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Docket No. 50-324

August 11, 1976

Carolina Power & Light Company
ATTN: Mr. J. A. Jones
Executive Vice President
336 Fayetteville Street
Raleigh, North Carolina 27602

Gentlemen:

The Commission has issued the enclosed Amendment No. 20 to Facility Operating License No. DPR-62 for the Brunswick Steam Electric Plant Unit 2. This amendment consists of changes to the Technical Specifications in response to your requests dated December 19, 1975 and March 22, 1976.

This amendment (1) lowers the main steamline low pressure isolation setpoint from 850 to 825 psig, (2) makes miscellaneous corrections and clarifications, and (3) clarifies the action to be taken in the event that the Rod Block Monitor is inoperable for more than 24 hours.

Copies of the Safety Evaluation and Federal Register Notice are also enclosed.

Sincerely,

Original signed by

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Enclosures:
1. Amendment No. 20 to DPR-62
2. Safety Evaluation
3. Federal Register Notice

cc w/enclosures:
See next page

CMTrammell

KRG

OFFICE ➤	ORB#1	OELD	ORB#1	DOR:AD/OR		
SURNAME ➤	CMTrammell:tsb	Burkett	ASchwencer	KRGoller		
DATE ➤	7/29/76	8/6/76	8/11/76	8/10/76		

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CAROLINA POWER AND LIGHT COMPANY

DOCKET NO. 50-324

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 20
License No. DPR-62

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

   A. The applications for amendment by Carolina Power and Light Company (the licensee) dated December 19, 1975 and March 22, 1976, comply 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.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Karl R. Goller

Karl R. Goller, Assistant Director
for Operating Reactors
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 11, 1976

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 20 TO FACILITY OPERATING LICENSE NO. DPR-62

DOCKET NO. 50-324

Revise Appendix A as follows:

Remove the following pages and insert identically numbered revised pages:

1.1-3/1.1-4	3.2-9/3.2-10
1.1-17/1.1-18	3.2-11/3.2-12
1.2-3/1.2-4	3.2-13/3.2-14
3.1-3/3.1-4	3.2-15/3.2-16
3.1-5/3.1-6	3.2-17/3.2-18
3.1-7/3.1-8	3.2-41/3.2-42
3.2-1/3.2-2	3.2-49/3.2-50
3.2-5/3.2-6	3.2-51/3.2-52
3.2-7/3.2-8	3.2-59/3.2-60

SAFETY LIMIT

1.1 Fuel Cladding Integrity (Cont'd)

C. Whenever the reactor is in the cold shutdown condition with irradiated fuel in the reactor vessel, the water level shall not be less than 18 inches above the top of the normal active fuel zone.

LIMITING SAFETY SYSTEM SETTING

2.1 Fuel Cladding Integrity (Cont'd)

C. Reactor low water level #1 scram setting shall be $\geq$ 12.5" on level instruments.

D. Turbine stop valve closure scram setting shall be $\leq$ 10 percent valve closure except that this is bypassed when power $\leq$ 30 percent.

E. Turbine control valve

1. Fast closure - Results from low hydraulic oil pressure.
2. Loss of control oil pressure - setting shall be $\geq$ 500 psig.
3. For Brunswick Unit No. 2 - fast closure will initiate select rod insert and a reactor protection system trip.
4. The turbine control valve scram function is bypassed when reactor power $\leq$ 30 percent.

F. Main steam isolation scram setting shall be $\leq$ 10 percent valve closure.

G. Main steam isolation on main steam line low pressure at inlet to turbine valves. Pressure setting shall be $\geq$ 825 psig.

H. Reactor low water level #3 initiation of LPCI, core spray and auto blowdown shall be set at or above -147.5 inches indicated level.

I. Reactor low water level #2 initiation of HPCI and RCIC shall be set at or above -38 inches indicated level.

BASES:

1.1 FUEL CLADDING INTEGRITY SAFETY LIMIT

The fuel cladding integrity limit is set such that no calculated fuel damage would occur as a result of an abnormal operational transient. Because fuel damage is not directly observable, a step-back approach is used to establish a safety limit such that the minimum critical power ratio (MCPR) is no less than 1.05. MCPR > 1.05 represents a conservative margin relative to the conditions required to maintain fuel cladding integrity.

The fuel cladding represents one of the physical barriers which separate radioactive materials from environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use-related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the protection system safety settings. While fission product migration from cladding perforation is just as measurable as that from use-related cracking, the thermally-caused cladding perforations signal a threshold, beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding safety limit is defined with margin to the conditions which would produce onset of transition boiling (MCPR of 1.0). These conditions represent a significant departure from the condition intended by design for planned operation.

Onset of transition boiling results in a decrease in heat transfer from the clad and, therefore, elevated clad temperature and the possibility of clad failure. However, the existence of critical power, or boiling transition, is not a directly observable parameter in an operating reactor. Therefore, the margin to boiling transition is calculated from plant operating parameters such as core power, core flow, feedwater temperature, and core power distribution. The margin for each fuel assembly is characterized by the critical power ratio (CPR) which is the ratio of the bundle power which would produce onset of transition boiling divided by the actual bundle power. The minimum value of this ratio for any bundle in

BASES:

2.1

LIMITING SAFETY SYSTEM SETTINGS RELATED TO FUEL CLADDING INTEGRITY (Cont'd)

2. Select Rod Insert

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 startup and fuel warranty service associated with each control rod pattern, on RSCS considerations, and 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.

F & G. Main Steamline Isolation on Low Pressure and Main Steamline Isolation Scram

The low pressure isolation of the main steamlines at 825 psig was provided to protect against rapid reactor depressurization and the resulting rapid cooldown of the vessel. Advantage is taken of the scram feature that occurs when the main steamline isolation valves are closed, to provide for reactor shutdown so that high power operation at low reactor pressure does not

BASES:

2.1

LIMITING SAFETY SYSTEM SETTINGS RELATED TO FUEL CLADDING INTEGRITY (Cont'd)

occur, thus providing protection for the fuel cladding integrity safety limit. Operation of the reactor at pressures lower than 825 psig requires that the reactor mode switch be in the STARTUP position, where protection of the fuel cladding integrity safety limit is provided by the IRM high neutron flux scram. Thus, the combination of main steamline low pressure isolation and isolation valve closure scram assures the availability of neutron flux scram protection over the entire range of applicability of the fuel cladding integrity safety limit. In addition, the isolation valve closure scram anticipates the pressure and flux transients that occur during normal or inadvertent isolation valve closure.

BASES:

1.2 Safety Limit on Reactor Coolant System Integrity (Cont'd)

The RHR system is normally operated at pressures of 75 psig or less while in the shutdown cooling mode. Should, however, the condenser system be isolated from the reactor system, it will be necessary to cool down the primary system by utilizing the steam condensing mode of the RHR system. Due to the RCIC inlet steam limitations, it may be necessary to go into the shutdown cooling mode at higher pressures. It is anticipated that this transition will be made at pressures of 125 psig or less. Instrumentation exists to automatically isolate the RHR system should pressure in the reactor vessel exceed 135 psig, which is well below the design pressure of all portions of the RHR piping system. Accordingly, operation of the RHR system will be limited to 125 psig at which time the operator will shut the isolation valves.

BASES:

2.2 Limiting Safety System Setting Related to Reactor Coolant System Integrity

The ASME Boiler and Pressure Vessel Code requires that each vessel designed to meet Section III be protected from overpressure. The code permits a peak allowable pressure of 110 percent of vessel design pressure. The code specifications for safety valves require that: (1) the lowest safety valve be set at or below vessel design pressure and (2) the highest safety valve be set to open at or below 105 percent of vessel design pressure.

The safety relief valves are set to open by self-actuation (overpressure safety mode) in the range from 1080 to 1100 psig. This satisfies the ASME Code specifications for safety valves, since the lowest set valve opens at less than 1250 psig (reactor coolant system design pressure) and the highest set valve opens at less than 1313 psig (105 percent of reactor coolant system design pressure).

BSEP 1 & 2

Change #1 & #5 Change #1

TABLE 3.1-1

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT REQUIREMENTS

Trip Function	Trip Settings	Modes in Which Functions Must be Operable: Refuel (1)	Startup	Run	Min. No. Operable Instrument Channels Per Trip System (2)	Required Conditions When Minimum Conditions for Operation Are Not Satisfied (3)
1. Mode switch C72A-S1		X	X	X	1	A
2. Manual trip C72A-S3A, B		X	X	X	1	A
3. IRM High flux	$\leq$120/125 of scale	X	X	(12)	3	A
Inoperative		X	X	(12)	3	A
4. APRM						
High Flux (4,14) (flow bias)	$\leq(0.66W+54)\frac{(2.60)}{MTPF}$			X	2	B
(fixed)(14)	$\leq$120% of rated power			X	2	B
Inoperative		X		X	2(5)	B
Downscale	$\geq$3/125 of scale			X(13)	2	B
Startup	$\leq$15% of rated power	X	X		2	A
5. High reactor pressure B21-PS-N023A, B,C,D	$\leq$1045 psig	X(10)	X	X	2	A
6. High drywell pressure C72-PS-N002A, B,C,D	$\leq$2 psig	X(11)	X(11)	X	2	A
7. Reactor low water level #1 B21-LIS-N017A, B,C,D	$\geq$12.5 inch (6)	X	X	X	2	A

Amendment No. 20

AUGUST 1975

TABLE 3.1-1 (Cont'd)

Trip Function	Trip Settings	Modes in Which Functions Must be Operable: Refuel (1)	Startup	Run	Min. No. Operable Instrument Channels Per Trip System (2)	Required Conditions When Minimum Conditions for Operation Are Not Satisfied (3)
8. Scram discharge volume high level C11/C12-LSH-N013A,B,C,D	≤ 109 Gallons	X	X	X	2	A
9. Main steamline high radiation D12-RM-K603A,B,C,D	≤ 3x normal background at rated power	X	X	X	2	C
10. Main steamline isolation valve closure B21-ZS-F022A,B,C,D B21-ZS-F028A,B,C,D	≤ 10% valve closure (7)	X	X	X	4	C
11. Turbine stop valve closure EHC-SVOS-1X, 2X, 3X, 4X	≤ 10% valve (9) closure			X	4	D

TABLE 3.1-1 (Cont'd)

Trip Function	Trip Settings	Modes in Which Functions Must be Operable Refuel (1) Startup Run	Min. No. Operable Instrument Channels Per Trip System (2)	Required Conditions When Minimum Conditions For Operations are not Satisfied (3)
First stage turbine pressure permissive C72-PS-N003A,B,C,D	(9)	X	2	D
12. Turbine Control valve fast closure EHC-PSL-1756 EHC-PSL-1757 EHC-PSL-1758 EHC-PSL-1759	$\geq$ 500 psig (8) control oil pressure	X	2	D

NOTES:

(1) When the reactor is subcritical and the reactor water temperature is less than 212°F, only the following trip functions need to be operable:

A. Mode switch in SHUTDOWN

B. Manual scram

C. High flux IRM

D. Scram discharge volume high water level

It is possible during reactor operation to switch to the refuel mode and remain critical. The requirement to have all other scram functions operable in the refuel mode is therefore to assure that shifting to this mode during reactor operation does not diminish the protection afforded by the RPS.

(2) There shall be two operable, one operable and one tripped, or two tripped trip systems for each function. However, when necessary, one channel may be inoperable without tripping the instrument channel for two (2) hours to conduct required functional tests and calibrations provided that at least one other operable channel in the same trip system is monitoring that parameter.

TABLE 3.1-1 (Cont'd)

NOTES (Cont'd)

(3) When the requirements in the column "Minimum Number of Operating Instrument Channels Per Trip System" cannot be met, the appropriate actions listed below shall be taken:

A. Initiate insertion of operable rods and complete insertion of all operable rods within eight hours.

B. Reduce power level to IRM range and place mode switch in the STARTUP position within eight hours.

C. Reduce turbine load and close main steam line isolation valves within eight hours.

D. Reduce reactor power to less than 30% of rated within eight hours.

(4) "W" is the reactor driving loop flow in percent of rated (see Specification 2.1.A.1).

(5) To be considered operable, an APRM must have at least 2 LPRM inputs per level and at least a total of 11 LPRM inputs.

(6) Twelve and one half inches on the water level instrumentation is 177 inches above the top of the active fuel.

(7) A main steam isolation valve closure bypass is permitted when the reactor mode switch is in either the SHUTDOWN, REFUEL, or STARTUP position.

(8) For Unit 2, low control oil pressure initiates select rod insert, and has the provision to delay reactor protection system trip until determination of turbine bypass valve status. The time delay for bypass valve status determination shall be set at 0.00 sec. In both units, this scram is bypassed if the first stage turbine pressure is less than 30 percent of normal rated power.

(9) A turbine stop valve closure bypass is permitted when the first stage turbine pressure is less than 30 percent of normal rated power.

(10) Not required to be operable when the reactor pressure vessel head is not bolted to the vessel.

(11) Not required to be operable when the primary containment integrity is not required.

(12) IRM's are bypassed when APRM's are on scale and the reactor mode switch is in the RUN position.

(13) The APRM downscale trip is automatically bypassed when the IRM instrumentation is operable and ≤ 120/125 of full scale. The APRM downscale trip function is only active when the reactor mode switch is in RUN.

(14) The APRM high flux signal is fed through a time constant curcuit of approximately 6 seconds. The APRM fixed scram does not incorporate the time constant, but responds directly to instantaneous neutron flux.

TABLE 4.1-1

MINIMUM FUNCTIONAL TEST AND CALIBRATION FREQUENCIES FOR THE REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION

Instrument Channel	Functional Test (7)	Calibration (7)	Instrument Check (7)	Group (2)
1. Mode switch in SHUTDOWN C72A-S1	once/operating cycle	NA	NA	A
2. Manual trip C72A-S3A,B	once/3 months	NA	NA	A
3. IRM				
High flux	once per week (1) during refueling and within 1 week prior to each startup	Comparison to APRM on controlled shutdowns	once/day	C
Inoperative	once per week(1) during refueling and within 1 week prior to each startup	NA	NA	C
4. APRM				
High flux	once/week (1)	Heat balance once/week	once/shift	B
Flow bias	once/month	once/operating cycle calibrate flow bias signal	once/shift	B
Inoperative	once/week (1)	NA	NA	B
Downscale	once/week (1)	NA	NA	B
Startup 15% of rated power	once/week (1) during refueling and within 1 week prior to each startup	NA	NA	C
LPRM Signal	NA	once/6 weeks of equivalent full power operation	once/shift	B

Amendment No. 20

Instrument Channel	Functional Test (7)	Calibration (7)	Instrument Check (7)	Group (2)
5. High Reactor pressure B21-PS-N023A,B,C,D	once/month	(3)	NA	A
6. High drywell pressure C72-PS-N002A,B,C,D	once/month	(3)	NA	A
7. Reactor low water level #1 B21-LIS-N017A,B,C,D	once/month (4)	(3)	once/day	A
8. High water level in scram discharge volume C11/C12-LSH-N013A,B,C,D	once/3 months	(3)	NA	A
9. High steamline radiation D12-RM-K603A,B,C,D	once/week (1)	(5)	once/shift	B
10. Main steamline isolation valve closure B21-ZS-F022A,B,C,D B21-ZS-F028A,B,C,D	once/month without setpoint verification	Physical inspection once/operating cycle	NA	A
11. Turbine stop valves closure EHC-SVOS- 1x,2x,3x,4x	once/month without setpoint verification	Physical inspection once/operating cycle	NA	A
First stage turbine pressure permissive C72-PS-N003A,B,C,D	once/3 months	(3)	NA	A

LIMITING CONDITION FOR OPERATION

3.2 Protective Instrumentation

Applicability:

Applies to the plant instrumentation which initiates and controls a protective function.

Objective:

To assure the operability of protective instrumentation.

Specifications:

A. Primary Containment Isolation Functions

When primary containment integrity is required, the limiting conditions of operation that initiates primary containment isolation are given in Tables 3.2-1 through 3.2-6.

B. Core and Containment Cooling Systems - Initiation & Control

The limiting conditons for operation for the instrumentation that initiates or controls the core and containment cooling systems are given in Tables 3.2-7 through 3.2-10. This instrumentation must be operable when the system(s) it initiates or controls is required to be operable as specified in Specification 3.5.

SURVEILLANCE REQUIREMENTS

4.2 Protective Instrumentation

Applicability:

Applies to the surveillance requirement of the instrumentation that initiates and controls protective function.

Objective:

To specify the type and frequency of surveillance to be applied to protective instrumentation.

Specifications:

A. Primary Containment Isolation Functions

Instrumentation shall be functionally tested and calibrated as indicated in Tables 4.2-1 through 4.2-6.

System logic shall be functionally tested as indicated in Tables 4.2-1 through 4.2-6.

B. Core and Containment Cooling Systems - Initiation & Control

Instrumentation shall be functionally tested, calbrated and checked as indicated in Tables 4.2-7 through 4.2-10.

System logic shall be functionally tested as indicated in Tables 4.2-7 through 4.2-10.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.2.C. Control Rod Block Actuation	4.2.C. Control Rod Block Actuation
1. The limiting conditions of operation for the instrumentation that initiates control rod block are given in Table 3.2-11.	1. Instrumentation shall be functionally tested, calibrated and checked as indicated in Table 4.2-11. System logic shall be functionally tested as indicated in Table 4.2-11.
2. The minimum number of operable instrument channels specified in Table 3.2-11 for the rod block monitor may be reduced by one for maintenance and/or testing provided that this condition does not last longer than 24 hours.	
3. If one channel of the rod block monitor has been inoperable for more than 24 hours, control rod withdrawal shall be blocked; or the operating power level shall be limited such that the MCPR will remain above 1.05 assuming a single error that results in complete withdrawal of any single operable control rod.	
D. Radiation Monitoring Systems - Isolation & Initiation Functions	D. Radiation Monitoring Systems - Isolation & Initiation Functions
1. The limiting conditions for operation for Reactor Building ventilation system isolation and standby gas treatment system are given in Table 3.2-12.	1. Instrumentation shall be functionally tested, calibrated and checked as indicated in Table 4.2-12. System logic shall be functionally tested as indicated in Table 4.2-12.
E. Drywell Leak Detection	E. Drywell Leak Detection
The limiting conditions of operation for the instrumentation that monitors drywell leak detection are given in Table 3.2-13.	Instrumentation shall be calibrated and checked as indicated in Table 4.2-13.
F. Post Accident Monitoring Instrumentation	F. Post Accident Monitoring Instrumentation
1. The limiting conditions for the instrumentation that provides surveillance information readouts are given in Table 3.2-14.	1. Instrumentation shall be calibrated and checked as indicated in Table 4.2-14.

TABLE 3.2-1

PRIMARY CONTAINMENT ISOLATION INSTRUMENTATION

GROUP I ISOLATION (1)

	Trip Function	Trip Setting	Minimum Number of Operable Instrument Channels per Trip System	Required Action When Minimum Condition for Operation are not Satisfied	Remarks
1.	Reactor low water level #2 B21-LIS-N024A,B B21-LIS-N025A,B	≥ - 38" indicated level	Two	(2.a.)	
2.	Steamline area high temperature B21-TS-N010A,B,C,D B21-TS-N011A,B,C,D B21-TS-N012A,B,C,D B21-TS-N013A,B,C,D	≤ 200 F	Two of four in each of two channels	(2.b.)	
3.	Steamline high flow B21-dPIS-N006A,B,C,D B21-dPIS-N007A,B,C,D B21-dPIS-N008A,B,C,D B21-dPIS-N009A,B,C,D	≤ 140% of rated flow	Two per steamline	(2.b.)	
4.	Main steamline low pressure B21-PS-N015A,B,C,D	≥ 825 psig	Two (3)	(2.b.)	
5.	Main steamline high radiation D12-RM-K603A,B,C,D	≤ 3 x background at rated power	Two (2)	(2.b.)	Has contacts in reactor protection system
6.	Steamline high flow while in STARTUP B21-dPIS-N006A B21-dPIS-N007B B21-dPIS-N008C B21-dPIS-N009D	≤ 40% of rated flow	Two (4)	(2.b.)	

TABLE 3.2-1

PRIMARY CONTAINMENT ISOLATION INSTRUMENTATION

GROUP I ISOLATION (1) (Cont'd)

	Trip Function	Trip Setting	Minimum Number of Operable Instrument Channels per Trip System	Required Action When Minimum Condition for Operation are not Satisfied	Remarks
7.	Low condenser vacuum B21-PS-N056A,B,C,D	≥18" Hg.	Two	(2.b.)	
8.	Turbine building area high temperature B21-TS-3225A,B,C,D B21-TS-3226A,B,C,D B21-TS-3227A,B,C,D B21-TS-3228A,B,C,D B21-TS-3229A,B,C,D B21-TS-3230A,B,C,D B21-TS-3231A,B,C,D B21-TS-3232A,B,C,D	≤200°F	Two of four in each of four channels	(2.b.)	

BSEP-1 & 2

NOTES:

(1) Group I isolation includes:

a. Main steamline isolation valves
b. Main steamline drain valves
c. Reactor water sample valves (only on low water level #2 or high main steam line radiation signal)
d. Mechanical vacuum pump trip (only on high main steamline radiation signal)

Amendment 1

(2) If the minimum number of operable instrument channels is not available for one trip system, that trip system shall be tripped. If the minimum number of operable of tripped instrument channels is not available for both trip systems, the appropriate actions listed below shall be taken:

a. Initiate an orderly shutdown and have reactor in the cold shutdown condition in 24 hours.
b. Initiate an orderly load reduction and have reactor in Hot Standby within 8 hours.

(3) The main steamline low pressure need be available only in the RUN mode.

(4) Not required in RUN mode. Applies to Unit 2 only.

Amendment No. 20

MARCH 1975

TABLE 4.2-1

MINIMUM TEST & CALIBRATION FREQUENCIES
PRIMARY CONTAINMENT ISOLATION INSTRUMENTATION
GROUP I ISOLATION

	Trip Function	Functional Test	Calibration	Instrument Check
1.	Reactor low water level #2 B21-LIS-N024A,B B21-LIS-N025A,B	once/month	(1)	once/day
2.	Steamline area high temperature B21-TS-N010A,B,C,D B21-TS-N011A,B,C,D B21-TS-N012A,B,C,D B21-TS-N013A,B,C,D	once/month without setpoint verifications	once/operating cycle	N/A
3.	Steamline high flow B21-dPIS-N006A,B,C,D B21-dPIS-N007A,B,C,D B21-dPIS-N008A,B,C,D B21-dPIS-N009A,B,C,D	once/month	(1)	N/A
4.	Main steamline low pressure B21-PS-N015A,B,C,D	once/month	(1)	N/A
5.	Main steamline high radiation D12-RM-K603A,B,C,D	check with reactor protection system	once/operating cycle	once/day
6.	Steamline high flow while in STARTUP (2) MODE B21-dPIS-N006A B21-dPIS-N007B B21-dPIS-N008C B21-dPIS-N009D	once/month	(1)	N/A

TABLE 4.2-1

MINIMUM TEST & CALIBRATION FREQUENCIES
PRIMARY CONTAINMENT ISOLATION INSTRUMENTATION
GROUP I ISOLATION (Cont'd)

Trip Function	Functional Test	Calibration	Instrument Check
7. Low condenser vacuum B21-PS-N056A,B,C,D	once/month	(1)	N/A
8. Turbine building area high temperature B21-TS-3225A,B,C,D B21-TS-3226A,B,C,D B21-TS-3227A,B,C,D B21-TS-3228A,B,C,D B21-TS-3229A,B,C,D B21-TS-3230A,B,C,D B21-TS-3231A,B,C,D B21-TS-3232A,B,C,D	once/month without setpoint verification	once/operating cycle	N/A

Group I - Isolation logic system functional test to be performed once/operating cycle.

Group I - Isolation included:

a. Main steamline isolation valves.
b. Main steamline drain valves.
c. Reactor water sample valves (low water level #2 or high main steam line radiation).
d. Mechanical vacuum pump trip (only on high main steamline radiation signal).

NOTES: (1) When a functional test shows the setpoints are out of specified limits, a calibration will be performed immediately.

(2) Applies to Unit 2 only.

TABLE 3.2-2

PRIMARY CONTAINMENT ISOLATION INSTRUMENTATION
HIGH PRESSURE COOLANT INJECTION SYSTEM
GROUP IV ISOLATION (2)

Trip Function	Trip Setting	Minimum number of Operable Instrument Channels per Trip System	Required Action When Minimum Conditions for Operation are not Met	Remarks
1. Steamline high d/p (steamline break) E41-dPIS-N004 E41-dPIS-N005	≤300% rated flow	2	(1)	
2. HPCI turbine steam-line low pressure E41-PSL-N001A,B,C,D	100 (± 3) psig	2	(1)	
3. HPCI turbine exhaust diaphragm high pressure E41-PSH-N012A,B,C,D	10 (± 0.7) psig	2	(1)	
4. Suppression pool high ambient temperature E51-TS-N603C,D	≤200F	2	(1)	
5. Suppression pool area vent inlet/outlet high differential temperature E51-dTS-N604C,D	≤50F	2	(1)	
6. Emergency area cooler high temperature E41-TS-N602A,B	≤175F	2	(1)	

Amendment #1

BSEP-1 & 2

Amendment No. 20

TABLE 4.2-2

PRIMARY CONTAINMENT ISOLATION INSTRUMENTATION
HIGH PRESSURE COOLANT INJECTION SYSTEM
GROUP IV ISOLATION (2) (Cont'd)

Trip Function	Trip Setting	Minimum Number of Operable Instrument Channels per Trip System	Required Action When Minimum Conditions for Operation are Not Met	Remarks
7. HPCI equipment room vent inlet/outlet high differential temperature E41-dTS-N601A,B	≤ 50F	2	(1)	
8. HPCI equipment room main steamline area high temperature E41-TS-3314 E41-TS-3315 E41-TS-3316 E41-TS-3317 E41-TS-3318 E41-TS-3354 E41-TS-3488 E41-TS-3489	≤ 175F	2	(1)	
9. Bus power monitor E41-K55 E41-K56	NA	1	(3)	Annunciate only

NOTES: (1) Close isolation valves in system and comply with Specification 3.5.

(2) Group IV isolation includes:

a. HPCI inboard steam isolation valve
b. HPCI outboard steam isolation valve
c. HPCI torus suction valve (does not close on HPCI equipment room main steam line area high temperature)

(3) Monitor bus power daily and comply with Specification 3.5 if power is lost.

TABLE 4.2-2

MINIMUM TEST & CALIBRATION FREQUENCIES
PRIMARY CONTAINMENT ISOLATION INSTRUMENTATION
HIGH PRESSURE COOLANT INJECTION SYSTEM
GROUP IV ISOLATION (2)

	Trip Function	Functional Test	Calibration	Instrument Check
1.	Steamline high d/p (steamline break) E41-dPIS-N004 E41-dPIS-N005	once/month	(1)	once/day
2.	HPCI turine steamline low pressure E41-PSL-N001A,B,C,D	once/month	(1)	N/A
3.	HPCI turbine exhaust diaphragm high pressure E41-PSH-N012A,B,C,D	once/month	(1)	N/A
4.	Suppression pool high ambient temperature E51-TS-N603C,D	once/month without setpoint verificaton	once/operating cycle	N/A
5.	Suppression pool area vent inlet/outlet high differential temperature E51-dTS-N604C,D	once/month without setpoint verification	once/operating cycle	N/A
6.	Emergency area cooler high temperature E41-TS-N602A,B	once/month without setpoint verification	once/3 months	N/A
7.	HPCI equipment room vent inlet/outlet high difference temperature E41-dTS-N601A,B	once/month without setpoint verification	once/3 months	N/A

TABLE 4.2-2

MINIMUM TEST & CALIBRATION FREQUENCIES
PRIMARY CONTAINMENT ISOLATION INSTRUMENTATION
HIGH PRESSURE COOLANT INJECTION SYSTEM
GROUP IV ISOLATION (2) (Cont'd)

	Trip Function	Functional Test	Calibration	Instrument Check
8.	HPCI equipment room main steamline area high temperature E41-TS-3314 E41-TS-3315 E41-TS-3316 E41-TS-3317 E41-TS-3318 E41-TS-3354 E41-TS-3488 E41-TS-3489	once/month without setpoint verification	once/3 months	N/A
9.	Bus power monitor E41-K55 E41-K56	once/month	N/A	N/A

HPCI subsystem auto isolation logic system functional test will be performed once/6 months.

NOTES: (1) When a functional test shows the setpoints are out of specified limits, a calibration will be performed immediately.

(2) Group IV isolation includes:

a. HPCI inboard steam isolation valve
b. HPCI outboard steam isolation valve
c. HPCI torus suction valve (does not close on HPCI equipment room main steam line area high temperature)

BSEP-1 & 2
Amendment #1

TABLE 3.2-3
PRIMARY CONTAINMENT ISOLATION INSTRUMENTATION
REACTOR CORE ISOLATION COOLING SYSTEM
GROUP V ISOLATION (2)

	Trip Function	Trip Setting	Minimum Number of Operable Instrument Channels per Trip System	Required Action When Minimum Conditons for Operation are not Met	Remarks
1.	RCIC steamline high d/p (steamline break) E51-dPIS-N017 E51-dPIS-N018	$\leq$300% rated flow	2	(1)	
2.	RCIC turbine steamline low pressure E51-PS-N019A,B,C,D	50 ($\pm$ 2) psig	2	(1)	
3.	RCIC turbine high exhaust diaphragm pressure, E51-PS-N012A,B,C,D	10 ($\pm$ 4) psig	2	(1)	
4.	Suppression pool high ambient temperature E51-TS-N603A,B	≤ 200F	2	(1)	
5.	Suppression pool area vent inlet/outlet high differential temperature. E51-dTS-N604A,B	≤ 50F	2	(1)	
6.	RCIC equipment room high ambient temperature, E51-TS-N602A,B	≤ 175F	2	(1)	
7.	RCIC equipment room vent inlet/outlet high differential temperature E51-dTS-N601A,B	≤ 50F	2	(1)	

Amendment No. 20

TABLE 3.2-3 (Cont'd)

PRIMARY CONTAINMENT ISOLATION INSTRUMENTATION
REACTOR CORE ISOLATION COOLING SYSTEM
GROUP V ISOLATION (2)

	Trip Function	Trip Setting	Minimum Number of Operable Instrument Channels per Trip System	Required Action When Minimum Conditions for Operation are not Met	Remarks
8.	RCIC equipment room main steamline area high temperature E51-TS-3319 E51-TS-3320 E51-TS-3321 E51-TS-3322 E51-TS-3323 E51-TS-3355 E51-TS-3487	$\leq$ 175F	2	(1)	
9.	Bus power monitor E51-K42 E51-K43	N/A	1	(3)	Annunciate only

NOTES: (1) Close isolation valve in system and comply with Specification 3.5.

(2) Group V includes

a. RCIC inboard steam isolation valve.
b. RCIC outboard steam isolation valve.

(3) Monitor bus power daily and comply with Specification 3.5 if power is lost.

TABLE 4.2-3
MINIMUM TEST & CALIBRATION FREQUENCIES
PRIMARY CONTAINMENT ISOLATION INSTRUMENTATION
REACTOR CORE ISOLATION COOLING SYSTEM
GROUP V ISOLATION (2)

	Trip Function	Functional Test	Calibration	Instrument Check
1.	RCIC steamline high d/p (steamline break) E51-dP15-N017 E51-dP15-N018	once/month	(1)	Once/day
2.	RCIC turbine steamline low pressure E51-PS-N019A,B,C,D	once/month	(1)	N/A
3.	RCIC turbine high exhaust diaphragm pressure E51-PS-N012A,B,C,D	once/month	(1)	N/A
4.	Suppression pool high ambient temperature E51-TS-N603A,B	once/month without setpoint verification	once/operating cycle	N/A
5.	Suppression pool area vent inlet/outlet high differential temperature E51-dTS-N604A,B	once/month without setpoint verification	once/operating cycle	N/A
6.	RCIC equipment room high ambient temperature E51-TS-N602A,B	once/month without setpoint verification	once/3 months	N/A
7.	RCIC equipment room vent inlet/outlet high differential temperature E51-dTS-N601A,B	once/month without setpoint verification	once/3 months	N/A

TABLE 4.2-3 (Cont'd)

MINIMUM TEST & CALIBRATION FREQUENCIES
PRIMARY CONTAINMENT ISOLATION INSTRUMENTATION
REACTOR CORE ISOLATION COOLING SYSTEM
GROUP V ISOLATION (2)

	Trip Function	Functional Test	Calibration	Instrument Check
8.	RCIC equipment room main steamline area high temperature E51-TS-3319 E51-TS-3320 E51-TS-3321 E51-TS-3322 E51-TS-3323 E51-TS-3355 E51-TS-3487	once/month without setpoint verification	once/3 month	NA
9.	Bus power monitor E51-K42/E51-K43	once/month	NA	NA

RCIC subsystem auto isolation logic system functional test will be performed once/6 months.

NOTES: (1) When a functional test shows the setpoints are out of specified limits, a calibration will be performed immediately.

(2) Group V isolation includes:

a. RCIC inboard steam isolation valve
b. RCIC outboard steam isolation valve

TABLE 3.2-4
PRIMARY CONTAINMENT ISOLATION INSTRUMENTATION
REACTOR WATER CLEANUP SYSTEM
GROUP III ISOLATION (2)

	Trip Function	Trip Setting	Minimum Number of Operable Instrument Channels per Trip System	Required Action When Minimum Conditions for Operation are Not Met	Remarks
1.	Reactor low level #1 B21-LIS-N017A,B,C,D	≥ 12.5" indicated level	2	(1)	Has contacts in Group II, Reactor Building ventilation isolation and SGTS initiation systems.
2.	Reactor water cleanup high temperature G31-TIS-N008	≤ 140°F	1	(1)	
3.	Reactor water cleanup high differential flow G31-FDI-R615	≤ 53 gpm	1	(1)	
4.	Standby liquid control system initiated C41-RMS-S1A,B	N/A	N/A	(1)	
5.	Reactor water cleanup space high temperature G31-TS-N600A,B,C,D,E,F	100 - 150°F	2	(1)	
6.	Vent air inlet/outlet high differential temperature G31-dTS-602A,B,C,D,E,F	≤ 50°F	2	(1)	

NOTES:

(1) Close isolation valves in cleanup system and comply with Specification 3.6.B.

(2) Group III isolation includes

a. RWCU outboard isolation valve
b. RWCU inboard isolation valve (does not close on SLC initiation or RWCU high temperature).

TABLE 4.2-4
MINIMUM TEST & CALIBRATION FREQUENCIES
PRIMARY CONTAINMENT ISOLATION INSTRUMENTATION
REACTOR WATER CLEANUP SYSTEM
GROUP III ISOLATION (2)

	Trip Function	Functional Test	Calibration	Instrument Check
1.	Reactor low water level #1 B21-LIS-N017A,B,C,D	This reactor low water level #1 switch is on the same instrument as the PCIS low water level #1 switch and will be functionally tested and calibrated at the same time.		
2.	Reactor water cleanup high temperature G31-TIS-N008	once/month	(1)	N/A
3.	RWCU high differential flow G31-FDI-R615	once/month	(1)	once/day
4.	Standby liquid control system initiated G41-RMS-S1A,B	once/operating cycle	N/A	N/A
5.	RWCU space high temperature G31-TS-N600A,B,C,D,E,F	once/month	(1)	N/A
6.	RWCU vent air inlet/outlet high differential temperature G31-dTS-602A,B,C,D,E,F	once/month	(1)	N/A

RWCU isolation logic system functional test will be performed once/6 months

NOTES:

(1) When a functional test shows the setpoints are out of specified limits, a calibration will be performed immediately.

(2) Group III isolation includes:

a. RWCU outboard isolation valve
b. RWCU inboard isolation valve (does not close on SLC initiation or RWCU high temperature).

TABLE 3.2-11 (Cont'd)

CONTROL ROD BLOCKS INITIATED FROM NEUTRON MONITORING SYSTEM

Trip Function	Minimum Number of Operable Instrument Channels (2)	Modes in Which Function Must Be Operable: Refuel	Startup	Run	Trip Setting	Remarks
c. Detector not in "full in" position, channels A through H, Relays C51-K9E through H, & J through M	6	X	X		Detector motor module limit switch LS-4 not closed (detector not full in)	Bypassed in run mode.
d. Downscale IRM channels A through H, Relay C51-K51	6	X	X		> 3/125 of Scale	Bypassed in run mode and when IRM is in RANGE 1.
3. Average power range monitor						
a. Upscale APRM channels A through F, Relays K1 & K7	4			X	$\leq (0.66W+42)\ \frac{2.60}{MTPF}$	
b. Inoperative APRM channels A through F Relays K2 & K8	4	X	X	X	(1)	
c. Downscale APRM channels A through F Relays K3 & K9	4			X	≥ 3/125 of Full Scale	Only active when mode switch is in RUN
d. Upscale startup APRM channels A through F Relay K18	4	X	X		≤ 12% power	Bypassed when in run mode.

Change #1

Change #1

Change #1 & #5

Change #1

BSEP-1 & 2

Amendment No. 20

AUGUST 1975

CONTROL ROD BLOCKS INITIATED FROM NEUTRON MONITORING SYSTEM

Trip Function	Minimum Number of Operable Instrument Channels (3)	Modes in Which Function Must Be Operable: Refuel	Startup	Run	Trip Setting	Remarks
4. Rod block monitor						
a. Upscale RBM channels A,B Relay K1	2			X (4)	≤ (0.66W+42) $\frac{2.60}{MTPF}$	
b. Downscale RBM channels A,B Relay K2	2			X (4)	≥ 3/125 of full scale	
c. Inoperative RBM channels A,B Relay K3	2			X (4)	(1)	

NOTES:

(1) The inoperative trips are produced by the following conditions:

(a) SRM and IRM

1) Mode switch not in OPERATE
2) High voltage power supply voltage low
3) Circuit boards not in circuit

(b) APRM

1) Mode switch not in OPERATE
2) Less than 11 LPRM inputs
3) Circuit boards not in circuit

(c) RBM

1) Mode switch not in OPERATE
2) Circuit boards not in circuit
3) RBM fails to null
4) Less than required number of LPRM inputs for rod selected.

(2) If the minimum number of channels cannot be met for one out of two trip systems, seven days are allowed before requiring the affected trip system to be tripped. If both trip systems do not meet the minimum number of operable channels for operation, both trip systems shall be tripped.

(3) If the minimum number of channels per trip system cannot be met, see Specifications 3.2.C and 3.3.B.5 for required action.

(4) Only required operable when mode switch is in RUN and reactor power is ≥30%.

TABLE 3.2-14

POST ACCIDENT MONITORING INSTRUMENTATION

	Instrument	Minimum No. of Operable Instrument Channels	Instrument No.	Type and Indication and Range	Notes
1.	Reactor water level	2	B21-LI-R604A,B B21-LR-R615	Indicator -150/0/+60" water Recorder -100" to +200" water	(1) (2)
2.	Reactor pressure	2	B21-PI-R004A,B C32-LPR-R608	Indicator 0-1500 psig Recorder 0-1200 psig	(1) (2)
3.	Containment pressure	2	CAC-PI-2599 CAC-PR-1257	Indicator 0-75 psig Recorder 0-75 psig	(1) (2)
4.	Containment temperature	2	CAC-TR-1258 C91-P602	Recorder 0-400 F Computer 0-400 F	(1) (2)
5.	Suppression chamber atmosphere temperature	2	CAC-TR-1258 C91-P602	Recorder 0-400 F Computer 0-400 F	(1) (2)
6.	Suppression chamber water level	2	CAC-LI-2601-3 CAC-LR-2602	Indicator -6 to 6' Recorder -6 to +6'	(1) (2)
7.	Suppression chamber water temperature	2	CAC-TR-1258 C91-P602	Recorder 0-400 F Computer 0-400 F	(1) (2)
8.	Containment radiation	2	CAC-AR-1259 CAC-AR-1260 CAC-AR-1261	Recorder 10^1 to 10^6 cpm Recorder 10^1 to 10^6 cpm Recorder 10^1 to 10^6 cpm	(1) (2)
9.	Containment oxygen	2	CAC-AR-1259 CAC-AR-1263	Recorder/Indicator 0-5%, 0-25% Recorder/Indicator 0-5%, 0-25%	(1) (2)
10.	Containment hydrogen	2	CAC-AR-1259 CAC-AR-1263	Recorder 0-10% Recorder 0-10%	(1) (2)

TABLE 3.2-14 (Cont'd)

NOTES:

(1) From the date one of these parameters is reduced to one indication, continued operation is permissible during the succeeding 30 days unless such instrumentation is made operable sooner.

(2) If the requirements of note (1) cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a cold condition within 24 hours.

TABLE 4.2-14

MINIMUM TEST AND CALIBRATION FREQUENCY FOR POST ACCIDENT MONITORING INSTRUMENTATION

	Instrument Channel	Instrument Numbers	Calibration Frequency	Instrument Check
1.	Reactor level	B21-LI-R604A,B B21-LR-R615	Once/6 months	Each shift
2.	Reactor pressure	B21-PI-R004A,B C32-LPR-R608	Once/6 months	Each shift
3.	Containment pressure	CAC-PI-2599 CAC-PR-1257	Once/6 months	Each shift
4.	Containment temperature	CAC-TR-1258 C91-P602	Once/6 months	Each shift
5.	Suppression chamber atmosphere temperature	CAC-TR-1258 C91-P602	Once/6 months	Each shift
6.	Suppression chamber water level	CAC-LI-2601-3 CAC-LR-2602	Once/6 months	Each shift
7.	Suppression chamber water temperature	CAC-TR-1258 C91-P602	Once/6 months	Each shift
8.	Containment radiation	CAC-AR-1259 CAC-AR-1260 CAC-AR-1261	Once/6 months	Each shift
9.	Containment oxygen	CAC-AR-1259 CAC-AR-1263	Once/6 months	Each shift
10.	Containment hydrogen	CAC-AR-1259 CAC-AR-1263	Once/6 months	Each shift

TABLE 3.2-15

INSTRUMENTATION THAT INITIATES RECIRCULATION PUMP TRIP

Trip Function	Trip Level Setting	Minimum Number of Operable Instrument Channels per Trip System (1)	Action
Reactor High Pressure B21-PS-N045 A,B,C,D	≤ 1120 psig	1	(2)
Reactor Low Water Level #2 B21-LIS-N024A,B B21-LIS-N025A,B	≥ -38 in. indicated level	1	(2)

NOTES:

1. Whenever the reactor is in the RUN Mode, there shall be one operable trip system for each parameter for each operating recirculation pump. If this cannot be met, the indicated action shall be taken.

2. Reduce power and place the mode selector-switch in a mode other than the RUN Mode.

Amendment No. 20

BASES:

3.2.B Core Standby Cooling System (CSCS) (Cont'd)

Section 3.5. Whenever an instrument in one subsystem is inoperable, the limiting condition for operation as specified in Section 3.5 applies. If an instrument is in more than one subsystem of CSCS, then Section 3.5 is too restrictive and the inoperable channel shall be tripped using special jacks or other permanently installed circuits.

C. Control Rod Blocks

The control rod block functions are provided to prevent excessive control rod withdrawal so that MCPR does not decrease to the safety limit. The trip logic for this function is one out of n; e.g., any trip on one of the six APRMs, eight IRMs, or four SRMs will result in a rod block. The minimum instrument channel requirements for the IRM may be reduced by one for a short period of time to allow for maintenance, testing or calibration. At least one RBM shall be operable when rod withdrawal is to be performed at $\geq$ 30% power to prevent excessive rod withdrawal and subsequent potential fuel damage.

The APRM rod block trip is flow referenced and prevents a significant reduction in MCPR, especially during operation at reduced flow. The APRM provided gross core protection; i.e., limits the gross core power increase from withdrawal of control rods in the normal withdrawal sequence. The rod block trips are set so that MCPR is maintained greater than the safety limit.

The IRM rod block function provides local as well as gross core protection. The scaling arrangement is such that the trip setting is less than a factor of 10 above the indicated level. Analysis of the worst case accident results in rod block action before MCPR approaches the safety limit.

BASES:

3.2.C Control Rod Blocks (Cont'd)

A downscale indication on an APRM or IRM is an indication the instrument has failed or the instrument is not sensitive enough. In either case, the instrument will not respond to changes in control rod motion; thus, control rod motion is prevented.

When minimum conditions for operation are not met, the required action is to leave the channel in the tripped condition until it is repaired.

D. Radiation Monitoring Systems - Isolation And Initiation Functions

Change #1

Two radiation monitors are provided which initiate isolation of the Reactor Building and operation of the standby gas treatment system. The monitors are located in the Reactor Building ventilation duct. Any one upscale trip will initiate the isolation. Trip settings for the monitors in the ventilation duct are based upon initiation of the normal ventilation isolation and standby gas treatment system operation to limit the dose rate at the nearest site boundary to less than the dose rate allowed by 10CFR20.

If the minimum conditions for operation are not met, the Reactor Building ventilation system shall be isolated and the standby gas treatment system operated until the instrumentation is repaired.

E. Drywell Leak Detection Monitors

The instrumentation that monitors drywell leak detection provides the information to determine whether Specification 3.6.C. (Coolant Leakage) is met, therefore, the limiting condition for operation is the same as Specification 3.6.C.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 20 TO FACILITY OPERATING LICENSE NO. DPR-62

CAROLINA POWER AND LIGHT COMPANY

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NO. 2

DOCKET NO. 50-324

Intoduction

By letters dated December 19, 1975, and March 22, 1976, Carolina Power and Light Company (the licensee) requested changes to the Technical Specifications for operation of the Brunswick Steam Electric Plant, Unit No. 2 (BSEP-2), located in Brunswick County, North Carolina. The requested changes include lowering the main steamline low pressure isolation setpoint from 850 to 825 psig, making miscellaneous corrections and clarifications, and modifying the operability requirements for the Rod Block Monitor (RBM).

Discussion and Evaluation

We have reviewed the above proposed changes to the Technical Specifications requested by the licensee. Our assessment of the acceptability of each change is discussed below.

1. Main Steamline Isolation Setpoint

   The main steamline low pressure isolation circuit is designed to close the main steamline isolation valves (MSIVs) in the event that steam pressure falls significantly below normal operating values. The purpose of such isolation is to reduce the rate of depressurization and cooldown of the reactor vessel which could otherwise result in exceeding the allowable rate of change of reactor vessel temperature. Such an event could be caused by a failure of the turbine pressure regulator which normally controls steam pressure between 920 and 950 psig. The present setpoint for MSIV isolation is 850 psig or greater. The licensee has requested that this setting be lowered to 825 psig or greater on the basis that the 850 psig or greater setting does not provide sufficient margin between the isolation trip point and normal operating pressure at the turbine inlet. The licensee states that a wider pressure margin is needed to accommodate normal pressure variations, and that the 825 psig setting should prove adequate to preclude spurious isolations and the resulting reactor scrams.

In the March 22, 1976 supplemental letter, the licensee reported that the analysis of the abnormal operational transient caused by pressure regulator failure, as described in FSAR for BSEP-2, was based on a setpoint value of 825 psig for MSIV closure, and that the BSEP FSAR incorrectly identifies the setpoint as 850 psig. The consequences of this operational transient were previously reviewed by the NRC staff and shown to be acceptable. Based on (1) the fact that this analysis was performed for an 825 psig setting (the requested value), and (2) a bounding analysis performed for a similar plant, we conclude that changing the pressure setpoint for MSIV isolation from 850 to 825 psig is acceptable. This same change has been approved by us for three other boiling water reactors.

2. Miscellaneous Corrections and Clarifications

The miscellaneous corrections and clarifications requested by the licensee are identified in detail in the enclosure to this Safety Evaluation. They are requested to (1) add instruments inadvertently omitted from several original instrument lists, (2) correct several instrument designations and (3) modify the language of several specifications to improve clarity. None of these changes involve any actual changes in intent of the original Technical Specifications. No action has been taken in this amendment with respect to the clarification requested concerning the time delay associated with the turbine control valve fast closure scram. That time delay has since been eliminated and the Technical Specifications have already been changed to reflect this. The balance of the requested changes are acceptable.

3. Rod Block Monitor Operability Requirements

a. Description of the Rod Block Monitor System (RBM)

The RBM is a system which is operable when the reactor mode switch is in the RUN position and reactor power is 30% or higher. The RBM is designed to prevent local fuel damage as a result of a single rod withdrawal error under the worst permitted condition of RBM bypass. The RBM also provides a signal to permit operator evaluation of the change in the local relative power level during control rod movement. The RBM prevents local fuel damage by stopping control rod withdrawal before the linear heat generation rate (LHGR) for any fuel rod exceeds the LHGR required to cause more than the design limit plastic strain of the cladding and before a fuel bundle power is such that the bundle is in a boiling transition heat transfer regime and, consequently, susceptible to fuel cladding perforation.

Whenever a control rod is selected for movement, the signals from the detectors in the adjacent Local Power Range Monitor (LPRM) strings (maximum of 4) are automatically divided between two RBM channels. There are a total of 16, 12, or 8 LPRM detectors available to supply information to the two RBM channels, depending on the location of the selected control rod. One RBM channel averages signals from the A and C level LPRMs while the other channel averages signals from the B and D level LPRMs. The average of the input signals to the RBM is modified to read the same as a reference Average Power Range Monitor (APRM). This assures that the RBM reads core average power whenever a control rod is selected for movement. The RBM automatically limits the local power change by restricting the local average neutron flux increase to a controlled amount. If the local power change is too great, the RBM prevents further control rod withdrawal.

The reactor operator has a display of the readings of up to four LPRM strings available on the console. The display presents the readings of the same LPRM strings that provide signals to the two RBM channels whenever a control rod is selected for movement. Thus, bypassed detectors are readily apparent to the operator as well as changes in individual detector readings as a control rod is moved.

The RBM has three trip levels where rod withdrawal permissives are removed. The trip levels may be adjusted, are nominally 8% of rated reactor power apart, and are automatically varied with reactor coolant flow. The lower two trip setpoints can be bypassed by manual operation of a reset button. Such passing of the trip setpoints can occur because a reset permissive is actuated. This is indicated by a light when the RBM senses a power level 2% less than the trip setpoint. The highest power trip setpoint cannot be reset or bypassed. A trip signal from either of the two RBM channels will inhibit further control rod withdrawal at this highest power trip setpoint.

A count of the available LPRM detectors is automatically made and rod motion inhibited when there are too few detectors. Control rod withdrawal is also inhibited when an RBM is inoperable and not bypassed. Only one RBM may be manually bypassed at any time. At low reactor power where a control rod withdrawal error could not result in local fuel damage, the RBM is automatically bypassed. The RBM is also automatically bypassed if a selected control rod has one or more adjacent fuel bundles comprising the outer boundary of the reactor core. An RBM which reads downscale is considered to have failed and control rod withdrawal is therefore inhibited.

b. The RBM and Rod Withdrawal Error

Since the RBM is designed to prevent local fuel damage as a result of a single control rod withdrawal error from a region of high neutron flux, an analysis of this event is conducted. This analysis is performed assuming that the maximum worth control rod is fully inserted while adjacent control rods are withdrawn in a manner such that full rated power is attained with operating limits attained near the inserted control rod. This is an abnormal control rod pattern, but control rod worth is maximized thereby resulting in a conservative analysis. The analysis is performed using a three-dimensional coupled nuclear-thermal-hydraulic representation of the core in order to determine neutron flux levels at instrument locations. Thus, a complex calculation is performed to determine the consequences of a control rod withdrawal error. This calculation results in an RBM trip setpoint which assures that no local fuel damage will occur in the event of a control rod withdrawal error since the RBM responds automatically and nearly instantaneously to local power changes.

c. Proposed Change

Present Technical Specifications with respect to RBM operability require that both RBM channels be operable if the reactor is above 30% power. However, Specification 3.2.C.2 allows the number of operable channels to be reduced to one for maintenance or testing of the other provided that this condition does not last longer than 24 hours in any 30 day period. Specification 3.3.B.5 places more stringent operability requirements on the RBM during power operation with a limiting control rod pattern.* These are:

a. Both RBM channels shall be operable; or

b. Control rod withdrawal shall be blocked; or

c. The operating power level shall be limited so that the minimum critical power ratio (MCPR) will remain above 1.05 assuming a single error that results in complete withdrawal of a single operable control rod.

Thus, present Technical Specifications (assuming a limiting control rod pattern does not exist) place a limit on out-of-service time (24 hours in any 30 day period) for the RBM, but no action is specified if this time is exceeded. This being the case, the

*A limiting control rod pattern is one which results in the core operating at a thermal-hydraulic limit.

licensee has requested that the specification be clarified, and has proposed the following:

If the number of RBM instrument channels has been reduced by one in one of the trip systems for maintenance or testing for more than 24 hours in a 30 day period, the RBM instrument channel may be bypassed and rods may be withdrawn if calculations by a qualified individual demonstrate that such a withdrawal is permissible.

The basis for this proposal is the licensee's conclusion that the RBM function can readily be performed by a qualified individual when a RBM channel is inoperable.

d. Evaluation

If the RBM were not available, the reactor operator would have to perform all of the functions of the RBM by working directly with the readings available to him on the reactor console for the applicable LPRM detectors. The operator would have to account for bypassed detectors, average the readings of the appropriate detectors, and normalize the average reading to a reference flow-biased power provided by an APRM to obtain the initial power level. The operator would then have to move the selected control rod a number of notches and repeat the procedure to obtain the change in the initial power level that had been previously determined. He would then have to check his new level to see if he has or has not exceeded his trip setpoint. He would not know if he had exceeded fuel damage limits except after the fact. The operator cannot act in time to stop control rod movement before fuel damage limits are exceeded because the calculations needed to make this judgement are made following control rod movement.

Moreover Brunswick Unit 2 has 137 control rods, many of which are moved during reactor startup and power ascension. During this period, the operator has to contend with a dynamic situation where recirculation flow may be changed, xenon buildup and redistribution may occur, and control rod movements are made. All of these considerations enter into assessing local power changes in various parts of the reactor core and are also an indication of the complexity of the problem. An additional factor of importance to the operator is that limiting control rod patterns may occur a number of times during a complex control rod maneuver.

Based on the above, we cannot agree with the licensee's proposal to use the calculations of a qualified individual in place of the RBM for determining the acceptability of rod withdrawal. Accordingly, we have modified the licensee's proposal to require that, in the event a RBM channel is inoperable for more than 24 hours, control rod withdrawal shall be blocked or the operating power level shall be limited such that the MCPR will remain above 1.05 assuming a single error that results in complete withdrawal of a single operable control rod. These additional controls are the same as those required when operating with a limiting control rod pattern. With these controls in effect, the possibility of a single rod withdrawal error is eliminated, and it is therefore acceptable to allow operation beyond 24 hours with a RBM channel inoperable. This modified specification provides the clarification requested by the licensee of the action to be taken in the event a RBM should be inoperable more than 24 hours, and represents an additional restriction not presently included in the Technical Specifications. Since the change does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the change does not involve a significant hazards consideration. The licensee has agreed with this modification to his proposal.

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and pursuant to 10 CFR §51.5(d)(4) that an environmental statement, negative declaration, or environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: August 11, 1976

ENCLOSURE

MISCELLANEOUS CORRECTIONS AND CLARIFICATIONS

1. The instrumentation lists in several tables do not include all of the instruments which monitor specific activities. In order to make these lists complete, the following instruments should be added:

   a. Under Item 8 on Pages 3.2-10 and 3.2-12, HPCI equipment room main steamline area high temperature instruments, add "E41-TS-3488" and "E41-TS-3489."

   b. Under Item 8 on Pages 3.2-14 and 3.2-16, RCIC equipment room main steamline area high temperature instruments, add "E51-TS-3487."

   c. Under Item 8 on Pages 3.2-49 and 3.2-51, containment radiation instruments, add "CAC-AR-1259." In addition, on Page 3.2-49 in the Type and Indication Range column corresponding to Instrument CAC-AR-1259, add "Recorder 10^1 to 10^6 cpm."

2. On pages 3.2-17 and 3.2-18, Item 3, the instrument number for measuring the reactor water cleanup differential flow should be changed from "G31-dFS-N603" to "G31-FDI-R615."

3. On page 3.1-4 and page 3.1-8, Item 11, the turbine stop valve switch designations should be changed from "EHC-SVOS-1, 2, 3, 4" to "EHC-SVOS-1X, 2X, 3X, 4X."

4. On pages 3.2-49 and 3.2-51, Item 6, the suppression chamber water level instrument number should be changed from "CAC-LI-2601" to "CAC-LI-2601-3."

5. The steamline area high temperature instrumentation listed on Page 3.2-7 and the turbine building area high temperature on Page 3.2-8 indicate an instrument check once per day. However, these instruments have no readouts and, as with similar instrumentation, the instrument check for these items should read "not applicable."

6. Item 10 on Page 3.2-49 concerns containment hydrogen instrumentation. This item indicates the presence of both a recorder and an indicator but, in fact, indicators are not present on these instruments. The reference to indicators should be deleted.

7. The turbine control valve fast closure scram is bypassed when the turbine is below 30% power. In order to make this clear, on Page 1.1-3, add Section 2.1.E.4 to read, "The turbine control valve scram function is bypassed when reactor power $\leq$ 30 percent".

8. On page 3.1-5 in Item 12, the turbine control valve fast closure scram function is shown as being required in the REFUEL, STARTUP, and RUN modes of operation. Since the reactor must be in the RUN mode for turbine power to be above 30% and this scram is bypassed automatically

below 30% power, delete the operability requirement for the turbine control valve fast closure scram function in the REFUEL and START-UP modes.

9. The next to the last sentence on Page 1.2-3 states in part "...the RHR piping is designed for pressures in excess of 600 psig..." While this statement does apply for part of the system, it is incorrect for some portions. Therefore, replace this next to the last sentence with the following: "Instrumentation exists to automatically isolate the RHR system should the pressure in the reactor vessel exceed 135 psig which is well below the design pressure of all portions of the RHR piping system."

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-324

CAROLINA POWER AND LIGHT COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 20 to Facility Operating License No. DPR-62 issued to Carolina Power and Light Company which revised Technical Specifications for operation of the Brunswick Steam Electric Plant, Unit No. 2, located in Brunswick County, North Carolina. The amendment is effective as of its date of issuance.

This amendment (1) lowers the main steamline low pressure isolation setpoint from 850 to 825 psig, (2) makes miscellaneous corrections and clarifications, and (3) clarifies the action to be taken in the event that the Rod Block Monitor is inoperable for more than 24 hours.

The applications for the amendment comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Notice of Proposed Issuance of Amendment to Facility Operating License in connection with item (1) above was published in the FEDERAL REGISTER on on March 29, 1976 (41 FR 12931). No request for a hearing or petition for leave to intervene was filed following notice of this proposed action.

Prior public notice of items (2) and (3) was not required since this amendment does not involve significant hazards considerations.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement, negative declaration or environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the applications for amendment dated December 19, 1975 and March 22, 1976, (2) Amendment No. 20 to License No. DPR-62, and (3) the Commission's Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W. Washington, D. C. 20555, and at the Southport Brunswick County Library, 109 W. Moore Street, Southport, North Carolina 28461. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 11th day of August 1976.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors