August 17, 2006

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

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION (RAI) - SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2 (SSES 1 AND 2) - APPLICATION FOR LICENSE AMENDMENT AND RELATED TECHNICAL SPECIFICATION CHANGES TO IMPLEMENT FULL-SCOPE ALTERNATIVE SOURCE TERM (TAC NOS. MC8730 AND MC8731)

Dear Mr. McKinney:

In reviewing your letter dated October 13, 2005, concerning the request for an amendment to the licensing basis for SSES 1 and 2 that supports a full implementation application of an Alternative Source Term methodology, the Nuclear Regulatory Commission staff has determined that additional information contained in the enclosure to this letter is needed to complete its review. These questions were discussed with your staff during a teleconference on August 15, 2006. As agreed to by your staff, we request you respond within 30 days of the date of this letter.

If you have any questions, please contact me at 301-415-1030.

Sincerely,

/RA/

Richard V. Guzman, Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-387 and 50-388

Enclosure:
RAI

cc w/encl: See next page

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DATE	8/17/06	8/17/06	8/17/06	8/17/06

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REQUEST FOR ADDITIONAL INFORMATION

RELATING TO THE

APPLICATION FOR LICENSE AMENDMENT FOR FULL IMPLEMENTATION OF AN

ALTERNATIVE SOURCE TERM (AST) METHODOLOGY

SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2 (SSES 1 AND 2)

PPL SUSQUEHANNA, LLC

DOCKET NOS. 50-387 AND 50-388

The Nuclear Regulatory Commission (NRC) staff is reviewing the request from PPL Susquehanna, LLC (PPL, the licensee) to support the full implementation application of an AST methodology for SSES 1 and 2. The NRC staff has determined that additional information requested below will be needed to complete its review.

1. Section 3.7 of Regulatory Guide (RG) 1.183 states that for boiling-water reactors (BWRs), primary containment leakage may be reduced after the first 24 hours, if supported by plant configuration and analyses, to a value not less than 50% of the technical specification leak rate. Please provide the requisite site-specific information to support the assumption of a 50% reduction in primary containment leakage after 24 hours. Section 3.9 of EC-RADN-1125, Rev. 0 provides some information; however, the bottom of sheet 8 (as provided in the submittal package and partially reproduced below) is blurred and difficult to read.

    > Based on the significant reduction of the calculated internal pressure of primary containment at 24 --------------------- Per PPL Drawing C-206-130, Sheet 1, Primary Containment Zones (Reference 29), the Drywell LOCA peak pressure <24 hours equals 41.3 psig and at 24 hours equals approximately 15.3 psig.

2. PPL's modeling of the primary containment leakage to the secondary containment pathway appears to treat the drywell and wetwell as a single, well-mixed volume of 388,190 feet-cubed (ft^3), from the start of the event. With the AST timing, as described in RG 1.183, Table 4, the initial blow-down of the reactor coolant system would be expected to have occurred prior to the onset of the early in-vessel release phase. Therefore, at the onset of the early in-vessel release phase (T = 30 minutes), the driving force for mixing between the two volumes will be less certain. Because of this uncertainty, the NRC staff has deterministically assumed that complete mixing does not occur until 2 hours post-loss-of-coolant accident (LOCA), when core re-flood is projected and necessary to end the early in-vessel release phase (as PPL assumed for the main steam isolation valve leakage and the secondary containment bypass pathways).

Enclosure

Please provide further justification describing why PPL's proposed approach is adequately conservative for the primary containment leakage pathway and justify the apparent inconsistency in the application of credit for drywell/wetwell mixing in the LOCA analysis.

3. Section 3.1 of RG 1.183 states that the suppression pool free air volume may be included provided there is a mechanism to ensure mixing between the drywell to the wetwell. Please provide a description of the site-specific mechanisms available to ensure mixing between the drywell and the wetwell.

4. In the main steam line break (MSLB) analysis, the current licensing basis mass releases are increased by 20% to provide additional margin for extended power uprate (EPU) conditions. PPL asserts that evaluations of steam line break masses for other EPU plants determined that the increases in mass releases were small compared to the pre-uprate MSLB masses while at power. Please provide additional information to show that a 20% increase in MSLB mass release would be bounding for a proposed future increase in rated thermal power of approximately 12%.

5. Table 4.5-1 in PPL's application describes the control rod drop accident (CRDA) assumptions and indicates that PPL assumed 100% of the noble gases and 50% of the iodines in the melted regions of the fuel is released to the reactor coolant system. The amount of solids released from the melted regions is not specified. However, Section 4.5.8 states that as a calculation conservatism, solids released from the melted fuel were included in the analysis. Table 4 of EC-RADN-1127 indicates that activity from solids as a result of the fuel melt is included in the analysis, and references RG 1.183, Table 1. Please clarify the assumptions regarding the inclusion of solids in the fuel melt portion of the CRDA source term. Also, please note that it appears that the fractions for Lanthanides and Ceriums in Table 4 of EC-RADN-1127 may have been inadvertently reversed; however, it is not expected that this would have a significant effect on the results of the calculation.

6. In Section 4.6.3 of the AST application describing the fuel-handling accident/equipment handling accident (FHA/EHA) analysis, it states that, "For this event, the CRHE [control room habitability envelope] automatically isolates and enters the emergency mode in sequence with the SGTS [standby gas treatment system] prior to commencement of the release of activity to environment." In Section 3.17 of EC-RADN-1126, "CRHE and Off-Site FHA/EHA Doses - AST", it states:

> Per References 22 (EC-RADN-0531) and 23 (EC-RADN-0319), the activity transport from the pool to the environment is via the SGTS filters. Reference 22 provides a conservative analysis using realistic assumptions and parameters for a fuel handling accident that demonstrates that the Refueling Floor High Exhaust Duct Radiation Monitors, Refueling Floor Wall Exhaust Duct Radiation Monitors and the Railroad Access Shaft Exhaust Duct Radiation Monitor will sense the event and provide the required signals to the SGTS. Reference 23 provides an analysis that demonstrates that the isolation damper closure

time is less than the air travel time. Therefore, the isolation damper will close prior to the activity reaching the damper.

From the cited references, is an estimate available of the CRHE and off-site consequences for an FHA that results in an activity release just below the threshold needed to activate the SGTS and the Control Room Emergency Outside Air Supply System as credited in the analysis?

7. In the following sections of the submittal the terms, “control room” and “control room operator,” may have been used in a general sense that could lead to confusion as to the areas and the personnel requiring access limitations. RG 1.196 defines the terms Control Room and Control Room Envelope (CRE) and makes a distinction between the two regarding the requirements for occupancy. The NRC staff is reluctant to make any exceptions to General Design Criterion 19 dose limits for the control room (and the technical support center). However, other areas within the CRE that do not support critical safety functions may be evaluated on a case-by-case basis to assess the appropriateness of limited access controls.

   Please provide a plan view of the affected areas to ensure that the proposed access controls for designated portions within the CRHE, will not impact the control room proper or the technical support center.

Susquehanna Steam Electric Station, Unit Nos. 1 and 2

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