December 18, 1987

Docket No. STN 50-457

Distribution:

Docket File	NRC & Local PDR	PDIII-2 r/f
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IBailey		

Mr. L. D. Butterfield, Jr.
Director of Nuclear Licensing
Commonwealth Edison Company
P.O. Box 767

Chicago, Illinois 60690

Dear Mr. Butterfield:

SUBJECT: ISSUANCE OF FACILITY OPERATING LICENSE NPF-75 - BRAIDWOOD STATION UNIT 2

The U. S. Nuclear Regulatory Commission (NRC) has issued the enclosed Facility Operating License NPF-75, together with the Environmental Protection Plan for Braidwood Station, Unit 2. License No. NPF-75 authorizes operation of Braidwood Station, Unit 2 at reactor power levels not in excess of 3411 megawatts thermal (100 percent rated power). Pending further Commission approval, operation is restricted to power levels not in excess of five percent rated power (170 megawatts thermal).

Enclosed is a copy of a related notice, the original of which has been forwarded to the Office of the Federal Register for publication.

Two copies of Amendment No. 5 to Indemnity Agreement No. B-102 which covers the activities authorized under License No. NPF-75 are also enclosed. Please return one signed copy to this office.

Safety Evaluation Report Supplement No. 5 (SSER 5) was prepared in support of issuing the enclosed license. Enclosed is a pre-printed copy of SSER 5. Twenty (20) bound copies of SSER 5 will be sent to you in the near future.

Sincerely,

/S/

Dennis M. Crutchfield, Director
Division of Reactor Projects - III,
IV, V, and Special Projects

Enclosures:
1. Facility Operating License NPF-75
2. Federal Register Notice
3. Amendment No. 5 to Indemnity Agreement No. B-102
4. Supplement No. 5 to the Safety Evaluation Report

cc: w/enclosures:
See next page

(*See Previous Concurrence)

PDIII-2	PDIII-2	PDIII-2	DRPS	DRPS
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See next page

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-457

BRAIDWOOD STATION, UNIT 2

FACILITY OPERATING LICENSE

License No. NPF-75

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

   A. The application for a license filed by Commonwealth Edison Company (the licensee) complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I, and all required notifications to other agencies or bodies have been duly made;

   B. Construction of Braidwood Station, Unit 2 (the facility) has been completed in conformity with Construction Permit No. CPPR-133 and the application, as amended, the provisions of the Act and the regulations of the Commission;

   C. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the regulations of the Commission (except as exempted from compliance in Section 2.D. below);

   D. There is reasonable assurance: (ii) that the activities authorized by this operating license 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 set forth in 10 CFR Chapter I (except as exempted from compliance in Section 2.D. below);

   E. Commonwealth Edison Company is technically qualified to engage in the activities authorized by this license in accordance with the Commission's regulations set forth in 10 CFR Chapter I;

   F. Commonwealth Edison Company has satisfied the applicable provisions of 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements," of the Commission's regulations;

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

H. After weighing the environmental, economic, technical, and other benefits of the facility against environmental and other costs and considering available alternatives, the issuance of Facility Operating License NPF-75, subject to the conditions for protection of the environment set forth in the Environmental Protection Plan attached as Appendix B to License No. NPF-72, issued July 2, 1987, is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied; and

I. The receipt, possession, and use of source, byproduct and special nuclear material as authorized by this license will be in accordance with the Commission's regulations in 10 CFR Parts 30, 40 and 70.

2. Based on the foregoing findings regarding this facility, Facility Operating License No. NPF-75 is hereby issued to Commonwealth Edison Company (the licensee) to read as follows:

A. This license applies to Braidwood Station, Unit 2, a pressurized water reactor, and associated equipment (the facility) owned by Commonwealth Edison Company. The facility is located in north-eastern Illinois, 3 miles southwest of the Kankakee River, 20 miles south-southwest of the town of Joliet, and 60 miles southwest of Chicago, Illinois. The facility is within Reed Township, Will County, Illinois and is described in the Byron/Braidwood Stations' Final Safety Analysis Report, as supplemented and amended, and in the Environmental Report, as supplemented and amended.

B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses:

(1) Commonwealth Edison Company (CECo), pursuant to Section 103 of the Act and 10 CFR Part 50, to possess, use and operate the facility at the above designated location in Will County, Illinois, in accordance with the procedures and limitations set forth in this license;

(2) CECo, pursuant to the Act and 10 CFR Part 70, to receive, possess and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;

(3) CECo, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron

sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) CECo, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) CECo, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at reactor core power levels not in excess of 3411 megawatts thermal (100 percent rated power) in accordance with the conditions specified herein and other items identified in Attachment 1 to this license. The items identified in Attachment 1 to this license shall be completed as specified. Attachment 1 is hereby incorporated into this license. Pending further Commission approval, this license is restricted to power levels not in excess of five percent of rated power (170 megawatts thermal).

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A 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. Attachment 2 contains revisions to Appendix A which are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Emergency Planning

In the event that the NRC finds that the lack of progress in completion of the procedures in the Federal Emergency Management Agency's final rule, 44 CFR Part 350, is an indication that a major substantive problem exists in achieving or maintaining an adequate state of emergency preparedness, the provisions of 10 CFR Section 50.54(s)(2) will apply.

(4) Initial Startup Test Program

Any changes to the Initial Test Program described in Section 14 of the FSAR made in accordance with the provisions of 10 CFR 50.59 shall be reported in accordance with 50.59(b) within one month of such change.

D. The facility requires an exemption from the requirements of Appendix J to 10 CFR Part 50, Paragraph III.D.2(b)(ii), the testing of containment air locks at times when containment integrity is not required (SER Section 6.2.6). This exemption is authorized by law, will not present an undue risk to the public health and safety, and is consistent with the common defense and security. This exemption is hereby granted. The special circumstances regarding this exemption are identified in the referenced section of the Safety Evaluation Report and the supplements thereto. This exemption is granted pursuant to 10 CFR 50.12. With this exemption, the facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission.

An exemption was previously granted pursuant to 10 CFR 70.24. The exemption was granted with NRC materials license No. SNM-1938, issued October 8, 1985, and relieved the licensee from the requirement of having a criticality alarm system. Therefore, the licensee is exempted from the criticality alarm system provision of 10 CFR 70.24 so far as this section applies to the storage of fuel assemblies held under this license.

E. The licensee shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report, as supplemented and amended, and as approved in the SER dated November 1983 and its supplements, subject to the following provision:

The licensee may make changes to the approved fire protection program without prior approval of the Commission, only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

F. The licensee shall fully implement and maintain in effect all provisions of the physical security, guard training and qualification, and safeguards contingency plans previously approved by the Commission and all amendments and revisions to such plans made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans, which contain Safeguards Information protected under 10 CFR 73.21, are entitled: "Braidwood Station Physical Security Plan, Security Personnel Training and Qualification Plan,* and Safeguards Contingency Plan*" with revisions submitted through May 27, 1986.

G. Except as otherwise provided in the Technical Specifications or Environmental Protection Plan, the licensee shall report any violations of the requirements contained in Section 2.C of this license in the following manner: initial notification shall be made within 24 hours to the NRC Operations Center via the Emergency Notification System with written followup within thirty days in accordance with the procedures described in 10 CFR 50.73(b), (c), and (e).

H. The licensee shall have and maintain financial protection of such type and in such amounts as the Commission shall require in accordance with Section 170 of the Atomic Energy Act of 1954, as amended, to cover public liability claims.

I. This license is effective as of the date of issuance and shall expire at midnight on December 18 , 2027.

FOR THE NUCLEAR REGULATORY COMMISSION

/s/

Thomas E. Murley, Director
Office of Nuclear Reactor Regulation

Attachments:
1. Work Items to be completed
2. Revisions to Appendix A - Technical Specifications (NUREG-1276)

Date of Issuance: December 18, 1987

Bill Olmstead concurred for OGC on 12/18/87

*The Security Personnel Training and Qualification Plan and the Safeguards Contingency Plan are Appendices to the Security Plan. As requested by CECo letter dated April 22, 1983, Revision 6 is to be considered "the initial formal submittal."

to OGC review SS 12/17/87

PDIII-2	PDIII-2	PTSB	PTSB	~~OGC~~	OGC
SSands/ww	LLuther	WLambe	IDinitz	~~EChan~~	EReis
12/15/87	12/15/87	12/15/87	12/15/87	12/16/87	/ /87
ECEB	EMTB	PDIII-2	AD:DRSP	D:DRSP	NRR
McCracken	CYCheng	DMuller	GHolahan	DCrutchfield	FMiraglia
12/15/87	12/15/87	12/15/87	12/18/87	12/18/8	12/18/87
NRR					
TMurley					
12/18/87					

ATTACHMENT 1

This attachment identifies specific items which must be completed to the Commission's satisfaction in accordance with the operational modes as identified below.

1. The preoperational testing exceptions identified in the December 16, 1987, letter from S. C. Hunsader to Thomas E. Murley and component demonstrations identified in the December 17, 1987, letter from S. C. Hunsader to Thomas E. Murley shall be completed in accordance with the scheduled commitments contained in those letters.

2. The 18 month surveillance activities for the Unit 2 2A Diesel Generator identified in the November 23, 1987 letter from S. C. Hunsader to Thomas E. Murley shall be accomplished as described in that letter.

3. The fire protection program work and compensatory measures identified in the December 16, 1987 letter from S. C. Hunsader to Thomas E. Murley shall be accomplished as described in that letter.

4. The 18 month testing of the control room ventilation system identified in the December 11, 1987 letter from S. C. Hunsader to Thomas E. Murley shall be accomplished as described in that letter.

ATTACHMENT 2 TO LICENSE NPF-75
REVISIONS TO NUREG-1276

Revise Appendix A, Technical Specifications (NUREG-1276), by removing the pages identified below and inserting the enclosed pages. Overleaf pages (*) have been provided to maintain document completeness.

REMOVE	INSERT
	3/4 6-1 (*)
3/4 6-2	3/4 6-2
	3/4 8-1 (*)
3/4 8-2	3/4 8-2
3/4 8-7	3/4 8-7
	3/4 8-8 (*)
3/4 8-9a	-
	3/4 8-10 (*)

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;

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

c. After each closing of each penetration subject to Type B testing, except the containment air locks, if opened following a Type A or B test, by leak rate testing the seal with gas at a pressure not less than P_a, 44.4 psig, and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Specification 4.6.1.2d. for all other Type B and C penetrations, the combined leakage rate is less than 0.60 L_a.

*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:

1) Less than or equal to L_a, 0.10% by weight of the containment air per 24 hours at P_a, 44.4 psig, or

2) Less than or equal to L_t, 0.07% by weight of the containment air per 24 hours for Unit 1 (0.07% by weight of the containment air per 24 hours for Unit 2) at P_t, 22.2 psig.

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 0.75 L_t, as applicable, 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 or less than 0.75 L_t, as applicable, 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 at the following test schedule and shall be determined in conformance with the criteria specified in Appendix J of 10 CFR Part 50 using the methods and provisions of ANSI N45.4-1972:

a. Three Type A tests (Overall Integrated Containment Leakage Rate) shall be conducted at 40 ± 10 month intervals during shutdown at a pressure not less than P_a, 44.4 psig, or P_t, 22.2 psig, during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection;

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.1.1 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. Each Class 1E 4160 volt bus capable of being powered from:

1) Either transformer of a given units normal System Auxiliary Transformer bank, and

2) Either transformer of the other units System Auxiliary Transformers bank, with

Each units System Auxiliary Transformer bank energized from an independent transmission circuit.

b. Two separate and independent diesel generators, each with:

1) A separate day tank containing a minimum volume of 450 gallons of fuel,

2) A separate Fuel Oil Storage System containing a minimum volume of 44,000 gallons of fuel, and

3) A separate fuel transfer pump.

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

ACTION:

a. With either an offsite circuit or diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Specification 4.8.1.1.1a within 1 hour and at least once per 8 hours thereafter and Specification 4.8.1.1.2.a.4 within 24 hours; restore at least two offsite circuits and two diesel generators to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one offsite circuit and one diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Specifications 4.8.1.1.1a within 1 hour and at least once per 8 hours thereafter and Specification 4.8.1.1.2.a.4 within 8 hours; restore at least one of the inoperable sources to OPERABLE status within 12 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore at least two offsite circuits and two diesel generators to OPERABLE status within 72 hours from the time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION (Continued)

c. With one diesel generator inoperable in addition to ACTION a. or b. above, verify that:

   1. All required systems, subsystems, trains, components and devices that depend on the remaining OPERABLE diesel generator as a source of emergency power are also OPERABLE, and

   2. When in MODE 1, 2, or 3, the diesel-driven auxiliary feedwater pump is OPERABLE and the other Unit's A Diesel Generator is OPERABLE, if the inoperable diesel generator is the emergency power supply for the motor-driven auxiliary feedwater pump.

   If these conditions are not satisfied within 2 hours be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

d. With two of the above required offsite A.C. circuits inoperable, demonstrate the OPERABILITY of two diesel generators by performing Specification 4.8.1.1.2a.4) within 8 hours, unless the diesel generators are already operating; restore at least one of the inoperable offsite sources to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours. With only one offsite source restored, restore at least two offsite circuits to OPERABLE status within 72 hours from time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

e. With two of the above required diesel generators inoperable, demonstrate the OPERABILITY of two offsite A.C. circuits by performing Specification 4.8.1.1.1a. within 1 hour and at least once per 8 hours thereafter; restore at least one of the inoperable diesel generators to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore at least two diesel generators to OPERABLE status within 72 hours from time of initial loss or be in least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

13) Verifying that the following diesel generator lockout features prevent diesel generator starting only when required:

a) Turning gear engaged, and
b) Emergency stop.

g. At least once per 10 years or after any modifications which could affect diesel generator interdependence by starting both diesel generators simultaneously, during shutdown, and verifying that both diesel generators accelerate to at least 600 rpm in less than or equal to 10 seconds;

h. At least once per 10 years by:

1) Draining each fuel oil storage tank, removing the accumulated sediment and cleaning the tank using a sodium hypochlorite solution, and

2) Performing a pressure test of those portions of the diesel fuel oil system designed to Section III, subsection ND of the ASME Code at a test pressure equal to 110 percent of the system design pressure.

i. At least once per 31 days by:

#1) Verifying the capability of crosstieing the Unit 2, A diesel generator to Bus 141 by independently performing the following:

a) Synchronizing the Unit 2, A diesel generator to Bus 241.
b) Closing breaker 1414, and.
c) Closing breaker 2414.

##2) Verifying the capability of crosstieing the Unit 1 A diesel generator to Bus 241 by independently performing the following:
a) Synchronizing the Unit 1, A diesel generator to Bus 141,
b) Closing breaker 1414, and
c) Closing breaker 2414.

j. At least once per 18 months by:

#1) Crosstieing the 2A diesel generator to Bus 141.
##2) Crosstieing the 1A diesel generator to Bus 241.

4.8.1.1.3 Reports - All diesel generator failures, valid or non-valid, shall be reported to the Commission pursuant to Specification 6.9.2. Reports of diesel generator failures shall include the information recommended in Regulatory Position C.3.b of Regulatory Guide 1.108, Revision 1, August 1977. If the number of failures in the last 100 valid tests (on a per nuclear unit basis) is greater than or equal to 7, the report shall be supplemented to include the additional information recommended in Regulatory Position C.3.b of Regulatory Guide 1.108, Revision 1, August 1977.

#Only required for Unit 1 operation in MODES 1, 2, or 3.
##Only required for Unit 2 operation in MODES 1, 2, or 3.

Table 4.8-1

DIESEL GENERATOR TEST SCHEDULE

NUMBER OF FAILURES IN LAST 20 VALID TESTS*	TEST FREQUENCY
≤ 1	At least once per 31 days
≥ 2	At least once per 7 days**

*Criteria for determining number of failures and number of valid tests shall be in accordance with Regulatory Position C.2.e of Regulatory Guide 1.108, Revision 1, August 1977, where the number of tests and failures is determined on a per diesel generator basis. For the purposes of this test schedule, only valid tests conducted after the completion of the preoperational test requirements of Regulatory Guide 1.108, Rev 1, Aug 1977, shall be included in the computation of the "last 20 valid tests."

**This test frequency shall be maintained until seven consecutive failure free demands have been performed and the number of failures in the last 20 valid demands has been reduced to one or less.

ELECTRICAL POWER SYSTEMS

3/4.8.2 D.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1* As a minimum the following D.C. electrical sources shall be OPERABLE:

a. 125-Volt D.C. Bus 111 fed from Battery 111 for Unit 1 (Bus 211 fed from Battery 211 for Unit 2), and its associated full capacity charger, and

b. 125-Volt D.C. Bus 112 fed from Battery 112 for Unit 1 (Bus 212 fed from Battery 212 for Unit 2), and its associated full capacity charger.

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

ACTION:

a. With one of the required battery banks and/or chargers inoperable, restore the inoperable battery bank and/or battery bus to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With the normal full capacity charger inoperable: 1) restore the affected battery and/or battery bus to operable status with the opposite units full capacity charger within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours, and 2) restore the normal full capacity charger 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

4.8.2.1.1 Each D.C. bus shall be determined OPERABLE and energized from its battery at least once per 7 days by verifying correct breaker alignment.

4.8.2.1.2 Each 125-volt battery bank and its associated charger shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

1) The parameters in Table 4.8-2 meet the Category A limits, and

2) The total battery terminal voltage is greater than or equal to 126 volts on float charge.

---

*This specification is only applicable prior to Unit 2 operation in MODE 4.

COMMONWEALTH EDISON COMPANY
BRAIDWOOD STATION, UNIT NO. 2
DOCKET NO. 50-457
NOTICE OF ISSUANCE OF FACILITY OPERATING LICENSE

Notice is hereby given that the U.S. Nuclear Regulatory Commission (the Commission or NRC), has issued Facility Operating License No. NPF-75 to Commonwealth Edison Company (the licensee) which authorizes operation of Braidwood Station, Unit No. 2 (the facility) at reactor core power levels not in excess of 3411 megawatts thermal (100 percent rated power). This license is restricted to power levels not in excess of five percent of rated power (170 megawatts thermal).

Braidwood Station, Unit No. 2 is a pressurized water reactor located in Will County, Illinois, about 20 miles south-southwest of Joliet, Illinois, in Reed Township. The license is effective as of the date of issuance.

The application for the license complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations. The Commission has made appropriate findings as required by the Act and the Commission's regulations in 10 CFR Chapter I which are set forth in the license. Prior public notice of the overall action involving the proposed issuance of an operating license was published in the Federal Register in December 15, 1978 (43 FR 58659).

The Commission has determined that the issuance of this license will not result in any environmental impacts other than those evaluated in the Final Environmental Statement and the Assessment of the Effect of License Duration on Matters Discussed in the Final Environmental Statement for the Braidwood Station, Units 1 and 2 (dated June 1984) since the activity authorized by the license is encompassed by the overall action evaluated in the Final Environmental Statement.

For further details with respect to this action, see (1) Facility Operating License No. NPF-75, with Technical Specifications and the Environmental Protection Plan; (2) the report of the Advisory Committee on Reactor Safeguards, dated February 11, 1985; (3) the Commission's Safety Evaluation Report, dated November 1983, (NUREG-1002), and Supplements 1 through 5; (4) the Final Safety Analysis Report and Amendments thereto; (5) the Environmental Report and supplements thereto; and (6) the Final Environmental Statement, dated June 1984, (NUREG-1026).

These items are available for inspection at the Commission's Public Document Room located at 1717 H Street, N.W. Washington, DC 20555 and in the Wilmington Township Public Library, 201 S. Kanakee Street, Wilmington, Illinois 60481. A copy of Facility Operating License NPF-75 may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Director, Division of Reactor Projects III/IV/V. Copies of the Safety Evaluation Report and Supplements 1 through 5 (NUREG-1002) and the Final Environmental Statement (NUREG-1026) may be purchased at current rates from the National Technical Information Service, Department of Commerce, 5285 Port Royal Road, Springfield, Virginia 22161, and through the NRC GPO sales program by writing to the Superintendent of Documents, U.S. Government Printing Office, Post Office Box 37082, Washington, DC 20013-7082.

Dated at Bethesda, Maryland this 18th day of December 1987.

FOR THE NUCLEAR REGULATORY COMMISSION

Stephen P. Sands

Stephen P. Sands, Project Manager
Project Directorate III-2
Division of Reactor Projects - III,
IV, V and Special Projects

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

Docket No. 50-457

AMENDMENT TO INDEMNITY AGREEMENT NO. B.-102
AMENDMENT NO. 5

Effective December 18, 1987, Indemnity Agreement No. B-102, between Commonwealth Edison Company and the Nuclear Regulatory Commission, dated October 8, 1985, as amended, is hereby further amended as follows:

Item 3 of the Attachment to the indemnity agreement is deleted in its entirety and the following substituted therefore:

Item 3 - License number or numbers

SNM - 1938	(From 12:01 a.m., October 8, 1985, to 12 midnight, October 16, 1986, inclusive)
SNM - 1945	(From 12:01, July 27, 1987, to 12 midnight, December 17, 1987, inclusive)
NPF - 59	(From 12:01 a.m., October 17, 1986, to 12 midnight, May 20, 1987, inclusive)
NPF - 70	(From 12:01 a.m., May 21, 1987, to 12 midnight, July 1, 1987, inclusive)
NPF - 72	(From 12:01 a.m., July 2, 1987)
NPF - 75	(From 12:01 a.m., December 18,) 1987

FOR THE U. S. NUCLEAR REGULATORY COMMISSION

Cecil O. Thomas, Jr.

Cecil O. Thomas, Jr., Chief
Policy Development and Technical Support Branch
Program Management, Policy Development
and Analysis Staff

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J PDR

Accepted ______________________,

By ______________________
COMMONWEALTH EDISON COMPANY

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

Docket No. 50-457

AMENDMENT TO INDEMNITY AGREEMENT NO. B.-102
AMENDMENT NO. 5

Effective December 18, 1987, Indemnity Agreement No. B-102, between Commonwealth Edison Company and the Nuclear Regulatory Commission, dated October 8, 1985, as amended, is hereby further amended as follows:

Item 3 of the Attachment to the indemnity agreement is deleted in its entirety and the following substituted therefore:

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NPF - 72	(From 12:01 a.m., July 2, 1987)
NPF - 75	(From 12:01 a.m., December 18,) 1987

FOR THE U. S. NUCLEAR REGULATORY COMMISSION

Cecil O. Thomas Jr.

Cecil O. Thomas, Jr., Chief
Policy Development and Technical Support Branch
Program Management, Policy Development
and Analysis Staff

Accepted ________________________,

By ________________________
COMMONWEALTH EDISON COMPANY

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DECEMBER 1987

SER RELATED TO THE OPERATION OF BRAIDWOOD STATION UNITS 1 AND 2

NUREG-1002, Supp. 5

NUREG-1002
Supplement No. 5

Safety Evaluation Report
related to the operation of Braidwood Station, Units 1 and 2

Docket Nos. STN 50-456 and STN 50-457

Commonwealth Edison Company

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

December 1987

ABSTRACT

In November 1983, the staff of the Nuclear Regulatory Commission issued its Safety Evaluation Report (NUREG-1002) regarding the application filed by the Commonwealth Edison Company, as applicant and owner, for a license to operate Braidwood Station, Units 1 and 2 (Docket Nos. 50-456 and 50-457). The first supplement to NUREG-1002 was issued in September 1986; the second supplement was issued in October 1986; the third supplement was issued in May 1987; the fourth supplement was issued in July 1987. This fifth supplement to NUREG-1002 is in support of the low-power license for Unit 2 and provides the status of certain items that remained unresolved at the time Supplement 4 was published. The facility is located in Reed Township, Will County, Illinois.

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APPENDIX A CONTINUATION OF CHRONOLOGY OF NRC STAFF RADIOLOGICAL SAFETY REVIEW OF BRAIDWOOD STATION, UNITS 1 AND 2

APPENDIX F NRC STAFF CONTRIBUTORS

APPENDIX K PRESERVICE INSPECTION RELIEF REQUEST EVALUATION

1 INTRODUCTION AND GENERAL DESCRIPTION OF FACILITY

1.1 Introduction

In November 1983, the Nuclear Regulatory Commission (NRC) staff issued its Safety Evaluation Report (SER) (NUREG-1002) on the application filed by the Commonwealth Edison Company, as applicant and owner, for a license to operate Braidwood Station, Units 1 and 2 (Docket Nos. 50-456 and 50-457). At that time, the staff identified items that had not been resolved with the applicant. The first supplement to NUREG-1002 was issued in September 1986; the second supplement to NUREG-1002 was issued in October 1986; the third supplement to NUREG-1002 was issued in May 1987; the fourth supplement was issued in July 1987. This fifth supplement to the SER provides the staff evaluation of the open items that have been resolved to date and addresses changes to the SER that resulted from the receipt of additional information from Commonwealth Edison Company (licensee); in addition, this supplement supports the issuance of the low-power license for Unit 2.

Each section or appendix that follows is numbered the same as the corresponding SER section or appendix that is being updated. Each section is supplementary to and not in lieu of the discussion in the SER unless otherwise noted. Appendix A continues the chronology of the staff's actions related to the processing of the application for Braidwood Units 1 and 2. Appendix F lists principal staff members who contributed to this supplement. Appendix K provides the staff evaluation of the licensee's request for relief from performing the Code-required volumetric examination on two welds.

Copies of this SER supplement are available for inspection at the NRC Public Document Room, 1717 H Street, N.W., Washington, D.C., and at the Wilmington Township Public Library, 201 South Kankakee Street, Wilmington, Illinois 60481.

The NRC Project Manager for Braidwood Station, Units 1 and 2, is Mr. Stephen P. Sands. Mr. Sands may be contacted by calling (301) 492-8298 or writing:

Stephen P. Sands
Office of Nuclear Reactor Regulation
Project Directorate III-2
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

1.7 Summary of Outstanding Items

The current status of the outstanding items listed in the SER follows:

Part A Items	Status	Section
(1) Pump and valve operability	Closed in Supplement 2	3.9.3.2*
(2) Seismic and dynamic qualification of equipment	Closed in Supplement 2	3.10*
(3) Environmental qualification of electrical and mechanical equipment	Closed in Supplement 2	3.11*
(4) Containment pressure boundary components	Closed in Supplement 1	6.2.7
(5) Organizational structure	Closed in Supplement 1	13.1, 13.4
(6) Emergency preparedness plans and facilities	Closed in Supplement 1	13.3*
(7) Procedures generation package (PGP)	Closed in Supplement 2	13.5.2
(8) Control room human factors review	Closed in Supplement 4	18.2*
(9) Safety parameter display system	Closed in Supplement 4	18.3*
(10) Control room habitability	Closed in Supplement 3	6.4
Part B Items		
(1) Turbine missile evaluation	Closed in Supplement 1	3.5.1.3
(2) Improved thermal design procedures	Closed in Supplement 1	4.4.1
(3) TMI Action Item II.F.2: Inadequate Core Cooling Instrumentation	Closed in Supplement 1	4.4.7
(4) Steam generator flow-induced vibrations	Closed in Supplement 1	5.4.2
(5) Conformance of ESF filter system to RG 1.52	Closed in Supplement 2	6.5.1
(6) Fire protection program	Closed in Supplement 3	9.5.1

*This section includes both site-specific-related information and duplicate-plant design features.

Part B Items (Continued)	Status	Section
(7) Volume reduction system	Closed in Supplement 2	11.1, 11.4.2

1.8 Confirmatory Issues

The current status of the confirmatory issues follows:

Part A Items	Status	Section
(1) Applicant compliance with the Commission's regulations	Closed in Supplement 2	1.1, 3.1*
(2) Site drainage	Closed in Supplement 1	2.4.3.3
(3) Piping vibration test program	Closed in Supplement 1	3.9.2.1*
(4) Preservice inspection program	Closed in Supplement 2	5.2.4, 6.6*
(5) Reactor vessel materials	Closed in Supplement 1	5.3
(6) Electrical distribution system voltage verification	Closed in Supplement 1	8.2.4*
(7) Independence of redundant electrical safety equipment	Closed in Supplement 1	8.4.4
(8) RPM qualifications	Closed in Supplement 1	12.5
(9) Revision to Physical Security Plan	Closed in Supplement 1	13.6
(10) Control room human factors review	Opened in Supplement 4	18.2*
(11) Safety parameter display system	Opened in Supplement 4	18.3*

Part B Items	Status	Section
(1) Inservice testing of pumps and valves	Partially closed in Supplement 2	3.9.6

*This section includes both site-specific-related information and duplicate-plant design features.

Part B Items (Continued)

	Status	Section
(2) Steam generator tube surveillance	Closed in Supplement 1	5.4.2.2
(3) Charging pump deadheading	Closed in Supplement 1	6.3.2, 7.3.2
(4) Minimum containment pressure analysis for performance capabilities of ECCS	Closed in Supplement 1	6.2.1.5
(5) Containment sump screen	Closed in Supplement 1	6.2.2
(6) Containment leakage testing vent and drain provisions	Closed in Supplement 1	6.2.6
(7) Confirmatory test for sump design	Closed in Supplement 1	6.3.4.1
(8) IE Bulletin 80-06	Closed in Supplement 1	7.3.2.2
(9) Remote shutdown capability	Closed in Supplement 2	7.4.2.2
(10) TMI Action Plan Item II.D.1	Partially closed in Supplement 1	3.9.3.3, 5.2.2
TMI Action Plan Item II.K.3.1	Closed in Supplement 1	7.6.2.7
TMI Action Plan Item III.D.1.1	Closed in Supplement 1	9.3.5
(11) SWS process control program	Closed in Supplement 2	11.4.1
(12) Noble gas monitor	Closed in Supplement 2	11.5.2
(13) RCP rotor seizure and shaft break	Closed in Supplement 1	15.3.6
(14) Anticipated transients without scram (ATWS)	Partially closed in Supplement 2	15.6
(15) Evaluation of compliance with 10 CFR 50.55a(a)(3)	Closed in Supplement 2	5.2.4.4
(16) Steam generator tube failure	Opened in Supplement 1	15.4.3

1.9 License Conditions

The current status of the license conditions follows:

Part A Items

	Status	Section
(1) Inservice inspection program	Closed in Supplement 3	5.2.4, 6.6*
(2) Natural circulation testing	Closed in Supplement 1	5.4.3*
(3) Response time testing	Closed in Supplement 1	7.2.2.5*
(4) Steam valve inservice inspection	Closed in Supplement 1	10.2*
(5) Implementation of secondary water chemistry monitoring and control program as proposed by the Byron/Braidwood FSAR	Closed in Supplement 1	10.3.3*
(6) TMI Item II.F.1: Iodine/Particulate Sampling	Closed in Supplement 3	11.5.2

Part B Items

(1) Masonry walls	Closed in Supplement 2	3.8.3
(2) TMI Item II.B.3 postaccident sampling	Closed in Supplement 1	9.3.2
(3) Fire protection program	Open	9.5.1
(4) Emergency diesel engine auxiliary support systems	Closed in Supplement 3	9.5.4.1

*This section includes both site-specific-related information and duplicate-plant design features.

5 REACTOR COOLANT SYSTEM

5.2 Integrity of Reactor Coolant Pressure Boundary

5.2.4 Reactor Coolant Pressure Boundary Inservice Inspection and Testing

5.2.4.5 Evaluation of Compliance With 10 CFR 50.55a(g) for Braidwood Unit 2

This evaluation supplements conclusions in Section 5.2.4.3 of Supplement No. 2 of the Braidwood Safety Evaluation Report (SER), NUREG-1002, dated October 1986. In Supplement No. 2 of the SER, the staff evaluated the preservice inspection program for Braidwood Unit 1 and concluded that the preservice inspection program is acceptable and in compliance with 10 CFR 50.55a(g). By letter dated September 23, 1987, the applicant submitted the preservice inspection program for Braidwood Unit 2 and stated that the Unit 2 preservice inspection requirements are the same as those used for Unit 1. Except for Relief Request 2NR-8, the relief requests for Unit 2 are identical to those for Unit 1, which were evaluated in Appendix K of Supplement No. 2. Relief Request 2NR-8 applies only to Unit 2 and is evaluated in Appendix K. All technical issues related to the Braidwood Unit 2 preservice inspection program have been addressed and the staff, therefore, concludes that it is acceptable.

By letter dated April 28, 1987, the licensee committed to submit the Braidwood Unit 2 inservice inspection (ISI) program within 12 months from the date of issuance of the first facility operating license for Braidwood Unit 2. This program will be evaluated on the basis of 10 CFR 50.55a(g)(4) which requires that the initial 120-month inspection interval shall comply with requirements in the latest edition and addenda of the ASME Code incorporated by reference in paragraph 50.55a(b). This program will be evaluated after the applicable ASME Code edition and addenda can be determined and before the first refueling outage when inservice inspection commences.

6 ENGINEERED SAFETY FEATURES

6.4 Control Room Habitability

In a letter dated October 30, 1987, the licensee proposed an interim operating plan for the control room ventilation (VC) system. Section 6.5.1 of Supplement No. 2 (SSER 2) of the Braidwood Safety Evaluation Report (SER), NUREG-1002, dated October 1986, provided the staff evaluation and acceptance of the plan that was intended for use during the startup of Braidwood Unit 1. This acceptance included the provision, during fuel loading and reactor system testing before initial criticality, that one train of the VC emergency makeup filter system be available, including an associated chiller system and control room air handling unit.

In a letter dated March 26, 1987, from S. C. Hunsader to H. R. Denton, the licensee provided a plan to utilize, on a temporary basis in the early summer of 1987, the service building chilled water system in lieu of the control room chiller. Section 6.4 of SSER 4, dated July 1987, provided the NRC evaluation and acceptance of this plan. On the basis of its review of the licensee's proposal, the staff concluded that the proposed cross-tie met General Design Criterion (GDC) 4 of Appendix A to 10 CFR 50 as described in the Standard Review Plan Section 9.4.1 (NUREG-0800) and was, therefore, an acceptable means for providing adequate temporary cooling to the control room envelope.

The licensee has proposed the same plan to again utilize the service building chiller system in an effort to complete the retubing work of the VC system chillers, before the initial criticality of Braidwood Unit 2. A temporary cross-tie of the service building and control room chilled water systems is again being considered. The proposed configuration is the same as the one presented in SSER 4. The evaluated finding of safety significance will remain unchanged since Braidwood Unit 1 will be in cold shutdown (Mode 5) and Unit 2 is expected to be in a condition no more than its pre-critical testing phase.

There are no changes necessary to the Braidwood Technical Specifications since Technical Specification 3/4.7.6 currently includes a note that renders it not applicable before initial criticality on Cycle 1. The licensee requests that this note remain in place for Braidwood Unit 2 in order that Unit 2 pre-critical testing can proceed during the anticipated Unit 1 outage and concurrent VC chiller retubing work. Additionally, Braidwood Unit 1 will be maintained in cold shutdown during the retubing effort and no positive reactivity changes or core alterations will be permitted.

On the basis of its review of the licensee's proposal, the staff concludes that the proposed cross-tie of the service building and control room chilled water systems is acceptable. The retubing of the VC chillers must be completed and satisfactorily tested before entering Mode 5 for Unit 1. The licensee will maintain compliance with the appropriate action statement requirements of Technical Specification 3.7.6.

In SSER 3, dated May 1987, the licensee indicated that representatives from Will County, Illinois, agreed to provide notification to the Braidwood Station in the event of a chlorine accident. The staff believes that this notification process is an acceptable means of alerting the Braidwood Station so that the control room can be isolated. In addition, Braidwood Station was required to include with its control room technical specifications: (1) a surveillance requirement to demonstrate, on an 18-month basis, that the control room envelope can be isolated and (2) a procedure to demonstrate, on an 18-month basis, that control room envelope integrity is maintained (i.e., infiltration into the control room envelope in the isolation mode does not negate the toxic gas analysis and, thus, the capability to protect the operators). The first demonstration that the control room envelope integrity is maintained was to be completed before the fuel loading date for Braidwood Unit 2. However, the licensee has proposed that this demonstration be deferred until the surveillance outage for Unit 1, scheduled for January 1988. Otherwise, Unit 1 would need to shut down as per Technical Specification 3/4.7.6.

On the basis of its review of the licensee's proposal, the staff concludes that the proposed deferral is acceptable because of the low probability of the occurrence of a chlorine release and the fact that compensatory measures are in place that would be used to mitigate the consequences of such an event. However, the control room envelope integrity demonstration must be completed before Unit 1 can enter into Mode 4. The licensee made this commitment by letter dated December 11, 1987 from S. C. Hunsader to T. E. Murley.

6.6 <u>Inservice Inspection of Class 2 and 3 Components</u>

6.6.4 Evaluation of Compliance With 10 CFR 50.55a(g) for Braidwood Unit 2

This evaluation supplements conclusions in Section 6.6.3 of SSER 2. In SSER 2, the staff evaluated the preservice inspection program (PSI) for Braidwood Unit 1 and concluded that the PSI program is acceptable and in compliance with 10 CFR 50.55a(g). By letter dated September 23, 1987, the applicant submitted the PSI program for Braidwood Unit 2 and stated that the Unit 2 PSI requirements are the same as those used for Unit 1. The relief requests for Class 2 components of Braidwood Unit 2 are identical to those for Braidwood Unit 1; these were evaluated in Appendix K of SSER 2. The staff therefore concludes that the PSI program for Class 2 and 3 components for Braidwood Unit 2 is acceptable.

By letter dated April 28, 1987, the licensee committed to submit the Braidwood Unit 2 inservice inspection (ISI) program within 12 months from the date of issuance of the first facility operating license for Braidwood Unit 2. This program will be evaluated on the basis of 10 CFR 50.55a(g)(4) which requires that the initial 120-month inspection interval shall comply with requirements in the latest edition and addenda of the ASME Code incorporated by reference in paragraph 50.55a(b). This program will be evaluated after the applicable ASME Code edition and addenda can be determined and before the first refueling outage when ISI commences.

9 AUXILIARY SYSTEMS

9.5 Other Auxiliary Systems

9.5.1 Fire Protection Program

By letter dated June 3, 1987, the licensee submitted Amendment No. 10 to the Fire Protection Report for Byron/Braidwood Units 1 and 2. The staff review relates to those changes that are specific to Braidwood Unit 2.

Amendment 10 provides a new Section 2.4 and Appendix A5.8 relating to the Braidwood Unit 2 safe shutdown analysis and Appendix R deviations, respectively. The licensee also provided three attachments to the submittal. Attachment A is an itemized summary and explanation of all changes included in the amendment. Appendix B compares the Braidwood Unit 2 safe shutdown analysis to the Braidwood Unit 1 and Byron Unit 2 analyses. Appendix C compares the Braidwood Unit 2 Appendix R deviations to the Byron Unit 2 deviations.

The staff has reviewed this submittal with respect to the fire protection program against the corresponding analysis that was previously reviewed as documented in the Safety Evaluation Report and its supplements for Byron Units 1 and 2 and Braidwood Unit 1. The Braidwood Unit 2 submittal does not indicate any significant change from the methodology previously reviewed and accepted at the other three nuclear reactor facilities. Further, the Braidwood Unit 2 alternate shutdown approach is identical to that of the other three nuclear reactor facilities.

On the basis of this review, the staff concludes that the same level of fire protection is being provided at Braidwood Unit 2 for the post-fire safe shutdown and alternate shutdown capability as was previously approved for Byron Units 1 and 2 and Braidwood Unit 1. The staff concludes that the previous safety evaluation documented in the Braidwood SER remains valid.

APPENDIX A

CONTINUATION OF CHRONOLOGY OF NRC STAFF RADIOLOGICAL SAFETY REVIEW OF BRAIDWOOD STATION, UNITS 1 AND 2

June 20, 1987	Representatives from NRC, Commonwealth Edison, and Business and Professional People for Public Interest meet in Bethesda, Maryland to provide information to assist in determination of significant hazards consideration regarding transfer of ownership. (Summary issued on July 13, 1987.)
June 26, 1987	Letter to licensee advising that appropriate offsite emergency response plan be revised to reflect provisions of Federal Emergency Management Agency (FEMA) Guidance Memorandum.
July 7, 1987	Letter from licensee concerning additional welds where Code Case N-340 is used at facility.
July 8, 1987	Letter from licensee transmitting Final Safety Analysis Report (FSAR) changes.
July 9, 1987	Letter to licensee concerning Generic Letter 87-12, loss of residual heat removal (RHR) while reactor coolant system (RCS) partially filled.
July 9, 1987	Letter from licensee transmitting additional information to justify FSAR changes.
July 10, 1987	Letter to licensee concerning Generic Letter 87-13, integrity of requalification exams at nonpower reactors.
July 14, 1987	Letter to licensee transmitting Supplement 4 to the Safety Evaluation Report (SSER 4) (NUREG-1002) regarding operation of facility.
July 14, 1987	Letter from licensee concerning list of 11 concern areas on emergency procedures.
July 15, 1987	Letter from licensee concerning changes to be made to plant FSAR.
July 16, 1987	Letter to licensee concerning proposed additions and clarifications regarding conformance with criteria of Regulatory Guides 1.52 and 1.140.

July 22, 1987	Letter from licensee transmitting Technical Specifications regarding surveillance testing of emergency diesel generators.
July 22, 1987	Letter from licensee concerning information regarding five separate issues, including DOE/NRC Forms 741 and 742 and review of safeguards classification information on forms.
July 23, 1987	Letter to licensee concerning amendment to Technical Specification 4.8.1.1.2, diesel generator.
July 30, 1987	Letter from licensee concerning applications for amendments.
July 30, 1987	Letter from licensee submitting "Final Summary Report of Human Factors Engineering Review for Byron and Braidwood Stations SPDS."
August 4, 1987	Letter to licensee concerning Generic Letter 87-14 regarding operator licensing exams.
August 5, 1987	Letter from licensee transmitting Amendment 48 to FSAR.
August 6, 1987	Letter to licensee transmitting Safety Issues Management System (SIMS) printouts for review for each plant.
August 7, 1987	Letter from licensee concerning proposed revisions to test in startup test program involving control rod drop measurements.
August 12, 1987	Letter from licensee transmitting additional proposed exception to FSAR, Appendix A.
August 24, 1987	Letter from licensee transmitting page 6-3 of final summary report of human factors review not included in initial submittal of facilities emergency reponse report.
August 31, 1987	Letter from licensee transmitting SER (NUREG-1002, Supplement 4) Section 18 response, addressing detailed control room design review (DCRDR) and safety parameter display system (SPDS) items.
August 31, 1987	Letter from licensee transmitting Revisions 6 and 7 to inservice testing pump and valve programs, respectively, for Byron Station and Revisions 3 and 3a to inservice testing pump and valve programs.
September 1, 1987	Letter from licensee transmitting final report "Evaluation of Plant Variables for Compliance," per Regulatory Guide 1.97 and Supplement 1 to NUREG-0737.
September 8, 1987	Letter from licensee clarifying the author's withdrawal of applications for amendments to Licenses NPF-66 and NPF-72.

September 8, 1987	Letter from licensee forwarding August 3, 1987 safety issues management system (SIMS) update.
September 9, 1987	Letter to licensee transmitting cost analyses for operating license (OL) application reviews.
September 9, 1987	Letter to licensee requesting that replacement parts on diesel generator auxiliary equipment be classified Safety Category 1, but Quality Group G rather than Quality Group C acceptable.
September 9, 1987	Letter to licensee transmitting request for additional information regarding Generic Letter 83-28, Items 4.2.3 and 4.2.4 concerning Salem anticipated transient without scram (ATWS).
September 10, 1987	Letter from licensee requesting approval for deviation to schedule for submittal of startup test as delineated in Appendix B of Revision 2 to Regulatory Guide 1.68.
September 18, 1987	Letter from licensee requesting one-time exemption to utility February 19, 1986 licensed operator requalification program topical report.
September 18, 1987	Letter from licensee transmitting additional information, requested per telephone conversations, on Item 1 of utility December 1, 1986 response to request for additional information about changes made in Amendment 47 to FSAR.
September 23, 1987	Letter from licensee transmitting Revision 0 to "Braidwood Unit 2 Preservice Inspection Program."
September 23, 1987	Letter from licensee transmitting updated response for Items 2.1 and 4.5.2 of Generic Letter 83-28.
September 25, 1987	Letter from licensee transmitting information assessing safe operation of pressurized-water reactors (PWRs) when reactor coolant systems (RCS) water level is below top of reactor vessel, per Generic Letter 87-12.
September 30, 1987	Letter from licensee transmitting revised emergency plan annexes for generating station emergency procedure manual.
October 6, 1987	Letter from licensee transmitting Policy 1 to MAELU Certificate M-115 and NELIA Certificate N-115.
October 7, 1987	Letter from licensee informing staff that fuel will be loaded on December 11, 1987 as scheduled.

October 13, 1987	Letter from licensee informing staff that J. Gallo and P. P. Steptoe will present oral arguments in proceeding on November 21, 1987 regarding harassment issues presented on appeal and issues concerning plan construction assessment program and overinspection results compiled by Pittsburgh Testing Laboratory.
October 19, 1987	Letter to licensee transmitting safety evaluation regarding SPDS.
October 26, 1987	Letter to licensee informing it of upcoming NRC site visit on November 9-13, 1987 to examine MESAC system.
October 26, 1987	Letter to licensee advising it to schedule submittal for startup tests, per Appendix B of Regulatory Guide 1.68.
November 25, 1987	Letter from licensee informing staff that portions of construction activities regarding plant fire protection program in safety-related areas not expected to be completed by start of fuel loading.
November 25, 1987	Letter from licensee requesting schedular relief for completion of review and evaluation of five preoperational tests and completion and review of retests beyond fuel loading.
November 25, 1987	Letter to licensee forwarding Amendments 12 (Byron Unit 1), 12 (Byron Unit 2), and 2 (Braidwood Unit 1) to Licenses NPF-37, NPF-66, and NPF-72, respectively. Amendments revise Technical Specifications to allow one-time extention to 32 months for interval for performing 18-month instrument surveillance.

APPENDIX F

NRC STAFF CONTRIBUTORS

Name	Title	Review Branch*
John W. Craig	Branch Chief	Plant Systems Branch, DEST
Richard J. Eckenrode	Human Factors Engineer	Human Factors Assessment Branch, DLPQ
George Johnson	Materials Engineer	Materials Engineering Branch, DEST
Dennis J. Kubicki	Fire Protection Engineer	Plant Systems Branch, DEST
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*Reflects reorganization since SER was issued.

APPENDIX K

COMMONWEALTH EDISON COMPANY
BRAIDWOOD GENERATING STATION - UNIT 2

DOCKET NUMBER 50-457

SAFETY EVALUATION REPORT SUPPLEMENT
PRESERVICE INSPECTION RELIEF REQUEST EVALUATION

Q. <u>RELIEF REQUEST NO. 2NR-8 (REV. 0), EXAMINATION CATEGORY B-J, ITEM NO. B9.31, PRESSURE RETAINING BRANCH CONNECTION WELDS IN CLASS 1 PIPING</u>

<u>CODE REQUIREMENTS:</u> Examination Category B-J, Item B9.31 requires a surface and volumetric examination of the areas described in Figures IWB-2500-9 thru IWB-2500-11 for pipe branch connections greater than 2 in. nominal pipe size. This examination includes essentially 100% of the weld length.

<u>Code Relief Request:</u> Relief is requested from performing the Code-required volumetric examination on the following welds:

Line Number	Weld Number
2RC04AB-12"	2RC-11-05
2RC04AA-12"	2SI-02-45

<u>Reason for Request</u>

The applicant reports that the above listed welds are 316 stainless steel weldolets. Due to the weld geometry and the metallurgical properties of the material, ultrasonic examination of these welds is not practical.

<u>Staff Evaluation:</u> This relief request is acceptable for PSI based on the following:

1. The subject welds received radiographic and surface examinations during fabrication in accordance with ASME Code requirements.

2. The welds are subjected to a system pressure test in accordance with Section XI requirements.

The staff therefore concludes that fabrication examinations and Section XI surface examination provide assurance of the preservice structural integrity of the branch connection welds and that compliance with the specific requirements of Section XI would result in hardship or unusual difficulties without a compensating increase in the level of quality and safety.

NRC FORM 335 (2-84) NRCM 1102, 3201, 3202

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

SEE INSTRUCTIONS ON THE REVERSE.

1. REPORT NUMBER (Assigned by TIDC, add Vol. No., if any)
NUREG-1002
Supplement No. 5

2. TITLE AND SUBTITLE
Safety Evaluation Report related to the operation of Braidwood Station, Units 1 and 2

3. LEAVE BLANK

4. DATE REPORT COMPLETED

MONTH	YEAR
December	1987

5. AUTHOR(S)

6. DATE REPORT ISSUED

MONTH	YEAR
December	1987

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Division of Reactor Projects - III, IV, V and Special Projects
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, DC 20555

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9. FIN OR GRANT NUMBER

10. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)
Same as 7, above

11a. TYPE OF REPORT
Technical

b. PERIOD COVERED (Inclusive dates)
November 1983 - December 1987

12. SUPPLEMENTARY NOTES
Docket Nos. STN 50-456 and STN 50-457

13. ABSTRACT (200 words or less)

In November 1983, the staff of the Nuclear Regulatory Commission issued its Safety Evaluation Report (NUREG-1002) regarding the application filed by the Commonwealth Edison Company, as applicant and owner, for a license to operate Braidwood Station, Units 1 and 2 (Docket Nos. 50-456 and 50-457). The first supplement to NUREG-1002 was issued in September 1986; the second supplement was issued in October 1986; the third supplement was issued in May 1987; the fourth supplement was issued in July 1987. This fifth supplement to NUREG-1002 is in support of the low-power license for Unit 2 and provides the status of certain items that remained unresolved at the time Supplement 4 was published. The facility is located in Reed Township, Will County, Illinois.

14. DOCUMENT ANALYSIS – a. KEYWORDS/DESCRIPTORS

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