December 5, 1990

Docket Nos. 50-325
and 50-324

DISTRIBUTION
See attached list

Mr. Lynn W. Eury
Executive Vice President
Power Supply
Carolina Power & Light Company
Post Office Box 1551
Raleigh, North Carolina 27602

Dear Mr. Eury:

SUBJECT: ISSUANCE OF AMENDMENT NO. 149 TO FACILITY OPERATING LICENSE NO. DPR-71 AND AMENDMENT NO. 179 TO FACILITY OPERATING LICENSE NO. DPR-62 - BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2, REGARDING PRIMARY CONTAINMENT ISOLATION SYSTEM (TAC NOS. 67991 AND 67992)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 149 to Facility Operating License No. DPR-71 and Amendment No. 179 to Facility Operating License No. DPR-62, for Brunswick Steam Electric Plant, Units 1 and 2. The amendments consist of changes to the Technical Specifications in response to your submittals dated February 29, 1988, superseded September 20, 1989, as supplemented December 5, 1989, February 15, August 9, and October 24, 1990.

The amendments change the Technical Specifications to revise TS 3/4.3.2 to clarify Instrument Tables 3.3.2-1, 3.3.2-2, 3.3.2-3 and 4.3.2-1 and revise Section 3/4.6.3 to move the primary containment isolation valve and secondary containment isolation damper tables to the plant procedures.

A copy of the related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's Bi-Weekly Federal Register Notice.

Sincerely,

Original Signed By:

Ngoc B. Le, Project Manager
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 149 to License No. DPR-71
2. Amendment No. 179 to License No. DPR-62
3. Safety Evaluation

cc w/enclosures:
See next page

OFC	LA:PD21:DRPR	PM:PD21:DRPR	D:PD21:DRPR			
NAME	PAnderson	NLe:bd	EAdensam			
DATE	12/5/90	12/6/90	12/6/90			

OFFICIAL RECORD COPY
Document Name: BSEP AMEND 67991/67992

DF01
~~DF03~~

Mr. L. W. Eury
Carolina Power & Light Company

Brunswick Steam Electric Plant
Units 1 and 2

cc:

Mr. Russell B. Starkey, Jr.
Vice President
Brunswick Nuclear Project
P. O. Box 10429
Southport, North Carolina 28461

Mr. R. E. Jones, General Counsel
Carolina Power & Light Company
P. O. Box 1551
Raleigh, North Carolina 27602

Ms. Frankie Rabon
Board of Commissioners
P. O. Box 249
Bolivia, North Carolina 28422

Resident Inspector
U. S. Nuclear Regulatory Commission
Star Route 1
P. O. Box 208
Southport, North Carolina 28461

Regional Administrator, Region II
U. S. Nuclear Regulatory Commission
101 Marietta Street, Suite 2900
Atlanta, Georgia 30323

Mr. Dayne H. Brown, Director
Division of Radiation Protection
N. C. Department of Environmental,
Commerce and Natural Resources
P. O. Box 27687
Raleigh, North Carolina 27611-7687

Mr. J. L. Harness
Plant General Manager
Brunswick Steam Electric Plant
P. O. Box 10429
Southport, North Carolina 28461

Mr. H. A. Cole
Special Deputy Attorney General
State of North Carolina
P. O. Box 629
Raleigh, North Carolina 27602

Mr. Robert P. Gruber
Executive Director
Public Staff - NCUC
P. O. Box 29520
Raleigh, North Carolina 27626-0520

AMENDMENT NO. 149 TO FACILITY OPERATING LICENSE NO. DPR-71 - BRUNSWICK, UNIT 1
AMENDMENT NO. 179 TO FACILITY OPERATING LICENSE NO. DPR-62 - BRUNSWICK, UNIT 2

DISTRIBUTION
Docket File
NRC & Local PDRs
PDII-1 Reading
S. Varga
G. Lainas
E. Adensam
P. Anderson
N. Le
OGC (info. only)
D. Hagan (MNBB 3302)
E. Jordan (MNBB 3302)
G. Hill (8) (P1-137)
W. Jones (P-130A)
J. Calvo (11D3)
C. McCracken
S. Newberry
ACRS (10)
GPA/PA
OC/LFMB

cc: Licensee/Applicant Service List

9012170109 901205
PDR ADOCK 05000325
P PDR

170013

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:

9012190122 901205
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	:LBachmann	:JCalvo	:SNewberry	:EAdensam
DATE	:8/15/90	:8/17/90	:08/30/90	:~~06/14/90~~ 8/17/90	:06/28/90 8/17/90	:12/5/90

(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

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

INDEX

BASES

SECTION		PAGE
3/4.4	REACTOR COOLANT SYSTEM (Continued)	
3/4.4.4	CHEMISTRY	B 3/4 4-2
3/4.4.5	SPECIFIC ACTIVITY	B 3/4 4-2
3/4.4.6	PRESSURE/TEMPERATURE LIMITS	B 3/4 4-3
3/4.4.7	MAIN STEAM LINE ISOLATION VALVES	B 3/4 4-7
3/4.4.8	STRUCTURAL INTEGRITY	B 3/4 4-7
3/4.5	EMERGENCY CORE COOLING SYSTEM	
3/4.5.1	HIGH PRESSURE COOLANT INJECTION SYSTEM	B 3/4 5-1
3/4.5.2	AUTOMATIC DEPRESSURIZATION SYSTEM (ADS)	B 3/4 5-1
3/4.5.3	LOW PRESSURE COOLING SYSTEMS	B 3/4 5-2
3/4.5.4	SUPPRESSION POOL	B 3/4 5-4
3/4.6	CONTAINMENT SYSTEMS	
3/4.6.1	PRIMARY CONTAINMENT	B 3/4 6-1
3/4.6.2	DEPRESSURIZATION AND COOLING SYSTEMS	B 3/4 6-3
3/4.6.3	PRIMARY CONTAINMENT ISOLATION VALVES	B 3/4 6-4
3/4.6.4	VACUUM RELIEF	B 3/4 6-5
3/4.6.5	SECONDARY CONTAINMENT	B 3/4 6-5
3/4.6.6	CONTAINMENT ATMOSPHERE CONTROL	B 3/4 6-6
3/4.7	PLANT SYSTEMS	
3/4.7.1	SERVICE WATER SYSTEMS	B 3/4 7-1
3/4.7.2	CONTROL ROOM EMERGENCY FILTRATION SYSTEM	B 3/4 7-1a

OPERABLE - OPERABILITY (Continued)

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-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 -	2, 6	2	1, 2, 3	20
Low, Level 1	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	≥ + 162.5 inches[a]	≥ + 162.5 inches[a]
2. Low, Level 3	≥ + 2.5 inches[a]	≥ + 2.5 inches[a]
b. Drywell Pressure - High	≤ 2 psig	≤ 2 psig
c. Main Steam Line		
1. Radiation - High	≤ 3 x full power background[c]	≤ 3.5 x full power background[c]
2. Pressure - Low	≥ 825 psig	≥ 825 psig
3. Flow - High	≤ 140% of rated flow	≤ 140% of rated flow
d. Main Steam Line Tunnel Temperature - High	≤ 200°F	≤ 200°F
e. Condenser Vacuum - Low	≥ 7 inches Hg vacuum	≥ 7 inches Hg vacuum
f. Turbine Building Area Temperature - High	≤ 200°F	≤ 200°F
g. Main Stack Radiation - High	(b)	(b)
h. Reactor Building Exhaust Radiation - High	≤ 11 mr/hr	≤ 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	≤ 11 mr/hr	≤ 11 mr/hr
b. Drywell Pressure - High	≤ 2 psig	≤ 2 psig
c. Reactor Vessel Water Level - Low, Level 2	≥ + 112 inches[a]	≥ + 112 inches[a]
3. REACTOR WATER CLEANUP SYSTEM ISOLATION		
a. Δ Flow - High	≤ 53 gal/min	≤ 53 gal/min
b. Area Temperature - High	≤ 150°F	≤ 150°F
c. Area Ventilation Δ Temperature - High	≤ 50°F	≤ 50°F
d. SLCS Initiation	NA	NA
e. Reactor Vessel Water Level - Low, Level 2	≥ + 112 inches[a]	≥ + 112 inches[a]
f. Δ Flow - High - Time Delay Relay	≤ 45 seconds	≤ 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	≤ 300% of rated flow	≤ 300% of rated flow
2. HPCI Steam Line Flow - High Time Delay Relay	3 ≤ t ≤ 7 seconds	3 ≤ t ≤ 12 seconds
3. HPCI Steam Supply Pressure - Low	≥ 100 psig	≥ 100 psig
4. HPCI Steam Line Tunnel Temperature - High	≤ 200°F	≤ 200°F
5. Bus Power Monitor	NA	NA
6. HPCI Turbine Exhaust Diaphragm Pressure - High	≤ 10 psig	≤ 10 psig
7. HPCI Steam Line Ambient Temperature - High	≤ 200°F	≤ 200°F
8. HPCI Steam Line Area Δ Temperature - High	≤ 50°F	≤ 50°F
9. HPCI Equipment Area Temperature - High	≤ 175°F	≤ 175°F
10. Drywell Pressure - High	≤ 2 psig	≤ 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	≤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]
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
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 $\geq$ 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)

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	: RBachmann	: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

OFFICIAL RECORD COPY
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

INDEX

BASES

SECTION		PAGE
3/4.4 REACTOR COOLANT SYSTEM (Continued)		
3/4.4.4	CHEMISTRY	B 3/4 4-2
3/4.4.5	SPECIFIC ACTIVITY	B 3/4 4-2
3/4.4.6	PRESSURE/TEMPERATURE LIMITS	B 3/4 4-3
3/4.4.7	MAIN STEAM LINE ISOLATION VALVES	B 3/4 4-7
3/4.4.8	STRUCTURAL INTEGRITY	B 3/4 4-7
3/4.5 EMERGENCY CORE COOLING SYSTEM		
3/4.5.1	HIGH PRESSURE COOLANT INJECTION SYSTEM	B 3/4 5-1
3/4.5.2	AUTOMATIC DEPRESSURIZATION SYSTEM (ADS)	B 3/4 5-1
3/4.5.3	LOW PRESSURE COOLING SYSTEMS	B 3/4 5-2
3/4.5.4	SUPPRESSION POOL	B 3/4 5-4
3/4.6 CONTAINMENT SYSTEMS		
3/4.6.1	PRIMARY CONTAINMENT	B 3/4 6-1
3/4.6.2	DEPRESSURIZATION AND COOLING SYSTEMS	B 3/4 6-3
3/4.6.3	PRIMARY CONTAINMENT ISOLATION VALVES	B 3/4 6-4
3/4.6.4	VACUUM RELIEF	B 3/4 6-5
3/4.6.5	SECONDARY CONTAINMENT	B 3/4 6-5
3/4.6.6	CONTAINMENT ATMOSPHERE CONTROL	B 3/4 6-6
3/4.7 PLANT SYSTEMS		
3/4.7.1	SERVICE WATER SYSTEMS	B 3/4 7-1
3/4.7.2	CONTROL ROOM EMERGENCY FILTRATION SYSTEM	B 3/4 7-1a

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

179

46, 88, 97, 142, 162

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 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	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 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 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	≥ + 162.5 inches[a]	≥ + 162.5 inches[a]
2. Low, Level 3	≥ + 2.5 inches[a]	≥ + 2.5 inches[a]
b. Drywell Pressure - High	≤ 2 psig	≤ 2 psig
c. Main Steam Line		
1. Radiation - High	≤ 3 x full power background[c]	≤ 3.5 x full power background[c]
2. Pressure - Low	≥ 825 psig	≥ 825 psig
3. Flow - High	≤ 140% of rated flow	≤ 140% of rated flow
4. Flow - High	≤ 40% of rated flow	≤ 40% of rated flow
d. Main Steam Line Tunnel Temperature - High	≤ 200°F	≤ 200°F
e. Condenser Vacuum - Low	≥ 7 inches Hg vacuum	≥ 7 inches Hg vacuum
f. Turbine Building Area Temperature - High	≤ 200°F	≤ 200°F
g. Main Stack Radiation - High	(b)	(b)
h. Reactor Building Exhaust Radiation - High	≤ 11 mr/hr	≤ 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	≤ 11 mr/hr	≤ 11 mr/hr
b. Drywell Pressure - High	≤ 2 psig	≤ 2 psig
c. Reactor Vessel Water Level - Low, Level 2	≥ + 112 inches[a]	≥ + 112 inches[a]
3. REACTOR WATER CLEANUP SYSTEM ISOLATION		
a. Δ Flow - High	≤ 53 gal/min	≤ 53 gal/min
b. Area Temperature - High	≤ 150°F	≤ 150°F
c. Area Ventilation Δ Temperature - High	≤ 50°F	≤ 50°F
d. SLCS Initiation	NA	NA
e. Reactor Vessel Water Level - Low, Level 2	≥ + 112 inches[a]	≥ + 112 inches[a]
f. Δ Flow - High - Time Delay Relay	≤ 45 seconds	≤ 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	≤ 300% of rated flow	≤ 300% of rated flow
2. HPCI Steam Line Flow - High Time Delay Relay	3 ≤ t ≤ 7 seconds	3 ≤ t ≤ 12 seconds
3. HPCI Steam Supply Pressure - Low	≥ 100 psig	≥ 100 psig
4. HPCI Steam Line Tunnel Temperature - High	≤ 200°F	≤ 200°F
5. Bus Power Monitor	NA	NA
6. HPCI Turbine Exhaust Diaphragm Pressure - High	≤ 10 psig	≤ 10 psig
7. HPCI Steam Line Ambient Temperature - High	≤ 200°F	≤ 200°F
8. HPCI Steam Line Area Δ Temperature - High	≤ 50°F	≤ 50°F
9. HPCI Equipment Area Temperature - High	≤ 175°F	≤ 175°F
10. Drywell Pressure - High	≤ 2 psig	≤ 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	≤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)

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 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."

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 149 TO FACILITY OPERATING LICENSE NO. DPR-71

AND AMENDMENT NO. 179 TO FACILITY OPERATING LICENSE NO. DPR-62

CAROLINA POWER & LIGHT COMPANY, et al.

BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2

DOCKET NOS. 50-325 AND 50-324

1.0 INTRODUCTION

By letter dated February 29, 1988, Carolina Power & Light Company (CP&L or the licensee), requested Technical Specification (TS) changes for the Brunswick Steam Electric Plant, Units 1 and 2. The February 29, 1988 letter was superseded by application dated September 20, 1989. Additional information was also provided by letters dated December 5, 1989 February 15, 1990, August 9, and October 24, 1990. The August 9, and October 24, 1990, submittals provided clarifying information that did not change the proposed no significant hazards consideration published in the Federal Register. The proposed changes revise TS 3/4.3.2 to modify Instrument Tables 3.3.2-1, 3.3.2-2, 3.3.2-3, and 4.3.2-1. Instrument Table 3.3.2-1 addressed isolation actuation instrumentation minimum channel operability, whereas Table 3.3.2-2 contains the isolation actuation instrumentation setpoints. Table 3.3.2-3 provides for isolation system instrumentation response times, and Table 4.3.2-1 addresses isolation actuation instrumentation surveillance requirements. The licensee also requests that the actual identification of containment isolation valves and secondary containment isolation dampers be removed from the TS and be located in a licensee control document that would only be changed pursuant to 10 CFR 50.59.

The licensee grouped the changes into five categories. The following staff evaluation is also formatted by the same categories. Category 1 would delete valve group 7 in Table 3.3.2-1. Category 2 would add new items and revise current items in Tables 3.3.2-1, 3.3.2-2, 3.3.2-3, and 4.3.2-1. Category 3 changes revise the tables to: (1) provide consistency, (2) correct administrative errors, and (3) provide clarification without impacting operation of the system. Category 4 changes revise the response times and associated footnotes in Table 3.3.2-3 to provide a more accurate description of the instrumentation. Category 5 changes relocate the primary containment isolation valve list, currently provided as Table 3.6.3-1, and the secondary containment automatic isolation damper list, currently provided as Table 3.6.5.2-1, to Plant Procedure RCI-02.6.

9012190140 901205
PDR ADOCK 05000325
P PDR

2.0 EVALUATION

Category 1 Changes

Category 1 changes would delete valve group 7 in Table 3.3.2-1. Valve group 7 currently includes the following items: Primary Containment Isolation/Reactor Vessel Water Level-Low, Level 1; Secondary Containment Isolation/Drywell Pressure High; Reactor Water Cleanup (RWCU) System Isolation/Drywell Pressure High; Shutdown Cooling System Isolation/Reactor Vessel Water Level-Low, Level 1; Shutdown Cooling System Isolation/Reactor Steam Dome Pressure High. The trip signal identified in the items would isolate certain valves in the systems identified. For example, a high reactor steam dome pressure would isolate certain valves of the shutdown cooling system. The valves, by groups, are identified in TS Table 3.6.3-1 entitled "Primary Containment Isolation Valves," and TS Table 3.6.3-2 entitled "Secondary Containment Automatic Isolation Dampers." The licensee states that Table 3.6.3-1 does not identify any valves as group 7 valves. Therefore, since there are no group 7 valves currently identified in the TS, the licensee reasons that the above named items identifying group 7 valves should be deleted. In addition, the licensee provided various reasons why valve group 7 should not be identified in Table 3.3.2-1. The reasons can be grouped as: (1) custom to standard technical specification conversion, (2) logic design changes, and (3) license amendments that were issued over ten years ago to support these efforts.

The staff reviewed Table 3.6.3-1 which contains an identification of the containment isolation valves for each unit. No group 7 valves are identified. The staff also reviewed the Updated Final Safety Analysis (UFSAR) Table 6.2.4-2 entitled "Automatic Primary Containment Isolation Valves." UFSAR Table 6.2.4-2 lists only one set of valves associated with group 7: high pressure coolant injection (HPCI) turbine exhaust vacuum breaker isolation. These two valves are addressed in the licensee's category 2 proposed changes where a new valve group 7 is proposed. Based upon the above evaluation, the staff concluded that valve group 7 in Table 3.3.2-1 should have been removed years ago and there are no group 7 valves except for the HPCI turbine exhaust vacuum breaker isolation valves which will be addressed in the following evaluation. The staff finds the change to delete valve group 7 in Table 3.3.2-1 acceptable.

Category 2 Changes

Category 2 changes add new items and revise current items in Tables 3.3.2-1, 3.3.2-2, 3.3.2-3, and 4.3.2-1. The licensee proposes to add the following new items: 1h (Reactor Building Exhaust Radiation-High), 3f (Delta Flow-High - Time Delay Relay), 4.a.10 (Drywell Pressure-High), 4.b.11 (RCIC Steam Line Tunnel Temperature - High Time Delay Relay), 4.b.12 (Drywell Pressure-High) and Footnotes (i), (j), (k), and (l) for Table 3.3.2-1 only.

The licensee proposes to revise the following items for Table 3.3.2-1 only: 1.c.2 (Main Steam Line Pressure-Low), 1.c.3 (Main Steam Line Flow-High), 1.c.4 (Main Steam Line Flow-High-Unit 2 only), 1.d (Main Steam Line Tunnel Temperature-High), 1e (Condenser Vacuum Low), 1.f. (Turbine Building Area Temperature - High), 4.b.2 (HPCI Steam Line Flow-High Time Delay Relay), 4.b.3 (RCIC Steam Supply Pressure-Low), and 5.b (Reactor Steam Dome Pressure-High). The details of these proposed changes follow.

The licensee proposes to add reactor building exhaust radiation-high trip function to the TS tables under PRIMARY CONTAINMENT ISOLATION. The licensee states that this signal causes the group 6 isolation valves to close during a loss-of-coolant accident. The licensee proposes the same applicable operational condition and action statement as the other group 6 valves under Item 1. The staff reviewed UFSAR Table 7.3-3 entitled "Isolation Signals and Setpoints." Reactor Building Exhaust High Radiation is listed as a signal for primary containment isolation, group 6. The staff agrees that this signal should be added to the TS and agrees that the operability and surveillance requirements are appropriate.

The licensee proposes to add the differential (delta) flow-high time delay relay trip function to the TS tables under REACTOR WATER CLEANUP SYSTEM ISOLATION. The licensee stated that current TS do not specifically reference operability surveillance requirements for the existing RWCU Delta Flow-High time delay relay. These relays do not initiate any isolation signal. However, they are an important part of the instrumentation. RWCU isolation under high flow conditions is already included in the Table 3.3.2-1 as Item 3.a, Delta Flow-High. The licensee proposes similar operability requirements to those already contained in Item 3.a., Delta Flow-High. The staff reviewed UFSAR Table 7.3.1-3 entitled " Isolation Signals and Setpoints." The RWCU high Delta Flow time delay is listed as a signal for primary containment isolation group 3. The staff agrees that the signal should be added to the TS and agrees that the operability and surveillance requirements are appropriate.

The licensee proposes to add drywell pressure-high trip function to the TS tables under HPCI System Isolation. This signal, when combined with existing signal, HPCI Steam Supply Pressure-Low, will close HPCI turbine exhaust vacuum breaker valves, which are identified by a new valve group 7. The HPCI steam supply pressure-low (4.a.3) will continue to close the group 4 valves. The HPCI system has vacuum breaker valves on a vacuum relief line for the HPCI turbine exhaust. These lines help prevent the creation of a water column in the exhaust line. Preventing filling of this column reduces the piping loads which could exist if the turbine is restarted. The valves isolate on coincident HPCI steam line pressure-low and drywell pressure-high. Proposed footnote k reflects this logic makeup. The licensee is proposing an operability requirement similar to other HPCI isolation signals. The staff reviewed UFSAR Table 7.3.1-3 entitled "Isolation Signals and Setpoints." Drywell pressure-high is an isolation signal with HPCI steam supply low pressure for primary

containment isolation group 7. In addition, UFSAR Table 6.2.4-2 entitled "Automatic Primary Containment Isolation Valves" identifies these valves as group 7 valves which close on low steam supply pressure coincident with high drywell pressure. The staff agrees that the signal should be added to the TS and that the operability and surveillance requirements are appropriate.

The licensee proposes to add the RCIC steam line tunnel temperature-high time delay relay trip function to the TS Table 3.3.2-3 under Reactor Core Isolation Cooling (RCIC) System Isolation. The licensee states that the current TS do not specifically reference operability and surveillance requirements for the existing RCIC steam line tunnel high temperature time delay relay. These relays do not initiate any isolation signal; however, they are an important part of the instrumentation. RCIC isolation under high steam line temperature conditions is already included in the table as Item 4.b.7, RCIC steam line ambient temperature high. The licensee proposes the same operability requirements as already contained for item 4.b.7. The staff reviewed UFSAR Table 7.3.1-3 entitled "Isolation Signals and Setpoints." The RCIC steam line tunnel temperature high time delay is listed as a signal for primary containment isolation, group 5. The staff agrees that the signal should be added to the TS and operability and surveillance requirements are appropriate

The licensee proposes to add the drywell pressure-high trip function to the TS Table 3.3.2-3 under Reactor Core Isolation Cooling System Isolation. This request is the same as the HPCI above. The staff performed a similar review of the licensee's request and reviewed the UFSAR. A new valve group 9 will be used for the RCIC turbine exhaust vacuum breaker isolation valves. Group 5 valves will continue to close under RCIC low steam pressure (4.b.3). The staff agrees that the signal should be added to the TS and agrees that the operability and surveillance requirements are appropriate.

In TS Table 3.3.2-1, footnote (i) is proposed to be added for valve group 8 for shutdown cooling system isolation under high reactor steam dome pressure conditions (Item 5.b.). Footnote (i) states "does not isolate E11-F015A, B." This fact is not reflected in the TS, according to the licensee. The staff reviewed UFSAR Table 6.2.4-2 entitled "Automatic Primary Containment Isolation Valves." This table states in footnote (d) that for these valves "does not isolate on high reactor pressure. Isolates on low level only in shutdown cooling mode." The staff agrees to the addition of footnote (i).

In TS 3.3.2-1, footnote (j) is proposed to be added for valve group 1 for Item 1.c.2, 1.c.3, 1.c.4 (BSEP-2 only), 1.d, 1.e, and 1.f. Footnote (j) states "does not isolate B22-F019 or B32-F20." This fact is not reflected in the TS, according to the licensee. The staff reviewed UFSAR Table 6.2.4-2 entitled "Automatic Primary Containment Isolation Valves." This table states in footnote (b) that these valves only isolate on reactor low water level and main steam line high radiation. Thus, the valves would not isolate for the signals associated with the above items. The staff agrees to the addition of footnote (j) as discussed above.

Category 3 Changes

Category 3 changes revise the tables to: (1) provide consistency, (2) correct administrative errors, and (3) provide clarification without impacting operation of the system. The licensee further subdivided this category into six subchanges. The staff's evaluation is formatted accordingly.

The licensee proposes to specify individualized valve groups on separate lines in Table 3.3.2-1 to show that their operability requirements are different (Subchange A). An in depth analysis of this change was provided in the licensee's December 5, 1989 submittal as follows.

Currently, valve groups actuated by an isolation signal are specified under the trip function regardless of whether the actuated valve groups are associated with that isolation function. As an example, valve groups 2, 6, and 8 are listed under both trip functions 1.a.1, Primary Containment Isolation, and 5.a, Shutdown Cooling System Isolation, even though valve groups 2 and 6 are associated with primary containment isolation and valve group 8 with shutdown cooling system isolation. As a result, the Technical Specifications are not consistent in specifying the correct action if the operability requirements are not met. In the above case, Item 1.a.1 requires Action 20 for valve groups 2, 6, and 8 whereas Item 5.a specifies Action 27 for these groups. Action 20, as specified by Item 1.a.1, is appropriate for valve groups 2 and 6 because both are associated with primary containment. Action 27, specified in Item 5.a, is appropriate for valve group 8 since it is associated with shutdown cooling system isolation. The nature of the action specified by Action 27 would not compensate for the inoperability of the valve group 2 and 6 instrumentation and, as such, is inappropriate for that condition.

Similar changes are necessary for Items 2.a, 2.b, and 2.c. These items address the operability requirements for the secondary containment isolation system instrumentation, however the secondary containment isolation dampers are not identified as a valve group operated by the items. This change adds footnote (1) which references the secondary containment isolation dampers. The specified Action 23 is correct for the secondary containment isolation dampers, however, it is not for valve groups 2, 3, and 6 which are related to secondary containment. The correct actions for these valve groups are specified by the primary containment isolation section for valve groups 2 and 6 and the reactor water cleanup system isolation system section for valve group 3. These sections are specifically provided for these isolation functions and contain the appropriate actions. The proposed changes do not reflect a change in the design or operation of the instrumentation. The staff agrees with the licensee's rationale and the proposed changes are acceptable because they clarify the valve groupings and their respective functions.

The licensee proposes to revise titles of the items in Tables 3.3.2-1, 3.3.2-2, 3.3.2-3, and 4.3.2-1 to provide consistency with other titles, correct administrative errors, and provide a more accurate description of the instrumentation (Subchange B). The licensee states that the changes

do not reflect changes in the design or operation of the instrumentation. The corrected titles are listed on pages E1-9 and E1-10 of the September 20, 1989 submittal. The staff reviewed the revised titles, and the changes are acceptable.

The licensee proposes to replace Footnote * with (f) in Item 2.a in Table 4.3.2-1 (Subchange C) on page 3/4 3-28. The footnote wording itself does not change. This is administrative in nature and is acceptable.

The licensee proposes to delete the word "or" from the end of Item a.2 under the "Primary Containment Integrity" definition on page 1-5 (Subchange D). According to the licensee, the conjunction is out of place and should be deleted. The licensee states that there is no technical basis for the current wording, nor does the conjunction represent any logic. The staff agrees that the "or" makes no sense in Item a.2 on page 1-5, and the change is acceptable.

The licensee proposes to delete Footnote * under Specification 4.6.3.4 (BSEP-1 only) (Subchange E). Footnote * was added to the TS via Amendment No. 72 to provide a one-time extension of the 18 month surveillance interval for four reactor instrumentation system isolation valves. The licensee stated that the surveillance has been completed and the footnote is no longer applicable. Based on the licensee's statements that the surveillance was completed and the footnote is no longer applicable, the change is acceptable.

The licensee proposes to delete Item 1.a.2, Reactor Vessel Water Level-Low, Level 2, from Tables 3.3.2-1, 3.3.2-2, 3.3.2-3, and 4.3.2-1 (Subchange F). Unit 1 Amendment No. 122, dated February 6, 1989 and Unit 2 Amendment No 146, dated April 18, 1989, revised the reactor vessel water level trip function for the valve group 1 isolation valves from low, level 2 to low, level 3. This resulted in only valve group 3 being actuated by the low, level 2 trip function. Valve group 3 isolates the reactor water cleanup system and is addressed specifically in Item 3.3 for the low, level 2 instrumentation. Item 1.a.2 is being deleted because the instrumentation no longer actuates any valve groups that need to be addressed under Item 1. This change does not represent any physical change to the design or operation of any systems. It only more accurately describes the trip function associated with the group 3 valves. The staff agrees that there is unnecessary duplication and agrees to the change.

Category 4 Changes

Category 4 changes consist of revising certain response times and associated footnotes in Table 3.3.2-3 to provide a more accurate description of the instrumentation.

The licensee proposes to change the wording of footnote (a) slightly and move the reference of this note from specific table items the (currently Item 4.a.1 and 4.b.1) to the heading entitled "RESPONSE TIME". Currently, part of footnote (a) states "Isolation system instrumentation response time specified includes the delay for diesel generator starting assumed in the accident analysis." The licensee desires to change the words "the delay" to "any delay. Some valves depend upon diesel generator AC power when loss of offsite power is assumed and some do not. The change is necessary when the footnote is placed at the heading "RESPONSE TIME." Thus, footnote (a) now addresses instrumentation response time as: (1) time for the instrument to respond, (2) any delay caused by the use of timers, and (3) any delay due the diesel generator starting. This time when added to the valve isolation time, as addressed in footnote (e), will give the isolation system response time. The staff agrees to the change.

The current TS specify an instrumentation response time of 1.0 second for the trip function instrumentation in Items 1.a.2, 1.c.1, 1.c.3, 2.c, and 3.e. A reference to Footnote (d), which states "Isolation actuation instrumentation response time only," is also provided. This footnote indicates that any time delay needed for diesel generator starting is not subject to the response time requirement since the requirement applies only to the instrumentation.

The trip functions covered by Items 1.a.2, 1.c.1, and 1.c.3 actuate both AC powered valves and the main steamline isolation valves (MSIV). Item 2.c and 3.e cover functions that actuate only AC powered valves. The MSIV are not dependent on AC power for closure. The response times currently specified are appropriate for the MSIV, but not for other AC powered valves actuated by this instrumentation since they are dependent upon the diesel generators. The proposed change establishes a 13-second response time requirement for the valves other than the MSIV to accommodate the diesel generator start time. A new footnote (f) is added with the new 13-second response time which states, "Isolation system instrumentation response time for associated valves except MSIV" for those functions that actuate both the MSIVs and AC powered valves. The proposed change also revises Footnote (d), which is associated with the 1.0-second response time to state, "Isolation system instrumentation response time for MSIVs only. No diesel generator delays assumed."

The above described changes do not represent a change to the plant. The changes clarify the existing information contained in the Table and more accurately portray it. On this basis, the changes are acceptable.

The last change associated with Category 4 changes replaces the response time for isolation of RWCU under high differential flow condition. The current value is less than or equal to thirteen seconds. The value proposed is 45 seconds. The licensee stated that a 45-second time delay is provided for this instrumentation to prevent spurious isolation signals resulting from RWCU pump starts or flow path changes. The time delay is not currently included in the TS response time. The staff reviewed UFSAR

Table 7.3.1-3 entitled "Isolation Signals and Setpoints" and confirmed the use of the 45-second timer. The licensee also confirmed by letter dated February 15, 1990, that the high energy line break analysis supports the 45-second time delay. Therefore, the staff finds the change acceptable.

Category 5 Changes

Category 5 changes relocate the primary containment isolation valve list, currently provided as Table 3.6.3-1, and the secondary containment automatic isolation damper list, currently provided as Table 3.6.5.2-1, to Plant Procedure RCI-02.6. This approach is similar to a TS change made for another nuclear plant owned and operated by the licensee. The procedure is referenced where the Table information used to be contained to ensure that any change to the Table would be subject to the provisions of 10 CFR 50.59. A paragraph is also added to the Bases statement to explain the removal of these Tables.

The staff does not believe that the listing of the actual valves/dampers need to be provided in the TS. Removal of such detailed listings of plant equipment from TS have been the subject of license amendments in the recent past. For example, snubber listings have been deleted from various plant TS because the actual snubber identification numbers listed in the TS were not needed for plant safety. The licensee's request to delete the actual valve/damper identification numbers is similar to the example, and the staff agrees to the change.

The licensee stated that CP&L has completed a review of the primary containment isolation system TS and determined that Table 3.6.3-1 needs to be revised to accurately reflect the as-built design of the primary containment isolation system. The staff requested the licensee to explain the differences between the TS valve listing and the similar listing in the procedure. The licensee's responses in their letter dated December 5, 1989, indicate that the current Technical Specifications list only automatic isolation valves with stroke time requirements. Technical Specification Interpretation 85-01 expanded the applicability of Technical Specification Section 3.6.3 to include the list of the primary containment isolation valves contained in existing Plant Procedure SD-12. It will also specify the same isolation times as currently specified by the Technical Specifications. The list of valves and requirements specified by the new procedure, therefore, will be the same as that currently specified by the Technical Specifications as expanded by the Technical Specification Interpretation. The proposed relocation primary containment isolation system valve listing is acceptable because it is consistent with TS simplifications already accepted in other nuclear plants, and this change does not impact on plant safety or operations.

3.0 ENVIRONMENTAL CONSIDERATIONS

These amendments change a requirement with respect to installation or use of a facility component located within the restricted areas as defined in 10 CFR Part 20 and change surveillance requirements. The staff has determined that these amendments involve no significant increase in the

amounts, and no significant change in the types, of any effluents that may be released off site; and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that these amendments involve no significant hazards consideration, and there has been no public comment on such finding. Accordingly, these amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of these amendments.

4.0 CONCLUSION

The Commission made a proposed determination that these amendments involve no significant hazards consideration which was published in the Federal Register (55 FR 8219) on March 7, 1990, and consulted with the State of North Carolina. No public comments or requests for hearing were received, and the State of North Carolina did not have any comments.

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations, and (3) 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: December 5, 1990

Principal Contributor: N. B. Le