July 25, 1991

Docket Nos. 50-387
and 50-388

Mr. Harold W. Keiser
Senior Vice President-Nuclear
Pennsylvania Power and Light Company
2 North Ninth Street
Allentown, Pennsylvania 18101

Dear Mr. Keiser:

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION REGARDING ENFORCEMENT ACTION 89-042 ON REACTOR VESSEL COOLDOWN RATE, SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2 (TAC NOS. 79893 AND 79894)

The staff has reviewed your February 21, 1991 response to NRC Enforcement Action 89-042 which dealt with Susquehanna Steam Electric Station, Unit 1 exceeding the Technical Specification on reactor vessel cooldown rate on January 12, 1989.

Based on this review, the staff has developed the enclosed request for additional information. Please provide the requested information within 30 days of your receipt of this letter.

If there are any questions regarding this matter, please contact me on (301) 492-1447.

This requirement affects 9 or fewer respondents and, therefore, is not subject to Office of Management and Budget Review under P.L. 96-511.

Sincerely,

/S/

James J. Raleigh, Acting Project Manager
Project Directorate I-2
Division Of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosure:
Request for Additional Information

cc w/enclosure:
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

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James J. Raleigh

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Project Directorate I-2
Division Of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosure:
Request for Additional Information

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ENCLOSURE

REQUEST FOR ADDITIONAL INFORMATION

ENFORCEMENT ACTION 89-042 ON REACTOR VESSEL COOLDOWN RATE

SUSQUEHANNA STEAM ELECTRIC STATION

DOCKET NOS. 50-387 & 50-388

By letters dated June 12, 1989 and December 21, 1989 the licensee responded to Enforcement Action 89-042 originated from the NRC Region I office. The following request for additional information was based on the staff's review of the licensee's response and associated GE report SASR 89-40 relating to the out-of-limit cooldown rate.

1. Was Tsat (saturation temperature derived from steam dome pressure) recorded during the 1/12/89 event, and, if so, was the cooldown rate based on Tsat less than $100^{o}F$ for any one hour period during the event?

2. As you have indicated in your submittal, the bottom drain line temperature is usually the lowest temperature in the vessel. However, it is the rate of change of temperature that should be considered. Is there analysis to support your conclusion that the rate of change of the bottom drain line temperature is a more conservative measure of cooldown rate than the rate of change of Tsat?

3. Section 3.2 of the GE report SASR 89-40 discusses that the temperature rate of changes of the steam dome coolant are representative of the beltline coolant due to the large percentage of core return flow in the downcomer region. Please provide further analysis to demonstrate that the rate of change of Tsat is indicative of the rate of change of the temperature of the coolant adjacent to the beltline region.

4. Provide an analysis to demonstrate that the rate of change of Tsat is satisfactory for determining the cooldown rate of the reactor vessel internals.

5. Have questions 2, 3 & 4 above been evaluated for natural

circulation conditions as well as forced circulation conditions?

Questions on the RESPONSE TO NOTICE OF VIOLATION in a letter dated June 12, 1989

6. In Response 1.b to Violation A.1, the licensee stated that "an engineering evaluation was performed to determine the effects of the out-of-limit condition." Technical Specifications 3/4.4.6 requires an evaluation to determine the structural integrity of the reactor coolant system and to justify continued operation.

   6.1 Provide the evaluation.

   6.2 Did the licensee perform an analysis in accordance with Appendix E, "Evaluation of Unanticipated Operating Events," to the ASME Code, Section XI?

   6.3 Which of the out-of-limit cooldown rates was used in the evaluation either the 137°F in the first 45 minutes, or, the 101°F by natural circulation in the first hour?

7. In Response 2.b to Violation A.1, the licensee stated that evaluation of the design bases for reactor pressure vessel fatigue and brittle fracture analyses were performed.

   7.1 The staff assumes that GE's report, SASR 89-40, is the evaluation that was mentioned in the PP&L's response? If it is not, provide the fatigue and brittle analyses.

   7.2 The staff has not found any discussion on the fatigue analysis in GE's report. The report discussed mainly about the brittle fracture analysis. Has there been a fatigue analysis performed to determine the effect of the out-of-limit cooldown rate on the reactor vessel?

Questions on the Summary Report prepared by PP&L in the letter dated December 21, 1989.

8. Second paragraph, page 6, discusses stresses considered for the feedwater nozzle analysis. It seems that the stress due to seismic event was not included in the analysis. Explain the exclusion.

9. Third paragraph, page 6, states that "normal pressure stresses in the bottom head region were conservatively adjusted by a factor of three to account for stress concentration at the CRD penetration...." Explain why a factor of three makes the analysis conservative.

Questions on GE's report, SASR 89-40.

10. Fourth paragraph, page 2-1, states that since the bottom head analysis covers a 178°F step change cooldown, temperature change of 100°F/hr can be tolerated. The fracture toughness analysis of the bottom head lacks analytical details and quantitative justification.

    10.1 The staff needs to review actual calculations and methodology.

    10.2 GE used Appendix G to the ASME Code section III to perform the fracture analysis for the CRD penetrations /bottom head. The Appendix G method assumes a plate with a postulated 1/4T flaw away from discontinuities; however, the geometry of the bottom head shell is a plate with many discontinuities (ie., CRD penetrations). Is a safety factor of three on the stress intensity factor that GE used bounding? Was a finite element analysis performed to justify the safety factor/stress intensity factor that were used?

    10.3 Provide quantitative values (input and output) and equations used in the analysis.

11. GE's report did not answer the following questions:

    11.1 How much additional stresses the bottom head was experienced due to the higher cooldown rate?

    11.2 Would the additional stresses cause degradation in the reactor vessel. Provide quantitative justification.

    11.3 GE performed a bounding analysis. Provide a comparison of stress results between the bounding analysis and the analysis of the out-of-limit cooldown rate.

    11.4 Was the structural integrity of the circumferential weld between the low shell plate and bottom head affected by the out-of-limit cooldown rate?

12. Section 4.2.2 CRD Penetration Limits

First paragraph, page 4-5, states that "Heatup/cooldown limits were calculated by increasing the safety factor in Section 4.1.2 from 1.5 to 2.0, on the assumption that the conservative factor of three on bottom head pressure stress bounds the thermal stresses occurring during heatup/cooldown." The staff has following questions:

12.1 Use of a factor of 2.0 for the stress intensity factor due to pressure for the heatup/cooldown limits and a factor of 1.5 for the hydrostatic test limits is a normal practice that follows Appendix G to the ASME Code, Section III. Therefore, it is not clear why GE made the statement that "...increasing the safety factor...from 1.5 to 2.0, on the assumption that the conservative factor of three...bounds the thermal stresses during heatup/cooldown."

12.2 Provide actual stress values to justify the statement, "A factor of three on bottom head pressure stress bounds the thermal stresses."

13. Second paragraph, page 6-3, states that the thermocouples (TCs) at the bottom head and the temperature elements (TEs) in the bottom head drain line were used to confirm safe conditions relative to Curves B and C of the heatup/cooldown limits. However, on page 6-4, the report stated that "For the purposes of determining conformance with 100°F in any one hour, bottom head shell TCs and the drain line TE should not be monitored." Explain why the temperature at the bottom head is used to confirm the pressure-temperature limits, but not to confirm the rate of heatup/cooldown. The discussion in the report was not convincing.

14. GE's response to PP&L questions in Appendix A

In GE's response to PP&L's question No.10, GE stated that the bounding analysis considered the worst transients and that heatup or cooldown could not approach the severity; therefore there is no need to monitor heatup/cooldown rates in non-beltline regions. The staff disagrees with this assessment. The staff believes that PP&L must monitor the non-beltline regions such as the closure flange area in accordance with Surveillance Requirements 4.4.6.1.4 in Technical Specifications 3/4.4.6.

15. Provide References 7-9 and 7-10.

General Questions

16. What was the cooldown rate at the beltline region of the reactor during the out-of-limit event?

17. Could this out-of-limit event occur in the future, i.e. Is this an inherent event due to the BWR system configuration?

18. How was the heatup/cooldown rate calculated? Provide the data to show your calculation.