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.)

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 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

ADMINISTRATIVE CONTROLS

SECTION PAGE

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 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.