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U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2
DOCKET NOS. 50-325 AND 50-324/LICENSE NOS. DPR-71 AND DPR-62
SUPPLEMENTAL INFORMATION TO REQUEST FOR LICENSE AMENDMENTS - POWER UPRATE

Gentlemen:

On May 23, 1996, Carolina Power & Light Company (CP&L) met with the NRC staff to discuss a proposed amendment to the Technical Specifications for the Brunswick Steam Electric Plant (BSEP), Unit Nos. 1 and 2 which was submitted on April 2, 1996. These proposed amendments revise the BSEP Technical Specifications to allow uprate of the units to 105% of rated thermal power.

Enclosure 1 provides CP&L's formal response to questions raised by the NRC staff during the May 23, 1996 meeting. Enclosure 2 provides Revision 2 of NEDC-32148, "Reactor Pressure Vessel Power Uprate Stress Report Reconciliation for the Brunswick Units 1 and 2 Power Plants." Enclosure 3 provides SIR-89-036 Rev. 1, "Design Report for Brunswick, Units 1 and 2, Core Spray System N5 Nozzle Safe-End, Thermal Sleeve, and Transition Piece Replacement. " Enclosure 4 provides a list of regulatory commitments.

Please refer any questions regarding this submittal to Mr. Tony Harris at (910) 457-3312.

Sincerely,

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William R Campbell

William R. Campbell

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Enclosures

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William R. Campbell, having been first duly sworn, did depose and say that the information contained herein is true and correct to the best of his information, knowledge and belief; and the sources of his information are officers, employees, and agents of Carolina Power & Light Company.

Notary (Seal)

My commission expires: August 12, 1996

pc: Mr. S. D. Ebneter, Regional Administrator, Region II
Mr. D. C. Trimble, Jr., NRR Project Manager - Brunswick Units 1 and 2
Mr. C. A. Patterson, Brunswick NRC Senior Resident Inspector
The Honorable H. Wells, Chairman - North Carolina Utilities Commission

ENCLOSURE 1

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2
NRC DOCKET NOS. 50-325 AND 50-324
OPERATING LICENSE NOS. DPR-71 AND DPR-62
REQUEST FOR LICENSE AMENDMENTS
POWER UPRATE

CP&L RESPONSE TO NRC STAFF QUESTIONS FROM MAY 23, 1996 MEETING

Question 1

Has CP&L incorporated the recommendations contained in General Electric (GE) SIL 377 for the Reactor Core Isolation Cooling (RCIC) system and SIL 480 for the High Pressure Coolant Injection (HPCI) system? Also, CP&L needs to provide a commitment to monitor the reliability of the HPCI and RCIC systems.

CP&L Response to Question 1

CP&L's evaluation of GE SIL 377 determined that these recommendations do not need to be incorporated into the Brunswick Plant design. At the time the SIL was issued, CP&L had installed a new exhaust check valve and noted that the number of overspeed trips had been reduced. Since that time, CP&L has taken data during the surveillance tests to verify this determination was correct. The surveillance data shows that the RCIC turbine has a steep initial ramp speed which drops off after a few seconds and eventually follows the ramp generator signal converter startup ramp until stable operation is obtained. The peak speed does not reach the overspeed trip setpoint (5625 rpm) and typically does not reach the rated speed (4500 rpm) during this initial phase of the turbine start.

As stated in Section 4.2.1 on page 4-7 of NEDC-32466P, Brunswick has implemented the recommendations of General Electric SIL 480 for the HPCI system.

In regards to monitoring system reliability, Brunswick has committed to implementing the Maintenance Rule in accordance with 10 CFR 50.65. Reliability of both the HPCI and RCIC systems are being monitored in accordance with the criteria developed by CP&L to comply with the Maintenance Rule, with specific criteria established for each system in accordance with NUMARC 93-01, "Industry Guidelines for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants."

Question 2

CP&L will need to provide a commitment to monitor recirculation pump vibration.

CP&L Response to Question 2

In the Brunswick specific system evaluation report, General Electric has evaluated the expected speed increase necessary to maintain the same core flow. The results of this evaluation shows an increase in speed of less than 1 percent is necessary. Both units have already operated their respective recirculation pumps at the speed where the vibration was noticed (1525 - 1550 rpm) with no increase in vibration/noise indicated. Brunswick Unit 2 has recently concluded a long operating cycle where the recirculation pumps were run at speeds which delivered flow above 105% of the rated core flow at a speed of 1550 rpm.

Power uprate does not increase the licensed core flow of the Brunswick units. The Brunswick units have been previously licensed for increased core flow to allow additional operating flexibility. Since GE SIL 600 has concluded that the vibration/noise encountered was not attributable to an increase in core thermal power, CP&L does not believe that additional monitoring of the recirculation pumps, beyond that normally performed for the system as part of the post refueling startup test program, is warranted. However, to ensure no increase in vibration or noise occurs with the Brunswick power uprate, during plant startup, CP&L will monitor the reactor recirculation system with existing reactor recirculation pump motor vibration annunciation instrumentation during the power ascension testing. CP&L does not plan to install any new or special test equipment.

Question 3

Determine why the difference exists between the analyzed pressure of the CRD system assumed on Page 2-3 of the Licensing Topical Report and the normal operating pressure assumed for power uprate.

CP&L Response to Question 3

The Safety Analysis Report for Brunswick Power Uprate (NEDC-32466P) states that the CRD system was evaluated at a reactor dome pressure of 1035 psig. The 1035 psig number is a typographical error. The analysis was actually performed at a reactor dome pressure of 1030 psig (1045 psia), which is the reactor dome pressure reported in the reactor heat balance (Figure 1-1) and which represents the reactor condition under power uprate. BNP will provide a revised page to the BNP topical report to indicate the correct value.

Question 4

Confirm that the LOCA analyses performed under SAFER/GESTR bound the power uprate. Also confirm that the break spectrum (and single failure analysis) is the same and ensure that the cycle specific analyses address uprate.

What impact does power uprate have on the Safety Limit MCPR?

CP&L Response to Question 4

The current SAFER/GESTR-LOCA (S/G) base analysis is described in Reference 1 and has been accepted by the NRC as the BNP LOCA licensing basis (Reference 2). Supplement 3 (Reference 3) to the base analysis documents the application of the approved SAFER/GESTR methodology to the GE13 fuel type that is currently in use for Brunswick Unit 2, Cycle 12 and that will be utilized for the uprated Brunswick Unit 1, Cycle 11. Supplement 3 was provided to the NRC staff with the Core Operating Limits Report on March 4, 1996 for the Brunswick Unit 2 Cycle 12 reload.

The base SAFER/GESTR-LOCA analysis and the Supplement 3 analysis were performed at a nominal power level of 2680 MWth (110% of the current rated power level of 2436 MWth) and an Appendix K power level of 2733 MWth (102% of 2680) in anticipation of a future power uprate. Therefore, this analysis bounds the requested uprated power level (2558 MWth). Additionally, the ECCS performance requirements (i.e., flow rates, start times, water levels, etc.) utilized for the S/G analyses are consistent with or conservative relative to the performance requirements assumed in the uprate evaluations. Therefore, the break spectrum analyses documented in the base report remain applicable to the planned uprate to 2558 MWth (105% of 2436 or 2558 MWth).

The single failure analysis is dependent upon plant configuration (i.e., equipment line-ups, equipment power supplies, electrical distribution systems, etc.) and not the power level at which the core is operated. Since the power uprate modifications do not affect the parameters that influence the single failure analysis, current single failure analyses remain bounding.

The B1C11 cycle specific LOCA report will be based on the analyses performed and documented in the base S/G report and the GE13 supplement. Since these reports assumed a nominal power level of 2680 MWth, the cycle specific LOCA report bounds the proposed uprate to 2558 MWth.

Safety Limit MCPR is not sensitive to reactor power level changes. The Safety Limit MCPR is dependent upon 1) the power peaking within the bundle and hence is impacted by the switch to the GE13 fuel type, and 2) the radial power distribution within the core, which is impacted by core design changes such as physical placement of the reload assemblies in the core. Therefore, power uprate does not affect the Safety Limit MCPR for the Brunswick Units.

A revision to the Safety Limit MCPR is being pursued for Unit 1 Cycle 11 as a part of the Cycle 11 reload analysis license amendment request, and was submitted to the NRC staff on April 9, 1996. Power uprate parameters have been included in this analysis.

The Cycle 11 reload analysis is currently being reviewed by CP&L and General Electric Company to address concerns with Safety Limit MCPR calculation methodology identified by

General Electric Company in a 10 CFR Part 21 Report issued to the NRC staff on May 24, 1996. CP&L will inform the NRC staff of any changes which may be necessary to the Safety Limit MCPR as a result of that review.

References:

1. "Brunswick Steam Electric Plant Units 1 and 2 SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis (Revision 2)", NEDC-31624P, July 1990.
2. NRC SER, "Revision of SAFER/GESTR-LOCA Analysis - Brunswick Steam Electric Plant, Units 1 and 2 (TAC Nos. 77585 and 77586)", January 10, 1991 (NRC-91-018).
3. "Brunswick Steam Electric Plant Units 1 and 2 SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis: Application to GE13 Fuel", NEDC-31624P, Supplement 3, Revision 0, January 1996.

Question 5

Confirm that the anticipated releases from power uprate are bounded by the Environmental Impact Statement.

CP&L Response to Question 5

The Non-Radiological Environmental Assessment submitted in November of 1995 with the BNP specific topical report addressed the effects of power uprate on the environment. This assessment stated that there would be no change to the current discharge permits as a result of power uprate. In addition, the radiological evaluation submitted as part of the topical report indicated that the radiological releases from normal operation are within the existing design of the plant and the releases as a result of a postulated accident are increased by a small fraction. The anticipated radiological releases were determined to be well within the guidelines of 10 CFR 20, 10 CFR 50, Appendix I, and 10 CFR 100. The evaluations performed have shown that there are no changes to the conclusions of the Environmental Impact Statement as a result of power uprate.

Question 6

Provide the basis for the differences between the Unit 1 and Unit 2 Feedwater (FW) nozzle usage factors. Also, provide a description (i.e., the analysis) of the conservatisms noted as being removed from the Core Spray (CS) nozzle usage factor calculations.

CP&L Response to Question 6

The results of the BNP Units 1 and 2 feedwater nozzle usage factor analyses are contained in NEDC-32148, Revision 2, "Reactor Pressure Vessel Power Uprate Stress Report Reconciliation for the Brunswick Units 1 and 2 Power Plants," sections 7.3 and 7.4. This document is included as Enclosure 2. The Unit 1 feedwater nozzle safe end was replaced after the original RPV vendor stress report was issued. The difference, in design fatigue usage from the Brunswick Unit 2 analysis, is mostly due to the modified design. The Unit 1 safe end has a welded thermal sleeve, while the Unit 2 safe end has a interference fit thermal sleeve. Different analysis assumptions, such as how events other than the most severe are modeled, also had an effect. Also, the Brunswick Unit 1 analysis covers several designs with similar but varying dimensions, so the dimensions used in the fatigue analysis were those that would give the highest usage.

The analysis for the Core Spray nozzle, safe end, thermal sleeve and transition piece is contained in Structural Integrity Associates (SIA) Report SIR-89-036 Revision 1. This document is included as Enclosure 3. The math model is divided into eight (8) regions. The conditions analyzed were A) Emergency Shutdown, B) Startup/Shutdown, C) Loss of Feedwater Pumps, and D) Test. The SIA report summarizes the usage factor for each analyzed condition based on the maximum enveloped stress from the eight (8) defined regions. This analysis resulted in an enveloped cumulative usage factor (CUF) of 0.98. When the CUFs are calculated by region, the maximum pre-uprate (current) CUF becomes 0.92. Core Spray nozzle fatigue evaluation methodology for Power Uprate is contained in NEDC-32148, Revision 2. When the Power Uprate methodology is applied, the CUF is scaled upwards to 0.96. This is the only conservatism removed from the calculation of the CUF for Power Uprate as stated in the November 1995 submittal. Other conservatisms which exist in the analysis were not removed for this evaluation.

ENCLOSURE 2

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"Reactor Pressure Vessel Power Uprate Stress Report Reconciliation for the Brunswick Units 1 and 2 Power Plants"