Progress Energy

Serial: RNP-RA/09-0054

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United States Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

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

REQUEST FOR TECHNICAL SPECIFICATIONS CHANGE TO
SECTION 3.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

Ladies and Gentlemen:

In accordance with the provisions of the Code of Federal Regulations, Title 10, Part 50.90, Carolina Power and Light Company, also known as Progress Energy Carolinas, Inc. (PEC), is submitting a request for an amendment to the Technical Specifications (TS) contained in Appendix A of the Operating License for H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2.

The proposed amendment will revise TS 3.3.1, "Reactor Protection System Instrumentation." The proposed change revises the requirements related to the reactor protection system interlock for the turbine trip input to the reactor protection system.

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

Attachment II provides a description of the current condition, a description and justification of the proposed change, a No Significant Hazards Consideration Determination, and an Environmental Impact Consideration.

Attachment III provides a markup of the affected TS pages. Attachment IV provides the retyped TS pages. Attachment V provides the proprietary version of Westinghouse Report LTR-SCS-06-22, Rev. 1, "H. B. Robinson Unit 2 Turbine Trip without Reactor Trip Transient from the P-8 Setpoint Analysis." Attachment VI provides the non-proprietary version of this report and Attachment VII provides the Westinghouse Affidavit, which provides the justification for withholding the proprietary version of the attached report.

In accordance with 10 CFR 50.91(b), Progress Energy Carolinas, Inc., is providing the State of South Carolina with a copy of this license amendment request.

Progress Energy Carolinas, Inc.
Robinson Nuclear Plant
3581 West Entrance Road
Hartsville, SC 29550

A020
NRR

Nuclear Regulatory Commission approval of the proposed license amendment is requested by March 31, 2010, to allow implementation of this change during Refueling Outage 26.

This license amendment request includes a commitment to complete a plant modification in accordance with the engineering change process, which will be implemented to conduct the required plant changes.

If you have any questions concerning this matter, please contact me at (843) 857-1253.

Sincerely,

C. T. Baucom
Manager – Support Services – Nuclear

Attachments:

I. Affirmation
II. Request for Technical Specifications Change Related to Reactor Protection System Instrumentation
III. Markup of Technical Specifications Pages
IV. Retyped Technical Specifications Pages
V. Westinghouse Report LTR-SCS-06-22, Rev. 1, "H. B. Robinson Unit 2 Turbine Trip without Reactor Trip Transient from the P-8 Setpoint Analysis," Proprietary
VI. Westinghouse Report LTR-SCS-06-22, Rev. 1, "H. B. Robinson Unit 2 Turbine Trip without Reactor Trip Transient from the P-8 Setpoint Analysis," Non-Proprietary
VII. Westinghouse Affidavit for Withholding Proprietary Information

CTB/cac

c: Ms. S. E. Jenkins, Manager, Infectious and Radioactive Waste Management Section (SC)
Mr. A. Gantt, Chief, Bureau of Radiological Health (SC)
Mr. L. A. Reyes, NRC, Region II
Mr. T. Orf, NRC Project Manager, NRR
NRC Resident Inspector, HBRSEP
Attorney General (SC)

AFFIRMATION

The information contained in letter RNP-RA/09-0054 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, also known as Progress Energy Carolinas, Inc. I declare under penalty of perjury that the foregoing is true and correct.

Executed On: 6/19/09

E. A. McCartney
Vice President, HBRSEP, Unit No. 2

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

REQUEST FOR TECHNICAL SPECIFICATIONS CHANGE RELATED TO REACTOR PROTECTION SYSTEM INSTRUMENTATION

Description of Current Condition

Technical Specifications (TS) Limiting Condition for Operation (LCO) 3.3.1, "Reactor Protection System Instrumentation," provides the operability requirements, allowed conditions, required actions, completion times, and surveillance requirements for the reactor protection system. Function 15 of Table 3.3.1-1 provides the operability requirements for the turbine trip input to the reactor protection system. The applicability requirements for Function 15 are modified by a note (f), which states that the turbine trip input to the reactor protection system is only required above the P-7 (Low Power Reactor Trips Block) interlock. TS Table 3.3.1-1, Function 15, also states that Condition P is applicable when the LCO requirements for the turbine trip function are not met. Condition P requires the channel to be placed in trip within 6 hours or thermal power be reduced to < P-7 within 10 hours.

Description and Justification of the Proposed Change

The proposed change revises the requirements related to the reactor protection system interlock for the turbine trip input to the reactor protection system. The applicability note (f) for TS Table 3.3.1-1, Function 15, "Turbine Trip," is proposed to be revised from the P-7 (Low Power Reactor Trips Block) to P-8 (Power Range Neutron Flux) interlock. This requires the associated Required Action P be revised to reduce thermal power to < P-8 within 10 hours.

This change is justified based on the analyses provided in the Westinghouse Report LTR-SCS-06-22, Rev. 1, "H. B. Robinson Unit 2 Turbine Trip without Reactor Trip Transient from the P-8 Setpoint Analysis," Proprietary, provided in Attachment V.

The Westinghouse evaluation was performed to assess the impact of increasing the power level for the turbine trip interlock. Following the Three Mile Island accident, the Nuclear Regulatory Commission (NRC) expressed concern about the implementation of blocking the reactor trip on turbine trip function on a permissive with an increased setpoint because of the potential to increase the probability of a stuck open pressurizer PORV. The NRC position is addressed in NUREG-0737, Item II.K.3.10. In NUREG-0737, the NRC has stated that any modifications to anticipatory trips should not be made until it has been shown by a licensee that the probability of a small break loss of coolant accident (LOCA) resulting from a stuck-open PORV is substantially unaffected by the modification. To satisfy the NRC requirements stated in Item II.K.3.10, the plant-specific analysis was performed to show that changing the reactor protection system interlock for the turbine trip input from the P-7 to the P-8 setpoint will not result in challenges to the pressurizer PORVs. The analysis was performed based on the P-8 nominal setpoint of 40 percent power. The results show that for this setpoint value, the pressurizer PORVs will not be challenged.

Option 2 from the Westinghouse report has been selected. The Westinghouse report shows that Option 2 results in fewer fluctuations in plant parameters during the transient response, when compared with Option 1. Additionally, plant response was evaluated by testing Option 2 on the plant-specific simulator for H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2, which supported the conclusions in the Westinghouse report. Therefore, Option 2 is the preferred option.

The Westinghouse report shows that either Option 1 or 2 can be used. Additionally, the acceptance criteria for this analysis, as documented in the Westinghouse report, is demonstration that the pressurizer power-operated relief valves are not expected to open during the transient conditions created by a turbine trip at approximately 40% power without a direct input to reactor trip.

Changes to the condenser steam dump control system can be made under licensee control in accordance with 10 CFR 50.59, provided that this and other applicable acceptance criteria for condenser steam dump system operation are met. Therefore, the statement that Option 2 has been chosen is not considered a commitment to implement Option 2, as described in the Westinghouse report. This license amendment request includes the commitment to complete a plant modification in accordance with the engineering change process, which will be implemented to conduct the required plant changes.

Plant modification is required for implementation of changes associated with the turbine trip reactor protection system interlock. The plant modification includes changes to the reactor protection system wiring that will change the turbine trip reactor protection function interlock from the P-7 permissive circuitry to the P-8 permissive circuitry. The plant modification also includes changes to the condenser steam dump control system consistent with those identified in the Westinghouse report.

The plant modification requirements are documented in Engineering Change (EC) 63785. The modification includes the installation requirements and procedure updates that will implement the required interlock and control system changes. The reactor protection system interlock changes involve moving and connecting wires within the reactor protection system. The condenser steam dump control system changes involve changes to calibration procedures that are required to be performed in accordance with the installation instructions for the modification.

Option 2 from the Westinghouse report provides the proposed condenser steam dump system control settings for the plant trip controller proportional gain, the plant trip controller trip-open setpoints for Bank 1 and Bank 2, and the sudden loss of load bistables. These settings will be established and controlled in accordance with the calibration procedure changes described in EC 63785. The procedure controls for these settings remain consistent with the current requirements for maintaining the condenser steam dump control system settings, including the calibration procedure being subject to the requirements of 10 CFR 50.59.

Procedure EGR-NGGC-0005, "Engineering Change," contains explicit instructions for modification turnover to operations. The turnover section of EC 63785 states that the required work tickets shall be in the "Finished" status and that the required documents are updated.

Therefore, the modification requirements for EC 63785 establish the necessary steps and conditions that ensure the required modifications will be completed prior to plant operation with the proposed interlock change for the turbine trip.

Also, a review of Updated Final Safety Analysis Report (UFSAR) safety analyses is provided, as follows, which shows that the safety analyses results are not adversely affected by this proposed change to the reactor protection system that revises the turbine trip function interlock from P-7 (approximately 10 percent power) to P-8 (approximately 40 percent power).

1. Uncontrolled Rod Cluster Control Assembly (RCCA) Bank Withdrawal from a Subcritical or Low Power Condition

Event Definition: An RCCA bank withdrawal accident is defined as an uncontrolled addition of reactivity to the reactor core caused by withdrawal of RCCA banks, resulting in a power excursion. Should a continuous RCCA withdrawal be initiated, there are four safety mechanisms that limit this event. These are:

Source range neutron flux level trip,
Intermediate range neutron flux level trip,
Intermediate range rod stop, and
Power range neutron flux level trip (low setting).

Plant Operating Conditions: The plant is assumed to be at the no-load reactor coolant average temperature.

Effect of Proposed Change: In this scenario, the reactor is at a power level of 2300 x 10E-9 MWt and the turbine generator is not on-line. The direct reactor trip from turbine trip is not credited. Therefore, the proposed change has no effect on this accident scenario and the conclusions of the UFSAR remain valid.

2. Uncontrolled RCCA Bank Withdrawal at Power

Event Definition: This event is defined as the inadvertent addition of positive reactivity to the core caused by the uncontrolled withdrawal of an RCCA bank(s) while at power. The automatic features of the reactor protection system which prevent core damage in an RCCA bank withdrawal incident at power include the following:

Power range neutron flux level trip (high setting),
Overtemperature delta-T.

Plant Operating Conditions: Analyses cases are evaluated for initial reactor power at 10.3, 60.3, and 100.3 percent of rated thermal power (2339 MWt).

Effect of Proposed Change: The direct reactor trip from turbine trip is not credited in the analysis. Therefore, the proposed change has no effect on this accident scenario and the conclusions of the UFSAR remain valid.

3. Control Rod Misoperation

Event Definition: RCCA misoperation accidents include:

Withdrawal of a single full length RCCA
A dropped RCCA,
A dropped RCCA bank, and
Statically misaligned RCCA.

Plant Operating Condition: Analyses are performed at full power conditions.

Effect of Proposed Change: The reactor trip from the turbine trip function is not credited for this event as either a primary or backup trip. Therefore, the proposed change has no effect on this accident scenario and the conclusions of the UFSAR remain valid.

4. Chemical and Volume Control System Malfunction

Event Definition: This event is the inadvertent dilution of the RCS boron concentration.

Plant Operating Condition: Boron dilution during shutdown, refueling, startup, and power operations were examined.

Effect of Proposed Change: The reactor trip on turbine trip function is not credited for this event as either a primary or backup trip. Therefore, the proposed change has no effect on this accident scenario and the conclusions of the UFSAR remain valid.

5. Loss of Reactor Coolant Flow

Event Definition: The loss of flow incident can result from a mechanical or electrical failure in a reactor coolant pump (RCP), or from a fault in the power supply of these pumps.

The following trip circuits provide the necessary protection against a loss of coolant flow incident:

Undervoltage or underfrequency on RCP power supply buses,
Pump circuit breaker opening, and
Low reactor coolant flow.

Plant Operating Condition: The following loss of flow cases are analyzed:

Loss of three pumps from nominal full power conditions with three loops operating, and
Locked rotor accident from nominal full power conditions with three loops operating.

Effect of Proposed Change: The low primary coolant loop flow, RCP undervoltage, RCP underfrequency, and RCP breaker position reactor trip functions provide the necessary protection for this event. These trips are not affected by the proposed change to the reactor protection system interlock for the turbine trip function.

Therefore, the proposed change has no effect on this accident scenario group and the conclusions of the UFSAR remain valid.

6. Startup of an Inactive Reactor Coolant Loop

Event Definition: The inadvertent startup of an idle loop while operating would result in the sudden introduction of colder water into the core from the idle loop which could cause an unplanned reactivity insertion and power increase.

Plant Operating Condition: The TS preclude operation of the plant with one or more loops out of service.

Effect of Proposed Change: This event is not part of the plant's licensing basis. Therefore, the proposed change has no effect on this accident scenario.

7. Loss of External Electrical Load

Event Definition: The loss of external electrical load and turbine trip events are defined as a complete loss of steam load from full power. Reactor protection is provided by:

High pressurizer pressure,
Overtemperature delta-T,
High pressurizer water level, and
Low-low steam generator water level.

Plant Operating Condition: The analysis of the loss of external electrical load assumes a complete loss of steam load from full power with no credit taken for the turbine trip function input to reactor protection. At power levels greater than the P-8 interlock, the turbine trip function will continue to provide an anticipatory trip function for reactor protection.

Effect of Proposed Change: Protection for this event is provided by the Overtemperature delta-T, high pressurizer pressure, high pressurizer water level, or low-low steam generator water level signals. The turbine trip event description states that the turbine trip causes a direct reactor trip with results in earlier trip than analyzed for the loss of external electrical load event. As previously stated, at power level greater than the P-8 interlock this will continue to occur. The loss of electrical load event is analyzed using inputs and conditions (i.e., full power), which bound the turbine trip without direct reactor trip below the P-8 interlock. Therefore, the proposed change has no effect on this accident scenario and the conclusions of the UFSAR remain valid.

8. Loss of Normal Feedwater Flow

Event Definition: The design basis loss of normal feedwater event is defined as a reduction in the capability of the secondary system to remove heat generated in the reactor core.

The reactor trip is initiated by:

Low – low steam generator water level, and
High pressurizer pressure.

Plant Operating Condition: A complete loss of main feedwater flow is assumed to occur from 102 percent of 2300 MWt.

Effect of Proposed Change: The credited trip signals are not affected by the change. Therefore, the proposed change has no effect on this accident scenario and the conclusions of the UFSAR remain valid.

9. Excessive Heat Removal due to Feedwater System Malfunctions

Event Definition: This event may result from an increase in feedwater flow to one or more of the steam generators or a decrease in feedwater temperature. This event will result in an increase in the heat transfer rate from primary to secondary in the steam generators and a consequential reduction in primary system temperature and pressure.

Plant Operating Condition: This event is analyzed at power levels corresponding to zero and full load.

Effect of Proposed Change: Opening of a low pressure heater bypass valve causes a reduction in feedwater temperature which increases the thermal load on the primary system. This reduction in feedwater temperature results in an increase in heat load on the primary system of less than 10 percent of full power. The increased thermal load would result in a transient very similar (but of reduced magnitude) to the consequences of a 10 percent step load increase. Reactor trip from turbine trip is not credited in Item 10, Excess Load Increase Event. The results of the feedwater temperature reduction analysis are not affected by the proposed change.

The increase in feedwater flow analysis states that the event is bounded by the results of the excess load increase event and the uncontrolled rod withdrawal event. These events do not rely upon the turbine trip function. Additionally, the increase in feedwater flow event evaluation does not rely upon the turbine trip function and will remain bounded by the results of the excess load increase event and the uncontrolled rod withdrawal event.

Therefore, the proposed change does not invalidate the results of the analysis and the conclusions of the UFSAR remain valid.

10. Excess Load Increase Event

Event Definition: An excess load increase event is defined as a rapid increase in the steam flow that causes a power mismatch between the reactor core power and the steam generator load demand.

Plant Operating Condition: The analysis is performed at 100 percent power.

Effect of Proposed Change: A reactor trip does not occur in this analysis. Instead, the plant reaches a new equilibrium condition at a higher power level corresponding to the increase in steam flow. A reactor trip from turbine trip is not among the credited trip actuations. Therefore, the proposed change has no effect on this accident scenario and the conclusions of the UFSAR remain valid.

11. Loss of All Alternating Current (AC) Power to the Plant Auxiliaries

Event Definition: A complete loss of non-emergency power (i.e., offsite power) may result in the loss of all power to the plant auxiliaries: i.e., the reactor coolant pumps (RCPs), condensate pumps, etc. The loss of power may be caused by a complete loss of the offsite grid accompanied by a turbine generator trip at the station, or by a loss of onsite non-emergency AC distribution system. The event is bounded by the loss of external electrical load, loss of feedwater flow, and loss of reactor coolant flow events.

Plant Operating Condition: See bounding events, which are loss of external electrical load, loss of feedwater flow, and loss of reactor coolant flow events.

Effect of Proposed Change: The reactor trip from turbine trip function is described in the analysis because it is expected to occur. The reactor trip on turbine trip is not required for core protection for the bounding events and was not credited in those analyses. Therefore, the proposed change has no effect on this accident scenario and the conclusions of the UFSAR remain valid.

12. Steam Generator Tube Rupture (SGTR)

Event Definition: This event is assumed to be the complete severance of a single tube.

Plant Operating Condition: The analysis is based on reactor decay heat equivalent to 120 percent of ANS standard with an infinite 100 percent power history.

Effect of Proposed Change: The basis and assumptions for this event are not affected by the proposed change. The UFSAR states that this event is similar to the primary valve malfunction event, Section 15.6.1, except the primary fluid relieves to the faulted generator, and that the Section 15.6.1 analysis shows the specified acceptable fuel design limit are not penetrated. The loss of coolant accident impact is evaluated in Section 16 (below) and shows the change to the turbine trip interlock has no effect on these analyses. Therefore, the proposed change has no effect on this accident scenario and the conclusions of the UFSAR remain valid.

13. Rupture of a Steam Pipe

Event Definition: A rupture of a steam pipe is assumed to include any accident which results in an uncontrolled steam release from a steam generator. Such a release may result from either the opening of a steam generator relief or safety valve, or from a steam system pipe break.

Plant Operating Condition: "Hot Zero Power" and "Hot Full Power" cases are analyzed.

Effect of Proposed Change: The limiting full power analysis credits the reactor trip system input from the safety injection signal input. Only the Engineered Safety Features Actuation System is needed to limit the consequence of the analyzed events. Therefore, the proposed change has no effect on this accident scenario and the conclusions of the UFSAR remain valid.

14. Rupture of a Control Rod Mechanism Housing – RCCA Ejection

Event Definition: This event is an assumed failure of a control rod mechanism pressure housing such that RCS pressure would eject the control rod and drive shaft. The reactor will trip on the power range high neutron flux.

Plant Operating Condition: Both full and zero power cases are analyzed.

Effect of Proposed Change: The trip function for this event is not affected by the proposed change. Therefore, the proposed change has no effect on this accident scenario and the conclusions of the UFSAR remain valid.

15. Major Rupture of Main Feedwater Pipe (Feedline Break)

Event Definition: This event is caused by the instantaneous severance of a feedwater line. This event is a cooldown event and is bounded by the steam line break analysis.

Plant Operating Condition: Bounded by the steam line break analysis.

Effect of Proposed Change: The trip functions for this event are not affected by the proposed change. Therefore, the proposed change has no effect on this accident scenario and the conclusions of the UFSAR remain valid.

16. Loss of Coolant Accident (LOCA) and LOCA-Related Analyses

Event Definition: A LOCA is the result of a pipe rupture of the RCS pressure boundary or inadvertent opening of a pressurizer safety or power operated relief valve. The following LOCA-related analyses have been reviewed for impact by the proposed change:

Large break LOCA,
Small break LOCA, and
Inadvertent opening of a pressurizer safety or power operated relief valve.

Plant Operating Condition: Full power conditions are assumed.

The low pressurizer pressure and safety injection actuation functions provide reactor protection for these events.

Effect of Proposed Change: The change does not affect the normal plant operating parameters, the safeguards systems actuation or accident mitigation capabilities important to LOCA, or the assumptions used in the LOCA-related accidents. The change does not create conditions more limiting than those determined in these analyses.

17. Containment Integrity Evaluation

Event Definition: Containment integrity analyses are performed for containments to verify that containment design pressure is not exceeded during design basis events.

Plant Operating Condition: Full power conditions are assumed.

Effect of Proposed Change: The turbine trip function is not credited in the containment integrity analyses. The proposed change does not adversely affect the short term or long term mass and energy releases of the containment analyses. Therefore, the conclusions presented in the UFSAR remain valid with respect to the containment analyses.

18. Main Steam Line Break (MSLB) Mass and Energy Release Analyses

Event Definition: Steam line ruptures occurring inside a reactor containment structure may result in significant releases of high energy fluid to the containment environment, possibly resulting in high containment temperatures and pressures.

Plant Operating Condition: Full power and hot-zero-power conditions are assumed.

Effect of Proposed Change: The turbine trip function is not credited in the UFSAR MSLB analyses. The conclusions presented in the UFSAR remain valid with respect to MSLB mass and energy release rates and steam mass release calculations.

<u>Summary</u>

The HBRSEP, Unit No. 2, UFSAR analyses of record do not credit the direct reactor trip from turbine trip in the analyses that show the accident analysis acceptance criteria are met. The conclusions of the UFSAR remain valid following the proposed change to the reactor protection system where the turbine trip interlock is changed from P-7 to P-8. The P-7 interlock receives input from the power range neutron flux instrumentation and the turbine first stage pressure. The P-8 interlock only receives input from the power range neutron flux instrumentation. This change is acceptable because the P-8 interlock will continue to receive reliable input from the power range neutron flux instrumentation.

The Westinghouse Report LTR-SCS-06-22, Rev. 1, "H. B. Robinson Unit 2 Turbine Trip without Reactor Trip Transient from the P-8 Setpoint Analysis," Proprietary, provided in Attachment V, describes the analysis associated with this change. The analysis shows the applicable acceptance criteria are met based on the identified modifications to the steam dump system. A plant modification in accordance with the engineering change process will be implemented to conduct the required plant changes.

Precedents

The NRC has approved similar submittals for plants changing the interlock at which the reactor trip from turbine trip is enabled. Specific precedents include:

Indian Point Unit 3	Accession No. ML003780834
North Anna	Accession No. ML013460457
Salem	Accession No. ML011690022
Braidwood/Byron	Accession No. ML020850675
D.C. Cook	Accession No. ML062840162

No Significant Hazards Consideration Determination

Carolina Power and Light Company, also known as Progress Energy Carolinas, Inc. (PEC), is proposing a change to Appendix A, Technical Specifications, of Facility Operating License No. DPR-23, for the H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2. The proposed change revises the requirements related to the reactor protection system interlock for the turbine trip input to the reactor protection system. The applicability note (f) for TS Table 3.3.1-1, Function 15, "Turbine Trip," is proposed to be revised from P-7 (Low Power Reactor Trips Block) to P-8 (Power Range Neutron Flux) interlock. This requires the associated Required Action P be revised to reduce thermal power to less than P-8 within 10 hours.

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 provides revised requirements for the reactor protection system interlock associated with the turbine trip protection function. The proposed change will allow the interlock for turbine trip function to be raised from the current interlock setting of about 10 percent reactor power to about 40 percent reactor power.

   This change will allow the reactor to continue operating safely at power levels up to about 40 percent when the turbine is not operating. The applicable accident analyses have been reviewed and it is concluded that the accident analyses are unaffected by the proposed change. An analysis of plant response to a turbine trip at approximately 40 percent power shows that the applicable acceptance criteria are met.

   The probability of the previously evaluated accidents remains unaffected because the proposed change to this interlock function does not increase the likelihood of any accident initiator or precursor.

Therefore, operation of the facility in accordance with the proposed amendment would 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.

   As described above, the proposed change provides revised requirements for the reactor protection system interlock associated with the turbine trip protection function. The proposed change will allow the interlock for turbine trip function to be raised from the current interlock setting of about 10 percent reactor power to about 40 percent reactor power.

   No new accident initiators or precursors are introduced by the proposed change. The required reactor protection system response as analyzed in the UFSAR is unchanged, except a reactor trip signal will not be generated due to a turbine trip when the reactor is operated below the P-8 interlock power level (approximately 40 percent power). As stated in the response to Criterion 1 (above), the applicable accident analyses have been reviewed and it is concluded that the accident analyses are unaffected by the proposed change. An analysis of plant response to a turbine trip at approximately 40 percent power shows that the applicable acceptance criteria are met. Therefore, no new or different accident sequences are created by this change.

   Therefore, operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any previously evaluated.

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

   As described above, the proposed change provides revised requirements for the reactor protection system interlock associated with the turbine trip protection function. The proposed change will allow the interlock for the turbine trip function to be raised from the current interlock setting of about 10 percent reactor power to about 40 percent reactor power.

   Also, as previously described, this change will allow the reactor to continue operating safely at power levels up to about 40 percent when the turbine is not operating. The applicable accident analyses have been reviewed and it is concluded that the accident analyses are unaffected by the proposed change. An analysis of plant response to a turbine trip at approximately 40 percent power shows that the applicable acceptance criteria are met.

   Therefore, operation of the facility in accordance with the proposed amendment would not involve a significant reduction in the margin of safety.

Based on the above discussion, Carolina Power and Light Company has determined that the requested change does not involve a significant hazards consideration.

Environmental Impact Consideration

10 CFR 51.22(c)(9) provides criteria for identification of licensing and regulatory actions for categorical exclusion from performing an environmental assessment. A proposed change for an operating license for a facility requires no environmental assessment if operation of the facility in accordance with the proposed change would not (1) involve a significant hazards consideration; (2) result in a significant change in the types or significant increases in the amounts of any effluents that may be released offsite; (3) result in a significant increase in individual or cumulative occupational radiation exposure. Carolina Power and Light Company, also known as Progress Energy Carolinas, Inc., has reviewed this request and determined the proposed change 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 needs to be prepared in connection with the issuance of the amendment. The basis for this determination follows.

Proposed Change

Carolina Power and Light Company, also known as Progress Energy Carolinas, Inc. (PEC), is proposing a change to Appendix A, Technical Specifications (TS), of Facility Operating License No. DPR-23, for the H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2. The proposed change revises the requirements related to the reactor protection system interlock associated with the turbine trip protection function. The proposed change will allow the interlock for turbine trip function to be raised from the current interlock setting of about 10 percent reactor power to about 40 percent reactor power.

Basis

The proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) for the following reasons:

1. As demonstrated in the No Significant Hazards Consideration Determination, the proposed change does not involve a significant hazards consideration.

2. As demonstrated in the No Significant Hazards Consideration Determination, the proposed change does not result in a significant increase in the consequences of an accident previously evaluated and does not result in the possibility of a new or different kind of accident. Therefore, the proposed change does not result in a significant change in the types or significant increases in the amounts of any effluents that may be released offsite.

3. The proposed change does not alter any parameters from which the individual and cumulative radiation exposure for HBRSEP, Unit No. 2, results. Therefore, the proposed change does not result in a significant increase in individual or cumulative occupational radiation exposures.

United States Nuclear Regulatory Commission
Attachment III to Serial: RNP-RA/09-0054
3 Pages (including cover page)

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

REQUEST FOR TECHNICAL SPECIFICATIONS CHANGE RELATED TO REACTOR PROTECTION SYSTEM INSTRUMENTATION

MARKUP OF TECHNICAL SPECIFICATIONS PAGES

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-7~~.	10 hours

$\boxed{\text{P-8}}$

(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-7 (Low Power Reactor Trips Block)~~ interlock.

P-8 (Power Range Neutron Flux)

United States Nuclear Regulatory Commission
Attachment IV to Serial: RNP-RA/09-0054
3 Pages (including cover page)

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

REQUEST FOR TECHNICAL SPECIFICATIONS CHANGE
<u>RELATED TO REACTOR PROTECTION SYSTEM INSTRUMENTATION</u>

RETYPED TECHNICAL SPECIFICATIONS PAGES

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.