January 16, 2004

Mr. John L. Skolds, President
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SUBJECT: PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3 - REQUEST FOR ADDITIONAL INFORMATION REGARDING USE OF ALTERNATE SOURCE TERM (TAC NOS. MC0154 AND MC0155)

Dear Mr. Skolds:

By letter dated July 14, 2003, Exelon Generation Company, LLC, submitted a request for amendment that would allow the use of alternate source term methodology for Peach Bottom Atomic Power Station, Units 2 and 3. In order to continue our review of your request, the Nuclear Regulatory Commission staff requires the additional information described in the enclosure. These questions were discussed in a telephone call with Messrs. David Helker and Tom Mscisz of your staff on December 18, 2003. In that call your staff agreed to try to send a response by January 30, 2004.

Sincerely,

/RA/

George F. Wunder, Project Manager, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-277 and 50-278

Enclosure: Request for Additional Information

cc w/encl: See next page

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REQUEST FOR ADDITIONAL INFORMATION

PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3

PROPOSED USE OF ALTERNATE SOURCE TERM (AST) METHODOLOGY

1. It is proposed that pages B 3.8.40, B 3.8.74, and B 3.8-94 of the Technical Specifications (TS) bases be changed to indicate that ac and dc electrical power are only required to mitigate fuel handling accidents involving recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours). Pages B 3.8.40, B 3.8.74, and B 3.8-94, however, also indicate that ac and dc are required to ensure (a) the facility can be maintained in the shutdown or refueling condition for extended periods, (b) sufficient instrumentation and control capability is available for monitoring and maintaining the unit status, and (c) adequate power is provided to mitigate events postulated during shutdown, such as an inadvertent draindown of the vessel. Please clarify your requirements for ac and dc power.

2. TSs 3.8.2, 3.8.5, and 3.8.8 (which are currently applicable in Modes 4 and 5 and during movement of irradiated fuel assemblies in the secondary containment) require, in part, immediate suspension of movement of irradiated fuel in secondary containment when both offsite preferred power sources, redundant safety-related electric onsite power sources, or redundant safety-related distribution systems are no longer operable. The proposed TSs relax these TS requirements such that TSs 3.8.2, 3.8.5, and 3.8.8 will be applicable when in Modes 4 and 5 and during movement of recently irradiated fuel. The proposed change thus allows, without TS restrictions, the movement of irradiated fuel assemblies that have decayed at least 24 hours when there is no offsite power, when there is no onsite power, or when there is no ac and dc electric power through the electric distribution system to safety system loads.

    a. The application for amendment indicates that the AST analyses take credit for standby liquid control (SLC) system operation. The amendment application also indicates that the SLC system is safety-related, is required to be operable by TSs, and is supplied with emergency power. Justify movement of irradiated fuel assemblies that have decayed at least 24 hours without the availability of the SLC safety system credited in the AST analyses.

    b. Justify movement of irradiated fuel assemblies that have decayed at least 24 hours without the availability of safety systems such as those needed to maintain plant shutdown, for monitoring and maintaining the unit status, or to mitigate events postulated during shutdown.

3. You describe the use of the SLC system to buffer the suppression pool following a loss-of-coolant accident (LOCA) involving significant fission product release. The buffering action of the sodium pentaborate from this system was analyzed to demonstrate that the suppression pool pH remains above 7 for the 30-day LOCA duration. Please provide a more detailed description of this analysis and its assumptions. This description should include the following:

    a. generation of hydrochloric acid by decomposition of the chlorine bearing cables,

    b. production of nitric acid in the post-accident radiation field, and

c. determination of the amount of sodium pentaborate required for maintaining the suppression pH below 7 (also listing the input and output to the computer code used in the analysis.

4. A review of the ARCON96 meteorological data input files for both Tower 1A and Tower 2 reveals that the 1984 wind speed and direction data are reported to the nearest mph and nearest 5 degrees, respectively, whereas the wind speed and wind direction data for the remaining period (1985-1988) are reported to the nearest 10^{th} mph and nearest degree, respectively. Please explain the data recording and processing procedures that resulted in reduced precision of the reported 1984 data as compared to the 1985-1988 data.

5. A review of the ARCON96 meteorological input files for Tower 1A also shows an unusually high occurrence of low wind speeds (less than 0.5 m/sec) during 1984 as compared to 1985-1988:

Tower 1A Wind Speed Frequency Distributions

Tower Level	Period of Record	Wind Speed Range (m/sec)							
		<0.5	0.5-1.0	1.0-1.5	1.5-2.0	2.0-3.0	3.0-5.0	5.0-10.0	>10.0
34-ft	1984	21.7%	19.4%	14.0%	13.9%	14.0%	15.1%	1.9%	0.0%
	1985-1988	1.8%	20.6%	20.8%	17.9%	20.5%	15.2%	3.2%	0.0%
92-ft	1984	9.0%	16.7%	11.7%	13.9%	22.0%	21.7%	5.1%	0.0%
	1985-1988	1.5%	16.0%	15.4%	13.2%	24.3%	22.5%	7.2%	0.0%

Please explain what might have caused these differences in reported wind speed frequency distributions between the 1984 data set and the 1985-1988 data set.

6. The LOCA analysis assumes that control room isolation and the main control room emergency ventilation system have been initiated by the start of gap release. During the isolation mode, unfiltered inleakage into the control room is assumed to be 1,600 cfm. This inleakage of unfiltered air, which can occur through doorways, envelope penetrations, and leakage in ventilation system components, was modeled using the control room intake χ/Q values. Verify that there are no other potential unfiltered inleakage pathways that could result in χ/Q values that are higher than the control room intake χ/Q values.

7. Explain the basis for the 131.4 meter release height used in the PAVAN computer runs for main stack releases to the control room.

8. Both Regulatory Guides (RG) 1.145 (Section 5.3) and 1.194 (Section 2) imply that the period with the most adverse release of radioactive materials to the environment should be assumed to occur coincident with the period of most unfavorable atmospheric dispersion. For the main stack releases, the highest control room χ/Q values are associated with 0-2 hour flow reversal conditions and the highest offsite χ/Q values are associated with the 0-0.5 hour fumigation conditions. Please describe how these highest χ/Q values were used coincident with the most limiting portion of

the release to the environment to estimate control room and offsite doses.

9. Explain in more detail the methodology used to model steam cloud transport for the main steamline break (MSLB) accident. Please also provide the resulting control room and offsite χ/Q values.

10. The proposed change to the Updated Final Safety Analysis Report (UFSAR) (Section 8 of the License Amendment Request (LAR)) states that the temperature profile presented in UFSAR Figure 14.6.12A includes a 2σ adder for decay heat. The figure is identified in the UFSAR as Revision 15 dated April 1998. During an amendment review in 2000, it was stated that the figure did not include this adder and additional information was provided to the staff to justify adequate conservatism in the minimum containment pressure available (MCPA) calculation without the adder at that time (Letter from B. C. Buckley, Sr., Nuclear Regulatory Commission (NRC), to J. A. Hutton, PECO Energy Company, August 14, 2000, “Peach Bottom Atomic Power Station, Units 2 and 3 - Issuance of Amendment Regarding Crediting of Containment Overpressure for Net Positive Suction Head Calculations For Emergency Core Cooling Pumps (TAC Nos. MA6291and MA6292).”). Explain the discrepancy in these two statements. Has the analysis and figure been updated to include the adder and approved for use? If the figure does include this adder then why was this not identified during the amendment review in 2000?

11. The proposed revised UFSAR text identifies a change in methodology regarding how the containment leakage is addressed in the MCPA analysis.

    a. Provide the MCPA and containment overpressure license (COPL) calculation for the NRC staff’s review.

    b. How is it different from the previously reviewed method described in PECO Energy Company’s Calculation PM-1013, “Minimum Containment Pressure Calculation,” Revision 3, February 2000?

    c. How are the main steam isolation valve (MSIV) and airlock leakages included in the calculation?

    d. How are the leakages conservatively varied with the containment pressure assuming turbulent flow?

12. Previously, containment leakage was assumed to be constant at L_a=0.5%/day throughout the event. The containment leakage has been increased to L_a=0.7%/day for the first 24 hours, based on the proposed change to TS 5.5.12, for a peak post-accident containment pressure of 49.1 psig. This leakage is then reduced to 0.56xL_a=0.392%/day from 24 to 38 hours and then reduced to 0.50xL_a=0.350%/day, for 38 to 720 hours. In addition, MSIV leakage of 174 scfh is included (based on the proposed change to TS 3.6.1.3) in the MCPA calculation, with leakage measured at a test pressure of 25 psig. After 24 hours, the MSIV leak rate is reduced to 77.2%, then to 65.4% at 48 hours, to 59.0% at 72 hours, to 55.5% at 96 hours, and finally to 50% at 157 hours for the remainder of the event. Leakage from the personal airlock of 9,000 sccm, for a peak post-accident containment pressure of 49.1 psig, is also included in the proposed change to the MCPA calculation.

a. How are the leakages conservatively varied with the containment pressure assuming turbulent flow?

b. How does this evaluation differ from the MCPA and COPL calculation in question 11 above, which is only carried out to 12.5 hours?

c. Identify the TS which controls the allowable airlock leakage rate.

13. During the previous amendment review (Hutton, J. A., PECO Energy Company, to NRC, "Peach Bottom Atomic Power Station, Units 2 and 3 Response to May 10, 2000, Telephone Questions Regarding PECO Energy License Amendment Request Related to Generic Letter 97-04," June 29, 2000) it was stated that the margin between the MCPA and COPL was set at 1 foot (0.42 psid). The proposed amendment would decrease this margin to about 0.28 psid.

    a. Provide a justification for reducing this agreed to margin.

    b. Provide a comparison of the COPL value to the COPR (containment overpressure required) value for the residual heat removal (RHR) and core spray pumps for the most limiting event(s), including the margin to the COPL value before and after the proposed change to the MCPA/COPL calculation.

    While not directly related to the MCPA calculation, justification for the inclusion of the suppression chamber air space in the mixing of the radioactive release needs to be provided.

14. In addressing RG 1.183, Appendix A, LOCA Item 6.1, it is stated in Table B that the radioactive release is mixed with the suppression chamber air space "based on expected steam flow from the drywell to the suppression chamber, even after the initial blowdown."

    a. Is this based on the results of thermal-hydraulic analyses performed for the duration of the release? If so, provide a summary of the analyses for staff review, or

    b. Provide justification for this assumption for the duration of the release.

15. What design-basis parameters, assumptions or methodologies (other than those provided in the July 14, 2003, submittal) were changed in the radiological design-basis accident analyses as a result of the proposed change? If there are many changes it would be helpful to compare and contrast them in a table. Also, please provide a justification for any changes.

16. Based upon a preliminary review of the proposed amendment the reviewer is unable to match the calculated doses for the accident analyses. It would be helpful if the licensee would provide their design-basis accident calculations. If the calculations are provided, answers to questions provided in this request for additional information (RAI) may reference the calculation.

17. Appendix B to Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, establishes quality assurance requirements for the design, construction, and operation

of those structure, system, and components (SSCs) that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public. Appendix B, Criterion III, “Design Control,” requires that design control measures be provided for verifying or checking the adequacy of a design. Appendix B, Criterion XVI, “Corrective Action,” requires measures to be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, defective material and equipment, and nonconformances are promptly identified and corrected. GL 2003-01, “Control Room Habitability,” addresses current issues with respect to previously assumed values of unfiltered inleakage. Generally, these issues can only be resolved by inleakage testing. In light of your Appendix B requirements and GL 2003-01, provide sufficient justification to explain why the value assumed for your control room’s unfiltered inleakage is appropriate for this proposed license amendment. Provide details regarding your control room, design, maintenance and assessments to justify the use of this number.

18. Regarding the conformance to Section 3.1 of RG 1.183, the NRC staff would like Exelon Nuclear to provide additional details regarding the source term utilized for each accident. Provide the calculated source term values (nuclide and Ci/MWt).

19. Questions regarding the use of the SLC are currently being developed and will be provide in a future RAI.

20. On Page 15 of Attachment 1 of the submittal, the second paragraph states that Exelon has used the Brockmann-Bixler model for main steamline deposition. The discussion and the data in Table 5 are insufficient to support an NRC staff confirmation. Please provide the following information.

    a. A single-line sketch of the four main steamlines and the isolation valves. Annotate this sketch to identify each of the control volumes assumed by Exelon in the deposition model.

    b. A tabulation of all of the parameters input into the Brockmann-Bixler model for each control volume shown in the sketch (and time step) for which Exelon is crediting deposition. This includes:

        - Flow rate
        - Gas pressure
        - Gas temperature
        - Volume
        - Inner surface area
        - Total pipe bend angle

    c. For each of the bulleted parameters in question 20.b., provide a brief derivation and an explanation of why that assumption is adequately conservative for a design-basis calculation. Address changes in parameters over time, e.g., plant cooldown.

    d. Clarify if your analysis addresses a single failure of one of the MSIVs. Such a failure could change the control volume parameters that are input to the deposition model. Previous implementations of main steam deposition have been found acceptable only if the licensee had modeled a limiting single failure. Please explain why Exelon feels that such a limiting failure need not be considered if it is not considered.

e. Since the crediting of main steamline deposition effectively establishes the main steam piping as a fission product mitigation system, the staff expects the piping to meet the requirements of an ESF system, including seismic and single failure considerations. Please confirm that the main steam piping and isolation valves that establish the control volumes for the modeling of deposition were designed and constructed to maintain integrity in the event of the safe shutdown basis earthquake for Peach Bottom. If the design basis for the piping and components does not include integrity during earthquakes, please provide an explanation of how the Peach Bottom design satisfies the prerequisites of the staff-approved NEDC-31858P-A, “BWROG Report for Increasing MSIV Leakage Rate Limits and Elimination of Leakage Control Systems.” If piping systems and components at Peach Bottom were previously found by the staff to be seismically rugged using the methodology of this BWROG report, please provide a specific reference to the staff’s approval.

21. On page 53 of Attachment 1 of your submittal, you state that your submittal is in compliance with paragraph 6.3 of Appendix A to RG 1.183, and reference the RADTRAD Brockman-Bixler approach apparently as establishing that conformance. However, paragraph 6.3 of RG 1.183 states that the model should be based on well-mixed volumes, but other models such as slug flow may be used if justified. The Brockman-Bixler model is a slug-flow model. This paragraph did not endorse RADTRAD as an acceptable approach. RG 1.183 states that main steamline deposition will be considered on a case-by-case basis.

    The staff documented its evaluation of the first application of main steamline deposition credit in an AST in Appendix A of NRC staff report, AEB-98-03, "Assessment of the Radiological Consequences for the Perry Pilot Plant Application using the Revised (NUREG-1465) Source Term." The methodology of this report, which can be found online in ADAMS at ML011230531, was used by at least two additional licensees.

    Generally, when the staff has accepted an application of slug flow, the licensee has (1) committed to maintaining a seismically rugged drain path from the 3rd MSIV to and through the condenser, (2) did not assume deposition in piping upstream of the inboard MSIV, (3) assumed a single failure of the one of the inboard MSIVs, (4) did not credit a delay time in the onset of the release, and (5) assumed a constant pressure and temperature in the steamline over 30 days. The added conservatism from the above assumptions provided additional margins to compensate for differences in conservatism in slug flow and well-mixed assumptions. Please provide a justification for your proposed modeling approach or re-perform the analyses.

22. Page 13 of Attachment 1 of your submittal provides text that states “...an initial 12 hours transport delay is determined.” The text suggests that the steamline volume and MSIV leak rate are used to establish this delay. This implies that a delay to fill the steamline is being taken:

    a. Your submittal does not identify this as an alternative to the guidance in RG 1.183. Please explain how this holdup is modeled in the LOCA analysis. Is this modeled as a delay in the onset of the release?

b. Please explain why this delay assumption is consistent with the assumption of slug flow (Item 6.3, Page 53 of Attachment 1).

23. Based on information provided in your submittal, you have assumed an MSIV leakage rate of 0.62 cfm for the 100 scfh lines, and 0.31 cfm for the 50 scfh line, prior to 24 hours post-accident and reduced values after 24 hours. The staff believes that these values are understated. When the proposed MSIV leakage, in scfh, at test conditions (typically 70 degrees and 25 psig) are scaled to peak containment pressure and temperature (typically 40-50 psig and about 250-350 degrees) the TS leakage past the inboard MSIV has been shown to be 1.3-1.6 cfm, at least double the value you have assumed. However, the temperature of the fluid in the steamlines is based on the steam piping temperatures, typically 500-600 degrees. At the steam piping conditions, the flow in scfm is even higher, typically 4-8 scfm. Please explain the basis of the values you used and why these values are adequately conservative since the effectiveness of deposition decreases with increasing flow.

24. Paragraph 5.5 of Appendix A of RG 1.183 states that the amount of iodine that becomes airborne should be assumed to be 10%, unless a smaller value can be justified on the actual sump pH history and area ventilation rates. In Figure 2 of Attachment 1 of the amendment request, the ECCS leakage flash fraction is set at 1.41%. On page 52 of Attachment 1 it is stated that the partition factor is based upon ORNL-TM-2412. Please explain why ORNL-TM-2412 is an acceptable alternative to the guidance in RG 1.183. Is this value in your current licensing basis or is this a new value? If the value is new please provide the details used to calculate this value. Please provide the pH history, and area ventilation rates in the areas of ECCS leakage.

25. Page 40 of Attachment 1, Table A, contains a comparison of the Peach Bottom analysis to RG 1.183, Section 4.2.1. The comments column of this table states “SGTS and MCREV filters are well away and/or shielded from the Control Room and have not historically been considered a source for operator doses. AST assessments would reduce filter loadings because of the credited natural deposition in containment. Therefore, historical conclusions continue to apply.” In light of the many changes proposed (no credit for the standby gas treatment system (SGTS) filters, a new allowable MSIV leakage value, and consideration for plateout in containment) provide a more quantitative assessment justifying why the historical conclusions continue to apply.

26. Section 12.3.3, “Design Considerations,” of the UFSAR states “The main control room, the Technical Support Center (TSC), and the Emergency Operation Facility (EOF) design is based on the airborne fission product inventory in the reactor building following the design-basis LOCA in Unit 2 or 3, using a TID-14844 source term. Shielding and ventilation air treatment are provided such that operators occupying the control room, the TSC, and the EOF and traveling to and from the control room across the site will receive an exposure of less than 5 Rem whole body or its equivalent over the course of the accident.”

Page 42 of Attachment 1 states “The Technical Support Center at PBAPS is in the Unit 1 Control Room. A review of the current TID-14844-based analysis indicates that it is unnecessary to reanalyze doses therein to assure accessability. For other areas requiring plant personnel access, a qualitative assessment of the regulatory positions on source terms indicates that, with no new operator actions required, radiation exposures are bounded by those previously analyzed."

Please provide more details regarding these assessments. Justify the conclusions reached by these qualitative assessments.

27. Page 49 of Attachment 1, Table B, contains a comparison of the Peach Bottom analysis to Section 4.5 of RG 1.183. The comment column of this table states "However, based on revised containment pressure analysis, the revised TS MSIV leakage is limited to 174 scfh." Proposed insert A (for SR 3.6.1.3.14 on TS page B 3.6-29) states that the total leakage through all four main steamlines must be less than 250 scfh. Please explain this apparent inconsistency.

28. Page 52 of Attachment 1, Table B, contains a comparison of the Peach Bottom analysis to Section 6.1 of RG 1.183. The PBAPS analysis column of this table states that it conforms with RG 1.183, but this RG does not endorse mixing between the drywell and the suppression chamber air volume to determine the source term for the MSIV leakage. The assumption that the radioactive release is assumed to instantaneously mix between these two volumes appears to be inconsistent with the timing of the AST.

    a. Is this based on the results of thermal-hydraulic analyses performed for the duration of the release? If so, provide a summary of the analyses for staff review, or

    b. Provide justification for this assumption for the duration of the release.

29. From the Peach Bottom UFSAR, Table 5.2.1, Rev. 17, the minimum drywell and suppression pool free volumes are 159,000 and 127,700 cubic feet, respectively. The minimum total containment free volume is therefore, 286,700 cubic feet. Justify the use of 293,900 cubic feet provided in Table 3, on page 27 of Attachment 1. Why is the more conservative UFSAR value not valid for the LOCA analysis?

30. On Page 15 of Attachment 1 to your submittal, your first paragraph states that Exelon has used the Powers model for main steamline deposition. The discussion and the data are insufficient to support staff confirmation. Please provide the values used to input into the RADTRAD model and justify these values. Confirm that the statement "at the 10% probability level" corresponds to the 10th percentile decontamination factors or "lower bound" as discussed in NUREG/CR-6189.

31. Confirm that the control room and SGTS flow rates assumed in the accident analysis are conservative. For example, the MCREV system flow rates in the TSs appear to allow flow rates from 2700 to 3300 cfm. The value used for the LOCA is 3000 cfm. Does the assumption of 3000 cfm provide the most limiting control room dose for the LOCA?

    Confirm that the intake flows assumed for the fuel handling accident provides the limiting control room doses. This evaluation should include other control room intake flow rates if they are allowed by operating procedures. For example, operating procedures may allow normal intake flows (20,600 cfm) for the first 2 hours and then change to emergency flow (3000 cfm). If the control room filters are credited this probably would not lead to a limiting operator dose. If the filters are not credited, this scenario would provide a more limiting dose than assuming the flow remained at 20,600 cfm.

32. Page 55 of Attachment 1, Table C, contains a comparison of the Peach Bottom analysis to Section 2.0 of RG 1.183. The PBAPS analysis column of this table states that it conforms with RG 1.183, but this RG does not find the use of a Decontamination Factor (DF) of 200 acceptable for less than 23 feet of water covering a damaged fuel assembly. Please provide the DF used for 21.5 feet of water and the parameters, methodology and justification used to calculate this value.

33. More detail regarding the main steamline break (MSLB) accident is needed. Please provide the reactor coolant system (RCS) activity used for the MSLB analysis and the parameters use to determine this activity.

    The second bulleted item on page 15 of Attachment 1, states that the activity in the steam cloud is based on the total mass of water released from the break. Confirm that the total activity released for this accident is the RCS specific activity times the break discharge mass (190,920 lbm). If this is not the methodology used, please provide more detail regarding the model utilized. Also, provide the input parameters used to calculate and justify the fraction of liquid water contained in the steam (2%) and the flashing fraction of liquid water released (40%).

34. In Attachment 1, page 30, Table 8, a value of 0.77% damaged fuel with melt is provided for the control rod drop accident (CRDA). The value typically used for fuel melt with General Electric 14 fuel is 1% for the CRDA. Please confirm this value of 0.77%.

35. Please provide the atmospheric dispersion factors used for the CRDA calculations for the exclusion area boundary and low-population zone doses. Are these the values provided in Table 14b on page 32?