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SUBJECT: Forwards comments on training packages,SGTS fusible link SE & evaluation of equipment impacted by higher room temps.

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February 21, 1994

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

SUBJECT: **SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 AND 2 DOCKET NOS. 50-387 AND 50-388 COMMENTS ON PP&L TRAINING PACKAGES, SGTS FUSIBLE LINK SAFETY EVALUATION, AND EVALUATION OF EQUIPMENT IMPACTED BY HIGHER ROOM TEMPERATURES**

Dear Mr. Shea:

We have received your letters dated February 1, 1994 transmitting copies of documents submitted to the NRC by the Pennsylvania Power and Light Company concerning nuclear safety issues raised in our November 27, 1992 10CFR21 report. Our comments on PP&L's submittals are attached to assist the NRC in its evaluation.

Please note that the PP&L safety evaluation prepared to support replacement of the standby gas treatment system fusible links is inadequate and does not assure that this safety related system will perform its required safety function in event of a boiling spent fuel pool. As indicated in PP&L and Bechtel documents, the standby gas treatment system is designed for a temperature of 125°F at the inlet of the filter trains and a temperature of 180°F downstream of the electric heaters before the HEPA filter units. Operating this system with a process temperature of 180°F conflicts with FSAR 6.5.1.1.1 and is outside the system design and licensing bases.

Please also note that the PP&L engineering evaluation of the impact of higher temperatures resulting from a boiling spent fuel pool on safety related equipment in the reactor building is valid only for the 30 day LOOP case and any other case with manual shedding of all non-Class 1E loads in the reactor building. Otherwise, the heat loads from non-Class 1E equipment operating in the reactor building will increase these calculated room temperatures.

We recommend that the NRC staff in evaluating this matter construct a timeline with markers at March 18, 1992 (discovery of problem), October 6, 1992 (PP&L's evaluation of EDR G20020), November 17, 1992 (PP&L's 10CFR50.9 report of concerns) and November 29, 1992 (our 10CFR21 report of concerns). The dates of procedure changes, plant modifications, calculations and engineering evaluations should be superimposed on this timeline. The dates for correspondence to PP&L from us prior to the 10CFR21 report and to PP&L from the NRC after the 10CFR21 report should also be superimposed on this timeline. This timeline should clearly indicate that PP&L's actions in addressing these nuclear safety

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concerns have been almost exclusively reactive. This timeline should also clearly reflect the extent of the plant modifications, procedure changes, and engineering evaluations completed since these problems were first identified at Susquehanna. This extent is an indicator of how far behind in terms of time and effort other operating nuclear power plants with the same or similar problems are in addressing these issues.

Thank you for your attention to this matter.

Sincerely,

David A. Lochbaum
David A. Lochbaum

David A. Lochbaum
for Donald C. Prevatte

10 CFR 2.790 (a)(6)

Attachment: COMMENTS ON PP&L TRAINING PACKAGES, SGTS FUSIBLE LINK SAFETY EVALUATION, AND EVALUATION OF EQUIPMENT IMPACTED BY HIGHER ROOM TEMPERATURES

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Comments on PP&L Training Packages, SGTS Fusible Link Safety Evaluation, and Evaluation of Equipment Impacted by Higher Room Temperatures

A. **Comments on PP&L Training Package SY015 B-10 Rev. 1, Residual Heat Removal System**

1) By letter PLA-4044 dated November 3, 1993, PP&L informed the NRC staff that:

> "During the pre-operational tests, the personnel recognized the need to raise water level in the SFP in order to support the higher flowrate of the RHR pumps (≈5,700 gpm vs. ≈1,800 gpm for SFP cooling)."

> "The calculation [M-RHR-039 dated May 17, 1993] shows that at the SFP level used for the successful test (≈8" above the skimmer weir) a flowrate of ≈5,700 gpm can be sustained; while at the normal water level (≈2" above the skimmer weir) a flowrate of only 2,000 gpm can be sustained."

This training package does not mention the need to raise the SFP level in order to support the desired RHR fuel pool cooling assist mode flow rate.

B. **Comments on PP&L Training Package SY017 C-1 Rev. 1, Residual Heat Removal**

1) By letter PLA-4044 dated November 3, 1993, PP&L informed the NRC staff that:

> "During the pre-operational tests, the personnel recognized the need to raise water level in the SFP in order to support the higher flowrate of the RHR pumps (≈5,700 gpm vs. ≈1,800 gpm for SFP cooling)."

> "The calculation [M-RHR-039 dated May 17, 1993] shows that at the SFP level used for the successful test (≈8" above the skimmer weir) a flowrate of ≈5,700 gpm can be sustained; while at the normal water level (≈2" above the skimmer weir) a flowrate of only 2,000 gpm can be sustained."

This training package does not mention the need to raise the SFP level in order to support the desired RHR fuel pool cooling assist mode flow rate, even though this training package states:

> "Pumps are arranged and located so adequate suction head is ensured for all operating conditions (NPSH required 18 psig)." {paragraph g, page 7}

C. **Comments on PP&L Safety Evaluation No. 92-9083, SGTS Fire Damper Fusible Link Replacement: Change to 285°F**

1) In PP&L Letter from James Kenny to Joseph Shea, "FSAR Change for SBGT Inlet Air Temperature", dated December 21, 1993, the justification for a change for SGTS inlet temperature from 180°F to 125°F in FSAR Section 6.5.1.1.1 was stated to be because "The SGTS can not handle 180°F." This justification supports the information contained in a February 1980 meeting summary between PP&L and Bechtel which indicated system components were not designed for a 180°F inlet temperature (copy of meeting summary previously provided to NRC OI investigator).

This Safety Evaluation is totally inadequate because it does not address the possibility of at least three malfunctions in safety-related equipment from different causes than previously analyzed in the FSAR. These are:

a) For the boiling spent fuel pool case, the refueling floor atmosphere will be ≈180°F at 100% relative humidity. As this atmosphere enters the ductwork enroute to the SGTS its temperature will drop, causing condensation in the ductwork. The system is presently designed to handle the condensation from incoming air at 125°F and 100% relative humidity. With the higher temperature from the boiling spent fuel pool case, two factors will significantly increase the volume of condensation that the system will be required to handle. First, the higher temperature of the incoming air will increase the heat transfer rate through the ductwork, causing a greater temperature drop and hence more condensation. Second, 100% relative humidity air at 180°F holds substantially more moisture than 100% relative humidity air at 125°F. This additional moisture also contributes additional condensation. The system is not designed to handle this additional volume of condensation. This can result in water collection in the ductwork for which it is not designed causing air flow blockage and/or collapse of the ductwork from the additional weight. In addition, the capacity of the SGTS demister could be exceeded causing moisture carryover into the SGTS filter trains.

b) High temperature cutout switches are located downstream of the SGTS electric heaters to protect them in low flow conditions. Typically these switches are set at 190°F ±5°F. If the incoming air is 180°F, the temperature rise provided by the electric heaters is likely to exceed the heater trip setpoint, thereby defeating the heater's function - prevention of condensation downstream in the SGTS filter train.

c) The higher temperature incoming air will cause the temperature in the SGTS fan room to exceed the EQ temperatures of various components in the room, including the fan motors.

The changing of the fusible links has created these three new possible safety-related equipment malfunctions which were not addressed in the licensee's safety evaluation. These possible malfunctions should have been addressed under the question concerning "malfunctions of a different type than any evaluated previously in the SAR". It should also be noted that the failures created represent common mode failures since both filter trains of SGTS would be affected.

This modification raises another issue. From the time the incorrect fusible links were identified until the modification was completed, the SGTS was in an inoperable condition with respect to its function during a boiling spent fuel pool event. In a letter dated October 20, 1992 (Attachment 25 to our 10CFR21 report dated November 27, 1992), Mr. Myers, PP&L's Manager of Nuclear Regulatory Affairs, informed Mr. Jones, PP&L's Manager of Nuclear Engineering, that if the boiling spent fuel pool event rendered the SGTS inoperable, that condition was reportable. Yet to date, no such report has been made to the NRC staff.

D. Comments on PP&L Engineering Report SEA-EE-550 Rev. 0, Evaluation of Impact on Equipment Due to Higher Room Temperature Due to Loss of Fuel Pool Cooling with LOCA and LOOP

1) By letter PLA-3132 dated January 17, 1989, PP&L informed the NRC staff that their original calculation of reactor building temperatures was non-conservative because it had assumed a LOCA/LOOP case whereas the room temperatures would be higher if heat inputs from non-Class 1E equipment were considered. Assumption 4.1.2 of this recent PP&L report states "Offsite power to Unit 1 & 2 is assumed to be lost for the entire 30 day duration of the accident." Restoration of Class 1E power does not immediately result in termination of the boiling spent fuel pool, since the non-Class 1E powered fuel pool cooling system is also non-safety related, non-single failure proof, and non-EQ. Therefore, PP&L has repeated their earlier mistake of neglecting the non-Class 1E loads in this current evaluation.

2) When the updated reactor building heat load calculation including the non-Class 1E loads was completed, several room temperatures were above the limiting EQ temperature for safety related components in these rooms. Therefore, PP&L implemented a procedure change to manually shed <u>all</u> non-Class

1E loads in the reactor building 24 hours after a LOCA. Subsequently in response to our concerns, PP&L has argued that the non-Class 1E powered spent fuel pool cooling system can be restored following a LOCA. For the LOCA/LOOP case, PP&L contends from a PRA standpoint that offsite power will be restored within 24 hours. However:

a) For the LOCA case, procedures require manually shedding of all non-Class 1E loads in the reactor building to maintain room temperatures within EQ temperature profiles.

b) For the LOCA/LOOP case, restoration of offsite power only provides a brief window of opportunity before manually shedding of all non-Class 1E loads in the reactor building to maintain room temperatures within EQ temperature profiles would be required.

If PP&L's PRA is correct and offsite power is restored within 24 hours following the LOCA, then SEA-EE-550 is significantly non-conservative because it assumes no heat loads from non-Class 1E equipment which will be operating in the reactor building. If on the other hand, PP&L concedes that non-Class 1E equipment in the reactor building cannot be relied upon post-LOCA (either due to LOOP or manual load shed), then PP&L cannot take credit for fuel pool cooling system restoration and SEA-EE-550 is adequate with respect to this point.

3) The reactor building response to a loss of fuel pool cooling is taken from calculation M-FPC-015, which is not available for review.

4) According to Step 3.1.7, "The HVAC Recirculating Fans are turned off when Fuel Pool Boiling is imminent (EP-PS-102 Rev. 6)." Apparently PP&L has already adopted this provision in their emergency procedures. However, it is not clear (and seriously doubted) that PP&L has properly supported this change for the post-LOCA case. Shutting down the HVAC recirculating fans following a LOCA with Reg Guide 1.3 source terms could significantly increase the EQ radiation levels in the reactor building. Since the reactor building air volume is no longer being mixed with the refueling floor air volume, the effective volume in which source terms are diluted is reduced by roughly 66.6%. Therefore, the radiation levels experienced by safety related equipment in the reactor building will be substantially increased if the HVAC recirc fans are turned off. This effect has not been accounted for in the current PP&L EQ Program. Additionally, the current offsite dose calculations do not account for these higher concentrations.

5) This evaluation concludes that <u>some</u> safety related equipment is rendered inoperable, although the long term cooling functions remain intact. Would this conclusion be reached without shutting down the HVAC recirculation fans? This would represent the case at time of discovery and at the time of the 10CFR21 report to the NRC.

6) PP&L's study concludes that the loss of safety-related components is acceptable since other backup components are available. However, Federal regulations state that it is not acceptable for any safety-related equipment to fail as a result of an inadequate design. Additionally, PP&L's evaluation does not account for single failures, as required by Federal regulations, which could render the backup component also inoperable. The components in question include the RHR Pump C Room Cooler and the Core Spray Pump C Room Cooler.

7) PP&L's study also concludes that the loss of function of certain safety-related valves is acceptable since they would have moved to their safe positions and would not change positions as a result of failure of their power supplies. However, two of the valves in question have safety functions in both the open and closed positions and may be required to operate to either position at any time during the thirty day period following the postulated accident. These valves are HV-E11-2F015A (RHR Injection Outboard Isolation Valve) and HV-E11-2F007A (RHR Minimum Flow Injection Valve). Therefore, the study's conclusion on this issue is incorrect.

8) These failures acknowledged in SEA-EE-550 must be considered in the context of the other previously conceded failures of safety-related components caused by flooding resulting from a boiling spent fuel pool. Again, Federal regulations do not allow the failure of safety-related components due to inadequate design.