

June 23, 1976

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Docket No. **50-324**

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

T. Verdery

Gentlemen:

The Commission has issued the enclosed Amendment No. 17 to Facility Operating License No. DPR-62 for the Brunswick Steam Electric Plant Unit No. 2. The amendment consists of changes to the Technical Specifications and is in response to your request dated May 21, 1976.

This amendment clarifies instrumentation operability requirements during periods of functional testing, and reduces the period of time a Low Pressure Coolant Injection pump can be inoperable during reactor operation from thirty to seven days.

Copies of the related Safety Evaluation and the 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. 17 to DPR-62
- 2. Safety Evaluation
- 3. Federal Register Notice

cc: See next page

not necessary. See background attached CMTrammell

6/11 OFFICE → x27433	ORB#1 CMTrammell	PSB WB	RSB RBaer	OELD Mitchell	ORB#1 ASchwencer
SURNAME →	CMTrammell:tb	WButler	RBaer	Mitchell	ASchwencer
DATE →	6/11/76	6/14/76	6/17/76	6/21/76	6/23/76

June 23, 1976

cc w/enclosures:

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Mr. W. A. Kopp, Jr.
Chairman, Board of County
Commissioners of Brunswick County
Bolivia, North Carolina 28422

cc w/enclosures & incoming:
Office of Intergovernmental
Relations
116 West Jones Street
Raleigh, North Carolina 27603



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

CAROLINA POWER AND LIGHT COMPANY

DOCKET NO. 50-324

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NO. 2

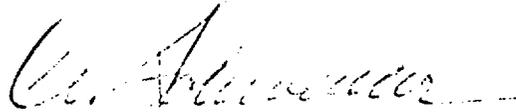
AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 17
License No. DPR-62

1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment by Carolina Power & Light Company (the licensee) dated May 21, 1976, 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 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. After weighing the environmental aspects involved, 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 a change 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



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

Attachment:
Changes to the
Technical Specifications

Date of Issuance: June 23, 1976

ATTACHMENT TO LICENSE AMENDMENT NO. 17

FACILITY OPERATING LICENSE NO. DPR-62

DOCKET NO. 50-324

Revise Appendix A as follows:

Remove the following pages and replace with identically numbered revised pages:

3.1-5 and 3.1-6
3.2-5 and 3.2-6
3.5-3 and 3.5-4

TABLE 3.1-1 (Cont'd)

Trip Function	Trip Settings	Modes in Which Functions Must Be Operable			Min. No. Operable Instrument Channels Per Trip System (2)	Required Conditions When Minimum Conditions For Operations are not Satisfied (3)
		Refuel (1)	Startup	Run		
First stage turbine pressure permissive CAD PS-N008A,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	≥ 850 psig (8) control oil pressure	X	X	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 1.1-1 (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 20% 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 12 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 IRM position.
- (13) The APRM downscale trip is automatically bypassed when the IRM instrumentation is operable and $\leq 100/105$ of full scale. The APRM downscale trip function is only active when the reactor mode switch is in IRM.
- (14) The APRM high flux signal is fed through a time constant circuit of approximately 6 seconds. The APRM high flux scram does not incorporate the time constant, but responds directly to instantaneous neutron flux.

TABLE 3.2-1

PRIMARY CONTAINMENT ISOLATION INSTRUMENTATIONGROUP I ISOLATION (1)

<u>Trip Function</u>	<u>Trip Setting</u>	<u>Minimum Number of Operable Instrument Channels per Trip System</u>	<u>Required Action When Minimum Condition for Operation are not Satisfied</u>	<u>Remarks</u>
1. Reactor low water level #2 B21-LIS-NO24A,B B21-LIS-NO25A,B	≥ - 38" indicated level	Two	(2.a.)	
2. Steamline area high temperature B21-TS-NO10A,B,C,D B21-TS-NO11A,B,C,D B21-TS-NO12A,B,C,D B21-TS-NO13A,B,C,D	≤ 200 F	Two of four in each of two channels	(2.b.)	
3. Steamline high flow B21-dPIS-NO06A,B,C,D B21-dPIS-NO07A,B,C,D B21-dPIS-NO08A,B,C,D B21-dPIS-NO09A,B,C,D	≤ 140% of rated flow	Two per steamline	(2.b.)	
4. Main steamline low pressure B21-PS-NO15A,B,C,D	≥ 850 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-NO06A B21-dPIS-NO07B B21-dPIS-NO08C B21-dPIS-NO09D	≤ 40% of rated flow	Two (4)	(2.b.)	

BSEP-1 & 2

Amendment #1

Amendment #1

TABLE 3.2-1

PRIMARY CONTAINMENT ISOLATION INSTRUMENTATIONGroup I Isolation (1) (Cont'd)

<u>Trip Function</u>	<u>Trip Setting</u>	<u>Minimum Number of Operable Instrument Channels per Trip System</u>	<u>Required Action When Minimum Condition for Operation is not Satisfied</u>	<u>Remarks</u>
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.)	

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 steamline radiation signal)
- d. Mechanical vacuum pump trip (only on high main steamline radiation signal)

(2) If the minimum number of operable instrument channels is not available for one trip system, that trip system shall be tripped. 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 operable channel in the same trip system is monitoring that parameter. If the minimum number of operable or 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.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENTS
<p>3.5.A <u>Core Spray and LPCI Subsystem (Cont'd)</u></p> <p>4. From the date that one of the RHR (LPCI) pumps is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding 7 days provided that during such 7 days the remaining active components of the LPCI subsystem and all active components of both core spray subsystems and the diesel generators are operable.</p> <p>5. From the date that the LPCI subsystem is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding seven days, provided that during such seven days all active components of both core spray subsystems, the containment cooling subsystems (including two LPCI pumps) and the diesel generators are operable.</p>	<p>4.5.A <u>Core Spray and LPCI Subsystem (Cont'd)</u></p> <p>Each pair of LPCI pumps discharging to a common header shall deliver 17,000 gpm against a system head corresponding to a reactor vessel pressure of 34.7 psig.</p> <p>e. Recirculation pump discharge valves travel from full open to full closed. Once/refueling outage</p> <p>4. When one of the RHR (LPCI) pumps is made or found to be inoperable at a time when it is required to be operable, the remaining active components of the LPCI subsystem, the containment cooling subsystem, both core spray subsystems and the diesel generators shall be demonstrated to be operable immediately and the operable LPCI pumps daily thereafter, until the pump is returned to an operable condition.</p> <p>5. When the LPCI subsystem is made or found to be inoperable, both core spray subsystems, the containment cooling subsystem and the diesel generators shall be demonstrated to be operable immediately and daily thereafter until the LPCI subsystem is returned to an operable condition.</p>

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
<p>3.5.A <u>Cold Spray and LPCI Subsystem (Cont'd)</u></p> <p>6. If the requirements of 3.5.A.1 through 5 cannot be met, an orderly shutdown of the reactor shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.</p>	
<p>3.5.B <u>Containment Cooling Subsystem</u></p> <p>1. Both subsystems of containment cooling mode, each including two RHR pumps and two RHR service water pumps, shall be operable whenever there is irradiated fuel in the reactor vessel and the reactor vessel is not vented except as in 3.5.B.2 below.</p> <p>2. Should one RHR pump and/or one RHR service water pump of the components required in 3.5.B.1 above be made or found inoperable, reactor power operation is permissible only during the succeeding 30 days provided that during such 30 days all remaining active components of the suppression pool cooling mode is operable.</p> <p>3. Should one of the containment cooling subsystems be made or found to be inoperable, reactor power operation is permissible for a period not to exceed seven days provided that during such seven days the other containment cooling subsystem is operable.</p>	<p>4.5.B <u>Containment Cooling Subsystem</u></p> <p>1. The subsystem of containment cooling mode is tested in conjunction with the test performed on the LPCI systems and given in 4.5.A.3.a, b, and c above. Residual heat removal service water pumps, two pumps operating in parallel will be included in testing, supplying 8000 gpm.</p> <p>2. When one RHR pump and/or one RHR service water pump of the components required in 3.5.B.1 above is made or found to be inoperable, the remaining active components of the containment cooling mode subsystems shall be operable.</p> <p>3. When one subsystem of the containment cooling mode is made or found to be inoperable when required, the other containment cooling subsystem shall be operable. The operable containment cooling subsystem shall be tested daily thereafter until the inoperable containment cooling components are returned to an operable condition.</p>



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 17 TO FACILITY OPERATING LICENSE NO. DPR-62

CAROLINA POWER & LIGHT COMPANY

BRUNSWICK STEAM ELECTRIC PLANT UNIT NO. 2

Introduction

By letter dated May 21, 1976, Carolina Power & Light Company (CP&L) proposed an amendment to Facility Operating License No. DPR-62 for Brunswick Steam Electric Plant Unit No. 2. The proposed amendment consists of changes to the Technical Specifications which would (1) clarify instrumentation operability requirements during periods of functional testing and calibration, and (2) reduce the period of time a Low Pressure Coolant Injection (LPCI) pump can be inoperable during reactor operation from thirty to seven days.

Discussion

The existing Technical Specifications provide minimum operability requirements for the instrumentation associated with the Reactor Protection System (RPS) and the Primary Containment Isolation System (PCIS). The objective of establishing minimum operability requirements is to assure that each protective instrumentation system is capable of tolerating a single failure of any component within its system without loss of the system's intended design function. To meet this objective the Technical Specifications require that (1) there shall be two operable or tripped trip systems for each protective function, and, (2) if the minimum required number of operable instrument channels cannot be met for one of the trip systems, the trip system shall be tripped.

During the periodic surveillance testing of an instrument channel within a trip system, the channel which is being tested is considered to be inoperable. Consequently, a literal interpretation of the Technical Specifications requires that the associated trip system be tripped during surveillance testing of one of its components (i.e., an instrument channel).

CP&L states that utilization of a half-scrum or isolation to trip the channel removes the channel redundancy, making it possible to trip the RPS or PCIS with a single signal, including spurious signals, and that such a requirement increases the probability of unnecessary reactor scrams, thereby reducing the reliability of the facility and causing additional thermal cycles of the reactor pressure vessel. As a result, CP&L has requested changes to the Technical Specifications which allow

the removal of an instrument channel from service for a maximum of two (2) hours during periods of required surveillance testing, without a requirement to trip the associated trip system.

The proposed change to the allowable out-of-service time for a LPCI pump was made following discussions with the NRC staff and would result in a more conservative limiting condition for operation by reducing the permissible out-of-service period from thirty to a maximum of seven days. The change is necessary because of a modification made to the actuation logic of the LPCI system when used as part of the emergency core cooling system (ECCS). Prior to the change, a logic matrix (LPCI loop selection) existed which determined the broken loop, and caused all four LPCI pumps to discharge into the unbroken loop. In this mode of operation, the flow from three LPCI pumps was sufficient for the ECCS function and therefore a relatively long thirty day maximum out-of-service period for a single LPCI pump was acceptable. The LPCI loop selection scheme was eliminated because an assumed single failure of a LPCI injection valve to open could cause a loss of all flow from the LPCI pumps. With its removal, the four LPCI pumps are now divided into two separate loops (2 pumps per loop) with one injection valve per loop. Therefore, a single failure of a LPCI injection valve no longer causes a complete loss of all LPCI flow, which results in a more favorable ECCS analysis. At the same time, however, more stringent requirements are placed on the operability of all LPCI pumps, and therefore CP&L has proposed the more restrictive maximum out-of-service period of seven days.

Evaluation

The existing operability requirements for RPS and PCIS instrumentation systems were established to assure that a single component failure within a trip system will not negate the intended protective function of the system during reactor operation. It was not intended that the RPS and PCIS systems should be unnecessarily placed in a condition where a single spurious trip signal could result in a reactor scram and/or isolation. In fact, the "bases" for the Technical Specifications related to protective instrumentation operability requirements state that:

"...it is prudent to limit the time [that a] bypass is in effect by requiring that surveillance testing proceed on a continuous basis and that the bypass be removed as soon as testing is completed."

During the time that functional tests and calibrations are being conducted on one channel of an instrument trip system, the capability of the system to perform its design function is maintained unless an unsafe failure of the remaining channel (or channels) in that same trip system occurs. Therefore, in order for an automatic protective function to be negated three events must occur simultaneously:

1. Surveillance testing of one channel in a specific trip system;
2. The unsafe failure of the remaining channel (or channels) in that same trip system; and
3. The onset of conditions which would require the specific instrument being tested to perform its intended tripping function.

The simultaneous occurrence of these three independent events in the proposed two hour time period during which an instrument channel may be inoperable for surveillance testing is highly unlikely. Consequently, we have concluded that it is acceptable to perform required surveillance testing of an RPS or PCIS instrument channel without placing the associated trip system in the tripped condition, provided that the allowable time interval for such testing does not exceed two hours. This same change was approved by us for a similar facility (E. I. Hatch Nuclear Plant Unit 1) on April 13, 1976.

The proposal to reduce the maximum out-of-service period for a single LPCI pump from thirty to seven days would make this period consistent with the present allowable out-of-service period for other ECCS equipment (High Pressure Coolant Injection, Automatic Depressurization System, Core Spray, and the diesel generators). A maximum outage time of seven days for this equipment was previously found to be acceptable on the basis that the probability of a loss-of-coolant accident requiring the use of this equipment occurring in any seven day period is extremely low. In addition, the Technical Specifications require increased testing of other ECCS equipment during such periods, which offers increased assurance that the remaining ECCS equipment will function reliably should the need arise. The proposed seven day maximum outage time for a single LPCI pump is acceptable on the same basis.

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 the amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because 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, (2) there is reasonable assurance that the health and safety of the public will not be endangered

by operation in the proposed manner, and (3) 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: June 23, 1976

UNITED STATES NUCLEAR REGULATORY COMMISSION

LOCKET NO. 50-324

CAROLINA POWER AND LIGHT COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY
OPERATING LICENSE

Notice is hereby given that the U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 17 to Facility Operating License No. DPR-62 issued to the 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 the date of issuance.

This amendment clarifies instrumentation operability requirements during periods of functional testing, and reduces the period of time a Low Pressure Coolant Injection pump can be inoperable during reactor operation from thirty to seven days.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

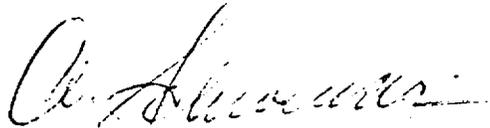
The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR § 51.5(d)(4) an environmental statement, negative declaration or environmental impact appraisal need not be prepared in connection with issuance of this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

For further details with respect to this action, see (1) the application for amendment dated May 21, 1976, (2) Amendment No. 17 to License No. DPR-62, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, NW., Washington, D.C. 20555, and at the Southport-Brunswick County Library, 109 W. Moore Street, Southport, North Carolina 28461.

A single 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 23rd day of June 1976.

FOR THE NUCLEAR REGULATORY COMMISSION



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