Docket Nos. 50-325
and 50-324

November 8, 1977

Carolina Power & Light Company
ATTN: Mr. J. A. Jones
Executive Vice President
336 Fayetteville Street
Raleigh, North Carolina 27602

Gentlemen:

By our letter dated August 5, 1975, we requested that you review the Brunswick Steam Electric Plant for compliance with the requirements of Appendix J to 10 CFR Part 50 and either eliminate areas of non-compliance or request specific exemptions from Appendix J.

In your responses of September 8, 1975, February 9, and September 16, 1977, and in an earlier letter dated May 2, 1975, you requested a number of specific exemptions from Appendix J requirements.

Based on our evaluation of your exemption requests, we have concluded that sufficient justification has been provided for three specific exemptions. In the area of Type C testing of main steam isolation valves, an exemption is granted from the requirements of Section III.C.2 of Appendix J for the purpose of permitting the main steam isolation valves to be locally leak-tested at 25 psig instead of 49 psig (Pa). In the area of testing of air locks, an exemption is granted to the air lock testing requirement of Section III.D.2 of Appendix J for the purpose of (1) extending the testing frequency for the air lock to once every three days during periods of high maintenance in the primary containment when two or more entries are expected per day and (2) limiting the test to the air lock door seals. Finally, an exemption is granted on a one-time basis from the requirement of Section III.D.3 of Appendix J as it applies to Type C testing of the main feedwater check valves at Brunswick for the purpose of deferring Type C testing of these valves until the end of the first major refueling outage of Unit 2 (fall 1977) for Unit 2, and until the end of the first scheduled outage requiring cold shutdown for at least 72 hours after the completion of the Unit 2 refueling outage for Unit 1.

The Commission has further concluded that the granting of the specific exemptions to Appendix J, 10 CFR Part 50, is authorized by law, will not endanger life or property or the common defense and security, and is otherwise in the public interest.

Unit 1

GD

OFFICE➤						
SURNAME➤						
DATE➤						

The Commission has issued the enclosed Amendments Nos. 10 and 36 to Facility Operating Licenses Nos. DPR-71 and DPR-62 for the Brunswick Steam Electric Plant, Units Nos. 1 and 2, which contain Technical Specification changes consistent with these exemptions. In addition, these amendments also reflect your commitment to test the air lock door seals at 49 psig (Pa) as required by Section III.B.2 of Appendix J.

With respect to your exemption request to conduct air lock tests at each refueling outage (instead of at six-month intervals per Appendix J), we find that the justification furnished is not sufficient to support this proposal, and therefore we cannot grant a specific exemption for this item at this time. You may desire to submit additional information and justification for this item for our consideration at some future date. For the present, we consider the issuance of these amendments and this letter to complete Commission action on your requests of May 2, 1975, September 8, 1975, February 9, 1977 and September 16, 1977.

Copies of the Safety Evaluation and Notice of Issuance related to this action are also enclosed.

Sincerely,

Original signed by

Victor Stello, Jr., Director
Division of Operating Reactors

Enclosures:
1. Amendment No. 10 to DPR-71
2. Amendment No. 36 to DPR-62
3. Safety Evaluation
4. Notice of Issuance

cc w/enclosures:
See next page

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NRC FORM 318 (9-76) NRCM 0240

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Atlanta, Georgia 30308

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

CAROLINA POWER & LIGHT COMPANY

DOCKET NO. 50-325

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 10
License No. DPR-71

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

   A. The applications for amendment by Carolina Power & Light Company (the licensee) dated May 2 and September 8, 1975, comply 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 License No. DPR-71 is hereby amended to read as follows:

"(2) Technical Specifications

The Technical Specifications contained in Appendices A, A-Prime, and B, as revised through Amendment No. 10, are hereby incorporated in this license. Appendix A shall be effective from the date of issuance of the Unit 1 operating license until the Appendix A-Prime becomes effective on or before the initial criticality of Brunswick Unit 2 following its initial refueling outage. Carolina Power & Light Company shall operate the facility in accordance with the Technical Specifications as indicated above. The licensee shall inform the Office of Inspection and Enforcement, Region II, of the date that the Appendix A-Prime becomes effective."

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 8, 1977

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

CAROLINA POWER & LIGHT COMPANY

DOCKET NO. 50-324

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 36
License No. DPR-62

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

   A. The applications for amendment by Carolina Power & Light Company (the licensee) dated May 2 and September 8, 1975, comply 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 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. 36, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 8, 1977

ATTACHMENT TO LICENSE AMENDMENT NO. 10 AND NO. 36

FACILITY OPERATING LICENSE NO. DPR-71 AND NO. DPR-62

DOCKETS NOS. 50-325 AND 50-324

Revise Appendix A as follows:

Remove pages 3.7-5 and 3.7-6 and replace with identically numbered pages.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.7.A.2 Containment Leak Rate Testing (Cont'd)	4.7.A.2 Containment Leak Rate Testing (Cont'd)
d. The combined leakage rate of all penetrations and isolation valves shall not exceed 60 percent of the design basis accident leakage rate L_a. No one steam isolation valve shall have a leakage rate in excess of 11.5 scf/hr @ 25 psig. Air lock leakage shall not exceed 5% of L_a.	d. (1) Primary containment testable penetrations shall be tested at 49 psig during each reactor shutdown for major fuel reloading but in no case greater than two-year intervals. (2) Isolation valves shall be tested at 49 psig during each major fuel reloading. (3) The main steamline isolation valves shall be tested at 25 psig* at the beginning of each fuel cycle. For the first two fuel cycles, the valves shall be tested two additional times during each fuel cycle, one of which should be performed at an outage near the middle cycle. If the results of these tests show that the valves meet the requirements of Specification 4.7.D.3, this test frequency shall be reduced to once during each refueling outage.** If acceptable leak rates cannot be obtained, appropriate modifications of equipment or procedures will be implemented to ensure that radioactive leakage through these valves will be within the requirements of Specification 4.7.D.3.

*Exemption to Appendix J, 10 CFR 50
**But in no case shall the test interval exceed 2 years.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.7.A.2.d Containment Leak Rate Testing (Cont'd)	4.7.A.2.d Containment Leak Rate Testing (Cont'd) (4) Bolted double-gasketed seals shall be tested whenever the seal is closed after being opened but in no case will the testing interval exceed one year. (5) Personnel airlock seals shall be tested at 49 psig whenever the doors are closed after opening. However, during periods of high maintenance in the primary containment when 2 or more entries are expected per day, personnel air lock seals shall be tested once every 3 days*. An overall air lock leakage test at 49 psig shall be conducted at least once per 6 months. (6) The drywell-suppression chamber vacuum breakers leakage shall be tested during each refueling outage by pressurizing the drywell to approximately one psig and then measuring the subsequent suppression chamber pressure response. The test objective will be to detect any leak path between drywell and suppression chamber whose capacity equals or exceeds that of a one-inch diameter orifice. e. Postoperational - Continuous Leak Rate Monitor When the primary containment is inerted, the containment shall be monitored for gross leakage by review of the inerting system makeup requirements. This monitoring system may be taken out of service for maintenance but shall be returned to service as soon as practicable.

*Exemption to Appendix J 10 CFR 50.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 10 TO FACILITY LICENSE NO. DPR-71

AMENDMENT NO. 36 TO FACILITY LICENSE NO. DPR-62

CAROLINA POWER & LIGHT COMPANY

BRUNSWICK STEAM ELECTRIC PLANT, UNITS NOS. 1 AND 2

DOCKETS NOS. 50-325 AND 50-324

Introduction

By letter dated May 2, 1975, Carolina Power & Light Company (CP&L) requested a specific exemption to a 10 CFR 50 Appendix J air lock testing requirement. By our letter dated August 5, 1975, CP&L was requested to review the Brunswick Steam Electric Plant (BSEP) in terms of the current containment leak testing program and the associated Technical Specifications for compliance with the requirements of Appendix J to 10 CFR Part 50. CP&L was also requested to determine the planned actions and their associated schedule for attaining conformance with the above cited regulation and for eliminating any inconsistencies by implementing design changes or proposing Technical Specification changes or specific exemptions from the regulation.

Appendix J to 10 CFR 50 was published on February 14, 1973. Since many operating nuclear plants had either received an operating license or were in advanced stages of design or construction at that time, it was possible that some plants may not have been in full compliance with the requirements of this regulation. Therefore, beginning in August 1975, (August 5, 1975 for Brunswick) NRC requests for review of the extent of compliance with the requirements of Appendix J were made of each licensee. CP&L responded on September 8, 1975, February 9 and September 16, 1977. Following the initial responses to these requests, NRC staff positions were developed regarding the extent to which current leak testing practices satisfy the requirements of Appendix J. These staff positions were applied in our review of the May 2 and September 8, 1975, February 9 and September 16, 1977, exemption requests filed by CP&L.

The results of our review are reflected in the following evaluation.

Evaluation

1. Testing of Main Steam Isolation Valves (MSIVs)

CP&L has requested an exemption to allow continued use of a 25 psig test pressure for the MSIVs, as stated in the approved Brunswick Technical Specifications, Sections 4.7.A.2.b.3 and 4.7.A.2.d.3 instead of peak calculated containment pressure, Pa (49 psig), required by Appendix J.

Section III.C.2 of Appendix J requires that containment isolation valves be locally leak tested (Type C) at the peak calculated containment pressure (Pa). The main steam system design in most operating BWR plants necessitates leak testing of the MSIVs by pressurizing between the valves. The MSIVs are angled in the main steam lines to afford better sealing in the direction of accident leakage. A test pressure of Pa acting on the inboard disc lifts the disc off the seat resulting in excessive leakage into the reactor vessel. Consideration was given to this matter by the NRC staff when the original test pressure of 25 psig was established for the MSIVs at the design stage of the plant. Since testing of the MSIVs at a reduced pressure between the valves gives rise to greater leakage than Pa applied upstream of the valves, the testing procedure results in a conservative determination of the leakage rate through the valves, so that we find the proposed exemption acceptable.

2. Testing of Air Locks

CP&L has requested an exemption to extend the testing frequency for the primary containment personnel airlock seals to once every three days during periods of high maintenance activity in the primary containment, instead of testing after each opening as required by Appendix J. Moreover, in its September 8 submittal, CP&L has revised the test pressure for the air locks to Pa (49 psig) which will conform to Appendix J requirements. CP&L originally proposed these tests be at 10 psig in its May 2, 1975 submittal. In addition, CP&L has proposed to test the total air lock system at every refueling outage.

Section III.B.2 of Appendix J requires that reactor containment airlocks be leak-tested at the peak calculated accident pressure (Pa) at six month intervals and should the airlocks be opened during such intervals, Section III.D.2 of Appendix J requires the

airlocks to be leak tested after each opening. Appendix J calls out these specific requirements for airlocks because they represent a potentially large leakage path that is more subject to human error than other isolation barriers.

The objectives of the Appendix J airlock leak testing requirements are: (1) that the six month test provide an integrated leakage rate for the entire airlock assembly, including electrical and mechanical penetrations, the airlock cylinder, hinge assemblies, welded connections, and other potential leakage paths; and (2) that the "after each opening" test provide a means of assuring that the door seals had not been damaged or sealed improperly during airlock use.

We have concluded that leak testing the airlock after each opening is an impractical requirement considering the time required to perform the test in relation to the frequency of airlock use. We find that testing the airlock seals within three days following the first in a series of openings will adequately demonstrate their continued integrity. We, therefore, conclude that CP&L's proposed exemption from the "after each opening" test frequency requirement in Appendix J Section III.D.2 is acceptable. As required by Appendix J, CP&L has proposed to perform the airlock leak tests at a pressure of Pa (49 psig). However, CP&L has not provided sufficient justification to support the proposed exemption which would allow the integrated airlock leak test to be performed at each refueling outage rather than every six months. Therefore, the requirement to test at six-month intervals is being retained.

3. Feedwater Check Valves

In its September 8, 1975 letter, CP&L requested an exemption from the requirements for Type C testing of feedwater check valves at BSEP on the basis that the design does not permit the performance of a local leak rate test because sufficient presure cannot be introduced at the test connection to seat the valves. Section II.H.4 of Appendix J defines the valves in BWR feedwater lines as isolation valves subject to Type C tests. CP&L proposed to test the valves as part of the required integrated leak rate test instead.

In later letters dated February 9 and September 16, 1977, CP&L modified its original request and requested instead an exemption from the local leak rate testing requirement until the first refueling outage of Brunswick Unit 2, scheduled for the fall of 1977. This would permit CP&L both the time and opportunity to develop a testing procedure to meet the requirements of Appendix J. For Unit 1, CP&L requested an exemption until the end of the first scheduled cold shutdown outage of greater than 72 hours duration following the completion of the Unit 2 refueling outage.

Appendix J requires that Type C tests be performed at each reactor shutdown for refueling but in no case at intervals greater than 2 years. The Type C test for the feedwater check valves for Unit 2 was due to be conducted in December 1976, but was overlooked because the original pre-operational test program did not include a test of these valves. When this was discovered on January 24, 1977, NRC was promptly informed, and the February 9, 1977 letter sent explaining the circumstances and requesting a modified exemption (an extension of the time interval for testing).

The feedwater lines at BSEP each contain a stop-check valve and a check valve. The Unit 2 stop-check valves successfully passed Type C testing in October 1976. In addition, the containment was subjected to an integrated leak rate test in 1974 which was also successful. These same tests were successfully conducted on Unit 1 in the summer of 1976.

In consideration of these factors, we conclude that the ability to isolate the feedwater line in the unlikely event of a loss-of-coolant accident has ben adequately demonstrated for the present, and that the CP&L proposal to defer the Type C test of the check valves as discussed above is acceptable.

4. Environmental Consideration

We have determined that the amendments do not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendments involve an action which is insignificant from the standpont of environmental impact and pursuant to 10 CFR § 51.5(d)(4) that an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of these amendments.

5. Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendments do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin, the amendments do not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Date: November 8, 1977

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKETS NOS. 50-325 AND 50-324

CAROLINA POWER & LIGHT COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

NOTICE OF GRANTING OF SPECIFIC EXEMPTIONS FROM REGULATIONS IN 10 CFR PART 50 APPENDIX J

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendments Nos. 10 and 36 to Facility Operating Licenses Nos. DPR-71 and DPR-62, respectively, issued to Carolina Power & Light Company (the licensee), which revised Technical Specifications for operation of the Brunswick Steam Electric Plant, Units Nos. 1 and 2 (the facilities) located in Brunswick County, North Carolina. The amendments are effective as of their date of issuance. The Commission has also granted three specific exemptions from the regulations in 10 CFR Part 50, Appendix J, "Primary Reactor Containment Leakage Testing For Water-Cooled Power Reactors".

The amendments and associated specific exemptions (1) allow the continued leakage testing of main steam isolation valves at a test presure of 25 pounds per square inch (gauge) (psig), (2) require that the containment personnel air lock door seal be tested at 49 psig after each opening except during maintenance periods involving frequent air lock usage, in which case the door seal shall be tested once every three days and (3) allow the licensee

to defer local leakage rate (Type C) testing of the feedwater check valves until the first refueling outage (scheduled for fall 1977) for Unit 2, and defer such testing for Unit 1 until the first cold shutdown outage of at least 72 hours duration following the refueling outage of Unit 2. An overall airlock leakage test will be conducted at 49 psig at least once per six months, consistent with Appendix J.

The applications for the amendments and specific exemptions comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments and related letter to the licensee. Prior public notice of the amendments were not required since the amendments do not involve a significant hazards consideration.

The Commission has determined that the issuance of these amendments will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of these amendments.

For further details with respect to this action, see (1) the applications for amendment and exemptions dated May 2 and September 8, 1975, February 9 and September 16, 1977, (2) Amendments Nos. 10 and 36 to Licenses Nos. DPR-71 and DPR-62, respectively, and (3) the Commission's

related Safety Evaluation and letter to the licensee dated 1977. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Southport-Brunswick County Library, 109 W. Moore Street, Southport, North Carolina 28461. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 8th day of November 1977.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors