November 21, 2000

MEMORANDUM TO: Docket File

FROM: George F. Dick, Jr., Senior Project Manager, Section 2 ***/RA/***
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

SUBJECT: BYRON STATION, UNITS 1 AND 2, AND BRAIDWOOD STATION, UNITS 1 AND 2 -- ELECTRONIC TRANSMISSION - DRAFT RAI QUESTIONS FOR POWER UPRATE (TAC NOS. MA9428, MA9429, MA9426 AND MA9427)

The attached document was transmitted electronically on November 16, 2000, to Commonwealth Edison Company (ComEd) to provide them with questions that are expected to be sent formally regarding ComEd's July 5, 2000, request for approval of a power increase for the units at Byron and Braidwood Stations. This memorandum and attachment do not convey a formal request for information or represent an NRC staff position.

Docket Numbers STN 50-454, STN 50-455,
STN 50-456, STN 50-457

Attachment: E-Mail to JA Bauer w/att

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ACCESSION NUMBER: ML003770722

OFFICE	PM:PD3-2	LA:PD3-2	SC:PD3-2
NAME	GDICK	CMOORE	AMENDIOLA
DATE	11/20/00	11/20/00	11/20/00

OFFICIAL RECORD COPY

From: George Dick
To: INTERNET:Joseph.A.Bauer@ucm.com
Date: Thu, Nov 16, 2000 9:04 AM
Subject: Mechanical Engineering RAI

Joe,

Attached is the RAI from mechanical engineering, pretty much as I received it. It looks like question 12 is a duplicate of one that you already have. We need to look at it more closely. If it is a duplicate, it will not be sent with the official RAI.

George

ATTACHMENT

REQUEST FOR ADDITIONAL INFORMATION
PROPOSED AMENDMENT FOR POWER UPRATE
BRAIDWOOD AND BYRON STATION, UNITS 1 AND 2

1. In Sections 5.1.1.3 of Attachment E to the reference transmittal, you stated that " the very conservative maximum range of stress intensity and cumulative fatigue usage factor results for the Byron inlet nozzle safety ends were applied to Braidwood to simplify the design transient evaluation." Provide the technical basis that the Byron results can be applicable to Braidwood. Confirm that the ASME Code Section III, 1971 Edition with addenda through the Summer 1973 is the Code of record specified in the UFSAR of both Braidwood and Byron Stations, and that they were used in the evaluation of the reactor vessel for the power uprate for each plant.

2. In Section 5.2.2.2, you stated that there is sufficient margin to accommodate the increase in Fluid induced vibration (FIV) loads for the power uprate. Discuss the potential for FIV with regard to the critical fluid elastic stability ratio and the acceptance criteria for the FIV loads for the power uprate. Also, provide a comparison of the maximum FIV load on the most critical component (i.e., guide tubes) at the uprated power level with the allowable FIV load at the Braidwood and Byron stations.

3. In Sections 5.2.3, you evaluated reactor internal components for the uprated power conditions including the lower core plate, baffle/barrel region components, core barrel, baffle plate, baffle/barrel bolts, and upper core plate. In Section 5.2.3.1, you stated that Table 5.2.3-1 lists the evaluation results. However, there are no results found in Table 5.2.3-1 for the reactor internal components mentioned above. Please provide such results including the maximum calculated stresses and CUFs for these components. Also provide the Code and Code Edition used for evaluation of the reactor internal components. If different from the Code of record, please justify and reconcile the differences.

4. In regard to Sections 5.4 of Attachment E to the reference transmittal, provide the maximum calculated stress and cumulative fatigue usage factor (CUF) at the critical locations of the CRDM's components. Also, provide the allowable code limits, and the Code and Code edition used in the evaluation for the power uprate. If different from the Code of record, provide the necessary justification.

5. In reference to Section 5.5, you stated that the system design transients are used in the evaluation of the piping fatigue, and that the impact of changes in the system thermal transients were factored into the ASME code stress and fatigue usage factor (CUF) determination. Discuss in detail the methodologies and assumptions for the determination of the ASME code stress and CUF for reactor coolant loop piping and branch piping. Provide the maximum

calculated stresses and CUFs at the most critical locations for RCL piping, primary equipment supports and nozzles, RCL branch nozzles and pressurizer surge nozzles, allowable limits, the Code of record and Code edition used for the power uprate conditions for NSSS piping and supports. If different from the Code of record, justify and reconcile the differences.

6. In reference to Section 5.5, you also stated that the current analysis for the reactor primary loop piping was performed by Framatome Technologies, Incorporated (FTI) for Byron and Braidwood Unit 1 due to steam generator replacement at Unit 1. The FTI analysis is also used to demonstrate primary equipment supports to be adequate for the power uprate. It is noted that the FTI analysis has not been reviewed by the NRC. Please describe the analysis method, assumptions, computer codes used for analysis (if different from those specified in the FSAR), the results of analysis , and the Code and Code Edition used. If different from the Code of record, justify and reconcile the differences. The results of analyses should include RCL piping, major equipment supports and nozzles, RCL branch nozzles and pressurizer surge nozzles.

7. In Section 5.7.1, the current design basis input parameters for the Unit 1 RCL piping analysis by FTI are provided in Table 5.7.1.1-1 where the steam flow for the 5 percent power uprate is 4.0×10^6 ibm/hr. This is different from Table 2.1-1 where the steam flow is 15.98×10^6 ibm/hr. Please explain the apparent discrepancy.

8. In Section 5.7.1.5, you indicated that the potential for vibration in the U-bends (including the small radius) and tubes due to fluid elastic instability at the power uprate conditions was assessed. Provide an evaluation of the flow-induced vibration of the steam generator U-bends tubes due to power uprate regarding the analysis methodology, vibration level, computer codes used in the analysis and the calculated elastic-fluid instability ratio. If any computer codes used for the analysis are different from those specified in the Braidwood and Byron FSAR, provide the basis for the acceptability of the computer codes that were used.

9. Discuss the functionality of safety-related mechanical components (i.e., all safety related valves and pumps, including power-operated relief valves) affected by the power uprate to ensure that the performance specifications and technical specification requirements (e.g., flow rate, close and open times) will be met for the proposed power uprate. Confirm that safety-related motor-operated valves (MOVs) in your Generic Letter 89-10 MOV program at Braidwood and Byron stations will be capable of performing their intended function(s) following the power uprate including such affected parameters as fluid flow, temperature, pressure and differential pressure, and ambient temperature conditions. Identify mechanical components for which functionality at the uprated power level could not be confirmed. Please discuss effects of the proposed power uprate on the pressure locking and thermal binding of safety-related power-operated gate valves for Generic Letter (GL)

binding of safety-related power-operated gate valves for Generic Letter (GL) 95-07 and on the evaluation of overpressurization of isolated piping segments for GL 96-06.

10. In reference to Section 9, list the balance-of-plant (BOP) piping systems that were evaluated for the power uprate. Discuss the methodology and assumptions used for evaluating BOP piping, components, and pipe supports, nozzles, penetrations, guides, valves, pumps, heat exchangers and anchorage for pipe supports. Provide the calculated maximum stresses for the critical BOP piping systems, the allowable limits, the Code of record and Code edition used for the power uprate conditions. If different from the Code of record, justify and reconcile the differences. Were the analytical computer codes used in the evaluation different from those used in the original design-basis analysis? If so, identify the new codes and provide justification for using the new codes and state how the codes were qualified for such applications.

11. Discuss the potential for flow-induced vibration in the heat exchangers following the power uprate. Provide a summary of evaluation for power uprate effects on the high energy line break analysis, jet impingement and pipewhip loads for the power uprate condition.

12. In Section 9.5.3, you indicated that the maximum temperature of 162.7°F (> 150°F allowed by ACI 349-97 Code) occurs in the spent fuel pool (SPF) for a full off load, and that this is acceptable since the full offload is a temporary condition and the spent temperature for the long term remains below 150°F. Please elaborate on the duration of the full offload condition at Braidwood and Byron. Provide a discussion regarding the evaluation of the SPF including analysis methodology, assumptions, computer codes used for the analysis, the results of the analysis (i.e., stresses), and allowable limits specified in the ACI-349 Code for application to the uprated power condition. If the computer codes used are different from the codes of record, Justify and reconcile the differences.

13. Do you project modifications to piping or equipment supports for the proposed power uprate? If any, provide examples of pipe supports requiring modification and discuss the nature of these modifications

REFERENCE

Commonwealth Edison Company Letter to the NRC, “Braidwood Station, Units 1 and 2, Facility Operating Licenses NPF-72, And NPF-77, NRC Docket Nos. 50-456, And 50-457, Respectively, And Byron Station, Units 1 and 2, Facility Operating Licenses NPF-37 And NPF-66, NRC Docket Nos. 50-454, And 50-455, Respectively, -Request for a License Amendment to permit Uprated Power Operations at Byron and Braidwood Stations," dated July 5, 2000.