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H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2  
DOCKET NO. 50-261/LICENSE NO. DPR-23

RESPONSE TO REQUEST FOR ADDITIONAL  
INFORMATION ON AMENDMENT REQUEST TO INCREASE  
AUTHORIZED REACTOR POWER LEVEL (TAC NO. MB5106)

Ladies and Gentlemen:

By letter dated May 16, 2002, Carolina Power and Light (CP&L) Company submitted a request for amendment to the Technical Specifications (TS) to increase the authorized reactor power level for the H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2. A previous response to an NRC request for additional information (RAI) was provided by letter dated July 25, 2002. A supplement to this license amendment request was provided by letter dated August 12, 2002.

Attachment I provides an Affirmation as required by 10 CFR 50.30(b). Attachment II provides the response to an NRC request for additional information that was provided by an electronic mail message dated July 29, 2002, and as part of an August 22, 2002, conference call between HBRSEP, Unit No. 2, and NRC staff personnel.

In accordance with 10 CFR 50.91(b), CP&L is providing the State of South Carolina with a copy of this response.

The response to the NRC RAI provide additional information that does not affect the basis or justification for the proposed TS change, including the evaluation of No Significant Hazards Consideration provided within the August 12, 2002, submittal.

If you have any questions concerning this matter, please contact Mr. C. T. Baucom.

Sincerely,

B. L. Fletcher III  
Manager - Regulatory Affairs

Robinson Nuclear Plant  
3581 West Entrance Road  
Hartsville, SC 29550

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CAC/cac


Attachments:

- I. Affirmation
  - II. Response To Request For Additional Information On Amendment Request to Increase Authorized Reactor Power Level
- c: Mr. L. A. Reyes, NRC, Region II  
Mr. H. J. Porter, Director, Division of Radioactive Waste Management (SC)  
Mr. R. M. Gandy, Division of Radioactive Waste Management (SC)  
Mr. R. Subbaratnam, NRC, NRR  
NRC Resident Inspector, HBRSEP  
Attorney General (SC)

**AFFIRMATION**

The information contained in letter RNP-RA/02-0133 is true and correct to the best of my information, knowledge and belief; and the sources of my information are officers, employees, contractors, and agents of Carolina Power and Light Company. I declare under penalty of perjury that the foregoing is true and correct.

Executed on: SEP 06 2002

  
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J.W. Moyer  
Vice President, HBRSEP, Unit No. 2

## **H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2**

### **RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION ON AMENDMENT REQUEST TO INCREASE AUTHORIZED REACTOR POWER LEVEL**

In an electronic mail message dated July 29, 2002, and in a telephone conference on August 22, 2002, NRC staff personnel provided further requests for additional information (RAI) regarding Carolina Power and Light (CP&L) Company's license amendment request to increase the authorized reactor power level, submitted by letter dated May 16, 2002. Responses to the RAI are provided below.

#### **NRC Request:**

The supplement to this license amendment request dated August 12, 2002, indicates that the main steam line break (MSLB) and steam generator tube rupture (SGTR) radiological analyses are not dependent on the power level of the H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2. Provide additional information on the effects of the reactor power level change on the break flow rates for these accidents. This response should also address the effects of changes to any other operating parameters associated with the operation at the increased power level.

#### **CP&L Response:**

For SGTR, the analysis of record (AOR) initial break flow rate into the affected steam generator and steam release flow rate from the affected steam generator are based on an initial core power level of 2346 MWt. This is 102% of the current authorized reactor power level and bounds the proposed increase in authorized reactor power level to 2339 MWt.

The reactor coolant system (RCS) temperatures used in the AOR are based on the full-power average RCS temperature of  $575.4 \pm 4$  degrees F. The proposed change in average RCS temperature for power uprate is +0.5 degrees F. Therefore, the uprated RCS average temperature remains bounded by the AOR. The RCS system pressure is not being changed for power uprate.

The AOR is based on a secondary system initial pressure of 800 psia. This remains conservative relative to the projected small change in the nominal secondary steam pressure, which is expected to be approximately a 3 psi reduction in nominal secondary steam pressure (approximately 809 psia for current power conditions compared to approximately 806 psia for the uprated conditions).

For MSLB, the AOR is described in the original licensing basis correspondence for HBRSEP, Unit No. 2. This information is also reported in the current HBRSEP, Unit No. 2, Updated

Final Safety Analysis Report (UFSAR) Section 15.1.5.4. The analysis basis states that in the event of a steam pipe rupture, the water boiled off in the affected steam generator, plus the steam released from the two unaffected steam generators during plant cooldown, are the sources for activity release. For the purpose of the radiological assessment, it was conservatively assumed that the affected steam generator pressure drops almost instantaneously to atmospheric pressure and primary coolant pressure remains essentially 2250 psia for the first two hours. As previously stated, the RCS system pressure is not being changed for power uprate. The total break flow from the affected loop is driven by the energy of the fluids (steam and liquid) in that loop prior to steam line isolation, plus energy associated with any primary to secondary heat transfer during the cooldown, and the energy in any primary to secondary leakage. The proposed changes in primary and secondary thermal hydraulic conditions (such as pressure and temperature) due to power uprate are small, as described for the SGTR event; therefore, the total break flow is not significantly affected.

The challenge to the fuel by departure from nucleate boiling ratio (DNBR) and fuel centerline melt is greatest, and therefore analyzed, at the zero power condition. The fuel integrity acceptance criteria are met for both the current and the Cycle 22 power uprate conditions. Attachment II to the May 16, 2002, submittal provides the results of the evaluation of the MSLB AOR for the fuel integrity acceptance criteria. That evaluation concludes that the MSLB analyses are not impacted by the power uprate and the associated changes to the main feedwater system.

Therefore, as stated in the August 12, 2002, letter and in accordance with the guidance contained RIS 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," the requested power uprate conditions continue to be bounded by the AOR for the radiological consequences of the SGTR and MSLB events.

**NRC Request:**

In the responses to questions on pages 14 and 17 of Attachment II to the HBRSEP, Unit No. 2, RAI response letter dated July 25, 2002, describe in what manner data and the FERRET code were used to determine the projected fluence values.

**CP&L Response:**

The methodology for the surveillance capsule and power uprate projections uses the ENDF/B-VI transport cross-section libraries and Regulatory Guide (RG) 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence." The calculated results are compared with the plant-specific measurements only to demonstrate that the measurements and calculations agree to within 20%. The FERRET code was used to compare the measured flux/fluence values with the calculated values. This comparison was performed to demonstrate uncertainty estimates, and was not used to determine the projected fluences.

**NRC Request:**

In the response to a question on page 17 of Attachment II to the HBRSEP, Unit No. 2, RAI response letter dated July 25, 2002, provide an explanation for the reduction in projected end-of-license (EOL) fluences for uprated conditions.

**CP&L Response:**

There are two main reasons for the reduction in EOL fluences:

1. The current fluences were calculated using ENDF/B-IV cross-section libraries and, due to deficiencies that led to under-predictions, were adjusted using measured data. In addition, the fluence values were rounded-up as a conservative measure and used in previous reactor vessel integrity regulatory submittals for HBRSEP, Unit No. 2. The new uprate fluences are calculated values based on the guidance of RG 1.190 and are therefore consistent with NRC-approved methodology.
2. The fluence reductions in the 20-30 degree azimuth range are due to changes in the radial core power distributions subsequent to the completion of Cycle 15. The part length shield assemblies (PLSA) used in Cycles 10 through 15 were fresh (unirradiated) fuel assemblies. In the design of Cycles 16 through 21, high burn-up fuel was placed in these locations. This resulted in a reduction in the calculated neutron flux at the vessel wall where these assemblies affect fluence. The EOL fluences for uprated conditions, which are also contained in the Surveillance Capsule X analysis report (see HBRSEP, Unit No. 2, letter dated April 25, 2002), are based on evaluations through the completion of Cycle 21 and on projections that were based on the average exposure rates for Cycles 16 through 21, appropriately increased to include the effect of uprated power conditions.

**NRC Request:**

Provide a summary of the HBRSEP, Unit No. 2, steam generator operational assessment for any active degradation mechanisms under uprated conditions.

**CP&L Response:**

There are no active degradation mechanisms identified for the HBRSEP, Unit No. 2, steam generators for current operating conditions or for uprated conditions. As stated in the HBRSEP, Unit No. 2, submittal dated May 16, 2002, the integrity of the steam generator tubes under uprated power conditions will continue to be verified through periodic inspections and measurements performed in accordance with the HBRSEP, Unit No. 2, Steam Generator Program. This program meets the requirements of Nuclear Energy Institute (NEI) 97-06, "Steam Generator Program Guidelines," and provides a balance of prevention, inspection, evaluation and repair, and leakage monitoring. The NEI 97-06 guideline refers to Electric Power Research Institute (EPRI) guidelines for the detailed development of these programmatic attributes.

**NRC Request:**

In the response to the NRC question number 8, on page 7 of Attachment II to the July 25, 2002, submittal, the licensee indicates that the reactor protection system low-low steam generator level setpoint is set at 16% and the ATWS mitigating system actuating circuitry (AMSAC) steam generator level setpoint is 11%. Neither of these setpoints are being changed as a result of proposed increase in authorized reactor power level. The licensee is requested to confirm that the effects of the Westinghouse Nuclear Safety Advisory Letter (NSAL) -02-3 issue are included in the low-low steam generator water level reactor trip and AMSAC actuation setpoints.

**CP&L Response:**

The effects of NSAL-02-3 have been evaluated for HBRSEP, Unit No. 2, and do not impact the low-low steam generator water level reactor trip and AMSAC actuation setpoints for current operating conditions, or for operation at the proposed uprated conditions.

**NRC Request:**

The NRC staff requested, in question number 10, on pages 12 and 13 of Attachment II to the July 25, 2002, submittal, the licensee address the effects of the steam generator level uncertainties, identified in NSAL -02-3, 02-4, and 02-5, on the analysis of record. In response, the licensee indicates that for the power uprate conditions, additional evaluation will be performed to verify that the NSAL issues are properly accounted for uprated power conditions. In order for the staff to continue the power uprate application, the staff requests the licensee to provide its results of the analysis addressing the steam generator level issues identified in the three NSAL letters for the power uprate conditions.

**CP&L Response:**

The evaluations of the steam generator level issues identified in the three NSAL letters conclude that no additional plant changes are required for the power uprate conditions. Therefore, the HBRSEP, Unit No. 2, evaluations of the NSAL issues have been completed and properly account for uprated power conditions. Additionally, the engineering change that was previously identified in Attachment II to the July 25, 2002, submittal, to implement the setpoint change identified in the evaluation of NSAL-02-4, is no longer needed and has been cancelled based on the results of the evaluation of this NSAL.