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The Honorable Shirley Jackson
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**SUBJECT: SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 AND 2, DOCKET NOS. 50-387 AND 50-38 NRC RESPONSES TO SPENT FUEL POOL COOLING SAFETY ISSUES**

Dear Chairman Jackson:

On November 27, 1992, the undersigned submitted a 10CFR21 report to the NRC reporting our grave concerns with the safety of the design of the Susquehanna Steam Electric Station and thirty-five other boiling water reactor nuclear plants. The thrust of these concerns was the inability to provide cooling to the spent fuel pools for design basis accident conditions, and the resultant potential for fuel meltdown in these pools and catastrophic radioactivity release to the environment. Today, three years later, the NRC Staff has completed a process to bury these concerns without resolving them. This letter is to inform you of this outrage and to request that you become personally involved in bringing about its resolution.

This request is a repeat from last year. On July 8, 1994, we met with your predecessor, Chairman Selin, and his Staff in his Rockville, Maryland office to discuss the NRC's response to our report. The meeting was called by the Chairman in response to our correspondence with the Staff and officials in Congress and the states in which these plants are located expressing our dismay with the Staff's efforts to dismiss these issues with invalid probability and legalistic arguments. Chairman Selin appeared to react with the appropriate trepidation and assured us that the Staff's attitude and approach would change.

However, by November of 1994 it was clear that there had been no fundamental change in the Staff's attitude or approach, and in a November 29, 1994 letter to Chairman Selin we expressed our disappointment. Several events had occurred which were contrary to the Chairman's instructions to the Staff in our July 8 meeting with him, including issuance of a draft Safety Evaluation Report (SER) concluding that the event of concern was unlikely, and a meeting between ourselves and the Staff on October 25, 1994 in which they continued in the same mode as before. Our November 29, 1994 letter also transmitted our detailed comments on the draft SER.

Since then, six other events have transpired which, considering the Chairman's direct involvement, have convinced us that there is, in reality, no commitment in the NRC's Staff or Management to properly resolve our concerns or any other significant safety issues that have the potential for costly resolution.

**First Event:** The first event was a presentation by the Staff to the ACRS on December 8, 1994 in which they dismissed our concerns based on a draft Probabalistic Risk Assessment (PRA). The ACRS accepted their position, as reported to the Chairman in their letter of December 19, 1994, with the proviso that they had not reviewed the PRA. Had the ACRS reviewed the PRA, they would have reached a different conclusion. Our detailed review revealed many non-conservative errors in inputs and assumptions, yielding, as would be expected, non-conservative conclusions. The two most glaring flaws were as follows:

First, the PRA analyzed the wrong event. The event we are concerned about is the loss of the ability to cool the fuel pool during a design basis LOCA. The PRA analyzed the "likelihood-of-core-damage event". It concluded that the core damage event had a probability of 5.0E-8 and 1.1E-8 per year for the "as-found" and "as-fixed" conditions at the Susquehanna Plant respectively, producing a conclusion that the event was incredible. However, for the loss-of-fuel-pool-cooling event, which the analysis addressed only as a precursor to the core damage event, the probability was calculated at 6.8E-5 and 2.1E-5 per year - almost two thousand times higher. This probability *is* credible. Additionally, these results were derived using the extremely non-conservative assumptions addressed below and in our attached detailed PRA comments. Had the correct assumptions been used, the probabilities would have been significantly higher.

Second, the analysis circumvented the regulations regarding containment and control of radioactivity resulting from a loss-of-coolant-accident (LOCA). These regulations specify the level of core damage, containment leakage, etc., that must be assumed, as well as the maximum allowable radiation exposures to operators and the general public. They specify the final design layers of "defense-in-depth", a fundamental design principle for American nuclear plants. Compliance with these regulations is not optional as the PRA implies, it is mandatory.

The PRA didn't stay within these regulations. Its basic starting premise was that in an accident there would be no core damage, therefore, there would be no threat to the fuel pool cooling system from the resultant environmental effects, and access to the reactor building to effect actions to cool the spent fuel pool would not be restricted by radiation. Using this logic, one could also justify eliminating the primary containment, secondary containment, the Standby Gas Treatment System, most plant shielding, environmental qualification of equipment, and many of the other plant features required by regulation to deal with the consequences of accident damaged fuel. Such logic directly contradicts the Three Mile Island accident experience in which there was virtually complete disintegration of the core. This was also precisely the logic that contributed to the Chernobyl disaster, logic that was roundly criticized by the NRC in NUREG-1250. Yet, this was the starting point for the PRA, and this PRA was the foundation of the Staff's position.

We also found that the PRA was based on numerous other incorrect, non-conservative, and unverified assumptions that would cause the results to be extremely non-conservative. Virtually all of these assumptions were also contained in the draft SER on which we provided comments in our November 29, 1994 letter - **before the Staff's ACRS presentation**. All of this had also been the topic of numerous conversations and letters between ourselves and the Staff over the previous two years. There is therefore no question that Staff was aware of the improper focus, the non-compliance with regulations, and the technical discrepancies in the PRA. That they would knowingly use such a flawed analysis as the basis for their ACRS presentation is particularly reprehensible.

**Second Event:** The second event was the generation of a memorandum by Mr. John Darby to the NRC dated March 19, 1995 concerning the NRC's review of Individual Plant Evaluations (IPEs) [PRAs performed on individual nuclear plants]. In this document he expressed grave concerns with the technical validity of many IPE assumptions and the NRC's reluctance to address his concerns. Mr. Darby is a highly qualified, highly regarded consultant who had performed twenty-five IPE reviews for the NRC. His memorandum came only after repeated futile attempts to resolve his concerns with the Staff. He stated that the situation had "...reached the point to where in good conscience I must cease to work on your IPE review program until my concerns are addressed." Because of the obvious impact such a memorandum could have on one's career, it would not be written on a whim. It, therefore, must be taken seriously, and it reinforced our contentions and demonstrated that our experience with the Staff was not unique.

It should also be pointed out that this was also not a unique case with the undersigned. One of the undersigned, Mr. Prevatte, has participated as a contractor in more than thirty team inspections for the

NRC in which numerous other serious technical issues have been similarly dismissed by the Staff based on invalid probabalistic and legalistic arguments. Such experiences are also commonplace and well documented with other NRC inspectors.

**Third Event:** The third event was the response of Mr. William T. Russell (Director, Office of Nuclear Reactor Regulation) to our November 28, 1994 petition pursuant to 10CFR2.206. This petition requested that the NRC not take any of the following licensing actions relating to the storage of spent nuclear fuel which might exacerbate our 10CFR21 safety concerns until they were resolved:

(1) Issue any new operating license.
(2) Issue any license amendments involving increased spent fuel pool storage capacity or increased licensed reactor power level.
(3) Issue any renewal or extension of an operating license.
(4) Issue any license or license amendment for independent spent fuel storage.

In Mr. Russell's response of March 8, 1995, our petition was rejected because "...it focuses entirely on licensing actions and, therefore, is not within the scope of 10 CFR 2.206." He further stated that in lieu of granting our petition, we were free to raise our safety concerns through other licensing processes in any individual cases involving the above described licensing actions.

Both of Mr. Russell's responses were invalid as follows:

First, not only are licensing actions within the scope of 10CFR2.206, they are its primary focus as described in its first sentence, "Any person may file a request for the Director of Nuclear Reactor Regulation, Director of Nuclear Material Safety and Safeguards, Director, Office of Inspection and Enforcement, as appropriate, to institute a proceeding pursuant to [Section] 2.202 to modify, suspend or revoke a *license* [emphasis added], or for such other action as may be proper."

Second, Mr. Russell's statements regarding our usage of other licensing processes circumvented the intent of this regulation. Its intent is to give "Any person" another avenue for assuring that relevant safety issues are considered in the licensing process. For Mr. Russell to say that we must address individual cases using other licensing processes is to deny the validity of 10CFR2.206. If it is not valid for cases such as ours, what is it valid for? Why is it there?

**Fourth Event:** The fourth event was the beginning of inspections of fuel pools at nuclear plants, an activity that was instituted in response to our insistence that the scope of the NRC's evaluations not be confined to the Susquehanna Plant. However, in reviewing the reports for the first two inspections at the Brunswick and Montecello Plants, it was revealed that they also did not address the real area of concern. Therefore, predictably, they did not discover serious problems, and they only served to reinforce the Staff's position that there are no serious problems. Although they did look at design, maintenance, operations, and other areas, they did not address the ability to provide cooling to the spent fuel pools *under post-LOCA conditions*, the real point of concern.

**Fifth Event:** The fifth event was issuance of the final SER for these concerns. This document was the NRC's final closure document for this issue. Although we provided detailed comments on the draft SER, we were not consulted on any of them, and virtually none were incorporated in the final document. When we confronted the Staff regarding this, their response was that there had been insufficient budget to address them.

**Sixth Event:** The sixth event was issuance of Information Notice No. 93-83, Supplement 1. This document notified the industry of the NRC's findings following its review of the Susquehanna concerns.

The Staff went to great lengths to explain the PRA results, yet failed to mention, even in passing, that it had determined that the Standby Gas Treatment System, and therefore secondary containment, would be unavoidably lost if the fuel pool boiled. The Staff also failed to emphasize that the Susquehanna Plant was modified to provide the control room operator with indication of spent fuel pool temperature and level or any of the other modifications made at Susquehanna in the design, procedures, training, etc. to address these concerns.

Our Part 21 report was made in good faith at great personal risk and cost. We expected that the NRC would also act in good faith. To-date, it has not; we have received no responses to our technical comments; we have never even been asked any technical questions regarding our concerns. By contrast, the Staff has encouraged the licensee to take refuting positions that have little or no technical bases.

The current attitude of the Staff toward potentially high cost issues such as ours is "shoot the messenger", make it go away with PRA, and it's not a safety concern until it becomes a fait accompli. However, the consequences of some issues, such as ours, are too grave to wait for the event to occur. That is why we cannot allow the Staff's perfunctory dismissal of these concerns to stand unchallenged.

This letter is being written in the hope that new leadership will foster a new attitude in NRC management. Such a new attitude will be demonstrated by our safety concerns being fully and completely resolved, not by burying them, as is the current practice, but by open dialogue where each and every concern is individually addressed and resolved strictly on its technical merit, or lack thereof. We expect that in your dedication to the NRC mission you will assure that this is indeed accomplished.

We look forward to working together, and we are at your service should you require any additional information on this matter.

Sincerely,

Donald C. Prevatte for
David A. Lochbaum

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Enclosure

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## COMMENTS OF DAVID A. LOCHBAUM & DONALD C. PREVATTE ON DRAFT NRC RISK EVALUATION REPORT DATED OCTOBER, 1994 RELATED TO LOSS OF SPENT FUEL POOL COOLING EVENTS AT SUSQUEHANNA STEAM ELECTRIC STATION

1. General Comment: Two plant conditions were examined in this analysis, the "As-found" and the "As-fixed" conditions for the Susquehanna Plant, the "As-fixed" condition being the plant condition after numerous modifications were made in its design, analyses, operating procedures, training, etc., to address the concerns raised by our 10CFR21 report. Implied in the NRC's acceptance of this analysis is their agreement that the Susquehanna Plant was indeed "not-fixed" at the time of the report. By extension then, all of the other nuclear plants with similar designs, of which there are approximately thirty-five, that have not made these modifications are still "not-fixed". What is being done to assure that these upgrades are being implemented at these other plants?.

2. As first described on Page v of the Executive Summary, the focus of this evaluation was to determine the "likelihood of core damage" as a result of a loss of fuel pool cooling. This focus was a fundamental error of this report; it completely missed the point of our 10CFR21 report, that the loss-of-fuel-pool-cooling event was the first event of concern, not just a precursor to core damage as it was handled in this analysis. This event in and of itself, without even considering core damage, would be catastrophic. It would result in meltdown or even burning of the fuel in the fuel pool, which would produce a much larger source term than core damage. It also would cause secondary containment failure (the only containment boundary around the fuel pools). The results would be offsite and operator exposures orders-of-magnitude higher than allowed by 10CFR100 and 10CFR50, Appendix A respectively. The probability of subsequent core damage would be a moot point, for by then the catastrophe would already have occurred. The subsequent core damage would only relate to the extent of the catastrophe, not the fact of it.

   Had this been addressed by the report, even without considering the gross non-conservatisms that were employed, the conclusions should have been vastly different. This can be seen from the report's determination of the relative probabilities of these two events, loss of fuel pool cooling versus core damage. The loss of fuel pool cooling event was calculated to be approximately two thousand times more likely. When combined with its consequences, this describes an event of extremely high risk.

3. Executive Summary, Page viii, the risk analysis results for loss of fuel pool cooling were stated as 6.8E-5 per year for the as-found case and 2.1E-5 for the as-fixed case. What was the criteria used to determine that these values were acceptable?

4. That the probabilities for loss of fuel pool cooling calculated by this analysis are as low as they are is not surprising considering the invalid assumptions that were made regarding alternate cooling methods for the spent fuel pool when normal cooling is lost. If these assumptions had not been made, the probabilities would have been orders of magnitude higher. The invalid assumptions used were as follows:

   a. As first documented on Page viii, 3rd paragraph, Case 3, it was assumed, without verification, that for loss of cooling for one fuel pool, if it were cross-connected to the opposite unit's pool through the cask storage pit by removal of the isolation gates, the opposite unit's spent fuel pool cooling system would be capable of cooling both pools. This was based on PP&L's assertion that with the cooled pool's temperature maintained at 170°F, the non-cooled pool's temperature would not exceed 200°F. It appeared to be assumed that cooling of the pool with the failed cooling system would be accomplished by natural circulation to and from the pool with the operable system. This was based on PP&L's statement that this ability had been "demonstrated". However, the "demonstration" which PP&L took credit for was a loss-of-service-water event which

occurred on one of the units' fuel pool cooling systems when the pools happened to be cross-connected. (Service water provides the cooling medium for the fuel pool cooling heat exchangers.) Since little difference in temperatures was observed between the pools over a period of about one hour, PP&L maintained that this "demonstrated" the claimed capability. However, this event was not a valid demonstration. Since the heat load was relatively low, the duration of the event was relatively short, and the temperatures of the pools were very low, it could not have produced meaningful data. Additionally, in the unit with the loss-of-service-water, the fuel pool cooling system pumps were left operating, creating forced circulation in that pool. This did not even approximate, much less simulate, a natural circulation condition. Finally, this event was not conducted under controlled test conditions. To maintain that this event "demonstrated" this ability is an absurd manipulation of the facts.

This ability has also never been "demonstrated" by a valid analysis. PP&L's "engineering judgement" was accepted in this report in lieu of analysis, and this judgement defies basic principles of natural circulation. Natural circulation requires a flow path from the hotter pool to the cooler pool at an upper elevation and a return path from the cooler pool to the hotter pool at a lower elevation. Although the gates between the pools could provide the hot flow path high in the pools, there is no return path. Therefore, natural circulation would not occur. The Staff has stated that they assumed that the expansion of the water in the hotter pool would cause flow to the cooler pool, carrying heat with it. However, this transfer mechanism would occur only during the initial heatup, and the mass/heat transferred would be relatively small. After the initial heatup, there would be little additional mass or heat transfer between the pools since there is no complete flow circuit.

It should also be recognized that this supposed "fix" for the Susquehanna Plant, even if it were valid, is not applicable to the other BWR plants with separated fuel pools or single unit plants since, for these, there is no opposite unit fuel pool to provide the alternate cooling or makeup sources.

b. As documented on page 3.3 under "top event" "ALT cooling", it was assumed without verification that pools could be cooled by a "feed and bleed". This consists of feeding a pool from a relatively cool source of water, and bleeding hot water from the pool. To date, this method is entirely imaginary and undeveloped, i.e., there are no plant systems with the capability to supply the water volume required to prevent boiling, there are no plant systems which can remove this volume of water from the pool, and no analyses or testing of this mode have been done. Even if sufficient volume could be supplied by the fire protection system, for instance, if there were no method of removing the hot water, it would simply overflow into the building, and if it overflowed, there are no methods available for removing this quantity of water from the building and safely disposing of it. If it were not removed, it would quickly threaten the operability of the ECCS pumps and other safety-related equipment. Please note that credit is also taken for this method of cooling in the discussion at the top of page 3.12 and in Conclusions 3 and 5 on page 4.2.

5. Section 2.2, Analysis Assumption 2.2, As-Found Assumption 2, same as Comment 4.a. above.

6. Section 2.2, Analysis Assumption 2.2, As-Found Assumptions 4 and 5 state, in effect, that the spent fuel pool (SFP) temperature is controlled to 115°F and that this would assure a time-to-boil greater than 25 hours. This may be an <u>objective</u> of the "as-fixed" procedures, but it was not addressed in the "as-found" procedures. Additionally, even in the "as-fixed" condition, plant operating conditions can force full core offload which may not allow these objectives to be met.

7. Section 2.2, Analysis Assumption, Page 2.7, Assumption No. 5 is confusing. If a minimum time-to-boiling of 25 hours is assumed, why did Case 5 for the as-found condition use 15 hours?

8. Section 2.2, Analysis Assumption, Page 2.7: Assumption No. 11 is **NOT** conservatively modelled. In the plant configuration with a reactor core entirely offloaded into the spent fuel pool and the reactor cavity gates installed, the Technical Specifications on the ECCS and electrical power distribution systems for that unit are not applicable, and therefore, their availability is not assured. This configuration was not prevented by any procedures and is one which Susquehanna and other plants are frequently in during outages. However, the manner in which these system unavailabilities was handled in the analysis was extremely unrealistic and non-conservative.

9. Section 2.2, Analysis Assumption 2.2, As-Found Assumption 12 states that the diesel generators can be aligned to supply the spent fuel pool cooling (SFPC) system. However, it does not mention the non-safety-related service water system which also must operate in order for the SFPC system to be operable. This system constitutes a significant additional electrical load for the diesel generators which does not appear to have been considered here and which therefore calls into question the validity of this assumption.

10. Section 2.2, Analysis Assumption 2.2, As-Found Assumption 13, same as Comment 4.a. above.

11. Section 2.2, Analysis Assumption, Page 2.7: Assumption No. 13 may be true now, but not always. The design heat removal capacity of the FPCS is 13.2 MBTU/hr. The maximum capacity after margin for heat exchanger tube plugging is 12.6 MBTU/hr. There are times, especially after more and more irradiated fuel assemblies are placed in the pools, when the combined heat loads of both pools exceeds the heat removal capacity of a single FPCS. This fact appears to have been overlooked in this analysis.

12. Section 2.2, Analysis Assumption 2.2, As-Found Assumption 14 addresses the availability of the non-1E powered reactor building cranes for removing the fuel shipping cask pit gates. This assumption is valid only if power can be supplied from the diesel generators and if radiological conditions allow operator access to the refueling floor. For a design basis LOCA, radiological conditions prescribed by Regulatory Guide 1.3 would preclude operator access.

13. Section 2.2, Analysis Assumption, Page 2.8: Assumption No. 15 is non-conservative for the as-found case. There were <u>NO</u> administrative controls in place at Susquehanna to assure the cask storage pit was always full of water, and there is no evidence that PP&L always maintained it that way.

14. Section 2.2, Analysis Assumption 2.2, As-Found Assumption 17 describes makeup to the SFPs from diesel fire pumps. As with Assumption 14, this assumption is valid only if radiological conditions would allow operator access to the refueling floor. As noted in Comment 7. above, operator access would be precluded.

15. Section 2.2, Analysis Assumption, Page 2.8: Assumption No. 19 is simply unrealistic. During a refueling outage, considerable maintenance is performed requiring safety systems to be taken out of service. Unlike maintenance and testing during normal operation, these systems are frequently disassembled such that they cannot be quickly returned to service, even in an emergency.

16. Section 2.2, Analysis Assumption 2.2, As-Found Assumption 20 states that "When HVAC Zone 3 is not isolated, the safety equipment in HVAC Zones 1 and 2 reaches equipment failing critical

temperatures approximately 8 hours after the onset of boiling in the SFP." It should be noted that in the "as found" condition there were no procedures for isolating Zone 3 or even awareness that isolation was necessary to preserve the operability of this equipment. Therefore, this equipment would have to be assumed to have failed.

17. Section 2.2, Analysis Assumption 2.2, As-Found Assumption 20 also states that "With the recirculation fans off, the SGTS would fail approximately 15 hours after the SFP begins to boil and the ECCS equipment would fail approximately 24 hours after the SFP begins to boil." It should be noted that turning off the recirculation fans is an "as-fixed" procedural change. In the "as found" condition, without the fans being turned off, the failure of the ECCS equipment would have occurred much earlier. It should also be noted that even in the "as fixed" condition, this major equipment for preventing core damage and mitigating offsite dose fails, a condition not allowed by regulations.

18. Section 2.2, Analysis Assumption 2.2, As-Found Assumption 20, it is not clear if the adverse temperature and moisture effects of SFP boiling on the ECCS pump power supplies was considered in determining the time to failure. These power supplies are also located in the reactor building, and their cooling units are not designed for latent heat cooling, i.e., condensing of vapor from a boiling spent fuel pool. It is also not clear if the ECCS pump room coolers, which also are not designed for latent heat cooling, were properly considered. In previous analyses, none of these components, whose operabilities have a huge impact on the time to failure of the ECCS pumps, were considered.

19. Section 2.2, Analysis Assumption, Page 2.8: Assumption No. 21 relies on action to be taken when area temperature in HVAC Zone III reaches 125°F. What instruments monitor this parameter? What is the effect on the risk analysis if this unsubstantiated assumption is not made?

20. Section 2.2, Analysis Assumption 2.2, As-Found Assumption 23 appears to imply that the components of the condensate and feedwater system will be available for core cooling. This is not necessarily the case since they are all non-1E powered.

21. Section 2.2, Analysis Assumption 2.2, As-Found Assumption 25 states that several other methods exist for backup SFPC, and it lists these - feed and bleed, use of diesel fire pumps, and use of RHR. None of these would be available in a design basis LOCA, and in the case of feed and bleed, this method has never been analyzed or otherwise demonstrated as capable of providing the required cooling for the SFPs.

22. Section 2.2, Analysis Assumption 2.2, As-Found Assumption 26 states that failure of ECCS equipment due to flooding from condensate from the boiling SFP is confined to only one train of core spray. This statement has two basic flaws. First, per regulations, failure of even one train of any ECCS as a result of inadequate design is not acceptable. Second, the assumption that failure would be confined to only one train of core spray is false; it is based on non-conservative estimates of water depth in the basement of the reactor building provided by the licensee and credit for "watertight" doors which for this event would have the water pressure acting in the wrong direction unseating them rather than seating them as per design. Both trains of core spray pumps, as well as all of the RHR pumps, would fail if the correct assumptions were used.

23. Section 2.2, Analysis Assumption, Page 2.9: Assumption No. 26 and Assumption No. 24 do not bound the seismic event case. If the FPCS piping breaks (as is assumed in the design basis SSES FSAR Appendix 9A analysis), an unisolated leakage pathway exists to dump water from the skimmer surge tank directly into the upper elevations of the reactor building. PP&L's strategy

to raise the spent fuel pool level 8" in order to use the RHR system in the fuel pool cooling mode would only maintain a constant flow of water through the break. Simply assuming that the water ends up in the reactor building sumps does not address the problem created by such a break.

24. Section 2.2, Analysis Assumption 2.2, As-Found Assumption 27 concerns response of the Technical Support Center (TSC) staff. This assumption is not valid for the "as found" condition. At that time, none of the complex interactions of this event had been provided in training to plant operations personnel or persons who would staff the TSC.

25. Section 2.2, Analysis Assumption, Page 2.10, Assumption No. 31: In the as-found condition, PP&L's accident procedures directed them to manually shed reactor building non-Class 1E electrical loads 24 hours into the event if there was no loss-of-offsite-power (LOOP). Since the FPC pump are non-Class 1E powered, this, in effect, <u>directed</u> the plant **<u>INTO</u>** the loss of fuel pool cooling event with which we are concerned, and this procedural error was not recognized by PP&L. The risk analysis does not reflect these facts, and thus is <u>grossly</u> non-conservative.

26. Section 2.2, Analysis Assumption, page 2.10: The analysis does not consider the setpoint of the as-found fusible links in the Standby Gas Treatment System (SGTS). They would have rendered the SGTS inoperable as the first SFP approached boiling in the as-found case.

27. Section 2.2, Analysis Assumption 2.2, As-Found Assumption 35 concerns the refueling floor hinged panels and how they would open to remove energy from the secondary containment and thereby reduce its temperature. This is also not a valid assumption. First, these panels are designed for tornado pressure relief. For SFP boiling, insufficient pressure would develop to actuate these panels since ample leak paths would exist for steam from the SFP to migrate to other parts of the building where it would condense on the relatively cool building structures and equipment. Second, if they did actuate, by so doing the safety-related secondary containment would be compromised

28. On page 2.12, under "<u>SFP Decay Heat Evaluation and Case selection</u>", same as Comment 4.a. above. It should also be pointed out that this method of cooling had not even been conceptualized for the as-found condition, and that at that time, the plant was normally operated with the gates installed. Therefore, no credit should have been assumed for this method of cooling for the as-found condition, even if it had been a viable cooling method, which it isn't.

29. Section 2.3, Page 2.12: The FPCS is designed to maintain **<u>ONE SPENT FUEL POOL</u>** below 125°F, not two pools when the combined decay heat load from both pools exceeds the MNHL.

30. Page 2.15, under "Manual cross-connection of the SFPs", same as Comment 12 above.

31. Page 2.17, the statement is made that "Failure sequences with a...time-to-boil of greater than 50 hours are considered incredible and are dropped from further analysis." Implicit in this statement is the assumption that for times-to-boil greater than 50 hours, plant staff will devise a way, as not yet described, to prevent boiling. To-date, none of the schemes suggested by the licensee can accomplish this under design basis LOCA conditions, regardless of the time available. If such a scheme has not yet been devised after more than three years since this concern was first identified, it's not reasonable to expect that it would be devised in the heat of battle of a LOCA. This assumption is therefore invalid.

32. Page 2.18, the last sentence addresses the failure of ECCS equipment as a result of the effects of a boiling SFP. Comment 16 above is also applicable here.

33. Section 2.3, Page 2.18: The decision to neglect failure sequences that require greater than 50 hours to develop is **ABSOLUTELY NOT** incredible for the DBA LOCA. Susquehanna is not a simulator that can be re-initialized with the flick of a switch. If the DBA LOCA occurs and significant radioactivity is released into containment, the event does not end within 50 hours. At TMI, the NRC and MetEd were struggling with a hydrogen bubble days after the event initiation.

34. The time-to-boil values developed in Appendix A, on page 2.19, and at other places in the analysis are non-conservative in that they use decay heat conditions that existed at the time of Susquehanna's spring, 1994 outage. This is significantly less than the heat load for filled-to-capacity conditions that will exist later in plant life.

35. In Section 3.1 the statement is made that "The potential for SFP drainage was considered not credible based on SSES having redundant and diverse features to preclude SFP drainage and the SFPs having redundant and diverse inventory makeup capabilities which reduce the potential for SFP drainage." Although this is true for the "as fixed" condition since the gates separating the pools have been removed, it was not true for the "as found" condition. In that condition, each pool could only be supplied by its own makeup features. For the design basis LOCA, these features would be rendered inaccessible in that unit due to the radiation conditions in the reactor building as prescribed by Regulatory Guide 1.3.

36. Section 4.0, Page 4.2: Item (3) states that event sequences that involve greater than 50 hours to attain near boiling conditions are neglected because "this allows sufficient time that PP&L can assemble the support necessary to provide event mitigation." This is invalid for two reasons: First, PP&L's emergency procedures at the time of our report made the DBA LOCA event worse by requiring manual shedding of non-Class 1E electrical loads in the reactor building as described above in Item 25 Second, sufficient time may be an advantage in handling a problem only if one is made aware there is a problem. For the as-found case, there was **NO FUEL POOL LEVEL OR TEMPERATURE INSTRUMENTATION AVAILABLE IN THE CONTROL ROOM** to indicate that the event was developing. What would have triggered PP&L's response?

37. Table 3.1 lists the initiating events and their frequencies used in this analysis. The frequency shown for the Station Blackout (SBO) is 2.73E-8/yr. SBO, which was not in the original licensing bases for most plants, has been deemed a credible event by the NRC, and licensees have been forced, at great expense, to upgrade their plants to survive this event. The ability to cool the fuel pool under all normal operating and accident conditions *is* in the licensing basis for all plants, and per this report, the probability of a failure-to-cool-the-fuel-pool event is approximately a thousand times higher than SBO. The consequences are essentially the same, catastrophic offsite releases of radiation. Please explain the inconsistency here; why does the NRC consider the loss of the ability to cool the fuel pool a low probability, low risk event when SBO is considered high risk?

38. Page 3.2, the "top event", "5$^{th}$ EDG" in the center of the page describes the use of "surplus emergency power supplies" to power the non-safety busses in order to power the SFPC system(s). This assumes the 5$^{th}$ EDG is always available in spite of there being no Technical Specifications or design or licensing bases which require its availability. Additionally, it is questionable if the large electrical load associated with the operation of the non-safety-related service water system in support of the SFPC system was considered with regard to the 5th EDG's capacity.

39. Page 3.2, the "top event" "crosstie" presumes that the cooling system for one pool can cool the other if they are crosstied. See Comment 4.a. above.

40. Appendix A, Section 2.0, Success Criteria, see Comment 4.a. above.

41. All of the failure rates described in Appendix B are based on the components being operated within their design basis environmental parameters. However, the fuel pool cooling system electrical components, since they are not qualified for the LOCA environment, would be operating well outside their design basis environmental parameters. Therefore, their failure rates would be expected to significantly higher than those used here.

42. Appendix D, "Assumptions - Bases and Impact" on page D.6, Item 2, see Comment 4.a. above.

43. Page D.10, Item 12 assumes that five emergency diesel generators are available. See Comment 38 above. Also, beyond the initial availability of the 5th EDG, this takes no account for single failure of one of the EDGs which may be initially available.

44. Page D.10, Item 13, see Comment 4.a. above.

45. Page D.11, Assumption 17, see Comment 4.b. above.

46. Page D.15, Assumption 25, see Comment 4.b. above.

47. Page D.15, Assumption 26, see Comment 22 above. Additionally, this assumption also relies on early isolation of the manual drain line isolation valves from the various ECCS pump rooms. No procedural direction was provided for isolating these valves for the "as found" condition of the plant. Therefore, no credit should be taken for their having been isolated for that condition, and general flooding of all of the ECCS pump rooms and therefore loss of these pumps should be assumed. Since these lines bypass the "watertight" doors, their ability to seal with pressure in the wrong direction becomes a moot point. It should also be considered that for the design basis LOCA, radiation conditions in the reactor building would preclude operator access to operate these valves for both the "as found" and "as fixed" conditions.

48. Page D.17, Assumption 35, see Comment 27 above.