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

August 24, 1995

50-434/435

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: REQUEST FOR ADDITIONAL INFORMATION REGARDING THE STEAM GENERATOR TUBE VOLTAGE-BASED REPAIR CRITERIA (TAC NOS. M91671, M91672, M91673 AND M91674)

Dear Mr. Farrar:

In the course of our review of the Commonwealth Edison Company (ComEd) pending request for license amendments submitted on July 7, 1995, regarding a revision to the technical specifications governing the steam generator (SG) tube voltage-based repair criteria for Byron Unit 1 and Braidwood Unit 1, we have identified a need for additional information. Your submittal of July 7, 1995, revised and superseded in its entirety, your original request for license amendments submitted on February 13, 1995. We have previously transmitted five requests for additional information (RAIs) in our letters dated May 31, 1995, June 22, 1995, August 3, 1995, August 11, 1995, and August 23, 1995. This latest RAI was developed during our further review of your responses to our second RAI issued on June 22, 1995. The RAI transmitted in our letter dated August 11, 1995, involved concerns developed as a result of our initial review. The issues in this RAI were discussed briefly in the meeting held in Rockville, Maryland, on August 17, 1995, between members of the NRC staff and representatives of ComEd. For convenience, we are continuing the numbering in the same sequence we established in our prior RAIs on this matter.

Our concerns in the present RAI are related to a number of specific areas including: (1) your proposed SG tube eddy current inspection criteria; (2) the methodology for calculating the maximum tube support plate (TSP) displacements and deflections under load; (3) the test program to establish a bounding leak rate for SG tube flaw indications restricted from burst (IRB); (4) your proposed inspection program after SG tube expansion at the TSP intersections to determine if circumferential cracking was initiated by the SG tube expansion process; (5) the updated data base of pulled SG tubes; and (6) your proposed structural integrity inspection program for those SG internal structures which serve to limit the deflections of the TSPs under postulated accident conditions.

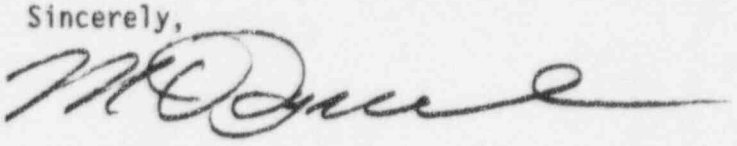
In this regard, we note that at the meeting cited above, you indicated your plans to conduct an inspection of the SG internal structures during the forthcoming refueling outage of Braidwood 1 but do not plan to conduct a similar inspection of the SG internal structures during the forthcoming Byron 1 mid-cycle SG inspection. Accordingly, we request that you address this issue by providing your justification for raising the lower voltage-based repair limit for Byron 1 from 1.0 volt to 3.0 volts without inspecting the Byron 1 SG internal structures required to function to ensure limited TSP displacements.

As stated in our previous RAIs, without timely and high quality technical resolution of the outstanding issues, it is unlikely that the staff will be able to reach a positive conclusion on your pending license amendments. Accordingly, we request that you state a date certain when you can respond to this latest RAI.

This requirement affects nine or fewer respondents and, therefore, is not subject to Office of Management and Budget review under P. L. 96-511.

If you have any questions on these matters, please contact Mr. M. D. Lynch at (301) 415-3023.

Sincerely,



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

Docket Nos. STN 50-454
STN 50-456

cc w/encl: See next page

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REQUEST FOR ADDITIONAL INFORMATION

REGARDING THE PROPOSED REVISIONS TO THE TECHNICAL SPECIFICATIONS

RELATED TO THE STEAM GENERATOR TUBE VOLTAGE-BASED REPAIR CRITERIA

BYRON UNIT 1 AND BRAIDWOOD UNIT 1

DOCKET NOS. STN 50-454 AND STN 40-456

58. (Refer to Item 35 in the staff's letter dated June 22, 1995.)

The staff believes that the use of a 0.590 inch diameter probe appears to be non-conservative when used to ensure that no dents exceed 65 mils, thereby ensuring the integrity of the tube support plate (TSP) ligaments. Accordingly, discuss the need to use either a different size probe, method, or criterion to ensure that the size of a dent is sufficiently small so as to ensure structural integrity of the TSP ligaments.

59. (Refer to Item 36 in the staff's letter dated June 22, 1995, for Items 59 and 60.)

In your response to Item 36, you provided further detail on how the TSP displacements were calculated. It appears that this calculation is non-conservative in that the reference position for the hot standby and full power SG conditions were taken to be equivalent. Accordingly, provide a reassessment of the TSP displacements for the SG tube expansion matrix presently proposed for the worst case postulated accident condition (e.g., a main steamline break (MSLB)) initiated from both the hot standby and full power conditions, assuming that the TSPs are free to move between all modes of operation (i.e., cold shutdown, hot standby and full power). In addition, the calculation of the TSP displacements should include any other effects which may result in relative movement between the SG tubes and the TSPs, unless exclusion of these other effects would result in more conservative estimates of the TSP displacements under all conditions (i.e., inclusion of these other effects would lower the displacements at all other locations of the TSPs). If the TSP displacements resulting from this assessment are greater than the currently estimated maximum displacement, provide an assessment of the significance of these larger displacements with respect to acceptable structural and leakage integrity of the SG tube indications of outer diameter stress corrosion cracks (ODSCC) accepted for continued service.

60. In response to Item 36.b., you indicated that a SG tube which has been expanded to create a new "tierod" and then plugged, may act to pull the TSPs down relative to the hotter, in service, SG tubes due to differences in the thermal growth between a plugged and unplugged SG tube. This assumes that the TSPs are locked in the "hot" condition and that the SG tube to TSP contact forces of the unplugged tubes are small

Enclosure

enough so that the new "tierod" can pull the TSP down. If this were to occur, the potential exists that SG tube ODSCC degradation previously confined within the TSP crevices, may be exposed. As a result, the ODSCC degraded area of the SG tube may be longer than the thickness of the TSP (i.e., 3/4-inch). Furthermore, the maximum length of an SG tube ODSCC indication exposed during a postulated MSLB accident may be greater than the maximum displacement calculated to date (i.e., 0.1-inch) which presently assumes that the SG tube ODSCC indications are fully confined within the TSPs. Consistent with the comments cited above, perform a calculation which determines the maximum length of a crack which may be exposed during a postulated MSLB in light of these assumptions. If the resultant crack length displacement (i.e., the TSP displacement plus the relative crack displacement as a result of the newly created tierods) are greater than the currently estimated 0.1-inch maximum displacement, provide an assessment of the significance of these larger crack length displacements with respect to ensuring acceptable structural and leakage integrity of the SG tube ODSCC indications accepted for continued service.

61. To quantify the uncertainty in the SG tube leakage measurements for an indication restricted from burst (IRB), the leakage from a set of orifices has been measured as part of your IRB test program. Discuss any modifications and/or repairs performed on the test rig and/or facility since the original SG tube specimen IRB leak rate testing was performed. Discuss whether these modifications and/or repairs, if any, would alter your conclusions derived from the orifice testing. For example, if a valve in the test rig was leaking by the seat, the leakage from this valve could result in an underestimate of the leakage. If a leaking valve was replaced or repaired prior to the orifice testing, uncertainties in the leakage measurements may not be fully quantified.

62. For those SG tubes which are proposed to be expanded, discuss the need for any rotating pancake coil (RPC) examinations to ensure that no circumferential cracks are present (i.e., circumferential cracks are neither initiated nor opened up as a result of the expansion process). Discuss the need for such an RPC examination to establish baseline data.

63. In attachment B to your letter dated July 7, 1995, you stated that the ODSCC database has been updated to include the latest Byron 1 and Braidwood 1 SG tube pull data. The staff believes that there is additional pulled tube data available from at least one other nuclear power plant (e.g., South Texas). Discuss whether this data will be included in the database.

64. Your proposed license amendments currently rely on several SG internal structures to limit the displacement of the TSPs during postulated accident conditions. As such, the structural integrity of these components is important to safety by ensuring that the displacements of the TSPs are limited to an acceptable value. Accordingly, provide your inspection plans which are intended to ensure the structural integrity

of those components necessary to limit the TSP displacements (e.g., wedges, vertical bars and tierods). Your response should address, but not be limited to, the following considerations:

a) The scope of the inspection of the SG internal structures; e.g., the TSPs, wedges, vertical bars and tierods. Provide a discussion of the available inspection technologies, including visual, eddy current and any other available state of the art inspection techniques, which have been considered in defining the scope of the inspection. Discuss any limitation in their application.

b) The capabilities, limitations, and qualification of the inspection techniques to be used. This discussion should address the capability of the proposed inspection techniques to identify cracking and other degradation mechanisms whose characteristics would impair the structural integrity of any SG internal component for which credit was taken in calculating the TSP displacements.

c) The need to clean or prepare the surface of each SG internal structural component required to limit the TSP displacements, prior to its inspection.

d) The applicability of the inspections of the SG internal structures performed at one location within a SG to assess the potential for degradation at other locations in the SG if only limited inspections can be performed. For example, if inspections are performed at the vertical bars at the bottom and top TSPs, discuss how the conditions at these specific SG locations are representative of other TSP locations.

65. In your pending request for license amendments, you propose to expand certain SG tubes into the TSPs, thereby adding additional structural restraint to the TSPs and resulting in limited TSP displacements under accident conditions. As part of this process, you propose to insert sleeve stabilizers into these selected SG tubes where they will then be hydraulically expanded at the TSP intersections. You cited certain corrosion tests and operating experience for similar hydraulically expanded joints, in part, as your basis for conducting delayed inspections of these joints rather than conducting earlier inspections. For example, you propose to inspect a minimum of three expanded SG tube joints every third planned SG inspection after installation. In light of the limitations of corrosion tests to simulate field conditions, including both installation and in-service conditions, and the importance of the expanded SG tubes in minimizing the TSP displacements during postulated accident conditions, the staff believes that conducting inspections of these expanded SG tube joints at the first planned SG inspection after installation would verify that no significant degradation had developed during the first portion of the in-service life of these expanded joints.

In this regard, we note that at the meeting cited above, you indicated your plans to conduct an inspection of the SG internal structures during the forthcoming refueling outage of Braidwood 1 but do not plan to conduct a similar inspection of the SG internal structures during the forthcoming Byron 1 mid-cycle SG inspection. Accordingly, we request that you address this issue by providing your justification for raising the lower voltage-based repair limit for Byron 1 from 1.0 volt to 3.0 volts without inspecting the Byron 1 SG internal structures required to function to ensure limited TSP displacements.

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Sincerely,
Original Signed By
M. D. 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 50-454
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