July 26, 2001

Mr. Robert G. Byram
Senior Vice President
and Chief Nuclear Officer
PPL Susquehanna, LLC
2 North Ninth Street
Allentown, PA 18101

SUBJECT: SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2 - REQUEST FOR ADDITIONAL INFORMATION REGARDING SUPPLEMENTAL RESPONSE TO GENERIC LETTER 96-06 (TAC NOS. M96875 AND M96876)

Dear Mr. Byram:

By letter dated August 3, 1999, you submitted a supplemental response to Nuclear Regulatory Commission (NRC) Generic Letter (GL) 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions." The supplement provided a risk-based assessment of the potential for thermally-induced over-pressurization of containment piping penetrations. In order to complete our review of your response to GL 96-06, the NRC staff requires responses to the questions in the enclosed request for additional information. The enclosed questions have been discussed with members of your staff and a mutually agreed date for your response of August 31, 2001, has been established.

Sincerely,

/RA/

Robert G. Schaaf, Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-387 and 50-388

Enclosure: As stated

cc w/encl: See next page

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REQUEST FOR ADDITIONAL INFORMATION

RELATED TO GENERIC LETTER (GL) 96-06, ASSURANCE OF EQUIPMENT OPERABILITY

AND CONTAINMENT INTEGRITY DURING DESIGN-BASIS ACCIDENT CONDITIONS

PPL SUSQUEHANNA, LLC

SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2

DOCKET NOS. 50-387 AND 50-388

By letter dated August 3, 1999, the licensee submitted a supplemental response to NRC Generic Letter (GL) 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions." The supplement provided a risk-based assessment of the potential for thermally-induced over-pressurization of containment piping penetrations. In order to complete our review of the response to GL 96-06, the NRC staff requires responses to the following questions.

1. Provide further justification for P[3] - containment heating causes heating and expansion of the water trapped between the isolation valves over-pressurizing the pipe until rupture. At a minimum, please, address additional failure modes such as station blackout and human reliability.

2. Page 16 of the submittal states, "Penetration failure is a concern when a large radioactive source term is available for release in the drywell." Please quantify the probability of a large radioactive source term in the drywell at the time of penetration failure.

3. During the resolution of Generic Safety Issue 150, "Over-pressurization of Containment Penetrations," the staff estimated a value of 0.1 for the probability that the penetration fails in a manner that results in a leakage path from the containment atmosphere to the environment. Although the staff believes that 0.1 is very conservative, barring further justification from PP&L, the staff believes that a value of 0.1 is more appropriate than PPL's estimate of between 10^{-5} and 10^{-2} with a point estimate of $3x10^{-4}$. The staff's concern in supporting a less conservative value is based on PPL's application of Branch Technical Position MEB 3-1, failure to address the effects of non-uniform strain, failure to identify the more likely failure points given over-pressurization, and a lack of relevant data.

4. Section 5.0 of the submittal discusses two mitigating measures that provide protection to primary containment integrity for the over-pressurization failure mode as well as other threats. The submittal goes on to say that these measures have been implemented in the plant's Emergency Operating Procedures (EOPs) via safety evaluations per 10 CFR 50.59. Please provide a copy and reference these safety evaluations in your submittal. What controls exist to assure that these improvements to the EOPs will not be modified without the consideration of the issues raised in GL 96-06?

5. Considering the safety importance of the drywell spray valves to open as described in Section 4.2.1 of the submittal, what monitoring program will be implemented to support the modeling assumptions of the drywell spray isolation valves? Have insights from the

engineering evaluation in Section 4.2.1 been incorporated into the drywell spray isolation valves' maintenance program? Have these valves been classified as having high safety significance? If so, will possible future changes to their classification consider their role in the disposition of GL 96-06?

6. Page 16 of the submittal states that, "Estimating the probability that the containment will reach a sustained temperature sufficient to rupture requires an evaluation of ... the containment temperature for a spectrum of accidents," and, "It is assumed that penetration failure will occur if cooling to the drywell is not restored." Please provide the evaluation of containment temperature for a spectrum of accidents described above. Have you quantified the impact of drywell sprays on containment temperature? If not, what is the basis for the assumption that the penetration will not fail given restoration of cooling to the drywell?

7. For those penetrations that are susceptible to thermally-induced over-pressure, provide the maximum-calculated temperature and pressure for the piping run. Describe in detail the method used to calculate these pressure and temperature values. This should include a discussion of the heat transfer model, and the basis for the heat transfer coefficients used in the analysis. Discuss any source of uncertainty associated with the calculated pressure and temperature.

8. Provide the results of piping and valve analysis based on the criteria contained in the American Society of Mechanical Engineers Code, Section III, Appendix F. For each component, provide a summary of the maximum faulted pressure, design load combination, calculated stress for design load combination including faulted pressure, and allowable stress based on the criteria contained in Appendix F. Also, you should include a reference to the specific provisions of Appendix F used as a basis in calculating the allowable stress (e.g., F-1331, F-1430, F-1420).

Mr. Robert G. Byram
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/RA/
Robert G. Schaaf, Project Manager, Section 1
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Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-387 and 50-388

Enclosure: As stated

cc w/encl: See next page

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