December 15, 1997

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

SUBJECT: BYRON STATION, UNITS 1 AND 2; AND BRAIDWOOD STATION, UNITS 1 AND 2, PRIMARY CONTAINMENT AND REACTOR COOLANT VOLUME (TAC NOS. M97851, M97852, M97853, AND M97854)

Dear Ms. Johnson:

The Commission has forwarded the enclosed, "Notice of Consideration of Issuance of Amendments to Facility Operating Licenses, Proposed No Significant Hazards Consideration Determination and Opportunity for a Hearing," to the Office of the Federal Register for publication.

This notice relates to the Commonwealth Edison Company (ComEd) submittal of January 30, 1997, as revised on December 9, 1997, regarding the plants' technical specifications. The changes are related to changes in the reactor coolant volume and the calculated peak containment pressure resulting from the increased volume in the event of an accident.

The staff's original "Proposed No Significant Hazards Consideration" was published in the Federal Register on April 23, 1997. However, the proposed revision of December 9, 1997, necessitates the publication of a revised Federal Register Notice.

Sincerely,

Original signed by:

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

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

DF01

Enclosure: As stated

cc w/encl: see next page

Distribution: Docket File, PUBLIC, PDIII-2 r/f, EAdensam, EGA1, RCapra, CMoore, GDick, OGC, O15B18, GHill, T5C3 [4], WBeckner, O13H15, ACRS, T2E26, MJordan, RIII

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OFFICIAL RECORD COPY

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

December 15, 1997

Ms. Irene Johnson, Acting Manager
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Commonwealth Edison Company
Executive Towers West III
1400 Opus Place, Suite 500
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Docket Nos: STN 50-454, STN 50-455,
STN 50-456 and STN 50-457

Enclosure: As stated

cc w/encl: see next page

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7590-01-P

UNITED STATES NUCLEAR REGULATORY COMMISSION

COMMONWEALTH EDISON COMPANY

DOCKET NOS. STN 50-454, STN 50-455, STN 50-456, AND STN 50-457

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

BYRON STATION, UNITS 1 AND 2 AND BRAIDWOOD STATION, UNITS 1 AND 2

The U.S. Nuclear Regulatory Commission (the Commission) is considering issuance of n amendment to Facility Operating License Nos. NPF-37, NPF-66, NPF-72, and NPF-77 issued to Commonwealth Edison Company (the licensee) for operation of the Byron Station, Units 1 and 2, located in Ogle County, Illinois and Braidwood Station, Units 1 and 2, located in Will County, Illinois.

The proposed amendment would revise technical specification (TS) 1.0, "Definitions", TS 3/4.6.1, "Primary Containment" and associated Bases; and TS 5.4.2, Reactor Coolant System Volume" for Byron and Braidwood to support the steam generator replacement for Unit 1 at each site. The replacement steam generators increase the reactor coolant system volume which results in a higher calculated peak containment pressure (Pa) value. The staff's proposed no significant hazards consideration determination for the requested change was published on April 23, 1997 (62 FR 19826).

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.**

Each of the RSGs has a larger RCS primary side volume than the original steam generators (OSGs). As a result of the RCS volume increase, the mass and energy release during the blowdown phase of the large break loss of coolant accident (LBLOCA) is increased. Additionally, the heat transfer rate of the RSGs is greater than the OSGs, and the RSGs will operate at a slightly higher pressure than that for the OSGs. Consequently, the steam enthalpy exiting the break during the reflood period, for the RSGs, will be greater than for the OSGs. This results in an increase in the containment building peak pressure, $P_a$.

The proposed revisions to the Technical Specifications involve the corrected value of the current Unit 1 and Unit 2 RCS volume and the incremental change in RCS volume for the RSGs. The proposed revisions also involve the defined value of Unit 1 $P_a$ following installation of the RSGs. Several editorial changes are also being made to improve clarity and consistency of the TS.

RCS volume is not an initiator for any event and an increase in volume does not affect any operating margin or requirements. Therefore, increasing the primary volume does not increase the probability of any event previously analyzed.

The current value of $P_a$ for Unit 2 is unchanged due to conservatism in the original analysis. The revised value of $P_a$ for Unit 1 continues to be less than the design basis pressure for the containment structure. The change represents only a revision to the containment test pressure for containment leakage testing. Such testing is only performed with the affected unit in the shutdown condition. Therefore, the proposed change in $P_a$ for Unit 1 does not involve a significant increase in the probability of an accident previously evaluated.

All accidents in the Updated Final Safety Analysis Report (UFSAR) were evaluated to determine the effect of an increase in primary volume on accident consequences. The events identified that may be impacted by an increase in primary volume are the Waste

Gas System Leak or Failure and LBLOCA. For the Waste Gas System Leak or Failure, the activity of the decay tank is controlled to Technical Specification limits which are unaffected by RCS volume. Therefore, an increase in RCS volume would not increase the offsite dose.

The offsite dose calculation for the LBLOCA is unaffected by the proposed change. The license basis offsite dose calculation is in accordance with NRC Reg Guide 1.4 "Assumptions Used for Evaluating The Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors." This Regulatory Guide states, in part, "...a number of appropriately conservative assumptions, based on engineering judgment and on applicable experimental results from safety research programs conducted by the AEC." These conservatisms include (but are not limited to) the following assumptions:

Twenty five percent of the equilibrium full power radioactive iodine inventory is immediately available for leakage from the primary containment. 100 % of the equilibrium full power radioactive noble gas inventory is immediately available for leakage from the primary containment. The primary containment should be assumed to leak at the (maximum) leak rate specified in the technical specifications for the first 24 hours and at 50% of this value for the remaining 29 days of the accident duration.

The design basis leakage corresponding to a peak containment pressure of 50 psig utilized in the design basis accident analysis is 0.10% per day of the containment free air mass. Therefore, the offsite dose calculation was performed with a leakage of .1 % per day for day one and .05 % per day for days 2 through 30. Isotopic inventories are unaffected by the increase in reactor coolant volume. Thus, the offsite dose is unaffected by the increase in the peak containment pressure. Therefore, this proposed change to $P_a$ does not involve a significant increase in the consequences of an accident previously evaluated.

The editorial changes proposed are for clarity and consistency within the Technical Specifications and do not affect either the probability or consequences of an accident previously evaluated.

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

The proposed change in RCS volume is a change in a plant parameter within the "Design Features" section of the Technical Specifications. Increasing the RCS volume does not create any new or different failure modes. The existing RCS design requirements continue to be met.

The revised value of $P_a$ for Unit 1 following replacement of steam generators continues to be less than the design basis pressure for the containment building structure. The change represents only a revision to the test pressure for containment leakage testing. Such testing is only performed with the affected unit in the shutdown condition. Therefore, no new or different failure modes are being introduced by modification of the testing parameters.

The editorial changes proposed are for clarity and consistency within the Technical Specifications and do not result in any physical changes to the facility or how it is operated. No new or different failure modes are being introduced by these changes.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Changing the RCS volume in the Technical Specifications does not reduce the margin of safety. RCS volume is a design feature. An evaluation of all UFSAR accidents was performed to determine the effect of an increase in RCS volume. This evaluation is summarized as follows:

An evaluation of the Chemical and Volume Control System Malfunction was performed to determine the effect of the increased RCS volume. The larger RCS volume reduces the reactivity insertion for a given dilution flow rate. Therefore, the UFSAR analyses remain bounding for Byron and Braidwood and there is no reduction in the margin of safety.

An evaluation of the Inadvertent Actuation of the Emergency Core Cooling System During Power Operation Event was performed to determine the effect of the increased RCS volume due to the RSGs. For this event, the injection of borated water causes a negative reactivity insertion, which increases DNBR. For a given Refueling Water Storage Tank (RWST) boron concentration, the larger RCS volume will cause a reduction in the negative reactivity insertion rate as compared to the current UFSAR analysis. However, negative reactivity would still be inserted and no fuel pins would experience DNB. Additionally, the increased RCS volume was evaluated to determine the effect on pressurizer level following the inadvertent actuation of ECCS and was found to be acceptable. Therefore, there is no reduction in the margin of safety.

An evaluation of the Small Break LOCA was performed to determine the effect of increased RCS volume. The additional RCS volume will cause a delay in the loop seal clearing which in turn delays the core uncovery as compared with the UFSAR analysis. A delay in core uncovery reduces the amount of core heatup which results in a lower peak clad temperature (PCT) because the core decay heat would be less than in the UFSAR analysis. The benefit is considered small, but there is still a benefit. Therefore, the increased RCS volume does not result in a reduction in the margin of safety.

An evaluation of the Large Break LOCA was performed to determine the effect of increased RCS volume for the RSGs. For a LB LOCA, the increased RCS volume causes the blowdown phase of the event to be longer. Increased blowdown phase, alone, could potentially result in a higher PCT. However, the RSGs also have less resistance to flow due to increased primary side steam generator flow area, which results in a higher blowdown flow compared to the OSGs. The increased blowdown flow will compensate for the longer blowdown phase associated with the increased RCS volume. The net effect is that the blowdown time (end of bypass) for the RSG will be the same or decrease compared to the OSG. Reduced resistance to break flow for the RSG compared to the OSG will result in a lower PCT for the RSG compared to the OSG.

The increase in the current value of RCS volume in Unit 2 is significantly less than the increase associated with the replacement of the steam generators in Unit 1. The small increase in the RCS volume will likely result in a slight increase in the blowdown period. This slight increase in the blowdown period will have no significant impact on the peak clad temperature (PCT) calculation for Unit 2. Any small changes in the PCT due to this small increase in the RCS volume can be easily accommodated for Unit 2 because of the significant margin in the PCT (over 100 degree) available to the Appendix K 10CFR50.46 acceptance criteria of 2200 °F. Therefore, there is no reduction in the margin of safety.

An evaluation of the Gas Waste System Leak or Failure was performed to determine the effect of the increased RCS volume. Because the activity of the decay tank is controlled within Technical Specification limits, an increase in RCS volume would not change the results of the event. Therefore, there is no reduction in the margin of safety.

An evaluation was performed to determine the effect of the increased RCS volume (associated with the RSGs) on the peak containment pressure following a LBLOCA. The increased RCS volume caused the peak containment pressure to increase to 47.8 psig. This is still below the containment design pressure of 50.0 psig. Therefore, there is no reduction in the margin of safety. The increase in RCS volume for the existing units (without RSGs) remains within the conservative volume used in the calculation of the current peak containment pressure value of 44.4 psig. Therefore, there is no reduction in the margin of safety.

This proposed change involves testing requirements designed to demonstrate acceptable leakage rates are maintained. If acceptable leakage rates are maintained as outlined in the Technical Specifications, there will be no reduction in the margin of safety. In the event of degradation of a containment seal that results in unacceptable leakage, plant shutdown will occur as required by Technical Specifications and administrative requirements in accordance with approved plant procedures. Therefore, this proposed change does not involve a significant reduction in a margin of safety. The editorial changes proposed are for clarity and consistency within the Technical Specifications and do not result in any physical changes to the facility or how it is operated. Therefore, the changes have no effect on the margin of safety.

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 Chief, Rules Review and Directives Branch, Division of Freedom of Information and Publications Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, 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.

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

By 01/20/97 , 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, and at the local public document room located for Byron, the Byron Public Library District, 109 N. Franklin, P.O. Box 434, Byron, Illinois 61010; for Braidwood, the Wilmington 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 20555-0001, Attention: Rulemakings and Adjudications Staff, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, by the above date. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, and to Michael I. Miller, Esquire; Sidley and Austin, One First National Plaza, Chicago, Illinois 60603, 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 January 30, 1997, as revised on December 9, 1997, which is available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document rooms: for Byron, located at the Byron Public Library District, 109 Franklin, P.O. Box 434, Byron, Illinois 61010; for Braidwood, the Wilmington Public Library, 201 S. Kankakee Street, Wilmington, Illinois 60481.

Dated at Rockville, Maryland, this 15th day of December, 1997.

FOR THE NUCLEAR REGULATORY COMMISSION

George F. Dick Jr.

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