September 5, 2007

Mr. Britt T. McKinney
Sr. Vice President and Chief Nuclear Officer
PPL Susquehanna, LLC
769 Salem Blvd., NUCSB3
Berwick, PA 18603-0467

SUBJECT: REQUESTS FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2, LICENSE RENEWAL APPLICATION

Dear Mr. McKinney:

By letter dated September 13, 2006, PPL Susquehanna, LLC submitted an application pursuant to Title 10 of the *Code of Federal Regulations*, Part 54, to renew the operating licenses for Susquehanna Steam Electric Station, Units 1 and 2, for review by the U.S. Nuclear Regulatory Commission (NRC or the staff). The staff is reviewing the information contained in the license renewal application and has identified, in the enclosure, areas where additional information is needed to complete the review.

These requests for additional information were discussed with Duane Filchner, and a mutually agreeable date for the response is within 30 days from the date of this letter. If you have any questions, please contact me at 301-415-4029 or via e-mail ehg2@nrc.gov.

Sincerely,

/RA/

Evelyn Gettys, Project Manager
License Renewal Branch A
Division of License Renewal
Office of Nuclear Reactor Regulation

Docket Nos. 50-387 and 50-388

Enclosure:
Requests for Additional Information

cc w/encl: See next page

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Letter to B. McKinney, from E. Gettys dated September 5, 2007

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Susquehanna Steam Electric Station,
Units 1 and 2

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SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2
LICENSE RENEWAL APPLICATION
REQUESTS FOR ADDITIONAL INFORMATION (RAIs)

RAI 4.2.3-1

License renewal application (LRA) Section 4.2.3 states, "It may be noted that ART [adjusted reference temperature] values are well below the 200 °F [as] suggested in Section 3 of Regulatory Guide 1.99 and are, thus, acceptable for the period of extended operation." Section 3 of Regulatory Guide 1.99, Revision 2 is for new plants and is not applicable to Susquehanna Units. Unlike the pressurized-thermal-shock (PTS) screening criteria used for evaluating the pressurized water reactor RPV material reference temperatures at the end of license fluence, there is no criteria for evaluating the RPV ARTs. The significance of ARTs is considered in the pressure and temperature (P-T) limit evaluation. Please revise LRA Section 4.2.3 and the associated updated final safety analysis renewal review (UFSAR) Supplement summary description so that Section 3 of Regulatory Guide 1.99 is not referenced.

RAI 4.2.4-1

LRA Section 4.2.4 states that calculations were performed to develop P-T limits for both units for the extended period of operation, using the 54 effective full-power year (EFPY) fluence values discussed in LRA Section 4.2.1. However, since the applicant did not include the revised P-T limits valid for 54 EFPY in the LRA for the staff review, it is inappropriate to state in LRA Section 4.2.4 that, "The 54 EFPY P-T curves for Units 1 and 2 demonstrate that there is sufficient operating margin for hydrostatic tests, heatup, cooldown, and core critical operation to the end of the period of extended operation." Please revise LRA Section 4.2.4 and its associated UFSAR Supplement summary description by taking this statement out. In addition, it is suggested to mention in the UFSAR Supplement summary description that Susquehanna Steam Electric Station (SSES) will submit the 54 EFPY P-T limits for NRC review and approval at the appropriate time to comply with Title 10 of the *Code of Federal Regulations*, Part 50, Appendix G.

RAI 4.2.5-1

In the July 28, 1998, safety evaluation report on Boiling Water Reactor Vessel and Internals Project (BWRVIP)-05, the NRC staff concluded that examination of the RPV circumferential shell welds would need to be performed if the corresponding volumetric examinations of the RPV axial shell welds revealed the presence of an age-related degradation mechanism. Confirm whether or not previous volumetric examinations of the RPV axial shell welds have shown any indication of cracking or other age-related degradation mechanisms in the welds. Please also confirm whether there are any flaw evaluations performed to date on RPV flaws as a result of previous volumetric examinations of the RPVs, and, if flaw evaluations exist, why they are not considered as time-limited aging analysis (TLAAs).

RAI 4.2.7-1

LRA Section 4.2.7 states that the SSES RPVs are bounded by the generic analysis that is discussed in NEDO-10029, “An Analytical Study on Brittle Fracture of GE-BWR Vessels Subject to the Design Basis Accident,” because the reference temperature nil ductility (RT_{NDT}) shifts at 54 EFPY (35.9 °F for Unit 1 and 38.4 °F for Unit 2) are less than the RT_{NDT} shift of 50 °F used in the generic analysis. Please confirm whether the current licensing basis relies on the NEDO-10029 conclusions in addressing the reflood thermal shock issue and whether NEDO-10029 has been reviewed and approved by the NRC. If NEDO-10029 has been approved by the staff and utilized within the current SSES licensing basis, summarize the technical basis for determining the adequacy of this TLAA based solely on RT_{NDT} shift (as opposed to the ART of the limiting material). If NEDO-10029 was not approved, please provide the report for staff review to determine the acceptance of the methodology and results for use in the extended period of operation. Further, please demonstrate that the driving force based on the plant-specific design basis accident and SSES RPV geometry is bounded by the generic analysis.

RAI 4.2.7-2

All recent LRAs for plants with BWRs (e.g., Monticello, Brunswick, and Browns Ferry units) addressed the issue of reflood thermal shock analysis of RPV core shroud. Please address this TLAA, or explain why this topic is not a TLAA for SSES, Units 1 and 2.

RAI 4.7.3-1

LRA Section 4.7.3 states that, based on BWRVIP-25, “BWR Core Plate Inspection and Flaw Evaluation Guidelines,” the core plate hold-down bolts will have at least 81-percent preload remaining at 54 EFPY and, based on the GE extended power uprate (EPU) analyses of the core plate hold-down bolts, the preload at the end of 60 years would be adequate to prevent lateral motion of the core plate for the period of extended operation. This conclusion is not supported by any SSES plant-specific evaluation. Please provide the following additional information:

(1) Demonstrate the applicability of the BWRVIP-25 loss of preload analysis to the SSES Units. Identify the temperature of the bolts during the normal operation and the projected bolt neutron fluence at the end of the period of extended operation for the SSES Units. Provide a plant-specific evaluation demonstrating that the loss of preload due to stress relaxation for the SSES RPV core plate hold-down bolts is bounded by the value of 19 percent from Appendix B of BWRVIP-25.

(2) Perform a plant-specific core plate hold-down bolt analysis using the BWRVIP-25 Appendix A methodology, demonstrating that the axial and bending stresses for the mean and highest loaded hold-down bolts will not exceed the American Society of Mechanical Engineers Boiler and Pressure Vessel Code Section III allowable stresses for P_m (primary membrane) and $P_m + P_b$ (primary membrane plus bending) as a result of a plant-specific reduction in the bolt preload at the end of the extended period of operation. State clearly the assumptions on which the plant-specific analysis was based.

(3) Provide sufficient information regarding the “GE EPU analyses” on the core plate hold-down bolts so that the staff can determine whether the SSES hold-down bolts are adequate to prevent lateral motion of the core plate for the period of extended operation.

RAI 4.7.3-2

LRA Appendix C discussed the applicant’s response to BWRVIP report application action items. The BWRVIP reports addressing the TLAA regarding irradiation assisted stress corrosion cracking (IASCC) in austenitic stainless steel RPV internals are BWRVIP-25, BWRVIP-26-A, “BWR Top Guide Inspection and Flaw Evaluation Guidelines,” BWRVIP-76, “BWR Core Shroud Inspection and Flaw Evaluation Guidelines,” and BWRVIP-47-A, “BWR Lower Plenum Inspection and Flaw Evaluation Guidelines.” Although managing TLAA using aging management programs or other measures is stated in Appendix C in your response to BWRVIP report application action items, IASCC in austenitic stainless steel RPV internals should be discussed in the TLAA section to build the connection between LRA Section 4.0 and Appendix C. Further, the fact that BWRVIP-47 is under the NRC staff review doesn’t imply that you don’t need to address it. In addition to your plant-specific evaluation using BWRVIP-76, you need to make a commitment to follow all BWRVIP-76 requirements and limitations, as adjusted by the NRC staff’s safety evaluation on the BWRVIP-76 report.