January 22, 1998

Mr. Oliver D. Kingsley, President
Nuclear Generation Group
Commonwealth Edison Company
Executive Towers West III
1400 Opus Place, Suite 500
Downers Grove, IL 60515

SUBJECT: ISSUANCE OF AMENDMENTS (TAC NOS. M97851, M97852, M97853 AND M97854)

Dear Mr. Kingsley:

The U.S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 97 to Facility Operating License No. NPF-37 and Amendment No. 97 to Facility Operating License No. NPF-66 for the Byron Station, Unit Nos. 1 and 2, respectively, and Amendment No. 88 to Facility Operating License No. NPF-72 and Amendment No. 88 to Facility Operating License No. NPF-77 for the Braidwood Station, Unit Nos. 1 and 2, respectively. The amendments are in response to Commonwealth Edison Company's (ComEd) application dated January 30, 1997, as supplemented by letter dated December 9, 1997. Additional information was submitted in ComEd's letters of May 23, 1997, August 8, 1997 and January 7, 1998.

The amendments revise the technical specifications and associated bases related to the primary containment pressure and reactor coolant system volume. The changes resulted from the replacement of the steam generators at Byron, Unit 1 and Braidwood, Unit 1.

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Original signed by:

George F. Dick, Senior Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

DF01

Docket Nos. STN 50-454, STN 50-455,
STN 50-456 and STN 50-457

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Enclosures:
1. Amendment No. 97 to NPF-37
2. Amendment No. 97 to NPF-66
3. Amendment No. 88 to NPF-72
4. Amendment No. 88 to NPF-77
5. Safety Evaluation

cc w/encls: see next page

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*concurrence provided by SE; no major revisions

**see previous page for concurrence

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PDR ADOCK 05000454
P PDR

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

January 22, 1998

Mr. Oliver D. Kingsley, President
Nuclear Generation Group
Commonwealth Edison Company
Executive Towers West III
1400 Opus Place, Suite 500
Downers Grove, IL 60515

SUBJECT: ISSUANCE OF AMENDMENTS (TAC NOS. M97851, M97852, M97853 AND M97854)

Dear Mr. Kingsley:

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The amendments revise the technical specifications and associated bases related to the primary containment pressure and reactor coolant system volume. The changes resulted from the replacement of the steam generators at Byron, Unit 1 and Braidwood, Unit 1.

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

George F. Dick, Jr.

George F. Dick, Senior Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Docket Nos. STN 50-454, STN 50-455,
STN 50-456 and STN 50-457

Enclosures:
1. Amendment No. 97 to NPF-37
2. Amendment No. 97 to NPF-66
3. Amendment No. 88 to NPF-72
4. Amendment No. 88 to NPF-77
5. Safety Evaluation

cc w/encls: see next page

cc:

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Commonwealth Edison Company
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-454

BYRON STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 97
License No. NPF-37

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Commonwealth Edison Company (the licensee) dated January 30, 1997, as supplemented on December 9, 1997, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-37 is hereby amended to read as follows:

9802030151 980122
PDR ADOCK 05000454
P PDR

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 97 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

George F. Dick, Jr.

George F. Dick, Senior Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 22, 1998

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-455

BYRON STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 97
License No. NPF-66

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Commonwealth Edison Company (the licensee) dated January 30, 1997, as supplemented on December 9, 1997, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-66 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A (NUREG-1113), as revised through Amendment No. 97 and revised by Attachment 2 to NPF-66, and the Environmental Protection Plan contained in Appendix B, both of which were attached to License No. NPF-37, dated February 14, 1985, are hereby incorporated into this license. Attachment 2 contains a revision to Appendix A which is hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

George F. Dick, Jr.

George F. Dick, Senior Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 22, 1998

ATTACHMENT TO LICENSE AMENDMENT NOS. 97 AND 97

FACILITY OPERATING LICENSE NOS. NPF-37 AND NPF-66

DOCKET NOS. STN 50-454 AND STN 50-455

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change.

Remove Pages	Insert Pages
1-4	1-4
3/4 6-5	3/4 6-5
3/4 6-12	3/4 6-12
B 3/4 6-1	B 3/4 6-1
B 3/4 6-2	B 3/4 6-2
5-4	5-4

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 (OLR) 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*, 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 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 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 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.

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.

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.

The maximum increase in peak pressure expected to be obtained from a cold leg double-ended break event is defined as P_a. The limit of 1.0 psig for initial positive containment pressure will limit the total pressure to 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

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 P_a 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 "Byron and Braidwood Spent Fuel Rack Criticality Analysis Using Soluble Boron Credit," May 1997, CAC-97-162 and "Criticality Analysis of the Byron/Braidwood Fresh Fuel Racks," June 1989.

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 12,340 cubic feet at a nominal T_{avg} of 588.4°F. An additional 1,280 cubic feet at a nominal T_{avg} of 588.4°F is added to the Unit 1 total Reactor Coolant System volume as a result of the four replacement steam generators installed after cycle 8.

5.5 METEOROLOGICAL TOWER LOCATION

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

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-456

BRAIDWOOD STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 88
License No. NPF-72

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Commonwealth Edison Company (the licensee) dated January 30, 1997, as supplemented on December 9, 1997, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-72 is hereby amended to read as follows:

9802050121 980122
PDR ADOCK 05000454
P PDR

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 88 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

George F. Dick, Jr.

George F. Dick, Senior Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 22, 1998

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-457

BRAIDWOOD STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 88
License No. NPF-77

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Commonwealth Edison Company (the licensee) dated January 30, 1997, as supplemented on December 9, 1997, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-77 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 88 and the Environmental Protection Plan contained in Appendix B, both of which were attached to License No. NPF-72, dated July 2, 1987, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date if its issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

George F. Dick, Senior Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 22, 1998

ATTACHMENT TO LICENSE AMENDMENT NOS. 88 AND 88

FACILITY OPERATING LICENSE NOS. NPF-72 AND NPF-77

DOCKET NOS. STN 50-456 AND STN 50-457

Replace the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change.

Remove Pages	Insert Pages
1-4	1-4
3/4 6-5	3/4 6-5
3/4 6-12	3/4 6-12
B 3/4 6-1	B 3/4 6-1
B 3/4 6-2	B 3/4 6-2
5-4	5-4

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 (OLR) 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*, 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 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 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 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.

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.

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.

The maximum increase in peak pressure expected to be obtained from a cold leg double-ended break event is defined as P_a. The limit of 1.0 psig for initial positive containment pressure will limit the total pressure to 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

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 P_a 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 "Byron and Braidwood Spent Fuel Rack Criticality Analysis Using Soluble Boron Credit," May 1997, CAC-97-162 and "Criticality Analysis of the Byron/Braidwood Fresh Fuel Racks," June 1989.

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 12,340 cubic feet at a nominal T_{avg} of 588.4°F. An additional 1,280 cubic feet at a nominal T_{avg} of 588.4°F is added to the Unit 1 total Reactor Coolant System volume as a result of the four replacement steam generators installed after cycle 7.

5.5 METEOROLOGICAL TOWER LOCATION

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

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 97 TO FACILITY OPERATING LICENSE NO. NPF-37,

AMENDMENT NO. 97 TO FACILITY OPERATING LICENSE NO. NPF-66,

AMENDMENT NO. 88 TO FACILITY OPERATING LICENSE NO. NPF-72,

AND AMENDMENT NO. 88 TO FACILITY OPERATING LICENSE NO. NPF-77

COMMONWEALTH EDISON COMPANY

BYRON STATION, UNIT NOS. 1 AND 2

BRAIDWOOD STATION, UNIT NOS. 1 AND 2

DOCKET NOS. STN 50-454, STN 50-455, STN 50-456 AND STN 50-457

1.0 INTRODUCTION

By application dated January 30, 1997, as supplemented by letter dated December 9, 1997, Commonwealth Edison Company (ComEd) the licensee for Byron and Braidwood Stations, requested changes to the Technical Specifications (TS) for the Byron and Braidwood facilities. Additional information that did not change the initial proposed no significant hazards consideration determination was provided by letters dated May 23, August 8, 1997, and January 7, 1998. The proposed changes would reflect the higher loss-of-coolant accident (LOCA) peak containment pressure calculation due to steam generator (SG) replacements and would also implement TS administrative improvements having no technical significance. ComEd will be replacing the original Westinghouse D4 SG in the Byron and Braidwood #1 units with Babcock & Wilcox International (BWI) SG. The SG replacements are a 10 CFR 50.59 modification that will increase the reactor coolant system (RCS) volume and secondary side pressure. This results in a greater steam mass and energy release in the event of a LOCA. The greater mass and energy release, in turn, results in an increase of the calculated peak containment internal pressure (Pa) related to the design basis LOCA. Since Pa is specified in the TS, changes are needed to increase the test pressure for the #1 units. The TS changes for the #2 units are necessary because common TS pages are used for Units 1 and 2.

2.0 AFFECTED TECHNICAL SPECIFICATIONS

The following changes are proposed:

TS 1.20.a defines Pa and specifies the current value as 44.4 psig. This TS would be changed to indicate that Pa will be 47.8 psig for Unit 1 after Cycle 8.

TS 4.6.1.3 would be changed to specify air lock door seal maximum leakage limits solely in terms of percentage of maximum allowable primary containment leakage rate (La) rather than in terms of both percentage of La and in standard cubic feet per hour (SCFH).

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P PDR

> TS 4.6.1.7.3 & 4 would be revised to specify that the containment purge supply and exhaust isolation valve resilient seal test pressure is "Pa" rather than "Pa, 44.4 psig."
>
> TS BASES 3/4.6.1.4 and 3/4.6.1.6 would be revised in several places to substitute the term Pa for "44.4. psig."

The above changes are due to the increase in calculated peak accident pressure from 44.4 psig to 47.8 psig. By (a) stating the exact value of Pa in one place only (i.e., the TS Definitions section) and using the term "Pa" alone elsewhere, and (b) stating specified leakage limits in terms of percentage of La, the TS are simplified so as to minimize the number of TS pages affected by differences among units.

Also:

> TS 5.4.2 2 would be revised to indicate a corrected value of 12,340 cubic feet for the RCS volume and that for each Unit 1, the RCS volume of 12,340 cubic feet is increased by an additional 1,280 cubic feet as a result of SG replacement.

This change would reflect the effects of SG replacement on the RCS volume.

3.0 EVALUATION

To support the increased RCS volume, the licensee has performed an evaluation of all accident analyses documented in the Updated Final Safety Analysis Report (UFSAR) to determine the effect of an increase in RCS volume on the consequences of the accidents analyzed. The results of the licensee's evaluation concluded that the increased RCS volume will not result in a reduction of the safety margin.

In the licensee's evaluation performed to determine the effect of the increased RCS volume on the peak containment pressure following a large break loss of coolant accident (LBLOCA), the calculation indicated that the increased RCS volume could cause the peak containment pressure to increase to 47.8 psig. However, the licensee states that this increased back pressure is still below the containment design pressure of 50.0 psig.

Pa is defined in Appendix J as:

> Pa (p.s.i.g.) means the calculated peak containment internal pressure related to the design basis accident and specified either in the technical specification or associated bases.

The design basis accident for calculation of Pa is the postulated LOCA case which produces the highest containment peak internal pressure considering the complete spectrum of primary coolant piping system break sizes, break locations, single-failures and initial operating conditions. A new maximum peak accident pressure was calculated by applying the incremental effects of the larger SG to the Final Safety Analysis Report (FSAR) mass and energy release analysis. The licensee increased the blowdown phase data by 54,256 lbm and 33.084E+07 BTU to account for the mass and energy added by the additional RCS fluid inventory of the replacement SG. For the reflood and post-reflood phases of a LOCA, the enthalphy of the fluid discharged from a break after passing through a SG would be higher with the replacement SG due to the higher secondary side temperature. To account for this effect, the licensee increased the enthalphy of

the fluid discharge during reflood and post-reflood by 3.0 BTU/lbm. The 3 BTU/lbm value corresponds to an increase in secondary side T(sat) of 5.6 degrees Fahrenheit. Also, the SG depressurization time was increased to allow for dissipation of the additional stored energy in the larger metal mass and secondary side water inventory of the replacement SG's. The revised mass and energy data were input into a new CONTEMPT-B&W containment model which had been previously benchmarked against the original licensing model using the original mass and energy data. The licensee's analytical methodology is sufficiently accurate and conservative for the purpose of calculating a new containment leakage rate test pressure. The licensee calculated the new Pa to be 47.8 psig. With regard to the qualification of equipment, the licensee determined that the revised temperature profile for the large break LOCA with the replacement steam generators is bounded by the equipment qualification envelope currently used to qualify equipment at the Byron and Braidwood Stations.

4.0 SUMMARY

The staff reviewed the licensee's submittals and methodology for calculation of the new containment peak accident (LOCA) pressure and found it acceptable. The proposed amendment will revise the containment test pressure "Pa" specification to conform with the new value. This is consistent with Appendix J to 10 CFR, Part 50 which requires that the LOCA peak accident pressure be identified in the TSs or TS Bases. The recalculated value of Pa remains bounded by the containment design pressure thereby maintaining containment structural margins and is acceptable. The other TS changes are of an editorial nature and are acceptable on the basis that the increased RCS volume will not result in a reduction of the margin of safety.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Illinois State official was notified of the proposed issuance of the amendments. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (62 FR 19826 and 62 FR 66699). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

7.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

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Date: January 22, 1998