CP&L
A Progress Energy Company

John S. Keenan
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10 CFR 50.90

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SERIAL: BSEP 02-0035
TSC-2001-06

U. S. Nuclear Regulatory Commission
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Washington, DC 20555-0001

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2
DOCKET NOS. 50-325 AND 50-324/LICENSE NOS. DPR-71 AND DPR-62
REVISION OF REQUEST FOR LICENSE AMENDMENTS - FREQUENCY OF PERFORMANCE-BASED LEAKAGE RATE TESTING
(NRC TAC NOS. MB3470 AND MB3471)

Ladies and Gentlemen:

On November 26, 2001 (Serial: BSEP 01-0070), Carolina Power & Light (CP&L) Company submitted a license amendment application for the Brunswick Steam Electric Plant (BSEP), Units 1 and 2. The application proposed a revision to Technical Specification 5.5.12, "Primary Containment Leakage Rate Testing Program," to incorporate a one-time exception to the 10-year frequency for performance-based Type A leakage rate tests. The proposed exception would allow performance of a Type A test within 15 years, one month from the last Type A test for Unit 1 and within 15 years from the last Type A test for Unit 2.

The purpose of this letter is to revise CP&L's license amendment application for BSEP, Unit 1. Rather than the previously proposed test frequency of 15 years, one month for the Unit 1 Type A leakage rate test, CP&L is proposing to extend the Unit 1 Type A test frequency to 13 years, two months. The 13 year, two month test frequency will result in performance of the next Type A test no later than April 15, 2004. Revised typed and marked-up Technical Specification pages for BSEP Unit 1, reflecting the revised Type A test frequency extension, are enclosed.

In support of the originally proposed extension of the Type A test frequency from once in 10 years to once in 15 years, one month, a plant-specific, risk-based evaluation was performed and the results compared to the guidance in NRC Regulatory Guide 1.174, "An Approach For Using Probabilistic Risk Assessment In Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis," dated July 1998. Regulatory Guide 1.174 defines very small changes in risk as resulting in increases of core damage frequency (CDF) below 1E-6 per year and increases in Large Early Release

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Frequency (LERF) below 1E-7 per year. The BSEP-specific risk evaluation determined: (1) the increase in LERF resulting from the proposed Type A test frequency change from once in 10 years to once in 15 years to be 5.14E-8, and (2) the increase in LERF resulting from a Type A test frequency change of once in 3 years to once in 15 years to be 1.54E-7 per year. Although the BSEP-specific risk evaluation has not been modified to reflect a type A test frequency of 13 years, two months for BSEP, Unit 1, the increase in LERF resulting from a change in Type A test frequency from once in 10 years to once in 13 years, two months would obviously be less than the previously determined values. Based on this qualitative assessment, the increases in LERF associated with changing the Type A test frequency from once in 10 years to once in 13 years, two months would be less than the 1E-7 per year threshold for a very small change, as stated in Regulatory Guide 1.174. As such, the proposed extension of the Unit 1 Type A test frequency to 13 years, two months does not represent a risk significant change.

The November 26, 2001, license amendment application included an evaluation of significant hazards considerations, as set forth in 10 CFR 50.92. On January 8, 2002, the NRC published, in the *Federal Register* (i.e., 67 FR 926), a proposed determination that the license amendment application involves no significant hazards considerations. The initial evaluation of significant hazards considerations addressed a one-time extension of the Unit 1 Type A testing frequency to 15 years, one month. The revised one-time extension of the Unit 1 Type A testing frequency to 13 years, two months is fully bounded by the bases and conclusions for the original evaluation of significant hazards considerations. Therefore, re-publication in the *Federal Register* of a no significant hazards determination is not necessary.

CP&L requests that the NRC continue review of the BSEP, Unit 2 license amendment application to incorporate an exception allowing performance of a Type A test within 15 years from the last Type A test for Unit 2. In recent telephone conference calls, the NRC has requested that CP&L provide, in the plant-specific risk analysis, additional information regarding the significance of potential degradation of the containment liner in uninspectable areas. CP&L is preparing the requested information in support of the currently submitted Unit 2 license amendment application.

Please refer any questions regarding this submittal to Mr. Leonard R. Beller, Manager - Regulatory Affairs, at (910) 457-2073.

Sincerely,

John S. Keenan

John S. Keenan

WRM/wrm

Enclosures:
1. Typed Technical Specification Pages - Unit 1
2. Marked-up Technical Specification Pages - Unit 1

John S. Keenan, having been first duly sworn, did depose and say that the information contained herein is true and correct to the best of his information, knowledge and belief; and the sources of his information are officers, employees, and agents of Carolina Power & Light Company.

Dean S. Mason

Notary (Seal)

My commission expires: 8/29/04

cc: U. S. Nuclear Regulatory Commission, Region II
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U. S. Nuclear Regulatory Commission
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Chair - North Carolina Utilities Commission
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Division of Boiler and Pressure Vessel
North Carolina Department of Labor
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Mr. Mel Fry
Director - Division of Radiation Protection
North Carolina Department of Environment and Natural Resources
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ENCLOSURE 1

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2
DOCKET NOS. 50-325 AND 50-324/LICENSE NOS. DPR-71 AND DPR-62
REVISION OF REQUEST FOR LICENSE AMENDMENTS - FREQUENCY OF PERFORMANCE-BASED LEAKAGE RATE TESTING
(NRC TAC NOS. MB3470 AND MB3471)

Typed Technical Specifications Pages - Unit 1

5.5 Programs and Manuals

Primary Containment Leakage Rate Testing Program (continued)

a. Compensation of instrument accuracies applied to the primary containment leakage total in accordance with ANSI/ANS 56.8-1987 instead of ANSI/ANS 56.8-1994;

b. Following air lock door seal replacement, performance of door seal leakage rate testing with the gap between the door seals pressurized to 10 psig instead of air lock testing at P_a as specified in Nuclear Energy Institute Guideline 94-01, Revision 0;

c. Reduced duration Type A tests may be performed using the criteria and Total Time method specified in Bechtel Topical Report BN-TOP-1, Revision 1.

d. Performance of Type C leak rate testing of the hydrogen and oxygen monitor isolation valves is not required; and

e. Performance of Type C leak rate testing of the main steam isolation valves at a pressure less than P_a instead of leak rate testing at P_a as specified in ANSI/ANS 56.8-1994.

f. NEI 94-01 - 1995, Section 9.2.3: The first Type A test performed after the February 15, 1991, Type A test shall be performed no later than April 15, 2004.

The peak calculated primary containment internal pressure for the design basis loss of coolant accident, P_a, is 49 psig.

The maximum allowable primary containment leakage rate, L_a, shall be 0.5% of primary containment air weight per day at P_a.

Leakage rate acceptance criteria are:

a. Primary containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.60\ L_a$ for Type B and C tests and $\leq 0.75\ L_a$ for Type A tests.

b. Air lock testing acceptance criteria are:

    1) Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

(continued)

5.5 Programs and Manuals

Primary Containment Leakage Rate Testing Program (continued)

2) For each air lock door, leakage rate is ≤ 5 scfh when the gap between the door seals is pressurized to ≥ 10 psig.

The provisions of SR 3.0.3 are applicable to the Primary Containment Leakage Rate Testing Program frequencies.

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

The following reports shall be submitted in accordance with 10 CFR 50.4.

5.6.1 Occupational Radiation Exposure Report

------------------------------NOTE------------------------------
A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station.

A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors), for whom monitoring was performed, receiving an annual deep dose equivalent > 100 mrem and their associated collective deep dose equivalent (reported in person-rem) according to work and job functions (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling). This tabulation supplements the requirements of 10 CFR 20.2206. The dose assignments to various duty functions may be estimated based on pocket dosimeter, thermoluminescent dosimeter (TLD), electronic dosimeter or film badge measurements. Small exposures totalling < 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total deep dose equivalent received from external sources should be assigned to specific major work functions. The report, covering the previous calendar year, shall be submitted by April 30 of each year.

5.6.2 Annual Radiological Environmental Operating Report

------------------------------NOTE------------------------------
A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station.

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with

(continued)

5.6 Reporting Requirements

5.6.2 Annual Radiological Environmental Operating Report (continued)

the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of Table 3 in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

5.6.3 Radioactive Effluent Release Report

-----------------------------NOTE-----------------------------
A single submittal may be made for a multiple unit station. The submittal shall combine sections common to all units at the station.

The Radioactive Effluent Release Report covering the operation of the unit during the previous year shall be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and the Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR Part 50, Appendix I, Section IV.B.1.

5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the main steam safety/relief valves, shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

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5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

1. The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) for Specification 3.2.1;

2. The MINIMUM CRITICAL POWER RATIO (MCPR) for Specification 3.2.2;

3. The Allowable Value for Function 2.b, APRM Flow Biased Simulated Thermal Power—High, for Specification 3.3.1.1; and

4. The Allowable Values and power range setpoints for Rod Block Monitor Upscale Functions for Specification 3.3.2.1.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel" (latest approved version).

2. NEDO-32339-A, "Reactor Stability Long Term Solution: Enhanced Option I-A," July 1995.

3. NEDC-32339-P Supplement 1, "Reactor Stability Long Term Solution: Enhanced Option I-A ODYSY Computer Code," March 1994 (Approved in NRC Safety Evaluation dated January 4, 1996).

4. NEDO-32339 Supplement 3, "Reactor Stability Long Term Solution: Enhanced Option I-A Flow Mapping Methodology," August 1995 (Approved in NRC Safety Evaluation dated May 28, 1996).

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 Post Accident Monitoring (PAM) Instrumentation Report

When a report is required by Condition B or F of LCO 3.3.3.1, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area

As provided in paragraph 20.1601(c) of 10 CFR Part 20, the following controls shall be applied to high radiation areas in place of the controls required by paragraph 20.1601(a) and (b) of 10 CFR Part 20:

5.7.1 <u>High Radiation Areas with Dose Rates not exceeding 1.0 rem/hour (at 30 centimeters from the radiation sources or from any surface penetrated by the radiation)</u>

a. Each accessible entryway to such an area shall be barricaded and conspicuously posted as a high radiation area. Such barricades may be opened as necessary to permit entry or exit of personnel or equipment.

b. Access to, and activities in, each such area shall be controlled by means of a Radiation Work Permit (RWP) or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures and personnel continuously escorted by such individuals may be exempted from the requirement for an RWP or equivalent while performing their assigned duties provided that they are following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual or group entering such an area shall possess:

1. A radiation monitoring device that continuously displays radiation dose rates in the area ("radiation monitoring and indicating device"); or

2. A radiation monitoring device that continuously integrates the radiation dose rates in the area and alarms when the device's dose alarm setpoint is reached ("alarming dosimeter"), with an appropriate alarm setpoint; or

3. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area; or

(continued)

5.7 High Radiation Area

5.7.1 High Radiation Areas with Dose Rates not exceeding 1.0 rem/hour (at 30 centimeters from the radiation sources or from any surface penetrated by the radiation) (continued)

4. A self-reading dosimeter and,

(a) Be under the surveillance, as specified in the RWP or equivalent, of an individual at the work site, qualified in radiation protection procedures, equipped with a radiation monitoring and indicating device who is responsible for controlling personnel radiation exposure within the area, or

(b) Be under the surveillance, as specified in the RWP or equivalent, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area.

e. Except for individuals qualified in radiation protection procedures, entry into such areas shall be made only after dose rates in the area have been established and entry personnel are knowledgeable of them.

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour (at 30 centimeters from the radiation source or from any surface penetrated by the radiation), but less than 500 rads/hour (at 1 meter from the radiation source or from any surface penetrated by the radiation)

a. Each accessible entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked door, gate, or guard that prevents unauthorized entry, and in addition:

1. All such door and gate keys shall be maintained under the administrative control of the shift superintendent or the radiation control supervisor or designated representative; and

2. Doors and gates shall remain locked or guarded except during periods of personnel or equipment entry or exit.

(continued)

5.7 High Radiation Area

5.7.2 <u>High Radiation Areas with Dose Rates Greater than 1.0 rem/hour (at 30 centimeters from the radiation source or from any surface penetrated by the radiation), but less than 500 rads/hour (at 1 meter from the radiation source or from any surface penetrated by the radiation)</u> (continued)

b. Access to, and activities in, each such area shall be controlled by means of an RWP or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures may be exempted from the requirement for an RWP or equivalent while performing radiation surveys in such areas provided that they are following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual (whether alone or in a group) entering such an area shall possess:

1. An alarming dosimeter with an appropriate alarm setpoint; or

2. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area with the means to communicate with and control every individual in the area; or

3. A direct-reading dosimeter and,

(a) Be under the surveillance, as specified in the RWP or equivalent, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring and indicating device who is responsible for controlling personnel exposure within the area, or

(b) Be under the surveillance, as specified in the RWP or equivalent, by means of closed circuit television, of personnel qualified in radiation

(continued)

5.7 High Radiation Area

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour (at 30 centimeters from the radiation source or from any surface penetrated by the radiation), but less than 500 rads/hour (at 1 meter from the radiation source or from any surface penetrated by the radiation) (continued)

protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with and control every individual in the area; or

4. A radiation monitoring and indicating device in those cases where the options of Specifications 5.7.2.d.2 and 5.7.2.d.3, above, are impractical or determined to be inconsistent with the "As Low As is Reasonably Achievable" principle.

e. Except for individuals qualified in radiation protection procedures, entry into such areas shall be made only after dose rates in the area have been established and entry personnel are knowledgeable of them.

f. Such individual areas that are within a larger area that is controlled as a high radiation area, where no enclosure exists for purpose of locking and where no enclosure can reasonably be constructed around the individual area need not be controlled by a locked door or gate, but shall be barricaded and conspicuously posted as a high radiation area, and a conspicuous, clearly visible flashing light shall be activated at the area as a warning device.

ENCLOSURE 2

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2
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Marked-up Technical Specifications Pages - Unit 1

5.5 Programs and Manuals

<u>Primary Containment Leakage Rate Testing Program</u> (continued)

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b. Following air lock door seal replacement, performance of door seal leakage rate testing with the gap between the door seals pressurized to 10 psig instead of air lock testing at P_a as specified in Nuclear Energy Institute Guideline 94-01, Revision 0;

c. Reduced duration Type A tests may be performed using the criteria and Total Time method specified in Bechtel Topical Report BN-TOP-1, Revision 1.

d. Performance of Type C leak rate testing of the hydrogen and oxygen monitor isolation valves is not required; and

e. Performance of Type C leak rate testing of the main steam isolation valves at a pressure less than P_a instead of leak rate testing at P_a as specified in ANSI/ANS 56.8-1994.

INSERT

The peak calculated primary containment internal pressure for the design basis loss of coolant accident, P_a, is 49 psig.

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a. Primary containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.60\ L_a$ for Type B and C tests and $\leq 0.75\ L_a$ for Type A tests.

b. Air lock testing acceptance criteria are:

   1) Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

   2) For each air lock door, leakage rate is ≤ 5 scfh when the gap between the door seals is pressurized to ≥ 10 psig.

The provisions of SR 3.0.3 are applicable to the Primary Containment Leakage Rate Testing Program frequencies.

INSERT for TS 5.5.12

f. NEI 94-01 - 1995, Section 9.2.3: The first Type A test performed after the February 15, 1991, Type A test shall be performed no later than April 15, 2004.