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SUBJECT: Forwards comments on personnel radiation doses for loss of SFP cooling manual actions,slide presentation,FSAR change & probabilistic risk analysis re nuclear safety issued raised in 921127 10CFR21 rept.

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January 24, 1994

Mr. Joseph W. Shea, Project Manager
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Washington, DC 20555-0001

**SUBJECT: SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 AND 2 DOCKET NOS. 50-387 AND 50-388 COMMENTS ON PERSONNEL RADIATION DOSES FOR LOSS OF SPENT FUEL POOL COOLING MANUAL ACTIONS, SLIDE PRESENTATION, FSAR CHANGE AND PROBABILISTIC RISK ANALYSIS**

Dear Mr. Shea:

We have received your letter dated January 11, 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 analyses are attached to assist the NRC in its evaluation.

We submitted a 10CFR21 report to the NRC almost fourteen months ago because PP&L's Susquehanna Steam Electric Station was not in compliance with several Federal regulations and requirements and this condition represented a significantly higher risk to nuclear safety than had been previously analyzed. While PP&L has performed extensive analyses and implemented a number of procedure changes and plant modifications since our 10CFR21 report, the fact remains that the Susquehanna Steam Electric Station is still not in compliance with Federal regulations and requirements such as 10CFR50 Appendix A General Design Criterion 44. We respectfully suggest that the NRC require PP&L to develop a justification for interim operation of the Susquehanna Steam Electric Station until this facility can be bought into compliance with applicable Federal regulations and requirements.

We are extremely concerned by the PP&L reliance on and the apparent NRC acceptance of the use of non-safety related systems to perform safety related functions following a design bases accident. Federal regulations require that systems, components and structures performing safety related functions be designed, constructed and maintained in accordance with strict standards to provide the necessary assurance that these functions will be performed. Any system, component, or structure designed, constructed, and maintained to these standards is therefore considered to be capable of performing its safety related function until proven otherwise. Conversely, any system, component or structure which is not designed, constructed, and maintained to these standards cannot be considered to be capable of performing a safety related function until so proven.



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The fuel pool cooling and service water systems at the Susquehanna Steam Electric Station have been designed, constructed and maintained as non-safety related systems. These systems are not Class 1E powered. These systems are not single failure proof. These systems are not seismically qualified. The components of these systems are not qualified for operation in the radiation, temperature and humidity environments to which they will be subjected following a design bases accident. It is totally inappropriate and in direct violation of numerous Federal regulations to then rely on these systems to perform safety related functions following a design bases event. From a design and licensing bases perspective, these systems are not available following a design bases event. From a design and licensing bases perspective, these systems should only be considered following a design bases event to ensure that their failure will not compromise the operation of any safety related system, component, or structure.

We readily concede that the fuel pool cooling and service water systems may be available following a loss of coolant accident at the Susquehanna Steam Electric Station and should be used if so available. However, for the design bases loss of coolant accident with required Reg Guide 1.3 accident source terms, these systems cannot be assumed to operate because they have not been designed, constructed and maintained for the conditions resulting from this postulated accident, and by Federal regulations they cannot be relied upon to perform nuclear safety related functions.

Thank you for your attention to this matter.

Sincerely,

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Attachment: Comments on Personnel Radiation Doses For Loss of Spent Fuel Pool Cooling Manual Actions, Slide Presentation, FSAR Change and Probabilistic Risk Analysis

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**Comments on Personnel Radiation Doses for Manual Actions, Slide Presentation, FSAR Change and Probabilistic Risk Analysis**

A. **Comments on PP&L Letter PLA-4069, R.G. Byram to Director of Nuclear Reactor Regulation, "Request for Additional Information on Loss of Spent Fuel Pool Cooling Events", January 4, 1994**

1) In Response 1 on page 2, PP&L stated it "...*would not expect it to be necessary to use ESW for make-up since the normal systems could be used or make-up from the non-accident unit could be provided.*" The fuel pool cooling and service water systems at SSES have been designed, constructed and maintained as non-safety related systems. These systems are not Class 1E powered. These systems are not single failure proof. These systems are not seismically qualified. The components of these systems are not qualified for operation in the radiation, temperature and humidity environments to which they will be subjected following a design bases accident. It is totally inappropriate and in direct violation of numerous Federal regulations to then rely on these systems to perform safety related functions following a design bases event. From a design and licensing bases perspective, these systems are not available following a design bases event. From a design and licensing bases perspective, these systems should only be considered following a design bases event to ensure that their failure will not compromise the operation of any safety related system, component, or structure. By analogy, PP&L could not hope to justify a deficiency in ECCS design by showing that the condensate and feedwater systems at SSES will function following a DBA LOCA.

2) In Response 1 on page 2, PP&L again stated it "...*would not expect it to be necessary to use ESW for make-up since the normal systems could be used...*". In 1988, PP&L implemented a change to the SSES emergency procedures for the postulated LOCA without loss of offsite power which imposes a manual shedding of all the non-Class 1E loads in the reactor building to prevent room temperatures from exceeding EQ limits. PP&L's current stated 'expectation' conflicts with the actions taken in 1988 since the fuel pool cooling pumps located in the reactor building require non-Class 1E power to operate.

3) In Response 1 on page 2, Section 1.1, first paragraph and again in Section 1.2, first paragraph, PP&L states that their analysis of operator radiation exposure to operate the service water valves for fuel pool makeup was based on "*realistic estimates of containment leakage rates*". They should be based instead on the maximum allowable design leakage rate, $L_A$, at pressure $P_A$, as specified in the

Technical Specifications and as required by 10CFR50 Appendix J. PP&L justifies an assumed primary containment leakage rate of 0.606%/day based on recent test results. The design leakage rate, $L_A$, is 1.0%/day. The design leakage rate should be used in this analysis unless PP&L commits to reanalyze the doses if subsequent containment tests result in measured leak rates greater than 0.606%. It should also be pointed out that Type A test results are not necessarily as conservative as they are portrayed by PP&L. For Type A tests, both the inboard and outboard isolation valves are closed, providing a double leakage barrier. For an actual design basis accident, if single failure is assumed, one or more of the containment penetrations may have only one valve closed, and therefore, the combined leakage may be greater than the Type A test results.

Another facet of PP&L's response is not clear. On page 4, last paragraph, PP&L states that for the clad and fuel damage cases, they used the measured leakage through the penetrations that are not water sealed. Does this mean that, for these cases, rather than use their Type A test results which are already non-conservative, they summed the Type C leakages from the penetrations which are not water sealed? If so, then this is even more non-conservative than using the Type A results.

4) Response 1, page 6, second paragraph, states that the spent fuel pool would begin to boil ≈48 hours after loss of cooling. This time is non-conservative. If the design basis spent fuel pool heat load of $13.32 \times 10^6$ BTU/hr is used, the time to boil is ≈19.8 hours. Therefore, their statement here is non-conservative and misleading.

5) Response 1 is incomplete since it does not cover all of the manual operator actions required to perform ESW makeup to a boiling spent fuel pool. The only instrumentation available to monitor spent fuel pool temperature and level is located in the reactor building. Per PP&L procedures, spent fuel pool level must be monitored to prevent overfilling the pool. In addition, ESW makeup to a boiling spent fuel pool is specified in PP&L procedures to be in batch mode. In Response 1, PP&L does not indicate how often the ESW makeups will be required. Since at least 6.18 man-Rem will be collected for each ESW makeup evolution with Reg Guide 1.3 source terms, the frequency of the ESW makeup batches is critical to determining the overall radiation dose consequence.

**Comments on Personnel Radiation Doses for Manual Actions, Slide Presentation, FSAR Change and Probabilistic Risk Analysis**

6) In Response 2 on page 11, PP&L assumes initial decay heat loads of 8.2 MBTU/hr in Unit 2 and 6.27 MBTU/hr in Unit 1. These decay heat values are bounding today but may not be bounding throughout the SSES lifetime, particularly if PP&L implements onsite spent fuel pool storage plans.

7) In Response 2 on page 11, PP&L assumes an initial fuel pool temperature of 110°F. This is the administrative limit on fuel pool temperature. The design temperature is 125°F which should be used in this analysis unless PP&L commits to reanalyzing if the fuel pool temperature exceeds 110°F.

8) In Response 1 [sic 3] on page 19, PP&L states that for a postulated DBA LOCA with Reg Guide 1.3 source terms, the RHR Fuel Pool Cooling assist mode on the accident unit cannot be used because radiation levels in the reactor building prevent the necessary operator actions. PP&L additionally states that for a postulated DBA LOCA with Reg Guide 1.3 sources terms, the non-accident unit's fuel pool cooling and RHR systems cannot be used to cool the accident unit's fuel pool if the cask storage pit gates are installed because radiation levels on the refueling floor prohibit the necessary manual actions. We concur with these statements which support the position we have consistently maintained.

**B. SLIDES FROM PP&L PRESENTATION TO THE NRC ON OCTOBER 20, 1993**

1) In the fourth slide, "PP&L's position", the statement is made, "*operators have time to react (50 to 130 hours)*". As described previously, the time is significantly less (≈19.8 hours) for the design basis spent fuel pool heat load.

2) In the fourth slide, PP&L makes the statement, "*safety-grade makeup source is always available*". This is <u>not</u> necessarily true without violating regulatory limits on operator radiation exposure, depending on the outcome of correctly performed exposure analyses per our comments on PP&L's Enclosure 1.

3) In the fourth slide, the statement is made, "*safety-grade cooling via RHR FPC mode*". This is contrary to PP&L's final concession on in Attachment 3 to their letter to the NRC, PLA-4069 that RHR could not be used in the FPC mode for Regulatory Guide 1.3 source terms.

4) In the fourth slide, the statement is made, "*boiling environment can be mitigated*". This is true <u>only</u> if other safety grade design features in the plant as well as

regulatory requirements are violated. Therefore, in effect, this statement also is not correct.

C. **PP&L Letter from James Kenny to Joseph Shea, "FSAR Change for SBGT Inlet Air Temperature", December 21, 1993**

1) The justification for the FSAR change from 180°F to 125°F 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).

2) According to the response to Mr. Dave Pai's (the responsible system engineer at the time) comments dated June 16, 1982, 180°F is the maximum temperature at the SBGT electric heater outlet for 125°F incoming air temperature. This response correctly concluded that under these design conditions, the fire damper upstream of the electric heaters will not isolate. However, if the incoming air temperature is actually 180°F, as a Bechtel calculation determined for the boiling spent fuel pool case, the operation of the electric heaters will increase the process temperature downstream of the heaters above 180°F. The boiling spent fuel pool event at the time of our 10CFR21 report in November 1992 presented two challenges to the SSES standby gas treatment system: (a) isolation of the system by the fire dampers at 160-165°F inlet temperature, and (b) process temperatures downstream of the electric heaters exceeding 180°F (the maximum design temperature) due to heater operation. PP&L may have remedied the first problem via replacement of the fusible links for the fire dampers (although they did not report this deficiency under 10CFR 50.72, 50.72 or 50.9), but to date they haven't addressed the second problem.

3) Mr. Pai's first comment on the review and comment sheet on the FSAR change was, *"delete word 'steam mixture' as no longer are postulating fuel pool boil off."* It is very clear from this statement that the design of the plant HVAC and SGTS systems at the time, and to date, did not include the effects of fuel pool boiling. Yet, the FSAR then and now takes credit for boiling of the fuel pool for emergency cooling when normal fuel pool cooling is lost. Therefore, the change of the temperature in the FSAR from 180°F to 125°F was not to correct a typo error as PP&L alleged. 125°F truly reflected the design temperature for the HVAC and SGTS which was not compatible with conditions of a

boiling spent fuel pool to which the plant was licensed, and this incompatibility was known at the time the FSAR change was made and at the time of PP&L's "voluntary" report of our concerns on November 17, 1992.

D. **PP&L Report SA-TSY-001 Rev. 0, Probabilistic Risk Analysis**

1) On page 3, a confidence level of 80% that the fuel pool cooling system will remain operable after a LOCA is assumed based on information provided by PP&L's Civil Engineer who analyzed the system piping integrity. Piping integrity is only one factor in determining system availability. The fuel pool cooling system is non-safety related, non-seismic, non-EQ, and non-Class 1E powered. The FPC system is automatically load shed for a LOCA even if power is available and cannot be manually restarted without exceeding the maximum allowable operator radiation dose. PP&L's piping studies may indicate that the system piping will withstand the hydrodynamic loads from the LOCA, but they do not demonstrate that the system MCCs, valves, pumps, motors, and other components will survive these loads. The system components will experience radiation, temperature and humidity conditions following a LOCA for which they are not designed. The system is not single failure proof so a component failure caused by the hydrodynamic loads, the environmental conditions or a random cause can incapacitate the system. Therefore, an assumed confidence level of 80% based solely on piping integrity represents the absolute maximum confidence level if all other confidence levels are 100%. Since these confidence levels clearly are not 100%, the assumed confidence level should not be 80%. We recommend that a parametric study using confidence levels between 20% and 80% would be more appropriate.

2) In Section 4.1.1 on page 5, the basis for neglecting the seismic event case is improper. As we have repeatedly pointed out to PP&L and to the NRC, the SSES FSAR Appendix 9A analysis is incomplete because it only addresses the radiological consequences from a boiling spent fuel pool. If all of the consequences are addressed, the event becomes more significant because multiple failures of safety related systems are encountered. PP&L has thus far acknowledged to the NRC such resultant failures as the loss of two core spray pumps as well as other unspecified failures.

3) In Section 4.1.2 on page 5, it is stated that operators will constantly monitor indications on panels 1C206 and/or 2C206 during a loss of fuel pool cooling event in accordance with

off-normal procedures. In the PP&L's response dated January 4, 1994 on radiation doses for manual actions, there is no discussion of monitoring indications during ESW makeup or prior to restoration of normal fuel pool cooling. These actions must be taken and an operator will be exposed to high radiation rates in performing these actions. Therefore, PP&L provided the NRC with an incomplete and misleading response by neglecting to address these required actions.

4) The list of initiating events in Section 4.1 covers the Station Blackout event, but does not address the Appendix R fire event. If the overall core damage contribution from loss of spent fuel pool cooling is quantified (as PP&L claims to have done), then the Appendix R initiating event should be analyzed.

5) The results table on page 11 provides numerous probabilities for the various cases. The most important probability value is missing - the probability of complying with Federal regulations on nuclear safety. This probability should be 1.0, but at SSES it is not.

6) Conclusion 6.4 on page 13 reports that "*the frequency of fuel pool boiling during a LOOP initiated loss of fuel pool cooling event is very high*", but justifies it based on the FSAR Appendix 9A analysis. As we pointed out in comment (D)(2) above, the FSAR Appendix 9A analysis is flawed.