June 26, 2002

Mr. J. W. Moyer, Vice President
Carolina Power & Light Company
H. B. Robinson Steam Electric Plant,
Unit No. 2
3581 West Entrance Road
Hartsville, South Carolina 29550

SUBJECT: H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2 - REQUEST FOR ADDITIONAL INFORMATION (RAI) ON AMENDMENT REQUEST REGARDING TECHNICAL SPECIFICATION CHANGES TO INCREASE AUTHORIZED POWER LEVEL (TAC NO. MB5106)

Dear Mr. Moyer:

By letter dated May 16, 2002, you submitted a request regarding technical specification changes to increase the authorized power level for the H. B. Robinson Steam Electric Plant, Unit No. 2. The NRC staff has reviewed the information that you provided in the subject amendment request, but needs additional information to complete the review.

This request for information was discussed with Mr. Chuck Baucom and your staff on June 18, 2002, and a mutually agreeable schedule of within 30 days of receipt of this RAI was established for your response. The RAI is enclosed.

In order to minimize the impact on the power uprate by the proposed alternate source term(AST) changes, you requested that the approval of the AST amendments (TAC Nos. MB 4632 and MB 5105) be separated from the power uprate submittal. The staff acceded to your proposal that you will submit a Supplement that will enable the power uprate be processed independent of the requested changes to the AST.

In your submittal you had requested approval date of October 7, 2002 , to be in time for implementing the modification during the upcoming refueling outage in October 2002. Due to the additional time now required to process the supplement and the review of your RAI responses, please note that your requested date will be impacted. However, the staff will make every effort to meet that requested date. We would appreciate any efforts on your part that may expedite the staff review in this regard.

If you have any questions, please contact me at (301) 415-1478.

Sincerely,

/RA/

Ram Subbaratnam, Project Manager, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-261

Enclosure: RAI

cc w/encl: See next page

In your submittal you had requested approval date of October 7, 2002 , to be in time for implementing the modification during the upcoming refueling outage in October 2002. Due to the additional time now required to process the supplement and the review of your RAI responses, please note that your requested date will be impacted. However, the staff will make every effort to meet that requested date. We would appreciate any efforts on your part that may expedite the staff review in this regard.

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REQUEST FOR ADDITIONAL INFORMATION

AMENDMENT REQUEST REGARDING TECHNICAL SPECIFICATION CHANGES TO

INCREASE AUTHORIZED POWER LEVEL

FOR

CAROLINA POWER & LIGHT COMPANY (CP&L)

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2 (HBRSEP2)

DOCKET NO. 50-261

1. The licensee should provide a discussion of allowed outage time for the Leading Edge Flow Meter (LEFM) CheckPlus™ system. This discussion should include the following:

   a. The length of time the plant can be operated at a power level above 2300 MWt if the LEFM CheckPlus™ system becomes unavailable.

   b. The actions needed to be taken to continue operation above 2300 MWt after the LEFM CheckPlus™ system becomes unavailable (i.e., if the LEFM can be returned to operation prior to the expiration of the time limit in Question 1.a).

   c. Identification of the Technical Specification (TS) Surveillance Requirement that governs the time and actions involved in returning the LEFM CheckPlus™ system to operation.

   d. The actions needed to be taken to increase the power level above 2300 MWt after reduction of power because of an inoperable LEFM CheckPlus™ system.

   e. The impact on power level of a single spool piece (or single LEFM CheckPlus™ channel) versus multiple spool pieces (or multiple LEFM CheckPlus™ channels) being unavailable.

2. The licensee should address the verification and validation of software.

Other questions (not directly related to NRC RIS 2002-03) are as follows:

3. In Attachment II, Section 3.2, page 12, the licensee stated that one spool piece will be installed in each of the feedwater headers that supply the steam generators. Robinson has two feedwater pumps and three steam generators (SGs). Therefore, it is not clear if two or three spool pieces will be installed. Please specify how many spool pieces are being installed.

4. In Attachment II, Section 3.2, page 13, the licensee refers to the currently installed feedwater temperature instrumentation. Please indicate what is the currently installed feedwater temperature instrumentation.

5. In Attachment II, Section 3.2.1.3, page 15, the licensee references Caldon Engineering Report ER-267, "Bounding Uncertainty Analysis for Thermal Power Determination at CP&L Robinson

Nuclear Power Station Using the LEFM Check Plus System," as the site-specific bounding uncertainty analysis that is the basis for the Robinson uncertainty values. Please submit ER-267 to the staff.

6. In Attachment II, Section 3.10.1.2, page 57, the licensee describes the change in Allowable Value for TS 3.3.2, “Engineered Safety Feature Actuation System Instrumentation,” Table 3.3.2-1, Function 1.e, "Safety Injection - Steam Line High Differential Pressure Between Steam Header and Steam Lines." This Allowable Value is being changed from a number with an upper bound to a number with both upper and lower bounds. Please indicate the rationale for the change from a number with an upper bound to a number with both upper and lower bounds.

7. Identify the loss-of-coolant accident (LOCA) and non-LOCA transient analyses of record that rely on the following isolation valves, reactor trip, and engineered safety features (ESF) for consequence mitigation. Provide the values for the valve isolation time, reactor trip, and ESF actuation system setpoints assumed in the applicable analyses and justify that the current analyses bound the cases with the revised values listed below.

   a. Page 37* indicates that the stroke time requirements specified in TS 3.7.3.1 for the main feedwater regulation valves and the main feedwater bypass valves are changed from $\leq$ 30 seconds to $\leq$ 20 seconds.

   b. Page 53* indicates that the values of T_{avg} in the reactor trip equations for the Overtemperature ΔT trip (TS Table 3.3.1-1, Note 1) and the Overpressure ΔT trip (TS Table 3.3.1-1, Note 2) are changed from 575.4 ^{o}F to 575.9 ^{o}F.

   c. Page 57* indicates that the allowable value in TS Table 3.3.2-1, Function 1.e for the “Steam Line High Differential Pressure Between Steam Header and Steam Lines - Safety Injection” is revised from $\leq$108.95 psig to an upper bound of $\leq$116.24 psig with an added lower bound of $\geq$ 83.76 psig.

      Also, page 55* states that “it will be necessary to revise calculations for the Steam/feedwater Flow Mismatch Trip to reflect the new nominal flow rates for feedwater and steam flow. The nominal trip setpoint is provided in TS Table 3.3.1-1.” Based on the statement, it appears to the staff that the trip setpoint needs to be changed. However, TS Table 3.3.1-1 is not changed to reflect the new trip setpoint. Provide clarification to the statement and the TS changes related to the Steam/Feedwater Flow Mismatch Trip.

8. Provide results of an anticipated transient without scram (ATWS) analysis demonstrating that the plant at power uprate conditions is within the bounds considered by the staff during the licensee’s documentation of compliance with the ATWS rule. If the licensee chooses to rely on the Westinghouse generic ATWS analyses to demonstrate the acceptance of the analytical results, the licensee is requested to provide a discussion of the ATWS analyses that are applicable to the specific plant and power uprate conditions, and justify that the assumptions for the applicable ATWS analyses are adequate as they relate to input parameters such as the initial power level, moderator temperature coefficient (MTC), pressurizer safety and relief valves capacity, RCS volume, SG pressure, auxiliary feedwater flow (AFW) rate and its actuation delay time, and the Setpoint for the ATWS mitigating system actuation circuitry system to actuate the AFW and trip the turbine. The submittal should include a discussion and applicable values of the unfavorable exposure time for the MTC assumed in the analyses.

9. Table 3.2-1 on page 83* lists uncertainties for six components used to calculate the total secondary calorimeter power measurement uncertainty. The six component uncertainties are represented by: A for the feedwater mass flow (LEFM), B for feedwater temperature, C for main steam pressure, D for main steam pressure (SG blowdown), E for blowdown flow, and F for core power correlation factor. Provide documents to show how each of the six component uncertainties (Items A through F) are determined and explain the differences of uncertainties contributed from items C, D, and E.

   The staff finds that in the equation used to calculate the total power measurement uncertainty, item A is not included and item H is not defined. It is also not clear why the items for the squares of C, D, E, and F in the equation are multiplied by 3 and Item B is not multiplied by 3 even though the licensee states that items B through E represents instrument uncertainties for each RCS loop (in a total of 3 loops) while F represents uncertainties from various heat gain and loss. The licensee is requested to clearly define each items used in the equation and confirm that the use of multiplier of 3 (instead of a divider of 3) for each item is correct and acceptable.

10. Westinghouse has issued three Nuclear Service Advisory Letters (NSALs), NSAL-02-3 and revision 1, 02-4 and 02-5, to document the problems with the Westinghouse-designed SG water level setpoint uncertainties. NSAL-02-3 and its revision, issued on February 15 and April 8, 2002, respectively, deal with the uncertainties caused by the mid-deck plate located between the upper and lower taps used for SG measurements and affect the low-low level trip setpoint (used in the analyses for events such as the feedwater line break, ATWS and steamline break). NSAL-02-4, issued on February 19, 2002, deals with the uncertainties created because the void contents of the two-phase mixture above the mid-deck plate were not reflected in the calculation and affect the high-high level trip setpoint. NSAL-02-5, issued on February 19, 2002, deals with the initial conditions assumed in the SG water level related safety analyses . The analyses may not be bounding because of velocity head effects or mid-deck plate pressure differential pressure which have resulted in significant increases in the control system uncertainties. Discuss how Robinson, Unit 2 accounts for all these uncertainties documented in these advisory letters in determining the SG water level setpoints, Also, discuss the effects of the water level uncertainties on the analyses of record for the LOCA and non-LOCA transients and the ATWS event, and verify that with consideration of all the water level uncertainties, the current analyses are still limiting.

11. The licensee is requested to discuss the methodology used in the calculation of the current vessel pressure-temperature curves (page 27*) and confirm that it adheres to the guidance in RG 1.190. Also, provide the results of calculations to show the change of the expected end-of-license value for RT_{PTS} from the current power level to the proposed power uprate conditions.

12. The definition of dose equivalent 131I should be defined using the effective dose conversion factor for 131I for inhalation taken from Table 2.1 of Federal Guidance Report 11 and not the thyroid dose conversion factor taken from NUREG/CR-6604. Please explain.

13. The appropriate dose limit for the release of the contents of the waste gas decay tank are the radiation dose limits for individual members of the public. When using total effective dose equivalent (TEDE) criteria, this is 0.1 rem TEDE and not 0.5 rem TEDE as proposed by the licensee. As a side note to this particular aspect, the calculation of the maximum allowable curie content of dose equivalent 133Xe in a waste gas decay tank should be based upon the 133Xe effective dose conversion factor for air submersion taken from Federal Guidance Report 12.

14. Provide fission product inventory in the fuel rod gap for each radionuclide of interest (noble gases and halogens) that is available for release after 8 and 56 hours after reactor shutdown to the water surrounding the failed fuel assembly. Also, provide assumed amounts of fission product activities (in curies) released to the environment from the containment and from the fuel handling building following the postulated design-basis fuel handling accident.

* Page number of Attachment II of the licensee's May 16, 2002, submittal.

cc:

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