February 1, 1995

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

Dear Mr. Anderson:

SUBJECT: ISSUANCE OF AMENDMENT NO. 174 TO FACILITY OPERATING LICENSE NO. DPR-71 AND AMENDMENT NO. 205 TO FACILITY OPERATING LICENSE NO. DPR-62 REGARDING ADMINISTRATIVE CHANGES - BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2 (TAC NOS. M89113 AND M89114)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 174 to Facility Operating License No. DPR-71 and Amendment No. 205 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 submittal dated March 25, 1994, as supplemented July 29, 1994, and August 24, 1994.

The amendments change the Technical Specifications to correct several of the licensees typographical errors, to add material implicitly contained in a footnote to an applicability statement, to provide detailed labels for items listed in a table, to correct the citation of references, and to remove references to the rod sequence control system that should have been included in a previous submittal.

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,
/s/
Patrick D. Milano, Senior 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. 174 to License No. DPR-71
2. Amendment No. 205 to License No. DPR-62
3. Safety Evaluation

cc w/enclosures: See next page

Document Name G:\BRUNSWIC\BR89113.AMD

OFFICE	LA:PDII-1	PM:PDII-1	SRXB	D:PDII-1	OGC
NAME	PAnderson	PMilano	RJones	WBateman	RBachmann
DATE	1/3 /95	1/3 /95	1/12/95	2/1 /95	1/20/95
COPY	Yes/No	Yes/No	Yes/No	Yes/No	Yes/No

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AMENDMENT NO.174 TO FACILITY OPERATING LICENSE NO. DPR-71 - BRUNSWICK, UNIT 1
AMENDMENT NO.205 TO FACILITY OPERATING LICENSE NO. DPR-62 - BRUNSWICK, UNIT 2

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cc: Brunswick Service List

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

February 1, 1995

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. 174 TO FACILITY OPERATING LICENSE NO. DPR-71 AND AMENDMENT NO. 205 TO FACILITY OPERATING LICENSE NO. DPR-62 REGARDING ADMINISTRATIVE CHANGES - BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2 (TAC NOS. M89113 AND M89114)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 174 to Facility Operating License No. DPR-71 and Amendment No. 205 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 submittal dated March 25, 1994.

The amendments change the Technical Specifications to correct several of the licensees typographical errors, to add material implicitly contained in a footnote to an applicability statement, to provide detailed labels for items listed in a table, to correct the citation of references, and to remove references to the rod sequence control system that should have been included in a previous submittal.

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,

Patrick D. Milano, Senior 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. 174 to License No. DPR-71
2. Amendment No. 205 to License No. DPR-62
3. Safety Evaluation

cc w/enclosures: See next page

Mr. R. A. Anderson
Carolina Power & Light Company

Brunswick Steam Electric Plant
Units 1 and 2

cc:

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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. 174
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 March 25, 1994, as supplemented on July 29, 1994, and August 24, 1994, 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. 174, 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

William H. Bateman

William 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: February 1, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 174

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
B 2-4	B 2-4
B 2-6	B 2-6
3/4 1-14	3/4 1-14
3/4 3-52	3/4 3-52
3/4 3-64a	3/4 3-64a
3/4 3-64c	3/4 3-64c
3/4 3-88	3/4 3-88
3/4 4-4	3/4 4-4

2.2.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

The Reactor Protection System Instrumentation Setpoints specified in Table 2.2.1-1 are the values at which the Reactor Trips are set for each parameter. The Trip Setpoints have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their safety limits.

1. Intermediate Range Monitor, Neutron Flux - High

The IRM system consists of 8 chambers, 4 in each of the reactor trip systems. The IRM is a 5-decade 10-range instrument. The trip setpoint of 120 divisions is active in each of the 10 ranges. Thus as the IRM is ranged up to accommodate the increase in power level, the trip setpoint is also ranged up. Range 10 allows the IRM instruments to remain on scale at higher power levels to provide for additional overlap and also permits calibration at these higher powers.

The most significant source of reactivity change during the power increase is due to control rod withdrawal. In order to ensure that the IRM provides the required protection, a range of rod withdrawal accidents have been analyzed, Section 7.5 of the FSAR. The most severe case involves an initial condition in which the reactor is just subcritical and the IRMs are not yet on scale. Additional conservatism was taken in this analysis by assuming the IRM channel closest to the rod being withdrawn is bypassed. The results of this analysis show that the reactor is shut down and peak power is limited to 1% of RATED THERMAL POWER, thus maintaining MCPR above the Safety Limit MCPR of Specification 2.1.2. Based on this analysis, the IRM provides protection against local control rod errors and continuous withdrawal of control rods in sequence and provides backup protection for the APRM.

2. Average Power Range Monitor

For operation at low pressure and low flow during STARTUP, the APRM scram setting of 15% of RATED THERMAL POWER provides adequate thermal margin between the setpoint and the Safety Limits. This margin accommodates the anticipated maneuvers associated with power plant startup. Effects of increasing pressure at zero or low void content are minor, cold water from sources available during startup is not much colder than that already in the system, temperature coefficients are small, and control rod patterns are constrained by the RWM. Of all the possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power increase. Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks and because several rods must be moved to change power by a significant amount, the rate of power rise is very slow. Generally, the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the trip level, the rate of power rise is not more than 5% of RATED THERMAL POWER per minute and the APRM system would

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. Main Steam Line Radiation - High

The Main Steam Line Radiation detectors are provided to detect a gross failure of the fuel cladding. When the high radiation is detected, a scram is initiated to reduce the continued failure of fuel cladding. At the same time, the Main Steam Line Isolation Valves are closed to limit the release of fission products. The trip setting is high enough above background radiation level to prevent spurious scrams, yet low enough to promptly detect gross failures in the fuel cladding.

The Main Steam Line Radiation detectors setpoints may be adjusted prior to placing the hydrogen water chemistry (HWC) system in service. If the setpoints are adjusted, the HWC system shall be placed in service or the setpoints shall be returned to the normal full power values within 24 hours. If the HWC system is not placed in service and the setpoints are not readjusted within 24 hours, control rod motion shall be suspended (except for scram or other emergency action) until the necessary adjustments are made. Hydrogen injection may cause the radiation levels in the main steam lines to increase. After shutting off the HWC system or decreasing power, the setpoints shall be returned to the normal full power values.

The Technical Specification wording was derived using the EPRI "Guidelines for Permanent BWR Hydrogen Water Chemistry Installations, 1987 Revision".

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.

3/4 1.4 CONTROL ROD PROGRAM CONTROLS

ROD WORTH MINIMIZER

LIMITING CONDITION FOR OPERATION

3.1.4.1 The Rod Worth Minimizer (RWM) shall be OPERABLE when THERMAL POWER is less than 10% of RATED THERMAL POWER.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2*.

ACTION:

a. With the RWM inoperable after the first 12 control rods have been fully withdrawn on a startup, operation may continue provided that control rod movement and compliance with the prescribed BPWS control rod pattern are verified by a second licensed operator or qualified member of the plant technical staff.

b. With the RWM inoperable before the first 12 control rods are withdrawn on a startup, one startup per calender year may be performed provided that control rod movement and compliance with the prescribed BPWS control rod pattern are verified by a second licensed operator or qualified member of the plant technical staff.

c. With RWM inoperable on a shutdown, shutdown may continue provided that control rod movement and compliance with the prescribed BPWS control rod pattern are verified by a second licensed operator or qualified member of the plant technical staff.

d. With RWM operable but individual control rod(s) declared inoperable, operation and control rod movement below the preset power level of the RWM may continue provided:

   1. No more than three (3) control rods are declared inoperable in any one BPWS group, and,

   2. The inoperable control rod(s) is bypassed on the RWM and control rod movement of the bypassed rod(s) is verified by a second licensed operator or qualified member of the plant technical staff.

e. With RWM inoperable, the provisions of Specification 3.0.4 are not applicable.

---

* Entry into OPERATIONAL CONDITION 2 and withdrawal of selected control rods is permitted for the purpose of determining the OPERABILITY of the RWM prior to withdrawal of control rods for the purpose of bringing the reactor to criticality.

TABLE 4.3.4-1 (Continued)

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

NOTES

(a) CHANNEL CALIBRATIONS are electronic.

(b) This calibration shall consist of the adjustment of the APRM flow biased setpoint to conform to a calibrated flow signal.

(c) Within 24 hours prior to 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) Placement of Reactor Mode Switch into 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.

(f) 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.

(g) When THERMAL POWER is greater than the preset power level of the RWM. |

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

(i) 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 92 days.

TABLE 3.3.5.5-1

CONTROL ROOM EMERGENCY VENTILATION SYSTEM INSTRUMENTATION

FUNCTION	REQUIRED NUMBER OF DETECTORS PER TRIP SYSTEM	APPLICABLE OPERATIONAL CONDITIONS	ACTION	ALARM/TRIP SETPOINT
1. CHLORINE ISOLATION:				
a. Control Building Air Intake (Local) Trip System	4 (a)	(b)	90	≤ 5ppm
b. Chlorine Tank Car Area (Remote) Trip System	4 (a)	(b)	90	≤ 5ppm
2. RADIATION PROTECTION:				
Control Building Air Intake	2	1, 2, 3, 4, 5, and (c)	91	≤ 7mR/hr (d)
3. CONTROL ROOM ENVELOPE SMOKE PROTECTION:				
a. Zone 4	2	1, 2, 3, 4, 5, and (c)	92	NA
b. Zone 5	2	1, 2, 3, 4, 5, and (c)	92	NA

(a) Four OPERABLE detectors per trip system, consisting of two detectors per trip subsystem.

(b) With the chlorine tank car within the exclusion area.

(c) During movement of irradiated fuel assemblies in the secondary containment.

(d) Allowable value of ≤ 10mR/hr.

TABLE 4.3.5.5-1

CONTROL ROOM EMERGENCY VENTILATION SYSTEM
INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
1. CHLORINE ISOLATION:			
a. Local Detection Trip System	NA	M	A
b. Remote Detection Trip System	NA	M	A
2. RADIATION PROTECTION:			
Control Building Air Intake	D	M	R
3. CONTROL ROOM ENVELOPE SMOKE PROTECTION:			
a. Zone 4	NA	6 months	(a)
b. Zone 5	NA	6 months	(a)

(a) See Surveillance Requirement 4.7.2.d.2

3/4.3.6 ATWS RECIRCULATION PUMP TRIP (RPT) SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.6.1 The ATWS-RPT system instrumentation trip systems shown in Table 3.3.6.1-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.6.1-2.

APPLICABILITY: OPERATIONAL CONDITION 1.

ACTION:

a. With an ATWS-RPT system instrumentation trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.6.1-2, declare the instrument channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels per Trip System requirement for one or both trip systems, place the inoperable channel(s) in the tripped condition within one hour.

c. With the total number of OPERABLE channels less than 3 as required by the Minimum OPERABLE Channels per Trip System requirement for one trip system and:

    1. If the inoperable channels consist of one reactor vessel water level channel and one reactor vessel pressure channel, place both inoperable channels in the tripped condition within one hour.

    2. If the inoperable channels include two reactor vessel water level channels or two reactor vessel pressure channels, declare the trip system inoperable.

d. With one trip system inoperable, restore the inoperable trip system to OPERABLE status within 14 days or be in at least STARTUP within the next 8 hours.

e. With both trip systems inoperable, restore at least one trip system to OPERABLE status within one hour or be in at least STARTUP within the next 8 hours.

SURVEILLANCE REQUIREMENTS

4.3.6.1.1 Each ATWS-RPT system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.6.1-1.

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

3/4.4.2 SAFETY/RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.2 The safety valve function of all reactor coolant system safety/relief valves shall be OPERABLE with lift settings within ± 1% of the following values.*

4 Safety-relief valves @ 1105 psig.
4 Safety-relief valves @ 1115 psig.
3 Safety-relief valves @ 1125 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With the safety valve function of one safety/relief valve inoperable, restore the inoperable safety valve function of the valve to OPERABLE status within 31 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With the safety valve function of two safety/relief valves inoperable, restore the inoperable safety valve function of at least one of the valves to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With the safety valve function of more than two safety/relief valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.2 The safety valve function of each of the above required safety/relief valves shall be demonstrated OPERABLE in accordance with the Surveillance Requirements of Specification 4.0.5.

* The lift setting pressure shall correspond to ambient conditions of the valves at normal operating temperature and pressure.

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. 205
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 March 25, 1994, as supplemented on July 29, 1994, and August 24, 1994, 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. 205 , 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

William H. Bateman

William 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: February 1, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 205

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
B 2-4	B 2-4
B 2-7	B 2-7
B 2-8	B 2-8
3/4 1-14	3/4 1-14
3/4 3-9	3/4 3-9
3/4 3-52	3/4 3-52
3/4 3-64a	3/4 3-64a
3/4 3-64c	3/4 3-64c
3/4 3-93	3/4 3-93
B 3/4 1-4	B 3/4 1-4

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. Main Steam Line Radiation - High [(h)]	≤ 3 x full power background	≤ 3.5 x full power background
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

2.2.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

The Reactor Protection System Instrumentation Setpoints specified in Table 2.2.1-1 are the values at which the Reactor Trips are set for each parameter. The Trip Setpoints have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their safety limits.

1. Intermediate Range Monitor, Neutron Flux - High

The IRM system consists of 8 chambers, 4 in each of the reactor trip systems. The IRM is a 5-decade, 10-range instrument. The trip setpoint of 120 divisions is active in each of the 10 ranges. Thus, as the IRM is ranged up to accommodate the increase in power level, the trip setpoint is also ranged up. Range 10 allows the IRM instruments to remain on scale at higher power levels to provide for additional overlap and also permits calibration at these higher powers.

The most significant source of reactivity change during the power increase is due to control rod withdrawal. In order to ensure that the IRM provides the required protection, a range of rod withdrawal accidents have been analyzed in Section 7.5 of the FSAR. The most severe case involves an initial condition in which the reactor is just subcritical and the IRMs are not yet on scale. Additional conservatism was taken in this analysis by assuming the IRM channel closest to the rod being withdrawn is bypassed. The results of this analysis show that the reactor is shut down and peak power is limited to 1% of RATED THERMAL POWER, thus maintaining MCPR above the Safety Limit MCPR of Specification 2.1.2. Based on this analysis, the IRM provides protection against local control rod errors and continuous withdrawal of control rods in sequence and provides backup protection for the APRM.

2. Average Power Range Monitor

For operation at low pressure and low flow during STARTUP, the APRM scram setting of 15% of RATED THERMAL POWER provides an adequate thermal margin between the setpoint and the Safety Limits. This margin accommodates the anticipated maneuvers associated with power plant startup. Effects of increasing pressure at zero or low void content are minor; cold water from sources available during startup is not much colder than that already in the system, temperature coefficients are small, and control rod patterns are constrained by the RWM. Of all the possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power increase. Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks and because several rods must be moved to change power by a significant amount, the rate of power rise is very slow. Generally, the heat flux is in near equilibrium with the fission rate In an assumed uniform rod withdrawal approach to the trip level, the rate of power rise is not more than 5% of RATED THERMAL POWER per

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.

8. Scram Discharge Volume Water Level-High

The scram discharge tank receives the water displaced by the motion of the control rod drive pistons during a reactor scram. Should this tank fill up to a point where there is insufficient volume to accept the displaced water, control rod movement would be hindered. The reactor is therefore tripped when the water level has reached a point high enough to indicate that it is indeed filling up, but the volume is still great enough to accommodate the water from the movement of the rods when they are tripped.

9. Turbine Stop Valve-Closure

The turbine stop valve closure trip anticipates the pressure, neutron flux, and heat flux increases that would result from closure of the stop valves. With a trip setting of 10% of valve closure from full open, the resultant increase in heat flux is such that adequate thermal margins are maintained even during the worst case transient that assumes the turbine bypass valves remain closed. This scram is bypassed when the turbine steam flow is below that corresponding to 30% of RATED THERMAL POWER, as measured by the turbine first-stage pressure.

10. Turbine Control Valve Fast Closure, Control Oil Pressure - Low

Low turbine control valve hydraulic pressure will initiate the Select Rod Insert function and the preselected group of control rods will be fully inserted. Select Rod Insert is an operational aid designed to insert a predetermined group of control rods immediately following either a generator load rejection, loss of turbine control valve hydraulic pressure, or by manual operator action using a switch on the R-T-G board. The assignment of control rods to the Select Rod Insert function is based on the start-up and fuel warranty service associated with each control rod pattern, on RWM considerations, and on a dynamic function of both time and core patterns.

Approximately ten percent of the control rods in the reactor will be assigned to the Select Rod Insert function by the operator. This selection will be accomplished by moving the rod scram test switch for those rods from the Normal position to the Select Rod Insert position.

10. Turbine Control Valve Fast Closure, Control Oil Pressure - Low (Continued)

Any rod selected for Select Rod Insert shall also have other rods in its BPWS group selected to ensure that the RWM criteria is met when THERMAL POWER < 10% of RATED THERMAL POWER. It is possible that a rod pattern within these limits may occur after the Select Rod Insert function operates.

In order to reduce the number of reactor scrams, a 200 millisecond time delay, referenced from the low turbine control valve hydraulic pressure and Select Rod Insert signals, was incorporated to determine turbine bypass valve status via limit switches prior to initiating a reactor scram. If the turbine bypass valves opened in < 200 milliseconds, the reactor scram was bypassed. It was found that during certain reload cycles the MCPR penalties involved with this time delay were more penalizing than the number of scrams saved; therefore, CP&L requested and received NRC approval to set this time at "0" in Amendment No. 14. With the timer set at "0", Select Rod Insert and RPS trip will be initiated simultaneously.

The control valve closure time is approximately twice as long as that for the stop valves which means that resulting transients, while similar, are less severe than for stop valve closure. No fuel damage occurs, and reactor system pressure does not exceed the safety relief valve setpoint. This is an anticipatory scram and results in reactor shutdown before any significant increase in pressure or neutron flux occurs. This scram is bypassed when turbine steam flow is below that corresponding to 30 percent of RATED THERMAL POWER, as measured by turbine first-stage pressure.

REACTIVITY CONTROL SYSTEMS

3/4 1.4 CONTROL ROD PROGRAM CONTROLS

ROD WORTH MINIMIZER

LIMITING CONDITION FOR OPERATION

3.1.4.1 The Rod Worth Minimizer (RWM) shall be OPERABLE when THERMAL POWER is less than 10% of RATED THERMAL POWER.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2*.

ACTION:

a. With the RWM inoperable after the first 12 control rods have been fully withdrawn on a startup, operation may continue provided that control rod movement and compliance with the prescribed BPWS control rod pattern are verified by a second licensed operator or qualified member of the plant technical staff.

b. With the RWM inoperable before the first 12 control rods are withdrawn on a startup, one startup per calender year may be performed provided that control rod movement and compliance with the prescribed BPWS control rod pattern are verified by a second licensed operator or qualified member of the plant technical staff.

c. With RWM inoperable on a shutdown, shutdown may continue provided that control rod movement and compliance with the prescribed BPWS control rod pattern are verified by a second licensed operator or qualified member of the plant technical staff.

d. With RWM operable but individual control rod(s) declared inoperable, operation and control rod movement below the preset power level of the RWM may continue provided:

   1. No more than three (3) control rods are declared inoperable in any one BPWS group, and,

   2. The inoperable control rod(s) is bypassed on the RWM and control rod movement of the bypassed rod(s) is verified by a second licensed operator or qualified member of the plant technical staff.

e. With RWM inoperable, the provisions of Specification 3.0.4 are not applicable.

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* Entry into OPERATIONAL CONDITION 2 and withdrawal of selected control rods is permitted for the purpose of determining the OPERABILITY of the RWM prior to withdrawal of control rods for the purpose of bringing the reactor to criticality.

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 setpoint 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) Instrument alignment using a standard current source.

(j) Calibration using a standard radiation source.

(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 monthly 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.

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

NOTES

(a) CHANNEL CALIBRATIONS are electronic.

(b) This calibration shall consist of the adjustment of the APRM flow biased setpoint to conform to a calibrated flow signal.

(c) Within 24 hours prior to 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) Placement of Reactor Mode Switch into 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.

(f) 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.

(g) When THERMAL POWER is greater than the preset power level of the RWM. |

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

(i) 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 92 days.

TABLE 3.3.5.5-1

CONTROL ROOM EMERGENCY VENTILATION SYSTEM INSTRUMENTATION

FUNCTION	REQUIRED NUMBER OF DETECTORS PER TRIP SYSTEM	APPLICABLE OPERATIONAL CONDITIONS	ACTION	ALARM/TRIP SETPOINT
1. CHLORINE ISOLATION:				
a. Control Room Air Intake (Local) Trip System	4 (a)	(b)	90	≤ 5ppm
b. Chlorine Tank Car Area (Remote) Trip System	4 (a)	(b)	90	≤ 5ppm
2. RADIATION PROTECTION:				
Control Room Air Intake	2	1, 2, 3, 4, 5, and (c)	91	≤ 7mR/hr (d)
3. CONTROL ROOM ENVELOPE SMOKE PROTECTION:				
a. Zone 4	2	1, 2, 3, 4, 5, and (c)	92	NA
b. Zone 5	2	1, 2, 3, 4, 5, and (c)	92	NA

(a) Four OPERABLE detectors per trip system, consisting of two detectors per trip subsystem.

(b) With the chlorine tank car within the exclusion area.

(c) During movement of irradiated fuel assemblies in the secondary containment.

(d) Allowable value of ≤ 10mR/hr.

TABLE 4.3.5.5-1

CONTROL ROOM EMERGENCY VENTILATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
1. CHLORINE ISOLATION:			
a. Local Detection Trip System	NA	M	A
b. Remote Detection Trip System	NA	M	A
2. RADIATION PROTECTION:			
Control Room Air Intake	D	M	R
3. CONTROL ROOM ENVELOPE SMOKE PROTECTION:			
a. Zone 4	NA	6 months	(a)
b. Zone 5	NA	6 months	(a)

(a) See Surveillance Requirement 4.7.2.d.2

INSTRUMENTATION

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.6.2 The end-of-cycle recirculation pump trip (EOC-RPT) system instrumentation channels shown in Table 3.3.6.2-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.6.2-2 and with the END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME as shown in Table 3.3.6.2-3.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 30% of RATED THERMAL POWER and the MCPR limits obtained from the COLR for use with Specification 3.2.2.1 require EOC-RPT.*

ACTION:

a. With an end-of-cycle recirculation pump trip system instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values Column of Table 3.3.6.2-2, declare the channel inoperable until the channel is restored to OPERABLE status with the channel setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels per Trip System requirement for one or both trip systems, place the inoperable channel(s) in the tripped condition within one hour.

c. With the number of OPERABLE channels two or more less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system and:

   1. If the operable channels consist of one turbine control valve channel and one turbine stop valve channel, place both inoperable channels in the tripped condition within one hour.

   2. If the inoperable channels include two turbine control valve channels or two turbine stop valve channels, declare the trip system operable.

d. With one trip system inoperable, restore the inoperable trip system to OPERABLE status within 72 hours or take the ACTION required by Specification 3.2.2.1.

e. With both trip systems inoperable, restore at least one trip system to OPERABLE status within one hour or take the ACTION required by Specification 3.2.2.1.

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* The provisions of Specification 3.0.4 are not applicable.

CONTROL ROD PROGRAM CONTROLS (Continued)

The RWM as a backup to procedural control provides an automatic control rod pattern monitoring function to ensure adherence to the BPWS control movement sequences from 100% control rod density to 10% RATED THERMAL POWER and, thus, eliminates the postulated control rod drop accident from resulting in a peak fuel enthalpy greater than 280 cal/gm (Reference 6).

The requirement that RWM be operable for the withdrawal of the first 12 control rods on a startup is to ensure that the RWM system maintains a high degree of availability.

Deviation from the BPWS control rod pattern may be allowed for the performance of Shutdown Margin Demonstration tests.

The analysis of the rod drop accident is presented in Section 15.4.6 of the Updated FSAR and the techniques of the analysis are presented in a topical report (Reference 1) and two supplements (References 2 and 3).

The RBM is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power operation. The RBM is only required to be operable when the limiting condition described in Specification 3.1.4.3 exists. Two channels are provided. Tripping one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. This system backs up the written sequence used by the operator for withdrawal of control rods. Further discussion of the RBM system is provided in Reference 5.

3/4.1.5 STANDBY LIQUID CONTROL SYSTEM

The standby liquid control system provides a backup capability for maintaining the reactor subcritical in the event that insufficient rods are inserted in the core when a scram is called for. The volume and weight percent of poison material in solution is based on being able to bring the reactor to the subcritical condition as the plant cools to ambient condition. The temperature requirement is necessary to keep the sodium pentaborate in solution. Checking the volume and temperature once each 24 hours assures that the solution is available for use.

With redundant pumps and a highly reliable control rod scram system, operation of the reactor is permitted to continue for short periods of time with the system inoperable or for longer periods of time with one of the redundant components inoperable.

Surveillance requirements are established on a frequency that assures a high reliability of the system. Once the solution is established, boron concentration will not vary unless more boron or water is added, thus a check on the temperature and volume once each 24 hours assures that the solution is available for use.

Replacement of the explosive charges in the valves at regular intervals will assure that these valves will not fail because of deterioration of the charges.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 174 TO FACILITY OPERATING LICENSE NO. DPR-71

AND AMENDMENT NO. 205 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 March 25, 1994, as supplemented July 29, 1994, and August 24, 1994, the Carolina Power & Light Company (the licensee) submitted a request for changes to the Brunswick Steam Electric Plant, Units 1 and 2, Technical Specifications (TS). The amendments would change the TS to correct several of the licensee's typographical errors, to add material implicitly contained in a footnote to an applicability statement, to provide detailed labels for items listed in a table, to correct the citation of references, and to remove references to the rod sequence control system that should have been included in a previous submittal. The July 29, 1994, and August 24, 1994, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

2.0 EVALUATION

The licensee proposed the changes to the TS to correct various administrative errors. In this regard, the licensee requested the following changes to the TS, referenced tables, and the supporting Bases sections.

Brunswick Steam Electric Plant, Unit 1

a. Bases Section 2.2.1, Item 2, Average Power Range Monitor

In Amendment No. 144 to the TS issued on September 11, 1990, the staff approved the licensees request to remove the rod sequence control system (RSCS) and to reduce the rod worth minimizer cutoff setpoint to 10 percent of reted thermal power. However, the licensee's submittal of March 14, 1990, did not propose the removal of the reference to the RSCS in the Bases. Therefore, the proposed change will correct this oversight. Since the RSCS removal was previously approved by the NRC staff, this change only corrects a previous admistrative error and is acceptable.

h. Section 3.4.2, Safety/Relief Valves Limiting Condition for Operation

The proposed change removes the footnote identifier "#" that idicates a second footnote that does not exist. The second footnote was removed in Amendment No. 66, dated March 6, 1984, but the footnote idicator was not corrected. Since this change corrects an administrative error, the staff finds it acceptable.

Brunswick Steam Electric Plant, Unit 2

a. Table 2.2.1-1, Reactor Protection System Instrumentation Setpoints

The proposed change would correct an error introduced in Amendment No. 171 on February 6, 1990, for the maximum allowable value of rated thermal power in the flow biased simulated thermal power - high trip. In Amendment No. 168, dated October 12, 1989, the NRC approved a change to the allowable value that was dependent on core flow but with a maximum value of 115.5 percent. However, when Amendment No. 171 was issued, approving a TS change related to the main steam line radiation monitors, Table 2.2.1-1 was revised and it did not include the change based on Amendment No. 168. Therefore, the allowable value stated in the table was incorrectly returned to 115 percent. The proposed change corrects an administratively introduced error. Since this error is now being corrected to the value given in Amendment No. 168, the staff finds this change acceptable.

b. Bases Section 2.2.1, Item 2, Average Power Range Monitor

The proposed change removes the references to the rod sequence control system and reduces the rod worth minimizer cutoff setpoint from 20 percent to 10 percent of rated thermal power. The NRC staff finds this change acceptable for the same reasons given for BSEP Unit 1.

c. Bases Section 2.2.1, Item 10, Turbine Control Valve Fast Closure, Control Oil Pressure - Low

As with the proposed change to Bases Section 2.2.1, Item 2, the TS change removes the reference to the rod sequence control system and reduces the rod worth minimizer cutoff setpoint from 20 percent to 10 percent of rated thermal power. The proposed change would also revise the Bases description for the select rod insertion that was not revised when the removal of the rod sequence control system was approved. The NRC has reviewed the Bases changes and finds them to be acceptable.

d. Section 3.1.4.1, Rod Worth Minimizer Limiting Condition for Operation (LCO)

In Action Statement d.1 of this LCO, the abbreviation for the blanked position withdrawl sequence would be changed from "BWS" to "BPWS." This change corrects a typographical error and is statisfactory.

e. Table 4.3.1-1, Reactor Protection System Instrumentation Surveillance Requirements

The proposed change to this table would add the word "is" after the words "THERMAL POWER" in Note (e). This change corrects a grammatical error in the sentence and makes it consistent with the wording used in the same table note for Unit 1. Since the change is grammatical in nature and does not alter the meaning of the note, the NRC staff finds the change acceptable.

f. Table 4.3.4-1, Control Rod Withdrawl Block Instrumentation Surveillance Requirements

The proposed change to Note (g) in this TS Table will remove the reference to the RSCS for the reasons stated above. Since the RSCS removal was previously approved by the NRC staff, this change only corrects a previous admistrative error and is, therefore, acceptable.

g. Table 3.3.5.5-1, Control Room Emergency Ventillation System Instrumentation

The proposed change modifies the numbering scheme (labeling) of the functions as listed in the table to be consistent other TS and the surveillance test scheduling system. This change is the same as proposed for Unit 1, For the same reasons as Unit 1, the NRC staff finds the changes to be acceptable.

h. Table 4.3.5.5-1, Control Room Emergency Ventillation System Instrumentation Surveillance Requirements

The proposed changes to the numbering of the functions in this table are consistent with the changes to Table 3.3.5.5-1. The staff finds these changes to be editorial and acceptable.

i. Section 3.3.6.2, End-of-Cycle Recirculation Pump Trip System Instrumentation Limiting Condition for Operation

The proposed change (a) eliminates the portion of the footnote regarding the inoperability and manual bypass of the end-of-cycle recirculation pump trip (EOC-RPT) instrumentation, (b) revises the applicability statement to require the EOC-RPT if the minimum critical power ratio (MCPR) limits from the core operating limits report (COLR) require its use, and (c) revises action statements d. and e. to reference the correct TS section.

The footnote to Specification 3.3.6.2 indicates a non-cycle specific statement concerning the inoperability of the EOC-RPT, and thus the non-applicability of this specification. The incorporation of the requirement for the EOC-RPT to be operable if required to satisfy the MCPR limits more correctly describes when the EOC-RPT should be operable and eliminates the need for the footnote. Since the changes to the applicability statement and the footnote better clarify the applicability of the EOC-RPT, the staff finds this change to be acceptable.

In Amendment No. 71, dated July 12, 1982, TS action statements 3.2.3.1 and 3.2.3.2 were created and contained the requirements previously in Specification 3.2.3. However, the references in Specification 3.3.6.2 were not revised at that time. Since the proposed change corrects the previous ommission error, the staff finds it acceptable.

j. Bases Section 3/4.1.4, Control Rod Program Controls

The proposed change corrects a typographical error in the Reference number listed for a General Electric report that provided the details for statements in this section. Since this change corrects a typographical error, the staff finds it 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 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 27052). 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: P. Milano

Date: February 1, 1995