August 4, 1989

Docket Nos. 50-325
and 50-324

DISTRIBUTION
See attached list

Mr. Lynn W. Eury
Executive Vice President
Power Supply
Carolina Power & Light Company
Post Office Box 1551
Raleigh, North Carolina 27602

Dear Mr. Eury:

SUBJECT: ISSUANCE OF AMENDMENT NO. 136 TO FACILITY OPERATING LICENSE NO. DPR-71 AND AMENDMENT NO. 166 TO FACILITY OPERATING LICENSE NO. DPR-62 - BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2, REGARDING CONTAINMENT INTEGRATED LEAK RATE TESTING (TAC NOS. 73030 AND 73031)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 136 to Facility Operating License No. DPR-71 and Amendment No. 166 to Facility Operating License No. DPR-62, for Brunswick Steam Electric Plant, Units 1 and 2. The amendments consist of changes to the Technical Specifications in response to your submittal dated May 1, 1989.

The amendments change the Technical Specifications (TS) Section 4.6.1.2, Containment Leakage, by deleting the requirement to use only the mass point method for Type A containment integrated leak rate testing. The amendments allow use of any containment leak rate determination method permitted by Appendix J.

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,

Brenda Mozafari/for

E. G. Tourigny, Senior Project Manager
Project Directorate II-1
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 136 to License No. DPR-71
2. Amendment No. 166 to License No. DPR-62
3. Safety Evaluation

cc w/enclosures:
See next page

[BSEP AMEND 73030/73031]

DF01
1/1

LA:PDII-1	PM:PDII-1	D:PDII-1
PAnderson	ETourigny:bd	EAdensam
06/14/89	06/14/89	06/13/89

Mr. L. W. Eury
Carolina Power & Light Company

Brunswick Steam Electric Plant
Units 1 and 2

cc:

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Raleigh, North Carolina 27626-0520

AMENDMENT NO. 136 TO FACILITY OPERATING LICENSE NO. DPR-71 - BRUNSWICK, UNIT 1
AMENDMENT NO. 166 TO FACILITY OPERATING LICENSE NO. DPR-62 - BRUNSWICK, UNIT 2

Docket File
NRC PDR
Local PDR
PDII-1 Reading
S. Varga (14E4)
G. Lainas
E. Adensam
P. Anderson
E. Tourigny
N. Le
L.Spessard (MNBB 3701)
OGC
D. Hagan (MNBB 3302)
E. Jordan (MNBB 3302)
B. Grimes (9A2)
T. Meeks (8) (P1-137)
W. Jones (P-130A)
J. Calvo
J. Craig
ACRS (10)
GPA/PA
ARM/LFMB

cc: Licensee/Applicant Service List

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

CAROLINA POWER & LIGHT COMPANY, et al.

DOCKET NO. 50-325

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 136
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 May 1, 1989, 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. 136 , 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

Original Signed By:

Elinor G. Adensam, Director
Project Directorate II-1
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 4, 1989

7/13/89

OFC	:LA:PD21:DRPR	:PM:PD21:DRPR	: OGC	:D:PD21:DRPR	:NRR:SPLB	:	:
NAME	: PAnderson	:ETourigny:bd	: S HLewis	: EAdensam	: ~~JCraig~~ C McCracken	:	:
DATE	:06/14/89	:06/1?/89	:07/25/89	: 08/4/89	:7/13/89	:	:

subject to changes

OFFICIAL RECORD COPY

ATTACHMENT TO LICENSE AMENDMENT NO. 136

FACILITY OPERATING LICENSE NO. DPR-71

DOCKET NO. 50-325

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised areas are indicated by marginal lines.

Remove Pages	Insert Pages
3/4 6-3	3/4 6-3
B 3/4 6-1	B 3/4 6-1
B 3/4 6-2	B 3/4 6-2

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

c. The leakage rate to less than or equal to 11.5 scf per hour for any one main steam line isolation valve,

prior to increasing reactor coolant system temperature above 212°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The primary containment leakage rates shall be demonstrated at the following schedule and shall be determined in conformance with the criteria specified in Appendix J of 10 CFR 50:

a. Three Type A Overall Integrated Containment Leakage Rate tests shall be conducted at 40 $\pm$ 10 month intervals during shutdown at P_a, 49 psig, or P_t, 25 psig, during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection.

b. If any periodic Type A test fails to meet either 0.75 L_a or 0.75 L_t, the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet 0.75 L_a or 0.75 L_t, a Type A test shall be performed at each plant shutdown for refueling or every 18 months, whichever occurs first, until two consecutive Type A tests meet 0.75 L_a or 0.75 L_t, at which time the above test schedule may be resumed.

c. The accuracy of each Type A test shall be verified by a supplemental test which:

1. Confirms the accuracy of the test by verifying that the difference between the supplemental data and the Type A test data is within 0.25 L_a or 0.25 L_t.

2. Has duration sufficient to establish accurately the change in leakage rate between the Type A test and the supplemental test.

3. Requires the quantity of gas injected into the containment or bled from the containment during the supplemental test to be equivalent to at least 25 percent of the total measured leakage rate at P_a, 49 psig, or P_t, 25 psig.

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 PRIMARY CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10 CFR Part 100 during accident conditions.

3/4.6.1.2 PRIMARY CONTAINMENT LEAKAGE

The limitations on primary containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure of 49 psig, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a or 0.75 L_t, as applicable, during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

Operating experience with the main steam line isolation valves has indicated that degradation has occasionally occurred in the leak tightness of the valves; therefore, the special requirement for testing these valves.

Exemptions from the requirements of 10 CFR Part 50 have been granted for main steam isolation valve leak testing, testing of airlocks after each opening, and leakage calculation methods.

Appendix J, paragraph III.A.3 requires that all Type A (Containment Integrated Leak Rate) tests be conducted in accordance with American National Standard (ANSI) N45.4-1972, "Leakage Rate Testing of Containment Structures for Nuclear Reactors," March 16, 1972. In addition to the Total Time and Point-to-Point methods described in that standard, the Mass Point method, when used with a test duration of at least 24 hours, is an acceptable method to use to calculate leakage rates. A typical description of the Mass Point method can be found in ANSI/ANS 56.8-1987, "Containment System Leakage Testing Requirements," January 20, 1987. Reduced duration Type A tests may be performed using the criteria and Total Time method specified in Bechtel Topical Report BN-TOP-1, Revision 1, November 1, 1972 (References 1 and 2).

References:

1. CP&L Letter to Mr. D. B. Vassallo, "Integrated Leak Rate Test," October 20, 1983.

2. NRC Letter from Mr. D. B. Vassallo to Mr. E. E. Utley, December 9, 1983.

3/4.6.1.3 PRIMARY CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on PRIMARY CONTAINMENT INTEGRITY and leak rate given in Specifications 3.6.1.1 and 3.6.1.2. The specification makes allowances for the fact that there may be long periods of time when the air locks will be in a closed and secured position during reactor operation.

3/4.6.1.4 PRIMARY CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the primary containment steel vessel will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the vessel will withstand the maximum pressure of 49 psig in the event of a LOCA. A visual inspection in conjunction with Type A leakage tests is sufficient to demonstrate this capability.

3/4.6.1.5 PRIMARY CONTAINMENT INTERNAL PRESSURE

The limitations of primary containment internal pressure ensure that the containment peak pressure of 49 psig does not exceed the design pressure of 62 psig during LOCA conditions. The limit of 1.75 psig for initial positive containment pressure will limit the total pressure to 49 psig, which is less than the design pressure and is consistent with the accident analyses.

3/4.6.1.6 PRIMARY CONTAINMENT AVERAGE AIR TEMPERATURE

The limitation in containment average air temperature ensures that the containment peak air temperature does not exceed the design temperature of 300°F during LOCA conditions and is consistent with the accident analyses.

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

CAROLINA POWER & LIGHT COMPANY, et al.

DOCKET NO. 50-324

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No.166
License No. DPR-62

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

   A. The application for amendment filed by Carolina Power & Light Company (the licensee), dated May 1, 1989, 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-62 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 166, 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

Original Signed By:

Elinor G. Adensam, Director
Project Directorate II-1
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 4, 1989

7/13/89 CQ

OFC	LA:PD21:DRPR	PM:PD21:DRPR	OGC	D:PD21:DRPR	NRR:SPLB		
NAME	PAnderson	ETourigny:bd	S H Lewis	EAdensam	~~SCraig~~ McCracken		
DATE	06/14/89	06/14/89	06/23/89	08/4/89	7/13/89		

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ATTACHMENT TO LICENSE AMENDMENT NO. 166

FACILITY OPERATING LICENSE NO. DPR-62

DOCKET NO. 50-324

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised areas are indicated by marginal lines.

Remove Pages	Insert Pages
3/4 6-3	3/4 6-3
B 3/4 6-1	B 3/4 6-1
B 3/4 6-2	B 3/4 6-2

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

c. The leakage rate to less than or equal to 11.5 scf per hour for any one main steam line isolation valve,

prior to increasing reactor coolant system temperature above 212°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The primary containment leakage rates shall be demonstrated at the following schedule and shall be determined in conformance with the criteria specified in Appendix J of 10CFR50:

a. Three Type A Overall Integrated Containment Leakage Rate tests shall be conducted at 40 ± 10 month intervals during shutdown at P_a, 49 psig, or P_t, 25 psig, during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection.

b. If any periodic Type A test fails to meet either 0.75 L_a or 0.75 L_t, the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet 0.75 L_a or 0.75 L_t, a Type A test shall be performed at each plant shutdown for refueling or every 18 months, whichever occurs first, until two consecutive Type A tests meet 0.75 L_a or 0.75 L_t, at which time the above test schedule may be resumed.

c. The accuracy of each Type A test shall be verified by a supplemental test which:

   1. Confirms the accuracy of the test by verifying that the difference between the supplemental data and the Type A test data is within 0.25 L_a or 0.25 L_t.

   2. Has duration sufficient to establish accurately the change in leakage rate between the Type A test and the supplemental test.

   3. Requires the quantity of gas injected into the containment or bled from the containment during the supplemental test to be equivalent to at least 25 percent of the total measured leakage at P_a, 49 psig or P_t, 25 psig.

3/4.6 CONTAINMENT SYSTEMS

BASES

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 PRIMARY CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10 CFR Part 100 during accident conditions.

3/4.6.1.2 PRIMARY CONTAINMENT LEAKAGE

The limitations on primary containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure of 49 psig, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a or 0.75 L_t, as applicable, during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

Operating experience with the main steam line isolation valves has indicated that degradation has occasionally occurred in the leak tightness of the valves; therefore, the special requirement for testing these valves.

Exemptions from the requirements of 10 CFR Part 50 have been granted for main steam isolation valve leak testing, testing of airlocks after each opening, and leakage calculation methods.

Appendix J, paragraph III.A.3 requires that all Type A (Containment Integrated Leak Rate) tests be conducted in accordance with American National Standard (ANSI) N45.4-1972, "Leakage Rate Testing of Containment Structures for Nuclear Reactors," March 16, 1972. In addition to the Total Time and Point-to-Point methods described in that standard, the Mass Point method, when used with a test duration of at least 24 hours, is an acceptable method to use to calculate leakage rates. A typical description of the Mass Point method can be found in ANSI/ANS 56.8-1987, "Containment System Leakage Testing Requirements," January 20, 1987. Reduced duration Type A tests may be performed using the criteria and Total Time method specified in Bechtel Topical Report BN-TOP-1, Revision 1, November 1, 1972 (References 1 and 2).

References:

1. CP&L Letter to Mr. D. B. Vassallo, "Integrated Leak Rate Test," October 20, 1983.

2. NRC Letter from Mr. D. B. Vassallo to Mr. E. E. Utley, December 9, 1983.

3/4.6.1.3 PRIMARY CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on PRIMARY CONTAINMENT INTEGRITY and leak rate given in Specifications 3.6.1.1 and 3.6.1.2. The specification makes allowances for the fact that there may be long periods of time when the air locks will be in a closed and secured position during reactor operation.

3/4.6.1.4 PRIMARY CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the primary containment steel vessel will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the vessel will withstand the maximum pressure of 49 psig in the event of a LOCA. A visual inspection in conjunction with Type A leakage tests is sufficient to demonstrate this capability.

3/4.6.1.5 PRIMARY CONTAINMENT INTERNAL PRESSURE

The limitations of primary containment internal pressure ensure that the containment peak pressure of 49 psig does not exceed the design pressure of 62 psig during LOCA conditions. The limit of 1.75 psig, for initial positive containment pressure will limit the total pressure to 49 psig, which is less than the design pressure and is consistent with the accident analyses.

3/4.6.1.6 PRIMARY CONTAINMENT AVERAGE AIR TEMPERATURE

The limitation in containment average air temperature ensures that the containment peak air temperature does not exceed the design temperature of 300°F during LOCA conditions and is consistent with the accident analyses.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 136 TO FACILITY OPERATING LICENSE NO. DPR-71

AND AMENDMENT NO. 166 TO FACILITY OPERATING LICENSE NO. DPR-62

CAROLINA POWER & LIGHT COMPANY, et al.

BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2

DOCKET NOS. 50-325 AND 50-324

1.0 INTRODUCTION

By letter dated May 1, 1989, Carolina Power & Light Company (the licensee) requested changes to the Brunswick Steam Electric Plant, Units 1 and 2, Technical Specifications (TS) 4.6.1.2, Containment Leakage, to delete the requirement to use only the mass point method for Type A containment integrated leak rate testing. The above change to the TS has been proposed by the licensee to allow for use of any containment leak rate determination method permitted by 10 CFR Part 50, Appendix J.

2.0 DISCUSSION AND EVALUATION

Appendix J to 10 CFR Part 50 entitled "Primary Reactor Containment Leakage Testing for Water - Cooled Power Reactors" permits three methods for determining containment leakage rates. These are point-to-point, total time, and mass point. Details associated with the point-to-point and total time methods are contained in ANSI N45.4-1972, and the mass point method is addressed in ANSI/ANS 56.8-1981 (revised 1987).

The licensee's present requirement to use only the mass point method, as specified in ANSI/ANS 56.8-1981, for Type A containment integrated leak rate testing was approved by the staff by letter and amendments dated February 17, 1988. Prior to February 17, 1988, the licensee was required by TS to use the point-to-point or total time methods described in ANSI N45.4-1972. The licensee realized after the amendments were issued that the use of the point-to-point or total time method was precluded. The licensee stated in the May 1, 1989 application that this was not intended, although it was what was requested.

The licensee is now requesting to reinstate use of the point-to-point or total time methods in the TS. The licensee has indicated that, when the total time method is utilized, a reduced duration Type A integrated leak rate test may be desirable. If so, it will be conducted using the criteria contained in Bechtel Topical Report BN-TOP-1, Revision 1, dated November 1, 1972. The use of the Bechtel Topical Report at Brunswick was previously approved by the staff in a letter to the licensee dated December 9, 1983. The continued use of BN-TOP-1, Revision 1 at Brunswick is acceptable.

Based upon the above, we conclude that the use of any methodology permitted by Appendix J is acceptable, as is the proposed change to TS 4.6.1.2.

3.0 ENVIRONMENTAL CONSIDERATIONS

These amendments change a requirement with respect to installation or use of a facility component located within the restricted areas as defined in 10 CFR Part 20 and changes to the surveillance requirements. The staff has determined that these amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released off site; and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that these amendments involve no significant hazards consideration, and there has been no public comment on such finding. Accordingly, these amendments meet 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 these amendments.

4.0 CONCLUSION

The Commission made a proposed determination that these amendments involve no significant hazards consideration which was published in the Federal Register (54 FR 25369) on June 14, 1989, and consulted with the State of North Carolina. No public comments or requests for hearing were received, and the State of North Carolina did not have any comments.

The staff 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, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: E. G. Tourigny

Dated: August 4, 1989