November 5, 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, you 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 the remainder of the questions because they require 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 4, 2001, except for questions 3.8 and 3.11, 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

cc:

John Skolds
Chief Operating Officer
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

John Cotton
Senior Vice President - Operations Support
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

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

Robert J. Hovey
Operations Vice President
Mid-West Regional Operating Group
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

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

Document Control Desk-Licensing
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

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

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

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JHopkins
THarris

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NAME	JHopkins	THarris	AMendiola
DATE	11/2/01	11/2/01	11/2/01

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CLINTON POWER STATION

DOCKET NO. 50-461

EXTENDED POWER UPRATE

REQUEST FOR ADDITIONAL INFORMATION

Questions for Sections 1.0 and 2.0 were asked in a letter dated October 3, 2001, and there are no additional questions in those sections.

3.0 Reactor Systems - Attachment E, NEDC-32989P

Questions 3.1, 3.2, and 3.3 were asked in the letter dated October 3, 2001; therefore, these questions begin with 3.4.

3.4 It is stated that "The plant conditions assumed in the ELTR2 Evaluations bound the conditions for this EPU." But ELTR2 evaluations were performed mainly for BWR/3 and BWR/4 plants with little reference to BWR/6 plants. Describe in detail how ELTR2 BWR/6 evaluations bound the conditions for Clinton. Identify the appropriate sections of ELTR2 which support the above conclusion. Also, Clinton extended power uprate (EPU) operation will not involve any reactor pressure increase, while ELTR2 assumes an increase in reactor pressure. Confirm that the above conclusion is valid for Clinton EPU operation.

3.5 In Section 1.2.3, "Approach (a) Reactor Core and Fuel design Performance," there is the following statement, "Analyses are performed for a representative equilibrium cycle with the reactor core operating at EPU conditions."

In Section 2.1, " Fuel Design and Operation," there is the following statement, "Detailed fuel cycle calculations of representative core design for this plant demonstrate a representative core design for this plant...."

Explain in detail what is "a representative equilibrium fuel cycle" and what is a "representative core design"? Also, explain in detail PUREC (Power Uprate Representative Equilibrium Cycle).

3.6 Reference Section 2.1, "Fuel Design and Operation"

The staff safety evaluation report (SER) for ELTR-2, states that "Each applicant for extended power uprate should adhere to existing radial power shape limitations when designing core reloads for uprated conditions. Provided that the radial power distribution remains within the bounds of the loss-of-coolant accident (LOCA)/emergency core cooling system (ECCS) assumptions, the effect of power uprate on the short-term response to a postulated LOCA should be minimal." Confirm that this is true for the Clinton EPU.

ENCLOSURE

3.7 Reference Section 2.2.1, “Minimum Critical Power Ratio”

The staff SER for ELTR-2, states that “A plant-specific power uprate and the reload submittal should contain analyses to confirm that the safety limit for minimum critical power (SLMCPR) is appropriate for the average bundle power at the uprated conditions.” Describe in detail that the SLMCPR is appropriate for the EPU?

3.8 The following transients are to be analyzed for EPU conditions as required by ELTR-1, Appendix E. Why were these transients not analyzed for the Clinton EPU?

Pressure Regulator Failure Downscale
Main Steam Isolation Valve (MSIV) valve closure-Direct Scram
One MSIV valve closure
Turbine Trip No Bypass-Fluxsram

3.9 Reference Section 2.4, “Stability”

Confirm that the Option III continues to be applicable to the EPU conditions.

Clinton is going to implement Option III OPRM system for EPU operation. General Electric Company (GE) issued a 10 CFR Part 21 interim report notifying the licensees that OPRM SCRAM set points are non-conservative due to a non-conservative GE analysis (Ref. Daily Event Report Number 38099 regarding Perry dated 7/27/01). Explain Clinton’s position on the OPRM SCRAM set points in conjunction with the DIVOM curve determination for GE14 fuel under Clinton EPU conditions

3.10 Reference Section 3.4, “Recirculation System”

(a) The staff SER for ELTR-2, states that ”Plant-specific data will be reviewed to confirm that the existing recirculation system will accommodate the increase in resistance, due to an increase in core average void fraction at the uprated condition when operating at maximum core flow.” Confirm that this review was performed for Clinton EPU operation.

(b) The staff SER for ELTR-2, states that “Each applicant for EPU will be expected to review plant-specific operating data to ensure that the recirculation system, including the recirculation pumps and its associated components, will accommodate the increase in system pressure as well as the increase in flow resistance that is expected due to the increase in core average void faction due to uprate.” Even though there is no pressure increase, flow resistance is changed due to power uprate. Confirm that the plant operating data was reviewed.

3.11 Reference Section 4.3, “ECCS performance”

The results of SAFER/GESTR-LOCA analysis are presented in Table-2. Explain the methodology used for the LOCA analysis. Describe in detail what parameters are used for this analysis which are different from the actual core design. Is this analysis based on an equilibrium core design?

3.12 Spray cooling of the core following the LOCA is important for Clinton post-accident long-term cooling. It is not clear whether a particular spray pattern is assumed in the LOCA analysis or whether the LOCA analysis simply assumes that a given amount of water is pumped inside through the top of the core without any spray distribution. Are you relying on a specific spray pattern to obtain the core spray heat transfer coefficient assumed in the analysis? What is the value of the core spray heat transfer coefficient assumed in the Appendix K analysis?

3.13 Reference Section 9.1, “Reactor Transients”

The staff SER for ELTR1 states, “Only the limiting transient need be included in the uprate amendment request, but a list of all transients analyzed in support of power uprate should be included, with an explanation of how the limiting transients were selected.” And the staff SER for ELTR2 states, “...Operating limit MCPR will be documented in each plant-specific power uprate submittal....” List the transients considered with an explanation how the limiting transient was selected.

3.14 Reference Section 9.3.1, “Anticipated Transient without Scram (ATWS)”

What ATWS events were analyzed at the EPU condition? Confirm that for all limiting ATWS conditions, the standby liquid control system (SLCS) will be able to inject whenever it can be actuated without lifting the SLCS pump discharge relief valves. For example, will the SLCS be able to inject the required flow rate at the assumed time for the ATWS/LOOP event without reaching the rated SLCS relief valve set point?

Regarding the ATWS recovery scenario and the ATWS rule, does the power uprate affect the time available for the recovery process? Is AmerGen satisfied that operators can identify and adequately respond to an ATWS event in four minutes?

3.15 Confirm that the EPU will be implemented in two stages over 2 operating cycles and the first power uprate will be about a 7 percent increase and the second increase about 13 percent.

3.16 ELTR-1, Section 5.6.2, “Recirculation System,” states that “A review of plant-specific operating data will be performed to confirm that the recirculation system will accommodate the expected insignificant increase in the flow resistance at the uprated power condition when operating at maximum core flow. Potential increases in system vibration will be evaluated from plant data.” Confirm that this plant-specific evaluation was conducted for Clinton EPU operation.

3.17 Reference Section 10.6, “Operator training and Human Factors” and ELTR-2, Section 2.3, “Emergency Operating Procedures.”

Confirm that emergency operating procedures (EOPs) will be reviewed and appropriate changes will be made to the plant variables and limit curves. Since Operator response time will be reduced due to power uprate, discuss the Operator training planned for EPU conditions.

3.18 The EPU submittal did not address whether operation at the higher MELLLA/EPU operation with introduction of GE14 fuel might affect the potential for and impact of thermal-hydraulic instability. Section L.3.1, “Power Conditions for ATWS Evaluation,” and L.3.2, “Operator Action,” of the ELTR1 discuss some aspects of the ATWS instability and typical ATWS operator actions. NEDO-32047-A, “ATWS Rule Issues Relative to BWR Core Thermal-hydraulic Stability,” provided generic evaluations of ATWS instability events for BWR/5 and BWR/6.

Confirm that the power shape assumed in NEDO-32047A bounds the conditions expected for Clinton during ATWS.

Confirm that the Clinton EOPs will be consistent with the recommendations of ELTR1 and the Nuclear Regulatory Commission (NRC) staff’s positions in NEDO-32047-A SER.

3.19 Table 5-1, “ATWS/Stability Transient Analysis Parameters,” of NEDO-32047-A provides the initial conditions assumed in the ATWS instability evaluations. Confirm that the key parameters (i.e initial feedwater temperature, core power density) used in the generic ATWS instability evaluation remain applicable and bounding for Clinton. Explain why the generic ATWS instability analysis is applicable to the Clinton EPU operation using the factors that affect thermal-hydraulic instability.

Questions for Section 4.0 were asked in the letter dated October 3, 2001, and there are no additional questions for that section.

5.0 Plant Systems

5.1 As a result of plant operations at the proposed EPU level, the decay heat load for any specific fuel discharge scenario will increase. In Section 6.3.1 of the Safety Analysis Report (SAR) for Clinton Power Station (CPS) EPU, AmerGen (licensee) stated that EPU does not adversely affect the capability of the fuel pool cooling and cleanup system (FPCCS) to keep the spent fuel pool (SFP) temperature at or below the design temperature and maintain adequate SFP cooling during normal refueling and under full-core offload conditions. However, AmerGen did not provide the detailed discussion of its SFP cooling evaluations in the SAR, please provide the following information for both pre-EPU and EPU conditions:

a. SFP heat loads and the corresponding peak calculated temperatures during planned[1] (normal) partial and/or full-core offload outages and unplanned[2] (abnormal) full-core offload outages for pre-EPU and EPU conditions.

b. Assumptions used in the SFP thermal-hydraulic analysis (i.e. fuel assemblies “in-reactor” hold time, number of the previously discharged spent fuel assemblies (SFAs) in the SFP, ultimate heat sink temperature, etc.) for each scenario.

c. For the planned refueling outages (with either partial or a full-core offload), discuss how the most severe single-failure (e.g. failure of: a FPCC system train, a residual heat removal system train, emergency diesel generator, etc.) has been identified and accounted for in the SFP thermal-hydraulic analyses. (A single-failure need not be assumed for the unplanned full-core offload events.)

d. Since the residual heat removal (RHR) system provides supplemental cooling, when needed, to maintain the SFP below 150 °F, prior to a planned or unplanned full-core offload event, how many trains of FPCC system and RHR system are required to be operable and available for SFP cooling?

e. For the planned refueling outages (with either partial or a full-core offload), If the calculated peak SFP temperature is above 150 °F, provide thermal stress analyses[3] to demonstrate that the SFP structure can withstand the new high temperature for the duration of time during which the SFP temperature is above 150 °F.

5.2 Section 9.1.3.3.a of the CPS Updated Safety Analysis Report (USAR) states that If the SFP water temperature rises above 120 °F, the FPCC heat exchanger cooling media may be transferred by operator from the component cooling water (CCW) system to the shutdown service water system to increase the heat removal capability of the FPCC system. Also, Section 9.1.3.3.b of the CPS USAR states that if it appears that the SFP water temperature will exceed 150 °F when the reactor is in a cold shutdown condition, the operator may connect the FPCC system to the RHR system to provide supplemental cooling to the SFP. Discuss the provisions that have been established in the plant operating procedures to ensure that the shutdown service water system and the RHR system will be properly aligned for SFP cooling, when needed.

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[1] A planned offload is the offload of fuel assemblies to the SFP for any expected (or planned) reason (e.g. refueling outage).

[2] An unplanned offload is the offload of fuel assemblies to the SFP due to an unforeseen condition (e.g., unexpected shutdown that includes an offload).

[3] Mechanical and Civil Engineering Branch (EMEB) has the primary review responsibility for structural thermal stress analyses.

5.3 Section 9.1.3 1.2.k of the CPS USAR states that:

"Refueling is typically done in the late fall, winter or early spring. During these times of the year, lake (ultimate heat sink) temperatures are low. Heat sink temperatures for service water and CCW are typically less than 60 °F and 80 °F, respectively. Under these conditions, with the vessel disassembled and the reactor pool flooded up to greater than 23 feet, the SFP cooling system alone can remove the core residual heat and maintain the SFP temperatures within limits."

Since the heat removal capability of the SFP cooling system is a function of the lake temperature, and the decay heat load is a function of the SFAs "in-reactor" hold time prior to being discharged from the reactor, AmerGen can alternately opt to perform a cycle-specific SFP thermal-hydraulic evaluation prior to every planned offload using the actual conditions at the time of the offload. The "in-reactor" hold time for offload can be adjusted, as long as the time is not shorter than what is assumed for the fuel handling accident.

If AmerGen opts to perform a cycle-specific SFP thermal-hydraulic evaluation prior to every planned offload using the actual conditions at the time of the offload, please provide the following information:

a. The calculated SFP peak temperatures at various lake water temperatures (i.e. 40 °F, 60 °F, 80 °F 90 °F, 95 °F, etc.) and their corresponding SFAs "in-reactor" hold time required; coincident[4] time after reactor shutdown; and coincident decay heat load. For the case with the highest decay heat load, also provide the "time-to-boil" and maximum boil off rate.

b. Discuss the provisions established or to be established in plant operating procedures to required evaluations being performed to determine/establish SFAs "in-reactor" hold time required prior to discharge SFAs from the reactor to ensure that the SFP operating temperature limit of 150 °F will not be exceeded.

5.4 In order to determine whether adequate controls exist to ensure the guidance of Standard Review Plan, Section, 9.1.3, "Spent Fuel Pool Cooling and Cleanup System" are met, the NRC staff needs to understand the provisions established or to be established in plant operating procedures to monitor and control the SFP water temperature during full-core offload events. Information should include:

a. How often the SFP water temperature will be monitored during planned and unplanned core off-load outages.

b. The set-point of the high water temperature alarm for the SFP.

c. Information supporting a determination that there is sufficient time for operators

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[4] The time after reactor shutdown at which the SFP water reaches its temperature limit of 150 °F.

to intervene in order to ensure that the temperature limit of 150 °F will not be exceeded.

d. The mitigative actions (i.e. prohibit fuel handing, aligning other systems to provide SFP cooling, etc.) to be taken in the event of a high SFP water temperature alarm.

5.5 In the unlikely event that there is a complete loss of SFP cooling capability, the SFP water temperature will rise and eventually reach boiling temperature. Provide the time to boil (from the pool high temperature alarm caused by loss-of-pool cooling to boiling) and the boil-off rate (based on the highest heat load from the planned or unplanned full core off-load). Also, discuss sources and capacity of make-up water and the methods/systems (indicating system seismic design Category) used to provide the make-up water.

5.6 With regard to the CCW system, in Section 6.4.3 of the SAR AmerGen stated that the only increases in heat loads due to EPU are the operation of the reactor recirculating pumps at higher power level, and an increase in the fuel pool coolers heat load. Based upon a service water temperature of 92 °F, the CCW system has sufficient heat removal capacity to accommodate the increased heat load due to EPU. However, Section 9.1.3 1.2.g in the CPS "USAR" states that the design temperature for service water is 95 °F. Also, Table 9.2.8 of the USAR indicates that the design service water inlet temperature for the CCW heat exchangers is 95 °F. Please provide detailed discussion to clarify the discrepancy.

5.7 For the containment pressure and temperature analyses in Section 4.1.1of the SAR, please provide all the main parameters for the EPU that are different from the USAR values and the reason for the differences (besides those that are power related). Also provide the peak pressure and temperature curves for the EPU analyses.

5.8 In Table 4-1, for peak suppression pool and containment temperatures, it is indicated that the current methods (EPU) calculated a peak suppression pool temperature of 167.5 °F and a peak containment temperature of 149.2 °F against the USAR values of 180.3 °F for the recirculation suction line break (RSLB). Please discuss the reasons for the above decreases in peak pool and containment temperatures.

5.9 In Table 4-1, for peak drywell pressure, it is indicated that the current methods (EPU) calculated a peak drywell pressure of 23.1 psig against a USAR value of 18.9 psig for main steamline break (MSLB). It is also indicated that the current methods calculated a peak wetwell pressure of 9.4 psig against a USAR value of 7.7 psig for MSLB and a peak containment pressure of 3.2 psig against a USAR value of 8.7 psig for the RSLB. Please discuss the reasons for the above changes in the peak drywell, wetwell and containment pressures.

5.10 In Section 4.1.2.1, for the containment dynamic loads due to LOCA, it is indicated that for conservatism, the USAR assumed a LOCA bubble pressure of 20.1 psig which is greater than the peak calculated drywell pressure of 18.9 psig for LOCA loads. But for the EPU, it is indicated that the use of the peak drywell pressure as the basis for the peak drywell wall pressure is very conservative. The peak drywell pressure occurs a few tenths of a second after vent clearing (calculated 23.2 psig) while the peak drywell wall pressure occur immediately after vent clearing (calculated 19.0 psig) when the LOCA bubble first forms. Please indicate why the LOCA bubble pressure was assumed greater than the peak drywell pressure for USAR and now calculated lower for EPU.

5.11 In Section 4.1.2.3, for the subcompartment pressurizing, it is indicated that the mass and energy releases for recirculation inlet line break in the USAR used the methods provided in NEDO-25458 and that the subcooled critical mass flux was conservatively calculated using the Henry-Fauske subcooled critical mass flux found in ANCR-NUREG-1335 “Relap4/MOD5 A Computer Program for Transient Thermal Hydraulic Analysis. But for the EPU, Moody slip flow model for subcooled blowdown recommended in NEDO-24548 was used. Please indicate why the Henry-Fauske method was used for USAR over the Moody slip flow model for subcooled blowdown now used for EPU.

5.12 In Section 4.7, for the post-LOCA combustible gas control system (CGCS), it is indicated that post-LOCA production of hydrogen and oxygen by radiolysis increases proportionally with power level and this increase has a minor impact on the time available to start the system to maintain hydrogen below the lower flammability limit. Please discuss the effect of EPU on other parameters that contribute in the production of hydrogen. Also provide curves of the volume percent concentration of hydrogen in the containment as a function of time with and without the operation of the CGCS after a LOCA.

5.13 In Section 4.4, for the main control room atmosphere control system (MCRACS), it is stated that the amount of charcoal in the makeup air train is more than adequate to handle additional iodine loading and the additional decay heat as a result of radionuclides deposited. Provide technical justification for this assessment. Also, discuss any additional heat loads affecting the MCRACS, as a result of EPU.

5.14 In Section 6.6, for the power dependent heating ventilation and air conditioning (PDHVAC), system assessment is generalized by stating at several places that EPU results in a small increase in the heat load, increased heat load is within the margin of the coolers, or other areas are unaffected by the EPU because the process temperatures and electrical heat loads remain relatively constant such as “the heat load in the containment and auxiliary building steam tunnels increases due to the increase in the feedwater process temperature. The increased heat load is within the margin of the steam tunnel area coolers. In the drywell, the increase in feedwater process temperature and the slight increase in the recirculation pump motor horse power are within the margin in the system capacity.” Provide your detailed technical justification for these generalized statements for the systems, structures or components served by the PDHVAC. Discuss the specific design capability and expected increase in heat loads due to the EPU.

6.0 Electrical Systems

6.1 Provide the grid stability analysis.

7.0 Instrumentation and Control Systems

7.1 For power uprates, the GE setpoint methodology discussed in GE document NEDC-32989P has been used to determine the acceptability or changing the setpoint. Therefore, this methodology should be referenced in the basis section of the technical specification. Also, confirm that this methodology has been used for both balance of plant (BOP) as well as nuclear steam supply system (NSSS).

7.2 Table 5.1 of NEDC-32989P provides changes in the analytical limit for certain plant parameters for the current and power uprate condition. Provide the instrument setpoints and allowable values at both the current and uprate power conditions for the instruments identified in Table 5.1.

7.3 Sections 5.1 and 5.2 provide the discussion on the effect of power uprate on the NSSS and BOP systems. However, a discussion of the effect of power uprate on instrumentation and control is lacking. This discussion should include all the changes to instrumentation and control required because of the changes in the setpoints (not covered by the Table 5.1), instrumentation scaling changes, obsolescence, or the changes in the control philosophy.