UNITED STATES NUCLEAR REGULATORY COMMISSION

COMMONWEALTH EDISON COMPANY

DOCKET NOS. STN 50-456 AND STN 50-457

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

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

The proposed amendments would effectively renew the present voltage-based repair criteria in the Braidwood, Unit 1, Technical Specifications (TS) which were added to the existing steam generator (SG) tube repair criteria by License Amendment No. 54, issued on August 18, 1994. The differences between the present repair criteria in the Braidwood, Unit 1, TSs and those in the pending request to continue their use, are discussed below. The need to take action on this matter arises partly from the limit placed on the use of the present voltage-based criteria for only one operating cycle when the license amendment cited above was issued.

The voltage-based repair criteria in the subject TSs are applicable only to a specific type of SG tube degradation which is predominantly axially-oriented outer diameter stress corrosion cracking (ODSCC). This particular form of SG tube degradation occurs entirely within the intersections of the SG tubes with the tube support plates (TSP).

The need to effectively renew the present voltage-based SG tube repair criteria is also predicated on the possibility that the NRC staff may not find acceptable, a pending request for license amendments dated September 1, 1995, for the Byron and Braidwood Stations in sufficient time to be applicable for the forthcoming refueling outage for Braidwood, Unit 1, presently scheduled to start on September 30, 1995.

This request for a 3.0 volt lower voltage limit was first submitted on February 13, 1995, and was subsequently superseded by requests for license amendments submitted on July 7, 1995, and September 1, 1995. All three of these requests for license amendments propose to raise the present value of the lower voltage repair limit from 1.0 volt to 3.0 volts. The license amendment request dated September 1, 1995, supersedes the prior two requests on this matter in their entirety.

The license amendment request dated September 1, 1995, is under active review by the staff; however, a number of technical issues associated with this pending revision to the present TSs may require considerable time to resolve. In the event that the staff is not able to resolve these outstanding technical issues prior to the repair of the Braidwood, Unit 1, SG tubes presently scheduled to start on or about October 15, 1995, the licensee proposes in its request dated August 15, 1995, to adopt the SG tube repair criteria contained in Generic Letter (GL) 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking," dated August 3, 1995.

The SG tube voltage-based repair criteria presently in the Braidwood, Unit 1, TSs differ slightly from those proposed in the licensee's submittal dated August 15, 1995, in that the present repair criteria in the TSs were similar to those in the draft generic letter on the issue of ODSCC published by the staff on August 12, 1994, while the pending proposal is consistent with GL 95-05. This generic letter contains repair criteria slightly different from those contained in the earlier draft version. These differences reflect the staff's further review of this matter, including a review of comments by industry and the public.

In summary, the request for license amendments dated August 15, 1995, to adopt the voltage-based repair criteria in GL 95-05 will be considered by the staff only in the event that the pending request to raise the lower voltage limit from 1.0 volt to 3.0 volts can not be addressed in a timely manner.

While the voltage-based repair criteria for ODSCC flaws are applicable only to Braidwood, Unit 1, the pending request for license amendments involves both units in that the Braidwood Station has a set of TSs applicable to both units. Before issuance of the proposed license amendments, 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 amendments 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 SG tube support plate APC for Braidwood Unit 1 meets the requirements of RG 1.121. The APC 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.

   During transients, the tube support plate (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 4.75 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 given by the upper voltage repair limit equation in Generic Letter 95-05. 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 outside diameter stress corrosion cracking (ODSCC) indications will be plugged or sleeved. Any ODSCC indications between 1.0 volt and the upper voltage repair limit 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 rotating pancake coil (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, considerations as well as the EPRI SG Inspection Guidelines. Enhancements in accordance with Generic Letter 95-05 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 APC.

Modification of the Braidwood Specifications for conformance with Generic Letter 95-05 requirements does not impact any accidents previously evaluated. The decrease in the allowed burst probability from 2.5×10^{-2} to 1.0×10^{-2} is conservative.

Calculations conducted for Braidwood have shown that the resulting 2-hour doses at the site boundaries will not currently exceed an appropriately small fraction of 10 CFR 100 dose guideline values in conjunction with the predicted MSLB leakage calculated in accordance with this submittal and a DE I-131 level of 1.0 μCi/gm. The site allowable leakage calculated using a DE I-131 level of 1.0 μCi/gm is 9.4 gallons per minute (gpm). This leakage includes accident leakage and the allowed 0.1 gpm primary-to-secondary leakage of the 3 unfaulted SGs per TS 3.4.6.2.c. However, in order to provide a defense in depth approach to application of this requested APC and to envelope any future increases in MSLB leakage due to tube degradation, Braidwood is lowering the RCS DE I-131 levels to 0.35 μCi/gm for all future cycles until SG replacement. The site allowable leak rate calculated using 0.35 μCi/gm DE I-131 is 26.8 gpm. This leakage also includes accident leakage and the allowed 0.1 gpm primary-to-secondary leakage of the 3 unfaulted SGs per TS 3.4.6.2.c. Lowering the limit to 0.35 μCi/gm DE I-131 is conservative and will not increase the probability or consequences of any accidents previously evaluated.

Renewal of the 1.0 volt IPC for Braidwood Unit 1 does not adversely affect steam generator tube integrity and results in acceptable dose consequences. Therefore, the proposed license amendment request does not result in any significant increase in the probability or consequences of an accident previously evaluated within the Braidwood 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.

Renewal of the proposed SG tube APC for Braidwood Unit 1 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. Therefore, neither a single or multiple tube rupture event would be expected in a steam generator in which APC has been applied.

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 Braidwood 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 APC.

ComEd has implemented a maximum primary to secondary leakage limit of 150 gallons per day (gpd) through any one SG at Braidwood 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.

Calculations conducted for Braidwood have shown that the resulting 2-hour doses at the site boundaries will not currently exceed an appropriately small fraction of 10 CFR 100 dose guideline values in conjunction with the predicted MSLB leakage calculated in accordance with this submittal and a DE I-131 level of 1.0 μCi/gm. The site allowable leakage calculated using a DE I-131 level of 1.0 μCi/gm is 9.4 gpm. This leakage includes accident leakage and the allowed 0.1 gpm primary-to-secondary leakage of the 3 unfaulted SGs per TS 3.4.6.2.c. However, in order to provide a defense in depth approach to application of this requested APC and to envelope any future increases in MSLB leakage due to tube degradation, Braidwood is lowering the RCS DE I-131 levels to 0.35 μCi/gm for all future cycles until SG replacement. The site allowable leak rate calculated using 0.35 μCi/gm DE I-131 is 26.8 gpm. This leakage also includes accident leakage and the allowed 0.1 gpm primary-to-secondary leakage of the 3 unfaulted SGs per TS 3.4.6.2.c. Lowering the Braidwood Unit 1 RCS DE I-131 concentration limit to the 0.35 μCi/gm is conservative and will not introduce any changes to the design basis for Braidwood Station.

Modification of the Braidwood Specifications for conformance with Generic Letter 95-05 requirements will not alter the plant design basis. The decrease in the allowed burst probability from 2.5×10^{-2} to 1.0×10^{-2} is conservative.

Upon renewal of the 1.0 volt APC for Braidwood Unit 1, 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 APC for Braidwood Unit 1 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 results in acceptable primary-to-secondary leakage during all plant conditions and radiological consequences are not adversely impacted by the application of APC.

The installation of SG tube plugs and sleeves reduces the RCS flow margin. As noted previously, renewal of the SG TSP APC will decrease the number of tubes which must be repaired by plugging or sleeving. Thus, renewal of APC 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 as a result of this proposed license amendment request.

Although not relied upon to prove adequacy of the proposed amendment request, the following analyses demonstrate that significant conservatisms 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 provides conservatism and reduces 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 0.043" wall thickness tubes. The results of this analysis clearly show that the detectability 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 Braidwood Unit 1, a risk evaluation was completed. The objective of this evaluation was to compare core damage frequency under containment bypass conditions, with and without the APC applied at Braidwood Unit 1. The total Braidwood 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 APC resulted in an insignificant increase in core damage frequency resulting from MSLB with containment bypass conditions.

Calculations conducted for Braidwood have shown that the resulting 2-hour doses at the site boundaries will not currently exceed an appropriately small fraction of 10 CFR 100 dose guideline values in conjunction with the predicted MSLB leakage calculated in accordance with this submittal and a DE I-131 level of 1.0 μCi/gm. The site allowable leakage calculated using a DE I-131 level of

1.0 µCi/gm is 9.4 gpm. This leakage includes accident leakage and the allowed 0.1 gpm primary-to-secondary leakage of the 3 unfaulted SGs per TS 3.4.6.2.c. However, in order to provide a defense in depth approach to application of this requested APC and to envelope any future increases in MSLB leakage due to tube degradation, Braidwood is lowering the RCS DE I-131 levels to 0.35 µCi/gm for all future cycles until SG replacement. The site allowable leak rate calculated using 0.35 µCi/gm DE I-131 is 26.8 gpm. This leakage also includes accident leakage and the allowed 0.1 gpm primary-to-secondary leakage of the 3 unfaulted SGs per TS 3.4.6.2.c. Lowering the Braidwood Unit 1 RCS DE I-131 concentration limit to the 0.35 µCi/gm is conservative and will not introduce any changes to the design basis for Braidwood Station. Thus this change is in conformance with Braidwood's current TS and does not involve a reduction in a margin of safety.

Modification of the Braidwood Specifications for conformance with Generic Letter 95-05 requirements will not reduce any safety margins. The decrease in the allowed burst probability from 2.5×10^{-2} to 1.0×10^{-2} is conservative.

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 amendments 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 amendments before the expiration of the 30-day notice period, provided that its final determination is that the amendments involve 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.

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

By November 6 , the licensee may file a request for a hearing with respect to issuance of the amendments to the subject facility operating licenses 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 at 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 amendments 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 amendments.

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 amendments.

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, 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; 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 amendments dated August 15, 1995, 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 room located at the Wilmington Public Library, 201 S. Kankakee Street, Wilmington, Illinois 60481.

Dated at Rockville, Maryland, this 29th day of September 1995.

FOR THE NUCLEAR REGULATORY COMMISSION

George F. Dick

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

September 29, 1995

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

SUBJECT: BRAIDWOOD STATION, UNITS 1 AND 2, INCREASED VOLTAGE LIMIT FOR STEAM GENERATOR TUBE VOLTAGE-BASED REPAIR CRITERIA (TAC NOS. M91671 AND M91672)

Dear Mr. Farrar:

The U.S. Nuclear Regulatory 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 your application of August 15, 1995. This application to revise the Braidwood, Unit 1, Technical Specifications (TSs) proposes to continue to use the voltage-based repair criteria which were added to the Braidwood, Unit 1, TSs by a license amendment issued on August 18, 1994. This August 15, 1995, request will be considered by the staff only in the event that the staff can not reach a timely decision on your pending request for license amendments dated September 1, 1995, to raise the present lower voltage repair limit from 1.0 volt to 3.0 volts.

Sincerely,

Original signed by
George F. Dick for:

M. David Lynch, Senior Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

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

Enclosure: Notice

cc w/encl: See next page

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