UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

October 14, 1993

Docket Nos. 50-325
and 50-324

Mr. R. A. Anderson
Vice President
Carolina Power & Light Company
Brunswick Steam Electric Plant
Post Office Box 10429
Southport, North Carolina 28461

Dear Mr. Anderson:

SUBJECT: ISSUANCE OF AMENDMENT NO.166 TO FACILITY OPERATING LICENSE NO. DPR-71 AND AMENDMENT NO. 197 TO FACILITY OPERATING LICENSE NO. DPR-62 REGARDING STEAM LEAK DETECTION EQUIPMENT - BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2 (TAC NOS. M84686 AND M84687)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 166 to Facility Operating License No. DPR-71 and Amendment No.197 to Facility Operating License No. DPR-62 for Brunswick Steam Electric Plant, Units 1 and 2. The amendments change the Technical Specifications in response to your submittals dated September 14, 1992, as supplemented January 13, January 25, February 8, May 11, June 18, July 26, and September 21, 1993.

The amendments revise the Technical Specifications to allow the replacement of existing Riley, GEMAC and Fenwal steam leak detection equipment with General Electric Company NUMAC leak detection equipment. The proposed amendments also revise surveillance requirements for steam leak detection instrumentation associated with the reactor core isolation cooling system, the high pressure coolant injection system, and the reactor water cleanup system. The specific changes include:

(1) Delete the channel check surveillance test for the reactor water cleanup system isolation high differential flow function.

(2) Extend and standardize the channel functional test and channel calibration surveillance frequencies for the reactor water cleanup, high pressure coolant injection, and reactor core isolation cooling system isolation ambient and differential temperature functions.

(3) Increase the reactor water cleanup system isolation differential flow time delay trip setpoint and allowable value from "less than or equal to 45 seconds" to "less than or equal to 30 minutes."

(4) Increase the reactor water cleanup system isolation differential flow trip setpoint and allowable value from "less than or equal to 53 gal/min" to "less than or equal to 73 gal/min."

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October 14, 1993

(5) Delete the instrument response time requirement for the high pressure coolant injection system isolation steam line tunnel temperature - high function.

(6) Delete the instrument response time requirement for the reactor water cleanup system isolation area temperature - high and area ventilation differential temperature - high functions.

(7) Delete the instrument response time requirement for the reactor water cleanup system isolation differential flow - high function.

(8) Revise the description of the reactor water cleanup isolation differential flow delay trip function to reflect elimination of the time delay relays per the new system configuration.

(9) Add a new reactor water cleanup system isolation area temperature function for piping outside of the reactor water cleanup room.

During the review of the Electro-Magnetic Interference and Radio Frequency Interference (EMI/RFI) qualifications for the NUMAC leak detection system, the NRC staff found that the final report on the additional EMI/RFI testing being performed by GE, and the final report on the results of the site mapping survey were not available. As indicated in the Safety Evaluation, you are required to provide these reports to the staff upon completion.

A copy of the related Safety Evaluation and a Notice of Issuance are also enclosed.

Sincerely,
ORIGINAL SIGNED BY:

Patrick D. Milano, Senior Project Manager
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 166 to License No. DPR-71
2. Amendment No. 197 to License No. DPR-62
3. Safety Evaluation
4. Notice of Issuance

cc w/enclosures:
See next page

*See Previous Concurrence

OFFICE	LA:PD21:DRPE*	PM:PD21:DRPE*	AD:PD21:DRPE*	OGC*
NAME	PAnderson	PMilano	SBajwa	APH
DATE	09/21/93	09/21/93	10/14/93	10/06/93

Document Name:G:\BRUNSWIC\BR84686.AMD

(5) Delete the instrument response time requirement for the high pressure coolant injection system isolation steam line tunnel temperature - high function.

(6) Delete the instrument response time requirement for the reactor water cleanup system isolation area temperature - high and area ventilation differential temperature - high functions.

(7) Delete the instrument response time requirement for the reactor water cleanup system isolation differential flow - high function.

(8) Revise the description of the reactor water cleanup isolation differential flow delay trip function to reflect elimination of the time delay relays per the new system configuration.

(9) Add a new reactor water cleanup system isolation area temperature function for piping outside of the reactor water cleanup room.

During the review of the Electro-Magnetic Interference and Radio Frequency Interference (EMI/RFI) qualifications for the NUMAC leak detection system, the NRC staff found that the final report on the additional EMI/RFI testing being performed by GE, and the final report on the results of the site mapping survey were not available. As indicated in the Safety Evaluation, you are required to provide these reports to the staff upon completion.

A copy of the related Safety Evaluation and a Notice of Issuance are also enclosed.

Sincerely,

Patrick D. Milano, Senior Project Manager
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 166 to
   License No. DPR-71
2. Amendment No. 197 to
   License No. DPR-62
3. Safety Evaluation
4. Notice of Issuance

cc w/enclosures:
See next page

AMENDMENT NO. 166 TO FACILITY OPERATING LICENSE NO. DPR-71 - BRUNSWICK, UNIT 1
AMENDMENT NO. 197 TO FACILITY OPERATING LICENSE NO. DPR-62 - BRUNSWICK, UNIT 2

DISTRIBUTION:

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

CAROLINA POWER & LIGHT COMPANY, et al.

DOCKET NO. 50-325

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 166
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 September 14, 1992, as supplemented January 13, January 25, February 8, May 11, June 18, July 26, and September 21, 1993, 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:

(2) <u>Technical Specifications</u>

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 166 , 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 90 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Bajwa

S. Singh Bajwa, Acting 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: October 14, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 166

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
3/4 3-13	3/4 3-13
3/4 3-19	3/4 3-19
3/4 3-23	3/4 3-23
3/4 3-24	3/4 3-24
3/4 3-28	3/4 3-28
3/4 3-29	3/4 3-29
3/4 3-30	3/4 3-30

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	24
e. Reactor Vessel Water Level - Low, Level 2	3	2	1, 2, 3	24
f. Δ Flow - High - Time Delay	NA	1	1, 2, 3	24
g. Piping Outside RWCU Rooms Area Temperature - High	3	1	1, 2, 3	24

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	≤ 73 gal/min	≤ 73 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	≤ 30 minutes	≤ 30 minutes
g. Piping Outside RWCU Rooms Area Temperature - High	≤ 120°F	≤ 120°F

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	NA
b. Area Temperature - High	NA
c. Area Ventilation Δ Temperature - High	NA
d. SLCS Initiation	NA
e. Reactor Vessel Water Level - Low, Level 2	≤13
f. Δ Flow - High - Time Delay	NA
g. Piping Outside RWCU Rooms Area Temperature - High	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	NA
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 Diaphragm 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 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	NA	SA	R	1, 2, 3
b. Area Temperature - High	NA	SA	R	1, 2, 3
c. Area Ventilation Δ Temperature - High	NA	SA	R	1, 2, 3
d. SLCS Initiation	NA	R	NA	1, 2
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	NA	SA	R	1, 2, 3
g. Piping Outside RWCU Rooms Area Temperature - High	NA	SA	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 High Flow 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	SA	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	SA	R	1, 2, 3
8. HPCI Steam Line Area Δ Temperature - High	NA	SA	R	1, 2, 3
9. HPCI Equipment Area Temperature - High	NA	SA	R	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	SA	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	SA	R	1, 2, 3
8. RCIC Steam Line Area Δ Temperature - High	NA	SA	R	1, 2, 3
9. RCIC Equipment Room Ambient Temperature - High	NA	SA	R	1, 2, 3
10. RCIC Equipment Room Δ Temperature - High	NA	SA	R	1, 2, 3
11. RCIC Steam Line Tunnel Temperature - High Time Delay Relay	NA	SA	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

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

CAROLINA POWER & LIGHT COMPANY, et al.

DOCKET NO. 50-324

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 197
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 September 14, 1992, as supplemented January 13, January 25, February 8, May 11, June 18, July 26, and September 21, 1993, 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. 197, 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 to be effective and implemented upon the completion of refueling outage no. 10 (B211R1).

FOR THE NUCLEAR REGULATORY COMMISSION

SBajwa

S. Singh Bajwa, Acting 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: October 14, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 197

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
3/4 3-13	3/4 3-13
3/4 3-19	3/4 3-19
3/4 3-23	3/4 3-23
3/4 3-24	3/4 3-24
3/4 3-28	3/4 3-28
3/4 3-29	3/4 3-29
3/4 3-30	3/4 3-30

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	24
e. Reactor Vessel Water Level - Low, Level 2	3	2	1, 2, 3	24
f. Δ Flow - High - Time Delay	NA	1	1, 2, 3	24
g. Piping Outside RWCU Rooms Area Temperature - High	3	1	1, 2, 3	24

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	≤ 73 gal/min	≤ 73 gal/min
b. Area Temperature - High	≤ 150°F	≤ 150°F
c. Area Ventilation Temperature Δ Temp - 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	≤ 30 minutes	≤ 30 minutes
g. Piping Outside RWCU Rooms Area Temperature - High	≤ 120°F	≤ 120°

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	NA
b. Area Temperature - High	NA
c. Area Ventilation Δ Temperature - High	NA
d. SLCS Initiation	NA
e. Reactor Vessel Water Level - Low, Level 2	≤13
f. Δ Flow - High - Time Delay	NA
g. Piping Outside RWCU Rooms Area Temperature - High	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	NA
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 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	NA	SA	R	1, 2, 3
b. Area Temperature - High	NA	SA	R	1, 2, 3
c. Area Ventilation Δ Temperature - High	NA	SA	R	1, 2, 3
d. SLCS Initiation	NA	R	NA	1, 2
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	NA	SA	R	1, 2, 3
g. Piping Outside RWCU Rooms Area Temperature - High	NA	SA	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	SA	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	SA	R	1, 2, 3
8. HPCI Steam Line Area Δ Temperature - High	NA	SA	R	1, 2, 3
9. HPCI Equipment Area Temperature - High	NA	SA	R	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 High - Flow 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	SA	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	SA	R	1, 2, 3
8. RCIC Steam Line Area Δ Temperature - High	NA	SA	R	1, 2, 3
9. RCIC Equipment Room Ambient Temperature - High	NA	SA	R	1, 2, 3
10. RCIC Equipment Room Δ Temperature - High	NA	SA	R	1, 2, 3
11. RCIC Steam Line Tunnel Temperature - High Time Delay Relay	NA	SA	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

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 166 TO FACILITY OPERATING LICENSE NO. DPR-71

AND AMENDMENT NO. 197 TO FACILITY OPERATING LICENSE NO. DPR-62

CAROLINA POWER & LIGHT COMPANY

BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2

DOCKET NOS. 50-325 AND 50-324

1.0 INTRODUCTION

By letter dated September 14, 1992, as supplemented January 13, January 25, February 8, May 11, June 18, July 26, and September 21, 1993, the Carolina Power & Light Company (the licensee or CP&L) submitted amendments to Facility Operating License Nos. DPR-71 and DPR-62 requesting changes to the Brunswick Steam Electric Plant, Units 1 and 2 (BSEP), Technical Specifications (TS). The proposed amendments reflect the replacement of the existing Riley, GEMAC and Fenwal steam leak detection system equipment with General Electric Company (GE) NUMAC leak detection equipment. The proposed amendments also revise the surveillance requirements for the steam leak detection instrumentation associated with the reactor core isolation cooling (RCIC) system, the high pressure coolant injection (HPCI) system, and the reactor water cleanup (RWCU) system. The NRC staff review included an evaluation of the safety significance of the replacement of the existing analog system with the digital equipment from the GE NUMAC line along with the specific changes to TS Table 4.3.2-1. The proposed amendments would revise TS Table 4.3.2-1 to:

(1) Delete of the channel check surveillance test for the RWCU system isolation high differential flow function.

(2) Extend and standardize the channel functional test and channel calibration surveillance frequencies for the RWCU, HPCI, and RCIC systems isolation ambient and differential temperature functions.

(3) Increase the RWCU system isolation differential flow time delay trip setpoint and allowable value from "less than or equal to 45 seconds" to "less than or equal to 30 minutes."

(4) Increase the RWCU system isolation differential flow trip setpoint and allowable value from "less than or equal to 53 gal/min" to "less than or equal to 73 gal/min."

(5) Delete the instrument response time requirement for the (HPCI) system isolation steam line tunnel temperature - high function.

(6) Delete the instrument response time requirement for the RWCU system isolation area temperature - high and area ventilation differential temperature - high functions.

(7) Delete the instrument response time requirement for the RWCU system isolation differential flow - high function.

(8) Revise the description of the RWCU isolation differential flow delay trip function to reflect elimination of the time delay relays under the new system configuration.

(9) Add a new RWCU system isolation area temperature function for piping outside of the RWCU system room.

2.0 EVALUATION

2.1 System Description

The steam leak detection system performs a safety-related function to (a) detect, measure and process temperature signals from the room ambient and the differential temperature sensors of the RWCU, HPCI, and RCIC systems and from the process flow sensors in the RWCU system, (b) compare the input signals to pre-selected levels, and (c) provide isolation signals at the pre-selected setpoints. The steam leak detection system is composed of Class 1E components.

The NUMAC leak detection monitor (LDM) supports the safety-related functions and other non-essential functions such as:

a. Measures input current from the thermocouple input unit and performs the specified temperature calculations

b. Provides high voltage DC power for operating the detector

c. Provides -15 VDC to power associated electronics

d. Provides output trip signals to external equipment

e. Performs automatic calibration

f. Performs automatic self-test and alarm

g. Displays self-test status on demand

h. Provides security by keylock and password against unauthorized changes to setpoints

2.2 Equipment Description

The LDM is a GE NUMAC class-1E system, with architecture consisting of a family of firmware-based 80C86 (16 bit) and 80376 (32 bit) controllers with application-specific analog and digital modules connected via a NUMAC bus. An independent display controller connects to the class-1E processor via a serial link and provides the man-machine interface without affecting the calculations of the class-1E process. The NUMAC architecture also includes both hardware and software watchdog timers and an integral self-test system. Each LDM chassis has the following modules:

- Microcomputer Module
- Display Microprocessor Module
- Front Panel Display and Control Module
- Thermocouple Input Module
- Open Drain Input/Output Relay Output Module
- Analog Module
- 16-Channel Analog Output Module
- GEDAC Communications and Memory Module

The LDM monitors ambient and differential temperatures via externally mounted thermocouples, provides display of local temperature and trip conditions, and provides trip outputs for system isolation and alarm functions. The differential flow functions monitors three temperature and three flow inputs to detect possible RWCU system leakage. Trip outputs are provided for RWCU system isolation and alarm functions. The microcomputer consists of the necessary hardware and software to process the received data and transmit appropriate control signals to other modules within the chassis using a high speed parallel data bus. The microcomputer also communicates with the display microprocessor/front panel display using a serial data link. The microcomputer also performs self-test system diagnostics when not processing instrument data.

The display microcomputer, which is based on the National Semiconductor NSC-800 microprocessor, processes the data from the essential microcomputer for display on the front panel display. The front panel display contains all of the circuitry necessary to interface with the display microcomputer, the front panel's keyboard, and electro-luminescent display.

The NUMAC thermocouple input unit (TCIU) interfaces the ambient and differential thermocouples to the LDM instrument chassis. The unit contains an isothermal terminal board interface that transmits the temperature measurements to the LDM. Six solid state temperature devices in the TCIU are used for determining the cold junction temperature.

The relay output module provides isolation, alarm, and INOP outputs using relay output contacts. An internal automatic bypass capability is provided for the trip functions to the assigned isolation and alarm output relays. The functions are automatically bypassed when a calibration, calibration check or trip check is being performed, when a thermocouple is open, or when a critical self-test fault exists.

The analog module interfaces analog signals from within the LDM chassis to the functional controller. The 16-channel analog output module produces analog outputs proportional to the RWCU system flow. This output is compensated for water density by using temperature measurements and the flow rates to normalize the actual flow.

GEDAC communications and memory module is a one-way high-speed output communications link to provide data for non-1E applications.

The NUMAC LDM has instrument power supplies and detector high voltage power supplies. The instrument power supplies power to the LDM chassis. Each LDM chassis has two redundant diode auctioneered low voltage power supplies for uninterruptable power in the event of a power supply failure.

2.3 System and Equipment Improvements

The licensee expects to improve the reliability and accuracy by replacing the existing LDM with the NUMAC LDM. The current analog LDM uses temperature switch modules that have historically experienced a high drift rate, have been prone to spurious alarms and trips, and have been difficult to maintain. The local switches currently in use do not provide an indication for the monitored areas and have been the cause of significant personnel radiation exposure due to the time spent inside the reactor building performing monthly testing of these switches. The existing RWCU differential flow leak detection instrumentation is subject to significant signal drift and inaccuracy. The loop inaccuracy is particularly evident during RWCU system transient conditions, such as system fill and start-up. These limitations have led to unwarranted process isolation signals that unnecessarily challenge the RWCU system containment isolation function, degrade operator confidence in the control system, create additional work load for operations personnel in dealing with the control system, and result in an excessive number of licensee event reports documenting those unnecessary system isolations.

By retrofitting the LDM with a GE microprocessor based NUMAC system, the licensee should be able to significantly reduce or eliminate these problems at BSEP. The NUMAC system will process signals from the ambient and differential temperature sensors of the RWCU, HPCI, and RCIC systems and from the flow sensors in the RWCU system. The NUMAC system will reduce instrument drift, improve instrument accuracy, simplify maintenance activities, enhance the operator/equipment interface, and provide a means of density compensation for the RWCU system differential flow measurement. In conjunction with NUMAC installation, the licensee will replace the local ambient temperature switches with thermocouples connected to NUMAC channels.

The licensee currently uses NUMAC in three applications at BSEP: the main steam line radiation monitor (MSLRM), the rod worth minimizer (RWM) and the steam jet air ejector radiation monitor (SJAE RM). The MSLRM is a safety-related system while the RWM and SJAE RM are non-safety related systems. These units have in-service histories ranging between three and six years, and have been highly reliable.

2.4 Equipment Evaluation

2.4.1 Review Criteria

Since the LDM is a Class 1E system, the General Design Criteria (GDC), the IEEE Standard 279, "Criteria for Protection Systems for Nuclear Power Generation Station" (10 CFR 50.55a(h)), and the applicable acceptance criteria listed in Section 7.5 of the "Standard Review Plan" (NUREG-0800) were used as review guidance. Additionally, the ANSI/IEEE Standard 7-4.3.2, 1982, "Application Criteria for Programmable Digital Computer Systems in Safety Systems of Nuclear Power Generating Stations," and the corresponding NRC Regulatory Guide (RG) 1.152, "Criteria for Programmable Digital Computer System Software in Safety-Related Systems of Nuclear Power Plants," were also used to evaluate the NUMAC LDM system software design verification and validation processes.

2.4.2 Hardware Evaluation

By GDC 2 and 4 to Appendix A of 10 CFR Part 50, the NRC staff requires that the safety system be designed to withstand the effects of natural phenomena and be qualified to operate in its environmental conditions, which consists of normal and postulated accident conditions. To ensure that the LDM will perform its intended function(s) under its environmental conditions, the staff reviewed the environmental qualification of the NUMAC equipment for (1) temperature and humidity, (2) seismic, (3) radiation, and (4) electro-magnetic and radio frequency interference.

2.4.2.1 Temperature and Humidity

GE performed temperature and humidity tests on the NUMAC instrument chassis and associated modules. The test procedures and results are documented in NEDC-31974P, Appendix C. The staff used IEEE Standard 323-1974, "IEEE Standard for Qualifying Class 1E equipment for Nuclear Power Generating Stations," for review guidance.

By testing equipment unique to LDM and analyzing equipment similar to the NUMAC product line equipment, GE qualified the NUMAC LDM equipment. The NUMAC LDM environmental tests consist of component aging, PC board (module) qualification, instrument qualification, and instrument heat rise. The NUMAC LDM instrument is qualified for operation between 40°F and 122°F and between 10 percent and 90 percent non-condensing humidity. All NUMAC LDM modules completed additional qualification programs, being operationally tested at 158°F. Maximum LDM instrument internal heat rise was measured at 18°F in the vicinity of the power supplies.

The licensee has stated that the design ranges for the BSEP control room temperature and humidity are:

Temperature:	Normal	40 - 120 °F	(75°F Avg)
	Design Basis Accident	40 - 120 °F	(75°F Avg)
Humidity:	Normal	30 - 60% RH	(45% Avg)
	Design Basis Accident	30 - 60% RH	(45% Avg)

The control room average air temperature is maintained at approximately 75 °F. The temperature is required to be observed and recorded daily as directed in BSEP procedure OI-03.4, Control Operator Daily Check Sheet. If the temperature exceeds 77 °F, the surveillance is increased to once per shift. As part of the licensee's response to the station blackout rule for BSEP, the licensee has committed to implement around-the-clock HVAC trouble shooting activities when temperatures exceed 84 °F. The basis for the 84 °F trigger point was a calculation that demonstrated that the control room temperature would not exceed 120 °F within one hour, after the loss of HVAC power supply and with an initial ambient temperature of 85 °F.

The margin between the normal control room general area ambient temperature and the NUMAC qualification temperature is adequate to accommodate potential local heating effects inside the H12-P614 panel.

Humidity is not a directly controlled parameter at BSEP. However, the design humidity range stated above is broadly bounded by the 10 percent to 90 percent humidity range to which the NUMAC is qualified.

Based on the foregoing review, the staff finds that the GE temperature and humidity qualification of the GE NUMAC product envelops the BSEP temperature and humidity requirements. In addition, the staff finds that temperature and humidity tests meet the intent of IEEE Standard 323-1974. Therefore, the staff finds the temperature and humidity qualification acceptable.

2.4.2.2 Seismic Qualification

The LDM equipment and panels which replace existing equipment and panels are safety-related, seismic Category 1 components. Since the replacement might alter some degree of mass and stiffness characteristic of the equipment control panels and structural supports, the seismic/dynamic qualification needed to be demonstrated for the installed equipment.

GE performed a similarity analysis of the NUMAC LDM chassis and interface panels for the licensee. GE performed the similarity analysis to show that the specific BSEP devices are mechanically the same or equivalent to the devices tested, and GE showed that for this reason, the NUMAC LDM is also capable of withstanding the tested seismic forces. All of the processes were performed in compliance with IEEE 334-1975, and is certified as such by GE on the product quality certifications provided with the equipment.

Based on the foregoing review, the staff concludes that seismic qualification performed is adequate.

2.4.2.3 Radiation

The LDM components located in the control room were qualified to a maximum total integrated dose of 1E+4 rad. This is well within the BSEP normal and accident doses for the associated areas. The test procedures and the test results are documented in Appendix C to NEDC-31974P, "NUMAC Qualification Report for Reactor Building Vents Radiation Monitor System for Tennessee

Valley Authority Browns Ferry Nuclear Plant Units 1, 2, and 3," dated November 1991.

2.4.2.4 Electromagnetic Interference and Radio Frequency Interference

Electromagnetic interference and radio frequency interference (EMI/RFI) are random noise produced by systems within the operating environment in a nuclear plant. This random noise can affect the safety of the plant since it can potentially lead to common cause failure of redundant safety-related equipment that are vulnerable to the noise.

In safety-related and non safety-related instrument and control (I&C) equipment at nuclear plants, digital equipment, which operate at higher speeds and lower voltages than the analog equipment it replaces, are especially vulnerable to EMI/RFI noises. Hence, in reviewing the application of digital I&C equipment at nuclear plants, the NRC staff has placed additional emphasis on addressing the vulnerabilities of this equipment to EMI/RFI noise. For general guidance for the review of EMI/RFI, the staff uses GDC 4, "Environmental and Dynamic Effects Design Base." For specific guidance for the review of EMI/RFI, the NRC currently uses the following standards and documents for reference:

MIL-STD-461(A,B,C), "Electro-magnetic Emission and Susceptibility Requirements for the Control of Electro-magnetic Interference"

MIL-STD-462, "Electro-magnetic Interference Characteristics Measurement"

MIL-STD-1399, "Interface Standard for Shipboard Systems, DC Magnetic Field Environment"

SAMA PMC 33.1-1978, "Electro-magnetic Susceptibility of Process Control Instrumentation"

IN83-83, "Use of Portable Radio Transmitters Inside Nuclear Power Plants"

IEC 801-2, "Electromagnetic Compatibility for Industrial-Process Measurement and Control Equipment Part 2: Electrostatic Discharge Requirements"

NUREG CR-3270, "Investigation of Electro-magnetic Interference (EMI) Levels in Commercial Nuclear Power Plants"

a. EMI/RFI Environmental Evaluation

Using the above as guidance, the staff followed four steps to review the effects of the EMI/RFI noise environment around safety-related digital equipment:

(1) Evaluate the plant and identifying the potential EMI/RFI sources

(2) Review the vendor EMI/RFI qualification methodology and range of frequency tested

(3) Review the licensee's EMI/RFI qualification process for the installed equipment

(4) Review licensee's measurement of EMI/RFI field at equipment installed at the plant and verifying that equipment is operating within its qualified environment

b. LDM Operating Environment Assessment

The staff visited the site and performed a walkdown to assess the environment. The walkdown area included the main control room and pump rooms. No concerns were identified during this walkdown.

c. Vendor Qualification Test and the NRC Review:

GE tested the LDM for four different EMI/RFI susceptibilities. The four EMI/RFI susceptibility tests are (1) radiated electric fields, (2) radiated magnetic fields, (3) conductive noise, and (4) static discharges. The vendor selected the test methodologies from various standards. The EMI/RFI test results for the NUMAC LDM are documented in NEDC-31974P, Appendix B.

The radiated electric field susceptibility test was conducted in accordance with SAMA Standard PMC 33.1, "Electromagnetic Susceptibility of Process Control Instrumentation," with a field strength of 65 V/m over frequency range of 20 to 990 Mhz. In addition, a keying test was performed to simulate the keying of a portable radio (walkie-talkie).

The radiated magnetic field susceptibility test was conducted in accordance with GE's test requirements for electromagnetic susceptibility of process control instrumentation test requirements. The test required fifty-foot wires to be attached to the inputs/outputs of the equipment being tested, and signals from a generator are injected into the test wires to simulate the noise induced on the power leads. The radiated magnetic field susceptibility test, however, did not test for low frequency radiated magnetic fields.

The conductive noise test was conducted in accordance with Swedish Standard, Svensk Standard SS 436 15 03, and GE's conductive noise test requirements.

The NUMAC LDM has been qualified by GE per IEC Standard 801-2, "Electromagnetic Compatibility for Industrial-Process Measurement and Control Equipment Part 2: Electrostatic Discharge Requirements."

The staff found that GE's testing methodologies are adequate for the tested frequency ranges. In particular, the radiated electric field test which injected 65 V/m over the frequency range of 20 to 990 Mhz seemed to be more than adequate. However, the staff found that the testing methodologies were inadequate for the low frequency range. In addition, the staff identified that the licensee needed to demonstrate that the BSEP EMI/RFI environmental conditions were within the tested envelope either by site survey or analysis. The staff discussed these findings and concerns with the licensee and GE.

In response to the staff's findings and concerns, the licensee performed additional EMI/RFI test for the frequency ranges that were not covered by the earlier tests. These additional tests are being performed by GE. The GE scheduled issue date for the results of these tests is September 1993. In addition, on May 18 through 20, 1993, the licensee conducted onsite EMI/RFI mapping of the equipment locations and surrounding areas, as detailed in the June 18, and July 26, 1993, supplements.

Based on the foregoing review, the NRC staff finds the licensee's and GE's conclusion that the LDM is EMI/RFI qualified for the BSEP environment to be acceptable. However, the staff's acceptances are contingent upon the licensee providing the staff a final report on the additional EMI/RFI testing being performed by GE, and a final report on the results of the site mapping survey.

2.4.3 Software Evaluation

The LDM application software consists of two principal modules (1) the functional software for the microcomputer, including the self test system, and (2) the front panel keyboard and display software for the display computer. The LDM digital equipment software is written in high level languages, to the maximum extent possible, to simplify software maintenance over the lifetime of the equipment. The total lines of the code required to perform the LDM's intended functions are under 20,000 lines. The functions performed by the software include (1) sampling and filtering sensor data, (2) comparing data to operator defined trip setpoints, (3) updating operator display, (4) generating analog indication and trip output signals, and (5) performing self-tests.

The LDM software has a large number of input and output combinations making it impractical to check for all the possibilities. The reliable operation of such software is assessed qualitatively based on the idea that the software development process and configuration management have a significant impact on producing reliable software.

The staff's LDM software review was assisted by its contractors, Lawrence Livermore National Laboratory and Sohar. By letter dated February 16, 1993, the contractors forwarded their evaluation, "GE NUMAC Verification Audit Report," to the staff.

The staff's software review included an audit of the products of the plans for the NUMAC LDM and examination of the NUMAC generic software development process on January 11-15, 1993, with particular attention to (1) software management plan (SMP), (2) software configuration management plan (SCMP), and (3) software verification and validation plan (SVVP). The staff examined these plans and their implementation for compliance with RG 1.152 and ANSI/IEEE-7-4.3.2-1982.

2.4.3.1 Verification and Validation

GE used nuclear quality assurance programs with supplemental Verification and Validation (V&V) procedures based on RG 1.152 to develop both Class 1E and non-Class 1E NUMAC LDM software. The GE NUMAC line of instruments consists of numerous modular components and uses NUMAC product codes where appropriate. The lines of code are stored in the three sets of firmware.

The GE V&V method is based on logical steps with baseline reviews performed at the completion of each phase of the development process. A list of open items is documented and maintained for each review. The reviewers are independent from the designers and communicate their review in written reports. The V&V review team, however, was not organizationally independent from the design team. The validation step includes a matrix relating each validation test to a functional requirement.

The staff's review of GE Report No. 23A5163, "Software Verification and Validation Plan," Revision 1, dated March 12, 1991, found that the document was weak in describing the V&V process. However, in actual practice, the GE software development method is consistently followed and provides an internally reviewed paper trail throughout the software development process. Testing is done using emulators, and each and every change requires testing. An organizationally independent configuration control engineer is required to sign-off on all baseline reviews (verification steps) and controls the NUMAC library of documents and firmware. The NUMAC review team has nine members and must approve all changes for resolutions of open items.

The design record file (DRF) has a standard format of six sections that include the associated baseline review documents:

- Definition and Planning
- Product Performance Definition (requirements)
- High Level Software Design
- Design/Code Module Test
- Integration Test
- V&V Test (verification and validation)

The DRF provided a very effective record of the project process steps and results, and it contributed substantially to the audit process.

2.4.3.2 Configuration Control

As a part of the configuration control review, the staff reviewed GE 23A5261, "Software Configuration Management Plan," Revision 1, dated March 12, 1991, and SMP GE 23A5262 "Software Management Plan," Revision 1, dated March 12, 1991. The review indicated that the strict configuration control standards are in place and all updates to the NUMAC instruments are performed at GE. Each version of the firmware included all software modules, whether modified or not. Each version is controlled with a separate revision and part number. The user's manual has a very extensive description of the system as well as instructions for its use.

2.4.3.3 NUMAC LDM Software Review Conclusion

The staff found that GE has a formally established design, code and test review process with associated formal documentation. The staff also found that GE has a formal configuration management plan and the plan is being consistently applied. The decision to maintain a library where each revision is a complete entity removes the problems associated with controlling different versions of the code.

The V&V process in actual use appears adequate. Personnel who conducted the V&V activities report to the same first line supervisor as those responsible for the software design. This is considered a deviation from the requirements for organizational independence in RG 1.152. However, the absence or existence of oversight or bias from lacking organizational independence could not be determined within the scope of this audit.

Based on this review, the staff concluded that the NUMAC LDM software is acceptable for use in BSEP. However, because of the weaknesses discussed in this section, the staff also concludes that this approval should not form the basis for accepting this software in any other application subject to regulatory review.

2.4.4 Defense Against the Common Mode Failure

The single failure criterion requires that any single failure within the protection system shall not prevent proper protective action at the system level when required. Common mode/common cause failures can prevent the safety system from performing its intended function. Common mode/common cause failures could also result in the loss of more than one echelon of defense-in-depth provided by the monitoring, control, reactor protection, and engineered safety functions performed by the digital instrumentation and control systems. Particularly, a microprocessor based digital system, such as the NUMAC LDM, which shares data bases and process equipment, has a potential for common mode/common cause failures in the area of software, hardware, and software and hardware interaction. Defense against common mode/common cause failure is provided by quality and/or diversity.

The potential sources of common cause failures for the LDM include the software, hardware, and their qualifications. The software V&V program strongly minimizes, although cannot eliminate, the likelihood of common mode NUMAC failure. The quality aspects of the NUMAC LDM defenses against such failures are discussed in section 2.4.3.

The safety-related functions of the NUMAC LDM consist of contribution to the primary containment isolation function for the HPCI, RCIC, and RWCU systems. In the event of common mode failure preventing the NUMAC high temperature isolation of either the HPCI and RCIC system, each of those systems features a 300 percent high flow isolation function. That function is physically separate from the NUMAC system and constitutes a diverse, redundant, safety-related backup that is capable of responding to a design basis line break within the same response time as the NUMAC temperature detection function.

Other contributions to the HPCI and RCIC isolation logic include low steam supply pressure and high turbine exhaust pressure.

For the RWCU system, there is no equivalent to the 300 percent flow monitoring feature for HPCI and RCIC. Other functions that contribute to an RWCU automatic isolation include reactor vessel water level low level 2, high process temperature entering the RWCU filters and Standby Liquid Control system initiation. Since RWCU is a non-safety related system, the design "fail-safe" NUMAC isolation logic configuration is to fail "closed". Upon loss of either external or internal power to NUMAC, and in response to an indeterminate percentage of possible internal failures, the NUMAC output relays that contribute to the RWCU isolation logic will de-energize to the trip state.

In the event that both division NUMACs were to fail to the non-isolate state concurrent with an actual RWCU line break, the alternate means for detection of that break would include such provisions as reactor building sump cycle and run time alarms, room flood alarms and direct observation by personnel. None of these methods are based on safety-related instrumentation, none contribute to an automatic isolation and none would result in isolation within the same response time as the NUMAC temperature function.

In the event of a failure of the NUMAC system's self-diagnostic, the NUMAC architecture includes an INOP relay output contact which causes a System Test/Trouble control room overhead annunciator to alarm under the following circumstances:

(a) loss of external or internal power to the NUMAC;

(b) placing the keylock switch out of operate;

(c) failure of a hardware module during self-test diagnostics (performed approximately each 1-2 minutes in the Leak Detection Monitor);

(d) detection of an open thermocouple or flow transmitter signal circuit;

(e) failure of the class-1E processor to update the hardware watchdog on regular intervals; and,

(f) any software task which is not running at its expected intervals.

In the event of command-mode failure of the NUMAC LDM, the operator can perform the LDM's actuation functions manually in the main control room (MCR) using the information available from the radiation indicators and alarms in the MCR. This is diverse from the LDM.

Based on the staff's review of the LDM's defense against the common mode failures or common cause failures, the staff finds that the LDM's defenses are adequate.

2.5 Training

Training is an important part of implementing the NUMAC LDM to the BSEP environment. The operators and technicians need to obtain a sufficient understanding by training on, using, and repairing the new system.

The licensee is planning to provide specific training for operations and maintenance personnel. The selected maintenance staff personnel will receive either Level 1 (1-2 hour) or Level 2 (½ to 1 day) courses on changes to the leak detection system, including changes to the physical configuration, technical specifications, manuals and drawings, and surveillance procedures. Level 2 will include in-depth training on hardware, software, testing and diagnostics. Both levels will include hands-on experience in the NUMAC training unit.

For the operations staff, all supervisors and on-shift control operators will receive an overview of the changes to the leak detection system, including changes to the physical configuration, technical specifications, manuals and drawings, and surveillance procedures. They will also receive training in the changes to LCO criteria and operational procedures. This training will also include hands-on experience in the NUMAC training unit.

The staff finds the licensee's approach to this issue to be acceptable.

2.6 Technical Specification Changes

2.6.1 Change (1)

The current requirement in TS Table 4.3.2-1, Isolation Actuation Instrumentation Surveillance Requirements, Item 3.a, Reactor Water Cleanup System Isolation, Differential Flow - High, currently specifies the performance of a channel check on a D (daily) frequency. The licensee proposed changing the surveillance frequency to NA (not applicable). The basis for this request is that the NUMAC system performs a continuous self-test and alerts the operator via annunciation when a problem is detected. These diagnostic and self-test features are provided below:

a. Continuous monitoring of each flow and each density compensation input signal for out-of-bounds values

b. Continuous monitoring of the two internal power supplies (NUMAC remains functional with only one internal power supply)

c. Continuous monitoring of the external power input

d. A self-check of each channel to confirm functionality at least once per 30 minutes

e. Continuous monitoring to assure that the system is not left in an inoperable condition (card out-of-file, status switch left in the inop mode)

The staff recognizes that, based upon the NUMAC system's self-diagnostic features, the D (daily) surveillance frequency for the RWCU system differential flow - high containment isolation actuation instrument channel check is no longer needed.

2.6.2 Change (2)

The licensee also proposed changing TS Table 4.3.2-1, Isolation Actuation Instrumentation Surveillance Requirements, to reduce the frequency of the tests required by this table as follows:

a. Reactor Water Cleanup System

Item 3.a, Differential Flow - High: revise the channel functional test from M (Monthly) to SA (semi-annual) frequency

Item 3.b, Area Temperature - High: revise the channel functional test from M to SA frequency

Item 3.c, Area Ventilation Δ Temperature - High: revise the channel functional test from M to SA frequency

Item 3.f, Differential Flow - High - Time Delay Relay): revise the channel functional test from M to SA frequency.

b. High Pressure Coolant Injection System Isolation

Item 4.a.4, HPCI Steam Line Tunnel Temperature - High: revise the channel functional test from M to SA frequency and the channel calibration from Q (quarterly) to R (refuel) frequency.

Item 4.a.7, HPCI Steam Line Ambient Temperature - High: revise the channel functional test from M to SA frequency.

Item 4.a.8, HPCI Steam Line Area Differential Temperature - High: revise the channel functional test from M to SA frequency.

Item 4.a.9, HPCI Equipment Area Temperature - High: revise the channel functional test from M to SA frequency and the channel calibration from Q to R frequency.

c. Reactor Core Isolation Cooling System Isolation

Item 4.b.4, RCIC Steam Line Tunnel Temperature - High: revise the channel functional test from M to SA frequency.

Item 4.b.7, RCIC Steam Line Ambient Temperature - High: revise the channel functional test from M to SA frequency.

Item 4.b.8, RCIC Steam Line Area Differential Temperature - High: revise the channel functional test from M to SA frequency.

Item 4.b.9, RCIC Equipment Room Ambient Temperature - High: revise the channel functional test from M to SA frequency and the channel calibration from Q to R frequency.

Item 4.b.10, RCIC Equipment Room Differential Temperature - High: revise the channel functional test from M to SA frequency and the channel calibration from Q to R frequency.

Item 4.b.11, RCIC Steam Line Tunnel Temperature - High Time Delay Relay: revise the channel functional test from M to SA frequency.

The basis for the increase in the channel functional test surveillance interval from M (monthly) to SA (semi-annual) is that the self-test feature can detect the potential failures that these channel functional tests are intended to identify. The basis for increasing the channel calibration surveillance interval from Q (quarterly) to R (refueling) for four of the temperature monitoring TS Trip Functions is the high degree of stability, permitting longer intervals between calibrations without significant instrument drift. The staff finds these conclusions to be acceptable.

2.6.3 Change (3)

The current requirement, in TS Table 3.3.2-2, Isolation Actuation Instrumentation Setpoints, Item 3.f, Reactor Water Cleanup System Isolation, Differential Flow - High - Time Delay Relay, specifies a trip setpoint and allowable value of less than or equal to 45 seconds. The requested change to less than or equal to 30 minutes is based on the difference between the input flow from the reactor coolant system and the sum of the two output flows (one is return to the feedwater system; the other is reject flow to the main condenser or the radwaste system). The difference between the input and output flows is assumed to be leakage. Since the NUMAC system will incorporate the enhanced feature of process flow density compensation based on process temperature, this will provide the operator more accurate RWCU system leak rate information. The design basis for the RWCU system high differential flow isolation function is to assure compliance with 10 CFR Part 100 and 10 CFR Part 20. The high differential flow isolation function is not intended for protection of reactor pressure vessel water level or for limiting the reactor building environment for equipment qualification purposes. Current RWCU system high energy line break scenarios rely on ambient temperature detection as the isolation initiation signal.

General Electric Company report, GE-NE-770-14-0592 (Proprietary) evaluates the consequences of a 300 gal/min RWCU cold leak remaining unisolated for 30 minutes. This document provides the basis for the establishment of 30 minutes as the allowable value/trip setpoint limit for this differential flow isolation time delay function. The licensee's calculations determined that doses for the control room, site boundary and low population zones were within the limits of GDC 19 and 10 CFR Part 100. The staff performed independent calculations. The staff utilized a reactor coolant concentration of 4 uCi/gm as the concentration for the release. This value was chosen because the coolant concentration could be this high before technical specifications require the unit to be shut down. Doses were calculated based upon this coolant concentration, a partition factor of 0.1 and a 300 gal/min release rate for 30 minute. Thyroid doses of 11.9, 5.9, and 1.2 rem were calculated for the control room operators, members of the public at the site boundary and the low population zone, respectively. These doses are acceptable and demonstrate that changing the RWCU system isolation differential flow time delay trip setpoint and allowable value from less than or equal to 45 seconds to less than or equal to 30 minutes is acceptable from a radiological dose aspect.

The licensee stated that plant sump monitoring instrumentation, room flood alarms and plant leakage response procedures preclude the possibility that adverse room flooding conditions could result from the increased time delay on the automatic RWCU system isolation.

The staff has determined that this analysis is satisfactory.

2.6.4 Change (4)

The licensee's calculation ORWCU-0010 defines the magnitude of the uncertainty associated with the reactor water cleanup system isolation differential flow trip function setpoint. The uncertainties are characterized as either "measurable" or "unmeasurable." The measurable uncertainties are those attributable to effects that may be present during surveillance testing. The unmeasurable uncertainties are those related to effects that will not be present during surveillance testing (e.g., flow orifice effects, seismic events, post-accident environmental conditions).

The requested increase in the TS allowable value is intended to establish a difference between the actual field calibration setpoint and the new allowable value that is large enough to bound the sum of the measurable uncertainties present during surveillance testing conditions and a nominal additional "LER avoidance" margin. The licensee's calculation demonstrates that satisfaction of the proposed allowable value during surveillance testing will assure that the 300 gal/min analytical limit will not be exceeded during any postulated plant events.

The staff has reviewed the licensee's submittal as discussed above and finds the proposed changes to the RWCU system high differential flow, time-delay trip setpoint and allowable value will have no adverse impact on safety and will not pose an undue risk to the public because the calculated offsite and control room doses continue to be less than the limits of 10 CFR Part 100 and GDC 19.

The staff has determined that this analysis is satisfactory.

2.6.5 Changes (5) and (6)

The response time requirements for containment isolation instrumentation were based on the need to limit damage to safety-related equipment or to mitigate the consequences of an accident or equipment failure. The sensing devices for the HPCI system isolation steam line tunnel temperature - high trip function channel are thermocouples. Thermocouples, being inherently stable devices with little potential for shift in response time, can be expected to either respond within manufacturer's time constant specifications or fail completely. A break in the HPCI system steam supply line will always result in a predictable response from the sensing thermocouples, regardless of how long the thermocouple has been installed (i.e., the response time of a thermocouple is constant). Therefore, the response time of a thermocouple need not be routinely measured.

Isolation of the HPCI system is required in the event of a process high energy line break and has been analyzed in the CP&L Report No. 9527-058-S-MS-001, "Reactor Building Environmental Report". The isolation trip signal initiators considered in that analysis included both temperature channels and high flow sensors. The high flow trips were used in the analysis to mitigate the large (greater than or equal to 300 percent flow) break and the temperature channels were used to mitigate smaller breaks. The high temperature isolation trip function is capable of providing a timely response to either size break.

The staff has determined that this analysis is satisfactory.

2.6.6 Change (7)

The intent of response time requirements for containment isolation instrumentation is to monitor subtle performance changes in trip channels whose prompt response is taken credit for in event detection and mitigation. The analysis reviewed in Section 5.3 and 5.4 above show that rapid response time for the differential flow trip channel is not critical.

The involved differential pressure transmitters, thermocouples and NUMAC equipment are subject to very little change in response time relative to the 30 minute setting proposed for the isolation time delay associated with that function. Any subtle response time changes that might occur would be insignificant relative margins present in the model and the results of GE report, GE-NE-770-14-0592.

The current TS identifies the existing 45 second differential flow delay time as both a response time requirement (Table 3.3.2-3, Item 3.a) and as a setpoint (Table 3.3.2-2, Item 3.f). This duplication is unnecessary and the 30 minute differential flow delay time parameter would best be treated as a setpoint, rather than as a response time. The surveillance testing of this parameter when treated as a setpoint will continue to ensure that the trip channel performance is adequate to satisfy the performance assumptions utilized in the GE leakage analysis report.

The staff finds this change to be acceptable.

2.6.7 Change (8)

In the existing configuration of the differential flow trip function, the time delay function is performed by actual time delay relays. Within the replacement NUMAC system, the time delay function is performed by software within the NUMAC microprocessor. This item description change is necessary to avoid potential confusion in future application and interpretation of this Technical Specification line item.

The staff finds this change to be acceptable.

2.6.8 Change (9)

The licensee proposed adding a new RWCU system isolation actuation instrumentation function covering piping outside of the RWCU system room to Technical Specification Tables 3.3.2-1, 3.3.2-2, 3.3.2-3, and 4.3.2-1. Two area high temperature instrumentation channels for initiating RWCU system isolation were installed and declared operable on February 13, 1991 (Unit 1), and August 10, 1990 (Unit 2). The design and function of these leak detection instruments is similar to the existing leak detection temperature monitoring instruments currently identified in the TS therefore, response times and surveillance frequencies that are the same as those established for existing leak detection temperature monitoring instrumentation are being proposed.

The licensee indicated that the existing setpoint for RWCU high differential flow time delay setpoint of less than or equal to 45 seconds has led to many unwarranted system isolations of the RWCU system and has been a chronic problem for operations personnel during initial system fill, pressurization and startup. The RWCU high differential flow alarm and trip are based on the difference between the input flow from the reactor coolant system and the sum of two output flows (to the feedwater system and the reject flow to the main condenser or the radwaste system). Density compensation is not provided to account for varying process water temperatures in the existing flow leak detection instrumentation. The proposed NUMAC system will incorporate the enhanced features of process flow density compensation based on process temperature. This will provide the operator more accurate RWCU system leak rate information.

The licensee stated that the sole design basis of the RWCU system high differential flow isolation function is to assure compliance with 10 CFR 100 and 10 CFR 20. The differential flow isolation function is not intended for protection of reactor vessel water level or for reactor building equipment qualification purposes. Current RWCU high energy line break scenarios rely on ambient temperature detection as the isolation initiation signal.

The licensee indicated that GE has evaluated the consequences of a 300 gal/min RWCU system cold water leak remaining unisolated for 30 minutes in its report GE-NE-770-14-0592. The licensee's calculations determined that the control room, site boundary and low population doses were within the limits of GDC 19 and 10 CFR Part 100. The staff performed independent calculations. The staff utilized a reactor coolant concentration of 4 uCi/gm as the concentration for the release. This value was chosen because the coolant concentration could be this high before technical specifications require the unit to be shut down. Based upon this coolant concentration, a partition factor of 0.1 and a 300 gallon/min release rate for 30 minutes, doses were calculated. Thyroid doses 11.9, 5.9, and 1.2 rem, were calculated for the control room operators and members of the public at the site boundary and the low population zone, respectively. These doses are acceptable and demonstrate that changing the

RWCU system are acceptable and demonstrate that changing the RWCU system isolation differential flow time delay trip setpoint and allowable value from less than or equal to 45 seconds to less than or equal to 30 minutes is acceptable from a radiological dose aspect.

The licensee stated that plant sump monitoring instrumentation, room flood alarms and plant leakage response procedures preclude the possibility that adverse room flooding conditions could result from the increased time delay on the automatic RWCU system isolation.

The requested increase in the differential flow trip setpoint and allowable value for RWCU system isolation from less than or equal to 53 gal/min to less than or equal to 73 gal/min is intended to establish a difference between the actual field calibration setpoint and the new allowable value that is large enough to bound the sum of uncertainties present during surveillance testing conditions and a nominal additional "LER avoidance margin". The proposed limit will assure that the 300 gal/min analytical limit will not be exceeded during any postulated plant events.

The staff has reviewed the licensee submittal as discussed above and finds the proposed changes to the RWCU system high differential flow and time-delay trip setpoint and allowable value will have no adverse impact on safety and will not pose an undue risk to the public because the calculated offsite and control room doses continue to be less than the limits of 10 CFR Part 100 and GDC 19.

The licensee indicated that the design and function of the new area high temperature instrumentation for initiating RWCU isolation is similar to the existing leak detection temperature monitoring instruments currently identified in the TS. The response time and surveillance frequencies that are being proposed are also the same as those established for existing leak detection temperature instrumentation.

The staff finds the addition of a new RWCU system isolation actuation function covering piping outside the RWCU as discussed above acceptable.

2.7 Summary

Based on the evaluations discussed above, the staff finds the equipment modifications, analyses, and supporting changes in the facility technical specifications to be acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the State of North Carolina official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 52.35, an environmental assessment and finding of no significant impact have been prepared and published in the Federal Register on August 30, 1993 (58 FR 45535). Accordingly, based upon the environmental assessment, the staff has determined that the issuance of the amendments will not have a significant effect on the quality of the human environment.

5.0 CONCLUSION

The Commission 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, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: P. Loeser
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Date: October 14, 1993

UNITED STATES NUCLEAR REGULATORY COMMISSION

CAROLINA POWER & LIGHT COMPANY

DOCKET NOS. 50-325 AND 50-324

NOTICE OF ISSUANCE OF AMENDMENT TO

FACILITY OPERATING LICENSE

The U.S. Nuclear Regulatory Commission (Commission) has issued Amendment Nos. 166 and 197 to Facility Operating License Nos. DPR-71 and DPR-62, respectively, issued to Carolina Power & Light Company (the licensee) that revised the Technical Specifications for operation of the Brunswick Steam Electric Plant, Units 1 and 2, located in Brunswick County, North Carolina. Amendment No. 166 for Unit 1 is effective as of the date of issuance, and Amendment No. 197 for Unit 2 will be effective upon completion of Refueling Outage No. 10.

The amendments revise the Technical Specifications to allow the replacement of existing Riley, GEMAC and Fenwal steam leak detection equipment with General Electric Company NUMAC leak detection equipment. The proposed amendment also revises surveillance requirements for steam leak detection instrumentation associated with the reactor core isolation cooling system, the high pressure coolant injection system, and the reactor water cleanup system. The specific changes include:

(1) Delete the channel check surveillance test for the reactor water cleanup system isolation high differential flow function.

(2) Extend and standardize the channel functional test and channel calibration surveillance frequencies for the reactor water cleanup, high pressure coolant injection, and reactor core

isolation cooling system isolation ambient and differential temperature functions.

(3) Increase the reactor water cleanup system isolation differential flow time delay trip setpoint and allowable value from "less than or equal to 45 seconds" to "less than or equal to 30 minutes."

(4) Increase the reactor water cleanup system isolation differential flow trip setpoint and allowable value from "less than or equal to 53 gal/min" to "less than or equal to 73 gal/min."

(5) Delete the instrument response time requirement for the high pressure coolant injection system isolation steam line tunnel temperature - high function.

(6) Delete the instrument response time requirement for the reactor water cleanup system isolation area temperature - high and area ventilation differential temperature - high functions.

(7) Delete the instrument response time requirement for the reactor water cleanup system isolation differential flow - high function.

(8) Revise the description of the reactor water cleanup isolation differential flow delay trip function to reflect elimination of the time delay relays per the new system configuration.

(9) Add a new reactor water cleanup system isolation area temperature function for piping outside of the reactor water cleanup room.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

Notice of Consideration of Issuance of Amendment and Opportunity for Hearing in connection with this action was published in the FEDERAL REGISTER on November 24, 1992 (57 FR 55287). No request for a hearing or petition for leave to intervene was filed following this notice.

The Commission has prepared an Environmental Assessment related to the action and has determined not to prepare an environmental impact statement. Based upon the environmental assessment, the Commission has concluded that the issuance of this amendment will not have a significant effect on the quality of the human environment (58 FR 45535).

For further details with respect to the action see (1) the application for amendment dated September 14, 1992, as supplemented January 13, January 25, February 8, May 11, June 18, July 26, and September 21, 1993; (2) Amendment No. 166 to License No. DPR-71 and Amendment No. 197 to License No. DPR-62; (3) the Commission's related Safety Evaluation and Environmental Assessment. All of these items are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street NW., Washington, DC 20555, and at the local public document room located at the University of North Carolina at Wilmington, William Madison Randall Library, 601 S. College Road, Wilmington, North Carolina 28403-3297.

Dated at Rockville, Maryland, this 14th day of October 1993.

FOR THE NUCLEAR REGULATORY COMMISSION

Patrick D. Milano, Project Manager
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