July 1, 1994

Docket No. STN 50-456

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

Dear Mr. Farrar:

SUBJECT: BRAIDWOOD STATION, UNIT 1 (TAC NO. M89697)

The U.S. Nuclear Regulatory Commission has forwarded the enclosed "Notice of Consideration of Issuance of Amendment to Facility Operating License, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing" to the Office of the Federal Register for publication.

This notice relates to your June 20, 1994, submittal to revise the Technical Specifications (TSs) to remove the condition limiting operation of the Braidwood, Unit 1, facility to 100 days during the present fuel cycle when T_{hot} is greater than 500°F, and to restore the reactor coolant dose equivalent Iodine-131 limit to 1 microcurie per gram of coolant from the present value of 0.35. Both the limit on permissible operational time and the reduction in the permissible level of Iodine-131 were incorporated into the TSs by Amendment No. 50 issued to Facility Operating License No. NPF-72 for Braidwood Station, Unit 1, on May 7, 1994.

Sincerely,

original signed by

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

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Notice

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

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DATE	6/30/94	6/30/94	06/30/94	6/30/94	7/01/94
COPY	YES/NO	YES/NO	YES/NO	YES/NO	YES/NO

Mr. D. L. Farrar
Commonwealth Edison Company

Braidwood Station
Units 1 and 2

cc:

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77 W. Jackson Blvd.
Chicago, Illinois 60604-3590

UNITED STATES NUCLEAR REGULATORY COMMISSION

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-456

NOTICE OF CONSIDERATION OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE, PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION, AND OPPORTUNITY FOR A HEARING

The U.S. Nuclear Regulatory Commission (the Commission) is considering issuance of an amendment to Facility Operating License No. NPF-72 issued to Commonwealth Edison Company (CECo, the licensee) for operation of the Braidwood Station, Unit 1, located in Will County, Illinois.

The proposed amendment would revise the Braidwood, Unit 1, Technical Specifications (TSs) to remove the condition limiting operation of the facility to 100 days during the present fuel cycle when T_{hot} is greater than 500°F and to restore the reactor coolant dose equivalent Iodine-131 limit to 1 microcurie per gram of coolant from the present value of 0.35. Both the limit on permissible operational time and the reduction in the permissible level of Iodine-131 were incorporated into the TSs by Amendment No. 50 issued to Facility Operating License No. NPF-72 for Braidwood Station, Unit 1, on May 7, 1994.

Before issuance of the proposed license amendment, the Commission will have made findings required by the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations.

The Commission has made a proposed determination that the amendment request involves no significant hazards consideration. Under the Commission's regulations in 10 CFR 50.92, this means that operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. **The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.**

   Braidwood Unit 1 TS Amendment 50 imposed a 100 calendar days with T_{hot} greater than 500°F operating limit on Unit 1. This limitation was a consequence of the amount of main steam line break (MSLB) leakage predicted in Braidwood Station's April 30, 1994, submittal. These predictions were made using the Log-Logistic method of draft NUREG 1477, "Voltage Based Interim Plugging for Steam Generator Tubes - Task Group Report," with the Dose Equivalent Iodine-131 limit of Specification 3.4.8 reduced from 1.0 microcurie per gram (μCi/gm) to 0.35 μCi/gm. However, WCAP 14046; "Braidwood Unit 1 Technical Support for Cycle 5 Steam Generator Interim Plugging Criteria" (WCAP-14046), docketed June 10, 1994, as required by Braidwood Station's April 25, 1994, submittal, has shown using the Electric Power Research Institute (EPRI) Leakrate Correlation that projected End Of Cycle (EOC)-5 MSLB leakage is 3.1 gallons per minute (gpm) which is less than the allowable limit of 9.1 gpm for Braidwood Unit 1. This analysis is discussed in detail in WCAP-14046.

   Thus the Unit 1 100 day operating limit and reactor coolant dose equivalent iodine restriction imposed by Amendment 50 on the basis of MSLB leakage is no longer required.

   In addition to the 100 day, leakage based limit, the Nuclear Regulatory Commissions (NRC) Safety Evaluation Report (SER), issued May 7, 1994, in support of Braidwood Station's Unit 1 TS Amendment 50 discusses a 4.6 month (138 day) limit derived from a deterministic assessment of SG tube burst probability. To address the issue of tube burst for full cycle operation Braidwood Station's April 25, 1994, submittal provided a probabilistic risk assessment which is restated below.

   As part of ComEd's evaluation of the operability of Braidwood Unit 1 Cycle 5, a risk evaluation was completed. The objective of this evaluation was to compare core damage frequency, with containment bypass, with and without the interim plugging criteria applied at Braidwood 1.

ComEd has evaluated the impact of operation using the proposed interim plugging criteria against the results of insights from the draft Braidwood Individual Plant Examination (IPE). Braidwood Station is scheduled to docket its IPE June 30, 1994. Byron Station's IPE was docketed April 20, 1994. The SG sections of these documents are identical.

While the Braidwood IPE is not in its final form, it is believed that the quantification in hand is sufficiently robust to allow a validation assessment of the impact of such operation. The ComEd evaluation parallels that described in the NRC Staff's SER for Palo Verde Unit 2 dated August 19, 1993.

The values calculated in WCAP-14046, for Beginning of Cycle (BOC) 5 and EOC 5 using 0.6 Probability Of Detection (POD) were used to develop a cycle average burst probability. Another BOC 5 burst probability assuming a POD of 0.6 for indications less than 3 volts and 1.0 for indications greater than 3 volts was used to evaluate the impact of POD on core damage frequency.

The total Braidwood core damage frequency is estimated to be 2.74E-5 per reactor year with a total contribution from containment bypass sequences of 2.9E-8 per reactor year in the current IPE. Operation with the alternate repair criteria with a variable POD is expected to increase the MSLB with containment bypass sequence frequency contribution by a factor of only 10%. An upper bound increase of a factor of two is derived when the fixed POD of 0.6 is employed in the calculation. Neither increase is significant from a risk perspective.

The reason for a reduced core damage frequency with a higher POD is that large voltage indications have a high assurance of being identified and removed from service during inspection. Therefore, the calculation of burst probability during MSLB changes because of differences in the assumed distribution of indications left in service at BOC. The EOC burst probability also changes because the growth distribution is added to the new BOC distribution of indications. The result of this change is a significant reduction in burst probability during MSLB.

Therefore, the operation of Braidwood Unit 1 Cycle 5 for a complete 18 month fuel cycle with the application of the one volt IPC does not significantly increase the core damage frequency even with the conservative assumption of a POD of 0.6 and application of the full growth rate distribution observed during Cycle 4.

To further address SG tube burst probability, the following qualitative discussion of limited tube support plate (TSP) displacement is provided. As part of ComEd's technical support for the implementation of IPC at Braidwood Unit 1, numerous quantitative analyses were completed to assure the structural integrity of the SG tubing. These quantitative determinations were provided as part of WCAP-14046. These analyses focused on the quantifiable elements of the IPC to evaluate the impact of crack length on steam generator tube leakage and burst, and were completed consistent with the guidance provided in draft RG 1.121, "Bases for Plugging Degraded SG Tubes."

The bases for these calculations are the analyses completed by the utility industry and reported to the NRC in the EPRI Draft Report TR-100407, "PWR Steam Generator Tube Repair Limits - Technical Support Document for Outside Diameter Stress Corrosion Cracking at Tube Support Plates", Revision 1, August 1993. As explained in this document, the analyses have been completed to assure that the general design criteria and the requirements of RG 1.121 are met during plant operation.

In the preparation of these industry documents and the Braidwood Unit 1 specific WCAP-14046, all analyses for leakage and burst potential were completed using the extremely conservative assumption that all Outside Diameter Stress Corrosion Cracking (ODSCC) indications occur on the tubing freespan. In fact, as indicated in both WCAP-14046 and EPRI Draft Report TR-100407, ODSCC degradation is confined to the region of the tube/TSP intersection. The burst capability of a section of tube containing ODSCC indications and located within the tube/TSP intersection substantially exceeds the burst capability of a freespan tube section without ODSCC indications. Therefore, tubing left in service by Braidwood's Unit 1 IPC amendment will not burst when confined by the tube support plates.

In fact, it is highly unlikely that a section of tubing within the tube support plate will leak, even with through wall cracks.

To assure structural integrity of the tubing, even during a MSLB accident, ComEd undertook extensive analyses, presented as part of WCAP-14046, to show analytically that the TSPs do not move far enough during a MSLB to allow degraded tubes to uncover, and subsequently, result in increased leakage.

A Generic Model D-4 SG Limited Support Plate Motion Analysis is also being conducted and should be submitted to the NRC by the end of August, 1994.

This analysis is being performed using the following assumptions:

1. The TSP crevices are clean,
2. The TSPs are free to move, depending on applied loads, along the length of the SG tube, and
3. Movement of the TSPs along the length of the tube is not restricted by bending or distortion of the SG tube hole.

Each of these base assumptions is extremely conservative in its own right:

1. For assumption 1, visual inspections of the secondary side of the tube bundles of Braidwood Unit 1 SGs show some quantity of deposits in the tube to TSP crevice, and along the length of the tube. Since these deposits are considered to be a possible factor in causing ODSCC, it is likely that any tube having ODSCC indications has deposits in the tube/TSP intersection. These deposits would tend to close the tube to TSP crevice, restricting by friction the ability of the TSP to move along the tubes as loads are applied to the TSP during a MSLB.

2. With regards to assumptions 2 and 3, the TSPs tend to flex and in some locations, are constrained by tie-rods and wedges attached to the tube bundle shroud. These constraints tend to cause the TSPs to ripple under the applied loads as indicated in WCAP-14046. This effect tends to distort the shape of the tube holes, which are fitted to a tight tolerance around the tubes. Therefore, any distortion of these tube holes caused by motion of the TSP will tend to cause the TSP to bind against the outside diameter of the tube, further constraining its movement away from the degraded area of the tubing.

The impact of these facts will lessen the ability of the TSP to move, thereby significantly reducing the possibility that a degraded section of tubing would become uncovered during a MSLB.

Thus, this proposed license amendment request does not result in any increase in the probability or consequences of an accident previously evaluated within the Braidwood Updated Final Safety Analysis Report (UFSAR).

2. **The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.**

Approval of this proposed change does not introduce any significant changes to the plant design basis. Removal of the Amendment 50 and SER operating limits for Unit 1 does not provide a mechanism which could result in a new or different kind of accident. Neither a single or multiple tube rupture event would be expected in a SG in which the IPC has been applied.

ComEd has implemented a maximum leakage rate limit of 150 gallons per day (gpd) through any one SG to help preclude the potential for excessive leakage during all plant conditions. The RG 1.121 criterion for establishing operational leakage rate limits that require plant shutdown are based upon leak-before-break considerations to detect a free span crack before potential tube rupture during faulted plant conditions. The 150 gpd limit will provide for leakage detection and plant shutdown in the event of the occurrence of an unexpected single crack resulting in leakage that is associated with the longest permissible free span crack length. Since tube burst is precluded during normal operation due to the proximity of the TSP to the tube and the potential exists for the crevice to become uncovered during MSLB conditions, the leakage from the maximum permissible crack must preclude tube burst at MSLB conditions. Thus, the 150 gpd limit provides for plant shutdown prior to reaching critical crack lengths for MSLB conditions.

As SG tube integrity will continue to be maintained upon approval of this amendment request through inservice inspection and primary-to-secondary leakage monitoring, the possibility of a new or different kind of accident from any previously evaluated is not created.

3. **The proposed change does not involve a significant reduction in a margin of safety.**

Braidwood Unit 1 TS Amendment 50 imposed a 100 calendar days with T_{hot} greater than 500°F operating limit on Unit 1. This limitation was a consequence of the amount of MSLB leakage predicted in Braidwood Station's April 30, 1994, submittal. These predictions were made using the Log-Logistic method of draft NUREG 1477, with the Dose Equivalent Iodine-131 limit of Specification 3.4.8 reduced from 1.0 μCi/gm to 0.35 μCi/gm. However, WCAP 14046, docketed June 10, 1994, as required by Braidwood Station's April 25, 1994, submittal, has shown using the EPRI Leakrate Correlation that projected EOC-5 MSLB leakage is 3.1 gpm which is less than the allowable limit of 9.1 gpm for Braidwood Unit 1. This analysis is discussed in detail in WCAP-14046.

Thus the Unit 1 operating limit imposed by Amendment 50 on the basis of MSLB leakage is no longer required.

In addition to the 100 day, leakage based limit, the Nuclear Regulatory Commissions (NRC) Safety Evaluation Report (SER), issued May 7, 1994, in support of Braidwood Station's Unit 1 TS Amendment 50 discusses a 4.6 month (138 day) limit derived from a deterministic assessment of SG tube burst probability. To address the issue of tube burst for full cycle operation Braidwood Station's April 25, 1994, submittal provided a probabilistic risk assessment which is restated below.

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ComEd has evaluated the impact of operation using the proposed interim plugging criteria against the results of insights from the draft Braidwood IPE. Braidwood Station is scheduled to docket its IPE June 30, 1994. Byron Station's IPE was docketed April 20, 1994. The SG sections of these documents are identical. While the Braidwood IPE is not in its final form, it is believed that the quantification in hand is sufficiently robust to allow a validation assessment of the impact of such operation. The ComEd evaluation parallels that described in the NRC Staff's SER for Palo Verde Unit 2 dated August 19, 1993.

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Therefore, the operation of Braidwood Unit 1 Cycle 5 for a complete 18 month fuel cycle with the application of the one volt IPC does not significantly increase the core damage frequency even with the conservative assumption of a POD of 0.6 and application of the full growth rate distribution observed during Cycle 4.

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The impact of these facts will lessen the ability of the TSP to move, thereby significantly reducing the possibility that a degraded section of tubing would become uncovered during a MSLB.

This evidence, in conjunction with the probability of occurrence of a MSLB, and the probabilistic assessment of the consequences of a MSLB, results in the substantially increased assurance that the

> consequences of a MSLB will be significantly less severe than those assessed in WCAP-14046 and the generic Model D-4 SG Limited Support Plate Motion Analyses.
>
> Thus, this proposed change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the pertinent portions of the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. This staff finding is partially based on the licensee's usage of a constant value for the Probability of Detection (POD) of 0.6 as recommended in draft NUREG-1447. This is consistent with the staff's position in the Safety Evaluation (SE) it issued in support of Amendment No. 50 to the Braidwood, Unit 1, operating license. While the licensee also discussed in its analysis the usage of a higher value for the POD, the staff did not rely on this. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

The Commission is seeking public comments on this proposed determination. Any comments received within 30 days after the date of publication of this notice will be considered in making any final determination.

Normally, the Commission will not issue the amendment until the expiration of the 30-day notice period. However, should circumstances change during the notice period such that failure to act in a timely way would result, for example, in derating or shutdown of the facility, the Commission may issue the license amendment before the expiration of the 30-day notice period, provided that its final determination is that the amendment involves no significant hazards consideration. The final determination will consider all public and State comments received. Should the Commission take this action, it will publish in the FEDERAL REGISTER a notice of issuance and provide for opportunity for a hearing after issuance. The Commission expects that the need to take this action will occur very infrequently.

Written comments may be submitted by mail to the Rules Review and Directives Branch, Division of Freedom of Information and Publications Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and should cite the publication date and page number of this FEDERAL REGISTER notice. Written comments may also be delivered to Room 6D22, Two White Flint North, 11545 Rockville Pike, Rockville Maryland, from 7:30 a.m. to 4:15 p.m. Federal workdays. Copies of written comments received may be examined at the NRC Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC 20555.

The filing of requests for hearing and petitions for leave to intervene is discussed below.

By August 10, 1994 , the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's "Rules of Practice for Domestic Licensing Proceedings" in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC 20555 and at the local public document room located at Wilmington Township Public Library, 201 S. Kankakee

Street, Wilmington, Illinois 60481. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of hearing or an appropriate order.

As required by 10 CFR 2.714, a petition for leave to intervene shall set forth with particularity the interest of the petitioner in the proceeding, and how that interest may be affected by the results of the proceeding. The petition should specifically explain the reasons why intervention should be permitted with particular reference to the following factors: (1) the nature of the petitioner's right under the Act to be made party to the proceeding; (2) the nature and extent of the petitioner's property, financial, or other interest in the proceeding; and (3) the possible effect of any order which may be entered in the proceeding on the petitioner's interest. The petition should also identify the specific aspect(s) of the subject matter of the proceeding as to which petitioner wishes to intervene. Any person who has filed a petition for leave to intervene or who has been admitted as a party may amend the petition without requesting leave of the Board up to 15 days prior to the first prehearing conference scheduled in the proceeding, but such an amended petition must satisfy the specificity requirements described above.

Not later than 15 days prior to the first prehearing conference scheduled in the proceeding, a petitioner shall file a supplement to the petition to intervene which must include a list of the contentions which are sought to be litigated in the matter. Each contention must consist of a specific statement of the issue of law or fact to be raised or controverted. In addition, the petitioner shall provide a brief explanation of the bases of the contention and a concise statement of the alleged facts or expert opinion which support the contention and on which the petitioner intends to rely in proving the contention at the hearing. The petitioner must also provide references to those specific sources and documents of which the petitioner is aware and on which the petitioner intends to rely to establish those facts or expert opinion. Petitioner must provide sufficient information to show that a genuine dispute exists with the applicant on a material issue of law or fact. Contentions shall be limited to matters within the scope of the amendment under consideration. The contention must be one which, if proven, would entitle the petitioner to relief. A petitioner who fails to file such a supplement which satisfies these requirements with respect to at least one contention will not be permitted to participate as a party.

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing, including the opportunity to present evidence and cross-examine witnesses.

If a hearing is requested, the Commission will make a final determination on the issue of no significant hazards consideration. The final determination will serve to decide when the hearing is held.

If the final determination is that the amendment request involves no significant hazards consideration, the Commission may issue the amendment and make it immediately effective, notwithstanding the request for a hearing. Any hearing held would take place after issuance of the amendment.

If the final determination is that the amendment request involves a significant hazards consideration, any hearing held would take place before the issuance of any amendment.

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Docketing and Services Branch, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC 20555, by the above date. Where petitions are filed during the last 10 days of the notice period, it is requested that the petitioner promptly so inform the Commission by a toll-free telephone call to Western Union at 1-(800) 248-5100 (in Missouri 1-(800) 342-6700). The Western Union operator should be given Datagram Identification Number N1023 and the following message addressed to Robert A. Capra: petitioner's name and telephone number, date petition was mailed, plant name, and publication date and page number of this FEDERAL REGISTER notice. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and to Michael I. Miller, Esquire; Sidley and Austin, One First National Plaza, Chicago, Illinois 60690, attorney for the licensee.

Nontimely filings of petitions for leave to intervene, amended petitions, supplemental petitions and/or requests for hearing will not be entertained absent a determination by the Commission, the presiding officer or the presiding Atomic Safety and Licensing Board that the petition and/or request should be granted based upon a balancing of the factors specified in 10 CFR 2.714(a)(1)(i)-(v) and 2.714(d).

For further details with respect to this action, see the application for amendment dated June 20, 1994, which is available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC 20555 and at the local public document room located at Wilmington Township Public Library, 201 S. Kankakee Street, Wilmington, Illinois 60481.

Dated at Rockville, Maryland, this 1st day of July 1994.

FOR THE NUCLEAR REGULATORY COMMISSION

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