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 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-37, NPF-66, NPF-72, and NPF-77, issued to Commonwealth Edison Company 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 amendments would revise the present voltage-based repair criteria in the Byron 1 and Braidwood 1 Technical Specifications (TSs). These proposed revisions would raise the lower voltage limit from its present value of 1.0 volt to 3.0 volts; there would no longer be an upper voltage limit.

The Braidwood 1 TSs were revised by License Amendment No. 54, issued on August 18, 1994, to add voltage-based repair criteria to the existing steam generator (SG) tube repair criteria. The Byron 1 TSs were revised in a similar manner by License Amendment No. 66, issued on October 24, 1994.

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 (TSPs).

The present voltage values for the ODSCC repair criteria are based on the assumption of a "free span" exposure of the SG tube flaw; i.e., no credit is given for any constraint against burst or leakage, which may be provided by the presence of the TSPs. This approach is, in turn, based on the assumption that under postulated accident conditions, the TSPs may be displaced sufficiently by blowdown hydrodynamic loads such that a SG tube flaw which was fully confined within the thickness of the TSP prior to the accident would then be fully exposed. This approach was first advanced by the NRC staff in a draft generic letter issued on August 12, 1994, which was subsequently modified slightly and issued as 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 previous license amendments related to the issue of ODSCC were based to a large extent on the draft generic letter cited above.

The fundamental difference between the pending proposal to raise the lower voltage repair limit to 3.0 volts and the methodology contained in GL 95-05, is that the licensee proposes to install certain modifications to the SG internal structures, thereby limiting to a small value, the maximum displacement of the TSPs under accident conditions. The proposed structural modifications consist of expanding a limited number of SG tubes only on the hot leg side of the TSP, at each of the intersections of the tubes with the TSPs. The purpose of this approach would be to greatly reduce the probability of SG tube burst under postulated accident conditions by several orders of magnitude. There would be a negligible impact on the primary-to-secondary SG tube leakage under accident conditions.

While the voltage-based repair criteria for ODSCC flaws are applicable only to Byron 1 and Braidwood 1, the pending request for license amendments involves all four units in that both stations have a common set of TSs.

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.
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> The previously evaluated accidents of interest are steam generator tube burst and main steam line break [MSLB]. Their potential impact on public health and safety due to the change in SG tube plugging criteria proposed in this amendment request is very low as discussed below. Tube burst related to the types of cracks under consideration is precluded during normal operating plant conditions since the tube support plates are adjacent to the degraded regions of the tube in the tube to tube support plate crevices.
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> During accident conditions, i.e., MSLB, the tubes and TSP may move relative to each other, which can expose a crack length portion to

freespan conditions. Testing has shown that the burst pressure correlates to the crack length that is exposed to the freespan, regardless of the length that is still contained within the TSP bounds.

Therefore, a more appropriate methodology has been established for addressing leakage and burst considerations that is based on limiting potential TSP displacements during postulated MSLB events, thus reducing the freespan exposed crack length to minimal levels. The tube expansion process to be employed in conjunction with this TS change is designed to provide postulated TSP displacements that result in negligible tube burst probabilities due to the minimal freespan exposed crack lengths.

Thermal hydraulic modeling was used to determine TSP loading during MSLB conditions. A safety factor was conservatively applied to these loads to envelope the collective uncertainties in the analyses. Various operating conditions were evaluated and the most limiting operating condition was used in the analyses. Additional models were used to verify the thermal hydraulic results.

Assessment of the tube burst probability was based on a conservative assumption that all hot-leg TSP intersections (32,046) contained throughwall cracks equal to the postulated displacement and that the crack lengths were located within the boundaries of the TSP. Alternatively, it was assumed that all hot-leg TSP intersections contained throughwall cracks with length equal to the thickness of the TSP. The postulated TSP motion was conservatively assumed to be uniform and equal to the maximum displacement calculated.

The total burst probability for all 32,046 throughwall indications given a uniform MSLB TSP displacement of 0.31" is calculated to be $1x10^{-5}$. This is a factor of 1000 less than the Generic Letter 95-05 burst probability limit of $1x10^{-2}$. Therefore, the functional design criteria for tube expansion is to limit the TSP motion to 0.31" or less. However, the design goal for tube expansion limits the TSP MSLB motion to less than 0.1", which results in a total tube burst probability of $1x10^{-10}$ for all 32,046 postulated throughwall indications. Additional tubes will be expanded to provide redundancy to the required expansions.

The structural limit for the hot-leg SG tube repair criteria with tube expansion is based on axial tensile loading requirements to preclude axial tensile severing of the tube. Axially oriented ODSCC does not significantly impact the axial tensile loading of the tube, therefore, the more limiting degradation mode with respect to affecting the tube structural limit at TSPs is cellular corrosion. Tensile tests that measure the force required to sever a tube with cellular corrosion and uncorroded cross sectional areas are used to establish the lower bound structural limit. Based upon these tests, a lower bound 95% confidence level structural voltage limit of 37 volts was established for cellular corrosion. This limit meets the Regulatory Guide (RG) 1.121, "Basis for Plugging Steam Generator Tubes," structural requirements based upon the normal operating pressure differential with a safety factor of 3.0 applied. Due to the limited database supporting this value, the structural limit was conservatively reduced to 20 volts. Accounting for voltage growth and Non-Destructive Examination (NDE) uncertainty, the full [interim plugging criteria] IPC upper limit exceeds 10 volts. However, for added conservatism a single voltage repair limit for hot-leg indications is specified in this request. All hot-leg indications with bobbin coil probe voltages greater than the hot-leg voltage repair limit will be plugged or repaired.

The freespan tube burst probability must be calculated for the cold-leg TSP indications to be within the requirements of Generic Letter 95-05. The freespan structural voltage limit is calculated using correlations from the database described in Generic Letter 95-05, with the inclusion of the recent Byron and Braidwood tube pull results. This structural limit is 4.75 volts. The lower voltage repair limit for cold-leg indications continues to be 1.0 volt. The upper voltage repair limit for cold-leg indications will be calculated in accordance with Generic Letter 95-05. Since flow distribution baffle indications are to be repaired to the 40% depth criteria, no leakage or burst analyses are required for these indications.

Per Generic Letter 95-05, MSLB leak rate and tube burst probability analyses are required prior to returning to power and are to be included in a report to the Nuclear Regulatory Commission (NRC) within 90 days of restart. If allowable limits on leak rates and burst probability are exceeded, the results are to be reported to the NRC and a safety assessment of the significance of the results is to be performed prior to returning the steam generators to service.

A postulated MSLB outside of containment but upstream of the Main Steam Isolation Valve (MSIV) represents the most limiting radiological condition relative to the IPC. The ODSCC voltage distribution at the TSP intersections are projected to the end of the cycle and MSLB leakage is calculated.

A site specific calculation has determined the allowable MSLB leakage limit for Byron Unit 1 and Braidwood Unit 1. These limits use the recommended dose equivalent Iodine-131 transient spiking values consistent with NUREG-0800, "Standard Review Plan" and ensure site boundary doses are within a small fraction of the

10 CFR 100 requirements. The projected MSLB leakage rate calculation methodology described in WCAP-14046, "Braidwood Unit 1 Technical Support for Cycle 5 Steam Generator Interim Plugging Criteria," and WCAP 14277, "SLB Leak Rate and Tube Burst Probability Analysis Methods for ODSCC at TSP Intersections," will be used to calculate end-of-cycle (EOC) leakage. This method includes a Probability Of Detection (POD) value of 0.6 for all voltage amplitude ranges and uses the accepted leak rate versus bobbin voltage correlation methodology (full Monte Carlo) for calculating leak rate, as described in Generic Letter 95-05. The database used for the leak and burst correlations is consistent with that described in Generic Letter 95-05 with the inclusion of the Byron Unit 1 and Braidwood Unit 1 tube pull results. The EOC voltage distribution is developed from the POD adjusted beginning-of-cycle (BOC) voltage distributions and uses Monte Carlo techniques to account for variances in growth and uncertainty.

The Electric Power Research Institute (EPRI) leak rate correlation has been used. It is based on free span indications that have burst pressures above the MSLB pressure differential. There is a low but finite probability that indications may burst at a pressure less than MSLB pressure. With limited TSP motion due to tube expansion, the tube is constrained by the TSP and tube burst is precluded. However, the flanks of the crack open up to contact the Inside Diameter (ID) of the TSP hole and result in a primary-to-secondary leak rate potentially exceeding that obtained from the EPRI correlation. This phenomenon is known as an Indication Restricted from Burst (IRB) condition.

ComEd has performed laboratory testing to determine the bounding leak rate obtainable in an IRB condition. The bounding leak rate value was then applied in a leak rate calculation methodology that accounts for the MSLB leak rate contribution from IRB indications to the total MSLB leak rate calculated as described above. Results indicate that the IRB contribution to the total leak rate value is negligible, however, ComEd will conservatively add a leakage contribution due to IRBs in addition to the leakage calculated in accordance with Generic Letter 95-05. When this is done, the dose at the site boundary resulting from the predicted leakage is shown to be a small fraction (less than 10%) of 10 CFR 100 limits.

Modification of the Byron and Braidwood Specifications for conformance with Generic Letter 95-05 requirements is primarily administrative and does not significantly increase the probability of any accidents previously evaluated. For Braidwood, the changes decrease the allowed burst probability from 2.5×10^{-2} to 1.0×10^{-2}. This change is in the conservative direction. Byron Station has previously incorporated this requirement.

In addition, defense in depth is provided by lowering the Unit 1 [reactor coolant system] RCS dose equivalent I-131 limit from 1.0 μCi/gm to 0.35 μCi/gm. Based on current predictions of MSLB leakage at the time of SG replacement, the lower RCS dose equivalent I-131 limit also ensures that the resulting 2-hour dose rates at the Braidwood and Byron site boundaries will not exceed an appropriately small fraction of 10 CFR 100 dose guideline values.

For these reasons, an increase in the IPC voltage repair limit to a maximum of 3.0 volts for the hot-leg support plate intersections does not adversely affect steam generator tube integrity and results in acceptable dose consequences. By effectively eliminating tube burst at hot-leg TSP intersections, the likelihood of a tube rupture is substantially reduced and the probability of occurrence of an accident previously evaluated is reduced.

This conclusion is not affected by recent foreign and domestic plant SG experiences. As the following evaluation shows, these experiences are not relevant to Byron and Braidwood. A foreign unit detected eddy current signal distortions in one area of the top tube support plate during a 1995 inspection. The steam generators had been chemically cleaned in 1992. Visual inspection showed that a small section of the top support plate had broken free and was resting next to the steam generator tube bundle wrapper. The support plate showed indications of metal loss. The chemical cleaning process used by the foreign unit was developed by the utility and differs significantly from the modified EPRI/SGOG process performed at Byron Unit 1 in 1994.

The foreign process, coupled with specific application of the process, resulted in tube support plate corrosion of up to 250 mils compared to a maximum of 2.16 mils (11 mils maximum allowed) measured at Byron. During the Byron eddy current inspection performed after the chemical cleaning, no distortion of the tube support plate signals was reported. Therefore, these differences in cleaning processes imply that this foreign experience is irrelevant to the effects of the chemical cleaning process on the TSPs at Byron.

A number of units have experienced TSP cracking associated with severe tube denting due to TSP corrosion at the tube to TSP crevice. WCAP 14273, Section 12.4, shows that a diametral reduction of 65 mils is required to develop stress levels above yield in the TSP ligaments at dented intersections. The bobbin voltage associated with a 1 mil radial dent is 20 to 25 volts.

Although, Byron Unit 1 and Braidwood Unit 1 have not seen corrosion induced denting, an appropriately sized bobbin probe will be used as a go/no-go gauge to assess hot-leg dents, if they occur in the future. If a tube has a dent at a hot-leg intersection that fails to pass the go/no-go test probe, cold-leg repair criteria will be applied to the affected tube and the adjacent tubes. In this way, any indications at these locations will be treated as free-span indications for the purposes of burst and leakage evaluation, which is bounded by the existing 1.0 volt IPC analysis. IPC repair limits will not be applied to tubes with dents > 5.0 volts since they could mask a 1.0 volt signal. Tubes with corrosion-induced dents > 5.0 volts and those tubes adjacent to such a tube will not be selected for tube expansion to preclude adverse effects of the failure of such a tube on limiting TSP displacement. Therefore, the denting experience at other plants is not relevant to Byron and Braidwood.

A foreign utility's steam generators have experienced cracking at the top tube support plate. The cause of the cracking appears to be the configuration of the single anti-rotation device, connected between the steam generator shell and wrapper, and the wrapper internals. The single anti-rotation device carries the full load associated with wrapper to shell motion. This rotational load is believed to be transferred to the TSP via the wrapper internals. The Byron/Braidwood Unit 1 steam generator design (D-4) uses three anti-rotation devices to spread the rotational load. The D-4 wrapper internals are configured such that this load is not directly transmitted to the TSP.

No top support plate cracking has been detected at Byron Unit 1 or Braidwood Unit 1 and very few (<1%) of the indications seen at Byron and Braidwood to date have been at the top TSP elevation.

Nevertheless, an analysis was performed to assess the impact of cracking of the top support plate. The results show an increase in top support plate deflection for a very limited number of tubes to greater than the 0.10" limit used in the 3.0 volt IPC analysis. The deflections of the lower support plates also increase, but remain within the 0.10" limit. Thus, hot-leg indications in a cracked top TSP continue to be bounded by the existing analysis. ComEd will develop an inspection plan for the SG internals to identify if indications detrimental to the load path exist. If the inspection determines that indications detrimental to the integrity of the load path necessary to support the 3 volt IPC are found, the results are to be reported to the NRC and a safety assessment of the significance of the results is to be performed prior to returning the steam generators to service.

A domestic utility reported several distorted TSP signals over the past three refueling outage tube inspections. It was determined that these signals were associated with the TSP geometry in an area where an access cover is welded into the TSP. These signal distortions are not attributed to TSP cracking or degradation. Since the distorted signals were due to TSP geometry which did not indicate or result in a defect of the TSP, there is no increase in the probability or consequences of an accident previously evaluated due to Byron Unit 1 and Braidwood Unit 1 steam generator TSP geometries which may result in distorted eddy current signals.

One foreign unit observed a dislocation of the tube bundle wrapper when they were unable to pass sludge lancing equipment through a handhole in the wrapper. The dislocation appears to be a result of improper attachment of the wrapper to the support structure. Steam generator sludge lance operations have been successfully performed on Byron Unit 1 and Braidwood Unit 1 which indicates that no problem with wrapper attachment exists. The foreign unit's wrapper support design is significantly different than that used on Byron Unit 1 and Braidwood Unit 1. Therefore, a similar wrapper dislocation will not occur and the foreign experience is not applicable to Byron and Braidwood.

Therefore, the proposed amendment does not result in any significant increase in the probability or consequences of an accident previously evaluated within the Byron Unit 1 and Braidwood Unit 1 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.

Implementation of the proposed steam generator tube plugging criteria with tube expansion 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 of the region of the tube support plate elevations as ODSCC does not extend beyond the thickness of the tube support plates. Neither a single nor multiple tube rupture event would be expected in a steam generator in which the plugging criteria has been applied.

The tube burst assessment involves a Monte Carlo simulation of the site specific voltage distribution to generate a total burst probability that includes the summation of the probabilities of 1 tube bursting, 2 tubes bursting, etc. For the hot-leg TSP intersections, the maximum total probability of burst, by design, is estimated to be 1×10^{-10} with all tube expansions functional.

Accounting for the unlikely event of expansion failures, a sufficient number of redundant expansions exist to ensure that the burst probability remains below 1×10^{-5}. This includes the conservative assumption that all 32,046 hot-leg TSP intersections contain throughwall indications. This level of burst probability is considered to be negligible when compared to the Generic Letter 95-05 limit of $1x10^{-2}$.

In addressing the combined effects of Loss Of Coolant Accident (LOCA) + Safe Shutdown Earthquake (SSE) on the SG as required by General Design Criteria (GDC) 2, it has been determined that tube collapse may occur in the steam generators at some plants. The tube support plates may become deformed as a result of lateral loads at the wedge supports located at the periphery of the plate due to the combined effects of the LOCA rarefaction wave and SSE loadings. The resulting pressure differential on the deformed tubes may cause some of the tubes to collapse. 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 throughwall cracks in tubes could progress to throughwall cracks during tube deformation or collapse. The tubes subject to collapse have been identified via a plant specific analysis and excluded from application of the voltage-based criteria. This analysis is included in revision 3 to WCAP-14046 which was submitted to the NRC June 19, 1995.

ComEd will continue to apply a maximum primary-to-secondary leakage limit of 150 gallons per day (gpd) through any one SG at Byron and Braidwood to help preclude the potential for excessive leakage during all plant conditions. The RG 1.121 criterion for establishing operational leakage limits that require plant shutdown are based on detecting a free span crack prior to resulting in primary-to-secondary operational leakage which could potentially develop into a tube rupture during faulted 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.

Tube burst is precluded during normal operation due to the proximity of the TSP to the tube and during a postulated MSLB event with tube expansion. The 150 gpd limit provides a conservative limit for plant shutdown prior to reaching critical crack lengths should significant crack extension unexpectedly occur outside the thickness of the TSP.

Lowering the Unit 1 RCS dose equivalent I-131 limit from 1.0 μCi/gm to 0.35 μCi/gm is conservative and provides a defense in depth approach to implementation of this IPC.

Based on current predictions of MSLB leakage at the time of SG replacement, the lower RCS dose equivalent I-131 limit also ensures that the resulting 2-hour dose rates at the Braidwood and

Byron site boundaries will not exceed an appropriately small fraction of 10 CFR 100 dose guideline values.

Modification of the Byron and Braidwood Specifications for conformance with Generic Letter 95-05 requirements is primarily administrative and will not alter the plant design basis. For Braidwood, the decrease in the allowed burst probability from 2.5×10^{-2} to 1.0×10^{-2} is conservative. Byron Station has previously incorporated this requirement.

With implementation of an increased IPC voltage repair limit (up to a maximum of 3.0 volts) using tube expansion for the hot-leg support plate intersections, steam generator tube integrity continues to be maintained through inservice inspection, tube repair and primary-to-secondary leakage monitoring. By effectively eliminating tube burst at hot-leg TSP intersections, the potential for multiple tube ruptures is essentially eliminated. Therefore, the possibility of a new or different kind of accident from any previously evaluated is not created.

ComEd has evaluated industry experiences with TSP degradation, eddy current signal distortions, and component misalignment. Eddy current signal distortions due to TSP geometry are not indicative of TSP degradation and do not result in any kind of accident.

The component misalignment experienced by one unit is not applicable to Byron Unit 1 or Braidwood Unit 1 and, thus, will not result in any kind of accident. Specific limitations, as discussed above, will be applied to indications at hot-leg intersections which contain dents. These limitations ensure that integrity of the SG tubes is maintained consistent with current analyses should tube denting or TSP cracking occur. Application of the 3.0 volt hot-leg IPC to Byron Unit 1 and Braidwood Unit 1, with the limitations specified, will not result in 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.

The use of the voltage-based, bobbin coil, tube support plate elevation plugging criteria with tube expansion at Byron Unit 1 and Braidwood Unit 1 is demonstrated to maintain steam generator tube integrity commensurate with the criteria of RG 1.121. RG 1.121 describes a method acceptable to the NRC staff for meeting GDC 14, 15, 31, and 32 by reducing the probability or the consequences of steam generator tube rupture.

This is accomplished by determining an eddy current inspection voltage value which represents a limit for leaving a SG tube in service. Tubes with ODSCC voltage indications beyond this limiting value must be removed from service by plugging or repaired by sleeving. Upon implementation of an increased IPC voltage repair limit (up to a maximum of 3.0 volts) for the hot-leg, even under the worst case conditions, the occurrence of ODSCC at the tube support plate elevations has been evaluated and shown not to present a credible potential for a steam generator tube rupture event during normal or faulted plant conditions. The End Of Cycle (EOC) distribution of crack indications at the tube support plate elevations will be confirmed to result in acceptable primary-to-secondary leakage during all plant conditions such that radiological consequences are not adversely impacted.

Addressing RG 1.83 considerations, implementation of the increased hot-leg tube support plate intersection bobbin coil voltage-based repair criteria is supplemented by enhanced eddy current inspection guidelines to provide consistency in voltage normalization and a 100% eddy current inspection sample size at the affected tube support plate elevations.

For the leak and burst assessments, the population of indications in the voltage distribution is dependant on the POD function. The purpose of the POD function is to account for indications that may not be identified by the data analyst.

In implementing this proposed IPC, ComEd will use the conservative Generic Letter 95-05 POD value of 0.6 for all voltage amplitude ranges.

Lowering the Unit 1 RCS dose equivalent I-131 limit from 1.0 μCi/gm to 0.35 μCi/gm is conservative and provides a defense in depth approach to implementation of this IPC. Based on current predictions of MSLB leakage at the time of SG replacement, the lower RCS dose equivalent I-131 limit also ensures that the resulting 2-hour dose rates at the Braidwood and Byron site boundaries will not exceed an appropriately small fraction of 10 CFR 100 dose guideline values.

Modification of the Byron and Braidwood Specifications for conformance with the Generic Letter 95-05 requirements is primarily administrative and will not reduce any safety margins. For Braidwood, the decrease in the allowed burst probability from 2.5×10^{-2} to 1.0×10^{-2} is conservative. Byron Station has previously incorporated this requirement.

Implementation of the tube support plate elevation repair limits will decrease the number of tubes which must be repaired. The installation of steam generator tube plugs or sleeves reduces the RCS flow margin. Thus, implementation of the interim plugging criteria will maintain the margin of flow that would otherwise be reduced in the event of increased tube plugging.

As discussed previously, ComEd has evaluated industry experiences with TSP degradation, eddy current signal distortions, and component misalignment. Eddy current signal distortions at tube support plates will be evaluated to attempt determination of the cause of the distortion. A signal distortion alone will not result in reduction in the margin of safety. The foreign unit that experienced the component misalignment was of a significantly different design than the Byron Unit 1 and Braidwood Unit 1 steam generators. Analysis of the design differences shows that component misalignment of that type is not applicable to Byron Unit 1 or Braidwood Unit 1 and, thus, will not result in a reduction in the margin of safety.

Specific limitations, as discussed previously, will be applied to indications at hot-leg intersections which contain dents. These limitations conservatively treat indications as freespan to ensure that integrity of the SG tubes is maintained consistent with current analyses should tube denting or TSP cracking occur. Also, tubes with large dents (> 5.0 volts) and tubes adjacent to these dented tubes will not be used for tube expansion to ensure success of tube support plate motion limitation under accident conditions. Application of the 3.0 volt hot-leg IPC to Byron Unit 1 and Braidwood Unit 1, with the limitations specified, will not result in a reduction in a margin of safety.

Thus, the implementation of this amendment does not result in a significant reduction in a margin of safety.

Therefore, based on the above evaluation, ComEd has concluded that these changes involve no significant hazards considerations.

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 October 27, 1995 , 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 rooms which for Byron is located at the Byron Public Library District, 109 N. Franklin, P.O. Box 434, Byron, Illinois 61010; and 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 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 September 1, 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 rooms which for Byron is located at the Byron Public Library District, 109 N. Franklin, P.O. Box 434, Byron, Illinois 61010; and for Braidwood, the Wilmington Public Library, 201 S. Kankakee Street, Wilmington, Illinois 60481.

Dated at Rockville, Maryland, this 19th day of September 1995.

FOR THE NUCLEAR REGULATORY COMMISSION

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