September 11, 1991

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

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Mr. Thomas J. Kovach
Nuclear Licensing Manager
Commonwealth Edison Company-Suite 300
OPUS West III
1400 OPUS Place
Downers Grove, Illinois 60515

Dear Mr. Kovach:

SUBJECT: ISSUANCE OF AMENDMENTS (TAC NOS. 69527, 69528, 69529 AND 69530)

The Commission has issued the enclosed Amendment No. 42 to Facility Operating License No. NPF-37 and Amendment No. 42 to Facility Operating License No. NPF-66 for the Byron Station, Unit Nos. 1 and 2, respectively, and Amendment No. 31 to Facility Operating License No. NPF-72 and Amendment No. 31 to Facility Operating License No. NPF-77 for the Braidwood Station, Unit Nos. 1 and 2, respectively. The amendments are in response to your application dated October 4, 1988, as supplemented on August 14, 1989, March 20 and April 8, 1991.

These amendments would change Technical Specification 4.6.1.3a to be consistent with the Standard Technical Specifications and allow continuous pressure testing of the containment air lock door seal gaskets. Bases section 3/4.6.1.3 is also changed to reflect the need to use precision flow measurements to determine seal leakage when the permanently installed continuous pressurization and monitoring system is incapable of providing continuous monitoring in the control room.

A copy of the related Safety Evaluation and Notice of Issuance are also enclosed.

Sincerely,
Original Signed By:
Robert M. Pulsifer, Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 42 to NPF-37
2. Amendment No. 42 to NPF-66
3. Amendment No. 31 to NPF-72
4. Amendment No. 31 to NPF-77
5. Safety Evaluation
6. Notice

cc w/enclosures:
See next page

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DOCUMENT NAME: [AMENDMENT 69527/8/9/30]

Office:	LA/PDIII-2	PM/PDIII-2	PM/PDIII-2	BC/SPLB	BD/PDIII-2	OGC-WF1
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Dat:	8/23/91	8/23/91	8/23/91	8/22/91	9/10/91	8/26/91

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Mr. Thomas J. Kovach
Commonwealth Edison Company

Byron/Braidwood Power Stations

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Chairman
Will County Board of Supervisors
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Joliet, Illinois 60434

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-454

BYRON STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 42
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 October 4, 1988, as supplemented on August 14, 1989, March 20 and April 8, 1991, 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:

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(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 42 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.

FOR THE NUCLEAR REGULATORY COMMISSION

for

Richard J. Barrett, Director
Project Directorate III-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 11, 1991

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-455

BYRON STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 42
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 October 4, 1988, as supplemented on August 14, 1989, March 20 and April 8, 1991, 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 1;

   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. 42 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.

FOR THE NUCLEAR REGULATORY COMMISSION

for

Richard J. Barrett, Director
Project Directorate III-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 11, 1991

ATTACHMENT TO LICENSE AMENDMENT NOS. 42 AND 42

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. Overleaf pages as indicated by an asterisk are provided for convenience.

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

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

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

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

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

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

1) At least once per 6 months,* and

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

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

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

*The provisions of Specification 4.0.2 are not applicable.

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

INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.4 Primary containment internal pressure shall be maintained between -0.1 and +1.0 psig.

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

ACTION:

With the containment internal pressure outside of the limits above, restore the internal pressure to within the limits 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

4.6.1.4 The primary containment internal pressure shall be determined to be within the limits at least once per 12 hours.

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with the leakage rate limitation, will limit the SITE BOUNDARY radiation doses to within the dose guideline values of 10 CFR Part 100 during accident conditions.

3/4.6.1.2 CONTAINMENT LEAKAGE

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

The surveillance testing for measuring leakage rates are consistent with the requirements of Appendix J of 10 CFR Part 50.

3/4.6.1.3 CONTAINMENT AIR LOCKS

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

3/4.6.1.4 INTERNAL PRESSURE

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

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

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

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-456

BRAIDWOOD STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 31
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 October 4, 1988, as supplemented on August 14, 1989, March 20 and April 8, 1991, 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:

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 31 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.

FOR THE NUCLEAR REGULATORY COMMISSION

for

Richard J. Barrett, Director
Project Directorate III-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 11, 1991

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-457

BRAIDWOOD STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 31
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 October 4, 1988, as supplemented on August 14, 1989, March 20 and April 8, 1991, 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 1;

   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. 31 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.

FOR THE NUCLEAR REGULATORY COMMISSION

for

Richard J. Barrett, Director
Project Directorate III-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 11, 1991

ATTACHMENT TO LICENSE AMENDMENT NOS. 31 AND 31

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. Overleaf pages as indicated by an asterisk are provided for convenience.

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

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

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

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

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

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

1) At least once per 6 months,* and

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

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

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

*The provisions of Specification 4.0.2 are not applicable.

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

INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.4 Primary containment internal pressure shall be maintained between -0.1 and +1.0 psig.

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

ACTION:

With the containment internal pressure outside of the limits above, restore the internal pressure to within the limits 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

4.6.1.4 The primary containment internal pressure shall be determined to be within the limits at least once per 12 hours.

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with the leakage rate limitation, will limit the SITE BOUNDARY radiation doses to within the dose guideline values of 10 CFR Part 100 during accident conditions.

3/4.6.1.2 CONTAINMENT LEAKAGE

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

The surveillance testing for measuring leakage rates are consistent with the requirements of Appendix J of 10 CFR Part 50.

3/4.6.1.3 CONTAINMENT AIR LOCKS

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

3/4.6.1.4 INTERNAL PRESSURE

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

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

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 ntainment 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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 42 TO FACILITY OPERATING LICENSE NO. NPF-37,

AMENDMENT NO. 42 TO FACILITY OPERATING LICENSE NO. NPF-66,

AMENDMENT NO. 31 TO FACILITY OPERATING LICENSE NO. NPF-72,

AND AMENDMENT NO. 31 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 letter dated October 4, 1988, and follow-up letters dated August 14, 1989, March 20 and April 8, 1991, Commonwealth Edison Company (CECo, the licensee) requested an amendment to change the Technical Specifications (TS) to (1) be consistent with NUREG-0452, Revision 4a, "Standard Technical Specifications for Westinghouse Pressurized Water Reactors," and (2) to reflect the installation of an automatic leakage rate detection system to provide continuous pressure testing of the containment air lock door seal gaskets. The first change reflects the NRC staff's position that, for the 72-hour air lock operability test, a constantly applied pressure of at least 10 psig for 30 seconds--instead of 15 minutes--may be sufficient to determine if seal leakage is less than the maximum allowable leakage rate of 0.01La. The second change reflects the installation of a continuous pressurization and leakage monitoring system (already installed at Byron 1, but proposed for all four units) which would continuously pressurize the volume between the door seals to greater than or equal to 3 psig, and monitor the leakage rate of each air lock door by a common flow instrument.

2.0 EVALUATION

The staff reviewed the licensee's proposed TS amendment which would reduce the pressurization time from 15 minutes to 30 seconds for the post-entry, 72-hour containment air lock operability test. This change would reflect the staff's position that 30 seconds of constantly applied pressure of at least 10 psig to the volume between the air lock door seals is sufficient to determine if there is a leakage from the seals which exceeds the maximum allowable leakage rate of 0.01La. This position has been reflected in Revision 4a to NUREG-0452, "Standard Technical Specifications for Westinghouse Pressurized Water Reactors," and has previously been incorporated into the TS

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of several other operating reactors in the country. CECo, in its submittal, stated that the volumes between the door seals for the Byron and Braidwood facilities are small and require only a short time period to stabilize at a constant pressure. The licensee assured the staff that once the pressure is stabilized, a reliable leak test can be performed within 30 seconds.

Based on the above, the staff finds the proposed change which requires the volume between the air lock door seals to be pressurized to a constant pressure of greater than or equal to 10 psig and then measured for leakage for a period of 30 seconds to be acceptable.

The staff also reviewed the licensee's proposed TS amendment which would reflect the installation of a continuous pressurization and leakage monitoring system. As an alternative to the above-mentioned 30 second test, this system modification would permit continuous reduced pressure (3 psig) testing of the air lock door seals. Instrument air lines would provide a regulated instrument air supply of greater than or equal to 3 psig between the door gasket interspace. The leakage rate of each air lock door would then be measured by a common flow element, and any excessive leakage condition would alarm in the control room. The licensee also proposes to use an extrapolation method to provide a correlation between actual observed leakage rates at 3 psig and maximum allowable leakage rates at 10 psig, the currently required test pressure. Currently, the Byron and Braidwood TS provide for a test for leakage less than 0.01La at 10 psig within 72 hours following each air lock closing, except when the air lock is being used for multiple entries, then at least once per 72 hours. The proposed amendments would change this to require, at the same frequency as above, a test for leakage less than 0.01La at 10 psig or greater, using precision flow measurements; or a test for leakage less than 0.0024La at 3 psig or greater using the automatic, continuous leakage monitoring system. In addition at least once per six months a test for leakage less than 0.01La at 10 psig is provided for. The licensee stated that the proposed testing would provide assurance that the limiting condition for operation would be met. The staff has reviewed the licensee's submittals and concurs with the licensee's conclusions and, therefore, finds the proposed changes described above to be acceptable.

The Bases for Specification 3/4.6.1.3 is changed to reflect that the precision flow measurement option must be used whenever the continuous monitoring capability is lost in the control room from the permanently installed continuous pressurization and monitoring system.

Based on the above, the licensee's proposed amendments to: (1) change the constant pressurization time for the post-entry 72-hour airlock test from 15 minutes to 30 seconds, and (2) reflect the installation of a continuous pressurization and leakage monitoring system which would allow for continuous reduced pressure testing (at 3 psig) of the containment air lock seals; the staff makes the following conclusions: (1) thirty seconds is a sufficient

time period for the volume between the door seals to be held at a constant pressure of 10 psig, and any leakage in excess of the maximum allowable leakage rate can be discovered in that time period, (2) test for leakage less than 0.0024La at 3 psig or greater using the automatic continuous leakage monitoring system or a test for leakage less than 0.01La at 10 psig or greater using precision flow measurements is adequate. Therefore, the staff finds revision (1) and (2) acceptable.

3.0 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.

4.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact has been prepared and published in the Federal Register on August 27, 1991 (56 FR 42362).

Accordingly, based upon the environmental assessment, the Commission has determined that the issuance of this amendment will not have a significant effect on the quality of the human environment.

5.0 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.

Principal Contributor: A. Drozd, D. Roberts, R. Pulsifer

Date: September 11, 1991

UNITED STATES NUCLEAR REGULATORY COMMISSION

COMMONWEALTH EDISON COMPANY

BYRON STATION, UNIT NOS. 1 AND 2

BRAIDWOOD STATION, UNIT NOS. 1 AND 2

NOTICE OF ISSUANCE OF AMENDMENT TO

FACILITY OPERATING LICENSE

The U.S. Nuclear Regulatory Commission (Commission) has issued Amendment No. 42 to Facility Operating License No. NPF-37, Amendment No. 42 to Facility Operating License No. NPF-66, Amendment No. 31 to Facility Operating License No. NPF-72, and Amendment No. 31 to Facility Operating License No. NPF-77, issued to Commonwealth Edision Company (CECo, the licensee), which revised the Technical Specifications for operation of the Byron Station, Unit Nos. 1 and 2, and Braidwood Station, Unit Nos. 1 and 2, located in Ogle County and Will County, Illinois, respectively. The amendments are effective as of the date of issuance.

The amendments modified the Technical Specifications to (1) be consistent with Standard Technical Specifications, and (2) to reflect the installation of an automatic leakage rate detection system to provide continuous pressure testing of the containment air lock door seal gaskets.

The application for the amendments comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments.

Notice of Consideration of Issuance of Amendment and Opportunity for Hearing in connection with this action was published in the FEDERAL REGISTER on

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November 4, 1988 (53 FR 44685). No request for a hearing or petition for leave to intervene was filed following this notice.

The Commission has prepared an Environmental Assessment related to the action and has determined not to prepare an environmental impact statement. Based upon the environmental assessment, the Commission has concluded that the issuance of these amendments will not have a significant effect on the quality of the human environment.

For further details with respect to the action see (1) the application for amendments dated October 4, 1988, as supplemented on August 14, 1989, March 20 and April 8, 1991, (2) Amendment Nos. 42, 42, 31, 31, to License Nos. NPF-37, NPF-66, NPF-72, NPF-77, respectively, (3) the Commission's related Safety Evaluation, and (4) the Commission's Environmental Assessment. All of these items are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street NW., Washington, DC and at the local public document rooms located at: for Byron, the Byron Public Library, 109 N. Franklin, P. O. Box 434, Byron, Illinois 61010; for Braidwood, the Wilmington Township Public Library, 201 S. Kankakee Street, Wilmington, Illinois 60481. A copy of items (2), (3), and (4) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Director, Division of Reactor Projects - III/IV/V.

Dated at Rockville, Maryland this 11th day of September 1991.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert M. Pulsifer, Project Manager,
Project Directorate III-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation