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Docket No. 50-325

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Docket
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Carolina Power & Light Company
ATTN: Mr. J. A. Jones
Executive Vice President
336 Fayetteville Street
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Gentlemen:

The Commission has issued the enclosed Amendment No. 14 to Facility Operating License No. DPR-71 for Brunswick Steam Electric Plant Unit No. 1. The amendment consists of changes to the Technical Specifications in response to your request dated August 14, 1978, as supplemented August 30, 1978.

The amendment raises the lift settings of the safety-relief valves by 25 psi.

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

Sincerely,

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Enclosures:
1. Amendment No. 14 to DPR-71
2. Safety Evaluation
3. Notice of Issuance

cc w/enclosures:
See next page

Const. 1
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DATE➤	9/5/78	9/8/78	9/8/78	9/11/78		

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

CAROLINA POWER & LIGHT COMPANY

DOCKET NO. 50-325

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 14
License No. DPR-71

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

   A. The application for amendment by Carolina Power & Light Company (the licensee) dated August 14, 1978, as supplemented August 30, 1978, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No.14, are hereby incorporated in the license. The license 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

T. A. Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 11, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 14

FACILITY OPERATING LICENSE NO. DPR-22

DOCKET NO. 50-263

Replace the following pages of the Technical Specifications contained in Appendix A of the above-indicated license with the attached pages bearing the same number. The changed area on the revised page is reflected by a marginal line.

Remove	Insert
3/4 4-3	3/4 4-3
3/4 4-4	3/4 4-4

REACTOR COOLANT SYSTEM

IDLE RECIRCULATION LOOP STARTUP

LIMITING CONDITION FOR OPERATION

3.4.1.3 An idle recirculation loop shall not be started unless:

a. The temperature differential between the reactor coolant within the dome and the bottom head drain is ≤ 145°F, and

b. The temperature differential between the reactor coolant within the idle loop that is to be put in operation and the coolant in the reactor pressure vessel is ≤ 50°F when both loops have been idle, or

c. The temperature differential between the reactor coolant within the idle and operating recirculation loops is ≤ 50°F when only one loop has been idle, and the operating loop flow rate is ≤ 50% of rated loop flow.

APPLICABILITY: CONDITIONS 1, 2, 3 and 4.

ACTION:

With temperature differences and/or flow rate exceeding the above limits, suspend startup of any idle recirculation loop.

SURVEILLANCE REQUIREMENTS

4.4.1.3 The temperature differential and flow rate shall be determined to be within the limit within 30 minutes prior to startup of an idle recirculation loop.

3/4.4.2 SAFETY/RELIEF VALVES

LIMITING CONDITION FOR OPERATION

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

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

APPLICABILITY: CONDITIONS 1, 2 and 3.

ACTION:

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

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

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

SURVEILLANCE REQUIREMENTS

4.4.2 The safety valve function of each of the above required safety/relief valves shall be demonstrated OPERABLE by verifying that the bellows on the safety/relief valves have integrity, by instrumentation indication, at least once per 24 hours.

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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 14 TO FACILITY OPERATING LICENSE NO. DPR-71

CAROLINA POWER & LIGHT COMPANY

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NO. 1

DOCKET NO. 50-325

1.0 Introduction

By letter(1) dated August 14, 1978, and supplemented(2) August 30, 1978, the Carolina Power & Light Company (the licensee) requested a change to the reactor coolant system safety/relief valve (SRV) setpoints appearing in the Brunswick Steam Electric Plant Unit No. 1 Technical Specifications. Currently, the safety/relief valves are grouped accordingly to setpoint pressure with: 4 valves @ 1080 psig, 4 valves @ 1090 psig and 3 valves @ 1100 psig. The proposed change would raise the lift setting of each valve group by 25 psi and thereby increase the margin between reactor coolant system normal operating pressure and the valve lift point. Increasing this "simmer margin" would, according to the licensee, reduce the probability of excessive pilot valve leakage, which can lead to unreliable operation in the form of spurious valve openings or failure to reseat following valve operation.

2.0 Evaluation

Raising the lift settings of the dual action SRVs affects those plant transients which result in an increase in reactor system pressure sufficient to cause safety-relief valve actuation. Accordingly, the licensee has reanalyzed the most severe pressurization transients. The limiting events for Brunswick Unit No. 1 are generator load rejection without bypass system operation (LRw/oBP) and main steam isolation valve (MSIV) closure with indirect high flux scram (vessel overpressure protection analysis). Additionally, the licensee has considered the effect of the proposed SRV setpoint change on the limiting small break Loss of Coolant Accident (LOCA) consequences peak centerline temperature (PCT) as well as on the reactor core isolation cooling (RCIC) system and high pressure coolant injection (HPCI) system high pressure injection capabilities.

2.1 Abnormal Operational Transients

For Brunswick Unit 1, the largest change in bundle critical power ratio (CPR) is caused by the load rejection without bypass pressurization event. This event, which is intitated by fast closure of the turbine control valves, causes a rapid pressurization of the reactor system. The pressurization causes a rapid collapse of moderator voids in the core. The collapse of the voids causes a significant addition of positive reactivity to the core, which results in a pronounced neutron flux spike, and subsequently, a rise in core heat flux. The event is terminated by reactor scram, caused by a fast closure trip signal developed at the turbine control valves, before core heat can rise substantially.

The licensee reanalyzed this event using methods which are the same as those used in the Brunswick Unit 1 and 2 Final Safety Analyses Report (FSAR)(3) and Reference 5. For the revised thermal margin analyses, in addition to the assumed 25 psi increase in valve lift pressure, the safety-relief valve capacities were modified to reflect the increase in steam relief rate at the higher setpoint pressure. However, the revised analysis conservatively did not include the effects of the Anticipated Transients Without Scram (ATWS) recirculation pump trip feature initiated on a high reactor system pressure signal.

2.2 Overpressurization Analysis

The licensee has also provided the results of a bounding overpressurization analysis, to demonstrate that an adequate margin exists to the ASME Code allowable pressure, with the proposed revised SRV settings. The ASME Code allows peak transient pressures up to 100% of vessel design pressure, i.e., 1375 psig. The most limiting event was taken to be the closure of all main steam isolation valves with a reactor trip on high neutron flux which was the same event analyzed in the FSAR. The analysis conservatively assumed an initial reactor power of 104.5% and 100% core flow, an end-of-cycle scram reactivity insertion rate curve and all safety/relief valves operative. As for the load rejection without bypass, the valve relief capacities reflected the higher opening pressure setpoint. The analysis included an ATWS recirculation pump trip on high reactor pressure since the attendant flow reduction has the effect of increasing peak transient pressure. The results show that the peak pressure at the bottom of the reactor vessel is 1256 psig leaving a margin of 119 psi to the 1375 psig Code allowable safety limit. Furthermore, a generic analysis(4), showing the sensitivity of peak transient pressure to total relief capacity, when applied to Brunswick Unit No. 1, shows that the failure-to-open of one SRV would cause pressure to increase by less than 20 psi. Therefore, the maximum transient reactor vessel pressure for MSIV closure at end-of-cycle, assuming an indirect high neutron flux scram and one failed safety valve, is no greater than 1276 psig. Since this peak pressure is still well within the 1375 psig pressure safety limit, these results are acceptable to the NRC staff.

Finally, the ASME Code requires that the lowest qualified valve setpoint be at or below the vessel design pressure (i.e., 1250 psig) and that the highest safety valve setpoint be no greater than 105% of the vessel design pressure (i.e., 1313 psig). The proposed SRV setpoints meet these additional requirements and are, therefore, acceptable from this viewpoint.

2.3 <u>HPCI and RCIC Injection Capability</u>

Increasing the lowest safety/relief valve setpoint would necessitate both the HPCI pump and the RCIC pump to discharge against a higher pressure head in the event of a reactor isolation. Additional information [(2)] provided by the licensee shows that the characteristics of the pumps in these two systems are adequate to discharge at design flow even when assuming that the safety/relief valve with the lowest setpoint opens at the high end of the tolerance band.

2.4 <u>LOCA Analysis</u>

Finally, in connection with the LOCA, the increase in SRV setpoints will affect only the small break results, when the worst single failure is taken to be the loss of the HPCI system. This is due to the small increase in rate of inventory loss which results when the automatic depressurization system relief valves open at a higher pressure. A new LOCA analysis for the limiting small break, using the NRC staff approved Emergency Core Cooling System (ECCS) evaluation model changes[(5)], shows a 76°F increase in the calculated PCT. This results in peak cladding temperature which are still well within the 2200°F cladding temperature limit.

3.0 <u>Conclusions</u>

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.

We have concluded, based on the considerations discussed above, that: (1) because the amendment 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 amendment 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.

Dated: September 11, 1978

4.0 References

1. CP&L letter (M. McDuffie) to NRC (T. Ippolito) dated August 14, 1978.

2. CP&L letter (M. McDuffie) to NRC (T. Ippolito) dated August 30, 1978.

3. Brunswick Steam Electric Plant Units 1and 2 Final Safety Analysis Report, Docket No. 50-325.

4. G.E. letter (I. Stuart) to NRC (V. Stello) "Code Overpressure Protection Analysis - Sensitivity of Peak Vessel Pressure to Valve Operability." dated December 23, 1975.

5. General Electric Boiling Water Reactor Generic Reload Fuel Application "NEDE-24011-P-3, March 1978.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-325

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. 14 to Facility Operating License No. DPR-71, issued to Carolina Power & Light Company (the licensee) for operation of the Brunswick Steam Electric Plant, Unit No. 1 (the facility), located in Brunswick County, North Carolina. The amendment is effective as of its date of issuance.

The amendment raises the lift settings of the safety-relief valves by 25 psi.

The application for 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 the amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of the 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 the amendment.

For further details with respect to this action, see (1) the application for amendment dated August 14, 1978, as supplemented August 30, 1978, (2) Amendment No. 14 to License No. DPR-71, and (2) the Commission's related Safety Evalaution. 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 West Moore Street, Southport, North Carolina 28461. A copy of items (2) and (3) may be obatined 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 September, 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

TA Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors