July 2, 1987

Docket No.: STN 50-456

Mr. Dennis L. Farrar
Director of Nuclear Licensing
Commonwealth Edison Company
P.O. Box 767
Chicago, Illinois 60690

Dear Mr. Farrar:

SUBJECT: ISSUANCE OF FACILITY OPERATING LICENSE NPF-72 - BRAIDWOOD STATION, UNIT 1

The U. S. Nuclear Regulatory Commission (NRC) has issued the enclosed Facility Operating License NPF-72, together with Technical Specifications and Environmental Protection Plan for Braidwood Station, Unit 1. Based upon the findings of the NRC, as reflected in the enclosed license and the favorable vote by the Commission on full-power operation, License No. NPF-72, which supercedes License No. NPF-70 issued on May 21, 1987, authorizes operation of Braidwood Station, Unit 1 at reactor power levels not in excess of 3411 megawatts thermal (100 percent rated power).

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. 3 to Indemnity Agreement No. B-103 which covers the activities authorized under License No. NPF-72 are also enclosed. Please return one signed copy to this office.

Safety Evaluation Report Supplement No. 4 (SSER 4) was prepared in support of issuing the enclosed license. Enclosed is a pre-printed copy of SSER 4. Twenty (20) bound copies of SSER 4 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-72
2. Federal Register Notice
3. Amendment No. 3 to Indemnity Agreement No. B-103
4. Supplement No. 4 to the Safety Evaluation Report

cc: w/enclosures:
See next page

PDIII-2
JStevens
6/30/87

PDI-1
CVogan
6/30/87

SP see cc 5 on pg of license
IDinitz
6/18/87

OGC see cc 5 on pg of license
EReis
6/16/87

D:PDIII-2
DMuller
7/ /87

AD:DRSP
GHolahan
7/1/87

D:DRSP
DCrutchfield
7/1/87

AD:NRR
FMiraglia
7/2/87

D:NRR
TMurley
7/2/87

SUBJECT: ISSUANCE OF FACILITY OPERATING LICENSE NO. NPF-72 FOR BRAIDWOOD STATION, UNIT 1

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E. Reis, OGC*
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JScinto, OGC
ACRS (10)
C. Miles, PA
DHagan*
EButcher*
CMiles, PA

* with Technical Specifications

Mr. Dennis L. Farrar
Commonwealth Edison Company

Braidwood Station
Units 1 and 2

cc:
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Atomic Power Distribution
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Nuclear Safeguards and
Licensing Division
Sargent & Lundy Engineers
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U. S. Nuclear Regulatory Commission
Resident Inspectors Office
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Braceville, Illinois 60407

Regional Administrator, Region III
U. S. Nuclear Regulatory Commission
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Mr. Dennis L. Farrar
Commonwealth Edison Company

Braidwood Station (other)
Units 1 and 2

cc:
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500 South 2nd Street
Springfield, Illinois 62701

EIS Review Coorindator
EPA Region V
230 S. Dearborn Street
Chicago, Illinois 60604

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

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

COMMONWEALTH EDISON COMPANY
DOCKET NO. STN 50-456
BRAIDWOOD STATION, UNIT 1
FACILITY OPERATING LICENSE

License No. NPF-72

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 1 (the facility) has been substantially completed in conformity with Construction Permit No. CPPR-132 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: (i) 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 No. NPF-72, subject to the conditions for protection of the environment set forth in the Environmental Protection Plan attached as Appendix B, 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, the Partial-Initial Decisions issued May 13 and 19, 1987, as amended on May 22, 1987, by the Atomic Safety and Licensing Board in regard to this facility and pursuant to approval by the Nuclear Regulatory Commission at a meeting on June 30, 1987, Facility Operating License No. NPF-72, which supersedes Facility Operating License No. NPF-70 issued on May 21, 1987, is hereby issued to Commonwealth Edison Company (the licensee) to read as follows:

A. This license applies to Braidwood Station, Unit 1, 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) <u>Maximum Power Level</u>

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.

(2) <u>Technical Specifications and Environmental Protection Plan</u>

The Technical Specifications contained in Appendix A 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) 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.

(5) Regulatory Guide 1.97, Revision 2 Compliance

The licensee shall submit the final report and a schedule for implementation within six months of NRC approval of the DCRDR.

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 October 17, 2026.

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas E Murley

Thomas E. Murley, Director
Office of Nuclear Reactor Regulation

Attachments:
1. Work Items to be completed
2. Appendix A - Technical Specifications (NUREG-1276)
3. Appendix B - Environmental Protection Plan

Date of Issuance: July 2, 1987

*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."

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.

The interim operation plan for the Auxiliary Building Ventilation (VA) System will be conducted in accordance with letters S. C. Hunsader to A. Bert Davis dated June 26, 1987, S. C. Hunsader to Thomas E. Murley dated June 23, 1987, S. C. Hunsader to Thomas E. Murley dated June 11, 1987, and A. D. Miosi to Harold R. Denton dated August 26, 1986.

COMMONWEALTH EDISON COMPANY
BRAIDWOOD STATION, UNIT NO. 1
DOCKET NO. 50-456
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-72 to Commonwealth Edison Company (the licensee) which authorizes operation of Braidwood Station, Unit No. 1 (the facility) at reactor core power levels not in excess of 3411 megawatts thermal (100 percent rated power).

Braidwood Station, Unit No. 1 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-72, 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 4; (4) the Final Safety Analysis Report and Amendments thereto; (5) the Environmental Report and supplements thereto; (6) and 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. Kankakee Street, Wilmington, Illinois 60481. A copy of Facility Operating License NPF-72 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 and Special Projects. Copies of the Safety Evaluation Report and Supplements 1 through 4 (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 2nd day of July 1987.

FOR THE NUCLEAR REGULATORY COMMISSION

Janice A. Stevens

Janice A. Stevens, 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

AMENDMENT TO INDEMNITY AGREEMENT NO. B-102
AMENDMENT NO. 3

Effective July 2, 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 therefor:

Item 3-License number or numbers

SNM-1938	(From 12:01 a.m., October 8, 1985, to 12 midnight, October 16, 1986, 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)

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Jesse L. Funches

Jesse L. Funches, Branch Chief
Policy Development and Technical Support Branch
Program Management, Policy Development and Analysis Staff
Office of Nuclear Reactor Regulation

Accepted: ______________ 1987

by ______________
Commonwealth Edison Company

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

AMENDMENT TO INDEMNITY AGREEMENT NO. B-102
AMENDMENT NO. 3

Effective July 2, 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 therefor:

Item 3-License number or numbers

SNM-1938	(From 12:01 a.m., October 8, 1985, to 12 midnight, October 16, 1986, 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)

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Jesse L. Funches, Branch Chief
Policy Development and Technical Support Branch
Program Management, Policy Development and Analysis Staff
Office of Nuclear Reactor Regulation

Accepted: ______________________ 1987

by ______________________
Commonwealth Edison Company

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NUREG-1002
Supplement No. 4

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

July 1987

NUREG-1002, Supp. No. 4

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

JULY 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; and the third supplement was issued in May 1987. This fourth supplement to NUREG-1002 reports the status of certain items that remained unresolved at the time Supplement 3 was published. The facility is located in Reed Township, Will County, Illinois.

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APPENDICES

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; and the third supplement to NUREG-1002 was issued in May 1987. The purpose of this fourth supplement to the SER is to provide the staff evaluation of the open items that have been resolved to date and to address changes to the SER that resulted from the receipt of additional information from Commonwealth Edison Company (licensee).

Each of the following sections or appendices 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 B lists references cited in this report.* Appendix F lists principal staff members who contributed to this supplement.

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 Ms. Janice A. Stevens. Ms. Stevens may be contacted by calling (301) 492-4993 or writing:

Janice A. Stevens
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**

*Availability of all material cited is described on the inside front cover of this report.
**This section includes both site-specific-related information and duplicate-plant design features.

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

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

1.8 Confirmatory Issues

The current status of the confirmatory issues follows:

	Status	Section
Part A Items		
(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		
(1) Inservice testing of pumps and valves	Partially closed in Supplement 2	3.9.6
(2) Steam generator tube surveillance	Closed in Supplement 1	5.4.2.2

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

Part B Items (Continued)	Status	Section
(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

5.2.4.6 Evaluation of Compliance With 10 CFR 50.55a(a)(3) for Braidwood Unit 2

In a previous input to this section of the SER, the staff indicated that the Braidwood Preservice Inspection (PSI) Program was based on the requirements of the ASME Code Section XI, 1977 Edition with Addenda through Summer 1978. In a letter from A. D. Miosi to H. R. Denton dated August 2, 1986, the licensee informed the NRC that a flaw was discovered in the weld between inlet nozzle F and the shell of the Braidwood Unit 2 reactor pressure vessel. The flaw was discovered during an ultrasonic examination performed from the inside of the vessel by Combustion Engineering. The examination indicated that the flaw was close to the outside surface and exceeded the preservice examination limits of ASME Code Section XI, 1977 Edition with Addenda through Summer 1978. In November 1986, the licensee performed an ultrasonic examination from the outside surface of the vessel to locate the flaw more precisely. This examination indicated that the flaw was located 1.25 inches below the outside surface and its size was less than the preservice examination limits of ASME Code Section XI, 1977 Edition with Addenda through Summer 1978. The distance sound traveled and the amount of beam spread were greater during the ultrasonic examination from the inside surface than during the examination from the outside surface. As the distance of sound travel and beam spread increase, the amount of flaw magnification increases. In addition, the licensee has obtained core samples and excavated the flaws in Byron Unit 2 steam generators. These examinations indicated that ultrasonic examination overestimated the size of the flaw. On the basis of magnification resulting from sound travel and beam spread, and the results of previous core samplings and flaw excavations, the ultrasonic examination from the outside surface appears to give more accurate size of the flaw than the examination from the inside surface. Since the flaw size resulting from the ultrasonic examination from the outside surface meets the preservice examination limits of the ASME Code, Section XI, 1977 Edition with Addenda through Summer 1978, the reactor vessel is acceptable for service without removing the flaw.

In a letter dated December 10, 1986, the licensee proposed to perform the inservice examination of the inlet nozzle F weld from the inside surface. If any measurable increase in the size of the defect is observed during the inservice examination, the size of the flaw will be measured by an ultrasonic examination from the outside surface. In a letter dated April 10, 1987, the licensee indicated that the ultrasonic examination from the outside surface is performed manually; the inside surface is examined automatically via remote control of the submerged ultrasonic transducer. By ultrasonically examining and remotely handling the submerged transducer, the licensee will significantly reduce the amount of radiation to which examination personnel are exposed. Because such exposure is to be kept as low as reasonably achievable, the staff considers the proposed inservice ultrasonic examination method acceptable for monitoring the growth of a flaw during service.

6 ENGINEERED SAFETY FEATURES

6.4 Control Room Habitability

In a letter dated March 26, 1987, the licensee proposed operating the control room air handling unit even though its associated chiller unit was inoperable. The licensee proposed that a temporary chilled water source be utilized during the time the air handling unit chiller was inoperable. A cross-tie would be made between the chilled water systems of the service building and the control room. By means of this cross-tie, the cooling load of the control room envelope would be added to the service building's chilled water system. The licensee indicated that this temporary chilling arrangement had been utilized during the summer of 1986 (while the control room's chilled water system was being preoperationally tested) and provided more than adequate cooling.

The licensee is proposing this operational mode so that maintenance can be performed on the control room chiller units before criticality. In SER Supplement 2 (SSER 2), the staff had noted the licensee's commitment to operate the control room ventilation system with one train of its emergency makeup filter system available along with its associated chiller system and air handling unit during fuel loading and reactor testing. The licensee's proposal would modify that portion of the commitment reading, "with its associated chiller system." The function associated with this portion of the commitment would be replaced by the cross-tie arrangement for the 14-day period.

On the basis of its review of the licensee's proposal, the staff concludes that the proposed cross-tie meets General Design Criterion (GDC) 4 of Appendix A to 10 CFR 50 as described in Standard Review Plan Section 9.4.1 of NUREG-0800 and is, therefore, an acceptable means for providing adequate temporary cooling to the control room envelope.

In SSER 2 the staff concluded that the Braidwood design meets GDC 19 even with the auxiliary building ventilation (VA) system inoperable provided: (1) the unfiltered inleakage to the control room is limited to 25 cubic feet per minute, (2) emergency core cooling system (ECCS) pump leakage is limited to 1 gallon per minute, and (3) reactor power is limited to 20 percent of rated power. By letters dated June 11, 1987, June 23, 1987, and June 26, 1987, the licensee proposed an interim operation plan for the VA system that would allow operation beyond 20 percent power until December 1, 1987. The actual power is not to exceed a percentage of rated power (P) given by

$$P = 97/(1+0.065L)$$

where L is the ECCS leakage in gallons per hour. This function represents the same graph proposed by the licensee in their letter dated June 11, 1987.

The staff has reviewed the proposed power limitation relationship and concludes that it is adequate to ensure compliance with both GDC 19 and 10 CFR 100 requirements. Therefore, the staff concludes that the power limitation relationship and the program for determining ECCS leakage are acceptable. The Unit 1 license contains a condition that the licensee adhere to the power limitations and leak rate determinations committed to in its letters until the VA system has been completed. Technical Specification Section 3.7.7 has been revised to allow the VA system to be completed by December 1, 1987.

6.5 Fission Product Removal and Control System

6.5.1 Engineered Safety Feature Atmospheric Cleanup Systems

In letters dated March 27, April 16, and May 11, 1987, the licensee provided information detailing the fact that it was unable to meet the testing criteria of ANSI N510-1980 for the control room recirculation charcoal adsorber. Specifically, the air flow distribution air-aerosol mixing test results were not within the acceptance criterion of ±20%. The charcoal adsorber was not designed in accordance with the criteria of ANSI N509-1980. However, the licensee committed to the criteria of ANSI N510-1980 for these adsorbers.

In the letters cited above, the licensee presented the results of charcoal adsorber testing which indicated that the assumed efficiency of 90% for the filters in the control room dose calculation is still valid despite the reduced residence time determined in the test data. Therefore, the original conclusion that doses will be within GDC 19 limits is not affected.

On this basis, the staff concludes that utilizing the recirculation adsorber units even though the air flow distribution and the air aerosol mixing test results are outside the criteria of ANSI N510-1980, is an acceptable deviation from the Standard Review Plan since doses are within the requirements of GDC 19. The staff, therefore, considers this issue resolved.

9 AUXILIARY SYSTEMS

9.5 Other Auxiliary Systems

9.5.1 Fire Protection Program

9.5.1.5 Fire Protection for Specific Plant Areas

In its letters dated June 10 and June 18, 1987, the licensee indicated that the installation of penetration seals in fire walls and fire-rated floor/ceiling assemblies and the installation of fireproofing for steel structural elements will not be completed before exceeding 5% of rated power as stated in SER Supplement 2. The licensee also stated in these letters that roving fire watch patrols have already been instituted for these affected areas and will remain in effect as required by the Station Administrative Procedures until all of the installation work has been completed by August 31, 1987. On the basis of this compensatory measure, the staff concludes that an adequate level of fire safety will be provided pending completion of the required modifications. Therefore, this matter is considered resolved.

9.5.4 Emergency Diesel Engine Fuel Oil Storage and Transfer System

9.5.4.2 Emergency Diesel Engine Fuel Oil Storage and Transfer System Design*

The SER stated that in order to minimize stirring up sediment when diesel fuel oil is added to the Unit 1 fuel oil storage tanks, the twin diesel fuel oil storage tanks for each diesel generator will be replenished by refilling one tank while the other tank provides fuel oil to the diesel. The fuel oil in the refilled tank will then be allowed to settle for 12 hours before the other tank is refilled or before fuel oil is suctioned from the refilled tank.

In a letter dated February 25, 1987, the licensee stated that it intended to perform a 100-hour reliability run on both diesel generators of Braidwood Unit 1. To perform this test without violating the Technical Specification requirement of maintaining a minimum 44,000 gallons of fuel oil in an operating plant, the licensee would have to continually refill the fuel oil storage tanks to maintain that minimum. Upon replenishing the tanks, the licensee would have to follow the FSAR/SER commitment stated above for refilling the fuel oil storage tank. However, the settling time restriction could not be satisfied for this test run.

In the letter dated February 25, 1987, the licensee proposed that as long as the fuel oil levels in each tank are maintained above the 50% level, the sediment in the tank would be stirred minimally and waiting time for settling can be eliminated. Should the level fall below 50% in a tank, the original FSAR/SER

*Title of Section 9.5.4.2 has been changed from SER to more accurately reflect the contents of this section.

commitment would apply. The staff has evaluated this proposal for other plants with similar tank designs and has found that it is an acceptable way of meeting Position C.2.g of Regulatory Guide 1.137 on minimizing stirring up sediment in fuel oil storage tanks while ensuring that a minimum of 44,000 gallons of fuel oil is available. Therefore, the staff finds the proposed change to the plant operating procedures/FSAR commitment acceptable.

18 HUMAN FACTORS ENGINEERING

18.2 Main Control Room and Remote Shutdown Panel

In its letter dated December 1, 1986, the licensee submitted the Detailed Control Room Design Review (DCRDR) Summary Report. This summary report included the human engineering discrepancies (HEDs) applicable to the Braidwood site-specific panels and instrumentation, and the survey findings along with proposed corrective actions and implementation schedules. The staff is satisfied with all proposed control room improvements and the schedule for their implementation. Therefore, Outstanding Item A(8) is closed.

On the basis of a preimplementation site audit conducted on March 10-11, 1987, the staff concluded that all requirements of Supplement 1 to NUREG-0737 had been satisfactorily completed except for: 1) the evaluation of the long term engineering solution to the problem of radio transmitters that can inadvertantly activate safeguard systems; 2) the evaluation of the lack of lamp test capability; 3) the determination of a long-term engineering solution to the indicator light bulb burnout problem; 4) the documentation of a detailed color versus use matrix for control room color coding and a commitment to make color coding in all control room contexts consistent with the green board concept; and 5) the documentation of a clarification of the licensee's response to HED No. 0165 regarding computer printer speed, in order to provide assurance that data will not be lost during a major event.

In its letter dated June 23, 1987, the licensee committed to submitting proposed resolutions to these five items by August 31, 1987. The staff finds this schedule acceptable. This is Confirmatory Issue A(10). With the control room design improvements that have already been completed, the staff has determined that the potential for operator error leading to serious consequences as a result of human factors considerations in the control room is sufficiently low to permit safe operation of Unit 1.

18.3 Safety Parameter Display System

In its letter dated June 23, 1987, the licensee stated that the identification of the wide-range and narrow-range iconic displays had been corrected for Braidwood Unit 1 in order to satisfy the requirements of Supplement 1 to NUREG-0737. Therefore, Outstanding Item A(9) is closed.

On the basis of a preimplementation site audit conducted on March 10-11, 1987, the staff concluded that all requirements of Supplement 1 to NUREG-0737 had been satisfactorily completed except for several operational problems with the SPDS. On the wide-range iconic screen, four of the eight parameters varied from the expected value enough to detract from the display. These were as follows:

(1) The reactor vessel level indication system (RVLIS) was reading properly but the value spiked to the alarm level periodically.

(2) The radiation level remained at the alarm setpoint.

(3) The steam generator level was reading far enough from the setpoint to be distracting.

(4) The containment pressure was indicating far enough from the setpoint to skew the iconic display.

On the narrow-range screen, the following three readings were off the setpoints:

(1) high radiation alarm

(2) containment temperature

(3) steam generator level

In its letter dated June 23, 1987, the licensee committed to submitting proposed resolutions to these items by August 31, 1987. The staff finds this schedule acceptable. In its letter dated October 10, 1986, the licensee committed to submitting the verification/validation report by August 1, 1987. Confirmation that the SPDS operational problems have been corrected and submittal of the verification/validation report constitute Confirmatory Issue A(11).

APPENDIX A

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

April 10, 1987	Letter from licensee concerning reactor vessel nozzle analysis.
April 28, 1987	Letter from licensee concerning Inservice Inspection (ISI) Program.
April 28, 1987	Letter from licensee concerning seismic Category I manhole covers.
April 29, 1987	Letter from Isham, Lincoln & Beale concerning inadvertent disclosure of materials subject to the Atomic Safety and Licensing Board protective order.
April 29, 1987	Letter from licensee concerning containment leak chase channels.
May 6, 1987	Filing from the Atomic Safety and Licensing Board Panel concerning motion to admit late-filed contention on financial qualifications.
May 6, 1987	Letter from licensee concerning removal of control room chlorine detectors.
May 13, 1987	Filing from the Atomic Safety and Licensing Board concerning Partial Initial Decision on Emergency Planning Issues.
May 15, 1987	Filing from Atomic Safety and Licensing Appeal Board concerning response to intervenors' motion seeking to reopen the record for the admission of a new contention.
May 18, 1987	Letter from licensee concerning Atomic Safety and Licensing Board's decision on emergency planning.
May 18, 1987	Filing concerning reconstitution of Atomic Safety and Licensing Appeal Board.
May 18, 1987	Memorandum from Atomic Safety and Licensing Appeal Board concerning reconstitution of Board.
May 19, 1987	Filing from the Atomic Safety and Licensing Board concerning concluding Partial Initial Decision (Operating License).

May 19, 1987	Filing from the Atomic Safety and Licensing Board concerning corrected pages to Minority Opinion.
May 26, 1987	Filing from the Atomic Safety and Licensing Board concerning "NRC Staff Response to Motion to Admit Late-Filed Contention on Financial Qualifications.
May 28, 1987	Letter from licensee concerning application for amendment.
May 28, 1987	Transcript concerning Immediate Effectiveness Issues.
May 29, 1987	Letter from Isham, Lincoln & Beale to the Atomic Safety and Licensing Board concerning the withdrawal of licensee's jurisdictional objection to intervenor's motion to admit late-filed contention.
June 1, 1987	Filing from the Atomic Safety Jurisdictional and Licensing Board concerning Notice of Appeal.
June 1, 1987	Letter from licensee concerning Emergency Response Facilities.
June 3, 1987	Letter from Isham, Lincoln & Beale to the Atomic Safety and Licensing Board concerning forwarding the May 28, 1987 application for amendment.
June 3, 1987	Letter to licensee concerning request for information concerning offsite medical services for Units 1 and 2.
June 9, 1987	Filing from the Atomic Safety and Licensing Board concerning emergency planning order.
June 10, 1987	Filing from the Atomic Safety and Licensing Board concerning notice of reconstitution of board.
June 10, 1987	Filing from the Atomic Safety and Licensing Board concerning Memorandum and order denying intervenors' motion to admit late-filed contentions on financial qualifications.
June 11, 1987	Letter from licensee concerning "Interim Operation of VA System."
June 12, 1987	Transcript of telephone conference in Washington, D.C.
June 15, 1987	Filing from the Atomic Safety and Licensing Board concerning memorandum on licensing board jurisdiction.
June 23, 1987	Letter from licensee concerning detailed control room design review and safety parameter display system.
June 23, 1987	Letter from licensee concerning "Interim Operation of VA System."
June 26, 1987	Letter from licensee concerning "Interim Operation of VA System."

APPENDIX B

BIBLIOGRAPHY

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APPENDIX F

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*Reflects reorganization since SER was issued.