertent opening of valves or inadvertent RCS drainage. Under the low-power and shutdown conditions, the available safety systems for accident mitigation are limited as compared to the Mode 1 conditions. The licensee is requested to address the effects of RCS drainage events initiated from the conditions of Modes 2 through 6.

9. Sections 6.1.3 and 4* discuss analyses of post-LOCA long-term core cooling and hot-leg switchover (HLSO). The licensee is requested to confirm that the hydraulic and boron mixing models used in the analyses are acceptable for the licensing applications. Also, compare the calculated range of the hot-leg switchover time with that specified in the post-LOCA HLSO procedure to show that the calculated HLSO time is appropriately reflected in the HLSO procedure.

10. Section 6.1.5* indicates that for the control rod ejection event, no fuel is expected to fail due to either departure from nucleate boiling (DNB) or fuel melting, and the overpressurization consequences are bounded by those of the turbine trip (TT) event. The licensee is requested to provide (1) a figure showing the calculated DNB ratios (DNBRs) during the transient, (2) a figure showing the calculated temperatures for the hot spot fuel centerline and outer fuel clad during the transient, and (3) an analysis showing that the overpressurization consequences are limited by those of the TT event.

11. The licensee classifies the design-basis events according to their anticipated frequency of occurrence identified as Condition 1 normal operation and operational transients; Condition II faults of moderate frequency; Condition III infrequent faults; and Condition IV limiting faults. In Table 6.2.0-1*, the licensee lists the design-basis events evaluated or analyzed under Conditions II, III and IV. The staff finds that the classification of these events is generally consistent with the guidance of the Standard Review Plan (SRP) and current licensing practices. However, the complete loss of forced reactor coolant flow and single rod withdrawal event are listed in the Table as a Condition III infrequent fault. This event categorization is inconsistent with SRP 15.3.1 and 15.4.3 that classify the complete loss of forced reactor coolant flow event and the single rod withdrawal event

as Condition II events (faults of moderate frequency) with the acceptance criteria that require the DNBR to not exceed the specified limit. The licensee is requested to justify the deviation from the SRP related to the event classification of the complete loss forced reactor coolant flow and the single rod withdrawal events and address the adequacy of the results of analysis for these events.

12. Provide a list of the setpoints with the associated uncertainties for normal operation and the setpoints assumed in the transient analysis for engineered safety feature actuation systems, pressurizer safety valves, power-operated relief valves (PORVs) and SG safety valves. Compare these analytical values with the applicable TS values and address the acceptability of the TS values.

13. As stated in the SRP, one of the acceptance criteria for the transient analysis is related to the calculated DNBRs. The staff finds that the analyses in Section 6.2* do not provide calculated DNBRs during a transient for most of the events. The licensee is requested to list the events that result in a decease in DNBRs and provide figures for these events to show the calculated DNBRs during transients. For cases (such as the locked rotor event) that are predicted to result in fuel rod damage because of the low calculated DNBRs, the licensee is requested to discuss the methods and input assumptions (such as pin census data and peak factors) used to determine the percentage of the damaged fuel rods and confirm the acceptance of the calculational methods and results.

14. Section 6.2.2.* states that the increased feedwater flow event is analyzed to ensure that "adequate margin to SAFDLs [specified acceptable fuel design limits] is maintained, and that protection against steam generator overfill is maintained." The staff finds that no figure is presented to show that the calculated DNBRs do not exceed the SAFDLs during the transient. The licensee is requested to provide the figure showing the calculated DNBRs. (This request is applied to all the transients that result in decreased DNBRs - see RAI 11.) The staff also finds that no sufficient information is presented for the assumptions used in the analysis to address the SG overfill issue. Specifically, for the case initiated from a full opening of a feedwater isolation valve without the isolation valve reclosure because of a single failure consideration, the licensee is requested to identify the safety-related equipment that are credible to isolate the feedwater in order to prevent SG overfill. If the licensee needs to credit non-safety-related systems or components (such as the feedwater control valves or feedwater pumps) to isolate or terminate the feedwater, the licensee should show that the non-safety-related system or component is reliable for feedwater isolation and provide a TS limiting condition for operation (LCO) to meet the requirements specified in item (c)2(ii)(C) of 10 CFR 50.36. (This request is applied to all the transients that credited the non-safety-related equipment for consequence mitigation - see RAI 4.)

15. Section 6.2.3* states that for the increased steam flow event, two cases are analyzed: one for minimum neutronics feedback (beginning-of-cycle (BOC) conditions) and the other for maximum neutronics feedback (end-of-cycle (EOC) conditions). Both cases are evaluated with automatic rod control. The licensee is requested to provide an analysis to show that the cases with automatic rod control are more limiting than the cases without automatic rod control. Also, provide the values of the moderator temperature and Doppler feedback coefficients assumed in the analysis for the BOC and EOC cores and confirm that the analytical values are bounded by the TS values.

16. Section 6.2.5* includes a discussion of the SLB analysis. The licensee states that the previously NRC-approved methodology (EMF-84-093) was used to perform the SLB events. The staff recognizes that the referenced SLB methodology was generically approved by the NRC. The staff also notes that the limitations of use of the methodology were identified in the NRC SER. The licensee should discuss the values used for the input parameters in the SLB analysis for HNP applications and confirm that it complies with the limitations for the input parameters or assumptions related to worst stuck control element assembly assumption, moderator reactivity coefficient, break size and location, blowdown fluid quality, single failure consideration, auxiliary feedwater flow and temperature. (The compliance with SER restrictions is applied to all the transient - see RAI 1.)

17. Table 6.2.8-4* lists the allowable high flux trip setpoints as a function of the number of inoperable main steam safety valves (MSSVs). The trip setpoints were credited in the analysis for transient such as the turbine trip event. In accordance with the requirements of item (c)2(ii)(C) of 10 CFR 50.36, the licensee is requested to provide a TS LCO for these high flux trip setpoints. (The licensee states (on page Enclosure 1-23 to a letter dated October 4, 2000) that proposed TS 3.7.1.1 includes the revised maximum power range neutron flux high setpoint with inoperable MSSVs. The proposed TS 3.7.1.1. is not available for the staff to review.)

18. Section 6.2.13* presents a discussion of the feedwater line break (FLB) accident analysis. The results show that for both FLB cases with and without offsite power available, the pressurizer becomes solid during the event. The safety relief valves are assumed to repeatedly open and close for an extended period of time in the water blowdown environment. TMI action Item II.D.1 requires that all RCS safety, relief, and blocked valves be tested to confirm the valve operability under expected operating conditions for design-basis transients and accidents. Accordingly, the licensee is requested to provide analysis or test data, or reference the NRC approval letter to show that (1) the safety relief valves (SRVs) can be operable (opening and closing on demand) under the water environment, and (2) the SRVs are reliable for repeated opening and closing during a transient for an extended period of time. Also, confirm that the value of initial pressurizer water level used in the pressurizer-overfill analysis maximizes the calculated pressurizer water level and is conservative as compared to the TS value.

19. Section 6.2.15* indicates that for the complete loss of forced reactor coolant flow event, two cases are analyzed: 100-percent power with an moderator temperature coefficient (MTC) of 0.0 pcm/^{o}F and 70 percent with MTC of +5.0 pcm/^{o}F. The licensee is requested to compare the analytical values of MTC with the TS values and confirm that the analytical values of MTC are bounded by the TS values.

    Table 6.2.15-1*, Event Summary for Loss of Forced Reactor Coolant Flow, indicates that pressurizer PORV is credited in the transient analysis for the event initiated from the full-power conditions. The licensee should reference a TS LCO for the PORV to show its compliance with item (c)2(ii)(C) of 10 CFR 50.36 requirements. (This RAI is applied to transient analysis for all the cases - see RAI 4.)

20. Section 6.2.19*, Uncontrolled Rod Cluster Control Assembly [RCCA] Withdrawal at

Power, indicates that a sensitivity study is performed to evaluate the effects of power level, reactivity insertion rate and reactivity feedback on the results of the transient. The licensee is requested to list all the cases (specifying initial conditions of power level, reactivity insertion rate and reactivity feedback representing the BOC and EOC cores) that are analyzed. Summarize the results of analyses and show that the limiting case is the full-power case at EOC core conditions with reactivity insertion rate of 27.6 pcm/sec as presented in Section 6.2.19*.

21. Section 6.2.20* indicates that the calculated minimum DNBR for the withdrawal of single full-length RCCA event is less than the safety limit. As a result, a total of one assembly (0.64 percent of total fuel rods) is predicted to fail. The licensee is requested to provide a figure showing the calculated DNBRs during the transient. Discuss the analytical methods, input parameters and assumptions used to determine the number of failed fuel rods, and show that the methods used for the analysis are acceptable and the input parameters and assumptions are conservative with respect to the fuel failure calculations.

22. Section 6.2.23* states that for the inadvertent boron dilution event, the analysis shows that there is adequate time for the operator to manually terminate the source of dilute flow during all modes of operation. However, no information is provided for the method used and the assumptions made in the analysis. The licensee is requested to confirm that (1) the method (especially, the boron mixing model applying to the condition without the reactor coolant pump running) used for the boron dilution analysis is acceptable, (2) the initial RCS water volumes and dilution flow rates assumed in the analysis for each mode of operation are conservative with respect to the calculated operator action time to terminate the diluted water flow, and (3) the staff's concern regarding nonconservative inputs for the deboration event analysis documented in NRC Information Notice 93-32 is satisfactorily addressed.

23. Section 6.2.26* submitted on December14, 2000, presents the results of analysis for the RCS inventory increase event resulting from inadvertent operation of the emergency core cooling system (ECCS). The event is initiated from inadvertent actuation of charging pumps. In the HNP plant, the ECCS contains the safety injection (SI) pumps, safety injection tanks (SITs) and charging pumps that inject water into the RCS. Inadvertent operations of any of the these ECC subsystems may increase RCS inventory. The licensee should expand the discussion of Section 6.2.26 to address the effects of the inadvertent operations of the SI pumps and SITs on the increased RCS inventory event.

24. Section 6.2.27*, CVCS [chemical and volume control system] Malfunction that Increases Reactor Coolant Inventory, states that the potential for water relief through the pressurizer through safety valves is addressed in Event 15.5.1 and the challenge to SAFDL is addressed in Event 15.4.6. The staff notes that the Event 15.5.1 is the inadvertent operation of the ECCS during power operation and Event 15.4.6 is the CVCS malfunction that results in a decrease in the boron concentration event. The referenced cases are caused by different initiators, need different safety systems to mitigate the consequences, may result in different system and thermal-hydraulic responses, and have different safety concerns. The licensee should provide a technical basis to justify that the increased reactor coolant inventory event due to CVCS malfunction is adequately represented by the analysis for Event 15.5.1 and Event 15.4.6, or provide the results of analysis for this event for the staff to review. Also, the

licensee states that for Modes 4 through 6, at least one pressurizer PORV (or vent) is available for pressure relief. The licensee should reference the TS for PORVs to satisfy the requirements specified in item (c)2(ii)(C) of 10 CFR 50.36.

25. Section 6.3.1* states that the major operator actions for SG tube rupture (SGTR) recovery provided in the licensee's EOPs Path-2 are explicitly modeled in the SG overfill analysis. Provide justification to show that the licensee's SGTR recovery procedures are acceptable for determining the operation actions and the associated action times.

26. Section 6.3.2* states that the block valve downsteam of the PORV is credited in the analysis to isolate the PORV from the SG with a ruptured tube. The licensee is requested to discuss the reliability of the block valves to function under the expected transient conditions and address the acceptability of the valves for the accident mitigation. Also, provide a TS LCO for the block valves to satisfy the requirements specified in item (c)2(ii)(C) of 10 CFR 50.36.

    As stated in Section 6.3.2*, the licensee determines that an operator can locally close the block valve for the PORV on the affected SG within 20 minutes following the SG PORV failure in the open position. The licensee is requested to discuss the method used to determine the action time for the operator to close the block valve and show that the method is acceptable and the proposed action time of 20 minutes is available.

27. The licensee's uprated power and SGR application will increase the operating power limit by 4.5 percent and change the SG heat capacity. The changes in the SG design and operating condition conditions may result in changes to the the setpoint of the low temperature overpressure protection (LTOP) system. The licensee is requested to provide an setpoint analysis for the LTOP system in accordance with guidance specified in Section II.B of SRP 5.2.2 and show that either the current LTOP setpoint remains valid or propose a new setpoint with an asssociated TS.

28. Page Enclosure 1-11 to a letter dated October 4, 2000, provides a basis to support the proposed TS change for the required water volume in the boric acid tank. It states that "Based on the analysis results, the minimum contained volume during shutdown specified for the boric acid tank in TS 3.1.2.5 is increased from 6650 to 7150." Discuss the referenced analysis used to draw the above conclusion on water volume in the boric acid tank and show that both the analytical method and results are acceptable.

29. Item G of Branch Technical Position (BTP) RSB 5.1 requires that a seismic Category I auxiliary feedwater supply be provided with sufficient inventory to permit operation at hot shutdown conditions for least 4 hours, followed by a cooldown to the condition permitting operation of the residual heat removal system. The auxiliary feedwater needed for the cooldown shall be based on the longest coldown time needed with either onsite or offsite power available and with the worst single failure. The licensee is requested to provide a discussion to address its compliance with the requirements specified in item G of BTP RSB 5-1. The discussion should include information to show that the analytical models and methods are acceptable; the assumptions used are consistent with the BTP RSB 5-1 and are conservative to maximize the required auxiliary feedwater supply; and the analyical results are bounded by the TS values for the auxiliary feedwater supply.

    Since the auxiliary feedwater supply is credited for event mitigation, a TS is required for the auxiliary feedwater system to specify the required water volume (item (c)2(ii)(C) of

10 CFR 50.36). The licensee indicates (on page Enclosure 1-25 to a letter dated October 4, 2000) that TS Bases 3.7.1.3 specifies the volume of 270,000 gallons for the condensate storage tank that is the primary source of supply for the auxiliary feedwater system. According to 10 CFR 50.36, the Bases for specifications are not part of the TS. Therefore, the staff determines that TS Bases 3.7.1.3 is not a TS and it alone is not adequate to satisfy the TS requirements for the auxiliary feedwater water supply.

* The section, table, and page numbers in the RAI refer to Enclosure 6 to the licensee's letter dated October 4, 2000.

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