ATTACHMENT B-1

MARKED UP PAGES FOR
PROPOSED CHANGES TO APPENDIX A
TECHNICAL SPECIFICATIONS OF
FACILITY OPERATING LICENSES
NPF-37, NPF-66

BYRON STATION UNITS 1 & 2
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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.

P_a

1.20.a P_a shall be the maximum calculated primary containment pressure ~~(44.4 psig)~~ (44.4 psig*, 47.8 psig**) for the design basis loss of coolant accident.

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.

* Applicable to Unit 1 through cycle 8 and to Unit 2.
** Not applicable to Unit 2. Applicable to Unit 1 after cycle 8.

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

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

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

d. By verifying that the airlock seal leakage tests are 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.

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~~.

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~~.

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 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.

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. defined as P_a

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~~, P_a 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.

3/4.6.1.5 AIR TEMPERATURE

The limitations on containment average air temperature ensure that the overall containment average air temperature does not exceed the initial temperature condition assumed in the accident analysis for a steam line break accident. Measurements shall be made at all of the listed running fan locations, whether by fixed or portable instruments, to determine the average air temperature.

3/4.6.1.6 CONTAINMENT VESSEL STRUCTURAL INTEGRITY

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This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the containment will withstand the maximum pressure of ~~44.4 psig~~ in the event of a cold leg double-ended break accident. The measurement of containment tendon lift-off force, the tensile tests of the tendon wires or strands, the visual examination of tendons, anchorages and exposed interior and exterior surfaces of the containment, and the Type A leakage test are sufficient to demonstrate this capability.

The Surveillance Requirements for demonstrating the containment's structural integrity are in compliance with the recommendations of proposed Rev. 3 to Regulatory Guide 1.35, "Inservice Surveillance of Ungrouted Tendons in Prestressed Concrete Containment Structures," April 1979 and proposed Regulatory Guide 1.35.1, "Determining Prestressing Forces for Inspection of Prestressed Concrete Containments," April 1979.

The required Special Reports from any engineering evaluation of containment abnormalties shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedure, the tolerances on cracking, the results of the engineering evaluation and the corrective actions taken.

3/4.6.1.7 CONTAINMENT PURGE VENTILATION SYSTEM

The 48-inch containment purge supply and exhaust isolation valves are required to be sealed closed (power removed) during plant operations since these valves have not been demonstrated capable of closing during a LOCA or steam line break accident. Maintaining these valves sealed closed during plant operation ensures that excessive quantities of radioactive material will not be released via the Containment Purge System. To provide assurance that the 48-inch containment valves cannot be inadvertently opened, the valves are sealed closed in accordance with Standard Review Plan 6.2.4 which includes mechanical devices to seal or lock the valve closed, or prevents power from being supplied to the valve operator.

The use of the containment purge lines is restricted to the 8-inch purge supply and exhaust isolation valves since, unlike the 48-inch valves, the 8-inch valves are capable of closing during a LOCA or steam line break accident. Therefore, the SITE BOUNDARY dose guideline values of 10 CFR Part 100 would not

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The core shall contain 193 fuel assemblies with each fuel assembly containing 264 fuel rods clad with Zircaloy-4 or ZIRLO, except that limited substitution of fuel rods by filler rods consisting of Zircaloy-4, ZIRLO, or stainless steel or by vacancies may be made if justified by a cycle specific reload analysis. Each fuel rod shall have a nominal active fuel length of 144 inches. The initial core loading shall have a maximum enrichment of less than 3.20 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading or previous cycle loading. The enrichment of any reload fuel design shall be determined to be acceptable for storage in either the spent fuel pool or the new fuel vault. Such acceptance criteria shall be based on the results of the CRITICALITY ANALYSIS OF BYRON AND BRAIDWOOD STATION FUEL STORAGE RACKS.

CONTROL ROD ASSEMBLIES

5.3.2 The core shall contain 53 full-length and no part-length control rod assemblies. The full-length control rod assemblies shall contain a nominal 142 inches of absorber material. All control rods shall be hafnium, silver-indium-cadmium, or a mixture of both types. All control rods shall be clad with stainless steel tubing.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The Reactor Coolant System is designed and shall be maintained:

a. In accordance with the Code requirements specified in Section 5.2 of the UFSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

5.4.2 The total water and steam volume of the Reactor Coolant System is 12,257 cubic feet at a nominal T_{avg} of 588.4°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-1.

An additional 1,251 ft³ of volume is added to the Unit 1 total RCS volume as a result of the four replacement steam generators installed after cycle 8

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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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 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.

P_a

1.20.a P_a shall be the maximum calculated primary containment pressure ~~(44.4 psig)~~ (44.4 psig*, 47.8 psig**) for the design basis loss of coolant accident.

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.

*Applicable to Unit 1 through cycle 7 and to Unit 2.
**Not applicable to Unit 2. Applicable to Unit 1 after cycle 7.

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

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

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b. By conducting overall air lock leakage tests in accordance with Regulatory Guide 1.163, September 1995, and 10 CFR 50, Appendix J, Option B.

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

d. By verifying that the airlock seal leakage tests are 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.

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~~.

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~~.

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 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.

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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.

defined as P_a

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.

3/4.6.1.5 AIR TEMPERATURE

The limitations on containment average air temperature ensure that the overall containment average air temperature does not exceed the initial temperature condition assumed in the accident analysis for a steam line break accident. Measurements shall be made at all of the listed running fan locations, whether by fixed or portable instruments, to determine the average air temperature.

3/4.6.1.6 CONTAINMENT VESSEL STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the containment will withstand the maximum pressure of ~~44.4 psig~~ in the event of a cold leg double-ended break accident. The measurement of containment tendon lift-off force, the tensile tests of the tendon wires or strands, the visual examination of tendons, anchorages and exposed interior and exterior surfaces of the containment, and the Type A leakage test are sufficient to demonstrate this capability.

The Surveillance Requirements for demonstrating the containment's structural integrity are in compliance with the recommendations of proposed Rev. 3 to Regulatory Guide 1.35, "Inservice Surveillance of Ungrouted Tendons in Prestressed Concrete Containment Structures," April 1979 and proposed Regulatory Guide 1.35.1, "Determining Prestressing Forces for Inspection of Prestressed Concrete Containments," April 1979.

The required Special Reports from any engineering evaluation of containment abnormalties shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedure, the tolerances on cracking, the results of the engineering evaluation and the corrective actions taken.

3/4.6.1.7 CONTAINMENT PURGE VENTILATION SYSTEM

The 48-inch containment purge supply and exhaust isolation valves are required to be sealed closed (power removed) during plant operations since these valves have not been demonstrated capable of closing during a LOCA or steam line break accident. Maintaining these valves sealed closed during plant operation ensures that excessive quantities of radioactive material will not be released via the Containment Purge System. To provide assurance that the 48-inch containment valves cannot be inadvertently opened, the valves are sealed closed in accordance with Standard Review Plan 6.2.4 which includes mechanical devices to seal or lock the valve closed, or prevents power from being supplied to the valve operator.

The use of the containment purge lines is restricted to the 8-inch purge supply and exhaust isolation valves since, unlike the 48-inch valves, the 8-inch valves are capable of closing during a LOCA or steam line break accident. Therefore, the SITE BOUNDARY dose guideline values of 10 CFR Part 100 would not

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The core shall contain 193 fuel assemblies with each fuel assembly containing 264 fuel rods clad with Zircaloy-4 or ZIRLO, except that limited substitution of fuel rods by filler rods consisting of Zircaloy-4, ZIRLO, or stainless steel or by vacancies may be made if justified by a cycle specific reload analysis. Each fuel rod shall have a nominal active fuel length of 144 inches. The initial core loading shall have a maximum enrichment of less than 3.20 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading or previous cycle loading. The enrichment of any reload fuel design shall be determined to be acceptable for storage in either the spent fuel pool or the new fuel vault. Such acceptance criteria shall be based on the results of the CRITICALITY ANALYSIS OF BYRON AND BRAIDWOOD STATION FUEL STORAGE RACKS.

CONTROL ROD ASSEMBLIES

5.3.2 The core shall contain 53 full-length and no part-length control rod assemblies. The full-length control rod assemblies shall contain a nominal 142 inches of absorber material. All control rods shall be hafnium, silver-indium-cadmium, or a mixture of both types. All control rods shall be clad with stainless steel tubing.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The Reactor Coolant System is designed and shall be maintained:

a. In accordance with the Code requirements specified in Section 5.2 of the UFSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

5.4.2 The total water and steam volume of the Reactor Coolant System is 12,257 cubic feet at a nominal T_{avg} of 588.4°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-1.

An additional 1,251 ft^3 of volume is added to the Unit 1 total RCS volume as a result of the four replacement steam generators installed after cycle 7

ATTACHMENT C

EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATIONS FOR PROPOSED CHANGES TO APPENDIX A TECHNICAL SPECIFICATIONS OF FACILITY OPERATING LICENSES NPF-37, NPF-66, NPF-72, AND NPF-77

ComEd has evaluated this proposed amendment and determined that it involves no significant hazards considerations. According to Title 10 Code of Federal Regulations Section 50 Subsection 92 Paragraph c (10 CFR 50.92 (c)), a proposed amendment to an operating license involves no significant hazards considerations 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.

A. INTRODUCTION

Commonwealth Edison (ComEd) proposes to revise Technical Specifications (TS) 1.0, "Definitions," 3/4.6.1, "Primary Containment" and associated Bases, and 5.4.2, "Reactor Coolant System Volume," for Byron Nuclear Power Station (Byron) and Braidwood Nuclear Power Station (Braidwood) to support steam generator replacement. ComEd will be replacing the original Westinghouse D4 steam generators at Byron and Braidwood with Babcock and Wilcox International (BWI) steam generators. The replacement steam generators (RSGs) increase the Reactor Coolant System (RCS) volume which results in a higher calculated peak containment pressure (P_a) value. The RCS volume and P_a for Unit 2 will remain unchanged. Additionally, several editorial changes are being made to improve clarity and consistency of the TS.

B. NO SIGNIFICANT HAZARDS ANALYSIS

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

Each of the RSGs has a larger RCS side volume than the original steam generators (OSGs). As a result of the RCS volume increase, the mass and energy release during the blowdown phase of the large break loss of coolant accident (LBLOCA) is increased. Additionally, the heat transfer rate of the RSGs is greater than the OSGs, and the RSGs will operate at a slightly higher pressure than that for the OSGs. Consequently, the steam enthalpy exiting the break during the reflood period, with the RSG, will be greater than that for the OSG. This results in an increase in the containment building peak pressure, P_a.

The proposed revisions to the Technical Specifications involve the specified value of Unit 1 RCS volume and the defined value of Unit 1 P_a. Several editorial changes are also being made to improve clarity and consistency of the TS.

RCS volume is not an initiator for any event and an increase in volume does not affect any operating margin or requirements. Therefore, increasing the primary volume does not increase the probability of any event previously analyzed.

The revised value of P_a continues to be less than the design basis pressure for the containment building structure. The change represents only a revision to the containment test pressure for containment leakage testing. Such testing is only performed with the affected unit in the shutdown condition. Therefore, the proposed change in P_a does not involve a significant increase in the probability of an accident previously evaluated.

All accidents in the Updated Final Safety Analysis Report (UFSAR) were evaluated to determine the effect of an increase in primary volume on accident consequences. The events identified that may be impacted by an increase in primary volume are the Waste Gas System Leak or Failure and LBLOCA. For the Waste Gas System Leak or Failure, the activity of the decay tank is controlled to Technical Specification limits which are unaffected by RCS volume. Therefore, an increase in RCS volume would not increase the offsite dose.

The offsite dose calculation for the LBLOCA is unaffected by the proposed change. The license basis offsite dose calculation is in accordance with NRC Reg Guide 1.4 "Assumptions Used for Evaluating The Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors." This Regulatory Guide states, in part, "...a number of appropriately conservative assumptions, based on engineering judgment and on applicable experimental results from safety research programs conducted by the AEC." These conservatisms include (but are not limited to) the following assumptions:

- Twenty five percent of the equilibrium radioactive full power inventory is immediately available for leakage from the primary containment.

- 100 % of the equilibrium full power radioactive noble gas inventory is immediately available for leakage from the primary containment.
- The primary containment should be assumed to leak at the (maximum) leak rate specified in the technical specifications for the first 24 hours and at 50% of this value for the remaining 29 days of the accident duration.

The design basis leakage corresponding to a peak containment pressure of 50 psig utilized in the design basis accident analysis is 0.10% per day of the containment free air mass. Therefore, the offsite dose calculation was performed with a leakage of .1 % per day for day one and .05 % per day for days two through 30. Isotopic inventories are unaffected by the increase in reactor coolant volume. Thus, the offsite dose is unaffected by the increase in the peak containment pressure. Therefore, this proposed change to P_a does not involve a significant increase in the consequences of an accident previously evaluated.

The editorial changes proposed are for clarity and consistency within the Technical Specifications and do not affect either 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 in RCS volume is a change in a plant parameter within the "Design Features" section of the Technical Specifications. Increasing the RCS volume does not create any new or different failure modes. The existing RCS design requirements continue to be met.

The revised value of P_a continues to be less than the design basis pressure for the containment building structure. The change represents only a revision to the test pressure for containment leakage testing. Such testing is only performed with the affected unit in the shutdown condition. Therefore, no new or different failure modes are being introduced by modification of the testing parameters.

The editorial changes proposed are for clarity and consistency within the Technical Specifications and do not result in any physical changes to the facility or how it is operated. No new or different failure modes are being introduced by these changes.

Therefore, these proposed change do 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.

Changing the RCS volume in the Technical Specifications does not reduce the margin of safety. RCS volume is a design feature. The change in RCS volume does not involve a change to any setpoint or design requirements. An evaluation of all UFSAR accidents was performed to determine the effect of an increase in RCS volume. This evaluation is summarized as follows:

An evaluation of the Chemical and Volume Control System Malfunction was performed to determine the effect of the increased RCS volume due to the RSGs. The larger RCS volume of the RSGs reduces the reactivity insertion for a given dilution flow rate. Therefore, the UFSAR analyses remain bounding for Byron Unit 1 and Braidwood Unit 1 with the RSGs and there is no reduction in the margin of safety.

An evaluation of the Inadvertent Actuation of the Emergency Core Cooling System During Power Operation Event was performed to determine the effect of the increased RCS volume due to the RSGs. For this event, the injection of borated water causes a negative reactivity insertion, which increases DNBR. For a given Refueling Water Storage Tank (RWST) boron concentration, the larger RCS volume will cause a reduction in the negativity insertion rate as compared to the current UFSAR analysis. However, negative reactivity would still be inserted, no fuel pins would experience DNB, and there is no reduction in the margin of safety.

An evaluation of the Small Break LOCA was performed to determine the effect of increased RCS volume. The additional RCS volume will cause a delay in the loop seal clearing which in turn delays the core uncovery as compared with the UFSAR analysis. A delay in core uncovery reduces the amount of core heatup which results in a lower peak clad temperature (PCT) because the core decay heat would be less than in the UFSAR analysis. The benefit is considered small, but there is still a benefit. Therefore, the increased RCS volume does not result in a reduction in the margin of safety.

An evaluation of the Large Break LOCA was performed to determine the effect of increased RCS volume. For a LBLOCA, the increased RCS volume causes the blowdown phase of the event to be longer. Increased blowdown phase, alone, could potentially result in a higher PCT. However, the RSGs also have less resistance to flow due to increased primary side steam generator flow area, which results in a higher blowdown flow compared to the OSGs. The increased blowdown flow more than compensates for the longer blowdown phase associated with the increased RCS volume. The net effect is a decrease in PCT for the RSG compared to the OSG. Therefore, there is no reduction in the margin of safety.

An evaluation of the Gas Waste System Leak or Failure was performed to determine the effect of the increased RCS volume. Because the activity of the decay tank is controlled within Technical Specification limits, an increase in RCS volume would not change the results of the event. Therefore, there is no reduction in the margin of safety.

An evaluation was performed to determine the effect of the increased RCS volume on the peak containment pressure following a LBLOCA. The increased RCS volume caused the peak containment pressure to increase to 47.8 psig. This is still below the containment design pressure of 50.0 psig. Therefore, there is no reduction in the margin of safety.

This proposed change involves testing requirements designed to demonstrate adequate leakage rates are maintained. If adequate leakage rates are maintained as outlined in the Technical Specifications, there will be no reduction in the margin of safety. In the event of degradation of a containment seal that results in unacceptable leakage, plant shutdown will occur as required by Technical Specifications and administrative requirements in accordance with approved plant procedures. Therefore, this proposed change does not involve a significant reduction in a margin of safety.

The editorial changes proposed are for clarity and consistency within the Technical Specifications and do not result in any physical changes to the facility or how it is operated. Therefore, the changes have no effect on the margin of safety.

Thus, this amendment request does not result in any decrease in a margin of safety.

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

ATTACHMENT D

ENVIRONMENTAL ASSESSMENT FOR
PROPOSED CHANGES TO APPENDIX A
TECHNICAL SPECIFICATIONS OF
FACILITY OPERATING LICENSES
NPF-37, NPF-66, NPF-72, AND NPF-77

Commonwealth Edison Company (ComEd) has evaluated this proposed License Amendment Request against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with Title 10, Code of Federal Regulations, Part 51, Section 21 (10 CFR 51.21). ComEd has determined that this proposed License Amendment Request meets the criteria for a categorical exclusion set forth in 10 CFR 51.22(c)(9). This determination is based upon the following:

1. The proposed licensing action involves the issuance of an amendment to a license for a reactor pursuant to 10 CFR 50 which 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 which changes an inspection or a surveillance requirement. This proposed License Amendment Request will allow ComEd to revise the Unit 1 RCS volume, to revise the Unit 1 P_a value, and to make editorial changes;

2. this proposed License Amendment Request involves no significant hazards considerations;

3. there is no significant change in the types or significant increase in the amounts of any effluent that may be released offsite; and

4. there is no significant increase in individual or cumulative occupational radiation exposure.

Therefore, pursuant to 10 CFR 51.22(b), neither an environmental impact statement nor an environmental assessment is necessary for this proposed License Amendment Request.