February 1, 1996

Mr. W. R. Campbell
Vice President
Brunswick Steam Electric Plant
Carolina Power & Light Company
Post Office Box 10429
Southport, North Carolina 28461

SUBJECT: ISSUANCE OF AMENDMENT NO. 181 TO FACILITY OPERATING LICENSE NO. DPR-71 AND AMENDMENT NO. 213 TO FACILITY OPERATING LICENSE NO. DPR-62 REGARDING 10 CFR PART 50 APPENDIX J, OPTION B - BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2 (BSEP 95-0316) (TAC NOS. M93679 AND M93680)

Dear Mr. Campbell:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 181 to Facility Operating License No. DPR-71 and Amendment No. 213 to Facility Operating License No. DPR-62 for Brunswick Steam Electric Plant, Units 1 (BSEP1) and 2 (BSEP2). The amendments change the Technical Specifications (TS) in response to your submittal dated September 13, 1995, as amended on November 27, 1995, and January 29, 1996.

The amendments revise the BSEP1 and BSEP2 TS to permit the use of 10 CFR Part 50, Appendix J, Option B, Performance-Based Containment Leakage Rate Testing.

A copy of the related Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly 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

Docket Nos. 50-325
and 50-324

Enclosures:
1. Amendment No. 181 to License No. DPR-71
2. Amendment No. 213 to License No. DPR-62
3. Safety Evaluation

cc w/enclosures: See next page

FILENAME - G:\BRUNSWICK\BR93679.AMD *See Previous Concurrence

LA:PDII-1*	PM:PDII-1*	SCSB *	OGC	D:PDII-1	
EDunnington	DTrimble	CBerlinger	CHoller	DMATTHEWS	
1/23 /96	1/26 /96	1/29 /96	2/1 /96	2/1 /96	
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AMENDMENT NO. 181 TO FACILITY OPERATING LICENSE NO. DPR-71 - BRUNSWICK, UNIT 1
AMENDMENT NO. 213 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. 181
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 September 13, 1995, as amended on November 27, 1995, and January 29, 1996, 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. 181 , 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 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

DBMatthews

David B. Matthews, 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: February 1, 1996

ATTACHMENT TO LICENSE AMENDMENT NO. 181

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
XV	XV
3/4 6-2	3/4 6-2
3/4 6-3	3/4 6-3
3/4 6-5	3/4 6-5
3/4 6-6	3/4 6-6
B 3/4 6-1	B 3/4 6-1
--	B 3/4 6-1a
--	B 3/4 6-1b
B 3/4 6-2	B 3/4 6-2
--	B 3/4 6-2a
6-16	6-16
6-17	6-17
--	6-17a

ADMINISTRATIVE CONTROLS

PRIMARY CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Primary containment leakage rates shall be limited to:

a. An overall integrated leakage rate of:

1. Less than or equal to L_a, 0.5 percent by weight of the containment air per 24 hours at P_a, 49 psig.

2. Deleted.

b. A combined leakage rate of less than or equal to 0.60 L_a for penetrations and valves subject to Type B and C tests when pressurized to P_a in accordance with the Primary Containment Leakage Rate Testing Program described in Specification 6.8.3.4, except for main steam line isolation valves*.

c. *Less than or equal to 11.5 scf per hour for any one main steam line isolation valve when tested at 25 psig.

APPLICABILITY: When PRIMARY CONTAINMENT INTEGRITY is required per Specification 3.6.1.1.

ACTION:

With:

a. The measured overall integrated primary containment leakage rate exceeding 0.75 L_a, or

b. The measured combined leakage rate for penetrations and valves subject to Type B and C tests in accordance with the Primary Containment Leakage Rate Testing Program, except for main steam line isolation valves*, exceeding 0.60 L_a, or

c. The measured leakage rate exceeding 11.5 scf per hour for any one main steam line isolation valve,

restore:

a. The overall integrated leakage rate(s) to less than or equal to 0.75 L_a, and

b. The combined leakage rate for penetrations and valves subject to Type B and C tests in accordance with the Primary Containment Leakage Rate Testing Program, except for main steam line isolation valves*, to less than or equal to 0.60 L_a, and

* Exemption to Appendix "J" of 10 CFR 50.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

c. The leakage rate to less than or equal to 11.5 scf per hour for any one main steam line isolation valve,

prior to increasing reactor coolant system temperature above 212°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2.1 Perform required primary containment leakage rate testing in accordance with the Primary Containment Leakage Rate Testing Program described in Specification 6.8.3.4.

4.6.1.2.2 Main steam line isolation valves shall be leak tested at least once per 18 months.

(Pages 3/4 6-3A and 3/4 6-3B have been deleted.)

4.6.1.3 Each primary containment air lock shall be demonstrated OPERABLE:

a. By verifying the seal leakage rate to be less than or equal to 5 scf per hour when the gap between the door seals is pressurized to 10 psig:

1. Within 7 days following each closing, except when the air lock is being used for multiple entries, then at least once per 30 days, and

2. Prior to establishing PRIMARY CONTAINMENT INTEGRITY when the air lock has been used and no maintenance has been performed on the air lock, and

3. When the air lock seal has been replaced.

b. By conducting an overall air lock leakage test at P_a, 49 psig, and by verifying that the overall air lock leakage is within its limit:

1. At least once per 30 months, and

2. Prior to establishing PRIMARY CONTAINMENT INTEGRITY when maintenance (except for seal replacement) has been performed on the air lock that would affect the air lock sealing capability.

c. By verification of air lock interlock OPERABILITY:

1. Prior to establishing PRIMARY CONTAINMENT INTEGRITY when the air lock has been used, and

2. prior to and following a drywell entry when PRIMARY CONTAINMENT INTEGRITY is required, and

3. Following the performance of maintenance affecting the air lock interlock.

PRIMARY CONTAINMENT STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.4 The structural integrity of the primary containment shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.4.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

With the structural integrity of the primary containment not conforming to the above requirements, restore the structural integrity to within the limits prior to increasing the Reactor Coolant System temperature above 212°F.

SURVEILLANCE REQUIREMENTS

4.6.1.4.1 The structural integrity of the exposed accessible interior and exterior surfaces of the primary containment, including the liner plate, shall be determined during the shutdown for each Type A containment leakage rate test by visual inspection of those surfaces. This inspection shall be performed prior to the Type A containment leakage rate test and during two other refueling outages before the next Type A test if the interval for the Type A test has been extended to 10 years, to verify no apparent changes in appearance or other abnormal degradation.

4.6.1.4.2 Reports Any abnormal degradation of the primary containment structure detected during the above required inspections shall be reported to the Commission pursuant to Specification 6.9.2. This Special Report shall include a description of the condition of the concrete, the inspection procedure, the tolerances on cracking, and the corrective actions taken.

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 PRIMARY CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10 CFR Part 100 during accident conditions.

3/4.6.1.2 PRIMARY CONTAINMENT LEAKAGE

The safety design basis for the primary containment is that it must withstand the pressures and temperatures of the limiting DBA without exceeding the design leakage rate.

The DBA that postulates the maximum release of radioactive material within primary containment is a LOCA. In analysis of this accident, it is assumed that primary containment is OPERABLE such that release of fission products to the environment is controlled by the rate of primary containment leakage.

Analytical methods and assumptions involving the primary containment are presented in References 6 and 7.

The maximum allowable leakage rate for the primary containment (L_a) is 0.5 percent by weight of the containment air per 24 hours at the maximum peak containment pressure (P_a) of 49 psig.

A Primary Containment Leakage Rate Testing Program has been established in accordance with 10 CFR 50.54(o) to implement the requirements of 10 CFR Part 50, Appendix J, Option B (Reference 1). The Primary Containment Leakage Rate Testing Program conforms with NRC Regulatory Guide 1.163, dated September 1995, "Performance-Based Containment Leak-Rate Testing Program" (Reference 2) and Nuclear Energy Institute (NEI) 94-01, Revision 0, dated July 26, 1995, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50 Appendix J" (Reference 3) with the exception of:

1. NEI 94-01, Section 8.0, "Testing Methodologies for Type A, B and C Tests" states that "Type A, Type B and Type C tests should be performed using the technical methods and techniques specified in ANSI/ANS 56.8-1994, or other alternative testing methods that have been approved by the NRC." The Brunswick Plant takes exception to ANSI 56.8 flowmeter accuracy requirements based upon compensation of instrument inaccuracies applied to the containment leakage total per the previous revision of the standard. Brunswick Plant administrative procedures and databases already effectively address instrument error. Brunswick Plant uses standard glass tube and ball type flowmeters with a 5 percent of full scale accuracy. Readings are compensated for back pressure, temperature, and test medium variables. To overcome the less accurate flowmeter use, an equipment error is applied to the results of each test. The square root of the sum of the squares of the equipment errors for the tests is also added to the cumulative containment leakage total. This method is consistent with ANSI 56.8-1987 Appendix E and provides conservative assurance that the

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.2 PRIMARY CONTAINMENT LEAKAGE (Continued)

cumulative containment leakage total accounts for instrument inaccuracy. No such instrument error analysis or accounting is required per ANSI/ANS 56.8-1994.

2. NEI 94-01, Section 10.2.2.2, "Repairs or Adjustments of Airlocks" states that following maintenance on an air lock pressure retaining boundary, one of the following tests shall be completed:

   a. The air lock shall be tested at a pressure of not less than P_a, or

   b. Leakage rate testing at P_a shall be performed on the affected area or component.

   A previously approved exemption to 10 CFR 50, Appendix J that allows the performance of air lock door seal leakage rate testing at a pressure less than P_a following door seal replacement instead of air lock testing at P_a has been retained and is listed as an exception in Technical Specification 6.8.3.4.

The leakage rate acceptance criteria of $\leq 0.60\ L_a$ for the combined Type B and C tests and $\leq 0.75\ L_a$ for the Type A test ensures a primary containment configuration, including equipment hatches, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analyses. Primary containment operability is maintained by limiting leakage to $\leq 1.0\ L_a$.

Individual leakage rates specified for the primary containment air lock are addressed in Specification 3.6.1.3.

Operating experience with the main steam line isolation valves has indicated that degradation has occasionally occurred in the leak tightness of the valves; therefore, the special requirement for testing these valves.

Exemptions from the requirements of 10 CFR Part 50 have been granted for the main steam isolation valve leak testing and leakage calculations.

NRC Regulatory Guide 1.163, (Reference 2) endorses NEI 94-01 (Reference 3) which in turn identifies ANSI/ANS 56.8-1994, "Containment System Leakage Testing Requirements" (Reference 4) as an acceptable standard regarding leakage-rate test methods, procedures, and analyses. 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, November 1, 1972 (References 5 and 6).

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."

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.2 PRIMARY CONTAINMENT LEAKAGE (Continued)

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."

4. ANSI/ANS 56.8-1994, "Containment System Leakage Testing Requirements".

5. CP&L Letter to Mr. D. B. Vassallo, "Integrated Leak Rate Test," October 20, 1983.

6. NRC Letter from Mr. D. B. Vassallo to Mr. E. E. Utley, December 9, 1983.

7. Updated FSAR, Section 6.2.

8. Updated FSAR, Section 15.6.4.

3/4.6.1.3 PRIMARY CONTAINMENT AIR LOCKS

The primary containment air lock forms part of the primary containment pressure boundary. As such, air lock integrity and leak tightness are essential for maintaining primary containment leakage rate to within limits in the event of a DBA. Not maintaining air lock integrity or leak tightness may result in a leakage rate in excess of that assumed in unit safety analysis.

The DBA that postulates the maximum release of radioactive material within primary containment is a LOCA. In analysis of this accident, it is assumed that primary containment is OPERABLE such that release of fission products to the environment is controlled by the rate of primary containment leakage. In the analysis of this accident, it is assumed that primary containment is OPERABLE, such that release of fission products to the environment is controlled by the rate of primary containment leakage. The primary containment is designed with a maximum allowable leakage rate (L_a) of 0.5 percent by weight of the containment air per 24 hours at the maximum peak containment pressure (P_a) of 49 psig. This allowable leakage rate forms the basis for the acceptance criteria imposed on the surveillance requirements associated with the air lock.

The primary containment air lock is required to be OPERABLE. For the air lock to be considered OPERABLE, the air lock interlock mechanism must be OPERABLE, the air lock must be in compliance with the Type B air lock leakage test, and both air lock doors must be OPERABLE. The interlock allows only one air lock door to be opened at a time. This provision ensures that a gross breach of primary containment does not exist when primary containment is required to be OPERABLE. Closure of a single door in each air lock is sufficient to provide a leak tight barrier following postulated events. Nevertheless, both doors are kept closed when the air lock is not being used for normal entry and exit from primary containment.

Maintaining primary containment air locks OPERABLE requires compliance with the leakage rate test requirements of 10 CFR 50, Appendix J as established in the Primary Containment Leakage Rate Testing Program. The Primary Containment Leakage Rate Testing Program has been established in accordance with 10 CFR 50.54(o) to implement the requirements of 10 CFR Part 50, Appendix J, Option B (Reference 1). The Primary Containment Leakage Rate Testing Program conforms with NRC Regulatory Guide 1.163, dated September 1995, "Performance-Based Containment Leak-Rate Testing Program" and Nuclear Energy Institute (NEI) 94-01, Revision 0, dated July 26, 1995, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50 Appendix J" as modified by the exceptions listed in Specification 6.8.3.4 (References 2 and 3).

An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. This is considered reasonable since either air lock door is capable of providing a fission product barrier in the event of a DBA.

Only one closed door in each air lock is required to maintain the integrity of the containment. In the event of an inoperable door interlock, locking shut the inner door will ensure containment integrity while permitting access to the lock for maintenance and surveillance testing.

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 PRIMARY CONTAINMENT AVERAGE AIR TEMPERATURE

The limitation in containment average air temperature ensures that the containment peak air temperature does not exceed the design temperature of 300°F during LOCA conditions and is consistent with the accident analyses.

6.8 PROCEDURES, PROGRAMS, AND MANUALS

6.8.1 Written procedures shall be established, implemented, and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, November 1972.

b. Refueling operations.

c. Surveillance and test activities of safety related equipment.

d. Security Plan implementation.

e. Emergency Plan implementation.

f. Fire Protection Program implementation.

g. OFFSITE DOSE CALCULATION MANUAL implementation.

h. PROCESS CONTROL PROGRAM implementation.

i. Quality Assurance Program for effluent and environmental monitoring using the guidance in Regulatory Guide 1.21, Revision 1, June 1974, and Regulatory Guide 4.1, Revision 1, April 1975.

6.8.2 Temporary changes to procedures of Specification 6.8.1 above, any other procedures that affect nuclear safety, and proposed tests or experiments may be made provided:

a. The intent of the original procedure, proposed test or experiment is not altered.

b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator License on the unit affected.

c. The change is documented, reviewed pursuant to Specifications 6.5.2.1 and 6.5.2.2 and approved by the General Manager - Brunswick Plant or his previously designated alternate within 14 days of implementation.

6.8.3 Programs and Manuals

The following programs shall be established, implemented, and maintained:

6.8.3.1 Primary Coolant Sources Outside Containment

A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The program shall include the following:

PROCEDURES, PROGRAMS, AND MANUALS (Continued)

1. Preventive maintenance and periodic visual inspection requirements, and

2. Integrated leak test requirements for each system at refueling cycle intervals or less.

6.8.3.2 In-Plant Radiation Monitoring

A program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

1. Training of personnel,

2. Procedures for monitoring, and

3. Provisions for maintenance of sampling and analysis equipment.

6.8.3.3 Post-Accident Sampling

A program which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include the following:

1. Training of personnel,

2. Procedures for sampling and analysis, and

3. Provisions for maintenance of sampling and analysis equipment.

6.8.3.4 Primary Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995 as modified by the following exceptions:

1. Compensation of instrument inaccuracies applied to the containment leakage total per ANSI/ANS 56.8-1987 instead of ANSI/ANS 56.8-1994.

2. 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.

PROCEDURES, PROGRAMS, AND MANUALS (Continued)

The peak calculated 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.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Regional Administrator of the Regional Office unless otherwise noted.

STARTUP REPORTS

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant.

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. 213
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 September 13, 1995, as amended on November 27, 1995, and January 29, 1996, 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. 213, 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 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

DBMatthews

David B. Matthews, 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: February 1, 1996

ATTACHMENT TO LICENSE AMENDMENT NO. 213

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
XV	XV
3/4 6-2	3/4 6-2
3/4 6-3	3/4 6-3
3/4 6-5	3/4 6-5
3/4 6-6	3/4 6-6
B 3/4 6-1	B 3/4 6-1
--	B 3/4 6-1a
--	B 3/4 6-1b
B 3/4 6-2	B 3/4 6-2
--	B 3/4 6-2a
6-16	6-16
6-17	6-17
--	6-17a

PRIMARY CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Primary containment leakage rates shall be limited to:

a. An overall integrated leakage rate of:

1. Less than or equal to L_a, 0.5 percent by weight of the containment air per 24 hours at P_a, 49 psig.

2. Deleted.

b. A combined leakage rate of less than or equal to 0.60 L_a for penetrations and valves subject to Type B and C tests when pressurized to P_a in accordance with the Primary Containment Leakage Rate Testing Program described in Specification 6.8.3.4, except for main steam line isolation valves*.

c. *Less than or equal to 11.5 scf per hour for any one main steam line isolation valve when tested at 25 psig.

APPLICABILITY: When PRIMARY CONTAINMENT INTEGRITY is required per Specification 3.6.1.1.

ACTION:

With:

a. The measured overall integrated primary containment leakage rate exceeding 0.75 L_a; or

b. The measured combined leakage rate for penetrations and valves subject to Type B and C tests in accordance with the Primary Containment Leakage Rate Testing Program, except for main steam line isolation valves*, exceeding 0.60 L_a, or

c. The measured leakage rate exceeding 11.5 scf per hour for any one main steam line isolation valve,

restore:

a. The overall integrated leakage rate(s) to less than or equal to 0.75 L_a, and

b. The combined leakage rate for penetrations and valves subject to Type B and C tests in accordance with the Primary Containment Leakage Rate Testing Program, except for main steam line isolation valves*, to less than or equal to 0.60 L_a, and

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* Exemption to Appendix "J" of 10 CFR 50.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

c. The leakage rate to less than or equal to 11.5 scf per hour for any one main steam line isolation valve,

prior to increasing reactor coolant system temperature above 212°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2.1 Perform required primary containment leakage rate testing in accordance with the Primary Containment Leakage Rate Testing Program described in Specification 6.8.3.4.

4.6.1.2.2 Main steam line isolation valves shall be leak tested at least once per 18 months.

(Pages 3/4 6-3A has been deleted.)

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each primary containment air lock shall be demonstrated OPERABLE:

a. By verifying the seal leakage rate to be less than or equal to 5 scf per hour when the gap between the door seals is pressurized to 10 psig:

 1. Within 7 days following each closing, except when the air lock is being used for multiple entries, then at least once per 30 days, and

 2. Prior to establishing PRIMARY CONTAINMENT INTEGRITY when the air lock has been used and no maintenance has been performed on the air lock, and

 3. When the air lock seal has been replaced.

b. By conducting an overall air lock leakage test at P_a, 49 psig, and by verifying that the overall air lock leakage is within its limit:

 1. At least once per 30 months, and

 2. Prior to establishing PRIMARY CONTAINMENT INTEGRITY when maintenance (except for seal replacement) has been performed on the air lock that would affect the air lock sealing capability.

c. By verification of air lock interlock OPERABILITY:

 1. Prior to establishing PRIMARY CONTAINMENT INTEGRITY when the air lock has been used, and

 2. prior to and following a drywell entry when PRIMARY CONTAINMENT INTEGRITY is required, and

 3. Following the performance of maintenance affecting the air lock interlock.

PRIMARY CONTAINMENT STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.4 The structural integrity of the primary containment shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.4.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

With the structural integrity of the primary containment not conforming to the above requirements, restore the structural integrity to within the limits prior to increasing the Reactor Coolant System temperature above 212°F.

SURVEILLANCE REQUIREMENTS

4.6.1.4.1 The structural integrity of the exposed accessible interior and exterior surfaces of the primary containment, including the liner plate, shall be determined during the shutdown for each Type A containment leakage rate test by visual inspection of those surfaces. This inspection shall be performed prior to the Type A containment leakage rate test and during two other refueling outages before the next Type A test if the interval for the Type A test has been extended to 10 years, to verify no apparent changes in appearance or other abnormal degradation.

4.6.1.4.2 Reports Any abnormal degradation of the primary containment structure detected during the above required inspections shall be reported to the Commission pursuant to Specification 6.9.2. This Special Report shall include a description of the condition of the concrete, the inspection procedure, the tolerances on cracking, and the corrective actions taken.

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 PRIMARY CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10 CFR Part 100 during accident conditions.

3/4.6.1.2 PRIMARY CONTAINMENT LEAKAGE

The safety design basis for the primary containment is that it must withstand the pressures and temperatures of the limiting DBA without exceeding the design leakage rate.

The DBA that postulates the maximum release of radioactive material within primary containment is a LOCA. In analysis of this accident, it is assumed that primary containment is OPERABLE such that release of fission products to the environment is controlled by the rate of primary containment leakage.

Analytical methods and assumptions involving the primary containment are presented in References 6 and 7.

The maximum allowable leakage rate for the primary containment (L_a) is 0.5 percent by weight of the containment air per 24 hours at the maximum peak containment pressure (P_a) of 49 psig.

A Primary Containment Leakage Rate Testing Program has been established in accordance with 10 CFR 50.54(o) to implement the requirements of 10 CFR Part 50, Appendix J, Option B (Reference 1). The Primary Containment Leakage Rate Testing Program conforms with NRC Regulatory Guide 1.163, dated September 1995, "Performance-Based Containment Leak-Rate Testing Program" (Reference 2) and Nuclear Energy Institute (NEI) 94-01, Revision 0, dated July 26, 1995, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50 Appendix J" (Reference 3) with the exception of:

1. NEI 94-01, Section 8.0, "Testing Methodologies for Type A, B and C Tests" states that "Type A, Type B and Type C tests should be performed using the technical methods and techniques specified in ANSI/ANS 56.8-1994, or other alternative testing methods that have been approved by the NRC." The Brunswick Plant takes exception to ANSI 56.8 flowmeter accuracy requirements based upon compensation of instrument inaccuracies applied to the containment leakage total per the previous revision of the standard. Brunswick Plant administrative procedures and databases already effectively address instrument error. Brunswick Plant uses standard glass tube and ball type flowmeters with a 5 percent of full scale accuracy. Readings are compensated for back pressure, temperature, and test medium variables. To overcome the less accurate flowmeter use, an equipment error is applied to the results of each test. The square root of the sum of the squares of the equipment errors for the tests is also added to the cumulative containment leakage total. This method is consistent with ANSI N56.8-1987 Appendix E and provides conservative assurance that the

3/4.6.1.2 PRIMARY CONTAINMENT LEAKAGE (Continued)

cumulative containment leakage total accounts for instrument inaccuracy. No such instrument error analysis or accounting is required per ANSI/ANS 56.8-1994.

2. NEI 94-01, Section 10.2.2.2, "Repairs or Adjustments of Airlocks" states that following maintenance on an air lock pressure retaining boundary, one of the following tests shall be completed:

   a. The air lock shall be tested at a pressure of not less than P_a, or

   b. Leakage rate testing at P_a shall be performed on the affected area or component.

   A previously approved exemption to 10 CFR 50, Appendix J that allows the performance of air lock door seal leakage rate testing at a pressure less than P_a following door seal replacement instead of air lock testing at P_a has been retained and is listed as an exception in Technical Specification 6.8.3.4.

The leakage rate acceptance criteria of $\leq 0.60\ L_a$ for the combined Type B and C tests and $\leq 0.75\ L_a$ for the Type A test ensures a primary containment configuration, including equipment hatches, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analyses. Primary containment operability is maintained by limiting leakage to $\leq 1.0\ L_a$.

Individual leakage rates specified for the primary containment air lock are addressed in Specification 3.6.1.3.

Operating experience with the main steam line isolation valves has indicated that degradation has occasionally occurred in the leak tightness of the valves; therefore, the special requirement for testing these valves.

Exemptions from the requirements of 10 CFR Part 50 have been granted for the main steam isolation valve leak testing and leakage calculations.

NRC Regulatory Guide 1.163, (Reference 2) endorses NEI 94-01 (Reference 3) which in turn identifies ANSI/ANS 56.8-1994, "Containment System Leakage Testing Requirements" (Reference 4) as an acceptable standard regarding leakage-rate test methods, procedures, and analyses. 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, November 1, 1972 (References 5 and 6).

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."

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.2 PRIMARY CONTAINMENT LEAKAGE (Continued)

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."

4. ANSI/ANS 56.8-1994, "Containment System Leakage Testing Requirements".

5. CP&L Letter to Mr. D. B. Vassallo, "Integrated Leak Rate Test," October 20, 1983.

6. NRC Letter from Mr. D. B. Vassallo to Mr. E. E. Utley, December 9, 1983.

7. Updated FSAR, Section 6.2.

8. Updated FSAR, Section 15.6.4.

3/4.6.1.3 PRIMARY CONTAINMENT AIR LOCKS

The primary containment air lock forms part of the primary containment pressure boundary. As such, air lock integrity and leak tightness are essential for maintaining primary containment leakage rate to within limits in the event of a DBA. Not maintaining air lock integrity or leak tightness may result in a leakage rate in excess of that assumed in unit safety analysis.

The DBA that postulates the maximum release of radioactive material within primary containment is a LOCA. In analysis of this accident, it is assumed that primary containment is OPERABLE such that release of fission products to the environment is controlled by the rate of primary containment leakage. In the analysis of this accident, it is assumed that primary containment is OPERABLE, such that release of fission products to the environment is controlled by the rate of primary containment leakage. The primary containment is designed with a maximum allowable leakage rate (L_a) of 0.5 percent by weight of the containment air per 24 hours at the maximum peak containment pressure (P_a) of 49 psig. This allowable leakage rate forms the basis for the acceptance criteria imposed on the surveillance requirements associated with the air lock.

The primary containment air lock is required to be OPERABLE. For the air lock to be considered OPERABLE, the air lock interlock mechanism must be OPERABLE, the air lock must be in compliance with the Type B air lock leakage test, and both air lock doors must be OPERABLE. The interlock allows only one air lock door to be opened at a time. This provision ensures that a gross breach of primary containment does not exist when primary containment is required to be OPERABLE. Closure of a single door in each air lock is sufficient to provide a leak tight barrier following postulated events. Nevertheless, both doors are kept closed when the air lock is not being used for normal entry and exit from primary containment.

Maintaining primary containment air locks OPERABLE requires compliance with the leakage rate test requirements of 10 CFR 50, Appendix J as established in the Primary Containment Leakage Rate Testing Program. The Primary Containment Leakage Rate Testing Program has been established in accordance with 10 CFR 50.54(o) to implement the requirements of 10 CFR Part 50, Appendix J, Option B (Reference 1). The Primary Containment Leakage Rate Testing Program conforms with NRC Regulatory Guide 1.163, dated September 1995, "Performance-Based Containment Leak-Rate Testing Program" and Nuclear Energy Institute (NEI) 94-01, Revision 0, dated July 26, 1995, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50 Appendix J" as modified by the exceptions listed in Specification 6.8.3.4 (References 2 and 3).

An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. This is considered reasonable since either air lock door is capable of providing a fission product barrier in the event of a DBA.

Only one closed door in each air lock is required to maintain the integrity of the containment. In the event of an inoperable door interlock, locking shut the inner door will ensure containment integrity while permitting access to the lock for maintenance and surveillance testing.

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 PRIMARY CONTAINMENT AVERAGE AIR TEMPERATURE

The limitation in containment average air temperature ensures that the containment peak air temperature does not exceed the design temperature of 300°F during LOCA conditions and is consistent with the accident analyses.

6.8 PROCEDURES, PROGRAMS, AND MANUALS

6.8.1 Written procedures shall be established, implemented, and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, November 1972.

b. Refueling operations.

c. Surveillance and test activities of safety related equipment.

d. Security Plan implementation.

e. Emergency Plan implementation.

f. Fire Protection Program implementation.

g. OFFSITE DOSE CALCULATION MANUAL implementation.

h. PROCESS CONTROL PROGRAM implementation.

i. Quality Assurance Program for effluent and environmental monitoring using the guidance in Regulatory Guide 1.21, Revision 1, June 1974, and Regulatory Guide 4.1, Revision 1, April 1975.

6.8.2 Temporary changes to procedures of Specification 6.8.1 above, any other procedures that affect nuclear safety, and proposed tests or experiments may be made provided:

a. The intent of the original procedure, proposed test or experiment is not altered.

b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator License on the unit affected.

c. The change is documented, reviewed pursuant to Specifications 6.5.2.1 and 6.5.2.2 and approved by the General Manager - Brunswick Plant or his previously designated alternate within 14 days of implementation.

6.8.3 Programs and Manuals

The following programs shall be established, implemented, and maintained:

6.8.3.1 Primary Coolant Sources Outside Containment

A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The program shall include the following:

PROCEDURES, PROGRAMS, AND MANUALS (Continued)

1. Preventive maintenance and periodic visual inspection requirements, and

2. Integrated leak test requirements for each system at refueling cycle intervals or less.

6.8.3.2 In-Plant Radiation Monitoring

A program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

1. Training of personnel,

2. Procedures for monitoring, and

3. Provisions for maintenance of sampling and analysis equipment.

6.8.3.3 Post-Accident Sampling

A program which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include the following:

1. Training of personnel,

2. Procedures for sampling and analysis, and

3. Provisions for maintenance of sampling and analysis equipment.

6.8.3.4 Primary Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995 as modified by the following exceptions:

1. Compensation of instrument inaccuracies applied to the containment leakage total per ANSI/ANS 56.8-1987 instead of ANSI/ANS 56.8-1994.

2. 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.

PROCEDURES, PROGRAMS, AND MANUALS (Continued)

The peak calculated 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.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Regional Administrator of the Regional Office unless otherwise noted.

STARTUP REPORTS

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 181 TO FACILITY OPERATING LICENSE NO. DPR-71

AND AMENDMENT NO. 213 TO FACILITY OPERATING LICENSE NO. DPR-62

CAROLINA POWER & LIGHT COMPANY

BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2

DOCKET NOS. 50-325 AND 50-324

1.0 INTRODUCTION

By letter dated September 13, 1995, as amended on November 27, 1995, and January 29, 1996, the Carolina Power and Light Company (the licensee) proposed a change to the Technical Specifications (TS) for the Brunswick Steam Electric Plant, Units No. 1 and No. 2 (BSEP-1&2). The proposed TS change would permit the use of 10 CFR Part 50, Appendix J, Option B, Performance-Based Requirements.

2.0 BACKGROUND

Compliance with Appendix J provides assurance that the primary containment, and those systems and components which penetrate the primary containment, do not exceed the allowable leakage rate values specified in the Technical Specifications and Bases. The allowable leakage rate is determined so that the leakage assumed in the safety analyses is not exceeded.

On February 4, 1992, the NRC published a notice in the Federal Register (57 FR 4166) discussing a planned initiative to eliminate requirements marginal to safety which impose a significant regulatory burden. Appendix J of 10 CFR Part 50, "Primary Containment Leakage Testing for Water-Cooled Power Reactors" was considered for this initiative and the staff undertook a study of possible changes to this regulation. The study examined the previous performance history of domestic containments and examined the effect on risk of a revision to the requirements of Appendix J. The results of this study are reported in NUREG-1493, "Performance-Based Leak-Test Program."

Based on the results of this study, the staff developed a performance-based approach to containment leakage rate testing. On September 12, 1995, the NRC approved issuance of this revision to 10 CFR Part 50, Appendix J, which was subsequently published in the Federal Register on September 26, 1995, and became effective on October 26, 1995. The revision added Option B "Performance-Based Requirements" to Appendix J to allow licensees to voluntarily replace the prescriptive testing requirements of Appendix J with testing requirements based on both overall and individual component leakage rate performance.

Regulatory Guide 1.163, September 1995, "Performance-Based Containment Leak Test Program," was developed as a method acceptable to the NRC staff for implementing Option B. This regulatory guide states that the Nuclear Energy Institute (NEI) document NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J" provides methods acceptable to the NRC staff for complying with Option B with four exceptions which are described therein.

Option B requires that the regulatory guide or other implementation document used by a licensee to develop a performance-based leakage testing program must be included, by general reference, in the plant technical specifications.

Regulatory Guide 1.163 specifies an extension in Type A test frequency from three approximately equally spaced tests in 10 years to at least one test in 10 years based upon two consecutive successful tests. Type B tests may be extended up to a maximum of 10 years based upon completion of two consecutive successful tests and Type C tests may be extended up to 5 years based on two consecutive successful tests.

By letter dated October 20, 1995, NEI proposed TS implementing Option B. After some discussion, the staff and NEI agreed on a set of model TS which were transmitted to NEI in a letter dated November 2, 1995. These TS are to serve as a model for licensees to develop plant-specific TS in preparing amendment requests to implement Option B.

In order for a licensee to determine the performance of each component, factors that are indicative of or affect performance, such as an administrative leakage limit, must be established. The administrative limit is selected to be indicative of the potential onset of component degradation. Although these limits are subject to NRC inspection to assure that they are selected in a reasonable manner, they are not technical specification requirements. Failure to meet an administrative limit requires the licensee to return to the minimum test interval for that component.

Option B requires that the licensee maintain records to show that the criteria for Type A, B and C tests have been met. In addition, the licensee must maintain comparisons of the performance of the overall containment system and the individual components to show that the test intervals are adequate. These records are subject to NRC inspection.

3.0 SPECIFIC TS CHANGES

The licensee proposed the following changes to the BSEP 1&2 TS.

Specification 3/4.6.1.2, Primary Containment Leakage Rates

- Specification 3.6.1.2.a.2 - The leakage rate limit for reduced pressure containment leakage testing is being deleted since reduced pressure testing is not an option available under the Regulatory Guide 1.163/NEI 94-01 performance-based leakage testing program.

- Specification 3.6.1.2.b - This specification and the associated ACTION statements are being revised to indicate that the combined leakage rate for valves and penetrations shall be in accordance with the new Primary Containment Leakage Rate Testing Program. The Table 3.6.3-1 reference is being removed because this table was previously relocated from the Technical Specifications (Amendments 149 and 179 for Unit 1 and Unit 2, respectively).

- Specification 4.6.1.2 - This specification regarding the schedule and criteria for demonstrating primary containment leakage rates has been renumbered to 4.6.1.2.1 and revised to require the performance of primary containment leakage rate testing in accordance with the Primary Containment Leakage Rate Testing Program described in the new Specification 6.8.3.4.

    Since a performance-based leakage testing program is being established and will be controlled through the Primary Containment Leakage Rate Testing Program, the detailed requirements regarding Type B and C testing (Specification 4.6.1.2.d) are being deleted.

    - Containment air locks are required to be tested pursuant to Specification 4.6.1.3; therefore, Specification 4.6.1.2.e is duplicative and is being deleted.

    - Specification 4.6.1.2.f, which requires that main steam isolation valves be tested at least once per 18 months, is being retained and renumbered to 4.6.1.2.2. Main steam line isolation valves are outside the scope of performance-based testing and leakage testing of these valves will continue to be performed in accordance with current Technical Specification 4.6.1.2.f.

    - The statement that the provisions of Specification 4.0.2 are not applicable to 24 month surveillance intervals (Specification 4.6.1.2.h) is being deleted, since the only references to a 24-month test frequency (in Specification 4.6.1.2.d) are being removed. The performance-based leakage rate testing program will establish the specific test frequencies based on component and system performance.

Specification 3/4.6.1.3, Primary Containment Air Locks

- Specification 4.6.1.3.a.1 - This specification is being revised to require verification of the primary containment air lock seal leakage rate within 7 days (versus the current 72 hour period) after each closing. The 7-day test frequency is consistent with the test frequency specified in NEI 94-01, Section 10.2.2.1 (Containment Airlocks — Test Intervals). Verification of the primary containment air lock seal leakage rate after multiple entries is being required at least every 30 days (versus the current 72 hours). The 30-day test frequency for the period of multiple containment entries is also consistent with the frequency specified in NEI 94-01, Section 10.2.2.1 (Containment Airlocks — Test Intervals).

o Specification 4.6.1.3.b.1 - This specification is being revised to require performance of an overall air lock leakage rate test every 30 months instead of the current six-month frequency. The 30-month test frequency is consistent with the periodic test frequency specified in NEI 94-01, Section 10.2.2.1 (Containment Airlocks — Test Intervals).

Specification 3/4.6.1.4, Primary Containment Structural Integrity

o Specification 4.6.1.4.1 - This specification is being revised to require the performance of visual examinations of the accessible areas of the primary containment interior and exterior surfaces consistent with NRC Regulatory Guide 1.163, Section C, paragraph 3. These examinations will be conducted prior to performing a Type A test and during two other refueling outages before the next Type A test (if the interval for the Type A test has been extended to 10 years).

Specification 6.8, Procedures and Programs

o Specification 6.8 - The title for Specification 6.8 is being revised to "Procedures, Programs, and Manuals."

o Specification 6.8.3 - This specification is being revised to add a title "Programs and Manuals." Specification 6.8.3.a is being renumbered to 6.8.3.1, Specification 6.8.3.b is being renumbered to 6.8.3.2, and Specification 6.8.3.c is being renumbered to 6.8.3.3. A new Specification 6.8.3.4 is being added to describe the Primary Containment Leakage Rate Testing Program. The Specification states that the program implements primary containment leakage rate testing as required by 10 CFR 50, Appendix J, Option B and the guidelines contained in NRC Regulatory Guide 1.163, dated September 1995. Specification 6.8.3.4 will identify the plant-specific value for L_a, the maximum allowable primary containment leakage rate, and the value for P_a, the peak calculated primary containment internal pressure. The values of P_a and L_a are currently referenced in Specifications 3/4.6.1.2 and 3/4.6.1.3 and are not being changed as part of this license amendment request. Also, Specification 6.8.3.4 will identify the approved plant-specific exceptions to the implementation process stipulated in NRC Regulatory Guide 1.163 and NEI 94-01. The only exceptions requested by the licensee and to be included in this specification are:

(1) an exception to leakage rate flowmeter instrument accuracy requirements. The licensee's flowmeter does not meet the instrument accuracy requirements specified in ANSI/ANS 56.8-1994, which is the industry standard on testing methodologies referenced in NEI 94-01. To overcome the

effects of the less accurate flowmeter, the licensee will apply an instrument error to the results of each test. The licensee additionally will add the square root of the sum of the squares of the instrument errors for the tests to the cumulative containment leakage total. This approach is consistent with ANSI 56.8-1987, Appendix E, and conservatively accounts for instrument inaccuracy.

(2) an exception regarding the testing of containment airlocks following door seal replacements. Instead of requiring testing of the airlock at P_a, as called for by NEI 94-01, following such maintenance, the exception allows an alternative test to be performed wherein the gap between the door seals is pressurized to 10 psig. This approach is consistent with a previously approved exemption to 10 CFR Part 50, Appendix J.

Bases

- Bases for Specification 3/4.6.1.2 - This section has been expanded to clarify the safety objectives stipulated in 10 CFR 50, Appendix A, "General Design Criteria for Nuclear Power Plants" for primary containments. Also, the regulatory requirements contained in 10 CFR 50, Appendix J, Option B and the requirements provided in NRC Regulatory Guide 1.163 and NEI 94-01 for implementation of a performance-based containment leakage rate testing program have been described, along with any exceptions being taken to these regulatory positions. A reference to the granting of a previous exemption from 10 CFR 50 regarding the testing of air locks after each opening is being deleted since the requirements of Regulatory Guide 1.163 and NEI 94-01 will now be followed.

- Bases for Specification 3/4.6.1.3 - This section has been expanded to address the regulatory requirements for air locks for primary containments. The regulatory requirements contained in 10 CFR 50, Appendix J, Option B and the implementation requirements provided in NRC Regulatory Guide 1.163 and NEI 94-01 pertaining to air lock leakage testing have been described.

- Bases for Specification 3/4.6.1.4 - This section has been modified to include the regulatory basis for performing the visual examinations of the accessible containment interior and exterior surfaces.

4.0 EVALUATION

The licensee's September 13, 1995 letter to the NRC, as amended on November 27, 1995, and January 29, 1996, proposed TS changes that permit the use of Option B of the revised 10 CFR Part 50 Appendix J. These TS changes refer to Regulatory Guide 1.163, September 1995, "Performance-Based Containment Leak Test Program" which specifies a method acceptable to the NRC for complying with Option B. This requires the TS changes listed above.

Option B permits a licensee to choose Type A; or Type B and C; or Type A, B and C testing to be done on a performance basis. The licensee has elected to perform Type A, B and C testing on a performance basis.

The exception to the guidelines contained in Regulatory Guide 1.163 and NEI 94-01 regarding leakage rate flowmeter accuracy is acceptable in that the licensee will conservatively account for the inaccuracy of its flowmeter through adherence to the methodology described in ANSI/ANS 56.8-1987, Appendix E.

The exception to the guidelines contained in Regulatory Guide 1.163 and NEI 94-01 regarding the testing of containment airlocks after seal replacement by means of pressurization between the seals to 10 psig is consistent with a previously approved exemption to 10 CFR Part 50, Appendix J and, therefore, is acceptable.

These TS changes replace specific surveillance requirements related to primary containment leakage rate testing and the corresponding acceptance criteria and test methods with a requirement to perform the required testing in accordance with Option B and approved exemptions using the guidance in Regulatory Guide 1.163 and the exceptions thereto discussed above. The staff has reviewed the licensee's proposed changes and finds that all the important elements of the guidance provided in the NRC letter to NEI dated November 2, 1995, are included in the TS proposed by the licensee and that the proposed changes meet the requirements of 10 CFR Part 50, Appendix J, Option B. The staff therefore concludes that the licensee's request to implement 10 CFR Part 50, Appendix J, Option B is acceptable.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Virginia State official was notified of the proposed issuance of the amendment. The State official had no comment.

6.0 ENVIRONMENTAL CONSIDERATION

These 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 change 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 these amendments involve no significant hazards consideration and there has been no public comment on such finding (60 FR 63739 dated December 12, 1995). Accordingly, these 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.

7.0 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.

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