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SUBJECT: Submits comments re util evaluation of loss of SFP cooling events at plant.

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EV 4

PDR

December 6, 1993

Mr. Joseph W. Shea, Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation
United States Nuclear Regulatory Commission
Washington, DC 20555-0001

Dear Mr. Shea:

SUBJECT: COMMENTS ON PP&L EVALUATION OF LOSS OF SPENT FUEL POOL COOLING EVENTS, SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2

We received your letters dated November 22, 1993 <Ref. 1> and November 23, 1993 <Ref. 2>. In reviewing the technical information PP&L recently supplied to the NRC, we have the following comments which we bring to your attention:

1) Response 1 in the PP&L submittal dated November 3, 1993 <Ref. 3> reported the results of assessments of Fuel Pool Cooling and Cleanup System FPCCS piping integrity under LOCA hydrodynamic loading. It should be noted that the Service Water System (SWS) piping, which supplies cooling water to the FPCCS heat exchangers, is not designed for seismic or Loss of Coolant Accident (LOCA) hydrodynamic loading. In engineering report NE-92-002 <Attachment 30 to our 10 CFR 21 report dated November 27, 1992>, PP&L concluded that the SWS piping was more vulnerable to LOCA hydrodynamic loading. If the FPCCS is to be relied upon following a LOCA, the SWS piping integrity must be assured.

2) Response 2 in the PP&L submittal dated November 3, 1993 <Ref. 3> reported the results of RHR Fuel Pool Cooling (FPC) Assist mode pre-operational testing for Units 1 and 2 and stated that RHR FPC Assist mode "has been proven by pre-operational testing and verified by calculation to be capable of providing long term cooling to the SFP." We find numerous faults with this PP&L response:

   a) PP&L states that their recent calculation <Ref. 4> shows that with the spent fuel pool (SFP) level 8" above the weir, an RHR FPC Assist mode flow rate of ≈5700 gpm can be sustained, while at the normal SFP level, only 2000 gpm can be sustained. RHR FPC Assist mode operating procedures at the time of our 10 CFR 21 report specified a flow rate of ≈5700 gpm without providing any prerequisite or caution that the SFP level had to be raised to allow such operation.

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b) According to the PP&L submittal <Ref. 3>, during the pre-operational testing performed on August 23, 1982 for Unit 1 and on July 21, 1984 for Unit 2, "personnel recognized the need to raise water level in the SFP in order to support the higher flowrate of the RHR pumps." PP&L determined in 1982 and confirmed in 1984 that SFP level had to be raised to support RHR FPC Assist mode, yet did not revise the RHR FPC Assist mode operating procedures until 1993 to reflect this requirement.

c) The pre-operational tests were performed at SFP temperatures of 58°F on Unit 1 and 76°F on Unit 2 <Ref. 3 Attachment 3>. We have not reviewed the recent PP&L calculation <Ref. 4>, but it probably assumed a SFP temperature of at most 125°F. PP&L has stated that it will take a minimum of eight (8) hours to align the RHR FPC Assist mode. For the postulated Design Bases Accident (DBA) LOCA event, the SFP temperature would initially be 125°F. Since the FPCCS is automatically load shed upon a LOCA signal, the SFP temperature in the DBA LOCA event will be greater than 125°F. Increased SFP temperature corresponds to reduced NPSH for the RHR pumps. The maximum SFP temperature which provides adequate NPSH with the SFP level 8" above the weirs should be determined and shown to bound the SFP temperature when the RHR FPC Assist mode is initiated after a postulated LOCA.

d) The manual operator actions in the reactor building to align RHR FPC Assist mode are more involved and require more time than those actions required to provide Emergency Service Water (ESW) makeup to the SFP. We note that your RAI for expected operator doses <Ref. 2> addresses this currently unanalyzed condition.

e) The RHR System has many functions following a LOCA including Low Pressure Coolant Injection, containment cooling, suppression pool cooling, and alternate shutdown cooling. To date, we are not aware of any engineering analysis which conclusively demonstrates that the RHR FPC Assist mode can be used following a DBA LOCA without adversely affecting any other required RHR function, especially if a limiting single failure is assumed.

3) The points outlined above apply to the postulated DBA LOCA event. If a concurrent Loss of Offsite Power (LOOP) is postulated (and the LOCA/LOOP is within the SSES licensing bases), then the RHR FPC Assist mode also has to be evaluated from the affect on the emergency diesel generator loading. Little capacity margin is available on the SSES diesel generators.

PP&L has claimed that the seismic event is the only event within the SSES licensing bases which must be evaluated for boiling spent fuel pool consequences. This claim is based on SSES FSAR Appendix 9A. For all other design bases events, PP&L has argued that restoration of fuel pool cooling was assumed in the SSES design. We disagree with PP&L's position on the following grounds:

1) For the seismic event postulated in FSAR Appendix 9A, the integrity of FPCCS piping on both units was assumed to fail. Both SFPs were assumed to boil and ESW makeup was provided to each boiling pool to maintain level. Please note that following this seismic event, the reactor buildings remain freely accessible with no radiation or temperature effects.

2) For the postulated DBA LOCA, the FPCCS is automatically load shed. Neither the SSES FSAR nor any design document (with the exception of the reactor building heat load calculation we prepared for PP&L which prompted EDR G20020) reviewed to date discuss the use of the non-safety related FPCCS following a DBA LOCA.

3) SSES FSAR Chapter 9 describes the RHR FPC Assist mode as providing supplemental SFP cooling during refueling outages. Neither the SSES FSAR nor any design document (with the exception of the reactor building heat load calculation we prepared for PP&L which prompted EDR G20020) reviewed to date discuss the use of the non-safety related RHR FPC Assist mode following a DBA LOCA.

4) The dependence on the FPCCS or the RHR FPC Assist mode to provide SFP cooling after a postulated DBA LOCA is therefore beyond the SSES design and licensing bases and constitutes an unreviewed safety question. The dependence on the existing FPCCS or the RHR FPC Assist mode at SSES to provide SFP cooling after a postulated DBA LOCA conflicts with numerous Federal regulations including 10 CFR 50, Appendix A, General Design Criteria 4, 44, and 61.

5) It is inconsistent (and extremely improbable) that the SSES design would assume use of FPCCS or RHR FPC Assist mode to restore SFP cooling following a DBA LOCA when use of RHR FPC Assist mode was not assumed following a seismic event. In the seismic event, the reactor buildings are freely accessible and RHR System requirements are minimal. In the DBA LOCA event, the accident unit's reactor building is virtually inaccessible, the non-accident unit's reactor building may be inaccessible, and the RHR System is required to perform numerous safety related functions.

We appreciate your attention to our concerns. Thank you again for your periodic status reports on the NRC evaluation of our 10 CFR 21 report issues. We will support your evaluation of our concerns any way we can.

Sincerely,

David A. Lochbaum

for Donald C. Prevatte

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December 6, 1993

References:

1. NRC Letter from Joseph W. Shea to David A. Lochbaum and Donald C. Prevatte, "Susquehanna Steam Electric Station, Units 1 and 2, Spent Fuel Pool Cooling Issue (TAC No. M85337)", November 22, 1993

2. NRC Letter from Joseph W. Shea to Robert G. Byram, "Request for Additional Information (RAI) Concerning Loss of Spent Fuel Pool Cooling Events, Susquehanna Steam Electric Station, Units 1 and 2 (TAC No. M85337)", November 23, 1993

3. PP&L Letter PLA-4044 from R. G. Byram to C. L. Miller, "Susquehanna Steam Electric Station, Response to 10/20/93 RAI on Spent Fuel Pool Cooling System", November 3, 1993

4. PP&L Calculation M-RHR-039 Rev. 0, May 17, 1993