**ATTACHMENT B-1**
**MARKED UP PAGES FOR**

**PROPOSED CHANGES TO APPENDIX A**
**TECHNICAL SPECIFICATIONS OF**
**FACILITY OPERATING LICENSES**
**NPF-37 and NPF-66,**

BYRON STATION UNITS 1 & 2

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

as modified by the approved exceptions,

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*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 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 $L_a$ at $P_a$.

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 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 $L_a$ 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 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 Regulatory Guide 1.163, September 1995, and 10 CFR 50, Appendix J, Option B.

as modified by the approved exceptions,

SURVEILLANCE REQUIREMENTS (Continued)

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

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

d. Type B and C tests shall be conducted in accordance with Regulatory Guide 1.163, September 1995, and 10 CFR 50, Appendix J, Option B.

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.

## 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 $L_a$ 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~~ is 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, insert bases text

### 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.

### 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 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.

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except as modified for Unit 2. Unit 2 is exempt from the requirement in NEI 94-01 to perform two consecutive successful Type A tests prior to extending the testing interval until September 10, 2003. Subsequent Type A test intervals for Unit 2 will be determined based on test results, in accordance with NEI 94-01.

**ATTACHMENT B-2**

**MARKED UP PAGES FOR
PROPOSED CHANGES TO APPENDIX A**

**TECHNICAL SPECIFICATIONS OF
FACILITY OPERATING LICENSES
NPF-72 and NPF-77**

BRAIDWOOD STATION UNITS 1 & 2

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## 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. By performing containment leakage testing in accordance with Regulatory Guide 1.163, September 1995, as modified by the approved exceptions, and 10 CFR 50, Appendix J, Option B.

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*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 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 $L_a$ at $P_a$.

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 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 $L_a$ 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 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 Regulatory Guide 1.163, September 1995, and 10 CFR 50, Appendix J, Option B.

as modified by the approved exceptions,

SURVEILLANCE REQUIREMENTS (Continued)

b. 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.
as modified by the approved exceptions

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

d. Type B and C tests shall be conducted in accordance with Regulatory Guide 1.163, September 1995, and 10 CFR 50, Appendix J, Option B.

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.

## 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 $L_a$ 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 is ~~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~~.~~, [insert bases text]

### 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.

### 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 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.

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except as modified for Unit 2. Unit 2 is exempt from the requirement in NEI 94-01 to perform two consecutive successful Type A tests prior to extending the testing interval until November 9, 2004. Subsequent Type A test intervals for Unit 2 will be determined based on test results, in accordance with NEI 94-01.

## ATTACHMENT C
## EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATION

Commonwealth Edison Company (ComEd) proposes to revise Technical Specifications (TS) Surveillance Requirements 4.6.1.1.c., 4.6.1.2.a, 4.6.1.2.b and the Bases to allow a one-time exception to a requirement in 10 CFR 50, Appendix J, Option B. The proposed change would allow the interval of Type A testing of the Byron Unit 2 and Braidwood Unit 2 containments to be determined based on one successful Type A test rather than two consecutive Type A tests. The test will be extended beyond the interval allowed in the Nuclear Energy Institute (NEI) document NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J," which is endorsed by Regulatory Guide 1.163, " Performance Based Containment Leak Test Program "to November 9, 2004 for Braidwood and September 10, 2003 for Byron.

1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   Performance of Type A tests at a different interval does not involve a change to any structures, systems, or components, does not affect reactor operations, is not an accident initiator, and does not change any existing safety analysis previously evaluated in the UFSAR. Therefore, there is no significant increase in the probability of an accident previously evaluated.

   Several tables of UFSAR Chapter 15,"Accident Analyses," provide containment leak rate values used in assessing the consequences of accidents discussed in this chapter. Although decreasing the test frequency can increase the probability that an increase in containment leakage could go undetected for an extended period of time, the risk resulting from this proposed change is inconsequential as documented in NUREG-1493, "Performance-Based Containment Leakage Test Program". This document indicated that given the insensitivity of reactor risk to containment leakage rate and a small fraction of leakage paths are detected solely by Type A testing, increasing the interval between integrated leak rate tests is possible with minimal impact on ublic risk. Further, industry experience presented in this document indicated that Type A testing has had insignificant impact on uncertainties involved with containment leak rates.

   Based on risk information presented in NUREG-1493, the proposed change does not increase the probability or consequences of an accident previously evaluated.

2. The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change does not alter the plant design, systems, components, or reactor operations, only the frequency of test performance. New conditions or parameters that contribute to the initiation of accidents would not be created as a result of this proposed change. The change does not involve new equipment and existing equipment does not have to be operated in a different manner, therefore there are no new failure modes to consider.

Changing test intervals as shown in NUREG 1493 has no impact on, nor contributes to the possibility of a new or different kind of accident as evaluated in the UFSAR. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change does not involve a significant reduction in a margin of safety.

With the exception of the test frequency, the actual tests will not change. Quantitative risk studies documented in NUREG-1493 regarding extended testing intervals demonstrated that there was minimal impact on the public health and safety. Reducing the frequency, as stated in the NUREG resulted in an "imperceptible" increase in risk to public safety. Further, a table in this NUREG regarding risk impacts due to a reduction in testing frequency suggested that there was also minimal difference in risk to the public safety when the test frequency was relaxed.

The proposed change will not reduce the availability of systems and components associated with containment integrity that would be required to mitigate accident conditions nor are any containment leakage rates, parameters or accident assumptions affected by the proposed change.

The proposed change does not involve a significant reduction in a margin of safety, based on the above information.

Based on the above evaluation, ComEd has concluded that these changes involve no significant hazards considerations.

## ATTACHMENT D

## ENVIRONMENTAL ASSESSMENT

ComEd has evaluated this proposed operating license amendment against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21. ComEd has determined that this proposed license amendment meets the criteria for a categorical exclusion set forth in 10 CFR 51.22(c)(9) and as such, has determined that no irreversible consequences exist in accordance with 10 CFR 50.92(b). 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 the 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 consideration.

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

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

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

The proposed amendment 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 proposed amendment 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.