May 23, 2000

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, UNIT 1 - ISSUANCE OF AMENDMENT REVISING CONTROL ROD TESTING REQUIREMENTS (TAC NO. MA8646)

Dear Mr. Keenan:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 210 to Facility Operating License No. DPR-71 for the Brunswick Steam Electric Plant, Unit 1. The amendment changes the Technical Specifications in response to your submittal dated April 14, 2000, as supplemented April 20, 2000.

The amendment changes Surveillance Requirement 3.1.3.3 to allow partial insertion of control rod 26-47 instead of insertion of one complete notch. This revised acceptance criterion is limited to the current Unit No. 1 operating cycle, after which the original one-notch requirement will be re-established.

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

Sincerely,

/RA/

Allen Hansen, Project Manager, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-325

Enclosures:
1. Amendment No. 210 to License No. DPR-71
2. Safety Evaluation

cc w/enclosures:
See next page

Document Name: G:\PDII-2\Brunswick\BRMA8646amd.wpd
*No substantive change to the safety evaluation

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Name	AHansen	Edunnington	RCaruso*	[illegible]	RCorreia
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON. D.C. 20555-0001

May 23, 2000

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AMENDMENT NO. 210 TO FACILITY OPERATING LICENSE NO. DPR-71 - BRUNSWICK, UNIT 1

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CAROLINA POWER & LIGHT COMPANY, et al.

DOCKET NO. 50-325

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 210
License No. DPR-71

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

   A. The application for amendment filed by Carolina Power & Light Company (the licensee), dated April 14, 2000, as supplemented April 20, 2000, 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 2.C.(2) of Facility Operating License No. DPR-71 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 210 , 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 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Richard P. Correia

Richard P. Correia, 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: May 23, 2000

ATTACHMENT TO LICENSE AMENDMENT NO. 210

FACILITY OPERATING LICENSE NO. DPR-71

DOCKET NO. 50-325

Replace the following page of the Appendix A Technical Specifications with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Remove Page	Insert Page
3.1-10	3.1-10

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.1.3.1	Determine the position of each control rod.	24 hours
SR 3.1.3.2	-------------------NOTE------------------- Not required to be performed until 7 days after the control rod is withdrawn and THERMAL POWER is greater than the LPSP of RWM. -- Insert each fully withdrawn control rod at least one notch.	7 days
SR 3.1.3.3	-------------------NOTES------------------- 1. Not required to be performed until 31 days after the control rod is withdrawn and THERMAL POWER is greater than the LPSP of the RWM. 2. For Cycle 13 only, SR 3.1.3.3 can be satisfied for control rod 26-47 by verifying inward motion versus inserting at least one notch. -- Insert each partially withdrawn control rod at least one notch.	31 days
SR 3.1.3.4	Verify each control rod scram time from fully withdrawn to notch position 06 is $\leq$ 7 seconds.	In accordance with SR 3.1.4.1, SR 3.1.4.2, SR 3.1.4.3, and SR 3.1.4.4

(continued)

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 210 TO FACILITY OPERATING LICENSE NO. DPR-71

CAROLINA POWER & LIGHT COMPANY

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 1

DOCKET NO. 50-325

1.0 INTRODUCTION

By letter dated April 14, 2000, as supplemented April 20, 2000, the Carolina Power & Light Company (the licensee) submitted a request for changes to the Brunswick Steam Electric Plant, Unit 1, Technical Specifications (TS). The requested changes would amend TS Surveillance Requirement (SR) 3.1.3.3 to allow partial insertion of control rod 26-47 instead of insertion of one complete notch. This revised acceptance criterion would be limited to the current Unit No. 1 operating cycle, after which the current one-notch requirement would be re-established. The April 20, 2000, letter provided clarifying information only, and did not expand the scope of the original *Federal Register* notice.

2.0 BACKGROUND

The control rod drive (CRD) system at Brunswick Unit 1 consists of 137 CRD mechanisms (CRDMs) and a hydraulic unit for each drive mechanism. The CRDM is a double acting, mechanically latched, hydraulic piston which uses condensate storage tank water as the operating fluid. Accumulators provide energy for scramming the control rods. An index tube and piston, coupled to the control rod, are locked at fixed increments by a collet mechanism. The collet fingers engage notches in the index tube to prevent accidental withdrawal of the control rod without restricting insertion. Control rod insertion, either manually or automatically, does not require the CRDM to actively control the position of the collet fingers. The tapered design of the index tube allows it to slip through the six collet fingers on an insert or scram signal. A jammed collet piston does not have any effect on the ability of the control rod to scram. Magnetically activated reed switches provide position indication in the control room.

3.0 EVALUATION

During the reactor startup tests for CRDs, the licensee determined that a problem had developed which resulted in an inability to withdraw control rod 26-47. Control rod 26-47 is currently at position 44, with position 48 being fully withdrawn and position 00 being fully inserted. The licensee has concluded that the most likely cause of this inability to withdraw the rod is a seal failure or collet piston ring fouling due to debris ingestion.

Control rod 26-47 has been inserted two notches since start up from the Unit 1 refueling outage. During the 7-day test of the fully withdrawn control rod on March 28, 2000, the control rod notched in normally to position 46. However, it could not be withdrawn to position 48. During the subsequent attempts to withdraw control rod 26-47, the control rod was inserted another notch, to position 44, its current position. Additionally, control rod 26-47 has continued to insert during the brief insert signal supplied during the withdraw attempts. Based on these two insertions and inward movement demonstrated during withdraw attempts, it can reasonably be assumed that there is no degradation in the insertion capability of control rod 26-47. Therefore, the control rod remains operable and can be automatically scrammed or manually inserted, maintaining its capability to perform its safety function.

The purpose of SR 3.1.3.3 is to demonstrate the control rod insertion capability by inserting the withdrawn control rod at least one notch position and observing that the control rod moves. This ensures that the control rod is not stuck and is free to insert on a scram signal. Since control rod 26-47 is at position 44, a partially withdrawn position, SR 3.1.3.3 requires one notch insertion once every 31 days. Since the control rod cannot be withdrawn one full notch, the licensee cannot satisfy the current SR 3.1.3.3 requirement. Brunswick requested a revision to SR 3.1.3.3 to verify the inward motion of the rod rather than insertion of the rod one full notch. The performance of SR 3.1.3.3 would be accomplished by observing the control rod position indication in the control room. The control rod would be inserted sufficiently to cause reed switch movement, as determined by intermediate position indication in the control room, without latching at the next position, and then it would be allowed to settle back into the original position. The proposed revision of SR 3.1.3.3 would change the requirement for control rod 26-47 for the current Cycle 13 only. Since the proposed revision would satisfy the intent of the current SR, it is acceptable.

An integrity test of the coupling between the control rod and its drive mechanism is required only for fully withdrawn control rods. Since control rod 26-47 is at position 44, not at 48, the coupling integrity test is not required to be performed. According to the licensee, control rod 26-47 is located in the periphery of the core and is a "moderate worth" rod and is not the "strongest" operable control rod. Hence, the shutdown margin requirements will still be satisfied. There are 137 CRDMs in Brunswick Unit 1, and at present all of the CRDMs are operable. However, all the CRDMs are not required for the safe shutdown of the plant. Control rod 26-47 is currently operable and can be inserted completely. Even if control rod 26-47 is inoperable, the remaining CRDMs would be sufficient to safely shut down the reactor.

The licensee proposed to add the following note, which is applicable for Cycle 13 operation only:

> For cycle 13 only, SR 3.1.3.3 can be satisfied for control rod 26-47 by verifying inward motion versus inserting at least one notch.

Since the intent of the current SR 3.1.3.3 is still satisfied by the proposed addition of the note, the note is acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the State of North 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 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 Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration (65 FR 21481). The supplemental information submitted on April 20, 2000, provided clarifying information only, and did not change the initial no significant hazards consideration determination. 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 Contributor: G. Thomas

Date: May 23, 2000

cc:

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