



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

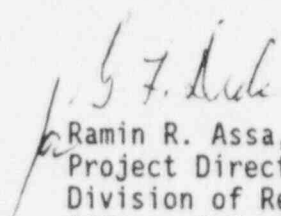
1. The Nuclear Regulatory Commission (the Commission) has found that:
  - A. The application for amendment by Commonwealth Edison Company (the licensee) dated December 6, 1995, as supplemented February 27, 1996, and March 28, 1996, complies with the standards and requirements of the Atomic Energy Act 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:

(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



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

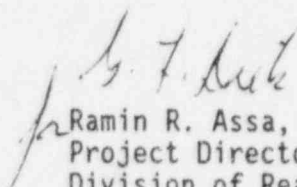
1. The Nuclear Regulatory Commission (the Commission) has found that:
  - A. The application for amendment by Commonwealth Edison Company (the licensee) dated December 6, 1995, as supplemented February 27, 1996, and March 28, 1996, complies with the standards and requirements of the Atomic Energy Act 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 of its issuance.

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

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

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1-4  
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3/4 6-2  
3/4 6-3  
3/4 6-4  
3/4 6-5  
3/4 6-12  
B 3/4 6-1

Insert Pages

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1-3  
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3/4 6-3  
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## DEFINITIONS

### $\bar{E}$ - 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.



## DEFINITIONS

### 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.

$P_0$

1.20.a  $P_0$  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

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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 SYSTEMS

### CONTAINMENT LEAKAGE

#### LIMITING CONDITION FOR OPERATION

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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  $200^\circ\text{F}$ .

#### SURVEILLANCE REQUIREMENTS

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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.

## CONTAINMENT SYSTEMS

### SURVEILLANCE REQUIREMENTS (Continued)

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- 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 SYSTEMS

### CONTAINMENT AIR LOCKS

#### LIMITING CONDITION FOR OPERATION

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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.

## CONTAINMENT SYSTEMS

### SURVEILLANCE REQUIREMENTS

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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.

## CONTAINMENT SYSTEMS

### SURVEILLANCE REQUIREMENTS

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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  $0.05 L_g$  when pressurized to at least  $P_g$ , 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  $0.01 L_g$  in accordance with Regulatory Guide 1.163, September 1995, and 10 CFR 50, Appendix J, Option B.



## 3/4.6 CONTAINMENT SYSTEMS

### BASES

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#### 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 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.