February 7, 2005

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 - ISSUANCE OF AN AMENDMENT ON REACTOR COOLANT SYSTEM PRESSURE AND TEMPERATURE LIMITS (TAC NO. MC4160)

Dear Mr. Moyer:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 202 to Renewed Facility Operating License No. DPR-23 for the H. B. Robinson Steam Electric Plant, Unit No. 2 (HBRSEP2). This amendment changes the HBRSEP2 Technical Specifications (TS) in response to your request dated August 19, 2004, as supplemented December 2, 2004.

The amendment revises the reactor coolant system (RCS) pressure and temperature limits by replacing TS Section 3.4.3, "RCS Pressure and Temperature (P/T) Limits," Figures 3.4.3-1 and 3.4.3-2, with figures that are applicable up to 35 effective full-power years

A copy of the related Safety Evaluation is enclosed. Notice of Issuance will be included in the Commission's bi-weekly Federal Register notice.

Sincerely,

/RA/

Chandu P. Patel, Project Manager, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-261

Enclosures:
1. Amendment No. 202 to DPR-23
2. Safety Evaluation

cc w/enclosures:
See next page

February 7, 2005

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

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SUBJECT: H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2 - ISSUANCE OF AN AMENDMENT ON REACTOR COOLANT SYSTEM PRESSURE AND TEMPERATURE LIMITS (TAC NO. MC4160)

Dated: February 7, 2005

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CAROLINA POWER & LIGHT COMPANY

DOCKET NO. 50-261

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

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 202
Renewed License No. DPR-23

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Carolina Power & Light Company (the licensee), dated August 19, 2004, as supplemented December 2, 2004, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications, as indicated in the attachment to this license amendment; and paragraph 3.B. of Renewed Facility Operating License No. DPR-23 is hereby amended to read as follows:

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 202, are hereby incorporated in the license. Carolina Power & Light Company shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Michael L. Marshall, Chief, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 7, 2005

ATTACHMENT TO LICENSE AMENDMENT NO. 202

RENEWED FACILITY OPERATING LICENSE NO. DPR-23

DOCKET NO. 50-261

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove Pages	Insert Pages
3.4-7	3.4-7
3.4-8	3.4-8

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 202 TO

RENEWED FACILITY OPERATING LICENSE NO. DPR-23

CAROLINA POWER & LIGHT COMPANY

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

DOCKET NO. 50-261

1.0 INTRODUCTION

By letter dated August 19, 2004, as supplemented December 2, 2004, the Carolina Power & Light Company (licensee) submitted a request for changes to the H. B. Robinson Steam Electric Plant, Unit No. 2 (HBRSEP2), Technical Specifications (TS). The requested changes would revise the reactor coolant system (RCS) pressure and temperature limits by replacing TS section 3.4.3, "RCS Pressure and Temperature (P/T) Limits," Figures 3.4.3-1 and 3.4.3-2, with figures that are applicable up to 35 effective full-power years (EFPY).

The calculations of the revised P-T limit curves are delineated in WCAP-15827, "H.B. Robinson Unit 2 Heatup and Cooldown Limit Curves for Normal Operation," dated March 2003. The proposed P-T limits were based, in part, on the use of certain elements of the American Society of Mechanical Engineers (ASME) Code Case N-641, "Alternative Pressure-Temperature Relationship and Low Temperature Overpressure Protection System Requirements." In addition, as a result of using this P-T limit methodology, the proposed P-T limits are less restrictive than the current P-T limits at lower temperatures. Therefore, the licensee requested no change to the current low temperature overpressure protection (LTOP) system limits.

The December 2, 2004, letter provided clarifying information only that did not change the initial proposed no significant hazards consideration determination or expand the scope of the initial application.

2.0 REGULATORY EVALUATION

Title 10 of Code of Federal Regulations (10 CFR) Part 50.60(a) states:

> Except as provided in paragraph (b) of this section, all light-water nuclear power reactors, other than reactor facilities for which the certifications required under §50.82(a)(1) have been submitted, must meet the fracture toughness and material surveillance program requirements for the reactor coolant pressure boundary set forth in appendices G and H to this part.

Appendix H to 10 CFR Part 50, “Reactor Vessel Material Surveillance Program Requirements,” establishes requirements related to facility reactor pressure vessel (RPV) material surveillance programs.

Regulatory Guide (RG) 1.99, Revision 2, “Radiation Embrittlement of Reactor Vessel Materials,” contains methodologies for determining the increase in transition temperature resulting from neutron radiation.

Appendix G to 10 CFR Part 50, “Fracture Toughness Requirements,” requires that facility P-T limit curves for the RPV be at least as conservative as those obtained by applying the methodology of Appendix G to Section XI of the ASME Code. The most recent version of Appendix G to Section XI of the ASME Code, which has been endorsed in 10 CFR 50.55a and therefore included by reference in 10 CFR Part 50, Appendix G, is the 2001 Edition through the 2003 Addenda of the ASME Code. This edition of Appendix G to Section XI continues to incorporate the provisions of ASME Code Cases N-588 and N-640. Additionally, Appendix G to 10 CFR Part 50 imposes minimum head flange temperatures when system pressure is at or above 20 percent of the preservice hydrostatic test pressure.

Generic Letter (GL) 92-01, Revision 1, requested that licensees submit the RPV data for their plants to the NRC staff for review, and GL 92-01, Revision 1, Supplement 1, requested that licensees provide and assess data from other licensees that could affect their RPV integrity evaluations.

NUREG-0800, “Standard Review Plan,” Section 5.3.2, “Pressure-Temperature Limits,” provides guidance on using these regulations and documents in the NRC staff’s review. Additionally, Section 5.3.2 provides guidance to the NRC staff on performing check calculations of the licensee’s submittal.

The licensee indicated in its submittal that Appendix G to the 1996 Edition of the ASME Code was used in generating the heatup and cooldown curves. In addition, two elements of Code Case N-641 were considered: (1) the postulation of a circumferential flaw for the limiting circumferential weld in the RPV, and (2) the use of the plane-strain fracture toughness (K_{IC}) instead of the crack arrest fracture toughness (K_{Ia}) in the P-T limit calculations. It should be mentioned that the first element used to be the subject of Code Case N-588, and the second element used to be the subject of Code Case N-640. Consequently, the licensee’s P-T limit methodology is equivalent to Appendix G to Section XI of the 2001 Edition through the 2003 Addenda of the ASME Code endorsed in 10 CFR 50.55a.

Section 50.61 of 10 CFR 50 defines RT_{NDT} as the reference temperature for a reactor vessel material under any conditions. It further specifies that the RT_{NDT} must account for the effects of neutron radiation for materials in the beltline region. RT_{PTS} is defined as the reference temperature, RT_{NDT}, evaluated for the end-of-life fluence for each of the vessel beltline materials using the procedures in this section. The pressurized thermal shock (PTS) screening criterion is defined as the value of RT_{PTS} for the vessel beltline material above which the plant cannot continue to operate without justification. The PTS screening criterion is 270EF for plates, forgings, and axial weld materials, and 300EF for circumferential weld materials. This section also specifies how the values of RT_{NDT} and RT_{PTS} are calculated.

RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," March 2001, provides the methods acceptable to the NRC staff for determining the pressure vessel fluence. The calculated fluence is used to determine RT_{NDT} and RT_{PTS} as specified in 10 CFR 50.61. The RG allows the use of the deterministic discrete ordinates method and the Monte Carlo transport method. The RG explains the proper conditions for the use of each calculation method. The RG also states that the uncertainty of the fluence calculations must be 20 percent (1σ) or less if the fluence will be used to determine RT_{NDT} and RT_{PTS} for complying with 10 CFR 50.61 and Revision 2 of RG 1.99, "Radiation Embrittlement of Reactor Vessel Materials."

3.0 TECHNICAL EVALUATION

3.1 Licensee's Evaluation

The licensee submitted adjusted reference temperature (ART) values and P-T limit curves valid for up to 35 EFPY of facility operation. The licensee identified the limiting material for the HBRSEP2 RPV as upper shell plate W10201-1, fabricated from plate heat A6623-1 and upper to intermediate shell plate circumferential weld 10-273, fabricated from weld heat W5214. The circumferential weld is limiting for only a portion of the cooldown curves. The licensee calculated the ART values for the limiting material for both the one-quarter of the RPV wall thickness (1/4t) and three-quarter of the RPV wall thickness (3/4t) locations. The key parameters for the licensee's ART determination for each of these locations are from WCAP-15827 and are shown in the table below for HBRSEP2.

Applicable Curves	Limiting Material	Location	Initial RT_{NDT} (EF)	Fluence (n/cm^2)	Chemistry Factor [1] (EF)	ΔRT_{NDT} (EF)	Margin [2] (EF)	ART (EF)
Heatup and Part of Cooldown	Upper Shell Plate W10201-1	1/4 T	69	1.05×10^{19}	62.9	63.5	34 ($\sigma_I = 0$, $\sigma_\Delta = 17$)	167
Heatup and Part of Cooldown	Upper Shell Plate W10201-1	3/4 T	69	0.344×10^{19}	62.9	44.4	34 ($\sigma_I = 0$, $\sigma_\Delta = 17$)	147
The remaining Part of Cooldown	Circ. Weld 10-273	1/4 T	-56	1.05×10^{19}	230.2	232.5	65.5 ($\sigma_I = 17$, $\sigma_\Delta = 28$)	242
The remaining Part of Cooldown	Circ. Weld 10-273	3/4 T	-56	0.344×10^{19}	230.2	162.5	65.5 ($\sigma_I = 17$, $\sigma_\Delta = 28$)	172

[1] The chemistry factors were determined from the chemistry factor table using Regulatory Guide 1.99, Revision 2 Position 1.1.

[2] The margin term for each ART calculation was based on the establishment of initial material property uncertainty (σ_I) and shift in material property uncertainty (σ_Δ) consistent with the guidance in RG 1.99, Revision 2.

The WCAP documented detailed thermal and fracture mechanics evaluations to establish the proposed HBRSEP2 P-T limits. The numerical representation of the proposed P-T limits can be found in Appendix A to the WCAP, and the RPV temperatures at the inner wall, 1/4t, 3/4t, and the outer wall locations for various heatup and cooldown rates can be found in Appendix B to the WCAP. Based on the temperature distribution across the RPV wall, the thermal stresses, and, subsequently, the applied thermal stress intensity factors (K_{It}) at the tip of the postulated axial, and circumferential flaw at the 1/4t location for the 100EF cooldown transient and at the 3/4t location for the 100EF heatup transient can be derived. Based on these applied K_{It} and K_{IC} values (values derived from ASME Code curves) at the crack tips, the WCAP calculated the corresponding applied pressure stress intensity factors (K_{Ip}) at the tip of the postulated flaw at the 1/4t and 3/4t locations, and subsequently the pressure itself. The licensee stated that the proposed P-T limit methodology, as applied to RPV beltline materials, is in accordance with Appendix G of Section XI of the ASME Code.

3.2 Staff Evaluation

3.2.1 Surveillance Capsule X

The licensee provided a surveillance capsule report for capsule X. Capsule X was evaluated using the HBRSEP2 surveillance program that is based on ASTM E185-66, “Recommended Practice for Surveillance Tests on Structural Materials in Nuclear Reactors.” This surveillance program is used to analyze fast neutron fluence values (E>1.0 MeV) to the pressure vessel and surveillance capsule. The licensee stated that the computational methods comply with RG 1.190. The capsule dosimetry was measured using techniques complying with ASTM standards. The measured dosimeter activities were compared to the calculated values for each dosimeter type at each exposure location and for the capsule exposure history. The calculated-to-measured (C/M) dosimeter ratios showed no indication of bias or abnormal distribution and were found to be well within the 20-percent uncertainty acceptance criteria limit as specified in RG 1.190. The NRC staff finds the methodology and the results of surveillance capsule X analysis to be acceptable.

3.2.2 Vessel Fluence Values

The proposed pressure vessel peak fluence values for HBRSEP2 were calculated using a methodology that adheres to the guidance in RG 1.190 as shown in WCAP-15805, “Analysis of Capsule X from the Carolina Power & Light Company H. B. Robinson Unit 2 Reactor Vessel Radiation Surveillance Program,” Westinghouse Electric Company LLC, March 2002. For example, the calculations and dosimetry evaluations were based on the latest available nuclear cross-section data derived from ENDF/B-VI and the approximation used follows the guidance in RG 1.190. The NRC staff finds the methodology specified in WCAP-15805 satisfies the requirements of RG 1.190 and therefore is acceptable. The RT_{PTS} values at 35 EFPY calculated using accepted fluence values are lower than the screening criteria of 10 CFR 50.61. Therefore, these RT_{PTS} values are acceptable.

The licensee requested a TS change to revise the RCS P-T limits to increase applicability up to 35 EFPY. The NRC staff has verified the RT_{PTS} was calculated with NRC-approved methodology contained in RG 1.190 and meets the screening criteria of 10 CFR 50.61. Therefore, the NRC staff finds the proposed values of RT_{PTS} acceptable for operation up to 35 EFPY.

3.2.3 P-T Limits

The evaluation of the embrittlement of the RPV beltline materials relies on neutron fluence prediction acceptable to the NRC staff. Evaluation of RPV neutron fluence calculation is discussed above. Since the proposed P-T limits are less restrictive than the current P-T limits at lower temperatures, the licensee requested no change to the current LTOP system limits. The remaining issues of P-T limits are discussed below.

To evaluate the proposed P-T limits for HBRSEP2, the NRC staff performed an independent calculation of the ART values for the limiting material of the HBRSEP2 RPV using the methodology in RG 1.99, Revision 2. Based on these calculations, the NRC staff verified that the licensee's limiting materials for the RPVs are the upper shell plate W10201-1 and upper to intermediate shell plate circumferential weld 10-273. The NRC staff's ART values for the limiting materials at the 1/4t and 3/4t locations are calculated using materials information for HBRSEP2 in the NRC Reactor Vessel Integrity Database. The ART values calculated by the NRC staff agree with the licensee's calculated values.

The NRC staff then evaluated the licensee's P-T limit curves for acceptability by performing independent calculations using the methodology referenced in Appendix G to the ASME Code (as indicated by SRP 5.3.2) based on information submitted by the licensee. The licensee stated that the proposed P-T limits were based on the elements of Code Case N-641, which permits postulation of a circumferential flaw for the limiting circumferential weld in the RPV and the use of the ASME K_{IC} curve instead of the K_{Ia} curve for the RPV materials in the P-T limit calculations. As discussed in Section 2.0 of this Safety Evaluation, the 2001 Edition through the 2003 Addenda of the ASME Code, endorsed in 10 CFR 50.55a, continues to incorporate the provisions of ASME Code Cases N-588 and N-640. Therefore, the postulation of a circumferential flaw for the limiting circumferential weld and the use of the K_{IC} curve is now in accordance with the Code. Appendix G permits two approaches to calculate K_{It}: use of the bounding K_{It} formulas based on heatup and cooldown rates, and use of the K_{It} formulas based on the thermal stress distribution from a thermal model (e.g., a finite element model) for heatup and cooldown. The WCAP used the latter approach and provided the RPV coolant temperature, pressure, and the metal temperatures at the tip of the postulated flaw at the 1/4t location during cooldown and the 3/4t location during heatup. The WCAP reported that the limiting material of the HBRSEP2 RPV would change from the circumferential weld 10-273 to the upper shell plate W10201-1 during cooldown. The NRC staff questioned this change of limiting beltline material in a request for additional information (RAI). The licensee, in a letter dated December 2, 2004, responded to this RAI by providing detailed K_{It} and K_{Ip} calculations for the cooldown curve. Based on the WCAP information and the additional information provided to the staff's RAI, the NRC staff verified that the licensee's proposed P-T limit methodology is in accordance with Appendix G to Section XI of the ASME Code, and the proposed P-T limits satisfy the requirements of Appendix G to 10 CFR Part 50.

In addition to beltline materials, Appendix G to 10 CFR Part 50 also imposes a minimum temperature at the closure head flange based on the most limiting reference temperature for the flange material. Section IV.A.2 of Appendix G states that when the pressure exceeds 20 percent of the pre-service system hydrostatic test pressure, the temperature of the closure flange regions, which are highly stressed by the bolt preload, must exceed the reference temperature of the material in those regions by at least 120EF for core (not critical) and by 160EF for core critical under normal operation, and by 90EF for hydrostatic pressure tests and

leak tests. The WCAP reported that the most limiting reference temperature for the flange material is 60EF. Based on this, the NRC staff determined that the vertical segment (60EF + 120EF) of the heatup and cooldown curves for core not critical and the vertical segment (60EF + 160EF) of the heatup and cooldown curves for core critical satisfy the closure flange requirement of Appendix G to 10 CFR Part 50. Therefore, the licensee's proposed P-T limit curves are acceptable for operation of the HBRSEP2 RPV through 35 EFPY of operation.

Based on the above evaluations, the NRC staff concludes that the proposed P-T limits for the pressure test, core not critical, and core critical conditions satisfy the requirements in Appendix G to 10 CFR Part 50 and Appendix G to Section XI of the ASME Code. Therefore, the proposed P-T limits are approved for incorporation into the HBRSEP2 TS and shall be valid until 35 EFPY of facility operation.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the State of South Carolina official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (69 FR 57981). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: J. Hardy
S. Sheng

Date: February 7, 2005

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