March 31, 1995

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

SUBJECT: ISSUANCE OF AMENDMENT NO. 176 TO FACILITY OPERATING LICENSE NO. DPR-71 AND AMENDMENT NO. 207 TO FACILITY OPERATING LICENSE NO. DPR-62 REGARDING INCREASES IN SURVEILLANCE TEST INTERVALS AND ALLOWABLE OUT-OF-SERVICE TIMES FOR SELECTED INSTRUMENTATION - BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2 (TAC NOS. M90794 AND M90795)

Dear Mr. Anderson:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 176 to Facility Operating License No. DPR-71 and Amendment No. 207 to Facility Operating License No. DPR-62 for the Brunswick Steam Electric Plant, Units 1 and 2. The amendments change the Technical Specifications (TS) in response to your submittal dated September 30, 1994, as supplemented on March 24, 1995.

The amendments eliminate the reactor scram and isolation functions of the main steam line radiation monitors.

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)

David C. Trimble, Project Manager
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-325
and 50-324

Enclosures:
1. Amendment No. 176 to License No. DPR-71
2. Amendment No. 207 to License No. DPR-62
3. Safety Evaluation

cc w/enclosures: See next page
DOCUMENT NAME: G:\BRUNSWICK\BR90573.AMD

OFFICE	LA:PDII-1	PM:PDII-1	D:PDII-1	OGC	
NAME	PAnderson	DTrimble	WBateman	R Bachmann	
DATE	03/27/95	03/27/95	03/31/95	03/30/95	
COPY	Yes/No	Yes/No	Yes/No	Yes/No	

OFFICIAL RECORD COPY

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PDR ADOCK 05000324
P PDR

cc:

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AMENDMENT NO. 176 TO FACILITY OPERATING LICENSE NO. DPR-71 - BRUNSWICK, UNIT 1
AMENDMENT NO. 207 TO FACILITY OPERATING LICENSE NO. DPR-62 - BRUNSWICK, UNIT 2

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060037

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. 176
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 30, 1994, as supplemented on March 24, 1995, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications, as indicated in the attachment to this license amendment; and paragraph 2.C.(2) of Facility Operating License No. DPR-71 is hereby amended to read as follows:

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P PDR

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 176, 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

Brenda Mozafari for

William H. Bateman, 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: March 31, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 176

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
2-4	2-4
2-5	2-5
B 2-6	B 2-6
3/4 3-2	3/4 3-2
3/4 3-4	3/4 3-4
3/4 3-5	3/4 3-5
3/4 3-8	3/4 3-8
3/4 3-9	3/4 3-9
3/4 3-12	3/4 3-12
3/4 3-18	3/4 3-18
3/4 3-22	3/4 3-22
3/4 3-27	3/4 3-27
3/4 3-32	3/4 3-32

TABLE 2.2.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Intermediate Range Monitor, Neutron Flux - High[a]	≤ 120 divisions of full scale	≤ 120 divisions of full scale
2. Average Power Range Monitor		
a. Neutron Flux - High, 15%[b]	≤ 15% of RATED THERMAL POWER	≤ 15% of RATED THERMAL POWER
b. Flow-Biased Simulated Thermal Power - High[c][d]	≤ (0.66W + 64%) with a maximum ≤ 113.5% of RATED THERMAL POWER	≤ (0.66W + 67%) with a maximum ≤ 115.5% of RATED THERMAL POWER
c. Fixed Neutron Flux - High[d]	≤ 120% of RATED THERMAL POWER	≤ 120% of RATED THERMAL POWER
3. Reactor Vessel Steam Dome Pressure - High	≤ 1045 psig	≤ 1045 psig
4. Reactor Vessel Water Level - Low, Level 1	≥ +162.5 inches[g]	≥ +162.5 inches[g]
5. Main Steam Line Isolation Valve - Closure[e]	≤ 10% closed	≤ 10% closed
6. (Deleted)		
7. Drywell Pressure - High	≤ 2 psig	≤ 2 psig
8. Scram Discharge Volume Water Level - High	≤ 109 gallons	≤ 109 gallons
9. Turbine Stop Valve - Closure[f]	≤ 10% closed	≤ 10% closed
10. Turbine Control Valve Fast Closure, Control Oil Pressure - Low[f]	≥ 500 psig	≥ 500 psig

TABLE 2.2.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

NOTES

(a) The Intermediate Range Monitor scram functions are automatically bypassed when the reactor mode switch is placed in the Run position and the Average Power Range Monitors are on scale.

(b) This Average Power Range Monitor scram function is a fixed point and is increased when the reactor mode switch is placed in the Run position.

(c) The Average Power Range Monitor scram function is varied, Figure 2.2.1-1, as a function of the fraction of rated recirculation loop flow (W) in percent.

(d) The APRM flow-biased simulated thermal power signal is fed through a time constant circuit of approximately 6 seconds. The APRM fixed high neutron flux signal does not incorporate the time constant, but responds directly to instantaneous neutron flux.

(e) The Main Steam Line Isolation Valve-Closure scram function is automatically bypassed when the reactor mode switch is in other than the Run position.

(f) These scram functions are bypassed when THERMAL POWER is less than 30% of RATED THERMAL POWER as measured by turbine first stage pressure.

(g) Vessel water levels refer to REFERENCE LEVEL ZERO.

4. Reactor Vessel Water Level-Low, Level #1

The reactor water level trip point was chosen far enough below the normal operating level to avoid spurious scrams but high enough above the fuel to assure that there is adequate water to account for evaporation losses and displacement of cooling following the most severe transients. This setting was also used to develop the thermal-hydraulic limits of power versus flow.

5. Main Steam Line Isolation Valve-Closure

The low-pressure isolation of the main steamline trip was provided to give protection against rapid depressurization and resulting cooldown of the reactor vessel. Advantage was taken of the shutdown feature in the run mode which occurs when the main steam line isolation valves are closed, to provide for reactor shutdown so that high power operation at low pressures does not occur. Thus, the combination of the low-pressure isolation and isolation valve closure reactor trip with the mode switch in the Run position assures the availability of neutron flux protection over the entire range of the Safety Limits. In addition, the isolation valve closure trip with the mode switch in the Run position anticipates the pressure and flux transients which occur during normal or inadvertent isolation valve closure.

6. (Deleted)

7. Drywell Pressure, High

High pressure in the drywell could indicate a break in the nuclear process systems. The reactor is tripped in order to minimize the possibility of fuel damage and reduce the amount of energy being added to the coolant. The trip setting was selected as low as possible without causing spurious trips.

TABLE 3.3.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	APPLICABLE OPERATIONAL CONDITIONS	MINIMUM NUMBER OPERABLE CHANNELS PER TRIP SYSTEM (a)	ACTION
1. Intermediate Range Monitors:			
a. Neutron Flux - High	2, 5[b] 3, 4	3 2	1 2
b. Inoperative	2, 5 3, 4	3 2	1 2
2. Average Power Range Monitor			
a. Neutron Flux - High, 15%	2, 5[b]	2	3
b. Flow Biased Simulated Thermal Power - High	1	2	4
c. Fixed Neutron Flux - High, 120%	1	2	4
d. Inoperative	1, 2, 5	2	5
e. Downscale	1	2	4
f. LPRM	1, 2, 5	(c)	NA
3. Reactor Vessel Steam Dome Pressure - High	1, 2[d]	2	6
4. Reactor Vessel Water Level - Low, Level 1	1, 2	2	6
5. Main Steam Isolation Valve - Closure	1	4	4
6. (Deleted)			

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

ACTIONS

ACTION 1 - In OPERATIONAL CONDITION 2, be in at least HOT SHUTDOWN within 6 hours.

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

ACTION 2 - Lock the reactor mode switch in the Shutdown position within one hour.

ACTION 3 - In OPERATIONAL CONDITION 2, be in at least HOT SHUTDOWN within 6 hours.

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

ACTION 4 - Be in at least STARTUP within 2 hours.

ACTION 5 - In OPERATIONAL CONDITION 1 or 2, be in at least HOT SHUTDOWN within 6 hours.

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

ACTION 6 - Be in at least HOT SHUTDOWN within 6 hours.

ACTION 7 - (Deleted)

ACTION 8 - Initiate a reduction in THERMAL POWER within 15 minutes and be at less than 30% of RATED THERMAL POWER within 2 hours.

ACTION 9 - In OPERATIONAL CONDITION 1 or 2, be in at least HOT SHUTDOWN within 6 hours.

In OPERATIONAL CONDITION 3 or 4, immediately and at least once per 12 hours verify that all control rods are fully inserted.

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

ACTION 10 - In OPERATIONAL CONDITION 1 or 2, be in at least HOT SHUTDOWN within 6 hours.

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

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

NOTES

(a) When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated ACTIONs may be delayed for up to 6 hours provided the Functional Unit maintains RPS trip capability.

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

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

(d) This function is not required to be OPERABLE when the reactor pressure vessel head is unbolted or removed.

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

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

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

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

TABLE 4.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[a]	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
5. Main Steam Line Isolation Valve - Closure	NA	Q	R[h]	1
6. (Deleted)				
7. Drywell Pressure - High				
Transmitter:	NA[k]	NA	R[l]	1, 2
Trip Logic:	D	Q	Q	1, 2
8. Scram Discharge Volume Water Level - High	NA	Q	R	1, 2, 5
9. Turbine Stop Valve - Closure	NA	Q	R[h]	1[o]
10. Turbine Control Valve Fast Closure, Control Oil Pressure - Low	NA	Q	R	1[o]
11. Reactor Mode Switch in Shutdown Position	NA	R	NA	1, 2, 3, 4, 5
12. Manual Scram	NA	Q	NA	1, 2, 3, 4, 5
13. Automatic Scram Contactors	NA	W	NA	1, 2, 3, 4, 5

TABLE 4.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

NOTES

(a) Neutron detectors may be excluded from CHANNEL CALIBRATION.

(b) Within 24 hours prior to startup, if not performed within the previous 7 days.

(c) The IRM channels shall be compared to the APRM channels and the SRM instruments for overlap during each startup, if not performed within the previous 7 days.

(d) When changing from OPERATIONAL CONDITION 1 to OPERATIONAL CONDITION 2, perform the required surveillance within 12 hours after entering OPERATIONAL CONDITION 2, if not performed within the previous 7 days.

(e) This calibration shall consist of the adjustment of the APRM readout to conform to the power values calculated by a heat balance during OPERATIONAL CONDITION 1 when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

(f) This calibration shall consist of the adjustment of the APRM flow-biased simulated thermal power channel to conform to a calibrated flow signal.

(g) The LPRMs shall be calibrated at least once per effective full power month (EFPM) using the TIP system.

(h) This calibration shall consist of a physical inspection and actuation of these position switches.

(i) (Deleted)

(j) (Deleted)

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

(l) Transmitters are exempted from the quarterly channel calibration.

(m) Placement of Reactor Mode Switch into the Startup/Hot Standby position is permitted for the purpose of performing the required surveillance prior to withdrawal of control rods for the purpose of bringing the reactor to criticality.

(n) Placement of Reactor Mode Switch into the Shutdown or Refuel position is permitted for the purpose of performing the required surveillance provided all control rods are fully inserted and the vessel head bolts are tensioned.

(o) Surveillance is not required when THERMAL POWER is less than 30% of RATED THERMAL POWER.

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. (Deleted)				
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	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 3.3.2-2

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. (Deleted)		
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
5. SHUTDOWN COOLING SYSTEM ISOLATION		
a. Reactor Vessel Water Level - Low Level 1	≥ 162.5 inches[a]	≥ 162.5 inches[a]
b. Reactor Steam Dome Pressure - High	≤ 140 psig	≤ 140 psig

(a) Vessel water levels refer to REFERENCE LEVEL ZERO.

(b) Establish alarm/trip setpoints per the methodology contained in the OFFSITE DOSE CALCULATION MANUAL (ODCM).

(c) (Deleted)

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	Q	Q	1, 2, 3
2. Low, Level 3				
Transmitter:	NA[a]	NA	R[b]	1, 2, 3
Trip Logic:	D	Q	Q	1, 2, 3
b. Drywell Pressure - High				
Transmitter:	NA[a]	NA	R[b]	1, 2, 3
Trip Logic:	D	Q	Q	1, 2, 3
c. Main Steam Line				
1. (Deleted)				
2. Pressure - Low				
Transmitter:	NA[a]	NA	R[b]	1
Trip Logic:	D	Q	Q	1
3. Flow - High				
Transmitter:	NA[a]	NA	R[b]	1
Trip Logic:	D	Q	Q	1
d. Main Steam Line Tunnel Temperature - High	NA	Q	R	1, 2, 3
e. Condenser Vacuum - Low				
Transmitter:	NA[a]	NA	R[b]	1, 2[e]
Trip Logic:	D	Q	Q	1, 2[e]
f. Turbine Building Area Temperature - High	NA	Q	R	1, 2, 3
g. Main Stack Radiation - High	NA	Q	R	1, 2, 3
h. Reactor Building Exhaust Radiation - High	D	Q	R	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 quarterly channel calibration.

(c) Deleted.

(d) Deleted.

(e) When reactor steam pressure ≥ 500 psig.

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

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.207
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 30, 1994, as supplemented on March 24, 1995, 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. 207 , 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

Brenda Mozafari for

William H. Bateman, 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: March 31, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 207

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
2-4	2-4
2-5	2-5
B 2-6	B 2-6
3/4 3-2	3/4 3-2
3/4 3-4	3/4 3-4
3/4 3-5	3/4 3-5
3/4 3-8	3/4 3-8
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3/4 3-27	3/4 3-27
3/4 3-32	3/4 3-32

TABLE 2.2.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Intermediate Range Monitor, Neutron Flux - High [a]	≤ 120 divisions of full scale	≤ 120 divisions of full scale
2. Average Power Range Monitor		
a. Neutron Flux - High, 15%[b]	≤ 15% of RATED THERMAL POWER	≤ 15% of RATED THERMAL POWER
b. Flow Biased Simulated Thermal Power - High [c][d]	≤ (0.66 W + 64%) with a maximum ≤ 113.5% of RATED THERMAL POWER	≤ (0.66 W + 67%) with a maximum ≤ 115.5% of RATED THERMAL POWER
c. Fixed Neutron Flux - High[d]	≤ 120% of RATED THERMAL POWER	≤ 120% of RATED THERMAL POWER
3. Reactor Vessel Steam Dome Pressure - High	≤ 1045 psig	≤ 1045 psig
4. Reactor Vessel Water Level - Low, Level 1	≥ +162.5 inches[g]	≥ +162.5 inches[g]
5. Main Steam Line Isolation Valve - Closure[e]	≤ 10% closed	≤ 10% closed
6. (Deleted)		
7. Drywell Pressure - High	≤ 2 psig	≤ 2 psig
8. Scram Discharge Volume Water Level - High	≤ 109 gallons	≤ 109 gallons
9. Turbine Stop Valve-Closure[f]	≤ 10% closed	≤ 10% closed
10. Turbine Control Valve Fast, Closure, Control Oil Pressure-Low[f]	≥ 500 psig	≥ 500 psig

TABLE 2.2.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

NOTES

(a) The Intermediate Range Monitor scram functions are automatically bypassed when the reactor mode switch is placed in the Run position and the Average Power Range Monitors are on scale.

(b) This Average Power Range Monitor scram function is a fixed point and is increased when the reactor mode switch is placed in the Run position.

(c) The Average Power Range Monitor scram function is varied, Figure 2.2.1-1, as a function of the fraction of rated recirculation loop flow (W) in percent.

(d) The APRM flow-biased simulated thermal power signal is fed through a time constant circuit of approximately 6 seconds. The APRM fixed high neutron flux signal does not incorporate the time constant, but responds directly to instantaneous neutron flux.

(e) The Main Steam Line Isolation Valve-Closure scram function is automatically bypassed when the reactor mode switch is in other than the Run position.

(f) These scram functions are bypassed when THERMAL POWER is less than 30% of RATED THERMAL POWER as measured by turbine first stage pressure.

(g) Vessel water levels refer to REFERENCE LEVEL ZERO.

4. Reactor Vessel Water Level-Low, Level #1

The reactor water level trip point was chosen far enough below the normal operating level to avoid spurious scrams but high enough above the fuel to assure that there is adequate water to account for evaporation losses and displacement of cooling following the most severe transients. This setting was also used to develop the thermal-hydraulic limits of power versus flow.

5. Main Steam Line Isolation Valve-Closure

The low-pressure isolation of the main steam line trip was provided to give protection against rapid depressurization and resulting cooldown of the reactor vessel. Advantage was taken of the shutdown feature in the run mode which occurs when the main steam line isolation valves are closed, to provide for reactor shutdown so that high power operation at low pressures does not occur. Thus, the combination of the low-pressure isolation and isolation valve closure reactor trip with the mode switch in the Run position assures the availability of neutron flux protection over the entire range of the Safety Limits. In addition, the isolation valve closure trip with the mode switch in the Run position anticipates the pressure and flux transients which occur during normal or inadvertent isolation valve closure.

6. (Deleted)

TABLE 3.3.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	APPLICABLE OPERATIONAL CONDITIONS	MINIMUM NUMBER OPERABLE CHANNELS PER TRIP SYSTEM (a)	ACTION
1. Intermediate Range Monitors:			
a. Neutron Flux - High	2, 5[b] 3, 4	3 2	1 2
b. Inoperative	2, 5 3, 4	3 2	1 2
2. Average Power Range Monitor			
a. Neutron Flux - High, 15%	2, 5[b]	2	3
b. Flow Biased Simulated Thermal Power - High	1	2	4
c. Fixed Neutron Flux - High, 120%	1	2	4
d. Inoperative	1, 2, 5	2	5
e. Downscale	1	2	4
f. LPRM	1, 2, 5	(c)	NA
3. Reactor Vessel Steam Dome Pressure - High	1, 2[d]	2	6
4. Reactor Vessel Water Level - Low, Level 1	1, 2	2	6
5. Main Steam Isolation Valve - Closure	1	4	4
6. (Deleted)			

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

ACTIONS

ACTION 1 - In OPERATIONAL CONDITION 2, be in at least HOT SHUTDOWN within 6 hours.

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

ACTION 2 - Lock the reactor mode switch in the Shutdown position within one hour.

ACTION 3 - In OPERATIONAL CONDITION 2, be in at least HOT SHUTDOWN within 6 hours.

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

ACTION 4 - Be in at least STARTUP within 2 hours.

ACTION 5 - In OPERATIONAL CONDITION 1 or 2, be in at least HOT SHUTDOWN within 6 hours.

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

ACTION 6 - Be in at least HOT SHUTDOWN within 6 hours.

ACTION 7 - (Deleted)

ACTION 8 - Initiate a reduction in THERMAL POWER within 15 minutes and be at less than 30% of RATED THERMAL POWER within 2 hours.

ACTION 9 - In OPERATIONAL CONDITION 1 or 2, be in at least HOT SHUTDOWN within 6 hours.

In OPERATIONAL CONDITION 3 or 4, immediately and at least once per 12 hours verify that all control rods are fully inserted.

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

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

ACTION 10 - In OPERATIONAL CONDITION 1 or 2, be in at least HOT SHUTDOWN within 6 hours.

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

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

NOTES

(a) When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated ACTIONs may be delayed for up to 6 hours provided the Functional Unit maintains RPS trip capability.

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

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

(d) This function is not required to be OPERABLE when the reactor pressure vessel head is unbolted or removed.

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

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

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

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

TABLE 4.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[a]	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
5. Main Steam Line Isolation Valve - Closure	NA	Q	R[h]	1
6. (Deleted)				
7. Drywell Pressure - High				
Transmitter:	NA[k]	NA	R[1]	1, 2
Trip Logic:	D	Q	Q	1, 2
8. Scram Discharge Volume Water Level - High	NA	Q	R	1, 2, 5
9. Turbine Stop Valve - Closure	NA	Q	R[h]	1[o]
10. Turbine Control Valve Fast Closure, Control Oil Pressure - Low	NA	Q	R	1[o]
11. Reactor Mode Switch in Shutdown Position	NA	R	NA	1, 2, 3, 4, 5
12. Manual Scram	NA	Q	NA	1, 2, 3, 4, 5
13. Automatic Scram Contactors	NA	W	NA	1, 2, 3, 4, 5

TABLE 4.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

NOTES

(a) Neutron detectors may be excluded from CHANNEL CALIBRATION.

(b) Within 24 hours prior to startup, if not performed within the previous 7 days.

(c) The IRM channels shall be compared to the APRM channels and the SRM instruments for overlap during each startup, if not performed within the previous 7 days.

(d) When changing from OPERATIONAL CONDITION 1 to OPERATIONAL CONDITION 2, perform the required surveillance within 12 hours after entering OPERATIONAL CONDITION 2, if not performed within the previous 7 days.

(e) This calibration shall consist of the adjustment of the APRM readout to conform to the power values calculated by a heat balance during OPERATIONAL CONDITION 1 when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

(f) This calibration shall consist of the adjustment of the APRM flow-biased simulated thermal power channel to conform to a calibrated flow signal.

(g) The LPRMs shall be calibrated at least once per effective full power month (EFPM) using the TIP system.

(h) This calibration shall consist of a physical inspection and actuation of these position switches.

(i) (Deleted) |

(j) (Deleted) |

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

(l) Transmitters are exempted from the quarterly channel calibration.

(m) Placement of Reactor Mode Switch into the Startup/Hot Standby position is permitted for the purpose of performing the required surveillance prior to withdrawal of control rods for the purpose of bringing the reactor to criticality.

(n) Placement of Reactor Mode Switch into the Shutdown or Refuel position is permitted for the purpose of performing the required surveillance provided all control rods are fully inserted and the vessel head bolts are tensioned.

(o) Surveillance is not required when THERMAL POWER is less than 30% of RATED THERMAL POWER.

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. (Deleted)				
2. Pressure - Low	1(j)	2	1	22
3. Flow - High	1(j)	2/line	1	22
4. Flow - High	1(j)	2	2, 3	21
d. Main Steam Line Tunnel Temperature - High	1(j)	2(d)	1, 2, 3	21
e. Condenser Vacuum - Low	1	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 3.3.2-2

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. (Deleted)		
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
5. SHUTDOWN COOLING SYSTEM ISOLATION		
a. Reactor Vessel Water Level - Low Level 1	≥ 162.5 inches[a]	≥ 162.5 inches[a]
b. Reactor Steam Dome Pressure - High	≤ 140 psig	≤ 140 psig

(a) Vessel water levels refer to REFERENCE LEVEL ZERO.

(b) Establish alarm/trip setpoints per the methodology contained in the OFFSITE DOSE CALCULATION MANUAL (ODCM).

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	Q	Q	1, 2, 3
2. Low, Level 3				
Transmitter:	NA[a]	NA	R[b]	1, 2, 3
Trip Logic:	D	Q	Q	1, 2, 3
b. Drywell Pressure - High				
Transmitter:	NA[a]	NA	R[b]	1, 2, 3
Trip Logic:	D	Q	Q	1, 2, 3
c. Main Steam Line				
1. (Deleted)				
2. Pressure - Low				
Transmitter:	NA[a]	NA	R[b]	1
Trip Logic:	D	Q	Q	1
3. Flow - High				
Transmitter:	NA[a]	NA	R[b]	1
Trip Logic:	D	Q	Q	1
4. Flow - High	D	Q	Q	2, 3
d. Main Steam Line Tunnel Temperature - High	NA	Q	R	1, 2, 3
e. Condenser Vacuum - Low				
Transmitter:	NA[a]	NA	R[b]	1, 2[e]
Trip Logic:	D	Q	Q	1, 2[e]
f. Turbine Building Area Temperature - High	NA	Q	R	1, 2, 3
g. Main Stack Radiation - High	NA	Q	R	1, 2, 3
h. Reactor Building Exhaust Radiation - High	D	Q	R	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 quarterly channel calibration.

(c) Deleted.

(d) Deleted.

(e) When reactor steam pressure ≥ 500 psig.

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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

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

AND AMENDMENT NO. TO FACILTY 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 30, 1994, as supplemented on March 24, 1995, Carolina Power & Light Company (CP&L) submitted a request to amend the Brunswick Steam Electric Plant, Units 1 and 2, (Brunswick) Technical Specifications (TS). The March 24, 1995, supplement provided clarifying information only and did not affect the NRC's determination of no significant hazards consideration. The proposed amendment will eliminate the main steam line radiation monitor (MSLRM) scram and isolation functions. As a result of the proposed amendment, the reactor scram, main steam isolation valves (MSIVs) closure, main steam line (MSL) drain valve closure, reactor water sample line isolation, and mechanical vacuum line isolation on a "High-High" radiation signal from the MSLRM will be removed from the TS. The MSLRM signal for the reactor water sample line isolation will be replaced with a low condenser vacuum signal. An isolation signal from the main stack radiation monitor will be provided for the mechanical vacuum pump line. This request was submitted as the plant specific portion which, in conjunction with the General Electric Company Licensing Topical Report NEDO-31400 and the NRC staff's May 15, 1991, Safety Evaluation (SE) on this topical report, formed the basis for the package to be evaluated.

The licensee states that elimination of this trip function would result in a reduced potential for unnecessary reactor shutdowns and engineered safety features actuations caused by spurious MSLRM "High-High" signals and would increase plant operational flexibility without compromising plant safety. The licensee's proposed changes are based on the May 1987 BWR Owners' Group Licensing Topical Report NEDO-31400[1] and on NRC's NUREG-0800[2].

In NEDO-31400, a reevaluation of the role of the MSLRM in the control rod drive accident (CRDA) analysis was performed, confirming that removal of the

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[1] NEDO-31400, "Safety Evaluation for Eliminating the Boiling Water Reactor Main Steam Line Isolation Valve Closure Function and Scram Function of the Main Steam Radiation Monitor".

[2] NUREG-0800, Standard Review Plan (SRP) 15.4.9, Rev.2, July 1981.

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MSLRM scram/isolation features would not compromise CRDA consequences. The topical report also evaluated the potential effect on occupational exposure in the event of a sudden release of radioactive material from the fuel and concluded that the elimination of the scram/isolation features would have no adverse effect.

Specifically, the licensee proposed the following changes for the TS for Units 1 and 2:

1. All references to the MSLRM automatic shutdown features have been deleted from Tables 2.2.1-1, 3.3.1-1 and 4.3.1-1.

2. All references to the MSLRM isolation function of the main steam lines have been deleted from Table 3.3.2-1, and 4.3.2-1.

3. The Bases for 2.2, Limiting Safety System Settings, have been amended to reflect the deletion of the main steam line radiation automatic reactor shutdown function and closure of the MSIVs.

4. The Actions and Notes items for Tables 2.2.1-1, Reactor Protection System Instrumentation Setpoints; 3.3.1-1, Reactor Protection System Instrumentation Actions, 4.3.1-1, Reactor Protection System Instrumentation Surveillance Requirements Notes; and 4.3.2-1, Isolation Actuation Instrumentation Surveillance Requirements Notes; have been amended to reflect the deletion of the main steam line radiation monitor automatic reactor shutdown and isolations.

2.0 EVALUATION

The MSLRM consists of four redundant radiation detectors located on the outside of the MSLs and external to the primary containment. The MSLRM was designed to provide an early indication of gross fuel cladding failures. The original intent of this monitor was to mitigate the releases of the detected fuel failure by providing a scram signal to terminate the initiating event and a MSIV closure signal to assure containment of the release. However, no credit is taken for the automatic reactor scram due to high radiation in the MSLs in any design basis event.

The only design basis accident in which either the MSLRM scram or MSIV isolation functions are mentioned is the control rod drop accident. To be consistent with Section 15.4.9 of the Standard Review Plan (SRP), all of the postulated radioactive material is assumed to be released to the condenser and turbine before the isolation occurs. Hence, no credit is assumed for the automatic isolation resulting from the MSLRM signal, and the resultant dose consequences from the control rod drop accident will remain unchanged. However, since the alarm function will be retained, an operator can initiate a manual closure of the MSIV after confirming a high radiation level in the MSLs. Additionally, upon a main stack high radiation signal, the mechanical vacuum pumps (MVPs) for the main condenser will automatically trip, and the MVP line will isolate. The MVPs are only in service at plant power levels of less than 5 percent. Above 5 percent power the offgass line from the

condenser will isolate automatically on a high radiation signal from a radiation monitor located on the offgas line.

In a May 15, 1991, Safety Evaluation on NEDO-31400, the staff concluded that removal of the MSLRM trips that automatically shutdown the reactor and close the main steam isolation valves was acceptable and that Licensing Topical Report NEDO-31400 may be referenced in support of an amendment request as long as the following three conditions were met:

1. The applicant demonstrates that the assumptions with regard to input values (including power per assembly, X/Q, and decay times) that are made in generic analysis bound those for the plant.

The licensee, in response to Condition 1, has provided two tables showing: a comparison of key input parameters and a comparison of the dose assessment between Brunswick design basis and NEDO-31400 analysis assumptions. The specific power level is used to determine the source term. This factor is offset by the lower calculated power for the failed fuel rods. The licensee also considered the two hour exclusion area boundary doses for the CRDA. The Brunswick site-specific atmospheric dispersion factor was lower than the value used in NEDO-31400. The other parameters are the same or more conservative than the NEDO-31400 values. Based on the above, the staff agrees that the generic analysis of the Licensing Topical Report is bounding for Brunswick. The staff finds that the licensee's analysis has met the applicable requirements of Condition 1, and is, therefore, acceptable.

2. The applicant includes evidence (implemented or proposed operating procedures, or equivalent commitments) to provide reasonable assurance that increased significant levels of radioactive material in the main steam lines will be controlled expeditiously to limit both occupational doses and effluent releases.

In the response to Condition 2, the licensee has in place the Offsite Dose Calculation Manual (ODCM), a Radiation Protection Program, including an ALARA program, and a Radiological Environmental Monitoring Program. Existing annunciator procedures for the MSLRM "High Radiation" annunciator and the condenser off-gas "High Radiation" annunciator initiate actions through emergency operating procedures which ensure that significant levels of radiation in the MSLs are controlled expeditiously to limit both occupational doses and environmental releases. Brunswick's emergency operating procedures will be revised as necessary to incorporate specific considerations to change isolation of the MSLs from an automatic to a manual function. Thus any significant increase in the levels of radioactivity in the MSLs will continue to be promptly controlled to limit effluent releases and on-site occupational exposure.

The MSLRM alarm setpoint of 1.5 times the normal full power background will be used to initiate sampling and surveillance actions. Confirmation of elevated activity will cause administrative controls to be implemented that ensure offsite and onsite doses are maintained ALARA. Manual action to close the MSLs and shutdown the reactor will occur when all the evidence has indicated the need for isolation and shutdown. Also, procedures will require immediate

notification of radiation protection and chemistry upon annunciation in the control room of the high radiation alarm of the MSLRM or offgas radiation monitor. The NRC staff concludes that the licensee's commitment is acceptable and responsive to Condition 2 that was addressed in the Topical Report NEDO-31400.

3. The applicant standardizes the MSLRM and offgas radiation monitor alarm setpoint at 1.5 times the nominal nitrogen-16 background dose rate at the monitor locations, and commits to promptly sample the reactor coolant to determine possible contamination levels in the plant reactor coolant and the need for additional corrective actions, if either the MSLRM or offgas radiation monitors or both exceed their alarm setpoint.

In response to Condition 3, CP&L will maintain the MSLRM alarm setpoint at 1.5 times the normal full power nitrogen-16 (N-16) background dose rate. CP&L will reduce the condenser off-gas monitor alarm setpoint to 1.5 times the nominal N-16 background but not less than 1.5 R/hr. This alarm will trigger entry into a procedure which will require a reactor coolant sample to be obtained and analyzed (TS 3.4.5.b.2). The alarm setpoint accounts for the normal full power N-16 carryover, due to hydrogen water chemistry, at the monitor location. The offgas radiation monitor is a more sensitive monitor than the MSLRM because the N-16, dominating the radiation signal to the MSLRM, has decayed by the time the offgas radiation monitor can be affected by any increased levels of activity. Therefore, setting the offgas radiation monitor at 1.5 times the N-16 background dose rate can lead to spurious activations of the alarm. The offgas radiation monitor alarm is set to satisfy Brunswick TS 3.4.5.b.2, which is based on the ODCM. The offgas monitor setpoint provides assurance that the total body exposure to an individual at the exclusion area boundary will not exceed the dose reference values of 10 CFR Part 100. The licensee has proposed to set the offgas radiation monitor alarm at five (5) times the normal full power background. If the monitor alarms at this setpoint the offgas will be immediately sampled and analyzed, followed by an analysis of reactor coolant sample. Based on a review of the licensee's commitment, the NRC staff has determined that Condition 3 has been satisfied.

Based on a review of the CP&L submittals and safety analysis, the NRC staff concludes that there are no adverse safety implications associated with removal of the MSLRM scram and MSIV closure function because the licensee has provided reasonable assurance that the offsite radiation exposure levels are within the reference values of 10 CFR Part 100 and SRP 15.4.9. The NRC staff concludes that the proposed changes to eliminate the reactor scram and MSIV requirements associated with the MSLRMs are 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 amendments. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes the Surveillance Requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (59 FR 55867). Accordingly, the 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 the amendments.

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

Principal Contributor: J. Minns

Date: March 31, 1995