January 15, 1998

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

SUBJECT: ISSUANCE OF AMENDMENTS (TAC NOS. M97991, M97992, M97993 AND M97994)

Dear Mr. Kingsley:

The U.S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 96 to Facility Operating License No. NPF-37 and Amendment No. 96 to Facility Operating License No. NPF-66 for the Byron Station, Unit Nos. 1 and 2, respectively, and Amendment No. 87 to Facility Operating License No. NPF-72 and Amendment No. 87 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 February 18, 1997, as supplemented by letter dated September 22, 1997.

The amendments change the Technical Specification requirements for steam generator water level to support steam generator replacement at Byron, Unit 1, and Braidwood, Unit 1.

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

Sincerely,

ORIGINAL SIGNED BY:

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

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

Enclosures:
1. Amendment No. 96 to NPF-37
2. Amendment No. 96 to NPF-66
3. Amendment No. 87 to NPF-72
4. Amendment No. 87 to NPF-77
5. Safety Evaluation

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cc w/encls: see next page

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* by memo dated 11/24/97

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

January 15, 1998

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Nuclear Generation Group
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Sincerely,

George F. Dick Jr.

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

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

Enclosures:
1. Amendment No. 96 to NPF-37
2. Amendment No. 96 to NPF-66
3. Amendment No. 87 to NPF-72
4. Amendment No. 87 to NPF-77
5. Safety Evaluation

cc w/encls: see next page

cc:

Mr. William P. Poirier, Director
Westinghouse Electric Corporation
Energy Systems Business Unit
Post Office Box 355, Bay 236 W.
Pittsburgh, Pennsylvania 15230

Joseph Gallo
Gallo & Ross
1250 Eye St., N.W., Suite 302
Washington, DC 20005

Michael I. Miller, Esquire
Sidley and Austin
One First National Plaza
Chicago, Illinois 60603

Howard A. Learner
Environmental law and Policy
Center of the Midwest
203 N. LaSalle St. Suite 1390
Chicago, Illinois 60601

U.S. Nuclear Regulatory Commission
Byron Resident Inspectors Office
4448 N. German Church Road
Byron, Illinois 61010-9750

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
801 Warrenville Road
Lisle, Illinois 60532-4351

Ms. Lorraine Creek
RR 1, Box 182
Manteno, Illinois 60950

Chairman, Ogle County Board
Post Office Box 357
Oregon, Illinois 61061

Mrs. Phillip B. Johnson
1907 Stratford Lane
Rockford, Illinois 61107

George L. Edgar
Morgan, Lewis and Bochius
1800 M Street, N.W.
Washington, DC 20036

Attorney General
500 S. Second Street
Springfield, Illinois 62701

Illinois Department of Nuclear Safety
Office of Nuclear Facility Safety
1035 Outer Park Drive
Springfield, Illinois 62704

Commonwealth Edison Company
Byron Station Manager
4450 N. German Church Road
Byron, Illinois 61010-9794

Commonwealth Edison Company
Site Vice President - Byron
4450 N. German Church Road
Byron, Illinois 61010-9794

U.S. Nuclear Regulatory Commission
Braidwood Resident Inspectors Office
RR 1, Box 79
Braceville, Illinois 60407

Mr. Ron Stephens
Illinois Emergency Services
and Disaster Agency
110 E. Adams Street
Springfield, Illinois 62706

Chairman
Will County Board of Supervisors
Will County Board Courthouse
Joliet, Illinois 60434

Commonwealth Edison Company
Braidwood Station Manager
RR 1, Box 84
Braceville, Illinois 60407

Ms. Bridget Little Rorem
Appleseed Coordinator
117 N. Linden Street
Essex, Illinois 60935

Document Control Desk-Licensing
Commonwealth Edison Company
1400 Opus Place, Suite 400
Downers Grove, Illinois 60515

Commonwealth Edison Company
Site Vice President - Braidwood
RR 1, Box 84
Braceville, IL 60407

Mr. Michael J. Wallace
Nuclear Services Senior Vice President
Commonwealth Edison Company
Executive Towers West III
1400 Opus Place, Suite 900
Downers Grove, IL 60515

Mr. Gene H. Stanley
PWR's Vice President
Commonwealth Edison Company
Executive Towers West III
1400 Opus Place, Suite 900
Downers Grove, IL 60515

Mr. Steve Perry
BWR's Vice President
Commonwealth Edison Company
Executive Towers West III
1400 Opus Place, Suite 900
Downers Grove, IL 60515

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

Ms. Irene Johnson, Licensing Director
Nuclear Regulatory Services
Commonwealth Edison Company
Executive Towers West III
1400 Opus Place, Suite 500
Downers Grove, IL 60515

Commonwealth Edison Company
Reg. Assurance Supervisor - Braidwood
RR 1, Box 79
Braceville, Illinois 60407

Commonwealth Edison Company
Reg. Assurance Supervisor - Byron
4450 N. German Church Road
Byron, Illinois 61010-9794

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. 96
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 February 18, 1997, as supplemented by letter dated September 22, 1997, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 96 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

George F. Dick /s/

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

Attachment:
Changes to the Technical Specifications

Date of Issuance: January 15, 1998

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-455

BYRON STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 96
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 February 18, 1997, as supplemented by letter dated September 22, 1997, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter 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. 96 and revised by Attachment 2 to NPF-66, and the Environmental Protection Plan contained in Appendix B, both of which were attached to License No. NPF-37, dated February 14, 1985, are hereby incorporated into this license. Attachment 2 contains a revision to Appendix A which is hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

George F. Dick, Jr.

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

Attachment:
Changes to the Technical Specifications

Date of Issuance: January 15, 1998

ATTACHMENT TO LICENSE AMENDMENT NOS. 96 AND 96

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. Pages indicated by an asterisk (*) are provided for convenience only.

Remove Pages	Insert Pages
2-5	2-5
3/4 3-25	3/4 3-25
3/4 3-26	3/4 3-26
*3/4 4-1	*3/4 4-1
3/4 4-2	3/4 4-2
*3/4 4-3	*3/4 4-3
3/4 4-4	3/4 4-4
3/4 4-5	3/4 4-5
*3/4 4-6	*3/4 4-6

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINTS	ALLOWABLE VALUE
12. Reactor Coolant Flow-Low	≥90% of loop minimum measured flow*	≥89.3% of loop minimum measured flow*
13. Steam Generator Water Level Low-Low		
a. Unit 1	≥33.0% (prior to cycle 9) ≥18.0% (cycle 9 and after) of narrow range instrument span	≥31.0% (prior to cycle 9) ≥16.1% (cycle 9 and after) of narrow range instrument span
b. Unit 2	≥36.3% of narrow range instrument span	≥34.8% of narrow range instrument span
14. Undervoltage - Reactor Coolant Pumps	≥5268 volts - each bus	≥4920 volts - each bus
15. Underfrequency - Reactor Coolant Pumps	≥57.0 Hz	≥56.08 Hz
16. Turbine Trip		
a. Emergency Trip Header Pressure	≥1000 psig	≥815 psig
b. Turbine Throttle Valve Closure	≥1% open	≥1% open
17. Safety Injection Input from ESF	N.A.	N.A.
18. Reactor Coolant Pump Breaker Position Trip	N.A.	N.A.

*Minimum measured flow = 92,850 gpm

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
4. Steam Line Isolation		
a. Manual Initiation	N.A.	N.A.
b. Automaticc Actuation Logic and Actuation Relays	N.A.	N.A.
c. Containment Pressure-High-2	≤8.2 psig	≤9.4 psig
d. Steam Line Pressure-Low (Above P-11)	≥640 psig*	≥614 psig*
e. Steam Line Pressure Negative Rate-High (Below P-11)	≤100 psi**	≤165.3 psi*
5. Turbine Trip and Feedwater Isolation		
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
b. Steam Generator Water Level-High-High (P-14)		
1) Unit 1	≤81.4% (prior to cycle 9) ≤88.0% (cycle 9 and after) of narrow range instrument span	≤83.4% (prior to cycle 9) ≤89.9% (cycle 9 and after) of narrow range instrument span
2) Unit 2	≤80.8% of narrow range instrument span	≤82.8% of narrow range instrument span

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
5. Turbine Trip and Feedwater Isolation (continued)		
c. Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.	
6. Auxiliary Feedwater		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Steam Generator Water Level-Low-Low-Start Motor-Driven Pump and Diesel-Driven Pump		
1) Unit 1	≥33.0% (prior to cycle 9) ≥18.0% (cycle 9 and after) of narrow range instrument span	≥31.0% (prior to cycle 9) ≥16.1% (cycle 9 and after) of narrow range instrument span
2) Unit 2	≥36.3% of narrow range instrument span	≥34.8% of narrow range instrument span
d. Undervoltage-RCP Bus-Start Motor Driven Pump and Diesel-Driven Pump	≥5268 volts	≥4920 volts
e. Safety Injection-Start Motor-Driven Pump and Diesel-Driven Pump	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.	

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

STARTUP AND POWER OPERATION

LIMITING CONDITION FOR OPERATIONS

3.4.1.1 All reactor coolant loops shall be in operation.

APPLICABILITY: MODES 1 and 2.*

ACTION:

With less than the above required reactor coolant loops in operation, be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.1.1 The above required reactor coolant loops shall be verified in operation and circulating reactor coolant at least once per 12 hours.

*See Special Test Exceptions Specification 3.10.4.

LIMITING CONDITION FOR OPERATION

3.4.1.2 At least two of the reactor coolant loops listed below shall be OPERABLE with two reactor coolant loops in operation when the Reactor Trip System breakers are closed and one reactor coolant loop in operation when the Reactor Trip System breakers are open:*

a. Reactor Coolant Loop A and its associated steam generator and reactor coolant pump,
b. Reactor Coolant Loop B and its associated steam generator and reactor coolant pump,
c. Reactor Coolant Loop C and its associated steam generator and reactor coolant pump, and
d. Reactor Coolant Loop D and its associated steam generator and reactor coolant pump.

APPLICABILITY: MODE 3.**

ACTION:

a. With less than the above required reactor coolant loops OPERABLE, restore the required loops to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.
b. With only one reactor coolant loop in operation and the Reactor Trip System breakers in the closed position, within 1 hour open the Reactor Trip System breakers.
c. With no reactor coolant loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required reactor coolant loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.2.1 At least the above required reactor coolant pumps, if not in operation, shall be determined OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.2.2 The required steam generators shall be determined OPERABLE by verifying secondary side narrow range water level to be greater than or equal to 18% (41% for Unit 1 prior to cycle 9) at least once per 12 hours.

4.4.1.2.3 The required coolant loops shall be verified in operation and circulating reactor coolant at least once per 12 hours.

*All Reactor Coolant pumps may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.
**See Special Test Exceptions Specification 3.10.4.

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.1.3 At least two of the loops listed below shall be OPERABLE and at least one of these loops shall be in operation:*

a. Reactor Coolant Loop A and its associated steam generator and reactor coolant pump,**

b. Reactor Coolant Loop B and its associated steam generator and reactor coolant pump,**

c. Reactor Coolant Loop C and its associated steam generator and reactor coolant pump,**

d. Reactor Coolant Loop D and its associated steam generator and reactor coolant pump,**

e. RHR Loop A, and

f. RHR Loop B.

APPLICABILITY: MODE 4.

ACTION:

a. With less than the above required reactor coolant and/or RHR loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible; if the remaining OPERABLE loop is an RHR loop, be in COLD SHUTDOWN within 24 hours.

b. With no reactor coolant or RHR loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required coolant loop to operation.

---

*All reactor coolant pumps and RHR pumps may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

**A reactor coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to 350°F unless the secondary water temperature of each steam generator is less than 50°F above each of the Reactor Coolant System cold leg temperatures.

SURVEILLANCE REQUIREMENTS

4.4.1.3.1 The required reactor coolant pump(s) and/or RHR pumps, if not in operation, shall be determined OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.3.2 The required steam generator(s) shall be determined OPERABLE by verifying secondary side narrow range water level to be greater than or equal to 18% (41% for Unit 1 prior to cycle 9) at least once per 12 hours.

4.4.1.3.3 At least one reactor coolant or RHR loop shall be verified in operation and circulating reactor coolant at least once per 12 hours.

COLD SHUTDOWN - LOOPS FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.1 At least one residual heat removal (RHR) loop shall be OPERABLE and in operation*, and either:

a. One additional RHR loop shall be OPERABLE#, or

b. The secondary side narrow range water level of at least two steam generators shall be greater than 18% (41% for Unit 1 prior to cycle 9).

APPLICABILITY: MODE 5 with reactor coolant loops filled##.

ACTION:

a. With one of the RHR loops inoperable and with less than the required steam generator level, immediately initiate corrective action to return the inoperable RHR loop to OPERABLE status or restore the required steam generator level as soon as possible.

b. With no RHR loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.4.1.1 The secondary side water level of at least two steam generators when required shall be determined to be within limits at least once per 12 hours.

4.4.1.4.1.2 At least one RHR loop shall be determined to be in operation and circulating reactor coolant at least once per 12 hours.

---

*The RHR pump may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

#One RHR loop may be inoperable for up to 2 hours for surveillance testing provided the other RHR loop is OPERABLE and in operation.

##A reactor coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to 350°F unless the secondary water temperature of each steam generator is less than 50°F above each of the Reactor Coolant System cold leg temperatures.

COLD SHUTDOWN - LOOPS NOT FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.2 Two residual heat removal (RHR) loops shall be OPERABLE* and at least one RHR loop shall be in operation.**

APPLICABILITY: MODE 5 with reactor coolant loops not filled.

ACTION:

a. With less than the above required RHR loops OPERABLE, immediately initiate corrective action to return the required RHR loops to OPERABLE status as soon as possible.

b. With no RHR loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.4.2 At least one RHR loop shall be determined to be in operation and circulating reactor coolant at least once per 12 hours.

*One RHR loop may be inoperable for up to 2 hours for surveillance testing provided the other RHR loop is OPERABLE and in operation.

**The RHR pump may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

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. 87
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 February 18, 1997, as supplemented by letter dated September 22, 1997, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 87 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

George F. Dick Jr.

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 15, 1998

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-457

BRAIDWOOD STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 87
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 February 18, 1997, as supplemented by letter dated September 22, 1997, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter 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. 87 and the Environmental Protection Plan contained in Appendix B, both of which were attached to License No. NPF-72, dated July 2, 1987, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

George F. Dick

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

Attachment:
Changes to the Technical Specifications

Date of Issuance: January 15, 1998

ATTACHMENT TO LICENSE AMENDMENT NOS. 87 AND 87

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. Pages indicated by an asterisk (*) are provided for convenience only.

Remove Pages	Insert Pages
2-5	2-5
3/4 3-25	3/4 3-25
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3/4 4-4	3/4 4-4
3/4 4-5	3/4 4-5
*3/4 4-6	*3/4 4-6

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
12. Reactor Coolant Flow-Low	≥90% of loop minimum measured flow*	≥89.3% of loop minimum measured flow*
13. Steam Generator Water Level Low-Low		
a. Unit 1	≥33.0% (prior to cycle 8) ≥18.0% (cycle 8 and after) of narrow range instrument span	≥31.0% (prior to cycle 8) ≥16.1% (cycle 8 and after) of narrow range instrument span
b. Unit 2	≥36.3% of narrow range instrument span	≥34.8% of narrow range instrument span
14. Undervoltage - Reactor Coolant Pumps	≥5268 volts - each bus	≥4920 volts - each bus
15. Underfrequency - Reactor Coolant Pumps	≥57.0 Hz	≥56.08 Hz
16. Turbine Trip		
a. Emergency Trip Header Pressure	≥1000 psig	≥815 psig
b. Turbine Throttle Valve Closure	≥1% open	≥1% open
17. Safety Injection Input from ESF	N.A.	N.A.
18. Reactor Coolant Pump Breaker Position Trip	N.A.	N.A.

*Minimum measured flow = 92,850 gpm

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
4. Steam Line Isolation		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Containment Pressure-High-2	≤8.2 psig	≤9.4 psig
d. Steam Line Pressure-Low (Above P-11)	≥640 psig*	≥614 psig*
e. Steam Line Pressure Negative Rate-High (Below P-11)	≤100 psi**	≤165.3 psi**
5. Turbine Trip and Feedwater Isolation		
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
b. Steam Generator Water Level-High-High (P-14)		
1) Unit 1	≤81.4% (prior to cycle 8) ≤88.0% (cycle 8 and after) of narrow range instrument span	≤83.4% (prior to cycle 8) ≤89.9% (cycle 8 and after) of narrow range instrument span
2) Unit 2	≤80.8% of narrow range instrument span	≤82.8% of narrow range instrument span

TABLE 3.3-4 (Continued)
ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT		TRIP SETPOINT	ALLOWABLE VALUE
5. Turbine Trip and Feedwater Isolation (continued)			
c. Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.		
6. Auxiliary Feedwater			
a. Manual Initiation		N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays		N.A.	N.A.
c. Steam Generator Water Level-Low-Low-Start Motor-Driven Pump and Diesel-Driven Pump			
1) Unit 1		≥33.0% (prior to cycle 8) ≥18.0% (cycle 8 and after) of narrow range instrument span	≥31.0% (prior to cycle 8) ≥16.1% (cycle 8 and after) of narrow range instrument span
2) Unit 2		≥36.3% of narrow range instrument span	≥34.8% of narrow range instrument span
d. Undervoltage-RCP Bus-Start Motor Driven Pump and Diesel-Driven Pump		≥5268 volts	≥4920 volts
e. Safety Injection-Start Motor-Driven Pump and Diesel-Driven Pump	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.		

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

STARTUP AND POWER OPERATION

LIMITING CONDITION FOR OPERATIONS

3.4.1.1 All reactor coolant loops shall be in operation.

APPLICABILITY: MODES 1 and 2.*

ACTION:

With less than the above required reactor coolant loops in operation, be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.1.1 The above required reactor coolant loops shall be verified in operation and circulating reactor coolant at least once per 12 hours.

*See Special Test Exceptions Specification 3.10.4.

LIMITING CONDITION FOR OPERATION

3.4.1.2 At least two of the reactor coolant loops listed below shall be OPERABLE with two reactor coolant loops in operation when the Reactor Trip System breakers are closed and one reactor coolant loop in operation when the Reactor Trip System breakers are open:*

a. Reactor Coolant Loop A and its associated steam generator and reactor coolant pump,
b. Reactor Coolant Loop B and its associated steam generator and reactor coolant pump,
c. Reactor Coolant Loop C and its associated steam generator and reactor coolant pump, and
d. Reactor Coolant Loop D and its associated steam generator and reactor coolant pump.

APPLICABILITY: MODE 3.**

ACTION:

a. With less than the above required reactor coolant loops OPERABLE, restore the required loops to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.
b. With only one reactor coolant loop in operation and the Reactor Trip System breakers in the closed position, within 1 hour open the Reactor Trip System breakers.
c. With no reactor coolant loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required reactor coolant loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.2.1 At least the above required reactor coolant pumps, if not in operation, shall be determined OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.2.2 The required steam generators shall be determined OPERABLE by verifying secondary side narrow range water level to be greater than or equal to 18% (41% for Unit 1 prior to cycle 8) at least once per 12 hours.

4.4.1.2.3 The required coolant loops shall be verified in operation and circulating reactor coolant at least once per 12 hours.

*All Reactor Coolant pumps may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

**See Special Test Exceptions Specification 3.10.4.

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.1.3 At least two of the loops listed below shall be OPERABLE and at least one of these loops shall be in operation:*

a. Reactor Coolant Loop A and its associated steam generator and reactor coolant pump,**

b. Reactor Coolant Loop B and its associated steam generator and reactor coolant pump,**

c. Reactor Coolant Loop C and its associated steam generator and reactor coolant pump,**

d. Reactor Coolant Loop D and its associated steam generator and reactor coolant pump,**

e. RHR Loop A, and

f. RHR Loop B.

APPLICABILITY: MODE 4.

ACTION:

a. With less than the above required reactor coolant and/or RHR loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible; if the remaining OPERABLE loop is an RHR loop, be in COLD SHUTDOWN within 24 hours.

b. With no reactor coolant or RHR loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required coolant loop to operation.

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*All reactor coolant pumps and RHR pumps may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

**A reactor coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to 350°F unless the secondary water temperature of each steam generator is less than 50°F above each of the Reactor Coolant System cold leg temperatures.

SURVEILLANCE REQUIREMENTS

4.4.1.3.1 The required reactor coolant pump(s) and/or RHR pumps, if not in operation, shall be determined OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.3.2 The required steam generator(s) shall be determined OPERABLE by verifying secondary side narrow range water level to be greater than or equal to 18% (41% for Unit 1 prior to cycle 8) at least once per 12 hours.

4.4.1.3.3 At least one reactor coolant or RHR loop shall be verified in operation and circulating reactor coolant at least once per 12 hours.

COLD SHUTDOWN - LOOPS FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.1 At least one residual heat removal (RHR) loop shall be OPERABLE and in operation*, and either:

a. One additional RHR loop shall be OPERABLE#, or

b. The secondary side narrow range water level of at least two steam generators shall be greater than 18% (41% for Unit 1 prior to cycle 8).

APPLICABILITY: MODE 5 with reactor coolant loops filled##.

ACTION:

a. With one of the RHR loops inoperable and with less than the required steam generator level, immediately initiate corrective action to return the inoperable RHR loop to OPERABLE status or restore the required steam generator level as soon as possible.

b. With no RHR loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.4.1.1 The secondary side water level of at least two steam generators when required shall be determined to be within limits at least once per 12 hours.

4.4.1.4.1.2 At least one RHR loop shall be determined to be in operation and circulating reactor coolant at least once per 12 hours.

*The RHR pump may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

#One RHR loop may be inoperable for up to 2 hours for surveillance testing provided the other RHR loop is OPERABLE and in operation.

##A reactor coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to 350°F unless the secondary water temperature of each steam generator is less than 50°F above each of the Reactor Coolant System cold leg temperatures.

REACTOR COOLANT SYSTEM

COLD SHUTDOWN - LOOPS NOT FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.2 Two residual heat removal (RHR) loops shall be OPERABLE* and at least one RHR loop shall be in operation.**

APPLICABILITY: MODE 5 with reactor coolant loops not filled.

ACTION:

a. With less than the above required RHR loops OPERABLE, immediately initiate corrective action to return the required RHR loops to OPERABLE status as soon as possible.

b. With no RHR loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.4.2 At least one RHR loop shall be determined to be in operation and circulating reactor coolant at least once per 12 hours.

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*One RHR loop may be inoperable for up to 2 hours for surveillance testing provided the other RHR loop is OPERABLE and in operation.

**The RHR pump may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 96 TO FACILITY OPERATING LICENSE NO. NPF-37,

AMENDMENT NO. 96 TO FACILITY OPERATING LICENSE NO. NPF-66,

AMENDMENT NO. 87 TO FACILITY OPERATING LICENSE NO. NPF-72,

AND AMENDMENT NO. 87 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 February 18, 1997, as supplemented by letter dated September 22, 1997, Commonwealth Edison Company (ComEd, or the licensee) proposed Technical Specification (TS) changes for Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, to support steam generator (SG) replacements at Byron, Unit 1, and Braidwood, Unit 1. The September 22, 1997, submittal provided additional clarifying information that did not change the initial proposed no significant hazards consideration determination.

The licensee will be replacing the original Westinghouse D4 SGs at Byron, Unit 1, and Braidwood, Unit 1, with Babcock & Wilcox International (BWI) SGs. Due to changes in the location of the SG level taps, the installation of the BWI SGs requires an increase of the SG water level operating range (i.e., the difference between the low-low and the high-high SG level setpoints in percent of narrow range span). Consequently, changes are necessary to the TS setpoints for SG water level reactor trip and engineered safety features actuation. These setpoints are found in TS 2.2.1 Table 2.2-1, TS 3.3.2 Table 3.3-4, TSSR 4.4.1.2.2, TSSR 4.4.1.3.2 and TS 3.4.1.4.1.6.

The TS setpoints and operating ranges for Byron, Unit 2, and Braidwood, Unit 2, which will continue to operate with the existing Westinghouse SGs, remain unchanged; however, due to the common Technical Specification pages being used for Byron, Units 1 and 2, and Braidwood, Units 1 and 2, these amendments will appear on the pages for both units.

2.0 EVALUATION

ComEd proposes to change the TS for Reactor Trip System Steam Generator Water Level Low-Low setpoint, the Engineered Safety Features Actuation System (ESFAS) Steam Generator Water Level Low-Low Auxiliary Feedwater (AFW) setpoint, and the ESFAS Steam Generator Water Level High-High Turbine Trip and Feedwater Isolation setpoint. ComEd also proposes to change the TS surveillance requirements for minimum water level in Modes 3, 4 and 5 (loops filled).

The current requirement for the Unit 1 low-low level setpoints is 33.0 percent of narrow range span (NRS) with an allowable value of 31.0 percent NRS. The current requirement for the Unit 1 high-high level setpoint is 81.4 percent NRS with an allowable value of 83.4 percent NRS. ComEd proposes to change these setpoints for Byron Unit 1 and Braidwood Unit 1 to 18.0 percent NRS with an allowable value of 16.1 percent NRS for the low-low setpoints and 88.0 percent NRS with an allowable value of 89.9 percent NRS for the high-high setpoint. While the narrow range span for the RSG (180 inches) has decreased as compared to the original Westinghouse Model D4 steam generators (OSG) (232 inches), the operating range for the replacement steam generators (RSG) (126 inches) is increased as compared to the OSG (112.3 inches). The licensee indicated that this increase in the operating range minimizes the possibility of inadvertent plant trips following load changes and feedwater transients.

The current surveillance requirement for the minimum SG water level in Modes 3, 4 and 5 (loops filled) is 41 percent NRS for Unit 1. ComEd proposes to change these requirements to 18 percent NRS.

The current SG level setpoints are based on the limiting accident analyses with Westinghouse steam generators. The limiting accidents for the low-low SG level reactor trip and AFW flow initiation setpoints are the Loss of Normal Feedwater and Feedwater Line Break. The limiting accident for the high-high SG level setpoint is the Feedwater System Malfunction, which results in an increase in feedwater flow to one or more steam generators. The intent of the surveillance requirement for a minimum SG inventory in Modes 3, 4 and 5 (loops filled) is to remove decay heat and is met by ensuring the SG tube bundle is completely covered.

The licensee determined the impact of the RSGs on the limiting low-low setpoint transients, the Loss of Normal Feedwater and Feedwater Line Break. The transients were analyzed with RELAP5/MOD2-B&W using the methodology approved by the staff in BAW-10169-A. The RELAP5 analysis incorporated a low-low setpoint of 0 percent NRS for the Feedwater Line Break and 10 percent NRS for the Loss of Normal Feedwater and demonstrated that all acceptance criteria (listed below) for each transient have been met.

Loss of Normal Feedwater

- Pressure in the reactor coolant and main steam systems did not exceed 110 percent of the design value;
- the minimum departure from nucleate boiling ratio (DNBR) remained above the 95/95 DNBR limit ; and
- the ultimate heat sink for decay heat removal was assured.

Feedwater Line Break

- Pressure in the reactor coolant and main steam systems did not exceed 110 percent of the design value;
- the ultimate heat sink for decay heat removal was assured;
- the core remained intact for effective cooling; and

- radiation doses remain bounded by those predicted for the steamline break event and, therefore, did not exceed a small fraction of the 10 CFR Part 100 exposure guidelines.

The licensee also determined the impact of the RSG on the limiting high-high setpoint transient, Feedwater System Malfunction resulting in increased feedwater flow. The transient was analyzed using the RELAP5/MOD2-B&W computer code. The licensee demonstrated that the acceptance criteria for the Feedwater Malfunction transient, listed below, were met with a high-high setpoint of 100 percent NRS. The licensee also demonstrated that the RSGs do not overfill.

Feedwater Malfunction

- Pressure in the reactor coolant and main steam systems did not exceed 110 percent of the design value; and

- the minimum DNBR remained above the 95/95 DNBR limit.

The TS low-low and high-high SG level setpoints and associated allowable values were calculated by the licensee using the approved methodology of WCAP-12583. Uncertainties in the setpoint value were determined based on this methodology to be approximately 15 percent NRS for low-low level (approximately 5 percent NRS for the Loss of Normal Feedwater) and 9 percent NRS for high-high level. The setpoint was conservatively chosen by the licensee as 18 percent NRS for low-low level and 88 percent NRS for high-high level. The setpoint allowance determined per WCAP-12583 is 1.9 percent, which yields a TS allowable value of 16.1 percent NRS for low-low level and 89.9 percent NRS for high-high level. The staff finds the setpoints and associated allowable value changes to be acceptable based on the accident analysis described above and the use of the approved WCAP-12583 methodology.

The licensee determined the impact of the RSGs on the surveillance requirement for a minimum inventory to remove decay heat in Modes 3, 4 and 5 (loops filled). The licensee stated that the intent can be met by assuring that the tube bundle is completely covered. The licensee determined that the Unit 1 SG tubes are covered when the SG water level is within the span of the narrow range level indication, which can be assured by specifying a surveillance requirement water level that is equal to or greater than the low-low level setpoint (18 percent NRS). The staff finds the proposed surveillance acceptable.

3.0 SUMMARY

The licensee proposed changes to the TS for Byron, Units 1 and 2, and Braidwood, Units 1 and 2, to reflect necessary changes to the low-low and high-high steam generator level setpoints. These changes are necessitated by the replacement of the original Westinghouse D4 steam generators with BWI steam generators and the subsequent decrease in narrow range span.

The licensee analyzed the limiting transients for both the low-low and high-high steam generator level setpoints using approved methodologies. The licensee demonstrated that the acceptance criteria are met for the Updated Final Safety Analysis Report, Chapter 15, transients that are impacted by the setpoint changes. Therefore, the staff finds the licensee's safety analysis to be an acceptable basis for setpoint determination.

Further, the staff concludes that the proposed TS low-low and high-high SG level reactor trip and engineered safety feature actuation setpoints and associated allowable values are consistent with the methodology approved by the staff in WCAP-12583. The staff, therefore, finds the TS changes supporting the SG replacement at Byron, Unit 1, and Braidwood, Unit 1, and the corresponding TS changes for Byron, Unit 2, and Braidwood, Unit 2, to be 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 (62 FR 11491). 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 Contributors: S. Bailey
S. Brewer
S. Rhow

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