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. 73
License No. DPR-62

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

   A. The application for amendment by Carolina Power & Light Company dated June 16, 1982, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C(2) of Facility Operating License No. DPR-62 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 73, 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 issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 17, 1982

ATTACHMENT TO LICENSE AMENDMENT NO. 73

FACILITY OPERATING LICENSE NO. DPR-62

DOCKET NO. 50-324

Remove the following pages and replace with identically numbered pages.

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3/4	3-6
3/4	3-8
3/4	7-12

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT AND INSTRUMENT NUMBER	APPLICABLE OPERATIONAL CONDITIONS	MINIMUM NUMBER OPERABLE CHANNELS PER TRIP SYSTEM (a)(b)	ACTION
7. Drywell Pressure - High (C72-PS-N002 A,B,C,D)	1, 2(e)	2	6
8. Scram Discharge Volume Water Level - High (C12-LSH-N013 A,B,C,D) (C12-LSH-4516A,B,C,D)	1, 2, 5(f)	2	5
9. Turbine Stop Valve - Closure (EHC-SVOS-1X,2X,3X,4X)	1(g)	4	8
10. Turbine Control Valve Fast Closure, Control Oil Pressure - Low (EHC-PSL-1756,1757,1758,1759)	1(g)	2	8
11. Reactor Mode Switch in Shutdown Position (C72A-S1)	1, 2, 3, 4, 5	1	9
12. Manual Scram (C72A-S3A,B)	1, 2, 3, 4, 5	1	10

TABLE 3.3.1-2

REACTOR PROTECTION RESPONSE TIMES

FUNCTIONAL UNIT AND INSTRUMENT NUMBER	RESPONSE TIME (Seconds)
1. Intermediate Range Monitors (C51-IRM-K601 A,B,C,D,E,F,G,H):	
a. Neutron Flux - High*	NA
b. Inoperative	NA
2. Average Power Range Monitor* (C51-APRM-CH.A,B,C,D,E,F):	
a. Neutron Flux - High, 15%	≤ 0.09
b. Flow-Biased Neutron Flux - High	NA
c. Neutron Flux - High, 120%	≤ 0.09
d. Inoperative	NA
e. Downscale	NA
f. LPRM	NA
3. Reactor Vessel Steam Dome Pressure - High (B21-PS-N023 A,B,C,D)	≤ 0.55
4. Reactor Vessel Water Level - Level #1 (B21-LIS-N017 A,B,C,D)	≤ 1.05
5. Main Steam Line Isolation Valve-Closure (B21-F022 A,B,C,D and B21-F028 A,B,C,D)	≤ 0.06
6. Main Steam Line Radiation - High (D12-RM-K603 A,B,C,D)	NA
7. Drywell Pressure - High (C72-PS-N002 A,B,C,D)	NA
8. Scram Discharge Volume Water Level - High (C12-LSH-N013 A,B,C,D) (C12-LSH-4516A,B,C,D)	NA
9. Turbine Stop Valve - Closure (EHC-SVOS-1X,2X,3X,4X)	≤ 0.06
10. Turbine Control Valve Fast Closure, Control Oil Pressure - Low (EHC-PSL-1756,1757,1758,1759)	≤ 0.08
11. Reactor Mode Switch in Shutdown Position (C72A-S1)	NA
12. Manual Scram (C72A-S3 A,B)	NA

*Neutron detectors are exempt from response time testing. Response time shall be measured from detector output or input of first electronic component in channel.

TABLE 4.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT AND INSTRUMENT NUMBER	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
8. Scram Discharge Volume Water Level - High (C12-LSH-N013 A,B,C,D) (C12-LSH-4516A,B,C,D)	NA	Q	R	1, 2, 5
9. Turbine Stop Valve - Closure (EHC-SVOS-1X,2X,3X,4X)	NA	M	R(h)	1
10. Turbine Control Valve Fast Closure, Control Oil Pressure - Low (EHC-PSL-1756,1757,1758,1759)	NA	M	R	1
11. Reactor Mode Switch in Shutdown Position (C72A-S1)	NA	R	NA	1, 2, 3, 4, 5
12. Manual Scram (C72A-S3A,B)	NA	Q	NA	1, 2, 3, 4, 5

a. Neutron detectors may be excluded from CHANNEL CALIBRATION.
b. Within 24 hours prior to start-up, if not performed within the previous 7 days.
c. The IRM channels shall be compared to the APRM channels and the SRM instruments for overlap during each start-up, if not performed within the previous 7 days.
d. When changing from CONDITION 1 to CONDITION 2, perform the required surveillance within 12 hours after entering CONDITION 2.
e. This calibration shall consist of the adjustment of the APRM readout to conform to the power values calculated by a heat balance during CONDITION 1 when THERMAL POWER $\geq$ 25% of RATED THERMAL POWER.
f. This calibration shall consist of the adjustment of the APRM flow-biased setpoint to conform to a calibrated flow signal.
g. The LPRMs shall be calibrated at least once per effective full power month (EFPM) using the TIP system.
h. This calibration shall consist of a physical inspection and actuation of these position switches.
i. Instrument alignment using a standard current source.
j. Calibration using a standard radiation source.

TABLE 3.7.5-1 (Continued)

SAFETY-RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION		ACCESSIBLE OR INACCESSIBLE	HIGH RADIATION ZONE**	ESPECIALLY DIFFICULT TO REMOVE
	Reactor Water Cleanup System				
2G31-1SS3	Drywell	54'	A	No	No
	Condensate Drains System				
2B21-51SS103	Drywell	29'	I	No	No
51SS105		26'	I	No	No
51SS106		18'	I	No	No
51SS109		31'	I	No	No
5iSS111		28'	I	No	No
51SS113		23'	I	No	No
51SS115		24'	I	No	No
51SS118		24'	I	No	No
	High Pressure Coolant Injection System				
2E41-4SS44	Drywell	40'	I	No	No
4SS45		35'	I	No	No
4SS47		40'	I	No	No
4SS49		37'	I	No	No
4SS50		40'	I	No	No
4SS51		30'	I	No	No