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

CAROLINA POWER & LIGHT COMPANY, et al.

DOCKET NO. 50-325

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 149
License No. DPR-71

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

   A. The application for amendment filed by Carolina Power & Light Company (the licensee), dated February 29, 1988, superseded September 20, 1989, as supplemented December 5, 1989, February 15, August 9, and October 24, 1990, complies with the standards and requirements of the Atomic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-71 is hereby amended to read as follows:

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PDR ADOCK 05000325
P PDR

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 149 , are hereby incorporated in the license. Carolina Power & Light Company shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signed By:

Elinor G. Adensam, Director
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: December 5, 1990

OFC	:LA:PD21:DRPR	:PM:PD21:DRPR	: OGC	: OTSB	: SICB	:D:PD21:DRPR	:
NAME	:PAnderson	:NLe:bd:	: CBachmann	:JCalvo	:SNewberry	:EAdensam	:
DATE	:8 /16/90	:8 /17/90	:08/30/90	:~~06/14/90~~ 8/17/90	:06/28/90 8/17/90	:12/5 /90	:

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BSEP AMEND 67991/67992

(2) Technical Specifications

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FOR THE NUCLEAR REGULATORY COMMISSION

Elinor G. Adensam

Elinor G. Adensam, Director
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 5, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 149

FACILITY OPERATING LICENSE NO. DPR-71

DOCKET NO. 50-325

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised areas are indicated by marginal lines.

Remove Pages	Insert Pages
XI	XI
1-5	1-5
3/4 3-12	3/4 3-12
3/4 3-13	3/4 3-13
3/4 3-14	3/4 3-14
3/4 3-15	3/4 3-15
3/4 3-16	3/4 3-16
3/4 3-17a	3/4 3-17a
3/4 3-18	3/4 3-18
3/4 3-19	3/4 3-19
3/4 3-20	3/4 3-20
3/4 3-21	3/4 3-21
3/4 3-23	3/4 3-23
3/4 3-24	3/4 3-24
3/4 3-25	3/4 3-25
3/4 3-26	3/4 3-26
3/4 3-27	3/4 3-27
3/4 3-28	3/4 3-28
3/4 3-29	3/4 3-29
3/4 3-30	3/4 3-30
3/4 3-32	3/4 3-32
3/4 6-14	3/4 6-14
3/4 6-15	3/4 6-15
3/4 6-16	3/4 6-16
3/4 6-17	3/4 6-17
3/4 6-22	3/4 6-22
3/4 6-24	3/4 6-24
B3/4 6-5	B3/4 6-5
B3/4 6-6	B3/4 6-6

BASES

OPERABLE - OPERABILITY (Continued)

Implicit in this definition shall be t[illegible] [illegible]tion that all necessary attendant instrumentation, controls, no[illegible] [illegible] emergency electric power sources, cooling or seal water, lubric[illegible] [illegible] or other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its function(s) are also capable of performing their related support function(s).

OPERATIONAL CONDITION

An OPERATIONAL CONDITION [illegible]hall be any one inclusive combination of mode switch position and average reactor coolant temperature as indicated in Table 1.2.

PHYSICS TESTS

PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and re[illegible]d instrumentation and are 1) described in Section 14 of the Updated FSA[illegible] [illegible] authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.

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.

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.

TABLE 3.3.2-1

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL(a)	MINIMUM NUMBER OPERABLE CHANNELS PER TRIP SYSTEM(b)(c)	APPLICABLE OPERATIONAL CONDITION	ACTION
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level -				
1. Low, Level 1	2, 6	2	1, 2, 3	20
	8	2	1, 2, 3	27
2. Low, Level 3	1	2	1, 2, 3	20
b. Drywell Pressure - High	2, 6	2	1, 2, 3	20
c. Main Steam Line				
1. Radiation - High	1	2	1, 2, 3	21
2. Pressure - Low	1(j)	2	1	22
3. Flow - High	1(j)	2/line	1	22
d. Main Steam Line Tunnel Temperature - High	1(j)	2(d)	1, 2, 3	21
e. Condenser Vacuum - Low	1(j)	2	1, 2(e)	21
f. Turbine Building Area Temperature - High	1(j)	4(d)	1, 2, 3	21
g. Main Stack Radiation - High	(h)	1	1, 2, 3	28
h. Reactor Building Exhaust Radiation - High	6	1	1, 2, 3	20

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL(a)	MINIMUM NUMBER OPERABLE CHANNELS PER TRIP SYSTEM(b)(c)	APPLICABLE OPERATIONAL CONDITION	ACTION
2. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Building Exhaust Radiation - High	(1)	1	1, 2, 3, 5, and*	23
	6	1	1, 2, 3	20
b. Drywell Pressure - High	(1)	2	1, 2, 3	23
	2, 6	2	1, 2, 3	20
c. Reactor Vessel Water Level - Low, Level 2	(1)	2	1, 2, 3	23
	3	2	1, 2, 3	24
3. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. Δ Flow - High	3	1	1, 2, 3	24
b. Area Temperature - High	3	2	1, 2, 3	24
c. Area Ventilation Δ Temperature - High	3	2	1, 2, 3	24
d. SLCS Initiation	3 (f)	NA	1, 2, 3	24
e. Reactor Vessel Water Level - Low, Level 2	3	2	1, 2, 3	24
f. Δ Flow - High - Time Delay Relay	NA	1	1, 2, 3	24

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL(a)	MINIMUM NUMBER OPERABLE CHANNELS PER TRIP SYSTEM(b)(c)	APPLICABLE OPERATIONAL CONDITION	ACTION
4. CORE STANDBY COOLING SYSTEMS ISOLATION				
a. High Pressure Coolant Injection System Isolation				
1. HPCI Steam Line Flow - High	4	1	1, 2, 3	25
2. HPCI Steam Line Flow - High Time Delay Relay	NA	1	1, 2, 3	25
3. HPCI Steam Supply Pressure - Low	4	2	1, 2, 3	25
	7(k)	1	1, 2, 3	25
4. HPCI Steam Line Tunnel Temperature - High	4	2	1, 2, 3	25
5. Bus Power Monitor	NA(g)	1/bus	1, 2, 3	26
6. HPCI Turbine Exhaust Diaphragm Pressure - High	4	2	1, 2, 3	25
7. HPCI Steam Line Ambient Temperature - High	4	1	1, 2, 3	25
8. HPCI Steam Line Area Δ Temperature - High	4	1	1, 2, 3	25
9. HPCI Equipment Area Temperature - High	4	1	1, 2, 3	25
10. Drywell Pressure - High	7(k)	1	1, 2, 3	25

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL(a)	MINIMUM NUMBER OPERABLE CHANNELS PER TRIP SYSTEM(b)(c)	APPLICABLE OPERATIONAL CONDITION	ACTION
4. CORE STANDBY COOLING SYSTEMS ISOLATION (Continued)				
b. Reactor Core Isolation Cooling System Isolation				
1. RCIC Steam Line Flow - High	5	1	1, 2, 3	25
2. RCIC Steam Line Flow - High Time Delay Relay	NA	1	1, 2, 3	25
3. RCIC Steam Supply Pressure - Low	5	2	1, 2, 3	25
	9(k)	1	1, 2, 3	25
4. RCIC Steam Line Tunnel Temperature - High	5	2	1, 2, 3	25
5. Bus Power Monitor	NA(g)	1/bus	1, 2, 3	26
6. RCIC Turbine Exhaust Diaphragm Pressure - High	5	2	1, 2, 3	25
7. RCIC Steam Line Ambient Temperature - High	5	1	1, 2, 3	25
8. RCIC Steam Line Area Δ Temperature - High	5	1	1, 2, 3	25
9. RCIC Equipment Room Ambient Temperature - High	5	1	1, 2, 3	25
10. RCIC Equipment Room Δ Temperature - High	5	1	1, 2, 3	25
11. RCIC Steam Line Tunnel Temperature - High Time Delay Relay	NA	1	1, 2, 3	25
12. Drywell Pressure - High	9(k)	1	1, 2, 3	25

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL(a)	MINIMUM NUMBER OPERABLE CHANNELS PER TRIP SYSTEM(b)(c)	APPLICABLE OPERATIONAL CONDITION	ACTION
5. SHUTDOWN COOLING SYSTEM ISOLATION				
a. Reactor Vessel Water Level - Low, Level 1	2, 6	2	1, 2, 3	20
	8	2	1, 2, 3	27
b. Reactor Steam Dome Pressure - High	8(i)	1	1, 2, 3	27

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

NOTES

* When handling irradiated fuel in the secondary containment

(a) See Specification 3.6.3.1, Table 3.6.3-1 for valves in each valve group.

(b) 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 other OPERABLE channel in the same trip system is monitoring that parameter.

(c) With only one channel per trip system, an inoperable channel need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, the inoperable channel shall be restored to OPERABLE status within 2 hours or the ACTION required by Table 3.3.2-1 for that Trip Function shall be taken.

(d) A channel is OPERABLE if 2 of 4 instruments in that channel are OPERABLE.

(e) With reactor steam pressure $\geq$ 500 psig.

(f) Closes only RWCU outlet isolation valve.

(g) Alarm only.

(h) Isolates containment purge and vent valves.

(i) Does not isolate E11-F015A,B.

(j) Does not isolate B32-F019 or B32-F020.

(k) Valve isolation depends upon low steam supply pressure coincident with high drywell pressure.

(l) Secondary containment isolation dampers as listed in Table 3.6.5.2-1.

TABLE 3.3.2-2

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. PRIMARY CONTAINMENT ISOLATION		
a. Reactor Vessel Water Level -		
1. Low, Level 1	$\geq$ + 162.5 inches[a]	$\geq$ + 162.5 inches[a]
2. Low, Level 3	$\geq$ + 2.5 inches[a]	$\geq$ + 2.5 inches[a]
b. Drywell Pressure - High	$\leq$ 2 psig	$\leq$ 2 psig
c. Main Steam Line		
1. Radiation - High	$\leq$ 3 x full power background[c]	$\leq$ 3.5 x full power background[c]
2. Pressure - Low	$\geq$ 825 psig	$\geq$ 825 psig
3. Flow - High	$\leq$ 140% of rated flow	$\leq$ 140% of rated flow
d. Main Steam Line Tunnel Temperature - High	$\leq$ 200°F	$\leq$ 200°F
e. Condenser Vacuum - Low	$\geq$ 7 inches Hg vacuum	$\geq$ 7 inches Hg vacuum
f. Turbine Building Area Temperature - High	$\leq$ 200°F	$\leq$ 200°F
g. Main Stack Radiation - High	(b)	(b)
h. Reactor Building Exhaust Radiation - High	$\leq$ 11 mr/hr	$\leq$ 11 mr/hr

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
2. SECONDARY CONTAINMENT ISOLATION		
a. Reactor Building Exhaust Radiation - High	$\leq$ 11 mr/hr	$\leq$ 11 mr/hr
b. Drywell Pressure - High	$\leq$ 2 psig	$\leq$ 2 psig
c. Reactor Vessel Water Level - Low, Level 2	$\geq$ + 112 inches[a]	$\geq$ + 112 inches[a]
3. REACTOR WATER CLEANUP SYSTEM ISOLATION		
a. Δ Flow - High	$\leq$ 53 gal/min	$\leq$ 53 gal/min
b. Area Temperature - High	$\leq$ 150°F	$\leq$ 150°F
c. Area Ventilation Δ Temperature - High	$\leq$ 50°F	$\leq$ 50°F
d. SLCS Initiation	NA	NA
e. Reactor Vessel Water Level - Low, Level 2	$\geq$ + 112 inches[a]	$\geq$ + 112 inches[a]
f. Δ Flow - High - Time Delay Relay	$\leq$ 45 seconds	$\leq$ 45 seconds

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
4. CORE STANDBY COOLING SYSTEMS ISOLATION		
a. High Pressure Coolant Injection System Isolation		
1. HPCI Steam Line Flow - High	$\leq$ 300% of rated flow	$\leq$ 300% of rated flow
2. HPCI Steam Line Flow - High Time Delay Relay	$3 \leq t \leq 7$ seconds	$3 \leq t \leq 12$ seconds
3. HPCI Steam Supply Pressure - Low	$\geq$ 100 psig	$\geq$ 100 psig
4. HPCI Steam Line Tunnel Temperature - High	$\leq$ 200°F	$\leq$ 200°F
5. Bus Power Monitor	NA	NA
6. HPCI Turbine Exhaust Diaphragm Pressure - High	$\leq$ 10 psig	$\leq$ 10 psig
7. HPCI Steam Line Ambient Temperature - High	$\leq$ 200°F	$\leq$ 200°F
8. HPCI Steam Line Area Δ Temperature - High	$\leq$ 50°F	$\leq$ 50°F
9. HPCI Equipment Area Temperature - High	$\leq$ 175°F	$\leq$ 175°F
10. Drywell Pressure - High	$\leq$ 2 psig	$\leq$ 2 psig

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
4. CORE STANDBY COOLING SYSTEMS ISOLATION (Continued)		
b. Reactor Core Isolation Cooling System Isolation		
1. RCIC Steam Line Flow - High	≤ 300% of rated flow	≤ 300% of rated flow
2. RCIC Steam Line Flow - High Time Delay Relay	3 ≤ t ≤ 7 seconds	3 ≤ t ≤ 12 seconds
3. RCIC Steam Supply Pressure - Low	≥ 50 psig	≥ 50 psig
4. RCIC Steam Line Tunnel Temperature - High	≤ 175°F	≤ 175°F
5. Bus Power Monitor	NA	NA
6. RCIC Turbine Exhaust Diaphragm Pressure - High	≤ 10 psig	≤ 10 psig
7. RCIC Steam Line Ambient Temperature - High	≤ 200°F	≤ 200°F
8. RCIC Steam Line Area Δ Temperature - High	≤ 50°F	≤ 50°F
9. RCIC Equipment Room Ambient Temperature - High	≤ 175°F	≤ 175°F
10. RCIC Equipment Room Δ Temperature - High	≤ 50°F	≤ 50°F
11. RCIC Steam Line Tunnel Temperature - High Time Delay Relay	≤ 30 minutes	≤ 30 minutes
12. Drywell Pressure - High	≤ 2 psig	≤ 2 psig

TABLE 3.3.2-3

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Seconds)$^{(a)(e)}$
1. PRIMARY CONTAINMENT ISOLATION	
a. Reactor Vessel Water Level -	
1. Low, Level 1	$\leq$13
2. Low, Level 3	$\leq$1.0$^{(d)}$ $\leq$13$^{(f)}$
b. Drywell Pressure - High	$\leq$13
c. Main Steam Line	
1. Radiation - High$^{(b)}$	$\leq$1.0$^{(d)}$ $\leq$ 13$^{(f)}$
2. Pressure - Low	$\leq$13
3. Flow - High	$\leq$0.5$^{(d)}$ $\leq$13$^{(f)}$
d. Main Steam Line Tunnel Temperature - High	$\leq$13
e. Condenser Vacuum - Low	$\leq$13
f. Turbine Building Area Temperature - High	NA
g. Main Stack Radiation - High$^{(b)}$	$\leq$ 1.0$^{(d)}$
h. Reactor Building Exhaust Radiation - High$^{(b)}$	NA
2. SECONDARY CONTAINMENT ISOLATION	
a. Reactor Building Exhaust Radiation - High$^{(b)}$	$\leq$13
b. Drywell Pressure - High	$\leq$13
c. Reactor Vessel Water Level - Low, Level 2	$\leq$13
3. REACTOR WATER CLEANUP SYSTEM ISOLATION	
a. Δ Flow - High	$\leq$45$^{(c)}$
b. Area Temperature - High	$\leq$13
c. Area Ventilation Δ Temperature - High	$\leq$13
d. SLCS Initiation	NA
e. Reactor Vessel Water Level - Low, Level 2	$\leq$13
f. Δ Flow - High - Time Delay Relay	NA

TABLE 3.3.2-3 (Continued)

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Seconds)(a)(e)
4. CORE STANDBY COOLING SYSTEMS ISOLATION	
a. High Pressure Coolant Injection System Isolation	
1. HPCI Steam Line Flow - High	≤ 13 (c)
2. HPCI Steam Line Flow - High Time Delay Relay	NA
3. HPCI Steam Supply Pressure - Low	≤ 13
4. HPCI Steam Line Tunnel Temperature - High	≤ 13
5. Bus Power Monitor	NA
6. HPCI Turbine Exhaust Diaphragm Pressure - High	NA
7. HPCI Steam Line Ambient Temperature - High	NA
8. HPCI Steam Line Area Δ Temperature - High	NA
9. HPCI Equipment Area Temperature - High	NA
10. Drywell Pressure - High	NA
b. Reactor Core Isolation Cooling System Isolation	
1. RCIC Steam Line Flow - High	≤ 13 (c)
2. RCIC Steam Line Flow - High Time Delay Relay	NA
3. RCIC Steam Supply Pressure - Low	NA
4. RCIC Steam Line Tunnel Temperature - High	NA
5. Bus Power Monitor	NA
6. RCIC Turbine Exhaust Diaphram Pressure - High	NA
7. RCIC Steam Line Ambient Temperature - High	NA
8. RCIC Steam Line Area Δ Temperature - High	NA
9. RCIC Equipment Room Ambient Temperature - High	NA
10. RCIC Equipment Room Δ Temperature - High	NA
11. RCIC Steam Line Tunnel Temperature - High Time Delay Relay	NA
12. Drywell Pressure - High	NA

TABLE 3.3.2-3 (Continued)

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Seconds)$^{(a)(e)}$
5. SHUTDOWN COOLING SYSTEM ISOLATION	
a. Reactor Vessel Water Level - Low, Level 1	NA
b. Reactor Steam Dome Pressure - High	NA

TABLE 3.3.2-3 (Continued)

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

NOTES

(a) The isolation system instrumentation response time shall be measured and recorded as a part of the ISOLATION SYSTEM RESPONSE TIME. Isolation system instrumentation response time specified includes any delay for diesel generator starting assumed in the accident analysis.

(b) Radiation monitors are exempt from response time testing. Response time shall be measured from detector output or the input of the first electronic component in the channel.

(c) Includes time delay added by the time delay relay.

(d) Isolation actuation instrumentation response time for MSIVs only. No diesel generator delays assumed.

(e) Isolation system instrumentation response time specified for the Trip Function actuating each valve group/damper shall be added to the isolation time for valves in each valve group shown in Table 3.6.3-1 and secondary containment isolation dampers shown in Table 3.6.5.2-1 to obtain ISOLATION SYSTEM RESPONSE TIME for each valve/damper.

(f) Isolation system instrumentation response time for associated valves except MSIVs.

TABLE 4.3.2-1

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level -				
1. Low, Level 1				
Transmitter:	NA(a)	NA	R(b)	1, 2, 3
Trip Logic:	D	M	M	1, 2, 3
2. Low, Level 3				
Transmitter:	NA(a)	NA	R(b)	1, 2, 3
Trip Logic:	D	M	M	1, 2, 3
b. Drywell Pressure - High				
Transmitter:	NA(a)	NA	R(b)	1, 2, 3
Trip Logic:	D	M	M	1, 2, 3
c. Main Steam Line				
1. Radiation - High	D	W	R(d)	1, 2, 3
2. Pressure - Low				
Transmitter:	NA(a)	NA	R(b)	1
Trip Logic:	D	M	M	1
3. Flow - High				
Transmitter:	NA(a)	NA	R(b)	1
Trip Logic:	D	M	M	1
d. Main Steam Line Tunnel Temperature - High	NA	M	R	1, 2, 3
e. Condenser Vacuum - Low				
Transmitter:	NA(a)	NA	R(b)	1, 2(e)
Trip Logic:	D	M	M	1, 2(e)
f. Turbine Building Area Temperature - High	NA	M	R	1, 2, 3
g. Main Stack Radiation - High	NA	Q	R	1, 2, 3
h. Reactor Building Exhaust Radiation - High	D	M	R	1, 2, 3

TABLE 4.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
2. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Building Exhaust Radiation - High	D	M	R	1,2,3,5, and(f)
b. Drywell Pressure - High				
Transmitter:	NA(a)	NA	R(b)	1, 2, 3
Trip Logic	D	M	M	1, 2, 3
c. Reactor Vessel Water Level - Low, Level 2				
Transmitter:	NA(a)	NA	R(b)	1, 2, 3
Trip Logic:	D	M	M	1, 2, 3
3. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. Δ Flow - High	D	M	R	1, 2, 3
b. Area Temperature - High	NA	M	R	1, 2, 3
c. Area Ventilation Δ Temperature - High	NA	M	R	1, 2, 3
d. SLCS Initiation	NA	R	NA	1, 2, 3
e. Reactor Vessel Water Level - Low, Level 2				
Transmitter:	NA(a)	NA	R(b)	1, 2, 3
Trip Logic:	D	M	M	1, 2, 3
f. Δ Flow - High - Time Delay Relay	NA	M	R	1, 2, 3

TABLE 4.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
4. CORE STANDBY COOLING SYSTEMS ISOLATION				
a. High Pressure Coolant Injection System Isolation				
1. HPCI Steam Line Flow - High				
Transmitter:	NA(a)	NA	R(b)	1, 2, 3
Trip Logic:	D	M	M	1, 2, 3
2. HPCI Steam Line Flow - High Time Delay Relay	NA	R	R	1, 2, 3
3. HPCI Steam Supply Pressure - Low	NA	M	R	1, 2, 3
4. HPCI Steam Line Tunnel Temperature - High	NA	M	Q	1, 2, 3
5. Bus Power Monitor	NA	R	NA	1, 2, 3
6. HPCI Turbine Exhaust Diaphragm Pressure - High	NA	M	Q	1, 2, 3
7. HPCI Steam Line Ambient Temperature - High	NA	M	R	1, 2, 3
8. HPCI Steam Line Area Δ Temperature - High	NA	M	R	1, 2, 3
9. HPCI Equipment Area Temperature - High	NA	M	Q	1, 2, 3
10. Drywell Pressure - High				
Transmitter:	NA(a)	NA	R(b)	1, 2, 3
Trip Logic:	D	M	M	1, 2, 3

TABLE 4.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
4. CORE STANDBY COOLING SYSTEMS ISOLATION (Continued)				
b. Reactor Core Isolation Cooling System Isolation				
1. RCIC Steam Line Flow - High				
Transmitter:	NA[(a)]	NA	R[(b)]	1, 2, 3
Trip Logic:	D	M	M	1, 2, 3
2. RCIC Steam Line Flow - High Time Delay Relay	NA	R	R	1, 2, 3
3. RCIC Steam Supply Pressure - Low	NA	M	Q	1, 2, 3
4. RCIC Steam Line Tunnel Temperature - High	NA	M	R	1, 2, 3
5. Bus Power Monitor	NA	R	NA	1, 2, 3
6. RCIC Turbine Exhaust Diaphragm Pressure - High	NA	M	R	1, 2, 3
7. RCIC Steam Line Ambient Temperature - High	NA	M	R	1, 2, 3
8. RCIC Steam Line Area Δ Temperature - High	NA	M	R	1, 2, 3
9. RCIC Equipment Room Ambient Temperature - High	NA	M	Q	1, 2, 3
10. RCIC Equipment Room Δ Temperature - High	NA	M	Q	1, 2, 3
11. RCIC Steam Line Tunnel Temperature - High Time Delay Relay	NA	M	R	1, 2, 3
12. Drywell Pressure - High				
Transmitter:	NA[(a)]	NA	R[(b)]	1, 2, 3
Trip Logic:	D	M	M	1, 2, 3

TABLE 4.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

NOTES

(a) The transmitter channel check is satisfied by the trip unit channel check. A separate transmitter check is not required.

(b) Transmitters are exempted from the monthly channel calibration.

(c) If not performed within the previous 31 days.

(d) Testing shall verify that the mechanical vacuum pump trips and the mechanical vacuum pump line valve closes.

(e) When reactor steam pressure ≥ 500 psig.

(f) When handling irradiated fuel in the secondary containment.

TABLE 3.6.3-1

PRIMARY CONTAINMENT ISOLATION VALVES

Table 3.6.3-1 has been deleted.

Refer to Plant Procedure RCI-02.6.

Pages 3/4 6-15 through 3/4 6-17 have been deleted.

(Next page is 3/4 6-18)

CONTAINMENT SYSTEMS

SECONDARY CONTAINMENT AUTOMATIC ISOLATION DAMPERS

LIMITING CONDITION FOR OPERATION

3.6.5.2 The secondary containment automatic isolation dampers shown in Table 3.6.5.2-1 shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 5, and *.

ACTION:

With one or more of the secondary containment isolation dampers specified in Table 3.6.5.2-1 inoperable, operation may continue and the provisions of Specification 3.0.4 are not applicable, provided that at least one isolation damper is maintained OPERABLE in each affected penetration that is open, and:

a. The inoperable damper is restored to OPERABLE status within 8 hours, or

b. The affected penetration is isolated by use of a closed damper within 8 hours, or

c. SECONDARY CONTAINMENT INTEGRITY is demonstrated within 8 hours and the damper is restored to OPERABLE status within 7 days.

Otherwise, in OPERATIONAL CONDITIONS 1, 2, or 3, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

Otherwise, in OPERATIONAL CONDITION 5 or *, suspend irradiated fuel handling in the secondary containment, CORE ALTERATIONS, or activities that could reduce the SHUTDOWN MARGIN. The provisions of Specification 3.0.3 are not applicable.

*When irradiated fuel is being handled in the secondary containment.

TABLE 3.6.5.2-1

SECONDARY CONTAINMENT AUTOMATIC ISOLATION DAMPERS

Table 3.6.5.2-1 has been deleted.

Refer to Plant Procedure RCI-02.6.

3/4.6.3 PRIMARY CONTAINMENT ISOLATION VALVES (Continued)

A list of automatic closing primary containment isolation valves and their associated closure times shall be available at the plant in accordance with Section 50.71(c) of 10 CFR Part 50. The addition and deletion of primary containment isolation valves shall be made in accordance with Section 50.59 of 10 CFR Part 50.

3/4.6.4 VACUUM RELIEF

Vacuum relief breakers are provided to equalize the pressure between the drywell and suppression pool and the suppression pool and reactor building. This system will maintain the structural integrity of the containment under conditions of large differential pressures.

The vacuum breakers between the drywell and the suppression pool must not be inoperable in the open position since this would allow bypassing of the suppression pool in case of an accident. There are an adequate number of valves to provide some redundancy so that operation may continue with no more than 2 vacuum breakers inoperable and secured in the closed position.

Each set of vacuum relief valves between the suppression chamber and reactor building provides 100% relief, which may by required in the unlikely event that negative pressures develop in the primary containment.

The Nitrogen Backup System provides backup motive power for these suppression pool-reactor building vacuum breakers on a loss of instrument air. The normal non-interruptible instrument air system for these vacuum breakers is designed as a Seismic Class I system supplied by air compressors powered from the emergency buses. The Nitrogen System serves as a backup to that air system and thus the loss of the Nitrogen System, or portions thereof, does not make the vacuum breakers inoperable. The design allows for the out of service times in Actions b and c. The Nitrogen Backup System is added to the Suppression Pool-Reactor Building Vacuum Breaker specification to satisfy NRC concerns relative to 10 CFR 50.44(c)(3) as addressed in the Brunswick Safety Evaluation Report dated October 30, 1986 concerning Generic Letter 84-09. Pressurization to 1130 psig assures sufficient system capacity to provide 24 hours of operation with design valve actuation and system leakage.

3/4.6.5 SECONDARY CONTAINMENT

Secondary containment is designed to minimize any ground level release of radioactive material which may result from an accident. The reactor building provides secondary containment during normal operation when the drywell is sealed and in service. When the reactor is shut down, or during refueling, the drywell may be open and the reactor building then becomes the primary containment.

3/4.6.5 SECONDARY CONTAINMENT (Continued)

Establishing and maintaining a vacuum in the building with the standby gas treatment system, once per 18 months, along with the surveillance of the valves, is adequate to ensure that there are no violations of the integrity of the secondary containment.

A list of secondary containment automatic isolation dampers shall be available at the plant in accordance with Section 50.71(c) of 10 CFR Part 50. The addition and deletion of secondary containment automatic isolation dampers shall be made in accordance with Section 50.59 of 10 CFR Part 50.

3/4.6.6 CONTAINMENT ATMOSPHERE CONTROL

The OPERABILITY of the containment iodine filter trains ensures that sufficient iodine removal capability will be available in the event of a LOCA. The reduction in containment iodine inventory reduces the resulting site boundary radiation doses associated with containment leakage. The operation of this system and resultant iodine removal capacity are consistent with the assumptions used in the LOCA analyses.

The OPERABILITY of the equipment and systems required for the detection and control of hydrogen gas ensures that this equipment will be available to maintain the hydrogen concentration within containment below its flammable limit during post-LOCA conditions. The containment inerting system is capable of controlling the expected hydrogen generation associated with 1) zirconium-water reactions, 2) radiolytic decomposition of water, and 3) corrosion of metals within containment. The hydrogen control system is consistent with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a LOCA."

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

CAROLINA POWER & LIGHT COMPANY, et al.

DOCKET NO. 50-324

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 179
License No. DPR-62

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

   A. The application for amendment filed by Carolina Power & Light Company (the licensee), dated February 29, 1988, superseded September 20, 1989, as supplemented December 5, 1989, February 15, August 9, and October 24, 1990, 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. 179 , are hereby incorporated in the license. Carolina Power & Light Company shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signed By:

Elinor G. Adensam, Director
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: December 5, 1990

OFC	:LA:PD21:DRPR	:PM:PD21:DRPR	:OGC	:OTSB	:SICB	:D:PD21:DRPR	:
NAME	:PAnderson	:NLe:bd:	: CBachmann	:JCalvo	: SNewberry	:EAdensam	:
DATE	:8 /15/90	:8 /17/90	:08/30/90	:~~06/14/90~~ 08/17/90	:~~06/28/90~~ 8/17/90	:12/5/90	:

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BSEP AMEND 67991/67992

ATTACHMENT TO LICENSE AMENDMENT NO. 179

FACILITY OPERATING LICENSE NO. DPR-62

DOCKET NO. 50-324

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised areas are indicated by marginal lines.

Remove Pages	Insert Pages
XI	XI
1-5	1-5
3/4 3-12	3/4 3-12
3/4 3-13	3/4 3-13
3/4 3-14	3/4 3-14
3/4 3-15	3/4 3-15
3/4 3-16	3/4 3-16
3/4 3-17a	3/4 3-17a
3/4 3-18	3/4 3-18
3/4 3-19	3/4 3-19
3/4 3-20	3/4 3-20
3/4 3-21	3/4 3-21
3/4 3-23	3/4 3-23
3/4 3-24	3/4 3-24
3/4 3-25	3/4 3-25
3/4 3-26	3/4 3-26
3/4 3-27	3/4 3-27
3/4 3-28	3/4 3-28
3/4 3-29	3/4 3-29
3/4 3-30	3/4 3-30
3/4 3-32	3/4 3-32
3/4 6-14	3/4 6-14
3/4 6-15	3/4 6-15
3/4 6-16	3/4 6-16
3/4 6-17	3/4 6-17
3/4 6-22	3/4 6-22
3/4 6-24	3/4 6-24
B 3/4 6-5	B 3/4 6-5
B 3/4 6-6	B 3/4 6-6

BASES

OFFSITE DOSE CALCULATION MANUAL (ODCM)

The OFFSITE DOSE CALCULATIONAL MANUAL (ODCM) is a manual which contains the current methodology and parameters to be used to calculate offsite doses resulting from the release of radioactive gaseous and liquid effluents; the methodology to calculate gaseous and liquid effluent monitoring instrumentation alarm/trip setpoints; and, the requirements of the environmental radiological monitoring program.

OPERABLE - OPERABILITY

A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electric power sources, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its function(s) are also capable of performing their related support function(s).

OPERATIONAL CONDITION

An OPERATIONAL CONDITION shall be any one inclusive combination of mode switch position and average reactor coolant temperature as indicated in Table 1.2.

PHYSICS TESTS

PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and are 1) described in Section 14 of the Updated FSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

PRESSURE BOUNDARY LEAKAGE shall be leakage through a non-isolable 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.

TABLE 3.3.2-1

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL(a)	MINIMUM NUMBER OPERABLE CHANNELS PER TRIP SYSTEM(b)(c)	APPLICABLE OPERATIONAL CONDITION	ACTION
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level -				
1. Low, Level 1	2, 6	2	1, 2, 3	20
	8	2	1, 2, 3	27
2. Low, Level 3	1	2	1, 2, 3	20
b. Drywell Pressure - High	2, 6	2	1, 2, 3	20
c. Main Steam Line				
1. Radiation - High	1	2	1, 2, 3	21
2. Pressure - Low	$1^{(j)}$	2	1	22
3. Flow - High	$1^{(j)}$	2/line	1	22
4. Fl	$1^{(j)}$	2	2, 3	21
d. Main St Tempe	$1^{(j)}$	$2^{(d)}$	1, 2, 3	21
e. Condens	$1^{(j)}$	2	1, $2^{(e)}$	21
f. Turbine Tempe	$1^{(j)}$	$4^{(d)}$	1, 2, 3	21
g. Main Stack Radiation - High	(h)	1	1, 2, 3	28
h. Reactor Building Exhaust Radiation - High	6	1	1, 2, 3	20

~~46~~, ~~98~~, 99, ~~142~~, ~~162~~, 179

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TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL(a)	MINIMUM NUMBER OPERABLE CHANNELS PER TRIP SYSTEM(b)(c)	APPLICABLE OPERATIONAL CONDITION	ACTION
2. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Building Exhaust Radiation - High	(1)	1	1, 2, 3, 5, and *	23
	6	1	1, 2, 3	20
b. Drywell Pressure - High	(1)	2	1, 2, 3	23
	2, 6	2	1, 2, 3	20
c. Reactor Vessel Water Level - Low, Level 2	(1)	2	1, 2, 3	23
	3	2	1, 2, 3	24
3. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. Δ Flow - High	3	1	1, 2, 3	24
b. Area Temperature - High	3	2	1, 2, 3	24
c. Area Ventilation Δ Temperature - High	3	2	1, 2, 3	24
d. SLCS Initiation	3 (f)	NA	1, 2, 3	24
e. Reactor Vessel Water Level - Low, Level 2	3	2	1, 2, 3	24
f. Δ Flow - High - Time Delay Relay	NA	1	1, 2, 3	24

Amendment No. 46, 78, 97, 142, 146, 160, 179

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL(a)	MINIMUM NUMBER OPERABLE CHANNELS PER TRIP SYSTEM(b)(c)	APPLICABLE OPERATIONAL CONDITION	ACTION
4. CORE STANDBY COOLING SYSTEMS ISOLATION				
a. High Pressure Coolant Injection System Isolation				
1. HPCI Steam Line Flow - High	4	1	1, 2, 3	25
2. HPCI Steam Line Flow - High Time Delay Relay	NA	1	1, 2, 3	25
3. HPCI Steam Supply Pressure - Low	4 7(k)	2 1	1, 2, 3 1, 2, 3	25 25
4. HPCI Steam Line Tunnel Temperature - High	4	2	1, 2, 3	25
5. Bus Power Monitor	NA(g)	1/bus	1, 2, 3	26
6. HPCI Turbine Exhaust Diaphragm Pressure - High	4	2	1, 2, 3	25
7. HPCI Steam Line Ambient Temperature - High	4	2	1, 2, 3	25
8. HPCI Steam Line Area Δ Temperature - High	4	1	1, 2, 3	25
9. HPCI Equipment Area Temperature - High	4	1	1, 2, 3	25
10. Drywell Pressure - High	7(k)	1	1, 2, 3	25

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL(a)	MINIMUM NUMBER OPERABLE CHANNELS PER TRIP SYSTEM(b)(c)	APPLICABLE OPERATIONAL CONDITION	ACTION
4. CORE STANDBY COOLING SYSTEMS ISOLATION (Continued)				
b. Reactor Core Isolation Cooling System Isolation				
1. RCIC Steam Line Flow - High	5	1	1, 2, 3	25
2. RCIC Steam Line Flow - High Time Delay Relay	NA	1	1, 2, 3	25
3. RCIC Steam Supply Pressure - Low	5 9[k]	2 1	1, 2, 3 1, 2, 3	25 25
4. RCIC Steam Line Tunnel Temperature - High	5	2	1, 2, 3	25
5. Bus Power Monitor	NA [g]	1/bus	1, 2, 3	26
6. RCIC Turbine Exhaust Diaphragm Pressure - High	5	2	1, 2, 3	25
7. RCIC St[illegible]ine Ambient Temper[illegible]e - High	5	1	1, 2, 3	25
8. RCIC Steam Line Area Δ Temperature - High	5	1	1, 2, 3	25
9. RCIC Equipment Room Ambient Temperature - High	5	1	1, 2, 3	25
10. RCIC Equipment Room Δ Temperature - High	5	1	1, 2, 3	25
11. RCIC Steam Line Tunnel Temperature - High Time Delay Relay	NA	1	1, 2, 3	25
12. Drywell Pressure - High	9[k]	1	1, 2, 3	25

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL(a)	MINIMUM NUMBER OPERABLE CHANNELS PER TRIP SYSTEM(b)(c)	APPLICABLE OPERATIONAL CONDITION	ACTION
5. SHUTDOWN COOLING SYSTEM ISOLATION				
a. Reactor Vessel Water Level - Low, Level 1	2, 6	2	1, 2, 3	20
	8	2	1, 2, 3	27
b. Reactor Steam Dome Pressure - High	8(i)	1	1, 2, 3	27

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

NOTES

* When handling irradiated fuel in the secondary containment.

(a) See Specification 3.6.3.1, Table 3.6.3-1 for valves in each valve group.

(b) 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 other OPERABLE channel in the same trip system is monitoring that parameter.

(c) With only one channel per trip system, an inoperable channel need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, the inoperable channel shall be restored to OPERABLE status within 2 hours or the ACTION required by Table 3.3.2-1 for that Trip Function shall be taken.

(d) A channel is OPERABLE if 2 of 4 instruments in that channel are OPERABLE.

(e) With reactor steam pressure $\geq$ 500 psig.

(f) Closes only RWCU outlet isolation valve.

(g) Alarm only.

(h) Isolates containment purge and vent valves.

(i) Does not isolate E11-F015A,B.

(j) Does not isolate B32-F019 or B32-F020.

(k) Valve isolation depends upon low steam supply pressure coincident with high drywell pressure.

(l) Secondary containment isolation dampers as listed in Table 3.6.5.2-1.

TABLE 3.3.2-2

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. PRIMARY CONTAINMENT ISOLATION		
a. Reactor Vessel Water Level -		
1. Low, Level 1	$\geq$ + 162.5 inches[(a)]	$\geq$ + 162.5 inches[(a)]
2. Low, Level 3	$\geq$ + 2.5 inches[(a)]	$\geq$ + 2.5 inches[(a)]
b. Drywell Pressure - High	$\leq$ 2 psig	$\leq$ 2 psig
c. Main Steam Line		
1. Radiation - High	$\leq$ 3 x full power background[(c)]	$\leq$ 3.5 x full power background[(c)]
2. Pressure - Low	$\geq$ 825 psig	$\geq$ 825 psig
3. Flow - High	$\leq$ 140% of rated flow	$\leq$ 140% of rated flow
4. Flow - High	$\leq$ 40% of rated flow	$\leq$ 40% of rated flow
d. Main Steam Line Tunnel Temperature - High	$\leq$ 200°F	$\leq$ 200°F
e. Condenser Vacuum - Low	$\geq$ 7 inches Hg vacuum	$\geq$ 7 inches Hg vacuum
f. Turbine Building Area Temperature - High	$\leq$ 200°F	$\leq$ 200°F
g. Main Stack Radiation - High	(b)	(b)
h. Reactor Building Exhaust Radiation - High	$\leq$ 11 mr/hr	$\leq$ 11 mr/hr

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
2. SECONDARY CONTAINMENT ISOLATION		
a. Reactor Building Exhaust Radiation - High	$\leq$ 11 mr/hr	$\leq$ 11 mr/hr
b. Drywell Pressure - High	$\leq$ 2 psig	$\leq$ 2 psig
c. Reactor Vessel Water Level - Low, Level 2	$\geq$ + 112 inches[a]	$\geq$ + 112 inches[a]
3. REACTOR WATER CLEANUP SYSTEM ISOLATION		
a. Δ Flow - High	$\leq$ 53 gal/min	$\leq$ 53 gal/min
b. Area Temperature - High	$\leq$ 150°F	$\leq$ 150°F
c. Area Ventilation Δ Temperature - High	$\leq$ 50°F	$\leq$ 50°F
d. SLCS Initiation	NA	NA
e. Reactor Vessel Water Level - Low, Level 2	$\geq$ + 112 inches[a]	$\geq$ + 112 inches[a]
f. Δ Flow - High - Time Delay Relay	$\leq$ 45 seconds	$\leq$ 45 seconds

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
4. CORE STANDBY COOLING SYSTEMS ISOLATION		
a. High Pressure Coolant Injection System Isolation		
1. HPCI Steam Line Flow - High	$\leq$ 300% of rated flow	$\leq$ 300% of rated flow
2. HPCI Steam Line Flow - High Time Delay Relay	$3 \leq t \leq 7$ seconds	$3 \leq t \leq 12$ seconds
3. HPCI Steam Supply Pressure - Low	$\geq$ 100 psig	$\geq$ 100 psig
4. HPCI Steam Line Tunnel Temperature - High	$\leq$ 200°F	$\leq$ 200°F
5. Bus Power Monitor	NA	NA
6. HPCI Turbine Exhaust Diaphragm Pressure - High	$\leq$ 10 psig	$\leq$ 10 psig
7. HPCI Steam Line Ambient Temperature - High	$\leq$ 200°F	$\leq$ 200°F
8. HPCI Steam Line Area Δ Temperature - High	$\leq$ 50°F	$\leq$ 50°F
9. HPCI Equipment Area Temperature - High	$\leq$ 175°F	$\leq$ 175°F
10. Drywell Pressure - High	$\leq$ 2 psig	$\leq$ 2 psig

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
4. CORE STANDBY COOLING SYSTEMS ISOLATION (Continued)		
b. Reactor Core Isolation Cooling System Isolation		
1. RCIC Steam Line Flow - High	$\leq$ 300% of rated flow	$\leq$ 300% of rated flow
2. RCIC Steam Line Flow - High Time Delay Relay	$3 \leq t \leq 7$ seconds	$3 \leq t \leq 12$ seconds
3. RCIC Steam Supply Pressure - Low	$\geq$ 50 psig	$\geq$ 50 psig
4. RCIC Steam Line Tunnel Temperature - High	$\leq$ 175°F	$\leq$ 175°F
5. Bus Power Monitor	NA	NA
6. RCIC Turbine Exhaust Diaphragm Pressure - High	$\leq$ 10 psig	$\leq$ 10 psig
7. RCIC Steam Line Ambient Temperature - High	$\leq$ 200°F	$\leq$ 200°F
8. RCIC Steam Line Area Δ Temperature - High	$\leq$ 50 [illegible]	$\leq$ 50°F
9. RCIC Equipment Room Ambient Temperature - High	$\leq$ 175°F	$\leq$ 175°F
10. RCIC Equipment Room Δ Temperature - High	$\leq$ 50°F	$\leq$ 50°F
11. RCIC Steam Line Tunnel Temperature - High Time Delay Relay	$\leq$ 30 minutes	$\leq$ 30 minutes
12. Drywell Pressure - High	$\leq$ 2 psig	$\leq$ 2 psig

TABLE 3.3.2-3
ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Seconds)[a][e]
1. PRIMARY CONTAINMENT ISOLATION	
a. Reactor Vessel Water Level -	
1. Low, Level 1	≤13
2. Low, Level 3	≤ 1.0[d] ≤ 13[f]
b. Drywell Pressure - High	≤13
c. Main Steam Line	
1. Radiation - High[b]	≤1.0[d] ≤13[f]
2. Pressure - Low	≤13
3. Flow - High	≤0.5[d] ≤13[f]
4. Flow - High	≤0.5[d] ≤13[f]
d. Main Steam Line Tunnel Temperature - High	≤13
e. Condenser Vacuum - Low	≤13
f. Turbine Building Area Temperature - High	NA
g. Main Stack Radiation - High[b]	≤1.0[d]
h. Reactor Building Exhaust Radiation - High[b]	NA
2. SECONDARY CONTAINMENT ISOLATION	
a. Reactor Building Exhaust Radiation - High[b]	≤13
b. Drywell Pressure - High	≤13
c. Reactor Vessel Water Level - Low, Level 2	≤13
3. REACTOR WATER CLEANUP SYSTEM ISOLATION	
a. Δ Flow - High	≤45[c]
b. Area Temperature - High	≤13
c. Area Ventilation Δ Temperature - High	≤13
d. SLCS Initiation	NA
e. Reactor Vessel Water Level - Low, Level 2	≤13
f. Δ Flow - High - Time Delay Relay	NA

TABLE 3.3.2-3 (Continued)

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Seconds)[a][e]
4. CORE STANDBY COOLING SYSTEMS ISOLATION	
a. High Pressure Coolant Injection System Isolation	
1. HPCI Steam Line Flow - High	≤13[c]
2. HPCI Steam Line Flow - High Time Delay Relay	NA
3. HPCI Steam Supply Pressure - Low	≤13
4. HPCI Steam Line Tunnel Temperature - High	≤13
5. Bus Power Monitor	NA
6. HPCI Turbine Exhaust Diaphragm Pressure - High	NA
7. HPCI Steam Line Ambient Temperature - High	NA
8. HPCI Steam Line Area Δ Temperature - High	NA
9. HPCI Equipment Area Temperature - High	NA
10. Drywell Pressure - High	NA
b. Reactor Core Isolation Cooling System Isolation	
1. RCIC Steam Line Flow - High	≤13[c]
2. RCIC Steam Line Flow - High Time Delay Relay	NA
3. RCIC Steam Supply Pressure - Low	NA
4. RCIC Steam Line Tunnel Temperature - High	NA
5. Bus Power Monitor	NA
6. RCIC Turbine Exhaust Diaphram Pressure - High	NA
7. RCIC Steam Line Ambient Temperature - High	NA
8. RCIC Steam Line Area Δ Temperature - High	NA
9. RCIC Equipment Room Ambient Temperature - High	NA
10. RCIC Equipment Room Δ Temperature - High	NA
11. RCIC Steam Line Tunnel Temperature - High Time Delay Relay	NA
12. Drywell Pressure - High	NA

TABLE 3.3.2-3 (Continued)

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Seconds)(a)(e)
5. SHUTDOWN COOLING SYSTEM ISOLATION	
a. Reactor Vessel Water Level - Low, Level 1	NA
b. Reactor Steam Dome Pressure - High	NA

TABLE 3.3.2-3 (Continued)

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

NOTES

(a) The isolation system instrumentation response time shall be measured and recorded as a part of the ISOLATION SYSTEM RESPONSE TIME. Isolation system instrumentation response time specified includes any delay for diesel generator starting assumed in the accident analysis.

(b) Radiation monitors are exempt from response time testing. Response time shall be measured from detector output or the input of the first electronic component in the channel.

(c) Includes time delay added by the time delay relay.

(d) Isolation actuation instrumentation response time for MSIVs only. No diesel generator delays assumed.

(e) Isolation system instrumentation response time specified for the Trip Function actuating each valve group/damper shall be added to the isolation time for valves in each valve group shown in Table 3.6.3-1 and secondary containment isolation dampers shown in Table 3.6.5.2-1 to obtain ISOLATION SYSTEM RESPONSE TIME for each valve/damper.

(f) Isolation system instrumentation response time for associated valves except MSIVs.

TABLE 4.3.2-1

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level -				
1. Low, Level 1				
Transmitter:	NA(a)	NA	R(b)	1, 2, 3
Trip Logic:	D	M	M	1, 2, 3
2. Low, Level 3				
Transmitter:	NA(a)	NA	R(b)	1, 2, 3
Trip Logic	D	M	M	1, 2, 3
b. Drywell Pressure - High				
Transmitter:	NA(a)	NA	R(b)	1, 2, 3
Trip Logic:	D	M	M	1, 2, 3
c. Main Steam Line				
1. Radiation - High	D	W	R(d)	1, 2, 3
2. Pressure - Low				
Transmitter:	NA(a)	NA	R(b)	1
Trip Logic:	D	M	M	1
3. Flow - High				
Transmitter:	NA(a)	NA	R(b)	1
Trip Logic:	D	M	M	1
4. Flow - High	D	M	M	2, 3
d. Main Steam Line Tunnel Temperature - High	NA	M	R	1, 2, 3
e. Condenser Vacuum - Low				
Transmitter:	NA(a)	NA	R(b)	1, 2(e)
Trip Logic:	D	M	M	1, 2(e)
f. Turbine Building Area Temperature - High	NA	M	R	1, 2, 3
g. Main Stack Radiation - High	NA	Q	R	1, 2, 3
h. Reactor Building Exhaust Radiation - High	D	M	R	1, 2, 3

TABLE 4.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
2. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Building Exhaust Radiation - High	D	M	R	1,2,3,5, and (f)
b. Drywell Pressure - High				
Transmitter:	NA(a)	NA	R(b)	1, 2, 3
Trip Logic:	D	M	M	1, 2, 3
c. Reactor Vessel Water Level - Low, Level 2				
Transmitter:	NA(a)	NA	R(b)	1, 2, 3
Trip Logic:	D	M	M	1, 2, 3
3. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. Δ Flow - High	D	M	R	1, 2, 3
b. Area Temperature - High	NA	M	R	1, 2, 3
c. Area Ventilation Δ Temperature - High	NA	M	R	1, 2, 3
d. SLCS Initiation	NA	R	NA	1, 2, 3
e. Reactor Vessel Water Level - Low, Level 2				
Transmitter:	NA(a)	NA	R(b)	1, 2, 3
Trip Logic:	D	M	M	1, 2, 3
f. Δ Flow - High - Time Delay Relay	NA	M	R	1, 2, 3

TABLE 4.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
4. CORE STANDBY COOLING SYSTEMS ISOLATION				
a. High Pressure Coolant Injection System Isolation				
1. HPCI Steam Line Flow - High				
Transmitter:	NA[(a)]	NA	R[(b)]	1, 2, 3
Trip Logic:	D	M	M	1, 2, 3
2. HPCI Steam Line Flow - High Time Delay Relay	NA	R	R	1, 2, 3
3. HPCI Steam Supply Pressure - Low	NA	M	R	1, 2, 3
4. HPCI Steam Line Tunnel Temperature - High	NA	M	Q	1, 2, 3
5. Bus Power Monitor	NA	R	NA	1, 2, 3
6. HPCI Turbine Exhaust Diaphragm Pressure - High	NA	M	Q	1, 2, 3
7. HPCI Steam Line Ambient Temperature - High	NA	M	R	1, 2, 3
8. HPCI Steam Line Area Δ Temperature - High	NA	M	R	1, 2, 3
9. HPCI Equipment Area Temperature - High	NA	M	Q	1, 2, 3
10. Drywell Pressure - High				
Transmitter:	NA[(a)]	NA	R[(b)]	1, 2, 3
Trip Logic:	D	M	M	1, 2, 3

TABLE 4.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
4. CORE STANDBY COOLING SYSTEMS ISOLATION (Continued)				
b. Reactor Core Isolation Cooling System Isolation				
1. RCIC Steam Line Flow - High				
Transmitter:	NA[a]	NA	R[b]	1, 2, 3
Trip Logic:	D	M	M	1, 2, 3
2. RCIC Steam Line Flow - High Time Delay Relay	NA	R	R	1, 2, 3
3. RCIC Steam Supply Pressure - Low	NA	M	Q	1, 2, 3
4. RCIC Steam Line Tunnel Temperature - High	NA	M	R	1, 2, 3
5. Bus Power Monitor	NA	R	NA	1, 2, 3
6. RCIC Turbine Exhaust Diaphragm Pressure - High	NA	M	R	1, 2, 3
7. RCIC Steam Line Ambient Temperature - High	NA	M	R	1, 2, 3
8. RCIC Steam Line Area Δ Temperature - High	NA	M	R	1, 2, 3
9. RCIC Equipment Room Ambient Temperature - High	NA	M	Q	1, 2, 3
10. RCIC Equipment Room Δ Temperature - High	NA	M	Q	1, 2, 3
11. RCIC Steam Line Tunnel Temperature - High Time Delay Relay	NA	M	R	1, 2, 3
12. Drywell Pressure - High				
Transmitter:	NA[a]	NA	R[b]	1, 2, 3
Trip Logic:	D	M	M	1, 2, 3

TABLE 4.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

NOTES

(a) The transitter channel check is satisfied by the trip unit channel check. A separate transmitter check is not required.

(b) Transmitters are exempted from the monthly channel calibration.

(c) If not performed within the previous 31 days.

(d) Testing shall verify that the mechanical vacuum pump trips and the mechanical vacuum pump line valve closes.

(e) When reactor steam pressure $\geq$ 500 psig.

(f) When handling irradiated fuel in the secondary containment.

TABLE 3.6.3-1

PRIMARY CONTAINMENT ISOLATION VALVES

Table 3.6.3-1 has been deleted.

Refer to Plant Procedure RCI-02.6.

Pages 3/4 6-15 through 3/4 6-17 have been deleted.

(Next page is 3/4 6-18)

CONTAINMENT SYSTEMS

SECONDARY CONTAINMENT AUTOMAT ""LATION DAMPERS

LIMITING CONDITION FOR OPERATION

3.6.5.2 The secondary containment automatic isolation dampers shown in Table 3.6.5.2-1 shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 5, and *.

ACTION:

With one or more of the secondary containment isolation dampers specified in Table 3.6.5.2-1 inoperable, operation may continue and the provisions of Specification 3.0.4 are not applicable, provided that at least one isolation damper is maintained OPERABLE in each affected penetration that is open, and:

a. The inoperable damper is restored to OPERABLE status within 8 hours, or

b. The affected penetration is isolated by use of a closed damper within 8 hours, or

c. SECONDARY CONTAINMENT INTEGRITY is demonstrated within 8 hours and the damper is restored to OPERABLE status within 7 days.

Otherwise, in OPERATIONAL CONDITION 1, 2, or 3, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

Otherwise, in OPERATIONAL CONDITION 5 or *, suspend irradiated fuel handling in the secondary containment, CORE ALTERATIONS, or activities that could reduce the SHUTDOWN MARGIN. The provisions of Specification 3.0.3 are not applicable.

*When irradiated fuel is being handled in the secondary containment.

TABLE 3.6.5.2-1

SECONDARY CONTAINMENT AUTOMATIC ISOLATION DAMPERS

Table 3.6.5.2-1 has been deleted.

Refer to Plant Procedure RCI-02.6.

3/4.6.3 PRIMARY CONTAINMENT ISOLATION VALVES (Continued)

A list of automatic closing primary containment isolation valves and their associated closure times shall be available at the plant in accordance with Section 50.71(c) of 10 CFR Part 50. The addition and deletion of primary containment isolation valves shall be made in accordance with Section 50.59 of 10 CFR Part 50.

3/4.6.4 VACUUM RELIEF

Vacuum relief breakers are provided to equalize the pressure between the drywell and suppression pool and the suppression pool and reactor building. This system will maintain the structural integrity of the containment under conditions of large differential pressures.

The vacuum breakers between the drywell and the suppression pool must not be inoperable in the open position since this would allow bypassing of the suppression pool in case of an accident. There are an adequate number of valves to provide some redundancy so that operation may continue with no more than 2 vacuum breakers inoperable and secured in the closed position.

Each set of vacuum relief valves between the suppression chamber and reactor building provides 100% relief, which may be required in the unlikely event that negative pressures develop in the primary containment.

The Nitrogen Backup System provides backup motive power for these suppression pool-reactor building vacuum breakers on a loss of instrument air. The normal non-interruptible instrument air system for these vacuum breakers is designed as a Seismic Class I system supplied by air compressors powered from the emergency buses. The Nitrogen System serves as a backup to the air system and thus the loss of the Nitrogen System, or portions thereof, does not make the vacuum breakers inoperable. This design allows for the out of service times in Actions b and c. The Nitrogen Backup System is added to the Suppression Pool-Reactor Building Vacuum Breaker specification to satisfy NRC concerns relative to 10 CFR 50.44(c)(3) as addressed in the Brunswick Safety Evaluation Report dated October 30, 1986 concerning Generic Letter 84-09. Pressurization to 1130 psig assures sufficient system capacity to provide 24 hours of operation with design valve actuation and system leakage.

3/4.6.5 SECONDARY CONTAINMENT

Secondary containment is designed to minimize any ground level release of radioactive material which may result from an accident. The reactor building provides secondary containment during normal operation when the drywell is sealed and in service. When the reactor is shut down or during refueling the drywell may be open and the reactor building then becomes the primary containment.

3/4.6.5 SECONDARY CONTAINMENT (Continued)

Establishing and maintaining a vacuum in the building with the standby gas treatment system, once per 18 months, along with the surveillance of the valves, is adequate to ensure that there are no violations of the integrity of the secondary containment.

A list of secondary containment automatic isolation dampers shall be available at the plant in accordance with Section 50.71(c) of 10 CFR Part 50. The addition and deletion of secondary containment automatic isolation dampers shall be made in accordance with Section 50.59 of 10 CFR Part 50.

3/4.6.6 CONTAINMENT ATMOSPHERE CONTROL

The OPERABILITY of the containment iodine filter trains ensures that sufficient iodine removal capability will be available in the event of a LOCA. The reduction of containment iodine inventory reduces the resulting site boundary radiation doses associated with containment leakage. The operation of this system and resultant iodine removal capacity are consistent with the assumptions used in the LOCA analyses.

The OPERABILITY of the equipment and systems required for the detection and control of hydrogen gas ensures that this equipment will be available to maintain the hydrogen concentration within containment below its flammable limit during post-LOCA conditions. The containment inerting system is capable of controlling the expected hydrogen generation associated with 1) zirconium-water reactions, 2) radiolytic decomposition of water, and 3) corrosion of metals within containment. The hydrogen control system is consistent with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a LOCA."