September 21, 1994

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

SUBJECT: BYRON STATION, UNITS 1 AND 2, INTERIM PLUGGING CRITERIA (TAC NOS. M90052 AND M90053)

Dear Mr. Farrar:

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 applications of September 7, 1994, and September 17, 1994, (two letters) which supplemented your application of August 1, 1994. The proposed changes would revise the plant's technical specifications (TS) to incorporate a 1.0 volt steam generator tube interim plugging criteria for Unit 1, Cycle 7.

Sincerely,

Original signed by

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

Docket Nos. STN 50-454 and STN 50-455

Enclosure: Notice

cc w/encl: see next page

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NAME	CMoore		GDick:rc Rac for		~~MDLynch~~		RCapra Rac			
DATE	09/21/94		09/21/94		~~09/ /94~~		09/ /94		09/ /94	

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September 21, 1994

DOCKET NO. STN 50-454 and STN 50-455

MEMORANDUM TO: Rules Review and Directives Branch
Division of Freedom of Information and Publications Services
Office of Administration

FROM: Project Directorate III-2
Office of Nuclear Reactor Regulation

SUBJECT: COMMONWEALTH EDISON COMPANY - Byron Station, Units 1 & 2

One signed original of the *Federal Register* Notice identified below is attached for your transmittal to the Office of the Federal Register for publication. Additional conformed copies (5) of the Notice are enclosed for your use.

- [ ] Notice of Receipt of Application for Construction Permit(s) and Operating License(s).
- [ ] Notice of Receipt of Partial Application for Construction Permit(s) and Facility License(s): Time for submission of Views on Antitrust matters.
- [X] Notice of Consideration of Issuance of Amendment to Facility Operating License. (Call with 30-day insert date).
- [ ] Notice of Receipt of Application for Facility License(s); Notice of Availability of Applicant's Environmental Report; and Notice of Consideration of Issuance of Facility License(s) and Notice of Opportunity for Hearing.
- [ ] Notice of Availability of NRC Draft/Final Environmental Statement.
- [ ] Notice of Limited Work Authorization.
- [ ] Notice of Availability of Safety Evaluation Report.
- [ ] Notice of Issuance of Construction Permit(s).
- [ ] Notice of Issuance of Facility Operating License(s) or Amendment(s).
- [ ] Order.
- [ ] Exemption.
- [ ] Notice of Granting Exemption.
- [ ] Environmental Assessment.
- [ ] Notice of Preparation of Environmental Assessment.
- [ ] Receipt of Petition for Director's Decision Under 10 CFR 2.206.
- [ ] Issuance of Final Director's Decision Under 10 CFR 2.206.
- [ ] Other: ______________________

Robert A. Capra, Director, PDIII-2
Office of Nuclear Reactor Regulation

Attachment(s): As stated

Contact: G. Dick
Telephone: 504-3019

DOCUMENT NAME:

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DATE	9/21/94					

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September 21, 1994

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Nuclear Regulatory Services
Commonwealth Edison Company
Executive Towers West III, Suite 500
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Mr. D. L. Farrar
Commonwealth Edison Company

Byron Station
Unit Nos. 1 and 2

cc:

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Commonwealth Edison Company
Byron Station Manager
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Byron, Illinois 61010

UNITED STATES NUCLEAR REGULATORY COMMISSION

COMMONWEALTH EDISON COMPANY

DOCKET NOS. STN 50-454 AND STN 50-455

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 Nos. NPF-37 and NPF-66, issued to Commonwealth Edison Company (the licensee), for operation of Byron Station, Units 1 and 2, located in Ogle County, Illinois.

The proposed amendment would revise the technical specifications (TS) to incorporate a 1.0 volt steam generator tube interim plugging criteria (IPC) for Unit 1 beginning with Cycle 7, which has begun. This supplements the information that was published in the FEDERAL REGISTER on August 31, 1994 (59 FR 45019).

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.

   Consistent with Regulatory Guide (RG) 1.121, "Basis for Plugging Degraded PWR Steam Generator Tubes," Revision 0, August 1976, the traditional depth-based criteria for SG tube repair implicitly ensures that tubes accepted for continued service will retain adequate structural and leakage integrity during normal operating, transient, and postulated accident conditions. It is recognized that defects in tubes permitted to remain in service, especially cracks, occasionally grow entirely through-wall and develop small leaks. Limits on allowable primary-to-secondary leakage established in Technical Specifications ensure timely plant shutdown before the structural and leakage integrity of the affected tube is challenged.

   The proposed license amendment request to implement voltage amplitude [steam generator] SG tube support plate Interim Plugging Criteria for Byron Unit 1 Cycle 7 meets the requirements of RG 1.121. The IPC methodology demonstrates that tube leakage is acceptably low and tube burst is a highly improbable event during either normal operation or the most limiting accident condition, a postulated main steam line break (MSLB) event. Requesting a single cycle applicability is more conservative than the guidance contained in the draft Generic Letter on voltage-based repair criteria issued for comment on August 12, 1994.

   Adequate SG tube leakage integrity during normal operating conditions is assured by limiting allowable primary-to-secondary leakage to 150 gpd per SG or 600 gpd total. Currently, this limit is administratively controlled.

   However, a license amendment request was submitted on June 3, 1994, to incorporate this limit into the Byron Technical Specifications. During normal operating conditions, the tube support plate constrains the [outer diameter stress corrosion cracking] ODSCC affected area of the tube to provide additional strength that precludes burst. Any leakage of a tube exhibiting ODSCC at the [tube support plate] TSP is fully bounded by the existing SG tube rupture analysis included in the Byron [Updated Final Safety Analysis Report] UFSAR. Therefore, probability of failure of a tube left in service or consequences of tube failure during normal operating conditions is not significantly increased by the application of IPC.

During transients, the TSP is conservatively assumed to displace due to the thermal-hydraulic loads associated with the transient. This may partially expose a crack which is within the boundary of the TSP during normal operations to free span conditions. Burst is therefore conservatively evaluated assuming the crack is fully exposed to free span conditions. The structural eddy current bobbin coil voltage limit for free-span burst is 4.54 volts. This limit takes into consideration a 1.43 safety factor applied to the steam line break differential pressure that is consistent with RG 1.121 requirements. With additional considerations for growth rate assumptions and an upper 95% confidence estimate on voltage variability, the maximum voltage indication that could remain in service is reduced to 2.7 volts. For added conservatism, the allowable indication voltage is further reduced in the proposed amendment to a 1.0 volt confirmed ODSCC indication limit. All indications greater than 1.0 volt will be subject to an RPC examination. Tubes with RPC confirmed ODSCC indications will be plugged or sleeved. Any ODSCC indications between 1.0 volt and 2.7 volts which are not confirmed as ODSCC will be allowed to remain in service since these indications are not as likely to affect tube structural integrity or leakage integrity over the next operating cycle as the indications that are detectable by both bobbin and RPC inspections.

The eddy current inspection process has been enhanced to address RG 1.83, "Inservice Inspection of PWR Steam Generator Tubes," Revision 1, July 1975, consideration as well as the EPRI SG Inspection Guidelines. Enhancements in accordance with NUREG-1477 and Appendix A of the Catawba IPC report (WCAP-13698) are in place to increase detection of ODSCC indications and to ensure reliable, consistent acquisition and analysis of data. Based on the conservative selection of the voltage criteria and the increased ability to identify ODSCC, the probability of tube failure during an accident is also not significantly increased due to application of requested IPC.

For consistency with current offsite dose limits, the site allowable leakage limit during a MSLB has been conservatively calculated to be 12.8 gpm. This leakage limit includes maximum allowable operational leakage from the unaffected SGs and the accident leakage from the affected SG. As a requirement for operation following application of IPC, the projected distribution of crack indications over the operating period must be verified to result in primary to secondary accident leakage less than the site allowable leakage limit. Thus, the consequences of a MSLB remain unchanged.

For an unscheduled mid-cycle inspection as a result of leakage due to mechanisms other than ODSCC at support plates or some other cause, the ODSCC indication limit is represented by the following equation:

$$V < \frac{\frac{\Delta t}{CL}(V_{SL} - V_{BOC}) + V_{BOC}}{1 + (0.2)\left(\frac{\Delta T}{CL}\right)}$$

where:

V	=	measured voltage
V_{BOC}	=	voltage at BOC
Δt	=	time period of operation to unscheduled outage
CL	=	cycle length (full operating cycle length where operating cycle is the time between two scheduled steam generator inspections)
V_{SL}	=	4.5 volts

Assuming linear flaw growth from BOC to EOC and a maximum structural limit of 4.5 volts, the voltage expected for an identified flaw at any time in the cycle can be predicted. The allowed voltage limit for an unscheduled inspection, as identified by the equation given above, reduces the predicted straightline growth voltage to ensure conservatism in the limit. A flaw which has not exceeded the predicted voltage growth at any point in the cycle would not be expected to exceed the structural limit at end of cycle or negatively impact the burst probability calculated based on results from the last scheduled inspection. Therefore, it is acceptable to leave the tube in service.

Therefore, as implementation of the 1.0 volt IPC for Byron Unit 1 Cycle 7 does not adversely affect steam generator tube integrity and results in acceptable dose consequences, the proposed license amendment request does not result in any significant increase in the probability or consequences of an accident previously evaluated within the Byron Updated Final Safety Analysis Report.

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

   Implementation of the proposed SG tube IPC for Byron Unit 1 Cycle 7 does not introduce any significant changes to the plant design basis. Use of the criteria does not provide a mechanism which could result in an accident outside the tube support plate elevations since industry experience indicates that ODSCC originating within the tube support plate does not extend significantly beyond the thickness of the support plate. This criteria only applies to ODSCC contained within the region of the tube bounded by the tube support plate.

In addressing the combined effects of Loss of Coolant Accident (LOCA) coincident with a Safe Shutdown Earthquake (SSE) on the SG (as required by General Design Criteria 2), it has been determined that tube collapse of select tubes may occur in the SGs at some plants, including Byron Unit 1. There are two issues associated with SG tube collapse. First, the collapse of SG tubing reduces the RCS flow area through the tubes. The reduction in flow area increases the resistance to flow of steam from the core during a LOCA which, in turn, may potentially increase Peak Clad Temperature (PCT). Second, there is a potential that partial through-wall cracks in tubes could progress to through-wall cracks during tube deformation or collapse.

A number of tubes have been identified, in the "wedge" locations of the SG TSPs, that demonstrate the potential for tube collapse during a LOCA + SSE event. Because of this potential, these tubes have been excluded from application of the voltage-based SG TSP IPC.

Therefore, neither a single or multiple tube rupture event would be expected in a steam generator in which IPC has been applied.

ComEd has implemented a maximum primary to secondary leakage limit of 150 gpd through any one SG at Byron to help preclude the potential for excessive leakage during all plant conditions. The 150 gpd limit provides for leakage detection and plant shutdown in the event of an unexpected single crack leak associated with the longest permissible free span crack length. The 150 gpd limit provides adequate leakage detection and plant shutdown criteria in the event an unexpected single crack results 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 a conservative limit to prompt plant shutdown prior to reaching critical crack lengths under MSLB conditions.

Upon implementation of the 1.0 volt IPC for Byron Unit 1 Cycle 7, steam generator tube integrity continues to be maintained through inservice inspection and primary-to-secondary leakage monitoring. Therefore, 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.

   The use of the voltage based bobbin coil probe SG TSP IPC for Byron Unit 1 Cycle 7 will maintain steam generator tube integrity commensurate with the criteria of RG 1.121 as discussed above. Upon implementation of the criteria, even under the worst case conditions, the occurrence of ODSCC at the TSP elevations is not expected to lead to a steam generator tube rupture event during normal or faulted plant conditions. The distribution of crack indications at the TSP elevations result in acceptable primary-to-secondary leakage during all plant conditions and radiological consequences are not adversely impacted by the application of IPC.

   The installation of SG tube plugs and sleeves reduces the RCS flow margin. As noted previously, implementation of the SG TSP IPC will decrease the number of tubes which must be repaired by plugging or sleeving. Thus, implementation of IPC will retain additional flow margin that would otherwise be reduced due to increased tube plugging. Therefore, no significant reduction in the margin of safety will occur during Cycle 7 as a result of the implementation of this proposed license amendment request.

Although not relied upon to prove adequacy of the proposed amendment request, the following analyses demonstrate that significant conservatism exist in the methods and justifications described above:

LIMITED TUBE SUPPORT PLATE DISPLACEMENT

An analysis was performed to verify the extent of limited TSP displacement during accident conditions (MSLB). Application of minimum TSP displacement assumptions reduce the likelihood of a tube burst to negligible levels.

Consideration of limited TSP displacement would also reduce potential MSLB leakage when compared to the leakage calculated assuming free span indications.

PROBABILITY OF DETECTION

The Electric Power Research Institute (EPRI) Performance Demonstration Program analyzed the performance of approximately 20 eddy current data analysts evaluating data from a unit with 3/4" inside diameter and 0.049" wall thickness tubes. The results of this analysis clearly show that the detectabilty of larger voltage indications is increased which lends creditability for application of a POD of > 0.6 for ODSCC indications larger than 1.0 volt.

RISK EVALUATION OF CORE DAMAGE

As part of ComEd's evaluation of the operability of Byron Unit 1 Cycle 7, a risk evaluation was completed. The objective of this evaluation was to compare core damage frequency under containment bypass conditions, with and without the interim plugging criteria applied at Byron Unit 1.

The total Byron core damage frequency is estimated to be 3.09E-5 per reactor year with a total contribution from containment bypass sequences of 3.72E-8 per reactor year according to the results of the current individual plant evaluation (IPE). Operation with the requested IPC resulted in an insignificant increase in core damage frequency resulting from MSLB with containment bypass conditions.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. 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, 11455 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 October 24, 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 the Byron Public Library, 109 N. Franklin, Byron, Illinois 61010. 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 Mr. 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; Sidney 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 applications for amendment dated September 7, 1994, and September 17, 1994, (two letters) which are 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 the Byron Public Library, 109 N. Franklin, P.O. Box 434, Byron, Illinois 61010.

Dated at Rockville, Maryland, this 21st day of September 1994.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Capra

Robert A. Capra, Director
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation