Distribution
Docket File ASLAB
NRC PDR
Local PDR
ORB#2 Rdg.
D. Eisenhut
S. Norris
J. Van Vliet
OELD
SECY
L. J. Harmon 2
T. Barnhart 8
L. Schneider
D. Brinkman
ACRS 10
Clare Miles OPA
R. Diggs
NSIC
Gray

December 16, 1982

Docket Nos. 50-325
50-324

Mr. E. E. Utley
Executive Vice President
Carolina Power & Light Company
P. O. Box 1551
Raleigh, North Carolina 27602

Dear Mr. Utley:

The Commission has issued the enclosed Amendment No. 52 to Facility Operating License No. DPR-71 and Amendment No. 77 to Facility Operating License No. DPR-62 for the Brunswick Steam Electric Plant (BSEP), Units 1 and 2, respectively. These amendments consist of changes to the Technical Specifications in response to your applications dated September 25, 1981 (as modified by submittals dated April 7, 1982 and October 22, 1982) and November 18, 1981; and changes to the license conditions addressing Security personnel training.

The amendments change the Technical Specifications to institute reporting requirements for challenges to safety valves and relief valves per NUREG-0737, Item II.K.3.3, and introduce specific, standardized terminology for the reactor vessel water level reference point. The amendments also change the license conditions formerly titled "Guard Training and Qualification Plan" to "Security Personnel Training and Qualification Plan" and change the title of the plan specified in those license conditions. As previously discussed with your staff, this is being done to assure that operating license terminology is consistent with terminology in use at your facility.

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

Sincerely,

ORIGINAL SIGNED BY

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing

8301050097 821216
PDR ADOCK 05000324
P PDR

Enclosures:
1. Amendment No. 52 to DPR-71
2. Amendment No. 77 to DPR-62
3. Safety Evaluation
4. Notice

cc w/enclosures
See next page

F.R. NOTICE
+
AMENDMENT

OFFICE	ORB#2:DL	ORB#2:DL	ORB#2:DL	DL:OR	OELD		
SURNAME	S.Norris	J.VanVliet:pr	D.Vassallo	G. Lainas	M. KARMAN		
DATE	11/9/82	12/14/82	12/14/82	12/15/82	12/15/82		

OFFICIAL RECORD COPY

NRC FORM 318 (10-80) NRCM 0240 USGPO: 1981—335-960

Mr. E. E. Utley
Carolina Power & Light Company

cc:

Richard E. Jones, Esquire
Carolina Power & Light Company
336 Fayetteville Street
Raleigh, North Carolina 27602

George F. Trowbridge, Esquire
Shaw, Pittman, Potts & Trowbridge
1800 M Street, N. W.
Washington, D. C. 20036

Mr. Charles R. Dietz
Plant Manager
P. O. Box 458
Southport, North Carolina 28461

Mr. Franky Thomas, Chairman
Board of Commissioners
P. O. Box 249
Bolivia, North Carolina 28422

Mrs. Chrys Baggett
State Clearinghouse
Budget & Management
116 West Jones Street
Raleigh, North Carolina 27603

U. S. Environmental Protection Agency
Region IV Office
Regional Radiation Representative
345 Courtland Street, N. W.
Atlanta, Georgia 30308

Resident Inspector
U. S. Nuclear Regulatory Commission
P. O. Box 1057
Southport, North Carolina 28461

James P. O'Reilly
Regional Administrator, Region II
U.S. Nuclear Regulatory Commission
101 Marietta Street, Suite 3100
Atlanta, Georgia 30303

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

CAROLINA POWER & LIGHT COMPANY

DOCKET NO. 50-325

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No.52
License No. DPR-71

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

   A. The applications for amendments by Carolina Power & Light Company dated September 25, 1981 (as revised April 7, 1982 and October 22, 1982) and November 18, 1981 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, Facility Operating License No. DPR-71 is hereby amended by changing paragraphs 2.C.(2) and 2.D.(3) to read as follows:

   2.C.(2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 52 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

   2.D.(3) Security Personnel Training and Qualification Plan

   The licensee shall fully implement and maintain in effect all provisions of the Commission-approved Security Personnel Training and

Qualification Plan, including amendments and changes made pursuant to the authority of 10 CFR 50.54(p). The approved plan consists of documents withheld from public disclosure pursuant to 10 CFR 2.790(d) identified as Brunswick Steam Electric Plant Security Personnel Training and Qualification Plan dated August 17, 1979 as revised by Revision 1 pages dated May 26, 1981. This plan shall be implemented, in accordance with 10 CFR 73.55(b)(4), within 60 days after approval by the Commission. The licensee may make changes to this plan without prior Commission approval if the changes do not decrease the safeguards effectiveness of the plan. The licensee shall maintain records of and submit reports concerning such changes in the same manner as required for changes made to the Safeguards Contingency Plan pursuant to 10 CFR 50.54(p).

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

FOR THE NUCLEAR REGULATORY COMMISSION

DB Vassallo

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing

Attachment:
Changes to Technical Specifications

Date of Issuance: December 16, 1982

ATTACHMENT TO LICENSE AMENDMENT NO. 52

FACILITY OPERATING LICENSE NO. DPR-71

DOCKET NO. 50-325

Remove the following pages and replace with identically numbered pages.

1-5
1-6
2-4
2-6
3/4 3-17
3/4 3-18
3/4 3-21
3/4 3-34
3/4 3-35
3/4 3-64
6-16

PRIMARY CONTAINMENT INTEGRITY (Continued)

1.24.3 Each containment air lock is OPERABLE pursuant to Specification 3.6.1.3.

1.24.4 The containment leakage rates are within the limits of Specification 3.6.1.2.

1.24.5 The sealing mechanism associated with each penetration (e.g., welds, bellows or O-rings is) is OPERABLE.

RATED THERMAL POWER

1.25 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 2436 MWT.

REACTOR PROTECTION SYSTEM RESPONSE TIME

1.26 REACTOR PROTECTION SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids.

REFERENCE LEVEL ZERO

1.27 The REFERENCE LEVEL ZERO point is arbitrarily set at 367 inches above the vessel zero point. This REFERENCE LEVEL ZERO is approximately mid-point on the top fuel guide and is the single reference for all specifications of vessel water level.

REPORTABLE OCCURRENCE

1.28 A REPORTABLE OCCURRENCE shall be any of those conditions specified in Specification 6.8.1.8 and 6.9.1.9.

ROD DENSITY

1.29 ROD DENSITY shall be the number of control rod notches inserted as a fraction of the total number of notches. All rods fully inserted is equivalent to 100% ROD DENSITY.

SECONDARY CONTAINMENT INTEGRITY

1.30 SECONDARY CONTAINMENT INTEGRITY shall exist when:

1.30.1 All automatic reactor builidng ventilation system isolation valves or dampers are OPERABLE or secured in the isolated position,

1.30.2 The standby gas treatment system is OPERABLE pursuant to Specification 3.6.6.1.

1.30.3 At least one door in each access to the reactor building is closed.

SECONDARY CONTAINMENT INTEGRITY (continued)

1.30.4 The sealing mechanism associated with each penetration (e.g., welds, bellows or O-rings) is OPERABLE.

SHUTDOWN MARGIN

1.31 SHUTDOWN MARGIN shall be the amount of reactivity by which the reactor would be subcritical assuming that all control rods capable of insertion are fully inserted except for the analytically determined highest worth rod which is assumed to be fully withdrawn, and the reactor is in the shutdown condition, cold, 68°F, and Xenon free.

SPIRAL RELOAD

1.32 A SPIRAL RELOAD is the reverse of a SPIRAL UNLOAD. Except for two diagonal fuel bundles around each of the four SRMs, the fuel in the interior of the core, symmetric to the SRMs, is loaded first.

SPIRAL UNLOAD

1.33 A SPIRAL UNLOAD is a core unload performed by first removing the fuel from the outermost control cells (four bundles surrounding a control blade). Unloading continues in a spiral fashion by removing fuel from the outermost periphery to the interior of the core, symmetric about the SRMs, except for two diagonal fuel bundles around each of the four SRMs.

STAGGERED TEST BASIS

1.34 A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals.

b. The testing of one system, subsystem, train or other designated component at the beginning of each subinterval.

THERMAL POWER

1.35 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

TOTAL PEAKING FACTOR

1.36 The TOTAL PEAKING FACTOR (TPF) shall be the ratio of local LHGR for any specific location on a fuel rod divided by the average LHGR associated with the fuel bundles of the same type operating at the core average bundle power.

UNIDENTIFIED LEAKAGE

1.37 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE.

TABLE 2.2.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

FUNCTIONAL UNIT AND INSTRUMENT NUMBER	TRIP SETPOINT	ALLOWABLE VALUES
1. Intermediate Range Monitor, Neutron Flux - High(1) (C51-IRM-K601 A,B,C,D,E,F,G,H)	≤120 divisions of full scale	≤120 divisions of full scale
2. Average Power Range Monitor (C51-APRM-CH.A,B,C,D,E,F)		
a. Neutron Flux - High 15%(2)	≤15% of RATED THERMAL POWER	≤15% of RATED THERMAL POWER
b. Flow Biased Neutron Flux - High(3)(4)	≤(0.66 W + 54%)	≤(0.66 W + 54%)
c. Fixed Neutron Flux - High(4)	≤120% of RATED THERMAL POWER	≤120% of RATED THERMAL POWER
3. Reactor Vessel Steam Dome Pressure - High (B21-PTM-N023A-1,B-1,C-1,D-1)	≤1045 psig	≤1045 psig
4. Reactor Vessel Water Level - Low, Level#1(7) (B21-LTM-N017A-1,B-1,C-1,D-1)	≥+162.5 inches	≥+162.5 inches
5. Main Steam Line Isolation Valve - Closure(5) (B21-F022 A,B,C,D; B21-F028 A,B,C,D)	≤10% closed	≤10% closed
6. Main Steam Line Radiation - High (D12-RM-K603 A,B,C,D)	≤3 x full power background	≤3.5 x full power background

TABLE 2.2.1-1 (Continued)
REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS
TABLE NOTATION

(1) The Intermediate Range Monitor scram functions are automatically bypassed when the reactor mode switch is placed in the Run position and the Average Power Range Monitors are on scale.

(2) This Average Power Range Monitor scram function is a fixed point and is increased when the reactor mode switch is placed in the Run position.

(3) The Average Power Range Monitor scram function is varied, Figure 2.2.1-1, as a function of recirculation loop flow (W). The trip setting of this function must be maintained in accordance with Specification 3.2.2.

(4) The APRM flow biased high neutron flux signal is fed through a time constant circuit of approximately 6 seconds. The APRM fixed high neutron flux signal does not incorporate the time constant, but responds directly to instantaneous neutron flux.

(5) The Main Steam Line Isolation Valve-Closure scram function is automatically bypassed when the reactor mode switch is in other than the Run position.

(6) These scram functions are bypassed when THERMAL POWER is less than 30% of RATED THERMAL POWER.

(7) Vessel water levels refer to REFERENCE LEVEL ZERO.

TABLE 3.3.2-2

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION AND INSTRUMENT NUMBER	TRIP SETPOINT	ALLOWABLE VALUE
1. PRIMARY CONTAINMENT ISOLATION		
a. Reactor Vessel Water Level - Low*		
1. Level #1 (B21-LTM-N017A-1, B-1, C-1, D-1)	≥ +162.5 inches	≥ +162.5 inches
2. Level #2 (B21-LTM-N024A-1, B-1 and B21-LTM-N025A-1, B-1)	≥ +112 inches	≥ +112 inches
b. Drywell Pressure - High (C71-PS-N002 A,B,C,D)	≤ 2 psig	≤ 2 psig
c. Main Steam Line		
1. Radiation - High (D12-RM-K603 A, B, C, D)	≤ 3 x full power background	≤ 3.5 x full power background
2. Pressure - Low (B21-PS-N015A,B,C,D)	≥ 825 psig	≥ 825 psig
3. Flow - High (B21-dPIS-N006 A,B,C,D; B21-dPIS-N007 A,B,C,D; B21-dPIS N008 A,B,C,D; and B21-dPIS-N009 A,B,C,D)	≤ 140% of rated flow	≤ 140% of rated flow
d. Main Steam Line Tunnel Temperature - High (B21-TS-N010 A, B, C, D; B21-TS-N011 A, B, C, D; B21-TS-N012 A, B, C, D; and B21-TS-N013 A, B, C, D)	≤ 200° F	≤ 200° F
e. Condenser Vacuum - Low (B21-PS-N056 A,B,C,D)	≥ 7 inches Hg vacuum	≥ 7 inches Hg vacuum
f. Turbine Building Area Temp. - High (B21-TS-3225 A,B,C,D; B21-TS-3226 A,B,C,D; B21-TS-3227 A,B,C,D; B21-TS-3228 A,B,C,D; B21-TS-3229 A,B,C,D; B21-TS-3230 A,B,C,D; B21-TS-3231 A,B,C,D; B21-TS-3232 A,B,C,D)	≤ 200° F	≤ 200° F

*Vessel water levels refer to REFERENCE LEVEL ZERO.

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION AND INSTRUMENT NUMBER	TRIP SETPOINT	ALLOWABLE VALUE
2. SECONDARY CONTAINMENT ISOLATION		
a. Reactor Building Exhaust Radiation - High (D12-RM-N010 A, B)	≤ 11 mr/hr	≤ 11 mr/hr
b. Drywell Pressure - High (C71-PS-N002 A,B,C,D)	≤ 2 psig	≤ 2 psig
c. Reactor Vessel Water Level - Low, Level #2* (B21-LTM-N024A-1,B-1 and B21-LTM-N025A-1,B-1)	≥ +112 inches	≥ +112 inches
3. REACTOR WATER CLEANUP SYSTEM ISOLATION		
a. Δ Flow - High (G31-dFS-N603-1A, 1B)	≤ 53 gal/min	≤ 53 gal/min
b. Area Temperature - High (G31-TS-N600A, B, C, D, E, F)	≤ 150° F	≤ 150° F
c. Area Ventilation Temperature Δ Temp - High (G31-TS-N602A, B, C, D, E, F)	≤ 50° F	≤ 50° F
d. SLCS Initiation (C41A-S1)	NA	NA
e. Reactor Vessel Water - Low, Level #2* (B21-LTM-N024A-1, B-1 and B21-LTM-N025A-1, B-1)	≥ +112 inches	≥ +112 inches

*Vessel water levels refer to REFERENCE LEVEL ZERO.

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION AND INSTRUMENT NUMBER	TRIP SETPOINT	ALLOWABLE VALUE
5. SHUTDOWN COOLING SYSTEM ISOLATION		
a. Reactor Vessel Water - Low, Level #1* (B21-LTM-N017A-1, B-1, C-1, D-1)	$\geq$ +162.5 inches	$\geq$ +162.5 inches
b. Reactor Steam Dome Pressure - High (B32-PS-N018A, B)	$\leq$ 140 psig	$\leq$ 140 psig

*Vessel water levels refer to REFERENCE LEVEL ZERO.

TABLE 3.3.3-2

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION AND INSTRUMENT NUMBER	TRIP SETPOINT	ALLOWABLE VALUE
1. CORE SPRAY SYSTEM		
a. Reactor Vessel Water Level - Low, Level #3* (B21-LTS-N031A-4, B-4, C-4, D-4)	$\geq$ +2.5 inches	$\geq$ +2.5 inches
b. Reactor Steam Dome Pressure - Low (B21-PTS-N021A-2, B-2, C-2, D-2)	410 $\pm$ 15 psig	410 $\pm$ 15 psig
c. Drywell Pressure - High (E11-PS-N011A,B,C,D)	$\leq$ 2 psig	$\leq$ 2 psig
d. Time Delay Relay	14 $\leq$ t $\leq$ 16 secs	14 $\leq$ t $\leq$ 16 secs
e. Bus Power Monitor (E21-K1A, B)	NA	NA
2. LPCI MODE OF RHR SYSTEM		
a. Drywell Pressure - High (E11-PS-N011A,B,C,D)	$\leq$ 2 psig	$\leq$ 2 psig
b. Reactor Vessel Water Level - Low, Level #3* (B21-LTS-N031A-4, B-4, C-4, D-4)	$\geq$ +2.5 inches	$\geq$ +2.5 inches
c. Reactor Vessel Shroud Level* (B21-LTM-N036-1 and B21-LTM-N037-1)	$\geq$ -53 inches	$\geq$ -53 inches

*Vessel water levels refer to REFERENCE LEVEL ZERO.

TABLE 3.3.3-2 (continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION AND INSTRUMENT NUMBER	TRIP SETPOINT	ALLOWABLE VALUE
3. HPCI SYSTEM		
a. Reactor Vessel Water Level - Low, Level #2* (B21-LTS N031A-2, B-2, C-2, D-2)	$\geq$ +112 inches	$\geq$ +112 inches
b. Drywell Pressure - High (E11-PS-N011A,B,C,D)	$\leq$ 2 psig	$\leq$ 2 psig
c. Condensate Storage Tank Level - Low (E41-LS-N002, E41-LS-N003)	$\geq$ 23'4"	$\geq$ 23'4"
d. Suppression Chamber Water Level - High** (E41-LSH-N015A, B)	$\leq$ -2 feet	$\leq$ -2 feet
e. Bus Power Monitor (E41-K55 and E41-K56)	N/A	N/A
4. ADS		
a. Drywell Pressure - High (E11-PS-N010A,B,C,D)	$\leq$ 2 psig	$\leq$ 2 psig
b. Reactor Vessel Water Level - Low, Level #3* (B21-LTS-N031A-3, B-3, C-3, D-3)	$\geq$ +2.5 inches	$\geq$ +2.5 inches
c. Reactor Vessel Water Level - Low, Level #1* (B21-LTM-N042A-1, B-1)	$\geq$ +162.5 inches	$\geq$ +162.5 inches
d. ADS Timer (B21-TDPU-K5A, B)	$\leq$ 120 seconds	$\leq$ 120 seconds
e. Core Spray Pump Discharge Pressure - High (E21-PS-N008A, B and E21-PS-N009A, B)	$\geq$ 100 psig	$\geq$ 100 psig
f. RHR (LPCI Mode) Pump Discharge Pressure - High (E11-PS-N016A, B, C, D and E11-PS-N020A, B, C, D)	$\geq$ 100 psig	$\geq$ 100 psig

*Vessel water levels refer to REFERENCE LEVEL ZERO.
**Suppression chamber water level zero in the torus centerline minus 1 inch.

TABLE 3.3.6.1-2

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION SETPOINTS

TRIP FUNCTION AND INSTRUMENT NUMBER	TRIP SETPOINT	ALLOWABLE VALUE
1. Reactor Vessel, Water Level - Low Low, Level #2* (B21-LTM-N024A-2,B-2 and B21-LTM-N025A-2,B-2)	$\geq$ +112 inches	$\geq$ +112 inches
2. Reactor Vessel Pressure - High (B21-PS-N045A,B,C,D)	$\leq$ 1120 psig	$\leq$ 1120 psig

*Vessel water levels refer to REFERENCE LEVEL ZERO.

START-UP REPORT (Continued)

completion of start-up test program, and resumption or commencement of commercial power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.

ANNUAL REPORTS[1/]

6.9.1.4 Annual reports covering the activities of the unit as described below during the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.

6.9.1.5 Reports required on an annual basis shall include a tabulation of the number of station, utility, and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man rem exposure according to work and job functions, [2/] e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignment to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions.

MONTHLY OPERATING REPORT

6.9.1.6 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to main steam system safety/relief valves, shall be submitted on a monthly basis to the Office of Inspection and Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Office, to arrive no later than the tenth of each month following the calendar month covered by the report.

REPORTABLE OCCURRENCES

6.9.1.7 The REPORTABLE OCCURRENCES of Specifications 6.9.1.8 and 6.9.1.9 below, including corrective actions and measures to prevent recurrence, shall be reported to the NRC. Supplemental reports may be required to fully describe final resolution of occurrence. In case of corrected or supplemental reports, a licensee event report shall be completed and reference shall be made to the original report date.

1/A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.

2/This tabulation supplements the requirements of Section 20.407 of 10 CFR Part 20.

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. 77
License No. DPR-62

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

   A. The applications for amendments by Carolina Power & Light Company dated September 25, 1981 (as revised April 7, and October 22, 1982) and November 18, 1981 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, Facility Operating License No. DPR-62 is hereby amended by changing paragraphs 2.C.(2) and 2.C.(8) to read as follows:

   2.C.(2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 77, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

   2.C.(8) Security Personnel Training and Qualification Plan

   The licensee shall fully implement and maintain in effect all provisions of the Commission-approved Security Personnel Training and Qualification Plan, including amendments and changes made pursuant to the authority of 10 CFR 50.54(p). The approved plan consists of documents withheld from public disclosure pursuant to 10 CFR 2.790(d) identified as Brunswick Steam

Electric Plant Security Personnel Training and Qualification Plan dated August 17, 1979 as revised by Revision 1 pages dated May 26, 1981. This plan shall be implemented, in accordance with 10 CFR 73.55(b)(4), within 60 days after approval by the Commission. The licensee may make changes to this plan without prior Commission approval if the changes do not decrease the safeguards effectiveness of the plan. The licensee shall maintain records of and submit reports concerning such changes in the same manner as required for changes made to the Safeguards Contingency Plan pursuant to 10 CFR 50.54(p).

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

FOR THE NUCLEAR REGULATORY COMMISSION

D B Vassallo

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing

Attachment:
Changes to Technical Specifications

Date of Issuance: December 16, 1982

<u>ATTACHMENT TO LICENSE AMENDMENT NO. 77</u>

<u>FACILITY OPERATING LICENSE NO. DPR-62</u>

<u>DOCKET NO. 50-324</u>

Remove the following pages and replace with identically numbered pages.

II
1-5
1-6
1-6a
2-4
2-6
3/4 3-17
3/4 3-18
3/4 3-21
3/4 3-34
3/4 3-35
3/4 3-64
6-16

INDEX

DEFINITIONS

SECTION

1.0 DEFINITIONS (Continued)

PRESSURE BOUNDARY LEAKAGE

PRESSURE BOUNDARY LEAKAGE shall be leakage through a non-isolatable fault in a reactor coolant system component body, pipe wall, or vessel wall.

PRIMARY CONTAINMENT INTEGRITY

PRIMARY CONTAINMENT INTEGRITY shall exist when:

a. All penetrations required to be closed during accident conditions are either:

   1. Capable of being closed by an OPERABLE containment automatic isolation valve system, or

   2. Closed by at least one manual valve, blind flange, or deactivated automatic valve secured in its closed position, except as provided in Table 3.6.3-1 of Specification 3.6.3.1, or

b. All equipment hatches are closed and sealed.

c. Each containment air lock is OPERABLE pursuant to Specification 3.6.1.3.

d. The containment leakage rates are within the limits of Specification 3.6.1.2.

e. The sealing mechanism associated with each penetration (e.g., welds, bellows or O-rings) is OPERABLE.

RATED THERMAL POWER

RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 2436 MWT.

REACTOR PROTECTION SYSTEM RESPONSE TIME

REACTOR PROTECTION SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids.

REFERENCE LEVEL ZERO

The REFERENCE LEVEL ZERO point is arbitrarily set at 367 inches above the vessel zero point. This REFERENCE LEVEL ZERO is approximately mid-point on the top fuel guide and is the single reference for all specifications of vessel water level.

REPORTABLE OCCURRENCE

A REPORTABLE OCCURRENCE shall be any of those conditions specified in Specifications 6.9.1.8 and 6.9.1.9.

ROD DENSITY

ROD DENSITY shall be the number of control rod notches inserted as a fraction of the total number of notches. All rods fully inserted are equivalent to 100% ROD DENSITY.

SECONDARY CONTAINMENT INTEGRITY

SECONDARY CONTAINMENT INTEGRITY shall exist when:

a. All automatic Reactor Building ventilation system isolation valves or dampers are OPERABLE or secured in the isolated position,

b. The standby gas treatment system is OPERABLE pursuant to Specification 3.6.6.1.

c. At least one door in each access to the Reactor Building is closed.

d. The sealing mechanism associated with each penetration (e.g., welds, bellows or O-rings) is OPERABLE.

SHUTDOWN MARGIN

SHUTDOWN MARGIN shall be the amount of reactivity by which the reactor would be subcritical assuming that all control rods capable of insertion are fully inserted except for the analytically determined highest worth rod which is assumed to be fully withdrawn, and the reactor is in the shutdown condition, cold, 68°F, and Xenon free.

SPIRAL RELOAD

A SPIRAL RELOAD is the reverse of a SPIRAL UNLOAD. Except for two diagonal fuel bundles around each of the four SRMs, the fuel in the interior of the core, symmetric to the SRMs, is loaded first.

SPIRAL UNLOAD

A SPIRAL UNLOAD is a core unload performed by first removing the fuel from the outermost control cells (four bundles surrounding a control blade). Unloading continues in a spiral fashion by removing fuel from the outermost periphery to the interior of the core, symmetric about the SRMs, except for two diagonal fuel bundles around each of the four SRMs.

STAGGERED TEST BASIS

A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals.

STAGGERED TEST BASIS (continued)

b. The testing of one system, subsystem, train or other designated component at the beginning of each subinterval.

THERMAL POWER

THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

TOTAL PEAKING FACTOR

The TOTAL PEAKING FACTOR (TPF) shall be the ratio of local LHGR for any specific location on a fuel rod divided by the average LHGR associated with the fuel bundles of the same type operating at the core average bundle power.

UNIDENTIFIED LEAKAGE

UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE.

TABLE 2.2.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

FUNCTIONAL UNIT AND INSTRUMENT NUMBER	TRIP SETPOINT	ALLOWABLE VALUES
1. Intermediate Range Monitor, Neutron Flux - High(1) (C51-IRM-K601 A,B,C,D,E,F,G,H)	$\leq$ 120 divisions of full scale	$\leq$ 120 divisions of full scale
2. Average Power Range Monitor (C51-APRM-CH.A,B,C,D,E,F)		
a. Neutron Flux - High, 15%(2)	$\leq$ 15% of RATED THERMAL POWER	$\leq$ 15% of RATED THERMAL POWER
b. Flow Biased Neutron Flux - High(3)(4)	$\leq$ (0.66 W + 54%)	$\leq$ (0.66 W + 54%)
c. Fixed Neutron Flux - High(4)	$\leq$ 120% of RATED THERMAL POWER	$\leq$ 120% of RATED THERMAL POWER
3. Reactor Vessel Steam Dome Pressure - High (B21-PS-N023 A,B,C,D)	$\leq$ 1045 psig	$\leq$ 1045 psig
4. Reactor Vessel Water Level - Low, Level #1(7) (B21-LIS-N017 A,B,C,D)	$\geq$ +162.5 inches	$\geq$ +162.5 inches
5. Main Steam Line Isolation Valve - Closure(5) (B21-F022 A,B,C,D; B21-F028 A,B,C,D)	$\leq$ 10% closed	$\leq$ 10% closed
6. Main Steam Line Radiation - High (D12-RM-K603 A,B,C,D)	$\leq$ 3 x full power background	$\leq$ 3.5 x full power background
7. Drywell Pressure - High (C72-PS-N002 A,B,C,D)	$\leq$ 2 psig	$\leq$ 2 psig
8. Scram Discharge Volume Water Level - High (C12-LSH-N013 A,B,C,D) (C12-LSH-4516 A,B,C,D)	$\leq$ 109 gallons	$\leq$ 109 gallons

TABLE 2.2.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

TABLE NOTATION

(1) The Intermediate Range Monitor scram functions are automatically bypassed when the reactor mode switch is placed in the Run position and the Average Power Range Monitors are on scale.

(2) This Average Power Range Monitor scram function is a fixed point and is increased when the reactor mode switch is placed in the Run position.

(3) The Average Power Range Monitor scram function is varied, Figure 2.2.1-1, as a function of recirculation loop flow (W). The trip setting of this function must be maintained in accordance with Specification 3.2.2.

(4) The APRM flow biased high neutron flux signal is fed through a time constant circuit of approximately 6 seconds. The APRM fixed high neutron flux signal does not incorporate the time constant, but responds directly to instantaneous neutron flux.

(5) The Main Steam Line Isolation Valve-Closure scram function is automatically bypassed when the reactor mode switch is in other than the Run position.

(6) These scram functions are bypassed when THERMAL POWER is less than 30% of RATED THERMAL POWER.

(7) Vessel water levels refer to REFERENCE LEVEL ZERO.

TABLE 3.3.2-2

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION AND INSTRUMENT NUMBER	TRIP SETPOINT	ALLOWABLE VALUE
1. PRIMARY CONTAINMENT ISOLATION		
a. Reactor Vessel Water Level - Low*		
1. Level #1 (B21-LIS-N017 A,B,C,D)	$\geq$ +162.5 inches	$\geq$ +162.5 inches
2. Level #2 (B21-LIS-N024 A,B and B21-LIS-N025 A,B)	$\geq$ +112 inches	$\geq$ +112 inches
b. Drywell Pressure - High (C72-PS-N002 A,B,C,D)	$\leq$ 2 psig	$\leq$ 2 psig
c. Main Steam Line		
1. Radiation - High (D12-RM-K603 A,B,C,D)	$\leq$ 3 x full power background	$\leq$ 3.5 x full power background
2. Pressure - Low (B21-PS-N015 A,B,C,D)	$\geq$ 825 psig	$\geq$ 825 psig
3. Flow - High (B21-dPIS-N006 A,B,C,D; B21-dPIS-N007 A,B,C,D; B21-dPIS-N008 A,B,C,D; and B21-dPIS-N009 A,B,C,D)	$\leq$ 140% of rated flow	$\leq$ 140% of rated flow
4. Flow - High (B21-dPIS-N006A; B21-dPIS-N078; B21-dPIS-N008C and B21-dPIS-N009D)	$\leq$ 40% of rated flow	$\leq$ 40% of rated flow
d. Main Steam Line Tunnel Temperature - High (B21-TS-N010 A,B,C,D; B21-TS-N011 A,B,C,D; B21-TS N012 A,B,C,D; and B21-TS-N013 A,B,C,D)	$\leq$ 200°F	$\leq$ 200°F
e. Condenser Vacuum - Low (B21-PS-N056 A,B,C,D)	$\geq$ 7 inches Hg vacuum	$\geq$ 7 inches Hg vacuum
f. Turbine Building Area Temp - High (B21-TS-3225 A,B,C,D; B21-TS-3226 A,B,C,D; B21-TS-3227 A,B,C,D; B21-TS-3228 A,B,C,D; B21-TS-3229 A,B,C,D; B21-TS-3230 A,B,C,D; B21-TS-3231 A,B,C,D and B21-TS-3232 A,B,C,D)	$\leq$ 200°F	$\leq$ 200°F

*Vessel water levels refer to REFERENCE LEVEL ZERO.

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION AND INSTRUMENT NUMBER	TRIP SETPOINT	ALLOWABLE VALUE
2. SECONDARY CONTAINMENT ISOLATION		
a. Reactor Building Exhaust Radiation - High (D12-RM-N010 A,B)	≤ 11 mr/hr	≤ 11 mr/hr
b. Drywell Pressure - High (C72-PS-N002 A,B,C,D)	≤ 2 psig	≤ 2 psig
c. Reactor Vessel Water Level - Low, Level #2* (B21-LIS-N024 A,B and B21-LIS-N025 A,B)	≥ +112 inches	≥ +112 inches
3. REACTOR WATER CLEANUP SYSTEM ISOLATION		
a. Δ Flow - High (G31-dFS-N603-1A,1B)	≤ 53 gal/min	≤ 53 gal/min
b. Area Temperature - High (G31-TS-N600A,B,C,D,E,F)	≤ 150°F	≤ 150°F
c. Area Ventilation Temperature Δ Temp-High (G31-TS-N602A,B,C,D,E,F)	≤ 50°F	≤ 50°F
d. SLCS Initiation (C41A-S1)	NA	NA
e. Reactor Vessel Water - Low, Level #2* (B21-LIS-N024A,B and B21-LIS-N025A,B)	≥ +112 inches	≥ +112 inches

*Vessel water levels refer to REFERENCE LEVEL ZERO.

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION AND INSTRUMENT NUMBER	TRIP SETPOINT	ALLOWABLE VALUE
5. SHUTDOWN COOLING SYSTEM ISOLATION		
a. Reactor Vessel Water - Low, Level #1* (B21-LTM-N017A-1, B-1, C-1, D-1)	$\geq$ +162.5 inches	$\geq$ +162.5 inches
b. Reactor Steam Dome Pressure - High (B32-PS-N018A, B)	$\leq$ 140 psig	$\leq$ 140 psig

*Vessel water levels refer to REFERENCE LEVEL ZERO.

TABLE 3.3.3-2

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION AND INSTRUMENT NUMBER	TRIP SETPOINT	ALLOWABLE VALUE
1. CORE SPRAY SYSTEM		
a. Reactor Vessel Water Level - Low Level #3* (B21-LIS-N031A,B,C,D)	$\geq$ 2.5 inches	$\geq$ 2.5 inches
b. Reactor Steam Dome Pressure - Low (B21-PS-N021A,B,C,D)	410 ± 15 psig	410 ± 15 psig
c. Drywell Pressure - High (E11-PS-N011A,B,C,D)	$\leq$ 2 psig	$\leq$ 2 psig
d. Time Delay-Relay	14 $\leq$ t $\leq$ 16 secs	14 $\leq$ t $\leq$ 16 secs
e. Bus Power Monitor (E21-K1A,B)	NA	NA
2. LPCI MODE OF RHR SYSTEM		
a. Drywell Pressure - High (E11-PS-N011A,B,C,D)	$\leq$ 2 psig	$\leq$ 2 psig
b. Reactor Vessel Water Level - Low, Level #3* (B21-LIS-N031A,B,C,D)	$\geq$ +2.5 inches	$\geq$ +2.5 inches
c. Reactor Vessel Shroud Level* (B21-LITS-N036 and B21-LITS-N037)	$\geq$ -53 inches	$\geq$ -53 inches
d. Reactor Steam Dome Pressure - Low (B21-PS-N021A,B,C,D)		
1. RHR Pump Start and LCPI Valve Actuation	410 ± 15 psig	410 ± 15 psig
2. Recirculation Pump Discharge Valve Actuation	310 ± 15 psig	310 ± 15 psig
e. RHR Pump Start - Time Delay Relay	9 $\leq$ t $\leq$ 11 seconds	9 $\leq$ t $\leq$ 11 seconds
f. Bus Power Monitor (E11-K106A,B)	NA	NA

*Vessel water levels refer to REFERENCE LEVEL ZERO.

TABLE 3.3.3-2 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION AND INSTRUMENT NUMBER	TRIP SETPOINT	ALLOWABLE VALUE
3. HPCI SYSTEM		
a. Reactor Vessel Water Level - Low, Level #2* (B21-LIS-N031A,B,C,D)	$\geq$ +112 inches	$\geq$ +112 inches
b. Drywell Pressure-High (E11-PS-N011A,B,C,D)	$\leq$ 2 psig	$\leq$ 2 psig
c. Condensate Storage Tank Level - Low (E41-LS-N002, E41-LS-N003)	$\geq$ 23'4"	$\geq$ 23'4"
d. Suppression Chamber Water Level - High** (E41-LSH-N015A,B)	$\leq$ -2 feet	$\leq$ -2 feet
e. Bus Power Monitor (E44-K55 and E41-K56)	NA	NA
4. ADS		
a. Drywell Pressure-High (E11-PS-N010A,B,C,D)	$\leq$ 2 psig	$\leq$ 2 psig
b. Reactor Vessel Water Level - Low, Level #3* (B21-LIS-N031A,B,C,D)	$\geq$ +2.5 inches	$\geq$ +2.5 inches
c. Reactor Vessel Water Level - Low Level #1* (B21-LIS-N042 A,B)	$\geq$ + 162.5 inches	$\geq$ + 162.5 inches
d. ADS Timer (B21-TDPU-K5A,B)	$\leq$ 120 seconds	$\leq$ 120 seconds
e. Core Spray Pump Discharge Pressure - High (E21-PS-N008A,B and E21-PS-N009A,B)	$\geq$ 100 psig	$\geq$ 100 psig
f. RHR (LPCI MODE) Pump Discharge Pressure - High (E11-PS-N016A,B,C,D and E11-PS-N020A,B,C,D)	$\geq$ 100 psig	$\geq$ 100 psig
g. Bus Power Monitor (B21-K1A,B)	NA	NA

*Vessel water levels refer to REFERENCE LEVEL ZERO.
**Suppression chamber water level zero is the torus centerline minus 1 inch.

TABLE 3.3.6.1-2

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION SETPOINTS

TRIP FUNCTION AND INSTRUMENT NUMBER	TRIP SETPOINT	ALLOWABLE VALUE
1. Reactor Vessel, Water Level - Low low, Level #2* (B21-LIS-N024 A, B; B21-LIS-N025 A, B)	≥ +112 inches	≥ +112 inches
2. Reactor Vessel Pressure-High (B21-PS-N045 A, B, C, D)	≤1120 psig	≤1120 psig

*Vessel water levels refer to REFERENCE LEVEL ZERO.

START-UP REPORT (Continued)

completion of start-up test program, and resumption or commencement of commercial power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.

ANNUAL REPORTS[1/]

6.9.1.4 Annual reports covering the activities of the unit as described below during the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.

6.9.1.5 Reports required on an annual basis shall include a tabulation of the number of station, utility, and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man rem exposure according to work and job functions,[2/] e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignment to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions.

MONTHLY OPERATING REPORT

6.9.1.6 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to main steam system safety/relief valves, shall be submitted on a monthly basis to the Office of Inspection and Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Office, to arrive no later than the tenth of each month following the calendar month covered by the report.

REPORTABLE OCCURRENCES

6.9.1.7 The REPORTABLE OCCURRENCES of Specifications 6.9.1.8 and 6.9.1.9 below, including corrective actions and measures to prevent recurrence, shall be reported to the NRC. Supplemental reports may be required to fully describe final resolution of occurrence. In case of corrected or supplemental reports, a licensee event report shall be completed and reference shall be made to the original report date.

1/ A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.

2/ This tabulation supplements the requirements of Section 20.407 of 10 CFR Part 20.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
SUPPORTING AMENDMENT NO. 52 TO FACILITY LICENSE NO. DPR-71 AND
AMENDMENT NO. 77 TO FACILITY LICENSE NO. DPR-62
CAROLINA POWER & LIGHT COMPANY
BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2
DOCKET NOS. 50-325 AND 50-324

1.0 Introduction

By letters dated September 25, 1981 and November 18, 1981 Carolina Power & Light Company (the licensee) forwarded proposed changes to the Technical Specifications for the Brunswick Steam Electric Plant (BSEP) Units 1 and 2. By letters dated April 7, 1982 and October 22, 1982 the licensee revised the Technical Specification changes originally proposed by their September 25, 1981 submittal. The proposed changes: institute reporting requirements for challenges to safety valves and relief valves (S/RVs) per NUREG-0737, Item II.K.3.3, and introduce specific, standardized terminology for the reactor vessel water level reference point.

2.0 Evaluation

2.1 S/RV Reporting Requirements

NUREG-0737, Item II.K.3.3 specifies that all S/RV challenges and failures should be reported to the NRC. The Technical Spevification change proposed by the licensee would require all S/RV challenges to be reported monthly. (Reporting of S/RV failures is already required by Section 6.9.1.9 of the BSEP Technical Specifications.)

Since the proposed Technical Specification change conforms to the guidance of NUREG-0737, Item II.K.3.3., we find it to be acceptable.

2.2 Reactor Vessel Water Level Reference Point

NUREG-0737, Item II.K.3.27, Common Reference Level, requested licensees of all operating boiling water reactors to establish a common reference level to which all reactor vessel water level indicators would be zeroed. This was accomplished by license amendment No. 38 for BSEP Unit 1, and by license amendment nos. 56 and 60 for BSEP Unit 2. The safety evaluations accompanying those license amendments specify that the reference level is 367 inches above the vessel bottom, but the Technical Specifications refer to the reference level only as "top fuel guide." Since the top fuel guide is actually eight inches thick (the 367 inch level being about the mid-point on the top fuel guide) the licensee feels that the words "top fuel guide" are ambiguous and could be subject to misinterpretation. Consequently, the licensee has proposed to: (1) introduce a "Reference level zero" which would be defined in the Technical Specifications as an arbitrary point 367 inches above the vessel zero point; and (2) address all Technical Specification reactor vessel water level setpoints as "+" or "-" inches from Reference Level Zero.

We have verified that: (1) the Reference Level Zero is the same point (i.e. 367 inches above the vessel zero) that was evaluated in the safety evaluations mentioned in the preceding paragraph; (2) no changes are being proposed to the reactor vessel water level setpoints in the Technical Specifications; and (3) no hardware modifications are involved. We consider the proposal to clearly define Reference Level Zero in the Technical Specifications to be potentially beneficial in that the action can only serve to reduce the potential for misinterpretation. We, therefore, consider the proposed Technical Specification changes to be acceptable.

3.0 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 impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of the amendments.

4.0 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 an accident previously evaluated, do not create the possibility of an accident of a type different from any evaluated previously, and do not involve a significant reduction in a margin of safety, 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 this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: December 16, 1982

Principal Contributor: James A. Van Vliet

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-325 AND 50-324

CAROLINA POWER & LIGHT COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment Nos. 52 and 77 to Facility Operating License Nos. DPR-71 and DPR-62 issued to Carolina Power & Light Company (the licensee) which revised the Operating License and Technical Specifications for operation of the Brunswick Steam Electric Plant, Units Nos. 1 and 2 (the facility), located in Brunswick County, North Carolina. The amendments are effective as of the date of issuance.

The amendments revise the Technical Specifications to institute reporting requirements for challenges to safety valves and relief valves per NUREG-0737, Item II.K.3.3, and introduce specific, standardized terminology for the reactor vessel water level reference point. The amendments also change the license conditions formerly titled "Guard Training and Qualification Plan" to "Security Personnel Training and Qualification Plan" and change the title of the plan specified in those license conditions in order to maintain consistent terminology.

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 the amendments was not required since the amendments do not involve a significant hazards consideration.

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

For further details with respect to this action, see (1) the applications for amendments dated September 25, 1981 (supplemented April 7, 1982 and October 22, 1982) and November 18, 1981, (2) Amendment Nos.52 and 77 to License Nos. DPR-71 and DPR-62, and (3) the Commission's related Safety Evaluation. 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 West 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 Licensing

Dated at Bethesda, Maryland this 16th day of December, 1982

FOR THE NUCLEAR REGULATORY COMMISSION

DBVassallo

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing