ENCLOSURE 5

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2
NRC DOCKET NOS. 50-325 AND 50-324
OPERATING LICENSE NOS. DPR-71 AND DPR-62
SUPPLEMENT TO REQUESTS FOR LICENSE AMENDMENTS
CONTAINMENT LEAKAGE RATE TESTING

MARKED-UP TECHNICAL SPECIFICATION PAGES - UNIT 1

ADMINISTRATIVE CONTROLS

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,

CONTAINMENT SYSTEMS

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 ~~, or~~.

2. Deleted. ~~Less than or equal to L_t, 0.357 percent by weight of the containment air per 24 hours at a reduced pressure of P_t, 25 psig.~~

b. A combined leakage rate of less than or equal to 0.60 L_a for ~~all~~ penetrations and ~~all~~ valves ~~listed in Table 3.6.3-1~~, except for main steam line isolation valves* ~~, subject to Type B and C tests when pressurized to P_a, 49 psig.~~

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:

Subject to Type B and C tests in accordance with the Primary Containment Leakage Rate Testing Program,

With:

a. The measured overall integrated primary containment leakage rate exceeding 0.75 L_a ~~or 0.75 L_t, as applicable,~~ , or

b. The measured combined leakage rate for ~~all~~ penetrations and ~~all~~ valves ~~listed in Table 3.6.3-1~~, except for main steam line isolation valves*, ~~subject to Type B and C tests~~ 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 ~~or 0.75 L_t, as applicable,~~ and

b. The combined leakage rate for ~~all~~ penetrations and ~~all~~ valves ~~listed in Table 3.6.3-1~~, except for main steam line isolation valves*, ~~subject to Type B and C tests~~ 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 ~~The primary containment leakage rates shall be demonstrated in accordance with the schedule and criteria specified in 10 CFR 50, Appendix J, as modified by approved exemptions. The provisions of Technical Specification 4.0.2 are not applicable to the test intervals specified in 10 CFR 50, Appendix J.~~ Perform required primary containment leakage rate testing in accordance with the Primary Containment Leakage Rate Testing Program described in Specification 6.8.3.4.

~~a. Deleted.~~

~~b. Deleted.~~

~~c. Deleted.~~

~~d. Type B and C tests shall be conducted with gas at P_a, 49 psig, at intervals no greater than 24 months except for tests involving:~~

~~1. Air locks.~~

~~2. Main steam line isolation valves.~~

~~e. Air locks shall be tested and demonstrated OPERABLE per Surveillance Requirement 4.6.1.3.~~

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

~~g. All test leakage rates shall be calculated using observed data converted to absolute values. Error analyses shall be performed to select a balanced integrated leakage measurement system.~~

~~h. The provisions of Specification 4.0.2 are not applicable to 24 month surveillance intervals.~~

(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 ~~72 hours~~ 7 days following each closing, except when the air lock is being used for multiple entries, then at least once per ~~72 hours~~ 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 ~~six months#~~ 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.

* Exemption to Appendix J of 10 CFR 50.

~~# The provisions of Specification 4.0.2 are not applicable.~~

CONTAINMENT SYSTEMS

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 a visual inspection of those surfaces. This inspection shall be performed prior to the Type A containment leakage rate test 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.

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,

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

REPLACE WITH INSERT #1

The limitations on primary containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure of 49 psig, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a or 0.75 L_t, as applicable, during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

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 main steam isolation valve leak testing, ~~testing of airlocks after each opening,~~ and leakage calculation methods.

REPLACE WITH INSERT #2

Appendix J, paragraph III.A.3 requires that all Type A (Containment Integrated Leak Rate) tests be conducted in accordance with American National Standard (ANSI) N45.4-1972, "Leakage Rate Testing of Containment Structures for Nuclear Reactors," March 16, 1972. In addition to the Total Time and Point-to-Point methods described in that standard, the Mass Point method, when used with a test duration of at least 24 hours, is an acceptable method to use to calculate leakage rates. A typical description of the Mass Point method can be found in ANSI/ANS 56.8-1987, "Containment System Leakage Testing Requirements," January 20, 1987. 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 1 and 2).

References:

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

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

INSERT #1:

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, Revision 0, 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 cumulative containment leakage total

INSERT #1: (Continued)

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

The leakage rate acceptance criteria of ≤ 0.60 L_a for the combined Type B and C tests and ≤ 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 ≤ 1.0 L_a.

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

INSERT #2:

NRC Regulatory Guide 1.163, Revision 0 (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.

2. NRC Regulatory Guide 1.163, Revision 0, 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."

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.

REPLACE WITH INSERT # 3

3/4.6.1.3 PRIMARY CONTAINMENT AIR LOCKS

~~The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on PRIMARY CONTAINMENT INTEGRITY and leak rate given in Specifications 3.6.1.1 and 3.6.1.2. The specification makes allowances for the fact that there may be long periods of time when the air locks will be in a closed and secured position during reactor operation. 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.~~

3/4.6.1.4 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 ~~Type A leakage tests~~ the Primary Containment Leakage Rate Testing Program is sufficient to demonstrate this capability.

INSERT #4

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.

INSERT #3:

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, Revision 0, 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 approved exceptions (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.

References:

1. 10 CFR Part 50, Appendix J.

2. NRC Regulatory Guide 1.163, Revision 0, 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."

INSERT #4:

References:

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

2. NRC Regulatory Guide 1.163, Revision 0, dated September 1995, "Performance-Based Containment Leak-Rate Testing Program."

6.8 PROCEDURES ~~AND~~, 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

~~6.8.3~~ The following programs shall be established, implemented, and maintained:

6.8.3.1 ~~a.~~ 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, ~~AND~~ 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 ~~b.~~ 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 ~~c.~~ 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.

→ INSERT #5

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.

INSERT #5:

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, 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.

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.

ENCLOSURE 6

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2
NRC DOCKET NOS. 50-325 AND 50-324
OPERATING LICENSE NOS. DPR-71 AND DPR-62
SUPPLEMENT TO REQUESTS FOR LICENSE AMENDMENTS
CONTAINMENT LEAKAGE RATE TESTING

MARKED-UP TECHNICAL SPECIFICATION PAGES - UNIT 2

ADMINISTRATIVE CONTROLS

CONTAINMENT SYSTEMS

PRIMARY CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

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,

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 ~~or~~.

2. Deleted. ~~Less than or equal to L_t, 0.357 percent by weight of the containment air per 24 hours at a reduced pressure of P_t, 25 psig.~~

b. A combined leakage rate of less than or equal to 0.60 L_a for ~~all~~ penetrations and ~~all~~ valves ~~listed in Table 3.6.3-1~~, except for main steam line isolation valves*, ~~subject to Type B and C tests when pressurized to P_a, 49 psig.~~

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:

Subject to Type B and C tests in accordance with the Primary Containment Leakage Rate Testing Program,

With:

a. The measured overall integrated primary containment leakage rate exceeding 0.75 L_a, ~~or 0.75 L_t, as applicable,~~ or

b. The measured combined leakage rate for ~~all~~ penetrations and ~~all~~ valves ~~listed in Table 3.6.3-1~~, except for main steam line isolation valves*, ~~subject to Type B and C tests~~ 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, ~~or 0.75 L_t, as applicable,~~ and

b. The combined leakage rate for ~~all~~ penetrations and ~~all~~ valves ~~listed in Table 3.6.3-1~~, except for main steam line isolation valves*, ~~subject to Type B and C tests~~ 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 ~~The primary containment leakage rates shall be demonstrated in accordance with the schedule and criteria specified in 10 CFR 50, Appendix J, as modified by approved exemptions. The provisions of Technical Specification 4.0.2 are not applicable to the test intervals specified in 10 CFR 50, Appendix J.~~ Perform required primary containment leakage rate testing in accordance with the Primary Containment Leakage Rate Testing Program described in Specification 6.8.3.4.

~~a. Deleted.~~

~~b. Deleted.~~

~~c. Deleted.~~

~~d. Type B and C tests shall be conducted with gas at P_a, 49 psig, at intervals no greater than 24 months except for tests involving:~~

~~1. Air locks.~~

~~2. Main steam line isolation valves.~~

~~e. Air locks shall be tested and demonstrated OPERABLE per Surveillance Requirement 4.6.1.3.~~

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

~~g. All test leakage rates shall be calculated using observed data converted to absolute values. Error analyses shall be performed to select a balanced integrated leakage measurement system.~~

~~h. The provisions of Specification 4.0.2 are not applicable to 24 month surveillance intervals.~~

(Page 3/4 6-3A has 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 ~~72 hours~~ 7 days following each closing, except when the air lock is being used for multiple entries, then at lease once per ~~72 hours~~ 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 that at P_a, 49 psig, and by verifying that the overall air lock leakage is within its limit:

   1. At least once per ~~six months~~ 30 months, and

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

* Exemption of Appendix J of 10 CFR 50.

~~# The provisions of Specification 4.0.2 are not applicable~~

CONTAINMENT SYSTEMS

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 a visual inspection of those surfaces. This inspection shall be performed prior to the Type A containment leakage rate test 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.

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,

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.

REPLACE WITH INSERT #1

3/4.6.1.2 PRIMARY CONTAINMENT LEAKAGE

~~The limitations on primary containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure of 49 psig, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a or 0.75 L_t, as applicable, during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.~~

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 main steam isolation valve leak testing, ~~testing of airlocks after each opening,~~ and leakage calculation methods.

REPLACE WITH INSERT #2

~~Appendix J, paragraph III.A.3 requires that all Type A (Containment Integrated Leak Rate) tests be conducted in accordance with American National Standard (ANSI) N45.4-1972, "Leakage Rate Testing of Containment Structures for Nuclear Reactors," March 16, 1972. In addition to the Total Time and Point-to-Point methods described in that standard, the Mass Point method, when used with a test duration of at least 24 hours, is an acceptable method to use to calculate leakage rates. A typical description of the Mass Point method can be found in ANSI/ANS 56.8-1987, "Containment System Leakage Testing Requirements," January 20, 1987. 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 1 and 2).~~

~~References:~~

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

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

INSERT #1:

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, Revision 0, 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 cumulative containment leakage total

INSERT #1: (Continued)

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

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.

INSERT #2:

NRC Regulatory Guide 1.163, Revision 0 (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.

2. NRC Regulatory Guide 1.163, Revision 0, 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."

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.

REPLACE WITH INSERT #3

3/4.6.1.3 PRIMARY CONTAINMENT AIR LOCKS

~~The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on PRIMARY CONTAINMENT INTEGRITY and leak rate given in Specifications 3.6.1.1 and 3.6.1.2. The specification makes allowances for the fact that there may be long periods of time when the air locks will be in a closed and secured position during reactor operation. 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.~~

3/4.6.1.4 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 ~~Type A leakage tests~~ the Primary Containment Leakage Rate Testing Program is sufficient to demonstrate this capability.

→ INSERT #4

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.

INSERT #3:

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, Revision 0, 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 approved exceptions (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.
References:

1. 10 CFR Part 50, Appendix J.

2. NRC Regulatory Guide 1.163, Revision 0, 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."

INSERT #4:

References:

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

2. NRC Regulatory Guide 1.163, Revision 0, dated September 1995, "Performance-Based Containment Leak-Rate Testing Program."

6.8 PROCEDURES, ~~AND~~ 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

~~6.8.3~~ The following programs shall be established, implemented, and maintained:

6.8.3.1 ~~a.~~ 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, ~~AND~~ 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 ~~b.~~ 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 ~~c.~~ 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.

INSERT

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.

INSERT #5:

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, 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.

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.

ENCLOSURE 7

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2
NRC DOCKET NOS. 50-325 AND 50-324
OPERATING LICENSE NOS. DPR-71 AND DPR-62
SUPPLEMENT TO REQUESTS FOR LICENSE AMENDMENTS
CONTAINMENT LEAKAGE RATE TESTING

TYPED TECHNICAL SPECIFICATION PAGES - UNIT 1

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.

* Exemption to Appendix J of 10 CFR 50.

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, Revision 0, 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.

The leakage rate acceptance criteria of ≤ 0.60 L_a for the combined Type B and C tests and ≤ 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 ≤ 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, Revision 0 (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.
2. NRC Regulatory Guide 1.163, Revision 0, 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."
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, Revision 0, 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 approved exceptions (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.

2. NRC Regulatory Guide 1.163, Revision 0, 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, Revision 0, 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, 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.

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.

ENCLOSURE 8

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2
NRC DOCKET NOS. 50-325 AND 50-324
OPERATING LICENSE NOS. DPR-71 AND DPR-62
SUPPLEMENT TO REQUESTS FOR LICENSE AMENDMENTS
CONTAINMENT LEAKAGE RATE TESTING

1YPED TECHNICAL SPECIFICATION PAGES - UNIT 2

ADMINISTRATIVE CONTROLS

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 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.

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* Exemption to Appendix J of 10 CFR 50.

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, Revision 0, 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.

The leakage rate acceptance criteria of ≤ 0.60 L_a for the combined Type B and C tests and ≤ 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 ≤ 1.0 La.

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, Revision 0 (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.

2. NRC Regulatory Guide 1.163, Revision 0, 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."

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, Revision 0, 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 approved exceptions (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.

2. NRC Regulatory Guide 1.163, Revision 0, 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, Revision 0, 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, 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.

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.