July 21, 2003

Mr. J. S. Keenan
Vice President
Brunswick Steam Electric Plant
Carolina Power & Light Company
Post Office Box 10429
Southport, North Carolina 28461

SUBJECT: BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2 - REQUEST FOR ADDITIONAL INFORMATION ON AMENDMENT REQUEST TO EXPAND THE CORE FLOW OPERATING RANGE (MELLLA+) (TAC NOS. MB6692 AND MB6693)

Dear Mr. Keenan:

The U.S. Nuclear Regulatory Commission (NRC) staff is reviewing the subject amendment request as submitted by your letter dated November 12, 2002. In a teleconference on July 8, 2003, we discussed our need for additional information with members your staff. The draft request for additional information (RAI) had been e-mailed to your licensing staff on June 16, 2003. The enclosed RAI formalizes the NRC staff's information request.

As we discussed during the July 8 teleconference, we will need a timely response. Please respond within 45 days to meet the schedule established for this review. If more time is needed, or you need clarification of this RAI, feel free to contact me at 301-415-2020 or blm@nrc.gov.

Sincerely,

/RA by Chandu Patel for/

Brenda L. Mozafari, Senior Project Manager, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-325 and 50-324

Enclosure: RAI

cc w/encl: See next page

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DATE	7/18/03	7/18/03	7/18/03	7/ /03

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REQUEST FOR ADDITIONAL INFORMATION FOR
AMENDMENT TO EXPAND THE CORE FLOW OPERATING RANGE (MELLLA+)
BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2
DOCKET NOS. 50-325 AND 50-324
(TAC NOS. MB6692 AND MB6693)

On November 12, 2002, Carolina Power & Light Company (CP&L), the licensee for the Brunswick Steam Electric Plant (BSEP), Units 1 and 2, submitted a request to revise the Technical Specifications, as necessary, to support an expansion of the core flow operating range (i.e., Maximum Extended Load Line Limit Analysis Plus (MELLLA+) following the extended power uprate (EPU). The MELLLA+ application (NEDC-33063P) states that the anticipated transient without scram (ATWS) suppression pool temperature limit is 207.7°F (See table in Section 9.3.1 of NEDC-33063P). NEDC-33063P also indicates that the suppression pool design limit is 220°F. However, page 20 of the CP&L emergency operating procedure (EOP) document (0EOP-01NL, Rev. 12) shows the Heat Capacity Temperature Limit (HCTL) plot as a function of pressure. For a pressure of 1150 psig, the HCTL is between 120°F and 168°F, depending on the assumed suppression pool water level. In addition, page 109 of 0EOP-01-NL, Rev. 12, states that the suppression pool temperature requiring reactor depressurization is 120°F. It appears that the safety limits stated by NEDC-33063P and 0EOP-01-NL, Rev. 12, are contradictory. The following questions address the capability of BSEP, Units 1 and 2, to maintain containment integrity during ATWS for the EPU/MELLLA+ operation.

(1) Justify the use of the 207.7°F ATWS suppression pool temperature limit for the EPU/MELLLA+ ATWS analysis. Specifically, justify why the suppression pool temperature limit is higher than the temperature limit requiring reactor depressurization.

(2) Provide the basis for the ATWS suppression pool design limit of 220°F quoted in NEDC-33063P.

(3) The peak suppression pool temperature for EPU/MELLLA+ reported in NEDC-33063P is 197.7°F. While this number is below the 207.7°F loss-of-coolant accident (LOCA) limit, the reactor is still at full pressure. Thus, the reported 197.7°F is not the peak temperature but the initial condition prior to the reactor depressurization. Following a depressurization (which is required by the EOP for a temperature of 197.7°F), the suppression pool temperature would be greater than 207.7°F. Provide the actual peak suppression pool temperature when the ATWS transient is followed to completion according to the EOPs.

(4) Provide the assumptions used in the ATWS analysis for the EPU/MELLLA+ specific calculations reported in NEDC-33063P. Specifically, identify which ATWS transient is limiting in terms of each ATWS acceptance limit. Describe the initial conditions, including power, flow, and suppression pool level used. Identify the operator actions that are assumed. Provide the ATWS mitigation actions that are implemented during the transient. Provide the values that are used for the EOP variables (e.g., HCTL, HSBW, etc).

(5) Provide a plot of suppression pool temperature versus time (short-term and long-term).

(6) Provide a plot of reactor pressure and power versus time during the event.

(7) Provide a table showing the sequence of events (system and component actuations and operator actions) during the pressure regulator failure to open (PRFO) and Main Steam Isolation Valve Closure (MSIVC) ATWS events analyzed. Show the safety/relief valve actuation setpoints and indicate the time each safety/relief valve group first actuates.

cc:

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