OCT 12 1977

Docket No. 50-324

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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. 32 to Facility Operating License No. DPR-62 for the Brunswick Steam Electric Plant (BSEP), Unit 2. The amendment consists of changes to the Technical Specifications in response to your application dated February 2, 1976 as supplemented August 18, 1976.

The amendment changes the reactor water cleanup isolation signal, the standby gas treatment system actuation setpoint, and isolation of secondary containment from a reactor water level of plus 12.5 inches to minus 38 inches, and deletes high drywell pressure as an isolation signal for the residual heat removal system.

Copies of the Safety Evaluation and Notice of Issuance are also enclosed.

Sincerely,

TVWambach

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

Enclosures:
1. Amendment No. 32 to
   License No. DPR-62
2. Safety Evaluation
3. Notice

cc w/enclosures:
See next page

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SURNAME	SSheppard	CTrammell:esp	[illegible]	ASchwencer	KRGoller	
DATE	10/7/77	10/7/77	10/1/77	10/7/77	10/12/77	

NRC FORM 318 (9-76) NRCM 0240 ☆ U.S. GOVERNMENT PRINTING OFFICE: 1976 — 626-624

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OFFICE➤						
SURNAME➤						
DATE➤						

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

October 12, 1977

Docket No. 50-324

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

Gentlemen:

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The amendment changes the reactor water cleanup isolation signal, the standby gas treatment system actuation setpoint, and isolation of secondary containment from a reactor water level of plus 12.5 inches to minus 38 inches, and deletes high drywell pressure as an isolation signal for the residual heat removal system.

Copies of the Safety Evaluation and Notice of Issuance are also enclosed.

Sincerely,

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

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

CAROLINA POWER & LIGHT COMPANY

DOCKET NO. 50-324

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 32
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 February 2, 1976, as supplemented August 18, 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 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.C.(2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 32 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

for George Lear

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 12, 1977

ATTACHMENT TO LICENSE AMENDMENT NO. 32

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:

3.2-17/3.2-18
3.2-19/3.2-20
3.2-45/3.2-46
3.7-19/3.7-20
3.7-31/3.7-32

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 #2 B21-LIS-N024A,B B21-LIS-N025A,B	≥ -38" indicated level	2	(1)	Has contacts in Group I, 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-dFS-N603	≤ 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 #2 B21-LIS-N024A,B B21-LIS-N025A,B	This reactor low water level #2 switch is on the same instrument as the PCIS low water level #2 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-dFS-N603	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 3.2-5
PRIMARY CONTAINMENT ISOLATION INSTRUMENTATION
GROUP II ISOLATION (2)

	Trip Function	Trip Setting	Minimum Number of Operable Instrument Channels Per Trip System	Required Action When Minimum Condition for Operation is Not Satisfied	Remarks
1.	High drywell pressure C72-PS-N002A,B,C,D	≤ 2 psig	2	(1)	
2.	Reactor low water level #1 B21-LIS-N017A,B,C,D	≥+ 12.5" indicated	2	(1)	

NOTES:

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

(2) Group II isolation includes:
a. Drywell equipment drain discharge valves
b. Drywell floor drain discharge valves
c. Reactor head spray isolation valves (4)
d. Shutdown cooling suction valves (4) (5)
e. RHR system radwaste discharge isolation valves (5)
f. RHR process sampling valves (5)
g. Nitrogen makeup and inerting inlet valve (3)
h. Suppression chamber inerting inlet valve (3)
i. Drywell inerting inlet valve (3)
j. Suppression chamber purge exhaust isolation valve (3)
k. Suppression chamber vent valve bypass valve (3)
l. Drywell purge exhaust isolation valve (3)
m. Drywell purge exhaust backup valve (3)
n. Containment air purge isolation valve (3)
o. Suppression chamber vent valve (3)
p. Drywell purge exhaust backup valve bypass valve (3)
q. Suppression chamber makeup and containment atmosphere dilution inlet valve (3)

TABLE 3.2-5
PRIMARY CONTAINMENT ISOLATION INSTRUMENTATION
GROUP II ISOLATION (2) (Cont'd)

NOTES:

(2) Group II isolation (Cont'd)

r. Drywell makeup and containment atmosphere dilution inlet valve (3)
s. Drywell vent isolation valve (3)
t. Drywell vent backup valve (3)
u. Containment atmosphere dilution inlet valve (3)
v. Containment atmosphere dilution inlet bypass valve (3)
w. Traveling in-core probe isolation valves
x. RHR inboard injection valves (The signal only applies if the RHR system is in the shutdown cooling mode) (5)

(3) Also isolates on high radiation signal.

(4) Also isolates on high reactor pressure signal.

(5) Isolates on low reactor water level #1 only

TABLE 3.2-12
REACTOR BUILDING VENTILATION SYSTEM ISOLATION AND STANDBY GAS TREATMENT SYSTEM INITIATION (1)

	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 Water Level #2 B21-LIS-N024A, B B21-LIS-N025A, B	$\geq$ -38" Indicated	2	(2)	Has contact in Group I isolation
2.	High Drywell Pressure C72-PS-N002A,B,C,D	$\leq$ 2 psig	2	(2)	Has contact in Group II isolation
3.	Reactor Building Ventilation Monitors D12-RM-N010A,B	Upscale $\leq$ 11 mr/hr	1		
4.	Reactor Building Ventilation Monitors D12-RM-N010A,B	Downscale	1		Annunciate instrument failure in control room

NOTES:

(1) a. Start standby gas treatment
b. Shutdown and isolate Reactor Building vent system

(2) If the minimum number of operable instrument channels is not available in either trip system for more than 24 hours, the Reactor Building ventilation system shall be isolated and the standby gas treatment system operated until the instrumentation is repaired.

Amendment No. 32

TABLE 4.2-12

MINIMUM TEST & CALIBRATION FREQUENCIES

REACTOR BUILDING VENTILATION SYSTEM ISOLATION & STANDBY GAS TREATMENT SYSTEM INITIATION

Trip Function	Functional Test	Calibration	Instrument Check
1. Reactor low water level #2 B21-LIS-N024A,B B21-LIS-N025A,B	This reactor low water level #2 switch is on the same instrument as the PCIS Group 1 reactor low water level #2 switch and will be functionally tested and calibrated at the same time.		
2. High drywell pressure C72-PS-N002A,B,C,D	This high drywell pressure switch is on the same instrument as the reactor protection system high drywell pressure switch and will be functionally tested and calibrated at the same time.		
3. Reactor Bldg. ventilation monitors (upscale) D12-RM-N010A,B	once/month	once/operating cycle	once/day
4. Reactor Bldg. ventilation monitors (downscale) D12-RM-N010A,B	once/3 months	once/operating cycle	once/day

Logic System Functional Test	Frequency
a. Reactor Building isolation	once/6 months
b. Standby gas treatment system actuation	once/6 months

TABLE 3.7-1 (Cont'd)

Group	Valve Indentification	Number of Power Operated Valves Inboard	Number of Power Operated Valves Outboard	Maximum Operating Time (sec.)	Normal Position	Action on Initiating Signal
4	HPCI steamline isolation valves E41-F002 E41-F003	1	1	50	0	GC
4	HPCI torus suction isolation valves E41-F041		1	80	C	SC
5	RCIC steamline isolation valves E51-F007 E51-F008	1	1	20	0	GC

NOTES

KEY: 0 = Open
C = Closed
SC = Stays Closed
GC = Goes Closed

TABLE 3.7-1

Note: Isolation groupings are as follows:

GROUP 1: The valves in Group 1 are closed upon any one of the following conditions:

1. Reactor water level #2
2. Main steamline high radiation
3. Main steamline high flow
4. Main steamline tunnel high temperature
5. Main steamline low pressure when in the RUN mode
6. Main steamline high flow while in STARTUP mode (Unit 2 Only)
7. Low condenser vacuum
8. Turbine building area high temperature

GROUP 2: The actions in Group 2 are initiated by any one of the following conditions:

1. Reactor low water level #1
2. High drywell pressure (except of all RHR system valves)

GROUP 3: Actions in Group 3 are initiated by any one of the following conditions:

1. Reactor low water level #2
2. Reactor water cleanup high flow
3. Reactor water cleanup space high temperature
4. Reactor water cleanup space vent air inlet/outlet high differential temperature
5. Reactor water cleanup high temperature
6. Standby liquid control system initiated

Note: The inboard isolation valve does not close on RWCU high temperature or SLC initiation.

GROUP 4: Isolation valves in the high pressure coolant injection system (HPCI) are closed upon any of the following signals:

1. HPCI steamline high flow
2. HPCI turbine steamline low pressure
3. HPCI turbine exhaust diaphragm high pressure
4. Suppression pool high ambient temperature
5. Suppression pool area vent inlet/outlet high differential temperature
6. Emergency area cooler high temperature
7. HPCI equipment room vent inlet/outlet high differential temperature

BASES:

4.7.B and 4.7.C Standby Gas Treatment System and Secondary Containment (Cont'd)

All elements of the heater should be demonstrated to be functional and operable during the test of heater capacity. Operation of the heaters will prevent moisture buildup in the filters and adsorber system.

With doors closed and fan in operation, DOP aerosol shall be sprayed externally along the full linear periphery of each respective door to check the gasket seal. Any detection of DOP in the fan exhaust shall be considered an unacceptable test result and the gaskets repairs and test repeated.

If system drains are present in the filter/adsorber banks, loop-seals must be used with adequate water level to prevent by-pass leakage from the banks.

If significant painting, fire or chemical release occurs such that the HEPA filter or charcoal adsorber could become contaminated from the fumes, chemicals or foreign material, the same tests and sample analysis shall be performed as required for operational use. The determination of significant shall be made by the operator on duty at the time of the incident. Knowledgeable staff members should be consulted prior to making this determination.

Demonstration of the automatic initiation capability and operability of filter cooling is necessary to assure system performance capability. If one standby gas treatment system is inoperable, the other system must be tested daily. This substantiates the availability of the operable system and thus reactor operation or refueling operation can continue for a limited period of time.

BASES:

3.7.D and 4.7.D Primary Containment Isolation Valves

Double isolation valves are provided on lines penetrating the primary containment and open to the free space of the containment. Closure of one of the valves in each line would be sufficient to maintain the integrity of the pressure suppression system. Automatic initiation is required to minimize the potential leakage paths from the containment in the event of a loss of coolant accident.

Group 1 - Process lines are isolated by reactor vessel water level #2 in order to allow for removal of decay heat subsequent to a scram, yet isolate in time for proper operation of the core standby cooling systems. The valves in group 1 are also closed when process instrumentation detects excessive main steamline flow, high radiation, low pressure, main steam space high temperature, low reactor water level #2, low condenser vacuum, and turbine building area high temperature.

Group 2 - Process lines are not normally in use in power operation. Isolation valves are closed by reactor vessel water level #1, or high drywell pressure. RHR shutdown cooling mode isolation valves will close only on reactor water level #1 to assure availability of shutdown cooling when required.

Group 3 - Process lines are normally in use and it is therefore not desirable to cause spurious isolation due to high drywell pressure resulting from non-safety related causes. To protect the reactor from a possible pipe break in the system, isolation is provided by high temperature in the cleanup system area or high flow through the inlet to the cleanup system. Also, since the vessel could potentially be drained through the cleanup system, reactor vessel low level #2 isolation is provided.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 32 TO FACILITY OPERATING LICENSE NO. DPR-62

CAROLINA POWER & LIGHT COMPANY

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NO. 2

DOCKET NO. 50-324

Introduction

By letter dated February 2, 1976, as supplemented August 18, 1976, Carolina Power and Light requested NRC approval for changes in the reactor containment isolation system for Brunswick Steam Electric Plant (BSEP), Unit No. 2. The changes requested are as follows:

1. Change Group III isolation, Reactor Water Cleanup System (RWCS), from a reactor water level of plus 12.5 inches to minus 38 inches.

2. Change initiation of Standby Gas Treatment System (SGTS) operation and isolation of Reactor Building Ventilation from a reactor water level of plus 12.5 inches to minus 38 inches.

3. Eliminate high drywell pressure as an isolation signal for Residual Heat Removal (RHR) shutdown cooling suction valves, RHR inboard injection valves, RHR system radwaste discharge isolation valves and RHR process sampling valves.

Discussion and Evaluation

1. RWCS Isolation

   The primary concern for isolation of the RWCS is to prevent loss of coolant via this system in the event of a break in RWCS piping. In addition to the reactor low water level signal for isolation of the RWCS in case of a LOCA, a variety of other instrumentation is provided in RWCS piping and spaces to cause isolation in case of a LOCA in the RWCS. A list of the isolation signals follows:

1. Reactor low water level

2. RWCS high temperature

3. RWCS high differential flow

4. RWCS space high temperature

5. RWCS vent air inlet/outlet high differential temperature

6. Standby liquid control system initiation (SLCS)

The operation of the SLCS is initiated manually from the reactor control room. The proposed changes do not alter the RWCS isolation associated with SLCS.

The RWCS vent air inlet/outlet high differential temperature and the RWCS space high temperature functions are effective for detection of a leak in the RWCS outside containment. The RWCS space instrumentation is highly redundant and has sensors at many locations in the RWCS space.

The RWCS high differential flow function provides diversity for detection of a break or a large leak in the RWCS.

The RWCS high temperature function stops system flow to protect the filter-demineralizer bed and has no other safety function.

The RWCS isolation functions limit the radioactive release for a RWCS line break and are not affected by the proposed change.

The proposed change would isolate the RWCS at low water level No. 2 (minus 38 inches) rather than low water level No. 1 (plus 12.5 inches). Low water level No. 2 is the point of Core Standby Cooling System (CSCS) initiation. Thus, any accident that would start CSCS also would isolate RWCS.

According to FSAR table 6.5-1 for the design basis accident (DBA), low water level No. 1 is reached in about 1 second, followed by low water level No. 2 at about 3 seconds. Thus, the proposed change would result in only a slight delay in RWCS isolation for the DBA. This short delay compares to much longer closure times of 20 to 60 seconds for various reactor system isolation valves. There is a check valve on the RWCS return line to prevent any backflow.

For a LOCA in the RWCS, isolation would be affected by the other functions before reactor low water level No. 1 is reached. The proposed change would have no impact on this sequence.

For the Appendix K analysis, the total area of a suction line break was assumed to be 4.2 square feet. The LAMB code, used for the first 20 seconds of blowdown, does not have the RWCS line included but the long-term SAFE code does. This size of the RWCS line is 0.178 ft^2, which General Electric concludes is negligible in the short-term analysis. We concur with this conclusion. The DBA for BSEP is a discharge line break of 2.0 ft^2 and is not affected by the RWCS isolation set point change.

The operational benefits of the change result from not isolating the RWCS unnecessarily. Operating experience has shown that reactor scrams result in a reactor vessel water low level transient due to void collapse. Reactor scrams initiated from high power levels generally cause a reactor vessel water level No. 1 isolation signal to be generated. Isolation of the RWCS has a deleterious effect on the cleanup system filters, such as dropping the filter cake, and prevents the cleanup of reactor vessel water immediately after scram when thermal transient effects are degrading water quality. The proposed change will eliminate unnecessary isolations which prevent water chemistry from being tightly controlled, has no detrimental impact on safety, and is therefore acceptable.

2. SGTS and RBVS Isolation

The Reactor Building Ventilation System (RBVS) isolation and Standby Gas Treatment System (SGTS) initiation are affected by three trip functions:

(1) Reactor low water level

(2) High dry well pressure

(3) Reactor building ventilation monitors

CP&L proposes to change the RBVS isolation signal from low water level No. 1 to low water level No. 2 since isolation at low water level No. 1 results in an undesirable thermal transient on the reactor building and equipment. The available diverse RBVS isolation signals from high drywell pressure and RBVS monitors provide sufficient assurance that no increase in the release of radioactivity through the RBVS will result from this change.

The proposal to change SGTS initiation from low water level No. 1 to low water level No. 2 prevents unnecessary operation of the system and would still provide for SGTS operation coincident with ECCS actuation. In addition, SGTS operation is assured by the diverse initiation signal from high drywell pressure and the RBVS monitors. The staff concludes that the proposed changes have no adverse effect on safety and are acceptable.

3. RHR Isolation

The proposed change to the RHR isolation function would eliminate high drywell pressure as an isolation signal for the RHR suction valve, RHR injection valves, RHR radwaste discharge valves and RHR process sampling valves.

The changes to the RHR radwaste discharge valves and RHR process sampling valves are of secondary importance to the RHR suction and injection valves. This is because they connect to the RHR system outside the RHR suction and injection isolation valves and do not communicate directly with the reactor coolant system (RCS) pressure boundary. With respect to the RHR radwaste discharge valves and RHR process sampling valves, the staff finds no impact on safety due to the proposed change.

With respect to the RHR suction and injection valves, the proposed change only affects the system during the shutdown mode because, during normal power operations, the RHR suction and injection valves are closed.

During shutdown conditions, RHR pump suction is taken from one of the recirculation lines upstream of the reactor coolant pump and reinjected downstream of the reactor coolant pump in both recirculation loops. With or without the proposed change to the high drywell isolation signal, RHR system valves are manually aligned to a configuration for shutdown cooling. This alignment is only allowed at low system pressures and temperatures with the reactor completely shutdown. This RHR cooling configuration disables the normal LPCI suction and injection paths. Operator action is required to realign for LPCI mode operation.

During startup, the suction line is manually isolated and the RHR system is realigned for LPCI mode operation with LPCI (RHR) pump suction taken from the suppression pool. Thus, the proposed change has no impact on RHR (LPCI) during normal power operations because the system is already isolated manually.

CP&L states that elimination of the RHR isolation on high drywell pressure is beneficial in the case of a small leak in the RCS boundary while the reactor is shutdown. By elimination of that isolation signal, the RHR system in shutdown cooling mode could provide long-term shutdown cooling in the event of a small steam leak which could give high (over 2 psi) drywell pressure. Loss of coolant at low rates is detected by sump flow monitors.

If the loss of coolant is of such magnitude that the normal makeup systems cannot keep up, the reactor water level could drop and RHR isolation would be automatic when reactor low water level No. 1 (+ 12.5") were reached. Redundant core spray systems are available at this time to make up lost inventory.

There is only one possible situation where isolation of RHR on high drywell pressure while shutdown adds any amount of protection. That occurs if there is a small break in the RHR suction line between the inboard isolation valve and the containment boundary. In that case, high drywell pressure would isolate the leak, but the probability of such a leak in the pipe at the low shutdown pressure is very small, and all such leakage would be confined within containment so that the consequences of the break would have no impact outside containment.

In summary, we find on one hand that the proposed changes would have no impact during normal power operation and on the other hand under shutdown conditions these changes would provide an additional mode of cooling for very small breaks. Therefore we conclude that the proposed changes are acceptable.

Environmental Consideration

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 impact statement or negative declaration and 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: October 12, 1977

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-324

CAROLINA POWER & LIGHT COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 32 to Facility Operating License No. DPR-62, issued to Carolina Power & Light Company, which revised the Technical Specifications for operation of the Brunswick Steam Electric Plant, Unit No. 2 (the facility) located in Brunswick County, North Carolina. The amendment is effective as of the date of its issuance.

The amendment changes the reactor water cleanup isolation signal, the standby gas treatment system actuation setpoint, and isolation of secondary containment from a reactor water level of plus 12.5 inches to minus 38 inches, and deletes high drywell pressure as an isolation signal for the residual heat removal system.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Notice of Proposed Issuance of Amendment to Facility Operating License in connection with this action was published in the FEDERAL REGISTER on March 29, 1976 (41 F.R. 12932). No request for a hearing or petition for leave to intervene was filed following notice of the proposed action.

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, or negative declaration and 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 February 2, 1976 as supplemented August 18, 1976, (2) Amendment No. 32 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, N. W., Washington, D. C. 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 12th day of October 1977.

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas V. Wambach

Thomas V. Wambach, Acting Chief
Operating Reactors Branch #1
Division of Operating Reactors

PRELIMINARY DETERMINATION

NOTICING OF PROPOSED LICENSING AMENDMENT

LICENSEE: Carolina Power & Light Company

REQUEST FOR:

1. Modification to setpoint for Reactor Water Cleanup System (RWCS) isolation, Secondary Containment isolation, and Standby Gas Treatment System (SBGTS) initiation from present +12.5" reactor water level to -38" reactor water level.

2. Eliminate shutdown cooling isolation on High Drywell pressure.

REQUEST DATE: February 2, 1976

PROPOSED ACTION:
(X) Pre-notice Recommended
( ) Post-notice Recommended
( ) Determination delayed pending completion of Safety Evaluation

BASIS FOR DECISION: Both changes above constitute relaxations of Limiting Conditions for operation not accompanied by compensatory changes. Per RLOP 601, losure 1.a., this proposed change should be pre-noticed.

PROPOSED NEPA ACTION:
( ) EIS Required
( ) Negative Declaration (ND) and Environmental Impact Appraisal (EIA) Required
(XX) No EIS, ND or EIA Required
( ) Determination delayed pending completion of EIA

BASIS FOR DECISION: It is very unlikely that the modified set point for the RWCS isolation on low reactor water level would cause the RWCS to be isolated somewhat later in the event of a pipe break in the RWCS, because the RWCS is isolated on: RWCS high temperature, high differential flow, high room temperature and high ventilation differential temperature. This being the case, this proposed change does not represent a significant increase in the potential for accidental releases, and therefore this change involves an action which is insignificant from the standpoint of environmental impact.

The modified setpoint for SBGT initiation and reactor building isolation has no

	CONCURRENCES:	DATE:
1.	CM Tramell — C. Trammell	3/3/76
2.	R. A. Purple	3/3/76
3.	K. R. Goller	3/4/76
4.	OELD	3/8/76

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s' nificant effect on potential accidental releases because this action (SBGT tiation and reactor building isolation) is also initiated by high drywell pressure (leak inside primary containment) and high radiation detected by the reactor building radiation monitors (leak outside primary containment). These systems would act first to cause isolation under realistic conditions. Since isolation will continue to occur on high radiation, and since this is the effluent of concern with respect to environmental impact, this changed setpoint on reactor vessel level involves an action which is insignificant from the standpoint of environmental impact.

Eliminating the isolation of RHR valves on high containment pressure has no appreciable environmental effects beyond that previously evaluated since high containment pressure is indicative of leakage <u>inside</u> the primary containment. At 2 psig drywell pressure, reactor building isolation and SBGT initiation would occur, as before, thereby limiting effluents to the environment. Therefore, this change does not involve a change in accidental releases beyond those previously considered.