

November 14, 2001

Mr. Oliver D. Kingsley, President
and Chief Nuclear Officer
Exelon Nuclear
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: CLINTON POWER STATION, UNIT 1 - REQUEST FOR ADDITIONAL
INFORMATION (TAC NO. MB2210)

Dear Mr. Kingsley:

By letter dated June 18, 2001, AmerGen Energy Company, LLC, submitted a license amendment request for a 20-percent power uprate of the Clinton Power Station. The Nuclear Regulatory Commission staff has performed an initial review of your request and finds that it needs additional information to complete its review.

Therefore, I request that you respond to the enclosed request for additional information in order for the staff to complete its review in a timely manner. The questions were discussed and the response dates of November 16, 2001, for many of the questions, and November 30, 2001, for those questions that need input from your contractor were agreed upon with a member of your staff. The questions are unchanged from those sent by facsimile to a member of your staff on October 19, 2001, except for questions 10.5 and 10.12, which were slightly modified to eliminate proprietary information. Your staff indicated that the rest of the questions did not contain proprietary information.

Contact me if you have any questions.

Sincerely,

/RA/

Jon B. Hopkins, Senior Project Manager, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-461

Enclosure: As stated

cc w/encl: See next page

Oliver D. Kingsley

Clinton Power Station, Unit 1
AmerGen Energy Company, LLC

cc:

John Skolds
Chief Operating Officer
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, Illinois 60555

K. A. Ainger
Director-Licensing
Mid-West Regional Operating Group
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, Illinois 60555

William Bohlke
Senior Vice President Nuclear Services
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, Illinois 60555

Robert Helfrich
Senior Counsel, Nuclear
Mid-West Regional Operating Group
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, Illinois 60555

John Cotton
Senior Vice President - Operations
Support
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, Illinois 60555

Document Control Desk-Licensing
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, Illinois 60555

Christopher Crane
Senior Vice President - Mid-West
Regional Operating Group
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, Illinois 60555

Illinois Department of Nuclear Safety
Office of Nuclear Facility Safety
1035 Outer Park Drive
Springfield, IL 62704

Jeffrey Benjamin
Vice President - Licensing and
Regulatory Affairs
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, Illinois 60555

J. M. Heffley
Vice President
Clinton Power Station
RR 3, Box 228
Clinton, IL 61727-9351

Robert J. Hovey
Operations Vice President
Mid-West Regional Operating Group
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, Illinois 60555

M. J. Pacilio
Plant Manager
Clinton Power Station
RR 3, Box 228
Clinton, IL 61727-9351

W. S. Iliff
Regulatory Assurance Manager (Acting)
Clinton Power Station
RR 3, Box 228
Clinton, IL 61727-9351

Oliver D. Kingsley

Clinton Power Station, Unit 1
AmerGen Energy Company, LLC

cc:

Resident Inspector
U.S. Nuclear Regulatory Commission
RR#3, Box 229A
Clinton, IL 61727

R. T. Hill
Licensing Services Manager
General Electric Company
175 Curtner Avenue, M/C 481
San Jose, CA 95125

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
801 Warrenville Road
Lisle, IL 60532-4351

Chairman of DeWitt County
c/o County Clerk's Office
DeWitt County Courthouse
Clinton, IL 61727

J. W. Blattner
Project Manager
Sargent & Lundy Engineers
55 East Monroe Street
Chicago, IL 60603

Mr. Oliver D. Kingsley, President
and Chief Nuclear Officer
Exelon Nuclear
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

November 14, 2001

SUBJECT: CLINTON POWER STATION, UNIT 1 - REQUEST FOR ADDITIONAL
INFORMATION (TAC NO. MB2210)

Dear Mr. Kingsley:

By letter dated June 18, 2001, AmerGen Energy Company, LLC, submitted a license amendment request for a 20-percent power uprate of the Clinton Power Station. The Nuclear Regulatory Commission staff has performed an initial review of your request and finds that it needs additional information to complete its review.

Therefore, I request that you respond to the enclosed request for additional information in order for the staff to complete its review in a timely manner. The questions were discussed and the response dates of November 16, 2001, for many of the questions, and November 30, 2001, for those questions that need input from your contractor were agreed upon with a member of your staff. The questions are unchanged from those sent by facsimile to a member of your staff on October 19, 2001, except for questions 10.5 and 10.12, which were slightly modified to eliminate proprietary information. Your staff indicated that the rest of the questions did not contain proprietary information.

Contact me if you have any questions.

Sincerely,

/RA/

Jon B. Hopkins, Senior Project Manager, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-461

Enclosure: As stated

cc w/encl: See next page

Distribution:

PUBLIC GGrant, RIII
PD3/2 r/f OGC
AMendiola ACRS
JHopkins
THarris

ADAMS ACCESSION NO.: ML013110260

OFFICE	PM:PD3-2	LA:PD3-2	SC:PD3-2
NAME	JHopkins	THarris	AMendiola
DATE	11/8/01	11/8/01	11/14/01

OFFICIAL RECORD COPY

CLINTON POWER STATION

DOCKET NO. 50-461

EXTENDED POWER UPRATE

REQUEST FOR ADDITIONAL INFORMATION

Questions for Sections 1.0 through 7.0 were forwarded by previous letters and there are no additional questions in those sections.

8.0 Health Physics

- 8.1 Extended power uprate (EPU) submittal Section 6.3.3 states that the normal radiation levels around the spent fuel pool (SFP) may increase slightly, primarily during fuel handling operations. Explain the reason for and the magnitude of these postulated increases in dose rate levels in the area of the SFP. Verify that these postulated dose rate increases will be bounded by the current radiation zone designations in the SFP area. If this postulated dose rate increase is due to higher activation of spent fuel assemblies, discuss any effects that the storage of these spent fuel assemblies in the SFP may have on dose rates in accessible areas adjacent to the sides or bottom of the SFP.
- 8.2 Section 8.4.2 states that there may be an increase in the activated corrosion product production, but does not quantify the expected increase in dose rates from the increase in activated corrosion products. Provide the following information: 1) the expected magnitude of the dose rate increases associated with this increase in activated corrosion product production, 2) what plant areas will be affected by this increase in dose rates from the increased level of activated corrosion products, 3) what affects this will have on occupancy levels in the affected areas, and 4) what affect this increase in activated corrosion product levels will have on curie loading for resin waste shipments.
- 8.3 Section 8.5.2 states that the post-operation radiation levels in most areas of the plant are expected to increase by no more than the percentage increase in power level. This section also states; however, that there are a few areas near the reactor water piping and liquid radwaste equipment where the expected radiation level increase could be slightly higher. Provide the specific locations of these areas where higher dose rates are predicted, give the reasons for the expected increase in radiation levels in these areas, and state the percentage increase in dose rates expected.

ENCLOSURE

- 8.4 Section 8.5.3 states that access to vital areas needed for accident mitigation will not be significantly affected by the EPU. Provide a list of vital areas requiring post-accident occupancy per NUREG-0737, Item II.B.2. For each of these vital areas, provide the calculated pre-uprate and post-uprate mission doses to an operator performing vital tasks following a loss-of-coolant accident (LOCA). Verify that the mission doses to personnel in these vital areas, as well as the calculated dose estimates for personnel performing required post-accident duties in the plant's Technical Support Center, are within the dose guidelines of general design criteria 19 (10 CFR Part 50, Appendix A).
- 8.5 Section 8.5 states that the EPU will result in an increase in radiation levels in most areas of the plant during both normal and post-operations. Describe what measures will be taken (e.g. special surveys of area radiation levels) during the power ascension to 20 percent above the current 100 percent power level to assure that all radiation areas are properly designated, posted, and controlled, in a timely manner, as required by Part 20 and plant technical specifications.
- 8.6 The submittal for the proposed 20 percent EPU states that outage-related modifications to support the implementation of these proposed changes will be made during the next two planned refueling outages. Provide an estimate of the additional occupational dose that will result from these planned modifications.
- 8.7 In Section 8.1 and Table 8-1 of the Environmental Report for the EPU, you use the actual waste release quantities and effluent doses for the year 2000 as a basis for calculating the estimated waste quantities and effluent doses for the EPU. It is not clear from this data how the year 2000 data compares with waste release quantities and effluent doses for previous years at Clinton Power Station (CPS). Provide this data for the previous five-year period for CPS and show that the estimated waste quantities and effluent doses for the EPU (using this expanded data set) are still conservative when compared to the Final Environmental Statement values and 10 CFR Part 50 Appendix I limits.
- 8.8 Section 8.1.1 of the Environmental Report lists the actual and projected volumes of low-level solid radwaste generated at CPS for the years 2000 and 2001, respectively. Provide an estimate of the projected percentage increase in volume of low-level solid radwaste generated at CPS following EPU.
- 8.9 Section 8.2.2 of the Environmental Report states that radiation from shine (offsite) is not significantly affected by the EPU. Provide the present nominal value for the skyshine external dose component (before EPU) and the estimated value following EPU and compare these values to the 25 mRem whole body dose limit of 40 CFR 190. Describe how this external dose component for skyshine is calculated (i.e., is it a calculated dose, is it dependent upon plant power level). Identify the dose receptor for this skyshine component (i.e., is the dose receptor a member of the public located offsite).
- 8.10 Describe any programs that CPS has implemented or plans to implement to counteract any potential increases in dose rates resulting from the proposed EPU. Examples of some initiatives that other licensees have implemented to reduce the level of activated corrosion products and to further inhibit the buildup of corrosion products in the reactor

coolant system include the use of noble metals injection, zinc injection, cobalt reduction, and chemical decontaminations of various plant systems and components.

9.0 Materials Engineering

Reference: GE NEDC-32989P

- 9.1 The subject report states that the increase in core average power results in higher reactor internal pressure differences (RIPDs). Provide a technical assessment as to how the Nuclear Regulatory Commission (NRC) staff-approved Boiling Water Reactor Vessel and Internals Project (BWRVIP) inspection and flaw evaluation (I&E) guidelines will continue to bound Clinton given this increase in RIPD.
- 9.2 With regards to the shroud loads, discuss more fully how the change in pressure (P) and total stresses, including providing the calculated and design-basis numbers, will affect the Upset and Faulted conditions. Also discuss more fully, including providing the calculated and design-basis numbers, how the change in postulated recirculation system line break LOCA acoustic and flow-induced loads will affect the Upset and Faulted conditions.
- 9.3 The subject report states that there is no increase in P across the core support. Discuss more fully why this is since the report previously stated that there is an increase in RIPD.
- 9.4 For the shroud supports, provide the calculated and design-basis numbers for the beam buckling stresses and the fatigue usage factor, and discuss more fully why the proposed changes do not significantly reduce the design safety margins.
- 9.5 For the core plate, provide the calculated and design-basis numbers for the beam buckling stresses and the fatigue usage factor, and discuss more fully why the proposed changes do not significantly reduce the design safety margins.
- 9.6 Provide the calculated and design-basis numbers for the P and total stresses generated for the reactor vessel and internal components due to the EPU. Include a discussion of how the Upset and Faulted conditions will be affected, including how the change in annulus temperature will affect the internal components.

Reference: Flow Accelerated Corrosion

- 9.7 It is stated in the submittal that the evaluation of and inspection for flow accelerated corrosion (FAC) after EPU is in compliance with NRC Generic Letter 89-08, "Erosion/Corrosion in Piping." This letter requires that an effective program is implemented to maintain structural integrity of high-energy carbon steel systems. Describe how was this program modified to account for EPU. If the code used in predicting wall thinning by FAC in this program is a generic code, specify it. However, if the code is plant-specific provide its description.

- 9.8 Why in the evaluation of the FAC effects in the main steam and attached piping system, monitoring of wall thinning in a single-phase is specified when this system contains two-phase fluids?
- 9.9 What is the highest change in the predicted wall thinning caused by EPU?
- 10.0 Mechanical Engineering
- 10.1 In Section 3.1.1 of Attachment E, you state that EPU evaluations are performed using the existing safety relief valve (SRV) setpoint tolerance analytical limits as a basis. The in-service surveillance testing of the plant's SRVs has not shown a significant propensity for high setpoint drift greater than 3 percent. During the extended refueling outage RF-6, all 16 SRVs were tested. The "as found" setpoint lift verification tests found that three of the SRVs exceeded their setpoint by greater than +/- 3 percent. Confirm whether the Clinton EPU SRV analyses are performed using + 3 percent setpoint tolerance.
- 10.2 In Section 3.3.2 of Attachment E, you indicate that the effect of EPU was evaluated to ensure that the reactor vessel components continue to comply with the existing structural requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code. For the components under consideration, the 1971 Edition of the Code with addenda to and including summer 1973, which is the construction code of record, was used as the governing Code. You also indicate that if a component underwent a design modification, the governing code for that component was the code used in the stress analysis of the modified component. Provide a summary of the components that were modified and the code editions/code cases (if applicable) other than the code of record that were used for the EPU evaluation.
- 10.3 In Section 3.3.2, you indicate that new stresses are determined by scaling the "original" stresses based on the EPU conditions (temperature and flow). The analyses were performed for the design, normal and upset, and emergency and faulted conditions. Provide a summary discussion of how you arrived to the scaling factors for the EPU at various service conditions. Also, provide an example to illustrate how scale factors were calculated and used in calculating the EPU stress and cumulative usage factor (CUF) at the feedwater nozzle blend radius.
- 10.4 In Section 3.3.2, you also indicate that if there is an increase in annulus pressurization, jet reaction, pipe restraint or fuel lift loads, the changes are considered in the analysis of the components affected for upset, emergency and faulted conditions. Provide a summary discussion of how these loads are affected by the proposed power uprate. Confirm whether and how these loads are incorporated in the EPU evaluation of the reactor vessel and internal components.
- 10.5 CPS will apply ASME Section XI Appendix L for fatigue assessment of the feedwater nozzle safe end. What is the CPS plan for demonstrating that the nozzle is acceptable? CPS has a fatigue-monitoring program that tracks the plant-specific fatigue. Provide a summary description of the program and how it is used to arrive at an accurate

representation of the fatigue usage. Also, provide a comparison of the CPS design-basis cycles and calculated CUFs for the feedwater nozzle safe end and the actual plant operating data from the fatigue-monitoring program.

- 10.6 In Section 3.3.2.2, Table 3-1, you indicate that the reactor vessel main flanges and bolts do not experience a change in temperature or pressure due to EPU. Hence, EPU stresses and usage factors are the same as the current values. Provide a summary of loads and design transients considered in the evaluation of closure flanges and bolts. Confirm whether and how pressure and temperature used in the evaluation of the reactor vessel main flanges and bolts are not affected by the EPU.
- 10.7 In Section 3.3.3, you state that the original acoustic loads on the reactor internal components, following a postulated recirculation line break, were also updated in accordance with current methodology. Provide a summary of the methodology and assumptions used in calculating the acoustic loads and provide an example to illustrate how the acoustic loads and flow induced loads were calculated, at the critical locations (i.e., shroud), due to recirculation line break for the EPU condition.
- 10.8 In Section 3.3.4, you indicate that for components experiencing increased loads due to EPU, the existing stresses are scaled-up in proportion to the loads, and the combined stresses and fatigue usage factors were compared to the code allowables for the various service conditions. Provide a summary describing how you arrived to the scaling factors for the EPU at various service conditions. Also, provide an example to illustrate how scale factors were calculated and the calculation of the EPU stress and CUF at the feedwater sparger pipe/tee and at the Jet pump riser brace.
- 10.9 In Section 3.3.5, you evaluate the effects of the EPU on the potential for flow-induced vibration of the reactor internal components due to the increase in steam product (>20 percent) in the core, the increase in the core pressure drop, and the increase in the recirculation pump speed. You indicated that the evaluation was based on the vibration data for the reactor internal components recorded during the startup testing of the NRC designated prototype plant and on operating experience from similar plants. The expected vibration levels under EPU conditions were estimated by extrapolating the vibration data recorded during startup testing at Kuo Sheng 1, the prototype plant, and on GE Nuclear Energy BWR operating experience. Discuss whether and how the recorded vibration data at Kuo Sheng 1 can be applicable for CPS and provide the basis for using the operating experience of similar plants. Also, provide a sample evaluation for the most critical components (i.e., jet pump).
- 10.10 In Section 3.3.5, you provide a list of components (including steam dryer) that were evaluated for the flow-induced vibration. You also indicate that during EPU operation, the components in the upper zone of the reactor, such as the steam separators and dryers, are mostly affected by the increased steam flow. Provide recorded or testing data and a summary of the evaluation with regard to the flow-induced vibration affecting steam dryers. Discuss the potential for flow-induced vibration of the steam dryers due to various mechanisms, including, in particular, the fluid-elastic instability in the steam separators and dryers at the proposed power level. If the details of the analysis and the

results are documented in a report, submit the report for staff review. In light of the discussion in GE SIL No. 474 and BWRVIP-06 report, discuss how you can ensure that the steam dryer will maintain its structural integrity during the EPU operation.

- 10.11 Provide a discussion on the potential for excessive vibrations, high noise levels, and the instrument lines leakage that might be caused by the increased recirculation pump speed or flow for the proposed power uprate, as described in the NRC Information Notice 95-16. Confirm whether the jet pump riser brace will be susceptible to vibration from the recirculation pump vane passing frequency due to the EPU at CPS.
- 10.12 In reference to Section 3.5, provide a discussion of the methodology and assumptions used for evaluating the reactor coolant pressure boundary piping (RCPB) systems for the proposed power uprate. Also, provide the calculated maximum stresses and fatigue usage factors for both the current design-basis and the EPU conditions, critical locations on the evaluated RCPB piping systems, allowable stress limits, and the code and code edition used in the evaluation for the power uprate. 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 used and provide your justification for their use by specifying how these codes were benchmarked for such applications.
- 10.13 Provide a summary of your evaluation of pipe supports, nozzles, penetrations, guides, valves, pumps, heat exchangers and anchors at the power uprate condition. The evaluation should include the methodology, assumptions, and results of the evaluation for the critical piping systems affected by the proposed power uprate. 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 your justification for their use by specifying how were these codes benchmarked for such applications.
- 10.14 In Section 3.5.5, you indicate that the main steam (MS) and feedwater (FW) piping will experience increased vibration levels, approximately proportional to the square of the flow velocities. For the proposed power uprate, the flow rates and flow velocities will increase by more than 20 percent of the flow rate at the original rated thermal power for the MS and FW piping systems. Provide an evaluation of the cumulative fatigue usage factor (in addition to the startup and shutdown cycles), and the potential for flow-induced vibration in the MS and FW piping (during the normal and upset operations), and in heat exchangers following the power uprate. In Section 10.4.3, you indicated that the vibration level may even be higher if other flow induced vibration mechanisms occur. Provide a discussion on the potential for flow-induced vibration of the main steam and feedwater piping due to various mechanisms, including, in particular, the fluid-elastic instability at the proposed power level.
- 10.15 In Section 4.1.2.3 concerning subcompartment pressurization, you state that the increase in actual asymmetrical loads on the vessel, attached piping and biological shield wall, due to the postulated main steam and feedwater pipe breaks in the annulus between the reactor vessel and biological shield wall is bounded by the original analysis. The biological shield wall and component designs remain adequate, because there is

sufficient pressure margin available. Discuss how the feedwater line break mass and energy releases at EPU power level of 3473 MWth are bounded by the licensing basis mass and energy releases at the current power level of 2894 MWth and confirm whether the biological shield wall and the reactor vessel and internals will be affected by the proposed power uprate as a result of the EPU.

- 10.16 Discuss the functionality of safety-related mechanical components (i.e., safety-related valves and pumps, including air-operated valves (AOV) and safety and relief valves) affected by the proposed 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 AOV and motor-operated valves will be capable of performing their intended function(s) following the proposed power uprate including such affected parameters as fluid flow, temperature, pressure and differential pressure, and ambient temperature conditions. Identify the mechanical components that were not evaluated at the uprated power level. Also, discuss the effects of the proposed power uprate on the pressure locking and thermal binding of safety-related power-operated gate valves for Generic Letter (GL) 95-07. Confirm whether and how the EPU peak drywell temperature in exceedance of 330 °F does not affect the current CPS GL-96-06 evaluation.
- 10.17 In reference to Section 3.11, provide a summary addressing your evaluation of the effects of the proposed power uprate on the balance of plant (BOP) piping, components, and pipe supports, nozzles, penetrations, guides, valves, pumps, heat exchangers and anchorages. Also, provide the calculated maximum stresses and fatigue usage factors for the most 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.