DEC 10 1986

Docket No. 50-324

Mr. E. E. Utley
Senior Executive Vice President
Power Supply and Engineering & Construction
Carolina Power & Light Company
Post Office Box 1551
Raleigh, North Carolina 27602

Dear Mr. Utley:

The Commission has issued the enclosed Amendment No. 131 to Facility Operating License No. DPR-62 for the Brunswick Steam Electric Plant, Unit 2. The amendment consists of changes to the Technical Specifications in response to your submittal of August 22, 1986.

The amendment revises the main steam line high radiation scram and isolation setpoints, on a one time, short-term basis, to facilitate test injections of hydrogen into the reactor coolant. The test is scheduled to be performed in late November or mid December, 1986. The test is expected to be completed within about one week of its start.

A copy of the related Safety Evaluation is also enclosed. Notice of issuance will be included in the Commission's Bi-Weekly Federal Register Notice.

Sincerely,

Ernest D. Sylvester

Ernest D. Sylvester, Project Manager
BWR Project Directorate #2
Division of BWR Licensing

Enclosures:
1. Amendment No. 131 to License No. DPR-62
2. Safety Evaluation

cc w/enclosures:
See next page

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Mr. E. E. Utley
Carolina Power & Light Company

Brunswick Steam Electric Plant
Units 1 and 2

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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 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 131
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 August 22, 1986, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-62 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 131 , 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

Daniel R Muller

Daniel R. Muller, Director
BWR Project Directorate #2
Division of BWR Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 10, 1986

ATTACHMENT TO LICENSE AMENDMENT NO. 131

FACILITY OPERATING LICENSE NO. DPR-62

DOCKET NO. 50-324

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised areas are indicated by marginal lines.

Page No.

2-4

2-6

3/4 3-11

3/4 3-16a*

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*Page added

TABLE 2.2.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

FUNCTIONAL UNIT AND INSTRUMENT NUMBER	TRIP SETPOINT	ALLOWABLE VALUES
1. Intermediate Range Monitor, Neutron Flux - High(1) (C51-IRM-K601A,B,C,D,E,F,G,H)	≤ 120 divisions of full scale	≤ 120 divisions of full scale
2. Average Power Range Monitor (C51-APRM-CH.A,B,C,D,E,F)		
a. Neutron Flux - High, 15%(2)	≤ 15% of RATED THERMAL POWER	≤ 15% of RATED THERMAL POWER
b. Flow Biased Neutron Flux - High(3)(4)	≤ (0.66 W + 54%)	≤ (0.66 W + 54%)
c. Fixed Neutron Flux - High(4)	≤ 120% of RATED THERMAL POWER	≤ 120% of RATED THERMAL POWER
3. Reactor Vessel Steam Dome Pressure - High (B21-PTM-N023A-1,B-1,C-1,D-1)	≤ 1045 psig	≤ 1045 psig
4. Reactor Vessel Water Level - Low, Level 1 (B21-LTM-N017A-1,B-1,C-1,D-1)	≥ +162.5 inches*	≥ +162.5 inches*
5. Main Steam Line Isolation Valve - Closure(5) (B21-F022A,B,C,D; B21-F028A,B,C,D)	≤ 10% closed	≤ 10% closed
6. Main Steam Line Radiation - High(7) (D12-RM-K603A,B,C,D)	≤ 3 x full power background	≤ 3.5 x full power background
7. Drywell Pressure - High (C72-PTM-N002A-1,B-1,C-1,D-1)	≤ 2 psig	≤ 2 psig
8. Scram Discharge Volume Water Level - High (C12-LSH-N013A,B,C,D) (C12-LSH-4516A,B,C,D)	≤ 109 gallons	≤ 109 gallons

TABLE 2.2.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

TABLE NOTATION

* Vessel water levels refer to REFERENCE LEVEL ZERO.

(1) The Intermediate Range Monitor scram functions are automatically bypassed when the reactor mode switch is placed in the Run position and the Average Power Range Monitors are on scale.

(2) This Average Power Range Monitor scram function is a fixed point and is increased when the reactor mode switch is placed in the Run position.

(3) The Average Power Range Monitor scram function is varied, Figure 2.2.1-1, as a function of recirculation loop flow (W). The trip setting of this function must be maintained in accordance with Specification 3.2.2.

(4) The APRM flow-biased high neutron flux signal is fed through a time constant circuit of approximately 6 seconds. The APRM fixed high neutron flux signal does not incorporate the time constant, but responds directly to instantaneous neutron flux.

(5) The Main Steam Line Isolation Valve-Closure scram function is automatically bypassed when the reactor mode switch is in other than the Run position.

(6) These scram functions are bypassed when THERMAL POWER is less than 30% of RATED THERMAL POWER.

(7) Within 24 hours prior to the planned start of the hydrogen injection test, with reactor power at greater than 22% of rated thermal power, the normal full-power radiation background level and associated trip setpoints may be changed based on a calculated value of the radiation level expected during the test. The background radiation level and associated trip setpoints may be adjusted during the test program based on either calculations or measurements of actual radiation levels resulting from hydrogen injection. The background radiation level shall be determined and the associated trip setpoints shall be set within 24 hours of re-establishing normal radiation levels after completion of the hydrogen injection test or within 12 hours of establishing reactor power levels below 22% of rated thermal power, while these functions are required to be operable.

TABLE 3.3.2-1

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION AND INSTRUMENT NUMBER	VALVE GROUPS OPERATED BY SIGNAL(a)	MINIMUM NUMBER OPERABLE CHANNELS PER TRIP SYSTEM(b)(c)	APPLICABLE OPERATIONAL CONDITION	ACTION
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level -				
1. Low, Level 1 (B21-LT-N017A-1,B-1,C-1,D-1) (B21-LTM-N017A-1,B-1,C-1,D-1)	2, 6, 7, 8	2	1, 2, 3	20
2. Low, Level 2 (B21-LT-N024A-1,B-1, and B21-LT-N025A-1,B-1) (B21-LTM-N024A-1,B-1 and B21-LTM-N025A-1,B-1)	1, 3	2	1, 2, 3	20
b. Drywell Pressure - High (C72-PT-N002A,B,C,D) (C72-PTM-N002A-1,B-1,C-1,D-1)	2, 6, 7	2	1, 2, 3	20
c. Main Steam Line				
1. Radiation - High (d) (D12-RM-K603A,B,C,D)	1	2	1, 2, 3(i)	21
2. Pressure - Low (B21-PT-N015A,B,C,D) (B21-PTM-N015A-1,B-1,C-1,D-1)	1	2	1	22
3. Flow - High (B21-PDT-N006A,B,C,D; B21-PDT-N007A,B,C,D; B21-PDT-N008A,B,C,D; B21-PDT-N009A,B,C,D) (B21-PDTM-N006A-1,B-1,C-1,D-1; B21-PDTM-N007A-1,B-1,C-1,D-1; B21-PDTM-N008A-1,B-1,C-1,D-1; B21-PDTM-N009A-1,B-1,C-1,D-1)	1	2/line	1	22

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

ACTIONS

(i) Within 24 hours prior to the planned start of the hydrogen injection test, with reactor power at greater than 22% of rated thermal power, the normal full-power radiation background level and associated trip setpoints may be changed based on a calulated value of the radiation level expected during the test. The background radiation level and associated trip setpoints may be adjusted during the test program based on either calculations or measurements of actual radiation levels resulting from hydrogen injection. The background radiation level shall be determined and the associated trip setpoints shall be set within 24 hours of re-establishing normal radiation levels after completion of the hydrogen injection test or within 12 hours of establishing reactor power levels below 22% of rated thermal power, while these functions are required to be operable.

TABLE 3.3.2-2

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION AND INSTRUMENT NUMBER	TRIP SETPOINT	ALLOWABLE VALUE
1. **PRIMARY CONTAINMENT ISOLATION**		
a. Reactor Vessel Water Level -		
1. Low, Level 1 (B21-LTM-N017A-1,B-1,C-1,D-1)	≥ + 162.5 inches[a]	≥ + 162.5 inches[a]
2. Low, Level 2 (B21-LTM-N024A-1,B-1 and B21-LTM-N025A-1,B-1)	≥ + 112 inches[a]	≥ + 112 inches[a]
b. Drywell Pressure - High (C72-PTM-N002A-1,B-1,C-1,D-1)	≤ 2 psig	≤ 2 psig
c. Main Steam Line		
1. Radiation - High (D12-RM-K603A,B,C,D)	≤ 3 x full power background[b]	≤ 3.5 x full power[b] background
2. Pressure - Low (B21-PTM-N015A-1,B-1,C-1,D-1)	≥ 825 psig	≥ 825 psig
3. Flow - High (B21-PDTM-N006A-1,B-1,C-1,D-1; B21-PDTM-N007A-1,B-1,C-1,D-1; B21-PDTM-N008A-1,B-1,C-1,D-1; B21-PDTM-N009A-1,B-1,C-1,D-1)	≤ 140% of rated flow	≤ 140% of rated flow
4. Flow - High (B21-PDTS-N006A-2; B21-PDTS-N007B-2; B21-PDTS-N008C-2; B21-PDTS-N009D-2)	≤ 40% of rated flow	≤ 40% of rated flow

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION AND INSTRUMENT NUMBER	TRIP SETPOINT	ALLOWABLE VALUE
3. RCIC Steam Supply Pressure - Low (E51-PS-N019A,B,C,D)	≥ 50 psig	≥ 50 psig
4. RCIC Steam Line Tunnel Temp - High (E51-TS-3319; E51-TS-3320; E51-TS-3321; E51-TS-3322; E51-TS-3323; E51-TS-3355; E51-TS-3487)	≤ 175°F	≤ 175°F
5. Bus Power Monitor (E51-K42 and E51-K43)	NA	NA
6. RCIC Turbine Exhaust Diaphragm Pressure - High (E51-PS-N012A,B,C,D)	≤ 10 psig	≤ 10 psig
7. RCIC Steam Line Ambient Temp - High (E51-TS-N603A,B)	≤ 200°F	≤ 200°F
8. RCIC Steam Line Area Δ Temp - High (E51-dTS-N604A,B)	≤ 50°F	≤ 50°F
9. RCIC Equipment Room Ambient Temp - High (E51-TS-N602A,B)	≤ 175°F	≤ 175°F
10. RCIC Equpment Room Δ Temp - High (E51-dTS-N601A,B)	≤ 50°F	≤ 50°F
5. SHUTDOWN COOLING SYSTEM ISOLATION		
a. Reactor Vessel Water Level - Low, Level 1 (B21-LTM-N017A-1,B-1,C-1,D-1)	≥ + 162.5 inches[(a)]	≥ + 162.5 inches[(a)]
b. Reactor Steam Dome Pressure - High (B32-PS-N018A,B)	≤ 140 psig	≤ 140 psig

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

NOTES

(a) Vessel water levels refer to REFERENCE LEVEL ZERO.

(b) Within 24 hours prior to the planned start of the hydrogen injection test, with reactor power at greater than 22% of rated thermal power, the normal full-power radiation background level and associated trip setpoints may be changed based on a calulated value of the radiation level expected during the test. The background radiation level and associated trip setpoints may be adjusted during the test program based on either calculations or measurements of actual radiation levels resulting from hydrogen injection. The background radiation level shall be determined and the associated trip setpoints shall be set within 24 hours of re-establishing normal radiation levels after completion of the hydrogen injection test or within 12 hours of establishing reactor power levels below 22% of rated thermal power, while these functions are required to be operable.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 131 TO FACILITY LICENSE NO. DPR-62

CAROLINA POWER & LIGHT COMPANY

BRUNSWICK STEAM ELECTRIC PLANT UNIT 2

DOCKET NO. 50-324

1.0 INTRODUCTION

By submittal dated August 22 and December 2, 1986, the Carolina Power and Light Company has proposed a Technical Specification change to permit a temporary increase in the Brunswick Steam Electric Plant, Unit No. 2 main steam line high radiation scram and isolation setpoints. This change would facilitate the planned testing of hydrogen addition water chemistry at their Brunswick Plant Unit 2. On the basis of prior experience, it is anticipated that the main steam line radiation levels may increase during the test by a factor of five over the routinely experienced dose rates.

2.0 EVALUATION

2.1 HIGH RADIATION SCRAM AND ISOLATION SETPOINTS

The Main Steam Line Radiation Monitors (MSLRMs) provide reactor scram as well as reactor vessel and primary containment isolation signals upon detection of high activity levels in the main steam lines. Additionally, these monitors serve to limit radioactivity released in the event of fuel failures. The proposed Technical Specification changes (to Tables 2.2.1-1, 3.3.2-1 and 3.3.2-2) would allow adjustments to the normal background radiation level and associated trip setpoints for the MSLRMs at reactor power levels greater than 22% of rated power. The adjustments are needed to accommodate the expected increase in main steam activity levels as a result of planned tests involving hydrogen injection into the primary system. This is primarily due to increased nitrogen-16 (N-16) levels in the reactor coolant.

The licensee states that the only transient or postulated accident which takes credit for the main steam line high radiation scram and isolation signals is the control rod drop accident (CRDA). The staff notes that for a CRDA, the MSLRMs' primary function is to limit the transport of activity released from failed fuel to the turbine and condensers. This is done by initiating closure of the main steam isolation valves and thus isolating the reactor vessel. Main steam line high radiation will also produce a reactor scram signal (reactor scram in the event of a CRDA, however, would be initiated by signals from the Neutron Monitoring System) and will isolate the mechanical vacuum pump and the gland seal steam exhaust system to reduce leakage of fission products to the atmosphere from the turbine and condensers.

Generic analysis of the consequences of a CRDA have shown that fuel failures are not expected to result from a CRDA occurring at greater than 10% power. As power increases, the severity of the rod accident rapidly decreases due to the effects of increased void formation and increased Doppler reactivity feedback. Since the setpoint adjustments will be restricted to power levels above 22% of rated power, the staff concludes that the currently approved CRDA analysis for Brunswick 2 is appropriately bounded and remains valid.

2.2 RADIATION PROTECTION/ALARA

The staff also has reviewed the proposed Technical Specification change in reference to the radiological implications of the dose rate increases associated with N-16 equilibrium changes during hydrogen addition at BWRs. In addition, the review addressed the radiation protection/ALARA measures for the course of the planned test, in accordance with 10 CFR 20.1(c) and Regulatory Guide 8.8, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable."

An overall objective of the test is to determine general in-plant and site boundary dose rate increases due to hydrogen addition. The licensee has indicated that normal health physics/ALARA practices and procedures for Brunswick will be continued throughout the test. Additionally, the licensee has indicated that specific locations will be identified where temporary shielding may be needed for long-term implementation of hydrogen injection.

The staff also has reviewed the licensee's proposed dose control measures and surveillance efforts planned for the hydrogen addition test. Tests of this type have been proposed and conducted at other operating BWRs following staff review and approval of similar Technical Specification changes. The test conditions, as identified by the licensee, as well as the measures proposed for radiation protection/ALARA at the Brunswick Steam Electric Plant, Unit No. 2, are consistent with those utilized at the other BWRs during their successful hydrogen addition tests. None of these tests involved any significant, unanticipated, radiological exposures or releases.

2.3 HYDROGEN STORAGE AND DISTRIBUTION SYSTEM

The licensee's hydrogen addition system is designed to reduce the potential hazard to safety related systems and meets the applicable parts of the BWR Owners Group report, "Guidelines for Permanent BWR Hydrogen Water Chemistry Installations". The hydrogen addition system is being supplied by a new vendor, Innovative Technologies, Inc. All previous hydrogen water chemistry preimplementation tests have been performed by one other vendor.

Innovative Technologies, Inc. has extensive hydrogen water chemistry system experience on Swedish Reactor Plants.

The hydrogen injected into the condensate booster pump suction is stored as compressed gas in tube tank trucks. The hydrogen distribution system contains an excess flow check valve to limit the release of hydrogen in the event of a pipe break. To prevent the accumulation of combustible levels of hydrogen in the condensate booster pump rooms, at the pumps, near the control valves, and along hydrogen supply line, the hydrogen lines will be leak tested prior to the test and will be monitored for hydrogen concentrations during the test. These monitors will alarm and isolate the hydrogen supply when hydrogen concentrations exceed 2%.

Since the licensee stores substantial amounts of chlorine on site for the purpose of water treatment, the staff evaluated the potential synergistic effect associated with the storage of hydrogen. The combination of hydrogen gas and chlorine gas can explode in the presence of any form of energy, such as sunlight or heat (250°C). Therefore, it is prudent to maintain an adequate separation distance between the chlorine and hydrogen storage facilities. The hydrogen tube tank trucks will be parked a distance of 420 feet away from the chlorine tank car. The licensee has committed to limiting the closest approach the hydrogen supply truck will make to the chlorine tank car to approximately 300 feet. In addition, any movement of the chlorine tank car will not be brought closer than approximately 300 feet from the hydrogen supply truck. The 300 foot separation distance is judged to be sufficient to prevent interaction in the unlikely event of a simultaneous chlorine and hydrogen release.

On the basis of the above provisions, we conclude that the licensee's hydrogen addition system meets Section C.5.d of Branch Technical Position CMEB 9.5.1 of NUREG 0800 and, is therefore, acceptable.

3.0 EVALUATION SUMMARY

On the basis of the adequacy of the licensee's radiation protection/ALARA program, utilization of special surveys to monitor dose rates increases on site and at the site boundary, the capability to monitor for fuel failures, as well as the success of similar efforts at other operating BWRs, the staff finds the licensee's request acceptable.

4.0 ENVIRONMENTAL CONSIDERATIONS

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

5.0 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 the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: M. Lamastra

Dated: December 10, 1986