May 28, 1998

Mr. C. S. Hinnant, Vice President
Carolina Power & Light Company
Brunswick Steam Electric Plant
Post Office Box 10429
Southport, North Carolina 28461

SUBJECT: ISSUANCE OF AMENDMENT NO. 196 TO FACILITY OPERATING LICENSE NO. DPR-71 AND AMENDMENT NO. 226 TO FACILITY OPERATING LICENSE NO. DPR-62 REVISING DRYWELL AIR TEMPERATURE - BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2 (TAC NOS. M99367 AND M99368)

Dear Mr. Hinnant:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 196 to Facility Operating License No. DPR-71 and Amendment No. 226 to Facility Operating License No. DPR-62 for Brunswick Steam Electric Plant, Units 1 and 2.

The amendments consist of changes to the Technical Specifications (TS) in response to your application dated June 12, 1997, as supplemented by letter dated February 2, 1998. The amendments revise the Limiting Condition for operation of the governing TS to limit the drywell average air temperature rather than primary containment air temperature. Additionally, the amendments require that the drywell average air temperature be maintained less than or equal to 150°F during plant operation. The current primary containment average temperature limit is 135°F.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's bi-weekly Federal Register Notice.

Sincerely,
Original signed by:
David C. Trimble, Project Manager

Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

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PDR ADOCK 05000324
P PDR

Docket Nos. 50-325
and 50-324

Enclosures:
1. Amendment No. 196 to License No. DPR-71
2. Amendment No. 226 to License No. DPR-62
3. Safety Evaluation

cc w/enclosures: See next page

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PM:PDII-1	LA:PDII-1	OGC	APD:PDII-1	SCSB:DSSA	SCSB:DSSA
DTrimble	EDunnington		PTKuo	RGoel:bw	Lmarsh
5/11/98	5/11/98	5/14/98	5/28/98	12/18/97	01/29/98
Yes/No	Yes/No	Yes/No	Yes/No	Yes/No	Yes/No

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w/ change to SE notice
OGC comments incorporated 5/28/98

cc:

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AMENDMENT NO. 196 TO FACILITY OPERATING LICENSE NO. DPR-71 - BRUNSWICK, UNIT 1 AND AMENDMENT NO. 226 TO FACILITY OPERATING LICENSE NO. DPR-62 - BRUNSWICK, UNIT 2

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

CAROLINA POWER & LIGHT COMPANY, et al.

DOCKET NO. 50-325

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 196
License No. DPR-71

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

   A. The application for amendment filed by Carolina Power & Light Company (the licensee), dated June 12, 1997, as supplemented by letter dated February 2, 1998, 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:

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(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 196 , are hereby incorporated in the license. Carolina Power & Light Company shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

H E Echison for

Pao-Tsin Kuo, Acting Director
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 28, 1998

ATTACHMENT TO LICENSE AMENDMENT NO. 196

FACILITY OPERATING LICENSE NO. DPR-71

DOCKET NO. 50-325

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

Remove Pages	Insert Pages
VI	VI
3/4 6-8	3/4 6-8
B 3/4 6-2a	B 3/4 6-2a
---	B 3/4 6-2b

SECTION		PAGE
3/4.4	REACTOR COOLANT SYSTEM (Continued)	
3/4.4.4	CHEMISTRY	3/4 4-7
3/4.4.5	SPECIFIC ACTIVITY	3/4 4-10
3/4.4.6	PRESSURE/TEMPERATURE LIMITS	
	Reactor Coolant System	3/4 4-13
	Reactor Steam Dome	3/4 4-21
3/4.4.7	MAIN STEAM LINE ISOLATION VALVES	3/4 4-22
3/4.4.8	STRUCTURAL INTEGRITY	3/4 4-23
3/4.5	EMERGENCY CORE COOLING SYSTEMS	
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3/4.5.2	AUTOMATIC DEPRESSURIZATION SYSTEM	3/4 5-3
3/4.5.3	LOW PRESSURE COOLING SYSTEMS	
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3/4.5.4	SUPPRESSION POOL	3/4 5-9
3/4.6	CONTAINMENT SYSTEMS	
3/4.6.1	PRIMARY CONTAINMENT	
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	Primary Containment Leakage	3/4 6-2
	Primary Containment Air Lock	3/4 6-4
	Primary Containment Structural Integrity	3/4 6-6
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	Drywell Air Temperature	3/4 6-8

DRYWELL AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.6 Drywell average air temperature shall be ≤ 150°F.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

With the drywell average air temperature > 150°F, reduce the drywell average air temperature to within the limit within 8 hours, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 12 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.6 The drywell average air temperature shall be the volumetric average of the temperatures at the following locations and shall be determined at least once per 24 hours:

Location

a. Below 5′ elevation,

b. Between 10′ and 23′ elevation,

c. Between 28′ and 45′ elevation,

d. Between 70′ and 80′ elevation, and

e. Above 90′ elevation.

3/4.6.1.3 PRIMARY CONTAINMENT AIR LOCKS (Continued)

References:

1. 10 CFR Part 50, Appendix J, Option B.

2. NRC Regulatory Guide 1.163, dated September 1995, "Performance-Based Containment Leak-Rate Testing Program."

3. Nuclear Energy Institute Guideline 94-01, Revision 0, dated July 26, 1995, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50 Appendix J."

3/4.6.1.4 PRIMARY CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the primary containment steel vessel will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the vessel will withstand the maximum pressure of 49 psig in the event of a LOCA. A visual inspection in conjunction with the Primary Containment Leakage Rate Testing Program is sufficient to demonstrate this capability.

References:

1. 10 CFR Part 50, Appendix J, Option B, Section III.A.

2. NRC Regulatory Guide 1.163, dated September 1995, "Performance-Based Containment Leak-Rate Testing Program."

3/4.6.1.5 PRIMARY CONTAINMENT INTERNAL PRESSURE

The limitations of primary containment internal pressure ensure that the containment peak pressure of 49 psig does not exceed the design pressure of 62 psig during LOCA conditions. The limit of 1.75 psig, for initial positive containment pressure will limit the total pressure to 49 psig, which is less than the design pressure and is consistent with the accident analyses.

3/4.6.1.6 DRYWELL AIR TEMPERATURE

The drywell contains the reactor vessel and piping, which add heat to the airspace. Drywell coolers remove heat and maintain a suitable environment. The average airspace temperature affects the calculated response to postulated Design Basis Accidents (DBAs).

Primary containment performance is evaluated for a spectrum of break sizes for postulated loss of coolant accidents (LOCAs) (Reference 1). Among the inputs to the design basis analysis is the initial drywell average air temperature (Reference 1). Analyses assume an initial drywell average air temperature of 150°F. This limitation ensures that the safety analysis

3/4.6.1.6 DRYWELL AIR TEMPERATURE (Continued)

remains valid by maintaining the expected initial conditions and ensures that the peak LOCA drywell temperature does not result in the drywell structure exceeding the maximum allowable temperature of 300°F (Reference 2). Exceeding this design temperature may result in the degradation of the primary containment structure under accident loads. Equipment inside primary containment required to mitigate the effects of a DBA is designed to operate and be capable of operating under environmental conditions expected for the accident.

Drywell air temperature satisfies Criterion 2 of Reference 3.

In the event of a Design Basis Accident (DBA), with an initial drywell average air temperature less than or equal to the LCO temperature limit, the resultant peak accident temperature is maintained below the drywell design temperature. As a result, the ability of the primary containment to perform its design function is ensured.

References:

1. Updated FSAR, Section 6.2.1.1.3.2.

2. Updated FSAR, Section 6.2.1.

3. 10 CFR 50.36(c)(2)(ii).

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

CAROLINA POWER & LIGHT COMPANY, et al.

DOCKET NO. 50-324

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 226
License No. DPR-62

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

   A. The application for amendment filed by Carolina Power & Light Company (the licensee), dated June 12, 1997, as supplemented by letter dated February 2, 1998, 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. 226 , are hereby incorporated in the license. Carolina Power & Light Company shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

H E Edison for

Pao-Tsin Kuo, Acting Director
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 28, 1998

ATTACHMENT TO LICENSE AMENDMENT NO. 226

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.

Remove Pages	Insert Pages
VI	VI
3/4 6-8	3/4 6-8
B 3/4 6-2a	B 3/4 6-2a
---	B 3/4 6-2b

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION		PAGE
3/4.4	REACTOR COOLANT SYSTEM (Continued)	
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3/4.4.5	SPECIFIC ACTIVITY	3/4 4-10
3/4.4.6	PRESSURE/TEMPERATURE LIMITS	
	Reactor Coolant System	3/4 4-13
	Reactor Steam Dome	3/4 4-21
3/4.4.7	MAIN STEAM LINE ISOLATION VALVES	3/4 4-22
3/4.4.8	STRUCTURAL INTEGRITY	3/4 4-23
3/4.5	EMERGENCY CORE COOLING SYSTEMS	
3/4.5.1	HIGH PRESSURE COOLANT INJECTION SYSTEM	3/4 5-1
3/4.5.2	AUTOMATIC DEPRESSURIZATION SYSTEM	3/4 5-3
3/4.5.3	LOW PRESSURE COOLING SYSTEMS	
	Core Spray System	3/4 5-4
	Low Pressure Coolant Injection System	3/4 5-7
3/4.5.4	SUPPRESSION POOL	3/4 5-9
3/4.6	CONTAINMENT SYSTEMS	
3/4.6.1	PRIMARY CONTAINMENT	
	Primary Containment Integrity	3/4 6-1
	Primary Containment Leakage	3/4 6-2
	Primary Containment Air Lock	3/4 6-4
	Primary Containment Structural Integrity	3/4 6-6
	Primary Containment Internal Pressure	3/4 6-7
	Drywell Average Air Temperature	3/4 6-8

DRYWELL AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.6 Drywell average air temperature shall be ≤ 150°F.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

With the drywell average air temperature > 150°F, reduce the drywell average air temperature to within the limit within 8 hours, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 12 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.6 The drywell average air temperature shall be the volumetric average of the temperatures at the following locations and shall be determined at least once per 24 hours:

Location

a. Below 5′ elevation,

b. Between 10′ and 23′ elevation,

c. Between 28′ and 45′ elevation,

d. Between 70′ and 80′ elevation, and

e. Above 90′ elevation.

3/4.6.1.3 PRIMARY CONTAINMENT AIR LOCKS (Continued)

References:

1. 10 CFR Part 50, Appendix J, Option B.

2. NRC Regulatory Guide 1.163, dated September 1995, "Performance-Based Containment Leak-Rate Testing Program."

3. Nuclear Energy Institute Guideline 94-01, Revision 0, dated July 26, 1995, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50 Appendix J."

3/4.6.1.4 PRIMARY CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the primary containment steel vessel will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the vessel will withstand the maximum pressure of 49 psig in the event of a LOCA. A visual inspection in conjunction with the Primary Containment Leakage Rate Testing Program is sufficient to demonstrate this capability.

References:

1. 10 CFR Part 50, Appendix J, Option B, Section III.A.

2. NRC Regulatory Guide 1.163, dated September 1995, "Performance-Based Containment Leak-Rate Testing Program."

3/4.6.1.5 PRIMARY CONTAINMENT INTERNAL PRESSURE

The limitations of primary containment internal pressure ensure that the containment peak pressure of 49 psig does not exceed the design pressure of 62 psig during LOCA conditions. The limit of 1.75 psig, for initial positive containment pressure will limit the total pressure to 49 psig, which is less than the design pressure and is consistent with the accident analyses.

3/4.6.1.6 DRYWELL AIR TEMPERATURE

The drywell contains the reactor vessel and piping, which add heat to the airspace. Drywell coolers remove heat and maintain a suitable environment. The average airspace temperature affects the calculated response to postulated Design Basis Accidents (DBAs).

Primary containment performance is evaluated for a spectrum of break sizes for postulated loss of coolant accidents (LOCAs) (Reference 1). Among the inputs to the design basis analysis is the initial drywell average air temperature (Reference 1). Analyses assume an initial drywell average air temperature of 150°F. This limitation ensures that the safety analysis

3/4.6.1.6 DRYWELL AIR TEMPERATURE (Continued)

remains valid by maintaining the expected initial conditions and ensures that the peak LOCA drywell temperature does not result in the drywell structure exceeding the maximum allowable temperature of 300°F (Reference 2). Exceeding this design temperature may result in the degradation of the primary containment structure under accident loads. Equipment inside primary containment required to mitigate the effects of a DBA is designed to operate and be capable of operating under environmental conditions expected for the accident.

Drywell air temperature satisfies Criterion 2 of Reference 3.

In the event of a Design Basis Accident (DBA), with an initial drywell average air temperature less than or equal to the LCO temperature limit, the resultant peak accident temperature is maintained below the drywell design temperature. As a result, the ability of the primary containment to perform its design function is ensured.

References:

1. Updated FSAR, Section 6.2.1.1.3.2.

2. Updated FSAR, Section 6.2.1.

3. 10 CFR 50.36(c)(2)(ii).

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION

BRUNSWICK STEAM ELECTRIC PLANT (BSEP)

DRYWELL AIR TEMPERATURE

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

1.0 INTRODUCTION

By letter dated June 12, 1997, as supplemented by letter dated February 2, 1998, Carolina Power and Light Company (CP&L), the licensee for Brunswick Steam Electric Plant (BSEP), Units 1 and 2, proposed to revise Technical Specification (TS) 3.6.1.6 to require monitoring drywell average air temperature rather than primary containment average air temperature and to require that the drywell average air temperature be maintained less than or equal to 150°F during plant operation. The current primary containment average air temperature limit is 135°F. Also, a footnote which provided a one-time exception to the primary containment average air temperature limit is being deleted because the period of applicability for this exception has expired.

2.0 DISCUSSION AND EVALUATION

The licensee indicated that the drywell contains the reactor vessel and piping, which add heat to the airspace. Drywell coolers remove heat and maintain a suitable environment. The primary containment performance is evaluated for a spectrum of break sizes for postulated loss-of-coolant accidents (LOCAs). Among the inputs to these containment design basis analyses is the initial drywell average air temperature. The limitation on containment average air temperature is intended to ensure that the containment peak air temperature does not exceed the design temperature of 300°F during postulated LOCA conditions. The primary containment average air temperature is calculated using a volume-weighted average of both the drywell and the wetwell atmosphere at specific locations.

The current TS requirements pertaining to containment systems were re-evaluated as part of the Improved Technical Specifications (ITS) Conversion Project. The containment systems requirements provided in the new ITS for Boiling Water Reactors place a Limiting Condition for Operation on the drywell average air temperature rather than the primary containment air temperature. To support this change and the amendment request, CP&L had General Electric (GE) re-evaluate the containment performance design basis analyses using an initial drywell average air temperature of 150°F.

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The GE evaluation was performed using power uprate conditions, which are the limiting initial conditions, and included a review of the following containment analyses where the initial drywell temperature is an input. The results of these analyses indicate that the increase in initial drywell average air temperature has no significant effect on the accident peak pressure and temperature as discussed below:

Drywell and wetwell pressure and temperature during LOCAs - An initial drywell temperature of 150°F results in a slightly lower peak drywell and peak wetwell pressure (less than 1 psi) and a slightly lower peak drywell temperature (approximately 1°F) and a slightly higher peak wetwell temperature (approximately 0.1°F) than with an initial drywell temperature of 135°F.

Peak suppression pool temperature during LOCA - An increase in drywell temperature to 150°F has a negligible impact (less than 0.03°F increase) on the long-term peak suppression pool temperature following a LOCA or an abnormal transient.

Hydrodynamic loads defined for the LOCA and for safety/relief valve actuation - An increase in drywell temperature to 150°F has no adverse impact on the LOCA hydrodynamic loads including pool swell, vent-thrust, condensation oscillation, and chugging. It also has a negligible impact on the safety/relief valve discharge line (SRVDL) reflood height and, therefore, a negligible corresponding effect on the safety/relief valve discharge loads.

Net positive suction head (NPSH) available for the residual heat removal (RHR) and core spray (CS) pumps - An increase in drywell temperature to 150°F has a negligible impact (less than 1 inch of water) on available NPSH for the RHR and CS pumps that take suction from the suppression pool.

Appendix R event - An increase in drywell temperature to 150°F does not impact the capability of using safe shutdown systems to achieve cold shutdown within 72 hours as required by 10 CFR 50, Appendix R.

Station Blackout - An increase in drywell temperature to 150°F has a negligible impact on the drywell temperature response during a station blackout. Also, there is no adverse impact on the coping capability and 4-hour duration during a station blackout.

Small Break Analyses - The licensee indicated that an increase in drywell temperature to 150°F has a negligible impact on the final drywell atmosphere temperature following a small-break LOCA. This is based on a sensitivity study performed by GE in which the initial drywell temperature was varied from 135°F to 150°F for a large-break LOCA scenario. For the first case, the resultant drywell atmosphere peak temperature was 287.7°F and for the second case the peak drywell temperature was 286°F. The reason for the drop in peak temperature is that at 150°F, the drywell air has a lower density than at 135°F and therefore lower total mass of the fixed-volume. As a result, steam condensation can begin sooner and the resultant maximum temperature reached by the drywell is slightly lower. Although this evaluation was performed for a large-break LOCA scenario, the same phenomenon would occur for small-break LOCA, although the difference would be expected to be less.

Equipment Qualification

The licensee indicated that the drywell equipment necessary to mitigate the effects of a postulated design basis accident was reviewed in the areas of environmental qualification and instrumentation accuracy to assess the potential impacts of the proposed change and was determined to remain qualified to operate under environmental conditions expected during normal operations followed by conditions predicted for an accident.

The effect of ambient temperature on drywell equipment qualified life is evaluated periodically based upon input from individual temperature elements located within the drywell near the equipment rather than the average bulk temperature. Any change to the elevation-dependent temperatures will be factored into the qualified life calculations.

The pressure/temperature analyses for the design-basis accident LOCA were re-performed with an initial drywell average air temperature of 150°F. The result of these analyses indicate that the increase in initial drywell average air temperature has no significant effect on the accident peak pressure and temperature. Therefore, the use of a drywell average air temperature limit of 150°F is consistent with the pressure/temperature analysis presently used in the BSEP Environment Qualification Program.

Drywell Water Level Instrumentation Setpoint Calculations

The drywell temperature is used as an input to the setpoint scaling and uncertainty calculations. Changing the drywell average air temperature limit to 150°F does not impact the setpoint calculations because these calculations use the actual drywell temperature based on trending analysis in the vicinity of the instrument sensing lines. Therefore, changing the drywell average air temperature limit to 150°F does not impact the drywell water level instrument setpoint calculations.

Evaluation

The staff has reviewed the licensee submittal as discussed above and finds that the proposed change to monitor the drywell air temperature rather than the primary containment atmosphere air temperature and revising the temperature limit from 135°F to 150°F during plant operation has a negligible effect on containment response parameters and instrument setpoints or allowable values and, therefore, is acceptable.

The other proposed changes a) for Unit 1 to revise the term "CONDITIONS" to "OPERATIONAL CONDITIONS" to match Section 1.0 Definitions, and b) for Unit 2 to delete the footnote to TS 3.6.1.6 which provided a temporary, one-time exception to the primary containment average air temperature limit of 135°F because the period of the footnote's applicability has expired, are administrative in nature, and therefore, are acceptable.

Based on the above evaluation, the staff finds the following proposed amendments to revise BSEP Units 1 and 2 TS 3.6.1.6 acceptable: (1) require monitoring drywell average air temperature rather than primary average air temperature, (2) revising the associated temperature limit from 135°F to 150°F during plant operation, (3) deleting the expired footnote

to Unit 2 TS 3.6.1.6, and (4) revising the term "CONDITIONS" in the statement of "Applicability" for Unit 1 TS 3.6.1.6 to "OPERATIONAL CONDITIONS" to match Section 1.0 Definitions acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the State of North Carolina official was notified of the proposed issuance of the amendments. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendments change 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 the Surveillance Requirements. The NRC staff has determined that the amendments involve 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 amendments involve no significant hazards consideration, and there has been no public comment on such finding (62 FR 45454). Accordingly, the amendments meet 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 amendments.

5.0 CONCLUSION

The Commission has 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, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: Raj Goel, SCSB/ DSSA

Date: May 28, 1998