Distribution
Docket
ORB #3
Local PDR
NRC PDR
NRR Reading
HDenton
DEisenhut
RTedesco
WGammill
RVollmer
JMiller
LShao
BGrimes
TIppolito
JHannon
SNorris
Atty, OELD
OI&E (5)
BJones (8)
BScharf (10)
JWetmore
ACRS (16)
RDiggs
TERA
NSIC

MARCH 14 1980

Docket Nos. 50-325
and 50-324

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

Dear Mr. Jones:

The Commission has issued the enclosed Amendment No. 26 to Facility Operating License No. DPR-71 and Amendment No. 50 to Facility Operating License No. DPR-62 for the Brunswick Steam Electric Plant (BSEP), Unit Nos. 1 and 2, respectively. These amendments consist of changes to the Technical Specifications in response to your applications dated February 20, 1979, November 19, 1979, and January 24, 1980. The February 20, 1979 application was supplemented by letter dated January 14, 1980.

The amendments revise the Technical Specifications to (1) correct the table of safety related hydraulic snubbers, (2) provide for systematic implementation of instrumentation modifications, and (3) eliminate the requirement for removing the SRM "shorting links" during core alterations with control rods withdrawn. Certain modifications to the language in the proposed Technical Specifications were made to conform with the current Standard Technical Specifications. These modifications were discussed and agreed with by members of your staff.

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

Sincerely,

Original Signed by
T. A. Ippolito
Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Enclosures:
1. Amendment No. 26 to DPR-71
2. Amendment No. 50 to DPR-62
3. Safety Evaluation
4. Notice

cc w/enclosures: See next page

OFFICE	ORB #3	ORB #3	AD:ORP	OELD	ORB #3
SURNAME	SNorris:mjf	JHannon	WGammill	SHLewis	TIppolito
DATE	3/12/80	3/12/80	3/12/80	3/13/80	3/14/80

cc:

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Division of Policy Development
116 West Jones Street
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Southport - Brunswick County Library
109 W. Moore Street
Southport, North Carolina 28461

Director, Technical Assessment Division
Office of Radiation Programs (AW-459)
US EPA
Crystal Mall #2
Arlington, Virginia 20460

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

Resident Inspector
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Southport, North Carolina 28461

Mr. Fred Tollison
Plant Manager
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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. 26
License No. DPR-71

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

   A. The application for amendment by Carolina Power & Light Company dated February 20, 1979, as supplemented January 14, 1980, and applications dated November 19, 1979 and January 24, 1980, 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 Operating License No. DPR-71 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 26, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

8008250197

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

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 14, 1980

ATTACHMENT TO LICENSE AMENDMENT NO. 26

FACILITY OPERATING LICENSE NO. DPR-71

DOCKET NO. 50-325

Remove the following pages and replace with identically numbered pages.

3/4 3-1/3/4 3-2
3/4 3-5/3/4 3-6
3/4 7-11/3/4 7-12
3/4 7-13/3/4-7-14
3/4 7-15/3/4 7-16
3/4 7-17/3/4 7-18
3/4 7-19/3/4 7-20
3/4 7-21/3/4 7-22
3/4 7-23/3/4 7-24
3/4 7-25/3/4 7-26
3/4 7-27/3/4 7-28
3/4 7-29/3/4 7-30
3/4 9-3/3/4 9-4

The underlined pages are overleaf pages and are provided for convenience.

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the reactor protection system instrumentation channels shown in Table 3.3.1-1 shall be OPERABLE with REACTOR PROTECTION SYSTEM RESPONSE TIME as shown in Table 3.3.1-2. Set points and interlocks are given in Table 2.2.1-1.

APPLICABILITY: As shown in Table 3.3.1-1.

ACTION:

a. With the requirements for the minimum number of OPERABLE channels not satisfied for one trip system, place that trip system in the tripped condition within one hour or take the ACTION required by Table 3.3.1-1.

b. With the requirements for the minimum number of OPERABLE channels not satisfied for both trip systems, take the ACTION required by Table 3.3.1-1.

c. The provisions of Specification 3.0.3 are not applicable in OPERATIONAL CONDITION 5.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each reactor protection system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations during the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.1-1.

4.3.1.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months and shall include calibration of time delay relays and timers necessary for proper functioning of the trip system.

4.3.1.3 The REACTOR PROTECTION SYSTEM RESPONSE TIME of each reactor trip function of Table 3.3.1-2 shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one logic train such that both logic trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip function.

TABLE 3.3.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT AND INSTRUMENT NUMBER	APPLICABLE OPERATIONAL CONDITIONS	MINIMUM NUMBER OPERABLE CHANNELS PER TRIP SYSTEM(a)	ACTION
1. Intermediate Range Monitors: (C51-IRM-K601 A,B,C,D,E,F,G,H)			
a. Neutron Flux - High	2, 5[b]	3	1
	3, 4	2	2
b. Inoperative	2, 5	3	1
	3, 4	2	2
2. Average Power Range Monitor: (C51-APRM-CH.A,B,C,D,E,F)			
a. Neutron Flux - High, 15%	2, 5[b]	2	3
b. Flow Biased Neutron Flux - High	1	2	4
c. Fixed Neutron Flux-High, 120%	1	2	4
d. Inoperative	1, 2, 5	2	5
e. Downscale	1	2	4
f. LPRM	1, 2, 5	(c)	NA
3. Reactor Vessel Steam Dome Pressure - High (B21-PS-N023 A,B,C,D)	1, 2[d]	2	6
4. Reactor Vessel Water Level - Low, Level #1 (B21-LIS-N017 A,B,C,D)	1, 2	2	6
5. Main Steam Line Isolation Valve - Closure (B21-F022 A,B,C,D and B21-F028 A,B,C,D)	1	4	4
6. Main Steam Line Radiation - High (D12-RM-K603 A,B,C,D)	1, 2[d]	2	7

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

ACTION 10 - In OPERATIONAL CONDITION 1 or 2, be in at least HOT SHUTDOWN within 12 hours.

In OPERATIONAL CONDITION 3 or 4, lock the reactor mode switch in the Shutdown position within one hour.

In OPERATIONAL CONDITION 5, suspend all operations involving CORE ALTERATIONS or positive reactivity changes and fully insert all insertable control rods within one hour.

TABLE NOTATIONS

a. A channel may be placed in an inoperable status for up to 2 hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter.

b. The "shorting links" shall be removed from the RPS circuitry prior to and during the time any control rod is withdrawn* and shutdown margin demonstrations.

c. An APRM channel is inoperable if there are less than 2 LPRM inputs per level or less than eleven LPRM inputs to an APRM channel.

d. These functions are not required to be OPERABLE when the reactor pressure vessel head is unbolted or removed.

e. This function is not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required.

f. With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

g. These functions are bypassed when THERMAL POWER is less than 30% of RATED THERMAL POWER.

*Not required for control rods removed per Specification 3.9.10.1 or 3.9.10.2.

TABLE 3.3.1-2

REACTOR PROTECTION SYSTEM 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 (C71-PS-N002 A,B,C,D)	NA
8. Scram Discharge Volume Water Level - High (C11-LSH-N013 A,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 (C71A-S1)	NA
12. Manual Scram (C71A-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 3.7.5-1

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION		ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
	Core Spray System				
1E21-2SS16	Reactor Building	0'	A	No	No
2SS17		13'	A	No	No
2SS18		14'	A	No	No
15SS19		-3'	A	No	No
15SS20		-3'	A	No	No
28SS23		-4'	A	No	No
2SS31		68'	A	No	No
2SS32		66'	A	No	No
6SS41		70'	A	No	No
6SS42		69'	A	No	No
25SS91		-6'	A	No	No
25SS96		-6'	A	No	No
40SS106		-12'	A	No	No
40SS107		-12'	A	No	No
39SS108		-12'	A	No	No
39SS109		-12'	A	No	No
3SS46	Drywell	63'	I	No	No
3SS47		63'	I	No	No
3SS48		65'	I	No	No
3SS49		66'	I	No	No
7SS53		63'	I	No	No
7SS54		63'	I	No	No
7SS55		65'	I	No	No
7SS56		66'	I	No	No
	Reactor Water Cleanup System				
1G31-1SS3	Drywell	54'	I	No	No

TABLE 3.7.5-1 (Continued)

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION		ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
	Condensate Drain System				
1B21-51SS103	Drywell	29'	I	No	No
51SS105		29'	I	No	No
51SS106		26'	I	No	No
50SS109		18'	I	No	No
50SS111		31'	I	No	No
51SS113		28'	I	No	No
51SS115		23'	I	No	No
51SS118		24'	I	No	No
	Control Rod Drive System				
1C11-16SS1	Drywell	69'	I	No	No
16SS6		63'	I	No	No
16SS7		69'	I	No	No
16SS8		70'	I	No	No
16SS10		72'	I	No	No
16SS11		72'	I	No	No
16SS12		72'	I	No	No
	High Pressure Coolant Injection System				
1E41-4SS44	Drywell	40'	I	No	No
4SS45		35'	I	No	No
4SS47		40'	I	No	No
4SS49		37'	I	No	No
4SS50		50'	I	No	No
4SS51		30'	I	No	No
60SS9	Reactor Building	4'	A	No	No
6SS27		-5	A	No	No
6SS28		1'	A	No	No
6SS30		-1'	A	No	No
6SS32		-5'	A	No	No
6SS33		1'	A	No	No
6SS35		-1'	A	No	No
6SS36		-5	A	No	No

TABLE 3.7.5-1 (Continued)

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION		ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE** (Yes or NO)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
	High Pressure Coolant Injection System (Cont'd)				
1E41-6SS37	Reactor Building (Cont'd)	0'	A	No	No
6SS38		1'	A	No	No
6SS40		2'	A	No	No
6SS42		-4'	A	No	No
60SS52		41'	A	No	No
6SS64		-1	A	No	No
61SS71		-4	A	No	No
61SS72		-1	A	No	No
61SS73		-2	A	No	No
61SS76		37'	A	No	No
61SS77		37'	A	No	No
44SS84		14'	A	No	No
44SS86		14'	A	No	No
44SS98		12'	A	No	No
61SS99		22'	A	No	No
61SS100		17'	A	No	No
60SS101		42'	A	No	No
60SS102		42'	A	No	No
19SS103		-3'	A	No	No
2SS104		12'	A	No	No
2SS105		-17'	A	No	No
2SS106		-12'	A	No	No
22SS178		-11'	A	No	No
20SS195		4'	A	No	No
20SS196		4'	A	No	No
22SS197		13'	A	No	No
	Standby Gas Treatment System				
1SGT-8SS17	Reactor Building	69'	A	No	No

TABLE 3.7.5-1 (Continued)

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION		ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
	Instrument Sensing System				
1B21-701SS164	Drywell	104'	I	No	No
701SS167		104'	I	No	No
701SS169		100'	I	No	No
701SS170		103'	I	No	No
701SS171		99'	I	No	No
701SS172		101'	I	No	No
701SS175		100'	I	No	No
701SS177		94'	I	No	No
701SS178		97'	I	No	No
701SS179		96'	I	No	No
701SS184		88'	I	No	No
	Reactor Closed Cooling Water System				
1RCC-32SS30	Reactor Building	55'	A	No	No
32SS45		60'	A	No	No
36SS78		54'	A	No	No
37SS79		54'	A	No	No
39SS80		59'	A	No	No
38SS81		54'	A	No	No
7SS112		57'	A	No	No
47SS167		59'	A	No	No
47SS168		58'	A	No	No
48SS169		60'	A	No	No
50SS272		4'	A	No	No
60SS121	Drywell	17'	I	No	No
60SS122		16'	I	No	No
65SS128		7'	I	No	No
65SS129		9'	I	No	No
71SS139		9'	I	No	No
73SS145		5'	I	No	No
19SS157		21'	I	No	No
19SS160		29	I	No	No

TABLE 3.7.5-1 (Continued)

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION		ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
	Primary Steam System				
1PSN-A2SS30	Drywell	65'	I	No	No
A2SS31		64'	I	No	No
A3SS32		40'	I	No	No
A3SS33		35'	I	No	No
A3SS34		35'	I	No	No
A3SS35		41'	I	No	No
A5SS38		22'	I	No	No
B2SS40		63'	I	No	No
B2SS41		64'	I	No	No
B3SS42		40'	I	No	No
B3SS43		35'	I	No	No
B3SS44		40'	I	No	No
B3SS46		46'	I	No	No
B3SS47		35'	I	No	No
B3SS48		40'	I	No	No
B5SS50		18'	I	No	No
B5SS51		22'	I	No	No
C2SS54		63'	I	No	No
C2SS55		64'	I	No	No
C3SS56		40'	I	No	No
C3SS57		35'	I	No	No
C3SS58		40'	I	No	No
C3SS60		38'	I	No	No
C3SS61		35'	I	No	No
C3SS62		39'	I	No	No
C5SS64		18'	I	No	No
C5SS65		22'	I	No	No
D2SS68		65'	I	No	No
D2SS69		65'	I	No	No
D3SS70		40'	I	No	No
D3SS71		35'	I	No	No
D3SS72		35'	I	No	No
D3SS73		41'	I	No	No
D5SS76		22'	I	No	No
B3SS190		42'	I	No	No
C3SS272		42'	I	No	No
A3SS292		42'	I	No	No

TABLE 3.7.5-1 (Continued)

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION		ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
	Reactor Core Isolation Cooling System				
1E51-4SS45	Drywell	31'	I	No	No
3SS46		39'	I	No	No
3SS47		39'	I	No	No
4SS66		39'	I	No	No
4SS68		40'	I	No	No
4SS69		40'	I	No	No
4SS70		39'	I	No	No
4SS71		36'	I	No	No
4SS72		31'	I	No	No
4SS73		30'	I	No	No
41SS51	Reactor Building	40'	A	No	No
42SS74		20'	A	No	No
42SS75		20'	A	No	No
42SS76		18'	A	No	No
42SS77		5'	A	No	No
42SS78		0'	A	No	No
42SS79		4'	A	No	No
42SS80		-13'	A	No	No
42SS81		-16'	A	No	No
42SS82		-9'	A	No	No
40SS83		-9'	A	No	No
40SS84		-9'	A	No	No
40SS85		-12'	A	No	No
40SS86		-9	A	No	No
40SS87		-15'	A	No	No
40SS88		-13'	A	No	No
41SS89		41'	A	No	No
41SS95		-41'	A	No	No
19SS113		-17'	A	No	No
19SS114		-16'	A	No	No
19SS129		0'	A	No	No

TABLE 3.7.5-1 (Continued)

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION		ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
	Nuclear Steam Vent System				
1B21-44SS129	Drywell	104'	I	No	No
44SS131		93'	I	No	No
44SS134		99'	I	No	No
44SS136		97'	I	No	No
44SS137		96'	I	No	No
44SS138		95'	I	No	No
44SS141		87'	I	No	No
44SS142		87'	I	No	No
44SS143		87'	I	No	No
44SS146		87'	I	No	No
44SS147		82'	I	No	No
44SS149		85'	I	No	No
44SS150		83'	I	No	No
47SS155		75'	I	No	No
47SS156		78'	I	No	No
47SS157		75	I	No	No
	Standby Liquid Control System				
1C41-9SS4	Drywell	63'	I	No	No
9SS5		47'	I	No	No
9SS8		42'	I	No	No
9SS10		38'	I	No	No
9SS11		39'	I	No	No
9SS12		69'	I	No	No
9SS13		52'	I	No	No
9SS26	Reactor	72'	A	No	No
9SS27	Building	72'	A	No	No
6SS34		84	A	No	No

TABLE 3.7.5-1 (Continued)

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION		ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
	Fuel Pool Cooling System				
1G41-1SS22	Reactor Building	12'	A	No	No
1SS24		38'	A	No	No
1SS30		38'	A	No	No
12SS32		9'	A	No	No
12SS33		9'	A	No	No
15SS37		111'	A	No	No
20SS76		108'	A	No	No
11SS79		89'	A	No	No
22SS85		108'	A	No	No
12SS98		88'	A	No	No
6SS111		88'	A	No	No
7SS121		87	A	No	No
5SS152		82	A	No	No
	Reactor Recirculation System				
1B32-SSA1	Drywell	8'	I	No	No
SSB1		81'	I	No	No
SSA2		11'	I	No	No
SSB2		11'	I	No	No
SSA3		11'	I	No	No
SSB3		11'	I	No	No
SSA4		21'	I	No	No
SSB4		21'	I	No	No
SSA5		21'	I	No	No
SSB5		21'	I	No	No
SSA6		27'	I	No	No
SSB6		27'	I	No	No
SSB9A		30'	I	No	No
SSB9B		30'	I	No	No
SSA10		24'	I	No	No

TABLE 3.7.5-1 (Continued)

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION		ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
	Reactor Recirculation System (Continued)				
SSA11	Drywell (Cont'd)	11'	I	No	No
SSB11		11'	I	No	No
SSA12A		30'	I	No	No
SSA12B		30'	I	No	No
SSB12A		30'	I	No	No
SSB12B		30'	I	No	No
	Reactor Vessel Instrumentation				
1PS-3554	Drywell	32'	I	No	No
3558		32'	I	No	No
3561		32'	I	No	No
3562		60'	I	No	No
3567		63'	I	No	No
3570		32'	I	No	No
3613		32'	I	No	No
3617A		32'	I	No	No
3617B		32'	I	No	No
3751		34'	I	No	No
3752		34'	I	No	No
	Off Gas System				
1PS-3417	Nitrogen and Off Gas Bldg	31'	A	No	No
3418A		33'	A	No	No
3418B		33'	A	No	No
3419A		33'	A	No	No
3419B		33'	A	No	No
3819		37'	A	No	No
	Reactor Feedwater System				
1B21-2SS3	Drywell	38'	I	No	No
2SS4		56'	I	No	No
3SS6		41'	I	No	No
3SS9		39'	I	No	No
3SS11		41'	I	No	No
3SS12		40'	I	No	No
3SS13		61'	I	No	No
5SS17		38'	I	No	No
5SS18		56'	I	No	No

TABLE 3.7.5-1 (Continued)

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
	Reactor Feedwater System (Continued)			
6SS20	Drywell(Cont'd) 41'	I	No	No
6SS23	39'	I	No	No
6SS25	41'	I	No	No
6SS26	40'	I	No	No
6SS27	63'	I	No	No
1SS227	34'	I	No	No
1SS228	38'	I	No	No
1SS229	53'	I	No	No
2SS230	62'	I	No	No
2SS231	40'	I	No	No
2SS232	36'	I	No	No
3SS233	40'	I	No	No
3SS234	48'	I	No	No
3SS235	63'	I	No	No
4SS236	34'	I	No	No
4SS237	38'	I	No	No
5SS238	53'	I	No	No
5SS239	61'	I	No	No
6SS240	41'	I	No	No
6SS241	36'	I	No	No
6SS242	39'	I	No	No
6SS243	48'	I	No	No
6SS244	61'	I	No	No
	Residual Heat Removal System			
1E11-90SS267	Drywell 79'	I	No	No
90SS268	86'	I	No	No
90SS271	86'	I	No	No
90SS274	93'	I	No	No
90SS275	93'	I	No	No
90SS277	96'	I	No	No
90SS278	96'	I	No	No
90SS280	101'	I	No	No
90SS281	93'	I	No	No
90SS282	101'	I	No	No

TABLE 3.7.5-1 (Continued)

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
	Residual Heat Removal System (Continued)			
1E11-90SS283	Drywell(Cont'd) 93'	I	No	No
90SS284	101'	I	No	No
90SS285	100'	I	No	No
1SS302	35'	I	No	No
1SS303	34'	I	No	No
1SS305	20'	I	No	No
1SS306	19'	I	No	No
84SS309	33'	I	No	No
84SS311	35'	I	No	No
84SS312	35'	I	No	No
87SS315	33'	I	No	No
87SS317	35'	I	No	No
87SS318	35'	I	No	No
90SS388	79'	I	No	No
90SS389	96'	I	No	No
90SS390	96'	I	No	No
90SS391	93'	I	No	No
90SS392	99'	I	No	No
1E11-17SS3	Reactor Building 3'	A	No	No
17SS4	-1'	A	No	No
46SS7	12'	A	No	No
46SS9	12'	A	No	No
56SS13	5'	A	No	No
56SS15	4'	A	No	No
20SS19	-3'	A	No	No
95SS20	-3'	A	No	No
95SS22	9'	A	No	No
95SS23	9'	A	No	No
20SS28	-3'	A	No	No
58SS32	3'	A	No	No
58SS33	-4'	A	No	No
58SS35	8'	A	No	No
58SS36	8'	A	No	No

TABLE 3.7.5-1 (Continued)

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION		ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
	Residual Heat Removal System (Continued)				
1E11-18SS40	Reactor Bldg. (Cont'd)	8'	A	No	No
18SS47		12'	A	No	No
18SS48		13'	A	No	No
68SS59		15'	A	No	No
21SS63		8'	A	No	No
21SS70		13'	A	No	No
21SS71		10'	A	No	No
37SS95		-3'	A	No	No
61SS110		6'	A	No	No
116SS143		-11'	A	No	No
113SS157		-11'	A	No	No
37SS184		9'	A	No	No
53SS192		78'	A	No	No
53SS195		14'	A	No	No
53SS197		14'	A	No	No
53SS200		14'	A	No	No
50SS201		14'	A	No	No
89SS208		5'	A	No	No
46SS216		28'	A	No	No
46SS217		31'	A	No	No
46SS218		30'	A	No	No
47SS223		33'	A	No	No
47SS224		36'	A	No	No
47SS225		36'	A	No	No
47SS227		39'	A	No	No
47SS228		39'	A	No	No
95SS233		20'	A	No	No
95SS234		24'	A	No	No
95SS235		31'	A	No	No
131SS255		42'	A	No	No
131SS257		22'	A	No	No
132SS263		30'	A	No	No

TABLE 3.7.5-1 (Continued)

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION		ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
	Residual Heat Removal System (Continued)				
1E11-132SS264	Reactor Bldg. (Cont'd)	31'	A	No	No
21SS296		39'	A	No	No
21SS297		39'	A	No	No
47SS323		42'	A	No	No
47SS326		42'	A	No	No
47SS328		42'	A	No	No
49SS330		42'	A	No	No
49SS331		42'	A	No	No
49SS333		42'	A	No	No
49SS334		43'	A	No	No
49SS336		40'	A	No	No
128SS355		42'	A	No	No
49SS359		42'	A	No	No
127SS376		59'	A	No	No
128SS387		43'	A	No	No
2SS396		5'	A	No	No
2SS397		3'	A	No	No
5SS398		-41'	A	No	No
2SS399		-3'	A	No	No
5SS400		-3'	A	No	No
4SS401		-12'	A	No	No
5SS402		-12'	A	No	No
3SS403		-11'	A	No	No
6SS404		-12'	A	No	No
8SS405		-14'	A	No	No
6SS406		-14'	A	No	No
8SS407		-15'	A	No	No
12SS408		-14'	A	No	No
16SS409		-9'	A	No	No
113SS410		-9'	A	No	No
9SS411		-14'	A	No	No
109SS412		-14'	A	No	No
2SS413		0'	A	No	No
132SS414		43'	A	No	No

TABLE 3.7.5-1 (Continued)

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION		ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
	Residual Heat Removal System (Continued)				
1E11-20SS415	Reactor Bldg. (Cont'd)	-4'	A	No	No
20SS416		-4'	A	No	No
95SS417		0'	A	No	No
95SS418		-4'	A	No	No
45SS422		-2'	A	No	No
60SS423		-4'	A	No	No
60SS425		-2'	A	No	No
127SS426		43'	A	No	No
128SS427		21'	A	No	No
128SS428		28'	A	No	No
128SS429		39'	A	No	No
128SS430		31'	A	No	No
128SS431		23'	A	No	No
127SS433		14'	A	No	No
127SS434		37'	A	No	No
127SS435		15'	A	No	No
60SS437		12'	A	No	No
60SS438		12'	A	No	No
60SS440		13'	A	No	No
65SS441		3'	A	No	No
65SS442		3'	A	No	No
60SS443		11'	A	No	No
73SS444		21'	A	No	No
21SS445		5'	A	No	No
83SS446		10'	A	No	No
68SS448		13'	A	No	No
75SS449		7'	A	No	No
61SS450		2'	A	No	No
60SS451		13'	A	No	No
60SS452		13'	A	No	No
60SS453		10'	A	No	No
60SS454		10'	A	No	No
89SS459		11'	A	No	No

TABLE 3.7.5-1 (Continued)

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION		ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
	Residual Heat Eemoval System (Continued)				
1E11-89SS460	Reactor Bldg. (Cont'd)	10'	A	No	No
89SS461		6'	A	No	No
53SS462		15'	A	No	No
53SS463		14'	A	No	No
53SS464		14'	A	No	No
53SS465		14'	A	No	No
53SS466		14'	A	No	No
50SS467		14'	A	No	No
50SS468		17'	A	No	No
18SS469		53'	A	No	No
18SS470		43'	A	No	No
89SS480		67'	A	No	No
89SS487		67'	A	No	No
89SS489		67'	A	No	No
89SS491		67'	A	No	No
91SS499		69'	A	No	No
91SS500		57'	A	No	No
56SS504		14'	A	No	No
56SS505		7'	A	No	No
56SS506		3'	A	No	No
56SS507		3'	A	No	No
56SS508		4'	A	No	No
46SS509		8'	A	No	No
46SS510		11'	A	No	No
46SS511		10'	A	No	No
46SS512		-1'	A	No	No
58SS514		14'	A	No	No
49SS515		37'	A	No	No
49SS516		37'	A	No	No
49SS517		-5'	A	No	No
51SS546		32'	A	No	No
51SS547		28'	A	No	No
115SS549		31'	A	No	No

TABLE 3.7.5-1 (Continued)

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION		ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
	Residual Heat Removal System (Continued)				
1E11-54SS551	Reactor Bldg. (Cont'd)	31'	A	No	No
54SS552		28'	A	No	No
98SS554		32'	A	No	No
58SS563		7'	A	No	No
58SS565		13'	A	No	No
58SS566		6'	A	No	No
107SS573		15'	A	No	No
69SS574		6'	A	No	No
91SS575		53'	A	No	No
68SS577		8'	A	No	No
53SS596		14'	A	No	No
50SS597		26'	A	No	No
	Service Water System				
1SW-133SS22	Reactor Building	-6'	A	No	No
110SS35		-5'	A	No	No
174SS70		42'	A	No	No
173SS72		14'	A	No	No
142SS74		40'	A	No	No
142SS75		40'	A	No	No
140SS80		40'	A	No	No
140SS82		70'	A	No	No
140SS86		45'	A	No	No
153SS102		44'	A	No	No
153SS109		44'	A	No	No
173SS110		48'	A	No	No
173SS114		70'	A	No	No
153SS115		44'	A	No	No
103SS117		41'	A	No	No
103SS121		38'	A	No	No
103SS126		60'	A	No	No
103SS127		57'	A	No	No

TABLE 3.7.5-1 (Continued)

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION		ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
	Service Water System (Continued)				
1SW-106SS131	Reactor Bldg. (Cont'd)	60'	A	No	No
100SS145		59'	A	No	No
100SS149		60'	A	No	No
106SS151		59	A	No	No
106SS156		60'	A	No	No
142SS163		60'	A	No	No
142SS164		8'	A	No	No
142SS165		8'	A	No	No
175SS166		42'	A	No	No
140SS167		42'	A	No	No
142SS168		58'	A	No	No
142SS169		71'	A	No	No
138SS170		55'	A	No	No
140SS171		58'	A	No	No
139SS172		58'	A	No	No
174SS174		42'	A	No	No
173SS175		30'	A	No	No
133SS177		-5'	A	No	No
100SS193		62'	A	No	No
139SS209		56'	A	No	No
139SS210		56'	A	No	No
106SS211		60'	A	No	No
106SS212		60'	A	No	No
106SS213		59'	A	No	No
106SS214		60	A	No	No
127SS215		17'	A	No	No
123SS216		17'	A	No	No

TABLE 3.7.5-1 (Continued)

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION		ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
	Steam Relief Discharge System				
1B21-19SS192	Drywell	43'	I	No	No
19SS193		44'	I	No	No
19SS194		37'	I	No	No
19SS195		37'	I	No	No
19SS196		32'	I	No	No
19SS198		18'	I	No	No
19SS199		18'	I	No	No
26SS201		42'	I	No	No
26SS203		36'	I	No	No
26SS204		36'	I	No	No
26SS206		18'	I	No	No
26SS207		18'	I	No	No
26SS208		11'	I	No	No
18SS211		42'	I	No	No
18SS212		37'	I	No	No
18SS213		37'	I	No	No
18SS215		18'	I	No	No
18SS216		17'	I	No	No
18SS218		5'	I	No	No
11SS220		38'	I	No	No
11SS221		37'	I	No	No
11SS222		36'	I	No	No
11SS223		34'	I	No	No
11SS224		34'	I	No	No
33SS245		43'	I	No	No
33SS246		43'	I	No	No
33SS248		45'	I	No	No
33SS249		36'	I	No	No
33SS250		36'	I	No	No
33SS251		32'	I	No	No
20SS256		44'	I	No	No
20SS257		37'	I	No	No

TABLE 3.7.5-1 (Continued)

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
	Steam Relief Discharge System (Continued)			
1B21-20SS258	Drywell (Cont'd) 36'	I	No	No
20SS260	18'	I	No	No
20SS261	20'	I	No	No
20SS262	15'	I	No	No
27SS264	10'	I	No	No
27SS266	17'	I	No	No
27SS267	18'	I	No	No
27SS269	36'	I	No	No
27SS270	44'	I	No	No
58SS275	18'	I	No	No
58SS276	30'	I	No	No
58SS277	31'	I	No	No
58SS279	36'	I	No	No
58SS280	37'	I	No	No
58SS281	38'	I	No	No
12SS286	39'	I	No	No
12SS287	39'	I	No	No
12SS288	44'	I	No	No
12SS289	39'	I	No	No
12SS290	43'	I	No	No
34SS296	35'	I	No	No
34SS297	35'	I	No	No
34SS298	44'	I	No	No
34SS299	39'	I	No	No
34SS300	44'	I	No	No
26SS302	11'	I	No	No
26SS303	5'	I	No	No
58SS304	10'	I	No	No
58SS305	5'	I	No	No
58SS306	15'	I	No	No
58SS307	16'	I	No	No
12SS308	28'	I	No	No
12SS309	29'	I	No	No

TABLE 3.7.5-1 (Continued)

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
	Steam Relief Discharge System (Continued)			
1B21-12SS310	Drywell (Cont'd) 17'	I	No	No
12SS311	11'	I	No	No
12SS312	11'	I	No	No
11SS313	16'	I	No	No
11SS314	11'	I	No	No
11SS315	7'	I	No	No
19SS316	11'	I	No	No
19SS317	17'	I	No	No
18SS318	11'	I	No	No
20SS319	10'	I	No	No
59SS320	11'	I	No	No
59SS325	34'	I	No	No
59SS326	35'	I	No	No
59SS327	36'	I	No	No
59SS329	38'	I	No	No
59SS330	44'	I	No	No
33SS332	26'	I	No	No
33SS333	7'	I	No	No
33SS334	11'	I	No	No
33SS335	29'	I	No	No
34SS336	27'	I	No	No
34SS337	18'	I	No	No
34SS338	18'	I	No	No
34SS339	11'	I	No	No
34SS340	7'	I	No	No

*Snubbers may be added to safety related systems without prior License Amendment to Table 3.7.5-1 provided that safety evaluations, documentation and reporting are provided in accordance with 10 CFR 50.59 and that a proposed revision to Table 3.7.5-1 is included with the next License Amendment request.

**Modifications to this table due to changes in high radiation areas shall be submitted to the NRC as part of the next License Amendment request.

3/4.9.2 INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.9.2 At least 2 source range monitor (SRM) channels* shall be OPERABLE and inserted to the normal operation level with:

a. A continuous visual indication in the control room,

b. One of the SRM detectors located in the quadrant where CORE ALTERATIONS are being performed and the other SRM detector located in an adjacent quadrant, and

c. The "shorting links" removed from the RPS circuitry prior to and during the time any control rod is withdrawn** and shutdown margin demonstrations.

APPLICABILITY: CONDITION 5.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and fully insert all insertable control rods. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.2 Each of the above required SRM channels shall be demonstrated OPERABLE by:

a. At least once per 12 hours;

1. Performance of a CHANNEL CHECK,

*The use of special movable detectors during CORE ALTERATIONS in place of the normal SRM nuclear detectors is permissible as long as these special detectors are connected to the normal SRM circuits.

**Not required for control rods removed per Specifications 3.9.10.1 or 3.9.10.2.

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying the detectors are inserted to the normal operating level, and

3. During CORE ALTERATIONS, verifying that the detector of an OPERABLE SRM channel is located in the core quadrant where CORE ALTERATIONS are being performed and one is located in the adjacent quadrant.

b. Performance of a CHANNEL FUNCTIONAL TEST:

1. Within 24 hours prior to the start of CORE ALTERATIONS, and

2. At least once per 7 days.

c. Verifying at least once per 12 hours during CORE ALTERATIONS that the channel count rate is at least 3 cps.

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. 50
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 February 20, 1979, as supplemented January 14, 1980, and applications dated November 19, 1979 and January 24, 1980, 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 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. 50, 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

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 14, 1980

ATTACHMENT TO LICENSE AMENDMENT NO. 50

FACILITY OPERATING LICENSE NO. DPR-62

DOCKET NO. 50-324

Remove the following pages and replace with identically numbered pages.

3/4 3-1/3/4 3-2
3/4 3-5/3/4 3-6
3/4 7-11/3/4 7-12
3/4 7-13/3/4 7-14
3/4 7-15/3/4 7-16
3/4 7-17/3/4 7-18
3/4 7-19/3/4 7-20
3/4 7-21/3/4 7-22
3/4 7-23/3/4 7-24
3/4 7-25/3/4 7-26
3/4 7-27/3/4 7-28
3/4 7-29/3/4 7-30
3/4 9-3/3/4 9-4

The underlined pages are overleaf pages and are provided for convenience.

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the reactor protection system instrumentation channels shown in Table 3.3.1-1 shall be OPERABLE with REACTOR PROTECTION SYSTEM RESPONSE TIME as shown in Table 3.3.1-2. Set points and interlocks are given in Table 2.2.1-1.

APPLICABILITY: As shown in Table 3.3.1-1.

ACTION:

a. With the requirements for the minimum number of OPERABLE channels not satisfied for one trip system, place that trip system in the tripped condition within one hour or take the ACTION required by Table 3.3.1-1.

b. With the requirements for the minimum number of OPERABLE channels not satisfied for both trip systems, take the ACTION required by Table 3.3.1-1.

c. The provisions of Specification 3.0.3 are not applicable in OPERATIONAL CONDITION 5.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each reactor protection system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations during the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.1-1.

4.3.1.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months and shall include calibration of time delay relays and timers necessary for proper functioning of the trip system.

4.3.1.3 The REACTOR PROTECTION SYSTEM RESPONSE TIME of each reactor trip function of Table 3.3.1-2 shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one logic train such that both logic trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip function.

TABLE 3.3.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT AND INSTRUMENT NUMBER	APPLICABLE OPERATIONAL CONDITIONS	MINIMUM NUMBER OPERABLE CHANNELS PER TRIP SYSTEM (a)	ACTION
1. Intermediate Range Monitors: (C51-IRM-K601 A,B,C,D,E,F,G,H)			
a. Neutron Flux - High	2, 5[b]	3	1
	3, 4	2	2
b. Inoperative	2, 5	3	1
	3, 4	2	2
2. Average Power Range Monitor: (C51-APRM-CH.A,B,C,D,E,F)			
a. Neutron Flux - High, 15%	2, 5[b]	2	3
b. Flow Biased Neutron Flux - High	1	2	4
c. Fixed Neutron Flux-High, 120%	1	2	4
d. Inoperative	1, 2, 5	2	5
e. Downscale	1	2	4
f. LPRM	1, 2, 5	(c)	NA
3. Reactor Vessel Steam Dome Pressure - High (B21-PS-N023 A,B,C,D)	1, 2[d]	2	6
4. Reactor Vessel Water Level - Low, Level #1 (B21-LIS-N017 A,B,C,D)	1, 2	2	6
5. Main Steam Line Isolation Valve - Closure (B21-F022 A,B,C,D and B21-F028 A,B,C,D)	1	4	4
6. Main Steam Line Radiation - High (D12-RM-K603 A,B,C,D)	1, 2[d]	2	7

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

ACTION 10 - In OPERATIONAL CONDITION 1 or 2, be in at least HOT SHUTDOWN within 12 hours.

In OPERATIONAL CONDITION 3 or 4, lock the reactor mode switch in the Shutdown position within one hour.

In OPERATIONAL CONDITION 5, suspend all operations involving CORE ALTERATIONS or positive reactivity changes and fully insert all insertable control rods within one hour.

TABLE NOTATIONS

a. A channel may be placed in an inoperable status for up to 2 hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter.

b. The "shorting links" shall be removed from the RPS circuitry prior to and during the time any control rod is withdrawn* and shutdown margin demonstrations.

c. An APRM channel is inoperable if there are less than 2 LPRM inputs per level or less than eleven LPRM inputs to an APRM channel.

d. These functions are not required to be OPERABLE when the reactor pressure vessel head is unbolted or removed.

e. This function is not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required.

f. With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

g. These functions are bypassed when THERMAL POWER is less than 30% of RATED THERMAL POWER.

*Not required for control rods removed per Specification 3.9.10.1 or 3.9.10.2.

TABLE 3.3.1-2

REACTOR PROTECTION SYSTEM 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-GH.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)	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 3.7.5-1

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION		ACCESSIBLE OR INACCESSIBLE	HIGH RADIATION ZONE**	ESPECIALLY DIFFICULT TO REMOVE
	Core Spray System				
2E21-2SS32	Reactor Building	66'	A	No	No
26SS91		-6'	A	No	No
2SS16		0'	A	No	No
2SS17		13'	A	No	No
15SS19		-3'	A	No	No
15SS20		-3'	A	No	No
28SS23		-4'	A	No	No
25SS96		-6'	A	No	No
40SS106		-12'	A	No	No
40SS107		-12'	A	No	No
39SS108		-12'	A	No	No
39SS109		-12'	A	No	No
2SS31		68'	A	No	No
6SS41		70'	A	No	No
6SS42		69'	A	No	No
2SS18		14'	A	No	No
2E21-3SS46	Drywell	63'	I	No	No
3SS47		63'	I	No	No
3SS48		65'	I	No	No
3SS49		66'	I	No	No
7SS53		63'	I	No	No
7SS54		63'	I	No	No
7SS55		65'	I	No	No
7SS56		66'	I	No	No

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
51SS111		28'	I	No	No
51SS113		23'	I	No	No
51SS115		24'	I	No	No
51SS118		24'	I	No	No
	Control Rod Drive System				
2C12-16SS1	Drywell	69'	I	No	No
16SS6		68'	I	No	No
16SS7		69'	I	No	No
16SS8		70'	I	No	No
16SS10		72'	I	No	No
16SS11		72'	I	No	No
16SS12		72'	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

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
	High Pressure Coolant Injection System (Continued)				
2E41-2SS5	Reactor Building	-7'	A	No	No
61SS73		-2'	A	No	No
60SS9		4'	A	No	No
20SS15		-5'	A	No	No
44SS84		14'	A	No	No
44SS86		14'	A	No	No
44SS98		12'	A	No	No
19SS103		-3'	A	No	No
2SS104		12'	A	No	No
2SS105		-17'	A	No	No
2SS106		-12'	A	No	No
60SS52		41'	A	No	No
21SS76		37'	A	No	No
21SS77		37'	A	No	No
61SS99		22'	A	No	No
61SS100		17'	A	No	No
60SS101		42'	A	No	No
60SS102		42'	A	No	No
22SS178		-11'	A	No	No
6SS27		1'	A	No	No
6SS28		1'	A	No	No
6SS30		-1'	A	No	No
6SS32		-5'	A	No	No
6SS33		1'	A	No	No
6SS35		-1'	A	No	No
6SS36		-5'	A	No	No
6SS37		0	A	No	No
6SS38		1'	A	No	No
6SS40		2'	A	No	No
6SS42		-4'	A	No	No
6SS64		-1'	A	No	No
61SS71		-4'	A	No	No
61SS72		-1'	A	No	No

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
	Standby Gas Treatment System				
2-SGT8SS17	Reactor Building	69'	A	No	No
	Instrument Sensing System				
2B21-701SS164	Drywell	104'	I	No	No
701SS167		104'	I	No	No
701SS169		100'	I	No	No
701SS170		103'	I	No	No
701SS171		99'	I	No	No
701SS172		101'	I	No	No
701SS175		100'	I	No	No
701SS177		94'	I	No	No
701SS178		97'	I	No	No
701SS179		96'	I	No	No
701SS184		88'	I	No	No
	Reactor Building Closed Cooling Water System				
2RCC-58SS11	Reactor Building	39'	A	No	No
50SS33		48'	A	No	No
50SS34		48'	A	No	No
10SS56		48'	A	No	No
17SS74		38'	A	No	No
17SS75		48'	A	No	No
17SS76		39'	A	No	No
32SS30		55'	A	No	No
32SS52		64'	A	No	No
36SS78		54'	A	No	No
37SS79		54'	A	No	No

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 Building Closed Cooling Water System (Continued)				
2RCC-39SS80	Reactor Bldg (Cont'd)	59'	A	No	No
38SS81		54'	A	No	No
34SS82		57'	A	No	No
35SS83		57'	A	No	No
6SS107		60'	A	No	No
7SS112		57'	A	No	No
47SS168		58'	A	No	No
48SS169		60'	A	No	No
32SS45		60'	A	No	No
51SS58		88'	A	No	No
47SS167		59'	A	No	No
2RCC-60SS121	Drywell	17'	I	No	No
60SS122		16'	I	No	No
65SS128		7'	I	No	No
65SS129		7'	I	No	No
71SS139		9'	I	No	No
73SS145		5'	I	No	No
19SS157		21'	I	No	No
19SS160		29'	I	No	No
	Primary Steam System				
2PSN-A2SS30	Drywell	65'	I	No	No
A2SS31		64'	I	No	No
A3SS32		40'	I	No	No
A3SS33		35'	I	No	No
A3SS34		35'	I	No	No
A5SS38		22'	I	No	No
B2SS40		63'	I	No	No
B2SS41		64'	I	No	No
B3SS43		35'	I	No	No

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
	Primary Steam System (Continued)				
2PSN-B3SS44	Drywell (Cont'd)	40'	I	No	No
B3SS46		38'	I	No	No
B3SS47		35'	I	No	No
B3SS48		40'	I	No	No
B5SS50		18'	I	No	No
B5SS51		22'	I	No	No
C2SS54		63'	I	No	No
C2SS55		64'	I	No	No
C3SS56		40'	I	No	No
C3SS57		35'	I	No	No
C3SS58		40'	I	No	No
C3SS60		38'	I	No	No
C3SS61		35'	I	No	No
C3SS62		39'	I	No	No
C5SS64		18'	I	No	No
C5SS65		22'	I	No	No
D2SS68		65'	I	No	No
D2SS69		65'	I	No	No
D3SS70		40'	I	No	No
D3SS71		35'	I	No	No
D3SS72		35'	I	No	No
D3SS73		41'	I	No	No
D5SS76		22'	I	No	No
A3SS35		41'	I	No	No
B3SS42		40'	I	No	No

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 Core Isolation Cooling System				
2E51-4SS45	Drywell	31'	I	No	No
3SS46		39'	I	No	No
3SS47		39'	I	No	No
4SS66		39'	I	No	No
4SS100		40'	I	No	No
4SS101		40'	I	No	No
4SS102		39'	I	No	No
4SS103		36'	I	No	No
4SS104		31'	I	No	No
4SS105		30'	I	No	No
42SS76		18'	A	No	No
42SS77		5'	A	No	No
42SS79		4'	A	No	No
42SS80		-13'	A	No	No
42SS81		-16'	A	No	No
42SS82		-9'	A	No	No
40SS83		-9'	A	No	No
40SS84		-9'	A	No	No
40SS85		-12'	A	No	No
40SS86		-9'	A	No	No
40SS87		-15'	A	No	No
40SS88		-13'	A	No	No
41SS51		40'	A	No	No
42SS74		20'	A	No	No
42SS75		20'	A	No	No

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 Core Isolation Cooling System (Continued)				
2E51-41SS89	Drywell (Cont'd)	41'	A	No	No
41SS95		41'	A	No	No
19SS113		-17'	A	No	No
19SS114		-16'	A	No	No
16SS91		-6'	A	No	No
17SS94		-6'	A	No	No
42SS78		0	A	No	No
49SS129		0	A	No	No

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
	Nuclear Steam Vent System				
2B21-44SS129	Drywell	104'	I	No	No
44SS131		93'	I	No	No
44SS134		99'	I	No	No
44SS136		97'	I	No	No
44SS137		96'	I	No	No
44SS138		95'	I	No	No
44SS141		87'	I	No	No
44SS142		87'	I	No	No
44SS143		87'	I	No	No
44SS146		87'	I	No	No
44SS147		82'	I	No	No
44SS149		85'	I	No	No
44SS150		83'	I	No	No
44SS155		75'	I	No	No
44SS156		78'	I	No	No
44SS157		75'	I	No	No
	Standby Liquid Control System				
2C41-9SS4	Drywell	63'	I	No	No
9SS5		47'	I	No	No
9SS8		42'	I	No	No
9SS10		38'	I	No	No
9SS11		39'	I	No	No
9SS12		69'	I	No	No
9SS13		52'	I	No	No
2C41-9SS26	Reactor Building	72'	A	No	No
9SS27		72'	A	No	No
9SS34		84'	A	No	No

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
	Fuel Pool Cooling System				
2G41-1SS22	Reactor Building	12'	A	No	No
12SS32		9'	A	No	No
12SS33		9'	A	No	No
1SS24		38'	A	No	No
1SS30		38'	A	No	No
15SS37		111'	A	No	No
11SS54		88'	A	No	No
10SS67		89'	A	No	No
20SS76		108'	A	No	No
11SS79		89'	A	No	No
22SS85		108'	A	No	No
12SS98		88	A	No	No
6SS111		88	A	No	No
7SS121		87'	A	No	No
5SS152		82'	A	No	No
59SS159		84'	A	No	No
	Reactor Recirculation System				
2B32-SSA1	Drywell	8'	I	No	No
SSB1		8'	I	No	No
SSA2		11'	I	No	No
SSB2		11'	I	No	No
SSA3		11'	I	No	No
SSB3		11'	I	No	No
SSA4		21'	I	No	No
SSB4		21'	I	No	No
SSA5		21'	I	No	No
SSB5		21'	I	No	No

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 Recirculation System (Continued)					
2B32-SSA6	Drywell (Cont'd)	27'	I	No	No
SSB6		27'	I	No	No
SSB9A		30'	I	No	No
SSB9B		30'	I	No	No
SSA10		24'	I	No	No
SSA11		11'	I	No	No
SSB11		4'	I	No	No
SSA12A		30'	I	No	No
SSA12B		30'	I	No	No
SSB12A		30'	I	No	No
SSB12B		30'	I	No	No
Reactor Vessel Instrumentation					
2PS-3554	Drywell	63'	I	No	No
3558		68'	I	No	No
3561		60'	I	No	No
3562		60'	I	No	No
3567		65'	I	No	No
3570		65'	I	No	No
3613		90'	I	No	No
3705A		32'	I	No	No
3705B		32'	I	No	No
3706		32'	I	No	No
3707		32'	I	No	No
3708		32'	I	No	No
3709		32'	I	No	No
3710		32'	I	No	No
3722A		82'	I	No	No
3722B		82'	I	No	No

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
	Off Gas System				
2PS-3417	Nitrogen and Off Gas Bldg.	31'	A	No	No
3418A		33'	A	No	No
3418B		33'	A	No	No
3419A		33'	A	No	No
3419B		33'	A	No	No
3423		37'	A	No	No
	Reactor Feedwater System				
2B21-2SS3	Drywell	38'	I	No	No
2SS4		56'	I	No	No
3SS6		41'	I	No	No
3SS9		39'	I	No	No
3SS11		41'	I	No	No
3SS12		40'	I	No	No
3SS13		61'	I	No	No
5SS17		38'	I	No	No
5SS18		56'	I	No	No
6SS20		41'	I	No	No
6SS23		39'	I	No	No
6SS25		41'	I	No	No
6SS26		40'	I	No	No
6SS27		63'	I	No	No
1SS227		34'	I	No	No
1SS228		38'	I	No	No
2SS229		53'	I	No	No
2SS230		62'	I	No	No

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 Feedwater System (Continued)				
2B21-3SS231	Drywell (Cont'd)	40'	I	No	No
3SS232		36'	I	No	No
3SS233		40'	I	No	No
3SS234		48'	I	No	No
3SS235		63'	I	No	No
4SS236		34'	I	No	No
4SS237		38'	I	No	No
5SS238		53'	I	No	No
5SS239		61'	I	No	No
6SS240		41'	I	No	No
6SS241		36'	I	No	No
6SS242		39'	I	No	No
6SS243		48'	I	No	No
6SS244		61'	I	No	No
	Residual Heat Removal System				
2E11-90SS267	Drywell	79'	I	No	No
90SS268		86'	I	No	No
90SS271		86'	I	No	No
90SS274		93'	I	No	No
90SS275		93'	I	No	No
90SS277		96'	I	No	No
90SS278		96'	I	No	No
90SS280		101'	I	No	No
90SS281		93'	I	No	No
90SS282		101'	I	No	No
90SS283		93'	I	No	No
90SS284		101'	I	No	No
90SS285		100'	I	No	No
1SS302		35'	I	No	No
1SS303		34'	I	No	No
1SS305		201	I	No	No

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
	Residual Heat Removal System (Continued)				
2E11-1SS306	Drywell (Cont'd)	19'	I	No	No
84SS309		33'	I	No	No
84SS311		35'	I	No	No
84SS312		35'	I	No	No
87SS315		33'	I	No	No
87SS317		35'	I	No	No
87SS318		35'	I	No	No
90SS388		79'	I	No	No
90SS389		96'	I	No	No
90SS390		96'	I	No	No
90SS391		93'	I	No	No
90SS392		99	I	No	No
2E11-131SS255	Reactor Building	42'	A	No	No
131SS257		22'	A	No	No
132SS263		30'	A	No	No
132SS264		31'	A	No	No
128SS355		42'	A	No	No
128SS387		43'	A	No	No
132SS414		-14'	A	No	No
127SS426		43'	A	No	No
128SS427		21'	A	No	No
128SS428		28'	A	No	No
128SS429		39'	A	No	No
128SS430		31'	A	No	No
128SS431		23'	A	No	No
127SS376		59'	A	No	No

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
	Residual Heat Removal System (Continued)			
2E11-107SS573	Reactor Bldg (Cont'd) 15'	A	No	No
127SS433	14'	A	No	No
127SS435	15'	A	No	No
17SS3	-3'	A	No	No
17SS4	-1'	A	No	No
20SS19	-3'	A	No	No
95SS20	-3'	A	No	No
95SS22	9'	A	No	No
95SS23	9'	A	No	No
20SS28	-3'	A	No	No
37SS95	-3'	A	No	No
35SS360	-8'	A	No	No
35SS361	-16'	A	No	No
2SS397	-3'	A	No	No
5SS398	-4'	A	No	No
2SS399	-3'	A	No	No
5SS400	-3'	A	No	No
4SS401	-12'	A	No	No
5SS402	-12'	A	No	No
3SS403	-11'	A	No	No
6SS404	-12'	A	No	No
8SS405	-14'	A	No	No
6SS406	-14'	A	No	No
8SS407	-15'	A	No	No
12SS408	-14'	A	No	No
9SS411	-14'	A	No	No
109SS412	-14'	A	No	No
20SS415	-4'	A	No	No
20SS416	-4'	A	No	No
95SS417	0'	A	No	No
95SS418	-4'	A	No	No

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
	Residual Heat Removal System (Continued)			
2E11-45SS422	Reactor Bldg (Cont'd) -2'	A	No	No
56SS517	-5'	A	No	No
91SS575	53'	A	No	No
91SS500	57'	A	No	No
91SS499	69'	A	No	No
89SS491	67'	A	No	No
89SS489	67'	A	No	No
89SS487	67'	A	No	No
89SS480	67'	A	No	No
18SS469	53'	A	No	No
46SS216	28'	A	No	No
46SS217	31'	A	No	No
46SS218	30'	A	No	No
47SS223	33'	A	No	No
47SS224	36'	A	No	No
47SS225	36'	A	No	No
47SS227	39'	A	No	No
47SS228	39'	A	No	No
95SS233	20'	A	No	No
95SS234	24'	A	No	No
95SS235	31'	A	No	No
21SS296	39'	A	No	No
21SS297	39'	A	No	No
47SS326	42'	A	No	No
47SS328	42'	A	No	No
49SS330	42'	A	No	No
49SS331	42'	A	No	No
49SS333	42'	A	No	No
49SS334	43'	A	No	No

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
	Residual Heat Removal System (Continued)				
2E11-49SS336	Reactor Bldg (Cont'd)	40'	A	No	No
49SS359		42'	A	No	No
18SS470		43'	A	No	No
58SS514		14'	A	No	No
49SS515		37'	A	No	No
49SS516		37'	A	No	No
50SS597		20'	A	No	No
46SS7		12'	A	No	No
46SS9		12'	A	No	No
56SS13		5'	A	No	No
56SS15		4'	A	No	No
58SS32		3'	A	No	No
58SS33		-4'	A	No	No
58SS35		8'	A	No	No
58SS36		8'	A	No	No
18SS40		8'	A	No	No
18SS48		13'	A	No	No
68SS59		15'	A	No	No
21SS63		8'	A	No	No
21SS70		13'	A	No	No
21SS71		10'	A	No	No
61SS110		6'	A	No	No
53SS192		18'	A	No	No
53SS195		14'	A	No	No
53SS197		14'	A	No	No
53SS200		14'	A	No	No
50SS201		14'	A	No	No
89SS208		5'	A	No	No
60SS438		12'	A	No	No

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
Residual Heat Removal System (Continued)					
2E11-60SS440	Reactor Bldg. (Cont'd)	13'	A	No	No
65SS441		3'	A	No	No
65SS442		3'	A	No	No
60SS443		11'	A	No	No
73SS444		2'	A	No	No
21SS445		5'	A	No	No
68SS448		13'	A	No	No
75SS449		2'	A	No	No
61SS450		7'	A	No	No
60SS451		13'	A	No	No
60SS452		13'	A	No	No
60SS453		10'	A	No	No
60SS454		10'	A	No	No
89SS459		11'	A	No	No
89SS460		10'	A	No	No
89SS461		6'	A	No	No
53SS462		15'	A	No	No
53SS463		14'	A	No	No
53SS464		14'	A	No	No
53SS465		14'	A	No	No
53SS466		14'	A	No	No
53SS467		14'	A	No	No
50SS468		17'	A	No	No
56SS504		14'	A	No	No
56SS505		7'	A	No	No
56SS506		3'	A	No	No
56SS507		3'	A	No	No
56SS508		4'	A	No	No
46SS509		8'	A	No	No
46SS510		11'	A	No	No

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
Residual Heat Removal System (Continued)				
2E11-46SS511	Reactor Bldg. (Cont'd) 10'	A	No	No
46SS512	-1'	A	No	No
68SS577	8'	A	No	No
53SS596	14'	A	No	No
116SS143	-11'	A	No	No
113SS157	-11'	A	No	No
37SS184	9'	A	No	No
2SS396	5'	A	No	No
116SS409	-9'	A	No	No
113SS410	-9'	A	No	No
2SS413	0'	A	No	No
60SS423	-4'	A	No	No
60SS425	-2'	A	No	No
47SS323	42'	A	No	No
71SS393	9'	A	No	No
127SS434	37'	A	No	No
60SS437	13'	A	No	No
83SS446	10'	A	No	No
51SS546	32'	A	No	No
51SS547	28'	A	No	No
115SS549	31'	A	No	No
18SS47	12'	A	No	No
71SS174	-17'	A	No	No
71SS176	9'	A	No	No
54SS551	31'	A	No	No
54SS552	28'	A	No	No
98SS554	32'	A	No	No
58SS563	7'	A	No	No
58SS565	13'	A	No	No
58SS566	6'	A	No	No
69SS574	6'	A	No	No

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
	Service Water System				
2SW-133SS22	Reactor Building	-6'	A	No	No
110SS35		-5'	A	No	No
173SS72		14'	A	No	No
142SS164		8'	A	No	No
142SS165		8'	A	No	No
133SS176		14'	A	No	No
133SS177		-5'	A	No	No
142SS74		40'	A	No	No
142SS75		40'	A	No	No
140SS80		40'	A	No	No
140SS86		45'	A	No	No
153SS102		44'	A	No	No
153SS109		44'	A	No	No
173SS110		48'	A	No	No
153SS115		44'	A	No	No
103SS121		38'	A	No	No
140SS167		42'	A	No	No
173SS175		30'	A	No	No
142SS82		70'	A	No	No
173SS114		70'	A	No	No
103SS126		60'	A	No	No

3/4.9.2 INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.9.2 At least 2 source range monitor (SRM) channels* shall be OPERABLE and inserted to the normal operation level with:

a. A continuous visual indication in the control room,

b. One of the SRM detectors located in the quadrant where CORE ALTERATIONS are being performed and the other SRM detector located in an adjacent quadrant, and

c. The "shorting links" removed from the RPS circuitry prior to and during the time any control rod is withdrawn** and shutdown margin demonstrations.

APPLICABILITY: CONDITION 5.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and fully insert all insertable control rods. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.2 Each of the above required SRM channels shall be demonstrated OPERABLE by:

a. At least once per 12 hours;

1. Performance of a CHANNEL CHECK,

*The use of special movable detectors during CORE ALTERATIONS in place of the normal SRM nuclear detectors is permissible as long as these special detectors are connected to the normal SRM circuits.

**Not required for control rods removed per Specification 3.9.10.1 or 3.9.10.2.

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying the detectors are inserted to the normal operating level, and

3. During CORE ALTERATIONS, verifying that the detector of an OPERABLE SRM channel is located in the core quadrant where CORE ALTERATIONS are being performed and one is located in the adjacent quadrant.

b. Performance of a CHANNEL FUNCTIONAL TEST:

1. Within 24 hours prior to the start of CORE ALTERATIONS, and

2. At least once per 7 days.

c. Verifying at least once per 12 hours during CORE ALTERATIONS that the channel count rate is at least 3 cps.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 26 TO FACILITY LICENSE NO. DPR-71
AND
AMENDMENT NO. 50 TO FACILITY LICENSE NO. DPR-62

CAROLINA POWER & LIGHT COMPANY

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2

DOCKET NOS. 50-325 AND 50-324

A. Brunswick Steam Electric Plant Hydraulic Snubber Program

1.0 INTRODUCTION

By letter dated February 20, 1979, as supplemented January 14, 1980, Carolina Power and Light Company (the licensee) requested revisions to the Technical Specifications for the Brunswick Steam Electric Plant (BSEP) Unit Nos. 1 and 2 relative to Hydraulic Snubber Surveillance requirements.

2.0 DESCRIPTION

The proposed revisions were a result of the seismic reanalysis followup effort per I&E Bulletin 79-07. Based on the reanalysis, several support modifications were shown to be feasible. These include some hydraulic snubbers which could be replaced by rigid restraints, and some other hydraulic snubbers which could be removed without stressing the supported systems beyond the allowable stress limits. It is proposed that snubbers E21-6SS22 (A and B), E21-6SS21, E11-18SS447, E11-84SS308, E11-87SS314, and SW-103SS123 be removed from Table 3.7.5.1.

The proposal also includes the addition of several groups of hydraulic snubbers in Table 3.7.5-1. They are:

1. A group of BSEP 1 service water line snubbers (labeled SW)

2. One snubber of RHR line

3. A group of snubbers on instrument lines that tap off of the reactor vessel (labeled PS)

4. A group of snubbers in the off-gas building (labeled PS)

The omission of 1 and 2 from Table 3.7.5-1 was due to clerical errors while 3 and 4 were not included in that Table because of nomenclature inconsistencies.

A revision to Paragraph 4.7.5.2 of the Technical Specifications is also proposed to permit the establishment of snubber operability by test as well as by visual inspection as part of the surveillance requirement.

3.0 EVALUATION

The seismic reanalysis followup effort per I&E Bulletin 79-07 was made by CP&L for BSEP Units Nos. 1 and 2 and accepted by NRC on October 17, 1979. The revision of Table 3.7.5-1 based on the seismic reanalysis should be approved.

Some safety-related snubbers were omitted from the original Table 3.7.5-1 inadvertently. The correction of these errors is desirable. The addition of those snubbers into Table 3.7.5-1 should be approved.

When snubbers are found to be leaking fluid, their operability can be determined by a functional test in the AS FOUND condition. The proposed revisions to Paragraph 4.7.5.2 to permit the establishment of snubber operability by functional test should be approved but should be modified to be consistent with the current Standard Technical Specifications. The language of the STS is being clarified to describe exactly what is required to perform a functional test of a snubber in the AS FOUND condition. Until this language is agreed upon, we will not be in a position to approve this portion of the snubber Technical Specification change request.

4.0 CONCLUSIONS

The proposed revisions to Table 3.7.5-1 should be approved. The proposed revision to Paragraph 4.7.5.2 "--or functional test," will be deferred until the standard Technical Specification language is acceptably modified to describe the details of snubber functional testing.

B. Brunswick Steam Electric Plant Digital to Analog Instrumentation Change

1.0 INTRODUCTION

By letter dated November 19, 1979, CP&L filed an application for an amendment to the licenses for BSEP Units 1 and 2. The licensee proposed to replace certain existing safety and non-safety pressure and differential pressure switches with analog loops. The licensee provided proposed Technical Specification changes to update instrument designations, surveillance intervals, and battery test requirements. The licensee stated that the current schedule requires the equipment modifications to be performed during the 1980 refueling outages.

During subsequent telephone conversations, this schedule was revised such that only selected systems will be completed during the 1980 refueling out-
only selected systems will be completed during the 1980 refueling out-
ages, making it impractical and undesirable to change the Technical Specifications in advance. The licensee agreed to submit an administrative Technical Specification change at the conclusion of the 1980 refueling outages to request specific changes for those systems having the D/A modification completed. The licensee further agreed to

A revision to Paragraph 4.7.5.2 of the Technical Specifications is also proposed to permit the establishment of snubber operability by test as well as by visual inspection as part of the surveillance requirement.

3.0 EVALUATION

The seismic reanalysis followup effort per I&E Bulletin 79-07 was made by CP&L for BSEP Units Nos. 1 and 2 and accepted by NRC on October 17, 1979. The revision of Table 3.7.5-1 based on the seismic reanalysis should be approved.

Some safety-related snubbers were omitted from the original Table 3.7.5-1 inadvertently. The correction of these errors is desirable. The addition of those snubbers into Table 3.7.5-1 should be approved.

When snubbers are found to be leaking fluid, their operability can be determined by a functional test in the AS FOUND condition. The proposed revisions to Paragraph 4.7.5.2 to permit the establishment of snubber operability by functional test should be approved but should be modified to be consistent with the current Standard Technical Specifications. The language of the STS is being clarified to describe exactly what is required to perform a functional test of a snubber in the AS FOUND condition. Until this language is agreed upon, we will not be in a position to approve this portion of the snubber Technical Specification change request.

4.0 CONCLUSIONS

The proposed revisions to Table 3.7.5-1 should be approved. The proposed revision to Paragraph 4.7.5.2 "--or functional test," will be deferred until the standard Technical Specification language is acceptably modified to describe the details of snubber functional testing.

B. Brunswick Steam Electric Plant Digital to Analog Instrumentation Change

1.0 INTRODUCTION

By letter dated November 19, 1979, CP&L filed an application for an amendment to the licenses for BSEP Units 1 and 2. The licensee proposed to replace certain existing safety and non-safety pressure and differential pressure switches with analog loops. The licensee provided proposed Technical Specification changes to update instrument designations, surveillance intervals, and battery test requirements. The licensee stated that the current schedule requires the equipment modifications to be performed during the 1980 refueling outages.

During subsequent telephone conversations, this schedule was revised such that only selected systems will be completed during the 1980 refueling out-
only selected systems will be completed during the 1980 refueling out-
ages, making it impractical and undesirable to change the Technical Specifications in advance. The licensee agreed to submit an administrative Technical Specification change at the conclusion of the 1980 refueling outages to request specific changes for those systems having the D/A modification completed. The licensee further agreed to

operate the new analog instruments in accordance with the current Technical Specifications until such time as the administrative Technical Specification change is approved. This procedure has been discussed with the I&E Resident and will be in affect until all applicable instruments are modified.

2.0 DESCRIPTION

General Electric Licensing Topical Report NEDO-21617-A dated December 1978 describes the Analog Transmitter/Trip Unit System for Engineered Safeguards Sensor Trip Input. This report was reviewed and approved by the Staff on June 27, 1978. We required that operating plants be evaluated in accordance with specific divisional separation requirements for the plant. This necessitated the following application detail of the analog transmitter and trip unit hardware for BSEP:

Specific Instrument Loops

Information for each instrument loop that will be converted to the analog sensor system as identified below:

1. Variable name
2. Part number of device being deleted
3. System involved
4. The engineered safeguards division
5. Model number and vendor of the transmitter or RTD

Trip Unit Cabinet

Information for each trip unit cabinet as identified below:

1. Cabinet layout showing location areas of the power supplies, trip relays, and trip units.
2. Division to which the cabinet is assigned
3. Layout of each card file in the trip unit cabinet showing the trip variable for each card file slot.

Environmental Interface

The environment at each location where the retrofit hardware will be located must be compared to the maximum environment as stated in the topical report for the following factors:

1. Normal operation and post-accident temperature and humidity

2. Comparison of the floor seismic response spectra of the cabinet mounting location for the specific plant to seismic test envelope that the cabinet was tested too.

3. If the trip unit cabinets are not located in the preferred location, provide justification for the alternate selected location.

The following information must be supplied:

Specific Plant Interconnections

An interconnection diagram which shows the interconnections between the existing logic cabinets and instrument cabinets and the new trip cabinets.

Field Calibration Rack

The design and operational information on the "Field Calibration Rack"

3.0 EVALUATION

The staff has previously reviewed the use of this equipment and found that, provided certain interface requirements were satisfied, this equipment is acceptable. Our letter of approval, dated June 27, 1978, is a part of General Electric Topical Report NEDO-21617-A dated December 1978.

The licensee has submitted documentation demonstrating compliance with the interface requirements.

4.0 CONCLUSION

Based upon our review of the documentation, we conclude that the modifications proposed satisfy the constraints of our prior approval and are, therefore, acceptable.

C. Brunswick Steam Electric Plant Source Range Instrumentation

1.0 INTRODUCTION

By letter dated January 24, 1980, CP&L requested revisions to the Technical Specifications for the Brunswick Steam Electric Plant (BSEP) Unit Nos. 1 and 2 to delete a certain requirement for the source range monitor instrumentation.

2.0 DESCRIPTION

The proposed revision would delete the requirement for removing the SRM shorting links from the RPS circuitry during core alterations. With the SRM shorting links removed, any one SRM channel will initiate a SCRAM. With the SRM shorting links installed, a scram signal cannot be generated from the SRM instrumentation, although the rod control functions remain intact.

3.0 EVALUATION

Removal of the SRM shorting links enables the SRM scram with a setpoint of $5x10^5$cps (approximately $2.5x.0^{-4}$% power). In addition, the Reactor Protection Logic converts from a one-out-of-two taken twice to a one-out-of-four taken once logic. As a consequence of this requirement, experience at Brunswick has shown that spurious reactor scrams are common during refueling, with two undesirable side effects. First, these unnecessary scrams contribute to seal failures in the control rod drive mechanisms. Second, these scrams disturb the water clarity in the reactor vessel to the point of delaying refueling evoluations.

The plant was licensed with credit taken for the low level neutron monitoring system scrams which are based upon the Intermediate Range Monitoring (IRM) system. The IRM scram setpoint on low range is $1.6x10^{-3}$% power.

While no credit is given for the SRM scram in the licensing basis for the plant, this extra measure of conservatism is felt to be important during initial loading, normal fuel loadings with any control rods withdrawn, and during shutdown margin demonstrations. Accordingly, we cannot approve the deletion of this requirement in its entirety from the Technical Specifications. However, we concur that it is desirable to avoid spurious reactor scrams during refueling operations.

We have therefore concluded that the requirement can be modified such that the SRM shorting links be required to be removed during the refueling mode prior to withdrawing any control rod(s). Since the IRM scram is adequate to protect the core, modifying the requirement to enable the SRM scram prior to withdrawing any control rod(s) will not significantly affect the degree of conservatism, and should serve to minimize spurious scrams.

4.0 CONCLUSION

We have concluded that requiring the SRM scram to be enabled during refueling mode prior to and during the time any control rod(s) is(are) withdrawn, provides adequate assurance of reactor safety, and is not as likely to result in spurious scrams as the previous requirement.

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

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 the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Dated: March 14, 1980

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. 26 and 50 to Facility Operating License Nos. DPR-71 and DPR-62 issued to Carolina Power & Light Company (the licensee) which revised the 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 (1) correct the table of safety related hydraulic snubbers, (2) provide for systematic implementation of instrumentation modifications, and (3) eliminate the requirement for removing the SRM "shorting links" during core alterations with control rods withdrawn.

The applications for 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 amendment.

For further details with respect to this action, see (1) the application for amendment dated February 20, 1979, as supplemented January 14, 1980 and applications dated November 19, 1979 and January 24, 1980, (2) Amendment Nos. 26 and 50 to Licenses 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, NW., 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 Operating Reactors.

Dated at Bethesda, Maryland this 14th day of March 1980.

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors