February 7, 2002

Mr. Michael P. Gallagher
Director-Licensing
Exelon Corporation
200 Exelon Way
Kennett Square, PA 19348

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3

Dear Mr. Gallagher:

By letter dated July 2, 2001, Exelon Generation Company, LLC (Exelon), submitted for Nuclear Regulatory Commission (NRC) review an application, pursuant to 10 CFR Part 54, to renew the operating licenses for the Peach Bottom Atomic Power Station, Units 2 and 3. The NRC staff is reviewing the information contained in this license renewal application and has identified, in the enclosure, areas where additional information is needed to complete its review. Specifically, the enclosed request for additional information (RAI) is from Chapter 4.0 Time-Limited Aging Analyses, Section 4.1 Identification of TLAAs, Section 4.2 Reactor Vessel Embrittlement, Section 4.3 Metal Fatigue, and Section 4.7.1 Reactor Vessel Main Steam Nozzle Cladding Removal Corrosion Allowance.

Please provide a schedule by letter, or electronic mail for the submittal of your response within 30 days of the receipt of this letter. Additionally, the staff would be willing to meet with Exelon prior to the submittal of the response to provide clarification of the staff's request for additional information.

Sincerely,

/***RA***/

Raj K. Anand, Project Manager
License Renewal and Environmental Impacts Program
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Docket Nos. 50-277 and 50-278

Enclosure: As stated

cc w/encl: See next page

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REQUEST FOR ADDITIONAL INFORMATION
PEACH BOTTOM UNITS 2 AND 3

4.0 Time-Limited Aging Analyses

4.1 Identification of TLAAs

RAI 4.1-1

Table 4.1-1 of the LRA identifies flaw growth analysis as a TLAA for feedwater nozzle and control rod drive return line nozzle. The table does not identify the flaw growth analyses for other reactor coolant pressure boundary components as TLAAs. Flaws in Class 1 components that exceed the size of allowable flaws defined in IWB-3500 of the ASME Code need not be repaired if they are analytically evaluated to the criteria in IWB-3600 of the ASME Code. The analytic evaluation requires the applicant to project the amount of flaw growth due to fatigue and stress corrosion cracking mechanisms, or both, where applicable, during a specified evaluation period. The applicant is requested to identify all Class 1 components that have flaws exceeding the allowable flaw limits defined in IWB-3500 and that have been analytically evaluated to IWB-3600 of the ASME Code, and to submit the results of the analyses that indicate whether the flaws will satisfy the criteria in IWB-3600 for the period of extended operation.

4.2 Reactor Vessel Neutron Embrittlement

RAI 4.2-1

The applicant describes its evaluation of reactor vessel neutron embrittlement time-limiting analyses in Section 4.2 of the LRA. The evaluation shows that the RTNDT, reflood thermal shock analysis, Charpy USE, P-T limit, circumferential weld and axial weld integrity evaluations are all dependent upon the neutron fluence. The applicant states that it will initiate the calculations for end-of-life fluence for a 60-year licensed operating period (54 EFPY) using the GE fluence methodology after the NRC approves it.

In order to determine whether neutron irradiation embrittlement will satisfy the time-limited aging analyses criteria in 10 CFR Part 54.21(c)(1) the applicant must determine the adjusted reference temperature (ART) and the Charpy Upper-Shelf Energy (USE) at the end of the license renewal period (60 years of operation). These analyses require that the applicant determine the peak neutron fluence at the end of the license renewal period. Therefore, the applicant is requested to calculate the peak neutron fluence at the clad-steel interface and the 1/4 thickness location in the reactor vessels at the end of the license renewal period using a methodology approved by the staff and adheres to the guidance in Regulatory Guide (RG) 1.190,"Calculation and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence." Note: The staff approved a neutron fluence calculation methodology submitted by GE Nuclear Energy (NEDC-32983P,"General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluation") in a letter dated September 14, 2001 from S.A. Richards (NRC) to J.F. Klapproth (GENE).

RAI 4.2-2

The applicant has reviewed the reflood thermal shock analysis for Peach Bottom in Section 4.2.1 of the LRA. For the reflood thermal shock event, the peak stress intensity at 1/4 of vessel thickness from inside occurs about 300 seconds after LOCA. At 300 seconds, the analysis shows that the temperature of the vessel wall at 38.1-mm (1.5-inch) depth location is approximately 204^{o} C (400^{o} F). The applicant states that the reflood thermal shock analysis for 40-years of operation (32 EFPY) will be bounding and valid for the license renewal term because the vessel beltline material ART, even after 60 years of irradiation, is expected to be low enough to ensure that the material is in the Charpy upper shelf region at 204^{o} C. The applicant is requested to present technical basis for expecting the vessel beltline material ART after 60 years of irradiation to be low enough so that the material is in the Charpy upper shelf region at 204^{o} C.

RAI 4.2-3

EPRI TR-113596, "BWRVIP BWR Reactor Pressure Vessel Inspection and Flaw Evaluation Guidelines," BWRVIP-74, September 1999, performs a generic analysis and determines that the percent reduction in Charpy USE for the limiting BWR/3-6 plates and BWR non-Linde 80 submerged arc welds are 23.5 percent and 39 percent, respectively. Since this is a generic analysis, the applicant is requested to submit plant-specific information to demonstrate that the beltline materials of Peach Bottom Units 2 and 3 RPVs meet the criteria specified in the BWRVIP-74 report at the end of the license renewal period. The applicant is requested to submit the information specified in Tables B-4 and B-5 of EPRI TR-113596.

RAI 4.2-4

In Section 4.2.1 of the LRA, the applicant states that it will recalculate the vessel end-of-life RTNDT (54 EFPY operating period) according to Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves." But this code case is not about calculating the end-of-life RTNDT. It is about the use of the reference fracture toughness curve KIc, as found in Appendix A of ASME Section XI, in lieu of KIa, as given by Fig. G-2210-1 in Appendix G for the development of P-T limit curves. The applicant is requested to resolve this inconsistency about the use of Code Case N-640.

RAI 4.2-5

In Section 4.2.2 of the LRA, the applicant states that it will calculate vessel P-T limit curves for a 60 years, 54 EFPY operating period, after the NRC has approved GE fluence methodology. The applicant is requested to submit P-T limit curves for a 60-year design (54 EFPY operating period) for Peach Bottom using the neutron fluence calculation methodology discussed in RAI 4.2-1.

RAI 4.2-6

Sections 4.2.3 and A.5.1.2 of the LRA discuss inspection of the Peach Bottom RPV circumferential welds. These sections of the LRA indicate that Peach Bottom will use an approved technical alternative in lieu of ultrasonic testing of RPV circumferential shell welds.

The technical alternative is discussed in the staff's final SER of the BWR Vessel and Internals Project BWRVIP-05 Report, which is contained in a July 28, 1998 letter to Carl Terry, BWRVIP Chairman. This indicates BWR applicants may request relief from inservice inspection requirements of 10 CFR 50.55a(g) for volumetric examination of circumferential RPV welds by demonstrating: (1) at the expiration of the license, the circumferential welds satisfy the limiting conditional failure probability for circumferential welds in the evaluation, and (2) they have implemented operator training and established procedures that limit the frequency of cold over-pressure events to the amount specified in the report. The letter indicated that the requirements for inspection of circumferential RPV welds during an additional 20-year license renewal period would be reassessed, on a plant specific basis, as part of any BWR license renewal application.

Section A.4.5 of Report BWRVIP-74 indicates that the staff's SER conservatively evaluated BWR RPVs to 64 effective full power years (EFPY), which is 10 EFPY greater than what is realistically expected for the end of the license renewal period. It also discusses the impact of radiation embrittlement on circumferential RPV welds. Since this was a generic analysis, the applicant must submit plant-specific information to demonstrate that the Peach Bottom beltline materials meet the criteria specified in the report. To demonstrate that each of the Peach Bottom Unit 2 and 3 vessels has not been embrittled beyond the basis for the technical alternative, the applicant is requested to supply: (1) a comparison of the neutron fluence, initial RTNDT, Chemistry Factor, amounts of copper and nickel, delta RTNDT and Mean RT NDT of the limiting circumferential weld at the end of the renewal period to the 64 EFPY reference case in Appendix E of the staff's SER, and (2) an estimate of conditional failure probability of the RPV at the end of the license renewal term based on the comparison of the Mean RTNDT for the limiting circumferential weld and the reference case. Should the applicant request relief from augmented ISI requirements for volumetric examination of circumferential RPV welds during the period of extended operation, the applicant is requested to demonstrate that (1) at the expiration of the license, the circumferential welds satisfy the limiting conditional failure probability for circumferential welds in the evaluation, and (2) they have implemented operator training and established procedures that limit the frequency of cold over-pressure events to the amount specified in the report.

RAI 4.2-7

Sections 4.2.4 and A.5.1.3 of the LRA discuss inspection of the Peach Bottom RPV axial welds. These sections of the LRA state that Peach Bottom will perform plant-specific analyses following the generic analyses presented in BWRVIP-05 report. These analyses support a conclusion of an NRC SER, enclosed in the March 7, 2000 letter to Carl Terry, BWRVIP Chairman, that the RPV failure frequency due to failure of the limiting axial welds in the BWR fleet at the end of 40 years of operation is below 5 x 10-6 per reactor year, given the assumptions on flaw density, distribution and location described in the SER. Since the results apply only for the initial 40-year license period of BWR plants, applicants for license renewal must submit plant-specific information applicable to 60 years of operation.

Since the BWRVIP analysis was generic, the applicant is requested to submit plant-specific information to demonstrate that the Peach Bottom beltline materials meet the criteria specified in the report. To demonstrate that the vessel has not been embrittled beyond the basis for the staff and BWRVIP analyses, the applicant is requested to submit: (1) a comparison of the neutron fluence, initial RTNDT, Chemistry Factor, amounts of copper and nickel, delta RTNDT and Mean RTNDT of the limiting axial weld at the end of the renewal period to the reference cases in the BWRVIP and staff analyses and (2) an estimate of conditional failure probability of the RPV at the end of the license renewal term based on the comparison of the Mean RTNDT for the limiting axial welds and the reference case. If this comparison does not indicate that the RPV failure frequency for axial welds is less than 5 x 10-6 per reactor year, the applicant must submit a probabilistic analysis to determine the RPV failure frequency for axial welds.

4.3 METAL FATIGUE

RAI 4.3-1

Section 4.3.1 of the LRA indicates that the reactor vessel closure studs are projected to have a CUF>1.0 during the current period of operation. The LRA further indicates that the studs are included in the fatigue management program (FMP). Provide the reason the projected CUF for the closure studs is expected to exceed 1.0 during the current operating period. Discuss the potential corrective actions that will be implemented prior to the period of extended operation.

RAI 4.3-2

Section 4.3.1 of the LRA indicates that an improved program is being developed which will use temperature, pressure, and flow data to calculate and record accumulated usage factors for critical RPV locations and subcomponents. Describe how the monitored data will be used to calculate the usage factors for the monitored components. Indicate how the fatigue usage of the monitored components is estimated for the time prior to implementation of the improved program.

RAI 4.3-3

Section 4.3.2.1 of the LRA indicates that fatigue analyses of the core shroud supports were reevaluated for effects of increased recirculation pump starts with the loop outside thermal limits. Describe the reevaluations that were performed considering an increase in recirculation pump starts. Indicate the reason that the reevaluations were necessary.

RAI 4.3-4

Section 4.3.2.1 of the LRA indicates that the limiting fatigue usage for the core shroud and jet pump assembly is based on the evaluation of a plant with a configuration similar to PBAPS. As discussed in RAI 4.3-3, the PBAPS core shroud supports were reevaluated for the effects of increased recirculation pump starts with the loop outside thermal limits. Indicate whether the increase in recirculation pumps starts has any impact on the fatigue usage of the core shroud and jet pump assembly.

RAI 4.3-5

Section 4.3.3.3 of the LRA indicates that the NSSS vendor specified the RHR system for a finite number of cycles for each of its elevated-temperature operating modes. The LRA also indicates that no description of these design operating cycles was found in the BPAPS licensing basis documents. According to the LRA, Group 1 RHR piping inside the drywell was analyzed to the ASME Section III Class 1 rules. The LRA further indicates that an evaluation of the remaining Group I and Group II piping projected that the number of thermal cycles would be substantially less the 7,000 cycle limit contained in USAS B31.1. Provide further clarification regarding the details of the NSSS vendor specification. Describe the basis for assuming the 7,000 cycle limit contained in USAS B31.1 satisfies the vendor specification.

RAI 4.3-6

Section 4.3.4 of the LRA contains a discussion of Generic Safety Issue (GSI) 190, "Fatigue Evaluation of Metal Components For 60-year Plant Life." GSI-190 addresses the effect of the reactor water environment on the fatigue life of metal components. The discussion in Section 4.3.4 indicates that EPRI license renewal fatigue studies have demonstrated that sufficient conservatism exists in the design transient definitions to compensate for potential reactor water environmental effects. The staff does not agree with the contention that the EPRI fatigue studies have demonstrated that sufficient conservatism exists in the design transient definitions to compensate for potential reactor water environmental effects. The staff identified several technical concerns regarding the EPRI studies. The staff technical concerns are contained in an August 6, 1999, letter to NEI. Although these concerns involved the EPRI procedure and its application to PWRs, the technical concerns regarding the application of the Argonne National Laboratory (ANL) statistical correlations and strain threshold values are also relevant to BWRs. In addition to the concerns referenced above, the staff has additional concerns regarding the applicability of the EPRI BWR studies to PBAPS. EPRI Report TR-107943, "Environmental Fatigue Evaluations of Representative BWR Components," addressed a BWR-6 plant and EPRI Report TR-110356, "Evaluation of Environmental Thermal Fatigue Effects on Selected Components in a Boiling Water Reactor Plant," used plant transient data from a newer vintage BWR-4 plant. The applicability of the EPRI fatigue studies to PBAPS has not been demonstrated. Provide the following additional information regarding resolution of the environmental fatigue issue:

(a) Indicate whether the staff comments provided in the staff 's August 6, 1999, letter to NEI, which are applicable to PBAPS, have been considered in the assessment of the environmental fatigue issue at PBAPS. Discuss how the applicable staff comments were considered in the evaluation of environmental fatigue.

(b) Discuss the applicability of the component fatigue assessments in the EPRI Reports TR-107943 and TR-110356 to components in PBAPS. The discussion should include a comparison of design transients, operating cycles and fabrication details for each component. In addressing fabrication details, compare pipe diameters and thicknesses at PBAPS with the components evaluated in the EPRI reports. This comparison should also include a comparison of the fabrication details at the tee connections. Also include a comparison of the hydrogen water chemistry used at PBAPS with the hydrogen water chemistry considered in the EPRI reports.

(c) The staff assessed the impact of reactor water environment on fatigue life at high fatigue usage locations and presented the results in NUREG/CR-6260, "Application of NUREG/CR-5999, 'Interim Fatigue Curves to Selected Nuclear Power Plant Components'," March 1995. Formulas currently acceptable to the staff for calculating the environmental correction factors for carbon and low-alloy steels are contained in NUREG/CR-6583, "Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels," and those for austenitic stainless steels are contained in NUREG/CR-5704, "Effects of LWR Coolant Environments on Fatigue Design of Austenitic Stainless Steels." Provide an assessment of the 6 locations identified in NUREG/CR-6260 for an older vintage BWR-4 considering the applicable environmental fatigue correlations provided in NUREG/CR-6583 and NUREG/CR-5704 reports for PBAPS.

RAI 4.3-7

Table 4.3.4-3 of the LRA provides projected 60-year fatigue usage factors for selected PBAPS components. Confirm that the usage factors reported for the feedwater line (RCIC Tee) are correct

4.7.1 Reactor Vessel Main Steam Nozzle Cladding Removal Corrosion Allowance

RAI 4.7.1-1

The applicant should provide the basis for concluding that there will be 0.030 inches of corrosion over the 60 years of operation. Was it based on actual corrosion data? How was the data collected? Was the data specific to Peach Bottom?