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Docket Nos. 50-325
and 50-324

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

Gentlemen:

The Commission has issued the enclosed Amendment No. 12 to Facility Operating License No. DPR-71 and Amendment No. 39 to Facility Operating License No. DPR-62 for the Brunswick Steam Electric Plant, Unit Nos. 1 and 2, respectively. The amendments consist of changes to the operating licenses and to the Technical Specifications in response to your requests dated August 22 and September 14, 1977.

The amendments involve many revisions to the Standard Technical Specifications for Unit 1 and incorporate Standard Technical Specifications similar to those for Unit 1 into the operating license for Unit 2.

Copies of the Safety Evaluation and Notice of Issuance are also enclosed.

Sincerely,

151

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

Enclosures:

- 1. Amendment No. 12 to DPR-71 / see 50-325 for # 1)
 - 2. Amendment No. 39 to DPR-62
 - 3. Safety Evaluation
 - 4. Notice
- See tech spec*

DEisenhut
ACRS(16)
CMiles
DRoss
TBABernathy
JRBuchanan

cc w/encl: See next page

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60

OFFICE >	DOR:ORB#1	DOR:ORB#1				
SURNAME >	CMTrammell:1b	ASchwencer				
DATE >	11/23/77	11/ /77				



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

November 23, 1977

Docket Nos. 50-325
and 50-324

Carolina Power & Light Company
ATTN: Mr. J. A. Jones
Executive Vice President
336 Fayetteville Street
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The amendments involve many revisions to the Standard Technical Specifications for Unit 1 and incorporate Standard Technical Specifications similar to those for Unit 1 into the operating license for Unit 2.

Copies of the Safety Evaluation and Notice of Issuance are also enclosed.

Sincerely,

A handwritten signature in cursive script, appearing to read "A. Schwencer".

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

Enclosures:

1. Amendment No. 12 to DPR-71
2. Amendment No. 39 to DPR-62
3. Safety Evaluation
4. Notice

cc w/encl: See next page

Carolina Power & Light Company - 2 - November 23, 1977

cc: Richard E. Jones, Esquire
Carolina Power & Light Company
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Southport - Brunswick County Library
109 W. Moore Street
Southport, North Carolina 28461

Mr. Steve J. Varnam
Chairman, Board of County
Commissioners of Brunswick County
Southport, North Carolina 28461

Office of Intergovernmental
Relations
116 West Jones Street
Raleigh, North Carolina 27603

Chief, Energy Systems
Analyses Branch (AW-459)
Office of Radiation Programs
U.S. Environmental Protection Agency
Room 645, East Tower
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Washington, D.C. 20460

U.S. Environmental Protection Agency
Region IV Office
ATTN: EIS COORDINATOR
345 Courtland Street, NW
Atlanta, Georgia 30308



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

CAROLINA POWER AND LIGHT COMPANY

DOCKET NO. 50-325

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 12
License No. DPR-71

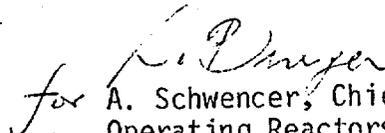
1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The applications for amendment by Carolina Power & Light Company (the licensee) dated August 22 and September 14, 1977, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter 1;
 - B. The facility will operate in conformity with the applications, 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 Appendix A (issued September 1976), Appendix A (issued November 23, 1977) and B, as revised through Amendment No.12, are hereby incorporated in this license. Appendix A (issued September 1976) shall be effective from the date of issuance of the Unit 1 operating license until the Appendix A (issued November 23, 1977) 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 Appendix A (issued November 23, 1977) becomes effective."

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

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment: -
Changes to the Technical
Specifications

Date of Issuance: November 23, 1977

ATTACHMENT TO LICENSE AMENDMENT NO. 12

FACILITY OPERATING LICENSE NO. DPR-71

DOCKET NO. 50-325

Revise the Technical Specifications as follows:

Discard Appendix A-Prime Technical Specifications in their entirety,
and add Appendix A Technical Specifications (issued November 23,
1977).



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

CAROLINA POWER AND LIGHT COMPANY

DOCKET NO. 50-324

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 39
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 August 22 and September 14, 1977, 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 applications, 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. 39, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective on or before Brunswick Unit No. 2 achieves initial criticality following its first refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 23, 1977

ATTACHMENT TO LICENSE AMENDMENT NO. 39

FACILITY OPERATING LICENSE NO. DPR-62

DOCKET NO. 50-324

Revise Appendix A Technical Specifications as follows:

Discard Appendix A (issued December 27, 1974) in its entirety, and replace with Appendix A Technical Specifications (issued November 23, 1977).



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
SUPPORTING AMENDMENT NO. 12 TO FACILITY OPERATING LICENSE NO. DPR-71
AND AMENDMENT NO. 39 TO FACILITY OPERATING LICENSE NO. DPR-62
CAROLINA POWER AND LIGHT COMPANY
BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 & 2
DOCKET NOS. 50-325 AND 50-324

INTRODUCTION

By letters dated August 22 and September 14, 1977, Carolina Power & Light Company (CP&L) requested numerous changes to the Standard Technical Specifications for Unit 1 (identified as Appendix A-Prime to Facility Operating License No. DPR-71) and also requested that Standard Technical Specifications similar to those for Unit 1 be issued for Unit 2.

The licensee has operated Brunswick Unit 1 in accordance with Appendix A custom Technical Specifications which were issued with the Facility Operating License in September 1976. At the same time, Appendix A-Prime Standard Technical Specifications were issued to be used (but not in effect) on a trial basis. Brunswick Unit 2 was licensed and has operated since December 1974 with Appendix A custom Technical Specifications.

Until the Brunswick Steam Electric Plant Unit 1 Appendix A "Prime" Technical Specifications were issued in September 1976, Technical Specifications for each boiling water reactor were developed by the particular applicant for staff approval and, therefore, reflected each applicant's approach to plant operation. In order to attain greater uniformity in the Technical Specifications for similar plants, the staff and the industry cooperated in the development of General Electric Boiling Water Reactor Standard Technical Specifications (GE-STTS) whose format and general content could be applied to all plants produced by the General Electric Company. Only minor variations need to be made to the GE-STTS to permit their application to a particular plant.

Brunswick Steam Electric Plant (Brunswick) Unit 1 was the first plant to which the General Electric Boiling Water Reactor Standard Technical Specifications were applied. On September 8, 1976, Facility Operating License DPR-71 incorporated these specifications as Appendix A "PRIME" to the license to be effective upon the startup of Brunswick Unit 2 from its first refueling outage, now scheduled to be completed in November 1977. It was intended that the Brunswick Unit 2 license would be amended

to include nearly identical Appendix A "Prime" Technical Specifications based on the GE-STS. Both would be effective at the same time.

The staff and the licensee both recognized at the time Unit 1 Standard Technical Specifications were issued that they might need to be changed as operating experience brought to light such things as inconsistencies between specifications that could not be envisioned before the specifications were actually used in operation. Such, in fact, was the case, and some changes have been made from time to time even though the Appendix A "Prime" Technical Specifications were not in effect.

Since the time Brunswick Unit 1 Appendix A "Prime" Technical Specifications were issued, further improvements in the GE-STS have been made. These improvements continue the elimination of inconsistencies and clarify the requirements so that they reflect the original intent of the specifications with less chance for misinterpretation. In order to provide Brunswick Unit 1 with the most up-to-date GE-STS consistent with its design and so that Unit 1 Technical Specifications would be consistent with Brunswick Unit 2 Technical Specifications (which are based on the latest GE-STS), the Brunswick Unit 1 Technical Specifications have been completely revised and are being reissued herewith.

When a large number of changes is contemplated, the Technical Specifications may be reprinted in toto, including both revised pages and unrevised pages. The revision is identified by license amendment number and date on the cover page. In the present instance, the Brunswick Unit 1 Technical Specifications have been extensively revised. All changes have been reviewed and concurred in by the licensee. The changes include:

1. Changes to clarify specifications that were difficult to understand and could be misinterpreted as to their application and intent.
2. Changes to correct errors that have been discovered since the original issue.
3. Changes to make specifications consistent with each other.
4. Changes in reporting requirements as reflected by the latest revision to Regulatory Guide 1.16.
5. Changes in frequencies with which surveillance must be performed, in out-of-service time permitted before an action must be taken, and in the time before an event that certain surveillance requirements must be satisfied. Many of the times and frequencies originally specified were arbitrary; operating experience indicates that these times can be adjusted to provide time for more orderly and thorough planning and accomplishment of the required tasks and reduce the radiation exposure of plant personnel without a significant impact on safety.

6. Changes to bring surveillance requirements into conformance with the most recent or applicable editions of codes, standards, and Regulatory Guides.

The above changes result in Technical Specifications for Brunswick Units 1 and 2 that are consistent with those now being issued for new GE Boiling Water Reactors.

The following changes were proposed in the August 22, 1977 letter. All page numbers refer to the Appendix A "Prime" Technical Specifications.

1. The typographical or editorial errors have been corrected in these Technical Specifications without further reference in this Safety Evaluation as they have been verified to be typographical or editorial errors and involve no significant change in the Appendix A "Prime" Technical Specifications.
2. The proposed changes on pages 3/4 5-5 and 3/4 7-1 to delete redundant reference to Specification 4.0.5 were included in these Technical Specifications for clarity and consistency.
3. The proposed changes on pages 3/4 3-10, 3/4 3-11, 3/4 3-14, 3/4 3-15, 3/4 3-17, 3/4 3-17a, 3/4 3-19 and 3/4 3-20 were included in these Technical Specifications to properly identify the instrumentation.
4. The proposed change on page 3/4 9-8 was included in these Technical Specifications to clarify that the intent of the water level requirement is relative to the height above the active fuel in the fuel assembly.
5. The proposed change on page 3/4 9-7 was included in these Technical Specifications to permit the necessary installation and removal of a spent fuel shipping cask head on a cask containing fuel.
6. The proposed change on page 6-4 to Table 6.2-1 has been withdrawn by the licensee.
7. The proposed changes on pages 3/4 3-14 and 3/4 3-40 relating to changing the instrumentation surveillance requirements for consistency within the Technical Specifications was included in these Technical Specifications.
8. The proposed change on pages 3/4 3-39 and 3/4 3-40 to delete the requirement for a drywell hydrogen concentration monitor because neither Unit has such a monitor was included in these Technical Specifications.

9. The proposed change on page 3/4 4-2 has been withdrawn by the licensee.
10. The proposed change on page 3/4 6-4 to establish a specific acceptable airlock door seal leakage rate is conservative with respect to the allowable airlock leakrate and is included in these Technical Specifications.
11. The proposed change on page 3/4 9-6 has been withdrawn by the licensee.
12. The proposed change to delete Section 6.12 of the Administrative Controls because of the issuance of the revised 10 CFR 20.103 has been included in these Technical Specifications.

The following changes were proposed in the September 14, 1977 letter. All page numbers refer to the Appendix A "Prime" Technical Specifications.

1. The proposed change on page 3/4 1-17 removing the limitation that system operability can only be demonstrated in the recirculation mode was included in these Technical Specifications. Operability can be demonstrated adequately without being in the recirculating mode.
2. The proposed change on page 3/4 3-9 that a note permitting one channel to be inoperable for up to 2 hours for testing of isolation instrumentation was included in these Technical Specifications. This change is consistent with the GE-STs and was inadvertently omitted from the Unit 1 Appendix A "Prime" Technical Specifications.
3. The proposed change on page 3/4 3-45 to reduce a chloride intrusion leak detector monitor alarm setpoint from 20 $\mu\text{mhos/cm}$ to 2.0 $\mu\text{mhos/cm}$ is an increase in conservatism and provides an alarm point with more margin.

Other changes have been made in the Brunswick Standard Technical Specifications, and these changes are discussed in the following Evaluation. The NRC staff's conclusions regarding these changes are given following the Evaluation.

Carolina Power and Light Company and the NRC staff are in agreement that the proposed new STS should become effective on or before Brunswick Unit 2 achieves initial criticality following the Unit 2 refueling outage which started in September 1977.

EVALUATION

We have evaluated the revised Appendix A "Prime" Technical Specifications* to verify that all requirements of the present Appendix A "Prime" Technical Specifications significant to reactor safety are retained. In addition, we have assured that new requirements were added, where necessary, to make the revised Appendix A "Prime" Technical Specifications consistent with the latest requirements of the GE-STs which are applicable to the design of Brunswick Units 1 and 2.

Our evaluation was based on existing information on the Brunswick docket, reports of operational experience, the present Technical Specifications, and current Commission requirements.

Section 1.0 - Definitions

This section includes standard definitions of terms used in the General Electric Standard Technical Specifications, modified as necessary for the Brunswick design. No significant changes were made.

Section 2.0 - Safety Limits and Limiting Safety System Settings

The Technical Specifications include Safety Limits, Limiting Safety System Settings and bases as currently required to emphasize the importance and purpose of these requirements. No changes were made to these limits except for the addition of a Safety Limit for reactor vessel water level during shutdown and refueling conditions to ensure capability to remove decay heat from irradiated fuel so as to prevent elevated cladding temperatures and cladding perforation.

Limiting Conditions For Operation and Surveillance Requirements

The Limiting Conditions For Operation (LCO) and Surveillance Requirements (SR) in the Technical Specifications are based on the Final Safety Analysis Report, as amended, and the present custom and Appendix A "Prime" Technical Specifications. Additions were made to account for operating experience; guidance provided by Regulatory Guides, where applicable; regulations and current Commission requirements. The most significant improvements in the Technical Specifications are clear delineation of actions to be taken in the event that required conditions cannot be met; testing of systems and equipment important to safety and verification of compliance with specified limits.

* The "A-Prime" Technical Specifications are redesignated "Appendix A" Technical Specifications for both Units in the amendments accompanying this Safety Evaluation.

Sections 3/4.1 Reactivity Control Systems

These sections contain LCO's and SR's for reactivity control and control rods. These LCO's are consistent with current requirements and include all significant requirements of the present Appendix A "Prime" Technical Specifications. Action statements have been added or changed consistent with current requirements. SR's on types and frequency of tests to demonstrate compliance with the LCO's have also been added or changed to be consistent with current requirements.

Sections 3/4.2 Power Distribution Limits

These sections contain LCO's and SR's limiting reactor power distribution. The limitations and surveillance are substantially the same as in the present Appendix A "Prime" Technical Specifications.

Sections 3/4.3 Instrumentation

These sections contain LCO's and SR's for the Reactor Protection System (RPS), Isolation Actuation System, Emergency Core Cooling System (ECCS), and for other instrumentation required for safety. These LCO's are consistent with current requirements and include all significant requirements of the present Appendix A "Prime" Technical Specifications.

The trip setpoints are the same as presently specified. Action statements have been revised, where necessary, consistent with current requirements. SR's on types and frequency of tests to demonstrate compliance with the LCO's have been augmented, where required, consistent with current requirements.

Sections 3/4.4 Reactor Coolant System

These sections contain LCO's and SR's for recirculation system operation, safety valves, leakage detection and limits, reactor coolant chemistry and specific activity, reactor coolant system pressure/temperature limits, main steam line isolation valves and reactor coolant system structural integrity. The LCO's are consistent with current requirements and include all significant requirements of the present Appendix A "Prime" Technical Specifications.

Action statements have been revised, where necessary, consistent with current requirements. SR's on types and frequency of tests to demonstrate compliance with the LCO's have been augmented, where required, consistent with current requirements.

Sections 3/4.5 Emergency Core Cooling Systems

These sections contain LCO's and SR's for the high and low pressure coolant injection systems, the source of injection water and the automatic

depressurization system. The LCO's contained in these sections are the same as but more explicit than specified in the present Appendix A "Prime" Technical Specifications.

Action statements have been revised, where necessary consistent with current requirements. SR's on types and frequency of tests to demonstrate compliance with the LCO's have been augmented, where required, consistent with current requirements.

Sections 3/4.6 Containment Systems

These sections contain LCO's and SR's for containment integrity, containment leak rate, air locks, internal pressure and temperature, structural integrity, depressurization and cooling systems, isolation and vacuum relief valves, and containment atmosphere control. These LCO's include all significant requirements in the present Appendix A "Prime" Technical Specifications but are more explicit.

LCO's have been added which require inservice inspection of the containment structure and limit containment internal pressure and temperature during operation. Action statements have been revised, where necessary, consistent with current requirements. SR's on types and frequency of tests to demonstrate compliance with the LCO's have been augmented, where required, and SR's have also been added for the new LCO's, consistent with current requirements.

Sections 3/4.7 Plant Systems

These sections contain LCO's and SR's for service water systems, control room isolation, flood protection, reactor core isolation cooling system, hydraulic snubbers, and sealed source contamination. These LCO's are consistent with current requirements and include all significant requirements of the present Appendix A "Prime" Specifications but are more explicit.

An LCO has been added which requires the service water system to be operable when safety related equipment requiring the service water is required to be operable. The LCO requiring reactor vessel water level to be above the top of irradiated fuel in the reactor vessel during shutdown and refueling has been deleted because this requirement is now a Safety Limit. Action statements have been revised, where necessary, consistent with current requirements. SR's on types and frequency of tests to demonstrate compliance with LCO's have been augmented, where required, consistent with current requirements.

Sections 3/4.8 Electrical Power Systems

These sections contain LCO's and SR's for offsite and onsite electrical power systems, including the diesel generator supplied onsite AC system and

the battery supplied onsite AC/DC system. The requirements of these LCO's are substantially the same as the present Appendix A "Prime" Technical Specifications. These LCO's are consistent with current requirements and include all significant requirements of the present Appendix A "Prime" Technical Specifications.

The LCO's have been revised to require the electrical power systems to be operable consistent with the operability requirements of safety related equipment. Action statements have been revised, where necessary, consistent with current requirements. SR's on types and frequency of tests to demonstrate compliance with the LCO's have been augmented, where required.

Sections 3/4.9 Refueling

These sections satisfy special requirements during the refueling mode to minimize the likelihood of inadvertent criticality, to assure adequate monitoring during core change operations, and to assure adequate shielding and cooling of the irradiated fuel. The LCO's and SR's involve requirements related to the reactor mode switch, neutron monitoring instrumentation, control rod position, decay time, communications, crane and hoist operability, crane travel, reactor vessel and spent fuel pool water levels, and control rod removal. The LCO's include all significant requirements in the present Appendix A "Prime" Technical Specifications and are consistent with current requirements.

Action statements have been revised, where required, consistent with current requirements. SR's on type and frequency of tests to demonstrate compliance with the LCO's have been augmented, where required, consistent with current requirements.

Sections 3/4.10 Special Test Exceptions

These sections provide special exceptions under which certain requirements of the Technical Specification may be temporarily waived so as to be able to perform certain required tests. Compensating safeguards have been added. The special test exceptions are substantially the same as contained in Appendix A "Prime" Technical Specifications except that the special test exception for oxygen concentration has been deleted because the provisions have been made a part of the LCO on containment oxygen concentration. A special test exception has been added to provide specific safeguards against accidental criticality during the measurement of shutdown margin during refueling.

Section 5.0 - Design Features

The Design Features section contains significant design information in the following subject areas: site, containment, reactor core, reactor coolant

system, fuel storage, meteorological tower location and component cyclic or transient limits. The subject areas are selected in accordance with 10 CFR Part 50.36(c)(4) and contain only those design features which, if altered or modified, would have a significant effect on safety and are not covered by other sections of the Technical Specifications. The Design Features section contains no significant changes, but has been slightly rearranged to be consistent with the GE-STs.

Section 6.0 - Administrative Controls

The Administrative Controls section contains provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner. All requirements of the present Appendix A "Prime" Technical Specifications have been retained and new requirements have been added, where necessary, consistent with current requirements.

The minimum shift crew requirements of Table 6.2-1 have been revised to account for all possible combinations of operational conditions for the two units.

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 standpoint of environmental impact and pursuant to 10 CFR §51.5(d)(4) that an environmental statement, or negative declaration and environmental appraisal need not be prepared in connection with the issuance of these amendments.

CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) because the changes 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 changes 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 23, 1977

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-325 AND 50-324

CAROLINA POWER AND LIGHT COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY
OPERATING LICENSES

The U.S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 12 to Facility Operating License No. DPR-71 and Amendment No. 39 to Facility Operating License No. DPR-62, issued to Carolina Power and Light Company (the licensee), which revised Technical Specifications for operation of Brunswick Steam Electric Plant, Unit Nos. 1 and 2 (the facility) located in Brunswick County, North Carolina. The changes become effective on or before Brunswick Unit No. 2 achieves initial criticality following its first refueling outage (on or about November 26, 1977).

The amendment for Unit 1 consists of numerous changes to the Standard Technical Specifications issued in tandem with Custom Technical Specifications. The Custom Technical Specifications were used for facility operation from September 1976 to the present. The Standard Technical Specifications were not placed in effect when originally issued in order to allow a suitable trial use period. The amendment for Unit 1 eliminates inconsistencies and corrects errors discovered during the trial use period. In addition, limiting conditions for operation and surveillance requirements have been modified to be consistent with NRC Staff requirements. For Unit 2, the amendment incorporates Standard Technical Specifications similar to those for Unit 1 for the first time.

The applications for the amendments 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. Prior public notice of these amendments was 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 dated August 22 and September 14, 1977, (2) Amendment No. 12 to License No. DPR-71, (3) Amendment No. 39 to License No. DPR-62, and (4) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, NW., Washington, D.C. and at the Southport-Brunswick County Library, 109 West Moore Street, Southport, North Carolina 28461. A copy of items (2), (3) and (4) 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 23rd day of November 1977.

FOR THE NUCLEAR REGULATORY COMMISSION

Alfred Burger
Alfred Burger, Acting Chief
Operating Reactors Branch #1
Division of Operating Reactors

File - 50-324

Amendment No. 38

Dated November 23, 1977

Am-38
11/23/77

PRELIMINARY DETERMINATION

NOTICING OF PROPOSED LICENSING AMENDMENT

LICENSEE: Carolina Power & Light Co. Brunswick 2
REQUEST FOR: Tech. Spec. changes associated with
ECCS reanalysis per our March 11,
1977 Order.

REQUEST DATE: September 22, 1977

PROPOSED ACTION: Pre-notice Recommended
 Post-notice Recommended
 Determination delayed pending
completion of Safety Evaluation

BASIS FOR DECISION:

On March 11, 1977, we ordered CP&L to perform a reanalysis of ECCS to correct errors outlined in the Order.

CP&L is now refueling and has submitted a reanalysis which corrects the errors, incorporate model changes, and, as a result allows substantially higher MAFHR for the recycled 2X7 fuel and establishes MAFHR limits for the new 8X8 fuel.

In view* of the substantially increased MAPLHC limits

for 7X7 fuel, this action should be pre-noted.

This is consistent w/

action taken on Fitzpatrick and consistent w/ J Scint

mems on the subject.

* of substitution and reliance on new ECCS and

AW 9/23/77

- Proposed NEPA Action:
- EIS Required
 - Negative Declaration (ND) and Environmental Impact Appraisal (EIA) Required
 - No EIS, ND or EIA Required
 - Determination delayed pending completion of EIA

BASIS FOR DECISION:

Action does not involve an increase in authorized power level or a change in effluent type or amounts.

CONCURRENCES:

DATE:

1. CM Trammell 9/23/77
2. A. Schwencer A. Schwencer 9/23/77
3. K. R. Goller KRG 9/23/77
4. OELD A. Mitchell 9/23/77

PRELIMINARY DETERMINATIONNOTICING OF PROPOSED LICENSING AMENDMENT

LICENSEE: Carolina Power & Light Company (CP&L)

REQUEST FOR: Technical Specifications for Cycle 2 (Unit 2) operation which (1) authorize operation with a reload of 8X8 drilled fuel (2) authorize drilling all remaining 7X7 fuel, (3) establish revised operating limits for both fuel types for Cycle 2.

REQUEST DATE: August 22, 1977

PROPOSED ACTION: (XX) Pre-notice Recommended
() Post-notice Recommended
() Determination delayed pending completion of Safety Evaluation

BASIS FOR DECISION: 1. Operation with drilled 7X7 fuel and new 8X8 (drilled) fuel is a modification to the plant which involves an unreviewed safety question and is a significant plant modification which meets the criterion for pre-noticing cited in DOR Memorandum No. 5 Enclosure 1, Item 4.

2. CP&L desires to raise the safety limit MCPR from 1.05 to 1.06, but at the same time lower the operating MCPR from 1.28 to 1.26 for 7X7 fuel. Taken together, this represents a significant reduction in a safety margin of about 13%.

3. Establishing new operating limits for 8X8 fuel (MCPR and LHGR) must be evaluated to determine that the safety margin has not been reduced for this fuel type.

- Proposed NEPA Action:
- EIS Required
 - Negative Declaration (ND) and Environmental Impact Appraisal (EIA) Required
 - No EIS, ND or EIA Required
 - Determination delayed pending completion of EIA

BASIS FOR DECISION: This action involves no changes in effluent types or amounts and no change in authorized power level.

CONCURRENCES

DATE:

1. C. M. Trammell

9/12/77

2. A. Schwencer

ASchwencer 9/12/77

3. K. R. Goller

KRG 9/13/77

4. OELD

A. M. J. Schell 9/19/77