CP&L
A Progress Energy Company

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

SUPPLEMENT TO AMENDMENT REQUEST REGARDING
INCREASE OF AUTHORIZED REACTOR POWER LEVEL (TAC NO. MB5106)

Ladies and Gentlemen:

By letter May 16, 2002, Carolina Power and Light (CP&L) Company submitted a license amendment request for an increase in the authorized reactor power for H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2. As originally submitted, the justification for the power uprate relied upon approval of another license amendment request that pertains to the full implementation of an alternative source term (AST). During a conference call with the NRC Staff on June 13, 2002, CP&L was informed that, due to resource limitations, the NRC review of the AST amendment request on a schedule that would support approval of the proposed power uprate by the end of the next refueling outage (RO-21), scheduled to begin on October 12, 2002, would be unlikely.

It was concluded that use of provisions described in Section II of Regulatory Issue Summary (RIS) 2001-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," would allow timely staff review of the proposed measurement uncertainty recapture power uprate license amendment request, without reliance on the AST license amendment.

Attachment I provides an affirmation as required by 10 CFR 50.30(b).

Attachment II contains additional information pertaining to the radiological accident analyses that were previously intended to be covered under the approval of the AST license amendment request. This additional information affects the "No Significant Hazards Consideration" evaluation provided with the May 16, 2002, submittal. Therefore, Attachment II also provides a revised "No Significant Hazards Consideration."

Robinson Nuclear Plant
3581 West Entrance Road
Hartsville, SC 29550

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Attachment III contains a response to a request for additional information regarding the Technical Specifications changes associated with the power uprate that was provided to CP&L by a letter dated July 29, 2002.

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

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

Sincerely,

B. L. Fletcher III
Manager - Regulatory Affairs

CAC/cac

Attachments:
I. Affirmation
II. Revised Evaluation of Radiological Consequences for Power Uprate
III. Response to Request for Additional Information

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/02-0115 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: AUG 1 2 2002

JWMoyer

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

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

REVISED EVALUATION OF RADIOLOGICAL CONSEQUENCES FOR POWER UPRATE

Carolina Power and Light (CP&L) Company submitted a request for an amendment to the Facility Operating License, including the Appendix A Technical Specifications (TS), for the H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2, by letter dated May 16, 2002. The proposed amendment would increase the authorized reactor core power level from 2300 MWt to 2339 MWt (approximately 1.7 percent). The justification for the power uprate, as originally submitted, partially depended upon discussions contained in a May 10, 2002, submittal requesting review and approval of a full implementation of an alternative source term (AST) for HBRSEP, Unit No. 2. The reanalysis of the dose consequences of accidents discussed in the May 10, 2002, submittal was performed in support of operation at an uprated reactor core power level.

In discussions with the NRC staff, it was determined that review and approval of the power increase license amendment prior to the upcoming refueling outage, which is scheduled to start on October 12, 2002, would be facilitated by modifying the May 16, 2002, power uprate request to remove the reliance on approval of the AST license amendment request. This revision provides an evaluation of the radiological accident analyses for the accidents that had relied upon the AST license amendment request. The affected accident analyses are the Loss of Coolant Accident (LOCA), Main Steam Line Break (MSLB), Steam Generator Tube Rupture (SGTR), Single Rod Cluster Control Assembly (RCCA) Withdrawal, Radioactive Waste Gas System Leak or Failure, and Reactor Coolant Pump Shaft Seizure (Locked Rotor) Accident.

An evaluation has been completed that concluded operation of HBRSEP, Unit No. 2, at the proposed 2339 MWt is bounded for approximately 95% of Cycle 22 (approximately 504 effective full power days [EFPD]). This evaluation was based on establishing a LOCA analysis of record (AOR) burnup limit for Cycle 22 that accounts for operation at the proposed 2339 MWt reactor power level. Therefore, the existing AOR for the LOCA, MSLB, SGTR, Single RCCA Withdrawal, Radioactive Waste Gas System Leak or Failure, and Reactor Coolant Pump Shaft Seizure (Locked Rotor) radiological accident analyses will bound operation at the proposed 2339 MWt for approximately 504 EFPD during Cycle 22.

Section II of Regulatory Issue Summary (RIS) 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," states that a matrix should be used to identify the information for each accident or transient analysis for which the existing analyses of record bound plant operation at the proposed uprated power level. The matrix for the analyses (LOCA, MSLB, SGTR, Single RCCA Withdrawal, and Reactor Coolant Pump

Shaft Seizure [Locked Rotor]) that were previously intended to be bounded by approval of the AST license amendment request is provided below:

Matrix for Affected Radiological Consequence Accident Analyses

Accident/Transient	UFSAR Section	Validity of Bounding Event Determination	Assumed Power Level (% of 2300 MWt)	NRC Approval
Radiological Consequences of a Loss of Coolant Accident	15.6.5.5	Remains Valid	102%	Method of Ref. 1 Approved by Ref. 2
Radiological Consequences of a Steam Generator Tube Rupture Accident	15.6.3.2	Remains Valid	Not used as an input to the analysis	Method of Ref. 1 Approved by Ref. 2
Radiological Consequences of a Main Steam Line Break Event	15.1.5.4	Remains Valid	Not used as an input to the analysis	Analysis Approved by Ref. 3
Single RCCA Withdrawal	15.4.3.1	Remains Valid	102%	Method of Ref. 1 Approved by Ref. 2
Radioactive Waste Gas System Leak or Failure	15.7.1	Remains Valid	Not used as an input to the analysis	Analysis Approved by Ref. 3
Reactor Coolant Pump Shaft Seizure (Locked Rotor)	15.3.2	Remains Valid	102%	Method of Ref. 1 Approved by Ref. 2

References:

1. The FRA-ANP generic dose analysis methodology (XN-NF-719(P), Rev 2, Assessment of Potential Radiological Consequences for High Exposure Fuel, Exxon Nuclear Company, October 1986).
2. NRC SER for Northern States Power Company, Prairie Island Units 1 and 2, Extended Fuel Burnup Reload Methodology, September 27, 1983.
3. Atomic Energy Commission SER, In the Matter of Carolina Power and Light Company, H. B. Robinson, Unit No. 2, Docket No. 50-261, May 18, 1970.

As stated previously, an evaluation has been completed that determines the current AOR for these accidents support operation at the uprated power level for approximately 95% of Cycle 22 (approximately 504 EFPD). Operation beyond 95% (504 EFPD) of Cycle 22 at the uprated power level is expected to be based on subsequent NRC Staff approval and HBRSEP, Unit No. 2, implementation of the AST analyses provided in the May 10, 2002, submittal.

The "No Significant Hazards Consideration" evaluation has also been revised, as shown below, to remove the reference to the analyses associated with the AST license amendment request.

REVISED NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Carolina Power and Light (CP&L) Company is proposing changes to the Facility Operating License (OL) for the H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2, including the Appendix A, Technical Specifications (TS). The following change is requested.

- Revision of the maximum reactor core power level stated in OL paragraph 3.A, and the TS 1.1 definition of "RATED THERMAL POWER (RTP)," from 2300 MWt to 2339 MWt.
- Revision of the reactor core safety limits (SLs) curve in TS 2.1.1, "Reactor Core SLs."
- Revision of the reference T_{avg} value in TS 3.3.1, "Reactor Protection System (RPS) Instrumentation."
- Revision of the Allowable Value for the "Steam Line High Differential Pressure Between Steam Header and Steam Lines" function in TS 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation."
- Revision of the RCS pressure-temperature limit curves in TS 3.4.3, "RCS Pressure and Temperature (P/T) Limits."
- Revision of the Required Actions in TS 3.7.1, "Main Steam Safety Valves (MSSVs)."
- Revision of the Main Feedwater Regulating Valve (MFRV) and Bypass Valve stroke time Surveillance Requirements in TS 3.7.3, "Main Feedwater Isolation Valves, (MFIVs), Main Feedwater Regulation Valves (MFRVs), and Bypass Valves."

An evaluation of the proposed change has been performed in accordance with 10 CFR 50.91(a)(1) regarding no significant hazards considerations using the standards in 10 CFR 50.92(c). A discussion of these standards as they relate to this amendment request follows:

1. The Proposed Change Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated.

   The proposed change described above does not involve a significant increase in the probability of an accident previously evaluated based on the results of comprehensive analytical efforts that were performed to demonstrate the acceptability of the proposed power uprate changes.

   An evaluation has been performed that identified the systems and components that could be affected by these proposed changes. The evaluation determined that these systems and components will function as designed and that performance requirements remain acceptable.

The primary loop components (reactor vessel, reactor internals, control rod drive mechanisms (CRDMs), loop piping and supports, reactor coolant pumps, steam generators and pressurizer) will continue to comply with their applicable structural limits and will continue to perform their intended design functions. Thus, there is no increase in the probability of a structural failure of these components leading to an accident.

The Leak-Before-Break analysis conclusions remain valid and the breaks previously exempted from structural considerations remain unchanged.

Systems included within the scope of the Nuclear Steam Supply System (NSSS) will continue to perform their intended design functions during normal and accident conditions. Additionally, NSSS components will continue to comply with applicable structural limits and will continue to perform their intended design functions. Thus, there is no increase in the probability of a structural failure of these components.

The NSSS/Balance of Plant interface systems will continue to perform their intended design functions. The MSSVs will provide adequate relief capacity to maintain the Main Steam System within design limits. The maximum feedwater flow rate and the isolation time for the MFRVs and Bypass Valves will continue to ensure that the analyzed containment pressure during postulated accidents remains below the allowable limit.

The current loss-of-coolant (LOCA) hydraulic analyses remain bounding.

The reduction in power measurement uncertainty achieved through the use of the Caldon Leading Edge Flow Meter (LEFM) Check-Plus™ system allows for certain safety analyses to continue to be used, without modification, at the 2346 MWt power level (102 percent of 2300 MWt). Other safety analyses performed at a nominal power level of 2300 MWt have been either re-performed or re-evaluated to support the 2339 MWt power level, and continue to meet their applicable acceptance criteria. Some existing safety analyses had been previously performed at a power level greater than or equal to 2346 MWt, and thus continue to bound the 2339 MWt power level.

The proposed changes to the RCS pressure-temperature limit curves impose a conservative projection of the increase in neutron fluence associated with the power uprate. This projection will ensure that the requirements of 10 CFR 50, Appendix G, "Fracture Toughness Requirements," will continue to be met following the proposed power uprate. The design basis events that were protected against by these limits have not changed, therefore, the probability of an accident previously evaluated is not increased.

Based on the foregoing, it is concluded that this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The Proposed Change Does Not Create the Possibility of a New or Different Kind of Accident From Any Previously Evaluated.

   The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated because no new accident scenarios, failure mechanisms, or single failures are introduced as a result of the proposed power uprate changes. Systems, structures, and components previously required for the mitigation of an event remain capable of fulfilling their intended design functions. The proposed changes have no adverse effects on any safety-related system or component, and do not challenge the performance or integrity of any safety-related system.

3. The Proposed Change Does Not Involve a Significant Reduction in the Margin of Safety.

   Extensive analyses of the primary fission product barriers conducted in support of the proposed power uprate have concluded that relevant design criteria remain satisfied, both from the standpoint of the integrity of the primary fission product barrier and compliance with regulatory acceptance criteria. As appropriate, evaluations have been performed using methods that have either been reviewed and approved by the Nuclear Regulatory Commission (NRC), or that are in compliance with applicable regulatory review guidance and standards.

   Therefore, this change does not involve a significant reduction in a margin of safety.

Based on the above discussion, CP&L has determined that the requested change does not involve a significant hazards consideration.

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

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

NRC Request:

Since the effects of flow-accelerated corrosion (FAC) on degradation of carbon steel components are plant-specific, the values of the parameters affecting FAC, i.e., velocity and temperature changes, must be included. In addition, the corresponding changes in components' wear rates due to FAC before and after the power uprate must be provided.

In Section 4.6.5, "Flow Accelerated Corrosion Program," on page 93 of Attachment II to the request, the licensee states the following:

> An evaluation of plant piping systems identified a number of feedwater heater components that may exhibit susceptibility to FAC under power uprate operating conditions.... In accordance with the provisions of the FAC Program, the projected power uprate operating conditions (i.e., flow and thermodynamic states) for these components are updated in the CHECWORKS model, as appropriate, and results of these models are factored into future pipe inspection and replacement plans.

The staff requests that you provide the predicted change of wear rates calculated by the revised CHECWORKS code for the components most susceptible to flow-accelerated corrosion. Specifically, the staff requests that the information be provided in a detailed table as illustrated below.

System	**Description**	**% Change in Predicted Wear Rate**	**Change in Predicted Wear Rate, mils / year**
FW	Feedwater (FW) to FW Pump to High Pressure FW Heater	+ 0.003 %	+ 0.02

CP&L Response:

The requested information is provided in a detailed table, as follows:

System	Description	% Change in Predicted Wear Rate		Change in Predicted Wear Rate (mils/year)	
		Average	Max	Average	Max
Steam Generator Blowdown	All trains, from steam generators to condenser	-0.15%	0.00%	-0.002	0.00
Extraction Steam	High pressure to 6th point heaters	+0.11%	+0.28%	+0.02	+0.16
Extraction Steam	High pressure to 5th point heaters	+0.12%	+0.14%	+0.04	+0.52
Extraction Steam	Low pressure to 3rd point heaters	-0.18%	0.00%	-0.01	0.00
Feedwater (FW)	FW Pump to 6th point heaters	0.00%	0.00%	0.00	0.00
Feedwater	FW from 6th point heaters to supply header	0.00%	0.00%	0.00	0.00
Feedwater	FW from supply header to steam generators (all trains)	0.00%	0.00%	0.00	0.00
Condensate	Condensate Pumps to 1st point heaters	-0.95%	0.00%	-0.013	0.00
Condensate	1st point to 2nd point heaters (both trains)	-0.94%	-0.90%	-0.015	-0.01
Condensate	2nd point to 3rd point heaters (both trains)	-0.92%	0.00%	-0.009	0.00
Condensate	3rd point to 4th point heaters (both trains)	-0.91%	0.00%	-0.009	0.00
Condensate	4th point to 5th point heaters (both trains)	-0.90%	0.00%	-0.013	0.00
Condensate	5th point heaters to FW pump suction header (both trains)	-0.58%	0.00%	-0.007	0.00
Condensate	FW pump suction piping	0.00%	0.00%	0.00	0.00
Heater Drains	6th point heater drains to drain tanks (both trains)	+0.46%	+2.17%	+0.004	+0.011
Heater Drains	4th point heater drains to 3rd point heaters (both trains)	-0.21%	0.00%	-0.001	0.00
Heater Drains	3rd point heater drains to 2nd point heaters (both trains)	-0.14%	+1.91%	0.00	+0.020
Heater Drains	2nd point heater drains to 1st point heaters (both trains)	-0.04%	+1.21%	0.00	+0.015
Heater Drains	Moisture separator drains to heater drain tanks (all trains)	+0.57%	+3.57%	+0.008	+0.023
Heater Drains	Reheater drains to 6th point heaters (all trains)	+0.43%	+2.04%	+0.005	+0.084
Heater Drains	Reheater scavenging steam to 6th point heaters (all trains)	+0.45%	+0.62%	+0.005	+0.034
Heater Drains	Moisture pre-separator drains to heater drain tank (all trains)	+0.57%	+2.27%	+0.004	+0.032
Heater Drains	Heater drain pump suction from heater drain tanks	-5.86%	+4.35%	-0.069	+0.087
Heater Drains	Heater drain pump discharge to FW pump suction header (both trains)	-6.98%	0.00%	-0.078	0.00
Heater Drains	Heater drain pump recirculation to heater drain tank (both trains)	-7.09%	0.00%	-0.044	0.00