ATTACHMENT B-1

PROPOSED CHANGES TO
APPENDIX A, TECHNICAL SPECIFICATIONS,
OF FACILITY OPERATING LICENSES NPF-37 AND NPF-66,
BYRON NUCLEAR POWER STATION, UNITS 1 & 2

Revised Pages:

I
1-3
1-4
3/4 6-1
3/4 6-2
3/4 6-3
3/4 6-4
3/4 6-5
3/4 6-12
B 3/4 6-1

INDEX

DEFINITIONS

SECTION PAGE

La

1.15.a The maximum allowable primary containment leakage rate, La, shall be 0.10% of the primary containment air weight per day at the calculated peak containment pressure (Pa).

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.12 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the sample) of the sum of the average beta and gamma energies per disintegration (MeV/d) for the radionuclides in the sample.

ENGINEERED SAFETY FEATURES RESPONSE TIME

1.13 The ENGINEERED SAFETY FEATURES (ESF) RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF Actuation Setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable.

FREQUENCY NOTATION

1.14 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1.

IDENTIFIED LEAKAGE

1.15 IDENTIFIED LEAKAGE shall be:

a. Leakage (except CONTROLLED LEAKAGE) into closed systems, such as pump seal or valve packing leaks that are captured and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of Leakage Detection Systems or not to be PRESSURE BOUNDARY LEAKAGE, or

c. Reactor Coolant System leakage through a steam generator to the Secondary Coolant System.

MASTER RELAY TEST

1.16 A MASTER RELAY TEST shall be the energization of each master relay and verification of OPERABILITY of each relay. The MASTER RELAY TEST shall include a continuity check of each associated slave relay.

MEMBER(S) OF THE PUBLIC

1.17 MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the licensee, its contractors or vendors and persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational, or other purposes not associated with the plant.

OFFSITE DOSE CALCULATION MANUAL

1.18 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Sections 6.8.4e and f, and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Radioactive Effluent Release Reports required by Specifications 6.9.1.6 and 6.9.1.7.

OPERABLE - OPERABILITY

1.19 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s), and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its function(s) are also capable of performing their related support function(s).

OPERATING LIMITS REPORT

1.19.a The OPERATING LIMITS REPORT is the unit-specific document that provides operating limits for the current operating reload cycle. These cycle-specific operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.9. Plant operation within these operating limits is addressed in individual specifications.

OPERATIONAL MODE - MODE

1.20 An OPERATIONAL MODE (i.e., MODE) shall correspond to any one inclusive combination of core reactivity condition, power level, and average reactor coolant temperature specified in Table 1.2.

PHYSICS TESTS

1.21 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the core and related instrumentation: (1) described in Chapter 14.0 of the FSAR, (2) authorized under the provisions of 10 CFR 50.59, or (3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

1.22 PRESSURE BOUNDARY LEAKAGE shall be leakage (except steam generator tube leakage) through a nonisolable fault in a Reactor Coolant System component body, pipe wall, or vessel wall.

P_a

1.20.a P_a shall be the maximum calculated primary containment pressure (44.4 psig) for the design basis loss of coolant accident.

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

Without primary CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

a. At least once per 31 days by verifying that all penetrations* not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions, except as provided in Table 3.6-1 of Specification 3.6.3, or for containment isolation valves that are open under administrative controls;

b. By verifying that each containment air lock is in compliance with the requirements of Specification 3.6.1.3; and

c. ~~After each closing of each penetration subject to Type B testing, except the containment air locks, if opened following a Type A or B test, by leak rate testing the seal with gas at a pressure not less than P_a, 44.4 psig, and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Specification 4.6.1.2d. for all other Type B and C penetrations, the combined leakage rate is less than 0.60 L_a.~~

*Except valves, blind flanges, and deactivated automatic valves which are located inside the containment and are locked, sealed or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except that such verification need not be performed more often than once per 92 days.

By performing containment leakage testing in accordance with Regulatory Guide 1.163, September 1995, and 10 CFR 50, Appendix J, Option B.

CONTAINMENT SYSTEMS

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited to:

a. An overall integrated leakage rate of~~:~~ less than or equal to La at Pa.

~~1) Less than or equal to L_a, 0.10% by weight of the containment air per 24 hours at P_a, 44.4 psig, or~~

~~2) Less than or equal to L_t, 0.07% by weight of the containment air per 24 hours for Unit 1 (0.07% by weight of the containment air per 24 hours for Unit 2) at P_t, 22.2 psig.~~

b. A combined leakage rate of less than 0.60 L_a for all penetrations and valves subject to Type B and C tests, when pressurized to P_a.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With either the measured overall integrated containment leakage rate exceeding 0.75 L_a ~~or 0.75 L_t, as applicable,~~ or the measured combined leakage rate for all penetrations and valves subject to Types B and C tests exceeding 0.60 L_a, restore the overall integrated leakage rate to less than 0.75 L_a ~~or less than 0.75 L_t, as applicable,~~ and the combined leakage rate for all penetrations subject to Type B and C tests to less than 0.60 L_a prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The containment leakage rates shall be demonstrated ~~at the following test schedule and shall be determined in conformance with the criteria specified in Appendix J of 10 CFR Part 50 using the methods and provisions of ANSI N45.4-1972:~~ in accordance with Regulatory Guide 1.163, September 1995, and 10 CFR 50, Appendix J, Option B.

a. Type A (Overall Integrated Containment Leakage Rate) testing shall be conducted in accordance with ~~the requirements specified in Appendix J to 10 CFR 50, as modified by approved exemptions;~~ Regulatory Guide 1.163, September 1995, and 10 CFR 50, Appendix J, Option B.

SURVEILLANCE REQUIREMENTS (Continued)

b. ~~If any periodic Type A test fails to meet either 0.75 L_a or 0.75 L_t, the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet 0.75 L_a, a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet 0.75 L_a;~~

c. The accuracy of each Type A test shall be verified by a supplemental test ~~which:~~ conducted in accordance with Regulatory Guide 1.163, September 1995, and 10 CFR 50, Appendix J, Option B.

~~1) Confirms the accuracy of the test by verifying that the supplemental test result, L_c, minus the sum of the Type A and the superimposed leak, L_o, is equal to or less than 0.25 L_a or 0.25 L_t;~~

~~2) Has a duration sufficient to establish accurately the change in leakage rate between the Type A test and the supplemental test; and~~

~~3) Requires that the rate at which gas is injected into the containment or bled from the containment during the supplemental test is between 0.75 L_a and 1.25 L_a.~~

d. Type B and C tests shall be conducted ~~with gas at a pressure not less than P_a, 44.4 psig, at intervals no greater than 24 months except for tests involving:~~ in accordance with Regulatory Guide 1.163, September 1995, and 10 CFR 50, Appendix J, Option B.

~~1) Air locks, and~~

~~2) Purge supply and exhaust isolation valves with resilient material seals.~~

e. Air locks shall be tested and demonstrated OPERABLE by the requirements of Specification 4.6.1.3;

f. Purge supply and exhaust isolation valves with resilient material seals shall be tested and demonstrated OPERABLE by the requirements of Specification 4.6.1.7.3 or 4.6.1.7.4, as applicable; and

g. The provisions of Specification 4.0.2 are not applicable.

The reporting requirements and frequency of Type A tests shall be in accordance with Regulatory Guide 1.163, September 1995, and 10 CFR 50, Appendix J, Option B.

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE with:

a. Both doors closed except when the air lock is being used for normal transit entry and exits through the containment, then at least one air lock door shall be closed, and

b. An overall air lock leakage rate of less than or equal to 0.05 L_a at P_a~~, 44.4 psig~~.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With one containment air lock door inoperable:

1. Maintain at least the OPERABLE air lock door closed and either restore the inoperable air lock door to OPERABLE status within 24 hours or lock the OPERABLE air lock door closed;

2. Operation may then continue until performance of the next required overall air lock leakage test provided that the OPERABLE air lock door is verified to be locked closed at least once per 31 days;

3. Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; and

4. The provisions of Specification 3.0.4 are not applicable.

b. With the containment air lock inoperable, except as the result of an inoperable air lock door, maintain at least one air lock door closed; restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

Insert A

~~a. Within 72 hours following each closing, except when the air lock is being used for multiple entries, then at least once per 72 hours, by~~

(1) Verifying that the door seal leakage is less than 0.0024La (1.11 SCFH) when the volume between the door seals is pressurized to greater than or equal to 3 psig by means of a permanently installed continuous pressurization and leakage monitoring system, or

(2) Verifying that the door seal leakage is less than 0.01La (4.63 SCFH) as determined by precision flow measurements when measured for at least 30 seconds with the volume between the seals at a constant pressure of greater than or equal to 10 psig;

b. By conducting overall air lock leakage tests ~~at not less than P_a, 44.4 psig, and verifying the overall air lock leakage rate is within its limit:~~

Insert B

~~1) At least once per 6 months,* and~~

~~2) Prior to establishing CONTAINMENT INTEGRITY when maintenance has been performed on the air lock that could affect the air lock sealing capability.**~~

c. At least once per 6 months by verifying that only one door in each air lock can be opened at a time.

~~d. At least once per 6 months by verifying that the seal leakage is less than 0.01La (4.63 SCFH) as determined by precision flow measurements when measured for at least 30 seconds with the volume between the seals at a constant pressure of greater than or equal to 10 psig;~~

Insert C

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

~~**This represents an exemption to Appendix J of 10 CFR Part 50, Paragraph III D.2(b)(ii).~~

Insert A

a. By conducting airlock seal leakage tests following each closing in accordance with Regulatory Guide 1.163, September 1995, and 10 CFR 50, Appendix J, Option B, by

Insert B

in accordance with Regulatory Guide 1.163, September 1995, and 10 CFR 50, Appendix J, Option B.

Insert C

d. By verifying that the airlock seal leakage tests is less than 0.01 La (4.63 SCFH) as determined by precision flow measurements when measured for at least 30 seconds with the volume between the seals at a constant pressure of greater than or equal to 10 psig in accordance with Regulatory Guide 1.163, September 1995, and 10 CFR 50, Appendix J, Option B.;

SURVEILLANCE REQUIREMENTS

4.6.1.7.1 Each 48-inch containment purge supply and exhaust isolation valve(s) shall be verified closed and power removed at least once per 31 days.

4.6.1.7.2 Each 8-inch containment purge supply and exhaust isolation valve shall be verified to be positioned in accordance with Specification 3.6.1.7b at least once per 31 days.

Leakage testing shall be conducted on

4.6.1.7.3 ~~At least once per 6 months on a STAGGERED TEST BASIS,~~ the inboard and outboard valves with resilient material seals in each closed 48-inch containment purge supply and exhaust penetration ~~shall be demonstrated OPERABLE~~ by verifying that the measured leakage rate is less than 0.05 L_a ~~when pressurized to at least P_a, 44.4 psig.~~ on a STAGGERED TEST BASIS in accordance with Regulatory Guide 1.163, September 1995, and 10 CFR 50, Appendix J, Option B.

4.6.1.7.4~~,~~ ~~At least once per 3 months,~~ each 8-inch containment purge supply and exhaust isolation valve with resilient material seals ~~shall be demonstrated OPERABLE~~ by verifying that the measured leakage rate is less than 0.01 L_a ~~when pressurized to at least P_a, 44.4 psig.~~ in accordance with Regulatory Guide 1.163, September 1995, and 10 CFR 50, Appendix J, Option B.

Leakage testing shall be conducted on

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 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 safety analyses. This restriction, in conjunction with the leakage rate limitation, will limit the SITE BOUNDARY radiation doses to within the dose guideline values of 10 CFR Part 100 during accident conditions.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on 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, 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 test to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates are consistent with the requirements of Appendix J of 10 CFR Part 50, Option B, Regulatory Guide 1.163, September 1995, Nuclear Energy Institute document NEI 94-01, and ANSI/ANS-56.8-1994.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and containment leak rate. Surveillance testing of the air lock seals provides assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests. ~~The use of precision flow measurements of Specification 4.6.1.3.a(2) must be used whenever the continuous monitoring capability in the control room is lost.~~

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that: (1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the outside atmosphere of 0.1 psig, and (2) the containment peak pressure does not exceed the design pressure of 50 psig during steam line break conditions.

The maximum increase in peak pressure expected to be obtained from a cold leg double-ended break event is 44.4 psig. The limit of 1.0 psig for initial positive containment pressure will limit the total pressure to 44.4 psig, which is higher than the ~~FSAR Chapter~~ accident analysis calculated peak pressure assuming a limit of 0.3 psig for initial positive containment pressure, but is considerably less than the design pressure of 50 psig.

UFSAR Chapter 15

ATTACHMENT B-2

PROPOSED CHANGES TO
APPENDIX A, TECHNICAL SPECIFICATIONS,
OF FACILITY OPERATING LICENSES NPF-72 AND NPF-77,
BRAIDWOOD NUCLEAR POWER STATION, UNITS 1 & 2

<u>Revised Pages</u>:

I
1-3
1-4
3/4 6-1
3/4 6-2
3/4 6-3
3/4 6-4
3/4 6-5
3/4 6-12
B 3/4 6-1

DEFINITIONS

SECTION PAGE

1.15.a $\frac{La}{}$ The maximum allowable primary containment leakage rate, La, shall be 0.10% of the primary containment air weight per day at the calculated peak containment pressure (Pa).

E - AVERAGE DISINTEGRATION ENERGY

1.12 E shall be the average (weighted in proportion to the concentration of each radionuclide in the sample) of the sum of the average beta and gamma energies per disintegration (MeV/d) for the radionuclides in the sample.

ENGINEERED SAFETY FEATURES RESPONSE TIME

1.13 The ENGINEERED SAFETY FEATURES (ESF) RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF Actuation Setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable.

FREQUENCY NOTATION

1.14 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1.

IDENTIFIED LEAKAGE

1.15 IDENTIFIED LEAKAGE shall be:

a. Leakage (except CONTROLLED LEAKAGE) into closed systems, such as pump seal or valve packing leaks that are captured and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of Leakage Detection Systems or not to be PRESSURE BOUNDARY LEAKAGE, or

c. Reactor Coolant System leakage through a steam generator to the Secondary Coolant System.

MASTER RELAY TEST

1.16 A MASTER RELAY TEST shall be the energization of each master relay and verification of OPERABILITY of each relay. The MASTER RELAY TEST shall include a continuity check of each associated slave relay.

MEMBER(S) OF THE PUBLIC

1.17 MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the licensee, its contractors or vendors and persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational, or other purposes not associated with the plant.

Amendment No.

OFFSITE DOSE CALCULATION MANUAL

1.18 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Sections 6.8.4.e and f, and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Radioactive Effluent Release Reports required by Specification 6.9.1.6 and 6.9.1.7.

OPERABLE - OPERABILITY

1.19 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s), and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its function(s) are also capable of performing their related support function(s).

OPERATING LIMITS REPORT

1.19.a The OPERATING LIMITS REPORT is the unit-specific document that provides operating limits for the current operating reload cycle. These cycle-specific operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.9. Plant Operation within these operating limits is addressed in individual specifications.

OPERATIONAL MODE - MODE

1.20 An OPERATIONAL MODE (i.e., MODE) shall correspond to any one inclusive combination of core reactivity condition, power level, and average reactor coolant temperature specified in Table 1.2.

PHYSICS TESTS

1.21 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the core and related instrumentation: (1) described in Chapter 14.0 of the FSAR, (2) authorized under the provisions of 10 CFR 50.59, or (3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

1.22 PRESSURE BOUNDARY LEAKAGE shall be leakage (except steam generator tube leakage) through a nonisolable fault in a Reactor Coolant System component body, pipe wall, or vessel wall.

Pa

1.20.a Pa shall be the maximum calculated primary containment pressure (44.4 psig) for the design basis loss of coolant accident.

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

Without primary CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

a. At least once per 31 days by verifying that all penetrations* not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions, except as provided in Table 3.6-1 of Specification 3.6.3~~.~~ or for containment isolation valves that are open under administrative controls;

b. By verifying that each containment air lock is in compliance with the requirements of Specification 3.6.1.3; and

c. ~~After each closing of each penetration subject to Type B testing, except the containment air locks, if opened following a Type A or B test, by leak rate testing the seal with gas at a pressure not less than P_a, 44.4 psig, and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Specification 4.6.1.2d. for all other Type B and C penetrations, the combined leakage rate is less than 0.60 L_a.~~

By performing containment leakage testing in accordance with Regulatory Guide 1.163, September 1995, and 10CFR50, Appendix J, Option B.

*Except valves, blind flanges, and deactivated automatic valves which are located inside the containment and are locked, sealed or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except that such verification need not be performed more often than once per 92 days.

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited to:

a. An overall integrated leakage rate of~~:~~ less than or equal to La at Pa

~~1) Less than or equal to L_a, 0.10% by weight of the containment air per 24 hours at P_a, 44.4 psig, or~~

~~2) Less than or equal to L_t, 0.07% by weight of the containment air per 24 hours for Unit 1 (0.07% by weight of the containment air per 24 hours for Unit 2) at P_t, 22.2 psig.~~

b. A combined leakage rate of less than 0.60 L_a for all penetrations and valves subject to Type B and C tests, when pressurized to P_a.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With either the measured overall integrated containment leakage rate exceeding 0.75 L_a ~~or 0.75 L_t, as applicable~~, or the measured combined leakage rate for all penetrations and valves subject to Types B and C tests exceeding 0.60 L_a, restore the overall integrated leakage rate to less than 0.75 L_a ~~or less than 0.75 L_t, as applicable,~~ and the combined leakage rate for all penetrations subject to Type B and C tests to less than 0.60 L_a prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The containment leakage rates shall be demonstrated ~~at the following test schedule and shall be determined in conformance with the criteria specified in Appendix J of 10 CFR Part 50 using the methods and provisions of ANSI N45.4-1972:~~ in accordance with Regulatory Guide 1.163, September 1995, and 10 CFR 50 Appendix J, Option B.

a. Type A (Overall Integrated Containment Leakage Rate) testing shall be conducted in accordance with ~~the requirements specified in Appendix J to 10 CFR 50, as modified by approved exemptions;~~ Regulatory Guide 1.163, September 1995, and 10CFR 50, Appendix J, Option B.

SURVEILLANCE REQUIREMENTS (Continued)

b. ~~If any periodic Type A test fails to meet either 0.75 L_a or 0.75 L_t, the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet either 0.75 L_a or 0.75 L_t, a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet either 0.75 L_a or 0.75 L_t;~~

c. The accuracy of each Type A test shall be verified by a supplemental test ~~which:~~ conducted in accordance with Regulatory Guide 1.163, September 1995 and 10 CFR 50, Appendix J, Option B.

   ~~1) Confirms the accuracy of the test by verifying that the supplemental test result, L_c, is in accordance with the appropriate following equation:~~

   $$|L_c - (L_{am} + L_o)| \le 0.25\ L_a \text{ or } |L_c - (L_{tm} + L_o)| \le 0.25\ L_t$$

   ~~where L_{am} or L_{tm} is the measured Type A test leakage and L_o is the superimposed leak;~~

   ~~2) Has a duration sufficient to establish accurately the change in leakage rate between the Type A test and the supplemental test; and~~

   ~~3) Requires that the rate at which gas is injected into the containment or bled from the containment during the supplemental test is between 0.75 L_a and 1.25 L_a, or 0.75 L_t and 1.25 L_t.~~

d. Type B and C tests shall be conducted ~~with gas at a pressure not less than P_a, 44.4 psig, at intervals no greater than 24 months except for tests involving:~~ in accordance with Regulatory Guide 1.163, September 1995, and 10 CFR 50, Appendix J, Option B.

   ~~1) Air locks, and~~

   ~~2) Purge supply and exhaust isolation valves with resilient material seals.~~

e. Air locks shall be tested and demonstrated OPERABLE by the requirements of Specification 4.6.1.3;

f. Purge supply and exhaust isolation valves with resilient material seals shall be tested and demonstrated OPERABLE by the requirements of Specification 4.6.1.7.3 or 4.6.1.7.4, as applicable; and

g. The provisions of Specification 4.0.2 are not applicable.

The reporting requirements and Frequency of Type A tests shall be in accordance with Reguatory Guide 1.163, September 1995, and 10 CFR 50 Appendix J, Option B.

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE with:

a. Both doors closed except when the air lock is being used for normal transit entry and exits through the containment, then at least one air lock door shall be closed, and

b. An overall air lock leakage rate of less than or equal to 0.05 L_a at P_a, ~~44.4 psig.~~

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With one containment air lock door inoperable:

1. Maintain at least the OPERABLE air lock door closed and either restore the inoperable air lock door to OPERABLE status within 24 hours or lock the OPERABLE air lock door closed;

2. Operation may then continue until performance of the next required overall air lock leakage test provided that the OPERABLE air lock door is verified to be locked closed at least once per 31 days;

3. Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; and

4. The provisions of Specification 3.0.4 are not applicable.

b. With the containment air lock inoperable, except as the result of an inoperable air lock door, maintain at least one air lock door closed; restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

Insert A

a. ~~Within 72 hours following each closing, except when the air lock is being used for multiple entries, then at least once per 72 hours, by~~

(1) Verifying that the door seal leakage is less than 0.0024La (1.11 SCFH) when the volume between the door seals is pressurized to greater than or equal to 3 psig by means of a permanently installed continuous pressurization and leakage monitoring system, or

(2) Verifying that the door seal leakage is less than 0.01La (4.63 SCFH) as determined by precision flow measurements when measured for at least 30 seconds with the volume between the seals at a constant pressure of greater than or equal to 10 psig;

b. By conducting overall air lock leakage ~~tests at not less than P_a, 44.4 psig, and verifying the overall air lock leakage rate is within its limit:~~ Insert B

~~1) At least once per 6 months,* and~~

~~2) Prior to establishing CONTAINMENT INTEGRITY when maintenance has been performed on the air lock that could affect the air lock sealing capability.**~~

c. At least once per 6 months by verifying that only one door in each air lock can be opened at a time.

~~d. At least once per 6 months by verifying that the seal leakage is less than 0.01La (4.63 SCFH) as determined by precision flow measurements when measured for at least 30 seconds with the volume between the seals at a constant pressure of greater than or equal to 10 psig;~~

Insert C

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

~~**This represents an exemption to Appendix J of 10 CFR Part 50, Paragraph III D.2(b)(ii).~~

Insert A

a. By conducting airlock seal leakage tests following each closing in accordance with Regulatory Guide 1.163, September 1995, and 10 CFR 50, Appendix J, Option B, by

Insert B

in accordance with Regulatory Guide 1.163, September 1995, and 10 CFR 50, Appendix J, Option B.

Insert C

d. By verifying that the airlock seal leakage tests is less than 0.01 La (4.63 SCFH) as determined by precision flow measurements when measured for at least 30 seconds with the volume between the seals at a constant pressure of greater than or equal to 10 psig in accordance with Regulatory Guide 1.163, September 1995, and 10 CFR 50, Appendix J, Option B.;

SURVEILLANCE REQUIREMENTS

4.6.1.7.1 Each 48-inch containment purge supply and exhaust isolation valve(s) shall be verified closed and power removed at least once per 31 days.

4.6.1.7.2 Each 8-inch containment purge supply and exhaust isolation valve shall be verified to be positioned in accordance with Specification 3.6.1.7b at least once per 31 days.

Leakage testing shall be conducted on

4.6.1.7.3 ~~At least once per 6 months on a STAGGERED TEST BASIS~~, the inboard and outboard valves with resilient material seals in each closed 48-inch containment purge supply and exhaust penetration ~~shall be demonstrated OPERABLE~~ by verifying that the measured leakage rate is less than 0.05 L_a ~~when pressurized to at least P_a, 44.4 psig~~. on a STAGGERED TEST BASIS in accordance with Regulatory Guide 1.163, September 1995, and 10 CFR 50, Appendix J, Option B.

4.6.1.7.4, ~~At least once per 3 months~~, each 8-inch containment purge supply and exhaust isolation valve with resilient material seals ~~shall be demonstrated OPERABLE~~ by verifying that the measured leakage rate is less than 0.01 L_a ~~when pressurized to at least P_a, 44.4 psig.~~ in accordance with Regulatory Guide 1.163, September 1995, and 10 CFR 50, Appendix J, Option B.

Leakage Testing shall be conducted on

3/4.6 CONTAINMENT SYSTEMS

BASES

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 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 safety analyses. This restriction, in conjunction with the leakage rate limitation, will limit the SITE BOUNDARY radiation doses to within the dose guideline values of 10 CFR Part 100 during accident conditions.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on 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, 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 test to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates are consistent with the requirements of Appendix J of 10 CFR Part 50, Option B, Regulatory Guide 1.163, September 1995, Nuclear Energy Institute document NEI 94-01, and ANSI/ANS-56.8-1994.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and containment leak rate. Surveillance testing of the air lock seals provides assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests. ~~The use of precision flow measurements of Specification 4.6.1.3.a(2) must be used whenever the continuous monitoring capability in the control room is lost~~.

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that: (1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the outside atmosphere of 0.1 psig, and (2) the containment peak pressure does not exceed the design pressure of 50 psig during steam line break conditions.

The maximum increase in peak pressure expected to be obtained from a cold leg double-ended break event is 44.4 psig. The limit of 1.0 psig for initial positive containment pressure will limit the total pressure to 44.4 psig, which is higher than the ~~FSAR Chapter~~ UFSAR Chapter 15 accident analysis calculated peak pressure assuming a limit of 0.3 psig for initial positive containment pressure, but is considerably less than the design pressure of 50 psig.

ATTACHMENT C

EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATIONS

Commonwealth Edison Company (ComEd) has evaluated this proposed amendment and determined that it involves no significant hazards considerations. According to Title 10, Code of Federal Regulations, Part 50, Section 92, Paragraph c [10 CFR 50.92(c)], a proposed amendment to an operating license involves no significant hazards if operation of the facility in accordance with the proposed amendment would not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated; or

2. Create the possibility of a new or different kind of accident from any accident previously evaluated; or

3. Involve a significant reduction in a margin of safety.

ComEd proposes to revise Byron Nuclear Power Station, Units 1 and 2 (Byron), and Braidwood Nuclear Power Station, Units 1 and 2 (Braidwood) Technical Specification (TS) Section 3/4.6.1, "Primary Containment," and the associated Bases to reflect recent changes to Appendix J to 10 CFR 50, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors." The proposed revisions include:

1. Adding TS Definitions for the maximum allowable primary containment leakage rate (L_a) and for the maximum calculated primary containment pressure (P_a). The redundant definitions throughout TS 3/4.6.1 are deleted,

2. Adding statements throughout TS 3/4.6.1 that leak rate testing is performed in accordance with 10 CFR 50, Appendix J, Option B, and Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program," September 1995, and its referenced documents,

3. Deleting TS requirements that are taken verbatim from 10 CFR 50, Appendix J. The specific requirements will be placed in the containment leakage rate test program in accordance with 10 CFR 50, Appendix J, Option B, and RG 1.163 and its referenced documents, and

4. Clarifying Technical Specification Surveillance Requirement (TSSR) 4.6.1.1.a for consistency with NUREG-1431, "Standard Technical Specifications for Westinghouse Plants," Revision 1.

A. **The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.**

10 CFR 50, Appendix J, has been amended to include provisions regarding performance-based leakage testing requirements (Option B). Option B allows plants with satisfactory Integrated Leak Rate Testing (ILRT) performance history to reduce the Type A testing frequency from three tests in ten years to one test in ten years. For Type B and Type C tests, Option B allows plants to reduce testing frequency based on the leak rate test history of each component. In addition, Option B establishes controls to ensure continued satisfactory performance of the affected penetrations during the extended testing interval. To be consistent with the requirements of Option B to 10 CFR 50, Appendix J, ComEd proposes to include appropriate changes to the TSs that incorporate the necessary revisions.

Some of the proposed changes represent minor curtailments to current TS requirements, but are based on the requirements specified by Option B to 10 CFR 50, Appendix J. Any such changes are consistent with the current plant safety analyses and have been determined to represent sufficient requirements for the assurance of the reliability of equipment assumed to operate in the safety analyses, or provide continued assurance that specified parameters associated with containment integrity remain within their acceptance limits. The other proposed changes maintain consistency with those requirements specified by Option B to 10 CFR 50, Appendix J and are consistent with the current plant safety analyses. Implementation of these changes will provide continued assurance that specified parameters associated with containment integrity will remain within their acceptance limits, and as such, will not significantly increase the probability or consequences of a previously evaluated accident.

The associated systems affecting the leak rate integrity are not assumed in any safety analyses to initiate any accident sequence; therefore, the probability of occurrence of any accident previously evaluated is not increased. In addition, the proposed changes to the limiting conditions for operation and surveillance requirements for such systems are consistent with the current 10 CFR 50, Appendix J, requirements. The proposed changes maintain an equivalent level of reliability and availability for all affected systems.

Maintaining allowable leakage within the analyzed limit assumed for the accident analyses does not adversely affect either the onsite or offsite dose consequences. Furthermore, containment leakage is not an accident initiator. As such, there is no adverse impact on the probability of accident initiators. Thus, there is no significant increase in the probability or occurrence of any previously analyzed accident, or increase the consequences of any previously analyzed accident.

B. The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

Option B of 10 CFR 50, Appendix J, specifies, in part, that a Type A test may be conducted at a periodic interval based on the performance of the overall containment system. Type A tests measure both the containment system overall integrated leakage rate at the containment pressure boundary and system alignments assumed during a large break loss-of-coolant accident (LOCA), and demonstrate the capability of the primary containment to withstand an internal pressure load. The acceptable leakage rates are specified in the TSs. For Type B and C tests, intervals are proposed for establishment based on the performance history of each component. Acceptance criteria for each component are based upon demonstration that the leakage rates at design basis pressure conditions for applicable penetrations are within the limits specified in the TSs.

The proposed changes reflect the requirements specified in the amended 10 CFR 50, Appendix J, and are consistent with the current plant safety analyses. Some minor curtailments of current TS requirements are based on generic guidance or similarly approved provisions for other plants. These changes do not involve revisions to the design of the plant. Some of the changes may involve revision in the testing of components at the plant; however, these are in accordance with the current plant safety analyses and provide for appropriate testing or surveillance that is consistent with Option B to 10 CFR 50, Appendix J. The proposed changes will not introduce new failure mechanisms beyond those already considered in the current plant safety analyses.

No new modes of operation are introduced by the proposed changes. Surveillance requirements are changed to reflect corresponding changes associated with Option B to 10 CFR 50, Appendix J. The proposed changes maintain at least the present level of operability of any such system that affects plant containment integrity. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

The associated systems that affect plant leak rate integrity related to the proposed amendment are not assumed to initiate any accident sequence. In addition, the proposed surveillance requirements for any such affected systems are consistent with the current requirements specified within the TSs and are consistent with the requirements of Option B to 10 CFR 50, Appendix J. The proposed surveillance requirements maintain an equivalent level of reliability and availability of all affected systems and, therefore, do not affect the consequences of any previously evaluated accident. As such, the probability of systems associated with leak rate test integrity failing to perform their intended function is unaffected by the proposed limiting conditions for operation and surveillance requirements.

C. **The proposed changes do not involve a significant reduction in a margin of safety.**

The provisions specified in Option B to 10 CFR 50 Appendix J, allows changes to Type A, B, and C test intervals based upon the performance of past leak rate tests. The effect of extending containment leak rate test intervals is a corresponding increase in the likelihood of containment leakage. The degree to which intervals can be extended has a direct impact on the potential effect on existing plant safety margins and the public health and safety that can occur due to an increased likelihood of containment leakage.

Changing Type A, B, and C test intervals from those currently provided in the TS to those provided for in 10 CFR 50, Appendix J, Option B, slightly increases the risk associated with Type A, B, and C specific accident sequences. Historical data suggest that increasing the Type C test interval can slightly increase the associated risk; however, this is compensated by the corresponding risk reduction benefits associated with reduction in component cycling, stress, and wear associated with increased test intervals. In addition, when considering the total integrated risk, which includes all analyzed accident sequences, the additional risk associated with increasing test intervals is negligible.

The proposed changes are consistent with those provisions specified in Option B of 10 CFR 50, Appendix J, and are consistent with current plant safety analyses. In addition, these proposed changes do not involve revisions to the design of the plant. As such, the proposed individual changes will maintain the same level of reliability of the equipment associated with containment integrity, assumed to operate in the plant safety analysis, or provide continued assurance that specified parameters affecting plant leak rate integrity, will remain within their acceptance limits. Therefore, the proposed changes provide continued assurance of the leakage integrity of the containment without adversely affecting the public health and safety and, as such, will not significantly reduce existing plant safety margins.

The proposed changes are based on United States Nuclear Regulatory Commission (USNRC) accepted provisions and maintain necessary levels of system or component reliability affecting plant containment integrity. The performance-based approach to leakage rate testing concludes that the impact on public health and safety due to revised testing intervals is negligible. The proposed changes will not reduce the availability of systems associated with containment integrity when they are required to mitigate accident conditions; therefore, the proposed changes do not involve a significant reduction in the margin of safety.

Guidance for the application of standards to license change requests for determination of the existence of significant hazards considerations has been provided in "Final Procedures and Standards on No Significant Hazards Considerations," Final Rule, 51 FR 7744. This document provides examples of amendments which are and are not considered likely to involve significant hazards considerations. The adoption of the requirements for the revised 10 CFR 50, Appendix J, most closely fits the example of a change which may either result in some increase to the probability or consequences of a previously analyzed accident or may reduce in some way a safety margin. However, the proposed amendment results in a change which is clearly within all acceptable criteria with respect to the system or component specified in NUREG-0800, Standard Review Plan, Section 6.2.6, Containment Leakage Testing. The proposed changes retain the current specification leak rate limits and acceptance criteria, thus preserving the safety margin, and will not significantly increase the consequences of an accident.

This proposed amendment does not involve a significant relaxation of the criteria used to establish safety limits, a significant relaxation of the bases for the limiting safety system settings, or a significant relaxation of the bases for the limiting conditions for operations. Therefore, based on the guidance provided in the Federal Register and the criteria established in 10 CFR 50.92(c), Commonwealth Edison has concluded that these changes involve no significant hazards considerations.

ATTACHMENT D

ENVIRONMENTAL ASSESSMENT

Commonwealth Edison Company (ComEd) has evaluated the proposed amendment against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with Title 10, Code of Federal Regulations, Part 50, Section 51 (10 CFR 51.21). ComEd has determined that the proposed change meets the criteria for a categorical exclusion set forth in 10 CFR 51.22(c)(9). This determination is based on the fact that this change is being proposed as an amendment to a license issued pursuant to 10 CFR 50 that changes a requirement with respect to installation or use of a facility component located within a restricted area, as defined in 10 CFR 20, or that changes an inspection or a surveillance requirement, and the amendment meets the following specific criteria:

(i) the amendment involves no significant hazards considerations,

As demonstrated in Attachment C, this proposed amendment does not involve any significant hazards considerations.

(ii) there is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite, and

As documented in Attachment C, there will be no change in the types or significant increase in the amounts of any effluents released offsite.

(iii) there is no significant increase in individual or cumulative occupational radiation exposure.

The proposed change will not result in changes in the operation or configuration of the facility. There will be no change in the level of controls or methodology used for processing of radioactive effluents or handling of solid radioactive waste; nor will the proposal result in any change in the normal radiation levels within the plant. Therefore there will be no increase in individual or cumulative occupational radiation exposure resulting from this change.

ATTACHMENT E

IMPLEMENTATION PLAN FOR 10 CFR 50, APPENDIX J, OPTION B

Byron and Braidwood will incorporate the performance oriented and risk-based approaches included in the following documents into their containment leakage rate testing programs:

- Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," Option B,

- Regulatory Guide (RG) 1.163, September 1995, "Performance-Based Containment Leak-Test Program,"

- Nuclear Energy Institute (NEI) 94-01, Revision 0, "Nuclear Energy Institute Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," and

- ANSI/ANS-56.8-1994, "American National Standard for Containment System Leakage Testing Requirements."

10 CFR 50, Appendix J, Option B provides a performance based option for Type A, B, and C leakage rate testing of primary containment. This option improves the focus of the regulation by eliminating prescriptive requirements that have been determined to be marginal to safety. The new rule allows for test intervals to be based on system and component performance and provides for greater flexibility for cost effective implementation methods for regulatory safety objectives.

ComEd has formed an Appendix J Implementation Task Force to implement and interpret the new 10 CFR 50, Appendix J in a consistent manner throughout ComEd. Each ComEd nuclear station (including Byron and Braidwood) is represented in the group. The task force will provide generic guidelines for all ComEd nuclear stations for the implementation of 10 CFR 50, Appendix J, Option B.

COMPONENT LEAKAGE LIMITS

Byron and Braidwood will use the administrative limits set by the ComEd Appendix J Implementation Task Force for each component requiring Types B and C leakage rate testing. To determine whether an as-found local leak rate test (LLRT) passed or failed, a component's measured leakage is compared against its administrative limit. The task force carefully evaluated the administrative leak rate limits to determine the proper limits, which are extremely important under the performance-based rule. These new administrative limits will be used to determine whether future or previous tests passed or failed. Thus, the limits chosen will affect each component's Type B or C testing frequency.

Two limits will be specified for each component, a warning limit and an alarm limit. When the component's leakage rate is above the warning limit and below the alarm limit, then the component should be evaluated for repair. This is not counted as a performance failure. When the component's leakage rate is above the alarm limit, then the component must be repaired, except as noted below. This is counted as a performance failure.

Although administrative limits are used to maintain the containment in good condition, it should be noted that the sum of the as-left maximum pathway leakage rates for all Appendix J barriers must be less than 0.6 L_a per plant Technical Specifications, where L_a is defined as the maximum allowable primary containment leakage rate. In the past, there have been instances where the leakage from one or more components has exceeded the alarm limits. To bring the leakage rate below the limit prior to start-up would have been very difficult and/or costly. For those special cases, a safety evaluation was performed . If this evaluation concluded that there was no significant safety impact, then the component(s) was(were) allowed to continue to leak in excess of the individual valve leakage limit until it could be repaired, provided that the Technical Specification limit of 0.6 L_a was not exceeded. It must be noted though, that the test was still considered to be a failure in spite of the safety evaluation. Byron and Braidwood reserve the option to continue to use this provision only on a critical, as needed basis.

BUILDING PERFORMANCE BASELINES/ESTABLISHING TEST FREQUENCIES

Type A Test

In accordance with the new requirements associated with 10 CFR 50, Appendix J, Option B, Type A testing shall be performed during a period of reactor shutdown at a frequency of at least once per 10 years based on acceptable performance history. Acceptable performance history is defined as completion of two consecutive periodic Type A tests where calculated as-found performance leakage rate was less than 1.0 La. Elapsed time between the first and last tests in a series of consecutive satisfactory tests used to determine performance shall be the normal Byron and Braidwood refuel interval. NEI 94-01 states that this interval shall be at least 24 months, however, the normal Byron and Braidwood refuel interval of 18 months is a more appropriate minimum interval between Type A tests.

The new rules allow for reviewing past performance history with several options to determine if past Type A tests were satisfactory:

a. As-Found Type A test results can be compared to 1.0 L_a rather than the previous 0.75 L_a criteria.

b. Leakage savings (repairs/adjustments) from Type B and C testable pathways which were added as penalties to the As-Found Type A test can be subtracted when reviewing previous Type A test results.

c. The Type A test upper confidence limit from previous Type A tests may be recalculated using the Mass Point Methodology described in ANS 56.8-1994.

Byron has reviewed Type A test results as compared to the current requirements and criteria to establish a test frequency for the primary containment integrated leak rate test (ILRT). In reviewing Byron Type A history, it has been determined that the two most recent as-found Type A tests for Unit 1 have been below the 1.0 L_a criteria. Therefore, Byron, Unit 1, will implement the 10 year Type A test frequency based on the criteria set forth in the new rule during the next refuel outage, Byron, Unit 1, Cycle 7, Refuel Outage (B1R07). Byron, Unit 2, and Braidwood data will be evaluated to determine applicable future test frequency requirements, based on the Type A test performance history. Braidwood is pursuing resolution of comments on previous ILRTs with the United States Nuclear Regulatory Commission (USNRC). If this effort is successful, Braidwood may implement the 10 year Type A test frequency of Option B to Appendix J.

Type B and C Tests

Byron and Braidwood will formulate administrative procedures for documenting Type B and C testing performance. A performance evaluation will be used to ensure that consistent criteria were applied to establish component baseline performance and their subsequent testing frequencies.

Byron and Braidwood have developed a computer database to compile all the required leak rate historical data to be used in the evaluation process. This database will continue to be updated with the most current as-found leak rate data acquired during the most recent refuel outages. The performance history of each component will be evaluated against the administrative limit to rate component performance over the last three refuel outages. In addition to a performance history evaluation, considerations such as service life, environment, design, system application, special service conditions, and safety impact/risk from failure will be reviewed and evaluated, and will be used to determine test frequency.

TECHNICAL CRITERIA & TESTING METHODOLOGY INTERPRETATION

The containment leakage rate testing program will follow the guidance in RG 1.163, NEI 94-01, ANSI/ANS-56.8-1994, and 10 CFR 50 Appendix J, Option B. The administrative procedure(s) for the containment leakage rate testing program will follow the requirements and contain the performance criteria for the Types A, B, and C testing. The administrative procedure(s) will also contain the description of the record keeping and methodology to establish test intervals for equipment and components in the containment leakage rate testing program. The equipment and component test procedures will contain information on the proper techniques and methods for performing the Type A, B, and C tests.