Docket File
50-454

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

April 4, 1996

Mr. D. L. Farrar
Manager, Nuclear Regulatory Services
Commonwealth Edison Company
Executive Towers West III
1400 Opus Place, Suite 500
Downers Grove, IL 60515

SUBJECT: ISSUANCE OF AMENDMENTS (TAC NOS. M94212, M94213, M94214 AND M94215)

Dear Mr. Farrar:

The U.S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 81 to Facility Operating License No. NPF-37 and Amendment No. 81 to Facility Operating License No. NPF-66 for the Byron Station, Unit Nos. 1 and 2, respectively, and Amendment No. 73 to Facility Operating License No. NPF-72 and Amendment No. 73 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 December 6, 1995, as supplemented February 27, 1996 and March 28, 1996.

The amendments modify the technical specifications (TSs) to replace the existing scheduling requirements for overall integrated and local containment leakage rate testing with a requirement to perform the testing in accordance with 10 CFR Part 50, Appendix J, Option B. Option B allows test scheduling to be adjusted based on past performance.

Proposed changes to TS 4.6.1.7.3 regarding the testing of valves with resilient seal material is outside the scope of the revisions to 10 CFR Part 50, Appendix J, and can not be approved. The basis for this finding is documented in the enclosed Safety Evaluation (SE). A copy of the Notice of Partial Denial of Amendment to be published in the Federal Register is enclosed for your information.

The proposed change to TS 4.6.1.2 regarding the interval for Type A tests included in your March 28, 1996, submittal is still under review.

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A copy of the SE 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 for:

Ramin R. Assa, 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. 81 to NPF-37
2. Amendment No. 81 to NPF-66
3. Amendment No. 73 to NPF-72
4. Amendment No. 73 to NPF-77
5. Notice of Partial Denial of Amendment
6. Safety Evaluation

cc w/encl: see next page

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R. Assa	C. Moore (2)	C. Grimes, 011E22

DOCUMENT NAME: BB94212.AMD * See previous concurrence

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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. 81
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 December 6, 1995, as supplemented February 27, 1996, and March 28, 1996, 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. 81 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

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: April 4, 1996

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. 81
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 December 6, 1995, as supplemented February 27, 1996, and March 28, 1996, 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. 81 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

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: April 4, 1996

ATTACHMENT TO LICENSE AMENDMENT NOS. 81 AND 81

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
I	I
1-3	1-3
1-4	1-4
3/4 6-1	3/4 6-1
3/4 6-2	3/4 6-2
3/4 6-3	3/4 6-3
3/4 6-4	3/4 6-4
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

INDEX

DEFINITIONS

SECTION	PAGE
1.0 DEFINITIONS	
1.1 ACTION	1-1
1.2 ACTUATION LOGIC TEST	1-1
1.3 ANALOG CHANNEL OPERATIONAL TEST	1-1
1.4 AXIAL FLUX DIFFERENCE	1-1
1.5 CHANNEL CALIBRATION	1-1
1.6 CHANNEL CHECK	1-1
1.7 CONTAINMENT INTEGRITY	1-2
1.8 CONTROLLED LEAKAGE	1-2
1.9 CORE ALTERATION	1-2
1.9.a CRITICALITY ANALYSIS OF BYRON AND BRAIDWOOD STATION FUEL STORAGE RACKS	1-2
1.10 DIGITAL CHANNEL OPERATIONAL TEST	1-2
1.11 DOSE EQUIVALENT I-131	1-2a
1.12 $\bar{E}$-AVERAGE DISINTEGRATION ENERGY	1-3
1.13 ENGINEERED SAFETY FEATURES RESPONSE TIME	1-3
1.14 FREQUENCY NOTATION	1-3
1.15 IDENTIFIED LEAKAGE	1-3
1.15.a L_a	1-3
1.16 MASTER RELAY TEST	1-3
1.17 MEMBER(S) OF THE PUBLIC	1-3
1.18 OFFSITE DOSE CALCULATION MANUAL	1-4
1.19 OPERABLE - OPERABILITY	1-4
1.19.a OPERATING LIMITS REPORT	1-4
1.20 OPERATIONAL MODE - MODE	1-4
1.20.a P_a	1-4
1.21 PHYSICS TESTS	1-4
1.22 PRESSURE BOUNDARY LEAKAGE	1-4
1.23 PROCESS CONTROL PROGRAM	1-5
1.24 PURGE - PURGING	1-5
1.25 QUADRANT POWER TILT RATIO	1-5
1.26 RATED THERMAL POWER	1-5
1.27 REACTOR TRIP SYSTEM RESPONSE TIME	1-5
1.28 REPORTABLE EVENT	1-5

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

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

ENGINEERED SAFETY FEATURES RESPONSE TIME

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

FREQUENCY NOTATION

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

IDENTIFIED LEAKAGE

1.15 IDENTIFIED LEAKAGE shall be:

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

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

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

L_a

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

MASTER RELAY TEST

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

MEMBER(S) OF THE PUBLIC

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

OFFSITE DOSE CALCULATION MANUAL

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

OPERABLE - OPERABILITY

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

OPERATING LIMITS REPORT

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

OPERATIONAL MODE - MODE

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

P_a

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

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.

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

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

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

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

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

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited to:

a. An overall integrated leakage rate of less than or equal to L_a at P_a.

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

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.1.2 The containment leakage rates shall be demonstrated in accordance with Regulatory Guide 1.163, September 1995, and 10 CFR 50, Appendix J, Option B.

a. Type A (Overall Integrated Containment Leakage Rate) testing shall be conducted in accordance with Regulatory Guide 1.163, September 1995, and 10 CFR 50, Appendix J, Option B.

SURVEILLANCE REQUIREMENTS (Continued)

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

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

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

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

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

g. The provisions of Specification 4.0.2 are not applicable.

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE with:

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

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

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

ACTION:

a. With one containment air lock door inoperable:

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

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

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

4. The provisions of Specification 3.0.4 are not applicable.

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

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

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 Leakage testing shall be conducted on each 8-inch containment purge supply and exhaust isolation valve with resilient material seals by verifying that the measured leakage rate is less than 0.01 L_a in accordance with Regulatory Guide 1.163, September 1995, and 10 CFR 50, Appendix J, Option B.

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 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.

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. 73
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 December 6, 1995, as supplemented February 27, 1996, and March 28, 1996, 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:

9604110277 960404
PDR ADOCK 05000454
P PDR

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 73 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 Ramin R. Assa, 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: April 4, 1996

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. 73
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 December 6, 1995, as supplemented February 27, 1996, and March 28, 1996, 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. 73 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 Ramin R. Assa, 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: April 4, 1996

ATTACHMENT TO LICENSE AMENDMENT NOS. 73 AND 73

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
I	I
1-3	1-3
1-4	1-4
3/4 6-1	3/4 6-1
3/4 6-2	3/4 6-2
3/4 6-3	3/4 6-3
3/4 6-4	3/4 6-4
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

INDEX

DEFINITIONS

SECTION	PAGE
1.0 DEFINITIONS	
1.1 ACTION	1-1
1.2 ACTUATION LOGIC TEST	1-1
1.3 ANALOG CHANNEL OPERATIONAL TEST	1-1
1.4 AXIAL FLUX DIFFERENCE	1-1
1.5 CHANNEL CALIBRATION	1-1
1.6 CHANNEL CHECK	1-1
1.7 CONTAINMENT INTEGRITY	1-2
1.8 CONTROLLED LEAKAGE	1-2
1.9 CORE ALTERATION	1-2
1.9.a CRITICALITY ANALYSIS OF BYRON AND BRAIDWOOD STATION FUEL STORAGE RACKS	1-2
1.10 DIGITAL CHANNEL OPERATIONAL TEST	1-2
1.11 DOSE EQUIVALENT I-131	1-2
1.12 $\bar{E}$-AVERAGE DISINTEGRATION ENERGY	1-3
1.13 ENGINEERED SAFETY FEATURES RESPONSE TIME	1-3
1.14 FREQUENCY NOTATION	1-3
1.15 IDENTIFIED LEAKAGE	1-3
1.15.a L_a	1-3
1.16 MASTER RELAY TEST	1-3
1.17 MEMBER(S) OF THE PUBLIC	1-3
1.18 OFFSITE DOSE CALCULATION MANUAL	1-4
1.19 OPERABLE - OPERABILITY	1-4
1.19.a OPERATING LIMITS REPORT	1-4
1.20 OPERATIONAL MODE - MODE	1-4
1.20.a P_a	1-4
1.21 PHYSICS TESTS	1-4
1.22 PRESSURE BOUNDARY LEAKAGE	1-4
1.23 PROCESS CONTROL PROGRAM	1-5
1.24 PURGE - PURGING	1-5
1.25 QUADRANT POWER TILT RATIO	1-5
1.26 RATED THERMAL POWER	1-5
1.27 REACTOR TRIP SYSTEM RESPONSE TIME	1-5
1.28 REPORTABLE EVENT	1-5

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

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

ENGINEERED SAFETY FEATURES RESPONSE TIME

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

FREQUENCY NOTATION

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

IDENTIFIED LEAKAGE

1.15 IDENTIFIED LEAKAGE shall be:

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

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

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

L_a

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

MASTER RELAY TEST

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

MEMBER(S) OF THE PUBLIC

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

OFFSITE DOSE CALCULATION MANUAL

1.18 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Sections 6.8.4.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) 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.

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

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

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

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

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

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited to:

a. An overall integrated leakage rate of less than or equal to L_a at P_a.

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

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.1.2 The containment leakage rates shall be demonstrated in accordance with Regulatory Guide 1.163, September 1995, and 10 CFR 50, Appendix J, Option B.

a. Type A (Overall Integrated Containment Leakage Rate) testing shall be conducted in accordance with Regulatory Guide 1.163, September 1995, and 10 CFR 50, Appendix J, Option B.

SURVEILLANCE REQUIREMENTS (Continued)

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

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

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

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

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

g. The provisions of Specification 4.0.2 are not applicable.

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE with:

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

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

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

ACTION:

a. With one containment air lock door inoperable:

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

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

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

4. The provisions of Specification 3.0.4 are not applicable.

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

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

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 Leakage testing shall be conducted on each 8-inch containment purge supply and exhaust isolation valve with resilient material seals by verifying that the measured leakage rate is less than 0.01 L_a in accordance with Regulatory Guide 1.163, September 1995, and 10 CFR 50, Appendix J, Option B.

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.

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.

UNITED STATES NUCLEAR REGULATORY COMMISSION

COMMONWEALTH EDISON COMPANY

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

NOTICE OF PARTIAL DENIAL OF AMENDMENT TO FACILITY OPERATING LICENSES

AND OPPORTUNITY FOR HEARING

The U.S. Nuclear Regulatory Commission (the Commission) has partially denied a request by Commonwealth Edison Company (ComEd, the licensee) for an amendment to Facility Operating License Nos. NPF-37, NPF-62, NPF-72 and NPF-77, issued to the licensee for operation of Byron Station, Units 1 and 2, located in Ogle County, Illinois and Braidwood Station, Units 1 and 2, located in Will County, Illinois. Notice of Consideration of Issuance of this amendment was published in the FEDERAL REGISTER on February 28, 1996 (61 FR 7547).

The purpose of the licensee's amendment request was to revise the Technical Specifications (TS) to replace the existing scheduling requirements for overall integrated and local containment leakage rate testing with a requirement to perform the testing in accordance with 10 CFR Part 50, Appendix J, Option B. As part of its submittal, ComEd proposed to revise the TS regarding the testing of valves with resilient seal material. The scope of the staff's revisions to 10 CFR Part 50, Appendix J, did not include changes to testing of such valves and ComEd's submittal did not include sufficient information for the staff to evaluate the proposed change independently of the others. Therefore, the staff has concluded that that portion of the licensee's request can not be granted. The licensee was notified of the Commission's partial denial of the proposed change by a letter dated April 4, 1996.

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By May 10, 1996, the licensee may demand a hearing with respect to the partial denial described above. Any person whose interest may be affected by this proceeding may file a written petition for leave to intervene.

A request for hearing or petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Docketing and Services Branch, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, by the above date.

A copy of any petitions should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and to Michael I. Miller, Esquire; Sidley and Austin, One First National Plaza, Chicago, Illinois 60603, attorney for the licensee.

For further details with respect to this action, see (1) the application for amendment dated December 6, 1995, as supplemented February 27, 1996, and March 28, 1996, and (2) the Commission's letter to the licensee dated April 4, 1996.

These documents 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 room located at: for Byron, the Byron Public Library District, 109 N. Franklin, P.O. Box 434, Byron, Illinois 61010; for Braidwood, the Wilmington Public Library, 201 S. Kankakee Street, Wilmington, Illinois 60481.

Dated at Rockville, Maryland, this 4th day of April 1996.

FOR THE NUCLEAR REGULATORY COMMISSION

Ramin R. Assa, Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 81 TO FACILITY OPERATING LICENSE NO. NPF-37,

AMENDMENT NO. 81 TO FACILITY OPERATING LICENSE NO. NPF-66,

AMENDMENT NO. 73 TO FACILITY OPERATING LICENSE NO. NPF-72,

AND AMENDMENT NO. 73 TO FACILITY OPERATING LICENSE NO. NPF-77

COMMONWEALTH EDISON COMPANY

BYRON STATION, UNITS 1 AND 2

BRAIDWOOD STATION, UNITS 1 AND 2

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

1.0 INTRODUCTION

By letter dated December 6, 1995, as supplemented February 27, 1996, and March 28, 1996, the Commonwealth Edison Company (ComEd, the licensee) requested amendments to the Technical Specifications (TS) appended to Facility Operating License Nos. NPF-37 and NPF-66 for the Byron Station, Units 1 and 2, and Operating License Nos. NPF-72 and NPF-77 for the Braidwood Station, Units 1 and 2. The proposed amendments would revise the TSs to reflect the approval for the licensee to use 10 CFR Part 50, Appendix J, Option B for the Byron and Braidwood containment leakage rate test program. The February 27, 1996, supplement modified the TS pages to be more consistent with the published guidance, but did not change the primary intent of the request. The March 28, 1996, letter modified TS 4.6.1.2 to indicate that the minimum interval between Type A tests is the nominal refueling outage and TS 4.6.1.3.a to be consistent with the intent of the TSs developed by the staff in cooperation with NEI. Neither supplemental submittal affected the initial proposed no significant hazards consideration determination.

2.0 BACKGROUND

Compliance with Appendix J provides assurance that the primary containment, including those systems and components which penetrate the primary containment, do not exceed the allowable leakage rate values specified in the TSs and bases. The allowable leakage rate is determined so that the leakage assumed in the safety analyses is not exceeded.

On February 4, 1992, the NRC published a notice in the Federal Register (57 FR 4166) discussing a planned initiative to begin eliminating requirements marginal to safety which impose a significant regulatory burden. 10 CFR Part 50, Appendix J, "Primary Containment Leakage Testing for Water-Cooled

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Power Reactors," was considered for this initiative and the staff undertook a study of possible changes to this regulation. The study examined the previous performance history of domestic containments and examined the effect on risk of a revision to the requirements of Appendix J. The results of this study are reported in NUREG-1493, "Performance-Based Containment Leak-Test Program."

Based on the results of this study, the staff developed a performance-based approach to containment leakage rate testing. On September 12, 1995, the NRC approved issuance of this revision to 10 CFR Part 50, Appendix J, which was subsequently published in the Federal Register on September 26, 1995, and became effective on October 26, 1995. The revision added Option B "Performance-Based Requirements" to Appendix J to allow licensees to voluntarily replace the prescriptive testing requirements of Appendix J with testing requirements based on both overall and individual component leakage rate performance.

Regulatory Guide 1.163, "Performance-Based Containment Leak Test Program," September 1995 was developed as a method acceptable to the NRC staff for implementing Option B. This regulatory guide states that the Nuclear Energy Institute (NEI) guidance document 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," provides methods acceptable to the NRC staff for complying with Option B with four exceptions which are described therein.

Option B requires that the regulatory guide or other implementation document used by a licensee to develop a performance-based leakage testing program must be included, by general reference, in the plant TSs.

Regulatory Guide 1.163 specifies an extension in Type A test frequency to at least one test in 10 years based upon two consecutive successful tests. Type B tests may be extended up to a maximum of 10 years based upon completion of two consecutive successful tests and Type C tests may be extended up to 5 years based on two consecutive successful tests.

By letter dated October 20, 1995, NEI proposed TSs for implementing Option B. After some discussion, the staff and NEI agreed on a set of model TSs that were transmitted to NEI in a letter dated November 2, 1995. These TSs are to serve as a model for licensees to develop plant-specific TSs in preparing amendment requests to implement Option B.

In order for a licensee to determine the performance of each component, a licensee must establish an administrative leakage limit. The administrative limit is selected to be indicative of the potential onset of component degradation. Although these limits are subject to NRC inspection to assure that they are selected in a reasonable manner, they are not TS requirements. Failure to meet an administrative limit requires the licensee to return to the minimum value of the test interval.

Option B requires that the licensee maintain records to show that the criteria for Type A, B, and C tests have been met. In addition, the licensee must

maintain comparisons of the performance of the overall containment system and the individual components to show that the test intervals are adequate. These records are subject to NRC inspection.

3.0 EVALUATION

The licensee's December 6, 1995, letter to the NRC proposes to revise Section 1, "Definitions;" Section 3/4.6.1.1, "Containment Integrity;" Section 3/4.6.1.2, "Containment Leakage;" Section 3/4.6.1.3, "Containment Air Locks;" Section 3/4.6.1.7, "Containment Purge Ventilation System;" and associated Indexes and Bases. With minor variations the proposal would change all the affected surveillances to require testing in accordance with 10 CFR Part 50, Appendix J, Option B, and Regulatory Guide 1.163, dated September 1995. The primary impact is to allow use of 10 CFR Part 50, Appendix J, Option B, for primary containment leakage rate testing.

Option B permits a licensee to choose Type A; or Type B and C; or Type A, B, and C testing to be done on a performance basis. The licensee has elected to perform Type A, B, and C testing on a performance basis. The licensee has committed to performance of primary containment leakage rate testing in accordance with the guidelines contained in Regulatory Guide 1.163. TSs meeting the intent of those transmitted to NEI in a letter dated November 2, 1995, as noted below, were also proposed.

The licensee's proposed TSs differed in format from the model TSs developed by the NRC staff in cooperation with NEI. However, all the essential elements were included. The peak calculated containment internal pressure for the design-basis loss-of-coolant accident, P_a, is specified in the TSs. The maximum allowable primary containment leakage rate, L_a, at P_a, is also specified. The surveillance requirements for containment leakage rate testing and containment air lock testing have been revised to be performed in accordance with 10 CFR Part 50, Appendix J, Option B, and Regulatory Guide 1.163, dated September 1995.

The licensee also proposed to delete references to P_t, L_t, L_c and L_o from TS 3/4.6.1.2., and its associated Bases, which were no longer appropriate with the change to Option B. The staff agrees that the deleted surveillances are unnecessary or inconsistent with Option B.

Based on the above the licensee's proposed changes implementing Option B of Appendix J are acceptable.

As part of its submittal, the licensee also proposed to revise TS 4.6.1.7.3 regarding the testing of valves with resilient seal material. However, the scope of 10 CFR Part 50, Appendix J, did not include changes to testing of such valves and the licensee's submittal did not include sufficient information for the staff to evaluate the proposed change independently of the others. Consequently, approval is not granted in this amendment. The staff would consider this proposed change in a future submittal accompanied by a supporting safety analysis.

In its March 28, 1996, letter, the license revised its proposed change to TS 4.6.1.2 to indicate that the minimum interval between Type A tests is the nominal refueling outage. The staff has that proposed change under review and will address it in a future amendment.

In addition to the above, the licensee proposed a change to surveillance requirement 4.6.1.1.a to add a statement that the valves that are open under administrative controls are not required to meet the surveillance requirement during the time the valves are open. This surveillance currently requires verifying that all penetrations not capable of being closed by operable containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions, except as provided in Table 3.6-1 of Specification 3.6.3. This surveillance helps to ensure that post-accident leakage of radioactive fluids or gases outside of the containment boundary is within design limits. This surveillance does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those containment isolation valves capable of being mispositioned are in the correct position. Valves that are open under administrative control are obviously already known to be open. The administrative controls are considered sufficient to assure that the valves will be closed as conditions require. Therefore, the proposed change is acceptable.

4.0 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.

5.0 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 (61 FR 7547). 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.

6.0 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: John Hickman
Richard Lobel

Date: April 4, 1996