October 19, 2000

Mr. Oliver D. Kingsley, President
Nuclear Generation Group
Commonwealth Edison Company
Executive Towers West III
1400 Opus Place, Suite 500
Downers Grove, IL 60515

SUBJECT: BYRON AND BRAIDWOOD - REQUEST FOR ADDITIONAL INFORMATION REGARDING THE POWER UPRATE REQUEST (TAC NOS. MA9428, MA9429, MA9426, AND MA9427)

Dear Mr. Kingsley:

By letter dated July 5, 2000, Commonwealth Edison Company (ComEd, the licensee) requested amendments to the licenses for Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2. The proposed changes would revise the maximum power level specified in each unit's license and the definition of rated thermal power in the technical specifications.

During the course of our review, we find that additional information is needed in order to complete our evaluation of ComEd's request. Consequently, please provide a response to the enclosed Request for Additional Information (RAI). The majority of the questions were discussed with members of your staff during a meeting with the NRC staff on September 20, 2000. Several additional questions have been discussed during telephone conversations.

In order for us to maintain our review schedule, we request a response to the RAI within 30 days of receipt of this letter.

Please note that because ComEd's response to the enclosed RAI is considered a supplement to the July 5, 2000, amendment application, we request that it be submitted under oath and affirmation in accordance with 10 CFR 50.90 and 10 CFR 50.30(b).

Please contact me if there are any questions regarding this RAI.

Sincerely,

/RA/

George F. Dick, Jr., Project Manager, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. STN 50-454, STN 50-455
STN 50-456, STN 50-457

Enclosure: RAI

cc w/encl: See next page

Mr. Oliver D. Kingsley, President
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Docket Nos. STN 50-454, STN 50-455
STN 50-456, STN 50-457

Enclosure: RAI

cc w/encl: See next page

ACCESSION NO.: ML003761594

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REQUEST FOR ADDITIONAL INFORMATION

RELATED TO AN INCREASE IN RATED THERMAL POWER

COMMONWEALTH EDISON COMPANY

BYRON STATION, UNITS 1 AND 2

BRAIDWOOD STATION, UNITS 1 AND 2

DOCKET NOS. STN 50-454, STN 50-455, STN 50-456 AND STN 50-457

The NRC staff is reviewing the July 5, 2000, request from Commonwealth Edison Company (ComEd, the licensee) to revise the maximum thermal power specified in the licenses for Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2. During the review, the staff has identified the areas where additional information is required. The specific questions follow.

Question Set A

A.1 It does not appear that there is any discussion addressing how the power uprate at Byron and Braidwood impacts the existing analysis performed for station blackout. Please discuss and verify the assumptions for the existing station blackout analysis are valid for the power uprate conditions, particularly as they relate to issues such as heat-up analysis, equipment operability and battery capacity.

A.2 It is indicated in Page 9-126 of the power uprate licensing report that the electrical equipment located inside and outside containment, that performs a safety-function, must remain qualified for the accident, pressure, and humidity environment at the uprate power conditions. However, the impact of power uprate on humidity environment during normal and accident conditions were not provided. Provide the impact of power uprate on humidity environment for electrical equipment inside and outside containment.

A.3 Explain, in detail, the bounding conditions of the containment revised temperature/pressure profiles as shown in Figures 9.3.21-1 and 9.3.21-2.

A.4 For electrical equipment within the scope of 10 CFR 50.49 outside the containment, explain, in detail, how the electrical equipment remains qualified when the peak compartment temperature of 518.4 degrees Fahrenheit (°F) from Case 102-L (Byron/Braidwood, Unit1) and the peak compartment temperature of 502.5 °F from Case 102-M (Byron/Braidwood, Unit 2) exceed the temperature of 419 °F previously used to demonstrate equipment qualification. In addition, explain how the post-accident monitoring equipment remains qualified when the peak long-term temperature used for evaluation of post-accident monitoring equipment outside containment in the steam tunnels and valve room exceeds the current peak temperature of 515.25 °F by 3 °F. In addition, please provide the temperature for which the post-accident monitoring equipment was previously qualified.

ENCLOSURE

Question Set B

In its submittal the licensee has stated, "Reactor trip and Engineered Safety feature (ESF) actuation setpoints have been assessed and no needed changes were identified as a result of uprated power operations.... All acceptance criteria including those for LBLOCA [large break loss-of-coolant accident], SBLOCA [small break loss-of-coolant accident], non-LOCA [loss-of-coolant accident] accidents, containment pressure and temperature, and radiological dose limits, continue to be met.... The results of the NSSS [nuclear steam supply system] analysis demonstrated that all systems were capable of performing their current design basis functions with either no changes or with appropriate changes to programs, setpoints and alarms." (Reference Attachment A to the submittal Pages A-2 and A-15.)

In regard to the above statements, please furnish the following additional information:

B.1 Identify safety-related instrumentation and control (I&C) functional units for which the power uprate evaluation indicated no change was needed in trip setpoint and allowable values for uprated power operation.

B.2 Identify-safety related I&C functional units where appropriate changes to programs, setpoints and alarms are being made and provide a short description of changes.

B.3 Provide a brief description of setpoint calculation methodology used for assessing acceptability of trip actuation setpoints and allowable values of safety-related instruments for uprated power operation, and confirm that uncertainty values were established with a 95 percent probability at a 95 percent confidence level. Also provide reference(s), if the setpoint methodology used at Byron and Braidwood has been reviewed and approved by the staff.

Question Set C

C.1 The licensee stated that certain WCAP reports provide the details of how the increase in core rated power will effect the pressure-temperature (P-T) limits and pressurized thermal shock (PTS) evaluations for the Byron and Braidwood units; however, it is not clear that the reports have been placed on the dockets for Byron and Braidwood. If these WCAPs have not been placed on the dockets for Byron and Braidwood, please provide the following documents: WCAP-15391 and WCAP-15391 on P-T limit curves and WCAP-15389 and WCAP-15390 on PTS evaluations for the Byron units; WCAP-15364 and WCAP-15373 on P-T limit curves and WCAP-15365 and WCAP-15381 on PTS evaluations for the Braidwood units.

C.2 It appears that discrepancies exist between the end-of-license (EOL) upper-shelf energy (USE) values reported by the licensee and those based on information from reactor vessel integrity database (RVID) for some beltline materials of Byron and Braidwood units. Provide detailed calculations for the EOL (32 effective full power years (EFPY)) USE values for the following material: intermediate to lower shell forging circumferential weld WF336 (heat 442002) for Byron, Unit 1; intermediate to lower shell forging circumferential weld WF447 (heat 442002) and nozzle shell to intermediate shell forging circumferential weld WF562 (heat 442001) for Byron, Unit 2; nozzle shell forging 5P-7016 and intermediate to lower shell forging circumferential weld WF562 (heat 442011) for Braidwood, Unit 1; and nozzle shell forging 5P-7056 and intermediate to lower shell forging circumferential weld WF562 (heat 442011) for Braidwood, Unit 2. The information submitted in the calculations for these materials should include initial USE, chemistry data, 1/4T fluence data, percent decrease in USE, method for calculating the change in USE, and the basis for any discrepancies with the corresponding USE data for these materials in the RVID.

C.3 In Section 5.1 of the Power Uprate Safety Evaluation, the licensee indicated that the methodology used to generate P-T limit curves for the Byron and Braidwood units is the methodology documented in WCAP-14040-NP-A, as modified by five alternatives (exceptions) to the methodology. Acceptance of the Pressure-Temperature Limits Report (PTLR) was predicated on acceptance of the methodology of WCAP-14040-NP-A, as modified by the methods of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code) Case N-514 for establishing low temperature overpressure (LTOP) system setpoints. The safety evaluation for acceptance of the license amendment for the PTLR never included or accepted the five alternatives as the basis for generating the P-T limit curves for the PTLR. With respect to these five alternatives, the following actions are necessary:

a. Since the alternatives to the methodology will change the previous methodology initially approved for the Byron/Braidwood PTLR, pursuant to the staff's position stated in Generic Letter (GL) 96-03 for accepting PTLR requests, the licensee will need to submit a license amendment request to change the appropriate administrative control section for the PTLR in the technical specifications to incorporate any proposed alternatives to the previously approved methodology, and submit these five alternative methods for review and approval. In this case, it should be noted that the staff, at this time, is not approving any proposal to eliminate the flange temperatures requirements, because the staff's review of WCAP-15315 is still pending.

b. Any proposals to modify the existing PTLR methodology by the methods of Code Case N-588 or N-640 will require, pursuant to 10 CFR 50.60(b), exemption requests to deviate from the requirements of Section IV.A.2 of Appendix G to 10 CFR Part 50. The staff will evaluate such exemptions on a case-by-case basis pursuant to the exemption approval criteria stated in 10 CFR 50.12. In this case, any proposal to use Code Case N-640 must be accompanied with a statement that the provisions in Code Case N-514 (or in Paragraph G-2215 of the 1996 Edition of Appendix G to the Code) for establishing the LTOP pressure setpoint at 110 percent of the allowable pressure provided by the P-T limit curves for normal operation will not be used, and that instead, the LTOP setpoint will be established at 100 percent of the allowable pressure established by the P-T limit curves for normal operation.

Question Set D

D.1 Summarize the contents and results of the assessment that was used to confirm that the current steam generator (SG) plugging criteria in the Technical Specifications for each unit will remain adequate for the power uprated conditions (Section 5.7).

D.2 For the Byron, Unit 2, and Braidwood, Unit 2, SGs, please provide a summary of the operational assessment for antivibration bar (AVB) wear due to the power uprate to determine the allowable operating interval between inspections (Section 5.7.2.2.5).

D.3 Page 5-104 states that, “The window for operating conditions at 5 percent power uprate (3600.6MWt) is the same as, or bounded by the window for, the operating conditions (3425 MWt) previously evaluated. For example, the maximum primary side T_{hot} (most important parameter with respect to tube degradation) is 618.4 °F for both 3425 MWt and 3600.6 MWt.” Page 5-124 states that, “To minimize the potential impact of the power uprate on SG tube degradation, selection of the optimum point (best estimate steam pressure) at which to design the high pressure turbine modifications includes maintaining the post-uprate T_{hot} at the same value as the current T_{hot}....” Because this information is inconsistent, please clarify if T_{hot} indeed stays the same after the uprate. If T_{hot} is increased after the power uprate, what effect will this have on the discussion in Section 5.7.2.4 on tube degradation.

Question Set E

E.1 Discuss whether the power uprate will change the type and scope of plant emergency and abnormal operating procedures. Will the power uprate change the type, scope, and nature of operator actions needed for accident mitigation and will new operator actions be required?

The licensee stated in its letter dated July 5, 2000 (page A-21) that, “The power uprate has the potential to affect plant procedures used to operate and maintain the facility in accordance with design basis and licensing requirements.... Procedures that are identified as being affected by the power uprate will be revised prior to the uprate implementation.” In Attachment E (page 12-1), the licensee stated that, “A physical review of each procedure identified [during the screening] will be conducted to determine the need for revision. Those procedures will be revised to incorporate changes. For example, changes due to modifications, operator response times, setpoint changes will result in revisions to existing procedures... and all required training will be conducted prior to the implementation of the power uprate.”

In addition to the information provided by the licensee in its July 5, 2000, submittal, what procedures will be changed, what changes will be made and, will new operator actions be required?

E.2 Provide examples of operator actions that are particularly sensitive to the proposed increase in power level and discuss how the power uprate will effect operator reliability or performance. Identify all operator actions that will have their response times changed because of the power uprate. Specify the expected response times before the power uprate and the new (reduced/increased) response times. If there are any reduced operator response times, discuss why they are needed. Discuss whether any reduction in time available for operator actions, due to the power uprate, will significantly affect the operator's ability to complete the required manual actions in the times allowed. Discuss results of simulator exercises conducted to assure that operator response times for operator actions that are potentially sensitive to power uprate, can be successfully achieved within allowable time limits.

E.3 Discuss all changes the power uprate will have on control room alarms, controls, and displays. For example, will zone markings on meters change (e.g., normal range, marginal range, and out-or-tolerance range)? If changes will occur, discuss how they will be addressed.

In Attachment C (page C-24) of the licensee's July 5, 2000, submittal, the licensee indicates that, "The basic design of all systems remains unchanged and no new equipment or systems have been installed which could potentially introduce new failure modes or accident sequences. No changes have been made to any reactor trip or ESF actuation setpoints." However, the licensee also stated that, "Minor modifications, to support implementation of uprated power conditions, will be made as required to existing systems and components."

In addition to addressing the specific questions in question E.3, what are the "minor modifications" referred to by the licensee and what effect will they have on operator performance?

E.4 Discuss all changes the power uprate will have on the Safety Parameter Display System (SPDS) and how they will be addressed.

E.5 Describe all changes the power uprate will have on the operator training program and the plant simulator.

<u>Question Set F</u>

With regard to radiological analyses, key elements of the staff review are comparisons of radiological consequence analyses discussed in the most recent evaluations of record (i.e., past analyses, which in most cases are expected to be those summarized in the Byron and Braidwood Updated Final Safety Analysis Report (UFSAR)) with the corresponding consequence analyses discussed in the proposed license amendment (i.e., current analyses, which are intended to support the proposed power up-rate). Specifically, for each postulated event or condition, the staff is interested in comparing and understanding the differences between past and current estimates of: (a) the concentrations of iodine and noble gas fission products available for release, (b) the concentrations of iodine and noble gases released to the environment as a result of the event or condition, as well as (c) the associated doses at the nearest exclusion area boundary (EAB), the low population zone (LPZ) outer boundary, and the control room. Therefore, please provide answers to the following:

F.1 What dose models were used for each of the accidents reanalyzed to support power uprate? Have the computer codes and methods of analysis used been approved by the NRC? If they have not been approved, please provide the basis for their use.

F.2 What models, codes, and methods of analysis were used previously, i.e., in the initial licensing of Byron and Braidwood or the analysis of record (UFSAR)?

F.3 All reanalyses for radiological consequences for the design basis accidents (discussed in Chapter 6) include the 2 percent instrument error margin; please provide the basis for not using the instrument error margin in the reanalysis of normal operation dose rates, shielding, and annual radwaste effluent release (discussed in Chapter 9). It should be noted that Regulatory Guide 1.49, which the licensee quotes in Chapter 9, specifically states that, "analyses and evaluation in support of the application should be made at an assumed core power level equal to 1.02 times the proposed licensed power level... for (a) normal operating conditions... (c) accident conditions...."

The following requests for comparisons and discussions are expected to focus on those changes of input data, assumptions, models, and methodology that make a <u>significant</u> contribution to the differences, as well as the rational for the changes. Receipt of this information should minimize staff interactions with the licensee concerning the details of the various specific calculations. To facilitate the comparison we request the following information for each event or condition analyzed:

F.4 Present and discuss the differences between the equilibrium fission product concentrations used in the analyses of record, i.e., past analyses, and those used in analyses for the proposed up-rated thermal power, i.e., the current analyses.

F.5 Present and discuss the differences between the concentrations of iodine and noble gases released to the environment used in past analyses and those used in the current analyses (include discussion of processes and systems which reduce the concentrations, e.g., plate-out, filtration, adsorption).

F.6 Present and discuss the differences between the doses calculated in the past analyses with those calculated in the current analyses.

F.7 Present and discuss the control room dose analyses for all postulated events (include discussion of event mitigation, transport paths, and iodine and noble gas releases).

<u>Question Set G</u>

G.1 Page 6-104 indicates that the methodology used for a feedwater line break accident has been changed from the current analysis by using the reactor coolant system (RCS) thick-metal mass model of the LOFTRAN computer program. Please confirm that the this model of the LOFTRAN has been reviewed and approved by NRC staff.

G.2 Page 6-137 indicates that the consequences of an inadvertent opening of a SG relief or safety valve is bounded by the main steamline break analysis discussed in Sections

6.2.4 and 6.4.5. Please provide transient departure from nucleate boiling (DNB) curves for the main steamline break events to confirm that there is no DNB in these events.

G.3 Page 6-186, Item a, indicates that the SG power operated relief valves are automatically opened during a loss of normal feedwater event. Please confirm that the SG power operated relieve valves' control systems are designed to safety grade requirements or the safety analysis should not give credit to the automatic function of these relief valves.

G.4 Page 6-193, Figure 6.2.8-1, indicated that the pressurizer level reaches 1800 cu-ft. If this curve is not accurate to reflect the results of the analysis, please provide the numerical value of the peak pressurizer level during the transient to confirm that the pressurizer does not reach a water-solid condition during a loss of main feedwater event as it is stated on page 6-190.

G.5 Please provide the results of an analysis for a loss of feedwater event regarding maximum RCS pressure (assume the main and auxiliary pressurizer sprays and pressurizer power-operated relief valves (PORVs) are inoperable, with plant initial conditions that would maximize the RCS pressure) to confirm that the peak RCS pressure is maintained below 110 percent of its design pressure.

G.6 Please provide the results of an analysis for a feedwater line break accident with respect to the maximum RCS pressure to confirm that the peak RCS pressure is maintained below 110 percent of its design pressure.

G.7 The results of the feedwater line break accident indicate that the pressurizer will reach water solid conditions. The NRC staff has generally not accepted a solid pressurizer in order to avoid the potential for all three pressurizer safety valves to be stuck open (a SBLOCA) due to liquid relief through these safety valves. Please propose necessary system modifications and provide the results of the reanalysis to confirm that the pressurizer will not reach solid conditions during this event.

G.8 Please provide the results of an analysis for reactor coolant pump (RCP) locked rotor/shaft break accident with respect to minimum transient departure from nucleate boiling ratio (DNBR) to determine the amount of fuel failure. In this analysis, a loss of offsite power should be assumed coincident with the event (see Final Safety Analysis Report, Section 15.3.3 and 15.3.5). Please assure that all fuel rods with a DNBR value below their allowable minimum DNBR should be assumed to experience DNB and become failed fuel rods. Provide transient curves which include DNBR and state the amount of failed fuel during this event.

G.9 The results of the analysis for an inadvertent operation of the emergency core cooling system (ECCS) during power operation indicate that the pressurizer will reach water solid during this event. The NRC staff has generally not accepted a solid pressurizer for this accident in order to avoid the potential for all three pressurizer safety valves to be stuck open (a SBLOCA) due to liquid relief through these safety valves. Please propose necessary plant modifications and provide the results of your reanalysis of this event to confirm that the pressurizer will not reach water solid conditions during this event.

G.10 Please provide the results of an analysis for another case of an inadvertent operation of the ECCS during power operation, which would lead to the maximum RCS pressure which uses only safety related equipment.

G.11 With respect to the analysis for SG tube rupture event regarding overfill, please provide transient curves for SG water levels and auxiliary feedwater flow rate to demonstrate that overfill does not occur during this event.

G.12 Please provide the results of a thermal hydraulic analysis for a SG tube rupture based on emergency operating procedures to determine the amount of radioactive steam released to the environment and assess the radiological consequences accordingly. In this analysis, a loss of offsite power should be assumed coincident with the steam generator tube rupture (SGTR) and a most limiting single failure of a stuck open SG power operated relief valve associated with the ruptured SG. Technical specification limit of tube leakage should be assumed as an initial condition. Transient curves including DNBR, RCS pressure, SG pressures, auxiliary feedwater flow, primary to secondary leak flow rate, integrated mass leaking from the primary to secondary side, steam release from the ruptured SGs, etc., should be provided.