UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

March 17, 2010

Mr. Eric McCartney, Vice President
H. B. Robinson Steam Electric Plant,
Unit No. 2
Carolina Power & Light Company
3581 West Entrance Road
Hartsville, South Carolina 29550-0790

SUBJECT: H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2 - ISSUANCE OF AMENDMENT REGARDING TECHNICAL SPECIFICATIONS CHANGES RELATED TO THE TURBINE TRIP INPUT TO THE REACTOR PROTECTION SYSTEM (TAC NO. ME1550)

Dear Mr. McCartney:

The Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 222 to Renewed Facility Operating License No. DPR-23 for the H.B. Robinson Steam Electric Plant, Unit No. 2, in response to your application dated June 19, 2009, as supplemented by letter dated October 20, 2009. The amendment consists of changes to Technical Specification Section 3.3.1, Reactor Protection System (RPS) Instrumentation.

The amendment changes TS 3.3.1 to revise the requirements related to the RPS interlock for the turbine trip input to the reactor protection system. Specifically, the change revises the requirement for the turbine trip input to the RPS above the P-7 (Low Power Reactor Trips Block) interlock to instead be required above the P-8 (Power Range Neutron Flux) interlock.

A copy of the related safety evaluation is also enclosed. The Notice of Issuance will be included in the NRC's next biweekly *Federal Register* notice.

Sincerely,

Tracy J. Orf, Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-261

Enclosures: 1. Amendment No. 222 to DPR-23
2. Safety Evaluation

cc w/enclosures: Distribution via Listserv

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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. 222
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 June 19, 2009, as supplemented by letter dated October 20, 2009, 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 Title 10 of the *Code of Federal Regulations* (10 CFR) Chapter I;

   B. The facility will operate in conformity with the application, as amended, 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. 222 are hereby incorporated in the license.

      The licensee 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 by the end of Refueling Outage 26.

FOR THE NUCLEAR REGULATORY COMMISSION

Douglas A. Broaddus, Acting Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to Renewed Facility Operating License No. DPR-23 and the Technical Specifications

Date of Issuance: March 17, 2010

ATTACHMENT TO LICENSE AMENDMENT NO. 222

RENEWED FACILITY OPERATING LICENSE NO. DPR-23

DOCKET NO. 50-261

Replace page 3 of Renewed Operating License No. DPR-23 with the attached page 3.

Replace the following pages of Appendix A of the 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 Page	Insert Page
3.3-5	3.3-5
3.3-16	3.3-16

neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

D. Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source, or special nuclear material without restriction to chemical or physical form for sample analysis or instrument and equipment calibration or associated with radioactive apparatus or components;

E. Pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by operation of the facility.

3. This renewed license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations: 10 CFR Part 20, Section 30.34 of 10 CFR Part 30, Section 40.41 of 10 CFR Part 40, Section 50.54 and 50.59 of 10 CFR Part 50, and Section 70.32 of 10 CFR Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

A. Maximum Power Level

The licensee is authorized to operate the facility at a steady state reactor core power level not in excess of 2339 megawatts thermal.

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 222 are hereby incorporated in the license.

The licensee shall operate the facility in accordance with the Technical Specifications.

(1) For Surveillance Requirements (SRs) that are new in Amendment 176 to Final Operating License DPR-23, the first performance is due at the end of the first surveillance interval that begins at implementation of Amendment 176. For SRs that existed prior to Amendment 176, including SRs with modified acceptance criteria and SRs whose frequency of performance is being extended, the first performance is due at the end of the first surveillance interval that begins on the date the Surveillance was last performed prior to implementation of Amendment 176.

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
M. One channel inoperable.	M.1	Place channel in trip.	6 hours
	OR		
	M.2	Reduce THERMAL POWER to < P-7.	12 hours
N. One Reactor Coolant Flow - Low (Single Loop) channel inoperable.	N.1	Place channel in trip.	6 hours
	OR		
	N.2	Reduce THERMAL POWER to < P-8.	10 hours
O. One Reactor Coolant Pump Breaker Position channel inoperable.	O.1	Restore channel to OPERABLE status.	6 hours
	OR		
	O.2	Reduce THERMAL POWER to < P-8.	10 hours
P. One Turbine Trip channel inoperable.	P.1	Place channel in trip.	6 hours
	OR		
	P.2	Reduce THERMAL POWER to < P-8.	10 hours

(continued)

Table 3.3.1-1 (page 4 of 7)
Reactor Protection System Instrumentation

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT (1)
14.	SG Water Level - Low	1,2	2 per SG	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	≥ 29.36%	30%
	Coincident with Steam Flow/Feedwater Flow Mismatch	1,2	2 per SG	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	≤ 7.01 E5 lbm/hr	6.4 E5 lbm/hr
15.	Turbine Trip						
	a. Low Auto Stop Oil Pressure	1(f)	3	P	SR 3.3.1.10 SR 3.3.1.15	≥ 40.87 psig	45 psig
	b. Turbine Stop Valve Closure	1(f)	2	P	SR 3.3.1.15	NA	NA
16.	Safety Injection (SI) Input from Engineered Safety Feature Actuation System (ESFAS)	1,2	2 trains	Q	SR 3.3.1.14	NA	NA

(continued)

(1) A channel is OPERABLE with an actual Trip Setpoint value found outside its calibration tolerance band provided the Trip Setpoint value is conservative with respect to its associated Allowable Value and the channel is re-adjusted to within the established calibration tolerance band of the Nominal Trip Setpoint.

(f) Above the P-8 (Power Range Neutron Flux) interlock.

UNITED STATES
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 222 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 June 19, 2009 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML091770309), as supplemented on October 20, 2009 (ADAMS Accession No. ML092990205), Carolina Power & Light Company (the licensee), now doing business as Progress Energy Carolinas, Inc., proposed changes to the Technical Specifications (TSs) for H. B. Robinson Steam Electric Plant (HBRSEP), Unit 2. The current TSs require a reactor trip when a turbine trip signal occurs and the reactor power is above the P-7 setpoint (at 10 percent of the rated thermal power (RTP)). The TS requirements allow the automatic reactor trip on a turbine trip function to be blocked when the reactor power decreases to the P-7 setpoint. The licensee proposed TS would increase the setpoint of the reactor trip on a turbine trip from the P-7 to the P-8 setpoint (at 40 percent of the RTP). The proposed changes would affect Function 15 of TS Table 3.3.1-1 and Condition P of TS 3.3.1, Reactor Protection System (RPS) Instrumentation.

Because most turbine trips occur at low power levels, the proposed TS would decrease unnecessary challenges to the reactor protection system and increase plant availability.

2.0 REGULATORY EVALUATION

The reactor trip actuated on a turbine trip signal is not assumed in transient and accident analyses since the turbine trip signal originated in the turbine building, which is in a non-seismically qualified area. However, the load rejection event was evaluated to address the Three Mile Island (TMI) Action Item II.K.3.10 requirements in NUREG-0737, "Clarification of TMI Action Plan Requirements," dated November 1980, which states:

> The anticipatory trip modification proposed by some licensees to confine the range of use to high-power levels should not be made until it has been shown on a plant-by-plant basis that the probability of a small-break loss-of-coolant accident (LOCA) resulting from a stuck-open power operated relief valve (PORV) is substantially unaffected by the modification.

The U.S. Nuclear Regulatory Commission (NRC) staff's review of the TS changes is to assure the licensee's compliance with the TMI Action Item II.K.3.10 requirements and compliance with regulations applicable to transient and accident analyses.

3.0 TECHNICAL EVALUATION

3.1 The LOCA and Transient Analyses

HBRSEP, Unit 2 is a three-loop, Westinghouse-designed pressurized-water reactor that was licensed to operate on September 23, 1970. The proposed TS change would increase the power level above which the reactor trip will occur in the event of a turbine trip. The licensee proposed to raise this power level from 10 percent of the RTP to 40 percent of the RTP. In support of the TS changes, the licensee identified the limiting cases in each event category discussed in the safety analysis sections of the updated final safety analysis report (UFSAR) and evaluated the effects of the TS changes on the loss-of-coolant-accident (LOCA) and non-LOCA transient analysis for each limiting case.

For the LOCA analysis, the licensee considered the following cases: (1) large break LOCAs; (2) small break LOCAs; and (3) inadvertent opening of a pressurizer safety or power operated relief valve (PORV). Since the proposed TS changes did not affect the normal operating parameters, the safeguards systems actuation or accident mitigation capabilities important to LOCA, or the assumptions used in the LOCA-related analysis, the licensee concluded, and the NRC staff agreed, that the proposed TS changes would not affect the results of the LOCA analysis in the UFSAR.

The licensee verified that the proposed TS changes have no effect on the following non-LOCA transient analyses:

1. Uncontrolled rod cluster control assembly (RCCA) bank withdrawal from a subcritical condition or low power condition

2. Uncontrolled RCCA bank withdrawal at power

3. Control rod mis-operation including withdrawal of a single full length RCCA, a dropped RCCA, a dropped RCCA bank and statistically misaligned RCCA.

4. Chemical and volume control system (CVCS) malfunction that results in a decrease in the boron concentration in the reactor coolant

5. Loss of reactor coolant flow

6. Startup of inactive reactor coolant loop

7. Loss of external electrical load

8. Turbine trip

9. Loss of normal feedwater flow

10. Excessive heat removal due to feedwater system malfunctions

11. Excessive load increase event

12. Loss of all alternating current power to the plant auxiliaries

13. Steam generator tube rupture

14. Rupture of a steam pipe

15. RCCA ejection

16. Major rupture of a main feedwater pipe

The licensee did not discuss the effect of the TS changes upon the analysis of the following events presented in the UFSAR:

1. Inadvertent opening of a steam generator relief or PORV

2. Loss of condenser vacuum and other events resulting in turbine trip

3. Locked rotor

4. Reactor coolant pump shaft break

5. Inadvertent loading and operation of a fuel assembly into the improper position

6. Inadvertent operation of emergency core cooling system

7. CVCS malfunction that increases reactor coolant inventory

Based on the transient analysis presented in the UFSAR, the NRC staff found that these events did not rely on the reactor trip on a turbine trip signal for consequence mitigation. Therefore, the NRC staff concluded that they would not be affected by the TS changes.

3.2 TMI Action Item II.K.3.10 analysis

The NRC staff position for TMI action Item II.K.3.10 in NUREG-0737 states that:

> The anticipatory trip modification proposed by some licensees to confine the range of use to high-power levels should not be made until it has been shown on a plant-by-plant basis that the probability of a small-break loss-of-coolant accident (LOCA) resulting from a stuck-open power operated relief valve (PORV) is substantially unaffected by the modification.

In addressing the TMI Action Item II.K.3.10 requirements, the licensee analyzed the turbine trip without reactor trip (TTWORT) event initiated from 40 percent power, identified by the licensee as the most limiting case affected by the proposed TS. The results of the analysis quantified the effect of the proposed TS on the potential of a small break LOCA resulting from stuck-open PORVs.

The licensee performed the TTWORT analysis using the Westinghouse LOFTRAN code. LOFTRAN, a reactor coolant system response code using a model containing a reactor vessel, hot and cold-leg piping, steam generators and a pressurizer, was previously approved by NRC for the non-LOCA transient analysis. In the analysis, the licensee assumed that the control systems including the rod control system, pressurizer pressure control system, steam dump control system, and steam generator control system were operable and in the automatic mode of control. The licensee performed the analysis based on a best-estimate approach without consideration of any instrument uncertainties. In addressing the TMI Action Item II.K.3.10 requirements, the NRC staff found that the assumptions of use of automatic control systems for consequence mitigation and a best-estimate approach for the analysis were consistent with those used by other Westinghouse plants approved by the NRC (such as Indian Point Unit 3 (ADAMS Accession No. ML003780834) and Salem (ADAMS Accession No. ML011690022)). Therefore, the NRC staff determined that they were acceptable.

The licensee discussed key assumptions (that contain Westinghouse proprietary information) in Attachment V, "Westinghouse Report LTR-SCS-06-22, Rev. 1, 'H. B. Robinson Unit 2 Turbine Trip without Reactor Trip Transient from P-8 Setpoint Analysis'" (LTR-SCS-06-22) to their June 19, 2009, application. Those assumptions include the turbine trip setpoint, the initial plant conditions (such as the reactor vessel average temperature, feedwater temperature, steam generator (SG) pressure and SG tube plugging level), reactivity parameters and specific models of automatic control systems credited in the analysis. The NRC staff found that the assumptions were conservative, resulting in a high predicted pressurizer pressure that would increase the potential to challenge the PORV to open. Therefore, the NRC staff concluded that the assumptions used in the analysis were adequate and acceptable.

The licensee analyzed the TTWORT event for two cases with specific values of the steam dump plant trip controller proportional gain and the trip-open setpoint for steam dump valves. The values are proprietary to Westinghouse and discussed on page 10 of LTR-SCS-06-22 as Options 1 and 2 cases.

The licensee's analysis for both Options 1 (Table 1 of LTR-SCS-06-22) and Option 2 (Table 3 of LTR-SCS-06-22) showed that for cases with all automatic control systems available, the pressurizer pressure increase, caused by the TTWORT event, would not be high enough to cause any of the PORVs to open. The licensee also performed a sensitivity analysis to assess the effect of failure modes in the control systems on the results of the TTWORT analysis. In the sensitivity study, the licensee considered the following credible single failures in automatic control systems:

1. Pressurizer pressure control system – 50 percent reduction in spray flow capacity (assuming that one spray valve fails to open).

2. Rod control system – Failure of the power mismatch channel. The power mismatch channel is to provide a fast signal to the rod control system during a rapid change in turbine load. If this signal is not present, then the rods are controlled by the T_{avg} error signal, which has a slower response and takes longer to drive the rod into the core at maximum speed following the turbine trip.

3. Steam Dump Control System - Single failure of one steam dump valve (SDV) in Bank 1 (one of three total valves failed).

4. Steam Dump Control System - Single failure of one SDV in Bank 2 (one of two total valves failed).

The results of the sensitivity study (Tables 2 and 4 of LTR-SCS-06-22) showed that for both Option 1 and Option 2, the TTWORT with the worst credible single failure event, the failure of one SDV in Bank 1, did not result in a peak pressurizer pressure that would actuate and open the PORV.

Since (1) the TTWORT event was analyzed using the NRC-approved LOFTRAN code and acceptable assumptions for the key input parameters, and (2) the results of the TTWORT with the worst credible single failure event showed that the likelihood of a small-break LOCA from a stuck-open PORV would be unaffected by the TS changes, meeting the TMI Action Item II.K.3.10 requirements, the NRC staff determined that the TTWORT analysis was acceptable for support of the TS changes.

The licensee stated that the steam dump control system settings, which are not in TSs, could be changed without staff approval under their 10 CFR 50.59, "Changes, tests, and experiments," process. The licensee submitted two options that meet the criteria of not challenging the PORV to lift and, therefore, are acceptable to the staff. In Attachment 2 of the licensee's June 19, 2009, application, the licensee indicated that Option 2 was a preferred option, because (1) the analysis in the Westinghouse report (LTR-SCS-06-22) showed that Option 2 resulted in fewer fluctuations in plant parameters during the transient response, when compared with Option 1, and (2) plant response was evaluated by testing Option 2 on the plant specific simulator for HBRSEP, Unit 2, which supported the conclusions in the Westinghouse report. The NRC staff's conclusions were limited to raising the RPS interlock for turbine trip function from the P-7 level to the P-8 level. The adjustments to the steam dump control system settings are subject to the requirements of 10 CFR 50.59.

3.3 Proposed TS Changes

The proposed changes would affect TS 3.3.1, which provides the operability requirements, allowable conditions, required actions, completion time, and surveillance requirements for the RPS. Specifically, Function 15 of TS Table 3.3.1-1 provides the operability requirements for the turbine trip input to the RPS. The application requirements for Function 15 are specified by a note (f), which states that the turbine trip input to the RPS is only required above P-7 (Low Power Reactor Trip Block) interlock. Function 15 of TS Table 3.3.1-1 states that Condition P is applicable when the Limiting Conditions for Operation requirements for the turbine trip function are not met. Condition P requires the turbine trip channel to be placed in trip within 6 hours or thermal power be reduced to less than P-7 within 10 hours.

The proposed TS Table 3.3.1-1 would change the turbine trip from a low auto stop oil pressure (Function 15.a), or closure of the turbine valve signal (Function 15.b), to be Applicable when power levels are above the P-8 (Power Range Neutron Flux) setting, rather than the current P-7 setting. The licensee proposed TS also would modify Condition P of TS 3.3.1, specifying that if one turbine trip channel is inoperable, the thermal power must be reduced to less than the P-8 setting, rather than the current P-7 setting. The NRC staff determined that the proposed TS changes were acceptable, since they were consistent with the technical bases discussed in Sections 3.1 and 3.2.

3.4 Summary

The NRC staff has found that: (1) the licensee demonstrated adequately that the proposed TSs did not change the results of transient and LOCA analyses in the UFSAR; (2) the licensee also satisfactorily addressed the TMI Action Item II.K.3.10 requirements by showing that the proposed TSs would not significantly affect the likelihood of a small-break LOCA resulting from a stuck-open pressurizer PORV during the limiting transient affected by the TS changes, the TTWORT event with the worst credible single failure initiated from 40 percent (P-8 interlock setpoint) of the RTP. Therefore, the NRC staff has concluded that the proposed Condition P of TS 3.3.1 and Function 15 in TS Table 3.3.1-1 with respect to the increased power level required for the reactor trip on a turbine trip are acceptable.

4.0 STATE CONSULTATION

In accordance with the NRC's regulations, the South Carolina State 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 the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. 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 NRC has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on the finding as published in the *Federal Register* on January 5, 2010 (75 FR 460). 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 issuance of the amendment.

6.0 CONCLUSION

The NRC 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 NRC'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 Contributor: Summer Sun, NRR

Date: March 17, 2010

Mr. Eric McCartney, Vice President
H. B. Robinson Steam Electric Plant,
Unit No. 2
Carolina Power & Light Company
3581 West Entrance Road
Hartsville, South Carolina 29550-0790

SUBJECT: H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2 - ISSUANCE OF AMENDMENT REGARDING TECHNICAL SPECIFICATIONS CHANGES RELATED TO THE TURBINE TRIP INPUT TO THE REACTOR PROTECTION SYSTEM (TAC NO. ME1550)

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Sincerely,

/RA/
Tracy J. Orf, Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-261

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2. Safety Evaluation

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