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SUBJECT: Responds to 940517 ltr & forwards comments on BWROG of spent fuel pool cooling sys.

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May 25, 1994

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SUBJECT: SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 AND 2 DOCKET NOS. 50-387 AND 50-388 COMMENTS ON BWROG LETTER DATED 03/04/94 AND PP&L SUBMITTALS DATED 04/29/94, 05/04/94 AND 05/05/94

Dear Mr. Shea:

We received your letter dated May 17, 1994 transmitting five enclosures related to the evaluation of the 10CFR21 report we submitted dated November 27, 1992. We have reviewed these documents and our comments are provided on the attached pages.

PP&L has informed the NRC that the Standby Gas Treatment System will fail due to water accumulation in the intake ductwork following fuel pool boiling event. Therefore, it seems imperative that the SSES licensing basis must explicitly require that fuel pool cooling be maintained or restored following every design bases event in which the Standby Gas Treatment System performs a mitigating function. Since fuel pool cooling is required to prevent failure of the safety-related Standby Gas Treatment System in the events, the systems relied upon to provide fuel pool cooling must be designed, tested and operated in a manner commensurate with this safety function.

With the documents enclosed with your May 17, 1994 letter, and with respect to the concerns expressed in the NRC's letter of May 19, 1994 to PP&L, it seems apparent that the NRC has sufficient knowledge of deficiencies in the design of nuclear power plants for loss of spent fuel pool cooling following design bases events to initiate generic actions to address these problems. As the NRC's experience with SSES has clearly demonstrated, it takes time to determine the plant specific licensing bases and as-built capabilities. It seems prudent therefore to start licensees on this process.

We appreciate your consideration of our comments on this issue. Please contact us if there are any questions regarding these comments.

Sincerely,

Donald C. Prevatte for
David A. Lochbaum

Donald C. Prevatte
Donald C. Prevatte

Attachment

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PDR ADOCK 05000387
S PDR

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1.0 <u>Comments on Enclosure 1, Letter from L. A. England, BWROG to A. C. Thadani, NRC, "*BWR Owner's Group Assessment of the Spent Fuel Pool Cooling System*", dated March 4, 1994</u>

1.1 We agree entirely with the statement in Section 1.3 that the design deficiencies identified in our 10CFR21 report dated November 27, 1992, are not applicable to plants with a safety-grade SFPC system. This is, in fact, the precise root of those problems - in the Susquehanna design, the non-safety related Fuel Pool Cooling System, the non-safety related Service Water System, and the non-safety related RHR Fuel Pool Cooling Assist mode are being relied upon to fulfill a safety function, contrary to both regulations and sound engineering principles.

1.2 The first conclusion under Section 2.0 is flawed in that it exclusively applies to the SFPC system licensing basis. We have repeatedly stated that with the sole exception of nonconforming fuel pool level and temperature instrumentation, the Susquehanna Fuel Pool Cooling System satisfied all applicable regulatory requirements. However, the Susquehanna secondary containment design failed, and still fails, to satisfy applicable regulatory requirements because the spent fuel pool is located within secondary containment without appropriate measures to remove the decay heat following design bases events for which secondary containment integrity is required.

1.3 The second conclusion under Section 2.0 reports an extremely low probability range for a LOCA with an extended loss of fuel pool cooling, but does not provide any clues as to the basis for these numbers so as to allow an independent confirmation of their validity. In addition, these unsubstantiated numbers are stated to have been calculated for 'some plants'. Are these representative plants? Are these the limiting plants? Are these the plants with the safety-grade SFPC systems? Is the BWROG dismissing safety concerns with extremely severe consequences based on a limited sample size?

We once again state that the single most important probability in this debate is the probability that SSES and other operating nuclear power plants are in compliance with Federal regulations which protect the public health and safety. That probability must be 1.0, unless explicitly justified by the licensee and approved by the NRC, and should override all other probability discussions. Within this industry, we formally double check calculations, independently verify valve lineups, receipt inspect all safety related components, and continually test safety systems. We do not do all of this, and much more, to create extra jobs or promote teamwork - we do it to guarantee the design and operation of the nuclear power plants in this country satisfy the high standards we have established to ensure nuclear safety. We cannot understand why these high standards are not being applied in this case in favor of significantly lower standards based more on 'what is there' instead of 'what is required.'

1.4 The third conclusion under Section 2.0 fails to address the common mode failure identified in the SSES design which prevents both restoration of fuel pool cooling and makeup to a boiling spent fuel pool - the presence of high radiation levels in the reactor building. The LOCA with postulated Reg Guide 1.3 source terms restrict, if not prohibit, access to the reactor building for necessary manual actions. In addition, this BWROG statement indicates that 'most BWRs' have the capability for backup cooling. What about the plants without such capability?

1.5 Section 3.1 states that "*Design and analysis of a LOCA coincident with the loss of normal spent fuel pool cooling and makeup are not required by the regulations.*" As we pointed out in our presentation to the NRC on October 1, 1994, reiterated in the NRC presentation on March 14, 1994, and include in virtually every submittal to the NRC on this issue, regulations require that all mechanistic consequences of a DBA LOCA be included in the evaluation of the plant's response to this postulated event. At SSES, a loss of fuel pool cooling is directly caused by the design on a LOCA signal. At SSES, a loss of fuel pool cooling is directly caused by emergency procedures following a LOCA. Therefore, to take the position that a loss of spent fuel pool cooling does not have to be analyzed concurrent with a DBA LOCA is contrary to the regulations and is basically irresponsible.

1.6 Section 3.2 states that the design deficiencies identified in the 10CFR21 report are not applicable to plants with a safety-grade SFPC system. At this time, eighteen months after the 10CFR21 report was submitted, does the NRC know how many of the 37 operating BWR units have safety-related fuel pool cooling systems? We suspect that the number of BWRs with safety-grade SFPC Systems is a small fraction of the total.

1.7 Under GDC 44/GDC 61/GDC 63, in the first paragraph it states, "Adequate makeup water is available from one or more of several sources in case of failure of the normal makeup water system." This is true only if those sources can be accessed and operated from outside the secondary containment (the reactor building). In most if not all BWRs, the radiation inside the reactor building for the design basis LOCA event, assuming the required Reg. Guide 1.3 source terms, prohibits operator entry. Therefore, this is not a true statement. This paragraph also states that pool water level instrumentation is available. This instrumentation generally is not qualified for the environmental conditions generated by the boiling spent fuel pool. Therefore, this statement also is not true.

1.8 In the third paragraph under GDC 44/GDC 61/GDC 63 it states that most BWRs also have the capability for backup cooling using the RHR system. This also is not a true statement in that this capability is not necessarily available in a LOCA event due to a variety of reasons including insufficient NPSH availability, insufficient ultimate heat sink capacity, inability of the RHR system to meet the single failure criteria as required when in this mode, and as with fuel pool makeup, the inability for operators to access the system components in the reactor building due to extremely high radiation levels.

1.9 The fourth paragraph under GDC 44/GDC 61/GDC 63 on page 5 of the attachment focuses on the SFPC system and neglects the regulations applicable to secondary containment. GDC 4 requires equipment performing safety functions to be designed for the conditions in which they must operate. GDC 44 requires heat from safety related structures to be transferred by safety related systems to the ultimate heat sink, including provisions for loss of offsite power and a single failure. The secondary containment structure is clearly designed to mitigate a LOCA. The secondary containment structure is clearly exposed to the temperature, pressure, humidity and radiation conditions resulting from a LOCA. Therefore, all systems and components located within the secondary containment structure which must operate to fulfill a safety function following a LOCA must also be designed for the temperature, pressure, humidity and radiation conditions resulting from a LOCA. It is the regulations and requirements for secondary containment at SSES which are violated by the fuel pool cooling design deficiencies - not the regulations and requirements for the fuel pool cooling system.

By narrowing its scope to the application of regulations to just the SFPC System, the BWROG is taking the well-travelled path which has resulted in these design deficiencies being overlooked numerous times in the past. All of the regulations must be considered collectively and applied to the entire plant. We recognize that the NRC, judging by its recent Requests for Additional Information to PP&L, is not following this path at this point.

1.10 The fifth paragraph under GDC 44/GDC 61/GDC 63 on page 5 of the attachment states that "*pertinent regulations and their application were determined during each plant's licensing process and establish that plant's licensing basis.*" At SSES in 1988/1989, emergency procedures were revised to implement a manual load shed of all non-Class 1E equipment in the reactor building twenty four hours after a LOCA in which offsite power was available. The reason for this procedure change was to ensure that Environmental Qualification (EQ) temperatures for safety related equipment inside the reactor building were not exceeded.

The original EQ temperatures were based on a Bechtel calculation for a LOCA/LOOP event. PP&L subsequently determined that this calculation was not a bounding case because it neglected the heat loads generated by the operating non-Class 1E equipment when offsite power was available. The SSES licensing basis for this non-safety related equipment does not require this equipment to operate following a LOCA. Regulations do not require this equipment to operate following a LOCA. But in the LOCA without LOOP event, this non-Class 1E equipment does operate and therefore rendered the original EQ calculations non-conservative, even though the licensing basis and regulations for the non-Class 1E equipment itself were not violated.

The decay heat in the SSES spent fuel pool is present before, during and following a LOCA event. Regulations and the SSES licensing basis require that this decay heat be handled before, during and following a LOCA event - not only because heat must be removed from the spent fuel in order to maintain its integrity, but also because of the safety related secondary containment structure and the safety related components within this structure. It is totally improper to focus on the Fuel Pool Cooling System licensing basis as a vehicle for excluding the spent fuel decay heat, just as it would have been improper to focus on the licensing basis of the non-Class 1E equipment as a vehicle for excluding the heat generated by this equipment from the EQ basis for safety related equipment in the reactor building.

1.11 The fifth paragraph under Section 3.5 on page 6 of the attachment reports "*that the longest power outage was approximately nine hours*" in dismissing the LOCA/LOOP event. How long was the offsite power outage at the Turkey Point plant following Hurricane Andrew? Four days?

1.12 The sixth paragraph under Section 3.5 on page 6 of the attachment states "*that the current regulatory guidance is sufficient and correct, and that changes to the regulations will not increase the current margins of safety and therefore are not needed in response to this issue.*" We agree wholeheartedly. We are not advocating new regulations - simply the enforcement of the existing regulations. If the NRC is not going to enforce the existing regulations to protect the public health and safety, the issuance of new regulations which may also not be enforced seems wasteful.

1.13 The discussion for GDC 4 on page 7 in Appendix A is improperly limited to external missiles affecting the SFPC System. GDC 4 also requires structures, systems and components performing safety functions to be designed for the conditions in which they must operate. The Fuel Pool Cooling and Service Water Systems at SSES are not designed for the post-LOCA environment.

1.14 We hasten to point out, in response to the discussion of GDC 63 on page 8 in Appendix A, that the SSES design at the time of the 10CFR21 report did not provide indication of spent fuel pool temperature and level except locally in the reactor building. The control room operators had absolutely no effective method of monitoring fuel pool temperature and level.

2.0 <u>Comments on Enclosure 2, Letter from R. G. Byram, PP&L to C. L. Miller, NRC, "*Follow-up Response to Request for Additional Information Concerning Standby Gas Treatment System*", PLA-4128 dated April 29, 1994</u>

2.1 In the last sentence on page 2, PP&L indicates that fuel pool cooling can be restored in a seismic event through use of the RHR System in the Fuel Pool Cooling (FPC) Assist mode. Prior to 1989, PP&L removed the RHR manual valves necessary to align the system in FPC Assist mode from their Inservice Inspection (ISI) program on the basis that these valves provide no safety function and that the ESW system providing makeup to a boiling fuel pool performs the safety function [1]. <u>AFTER</u> we brought this deletion to the attention of the NRC following the 10CFR21 report, PP&L indicated they would reinstate ISI testing of these valves. Unless the role of the RHR FPC Assist mode for the seismic event is explicitly incorporated into the SSES licensing basis, there is no legal requirement to include (i.e., RETAIN) these valves within the ISI program.

2.2 The first sentence on page 3 is misleading in that the RHR FPC Assist mode is a non-safety related function. This mode of operation does utilize some safety-related equipment and the entire mode uses Seismic Category I equipment, but not all of the equipment is safety related. The RHR FPC Assist mode does not satisfy single failure criteria. The RHR FPC Assist mode is not tested as a safety related system.

3.0 <u>Comments on Enclosure 3, Letter from R. G. Byram, PP&L to C. L. Miller, NRC, "*Additional Information in Response to 3/7/94 NRC Request Regarding Evaluation of SGTS Under Seismic Conditions*", PLA-4133 dated May 4, 1994</u>

3.1 Conclusion 2.3 on page 2 of the Attachment reports that the Standby Gas Treatment System is effectively rendered inoperable 36 hours after the start of fuel pool boiling. Although this specific evaluation was performed for the seismic event, this conclusion appears to be valid for any event which culminates in both fuel pools boiling. This conclusion may be true, albeit with a longer time frame from boiling, for any event which culminates in a single fuel pool boiling. Therefore, any SSES design bases event which relies on Standby Gas Treatment System operation to mitigate the consequences of the event must implicitly require that the fuel pools not be permitted to boil, unless the event duration is less than the time required to achieve fuel pool boiling and to fail the SGTS. This must be explicitly incorporated into the SSES licensing basis to ensure that the capability to restore fuel pool cooling is established and retained.

3.2 Conclusion 2.4 on Page 2 of the Attachment reports "that the resulting temperature of the SGTS room would be within the maximum normal operating temperature of the SGTS room *at the time in which the recirculation plenum reaches the analyzed limit*." (italics added for emphasis). Does this mean that the SGTS room temperature is not also a limiting result of this event, but at a later time? This statement only says that the water accumulation limit comes first. It does not say that temperature in the SGTS room is not also a later limiting result of the event. Considering the temperature of the fluid passing through the system (see Result 5.2.1 on Page 13 of the Attachment), it is very likely that it is. If it is, then this constitutes yet another mode of failure for the system. PP&L should be asked to make a clear, complete statement regarding this concern.

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[1] PP&L Engineering Work Requests MIS 86-0637 dated February 10, 1987 and MIS 85-0740 dated February 24, 1987

3.3 Section 5.2.3 on Page 13 of the Attachment reports that the maximum temperature of the refueling floor is 197.63°F for the seismic event which produces boiling of the fuel pools, and that this reduces the qualified life for the safety-related pressure transmitters PDT07554A1, A3, B1, and B3 from 100 days to 30 days. It also reports that for the LOCA/LOOP event (without boiling of the pools), the temperature is 171°F. For the event which is our primary concern, boiling of the pools *as a result of a LOCA event*, the temperature would be the result of the combined effects of pool boiling and the LOCA, which would undoubtedly produce temperatures somewhat higher than for the pool boiling event alone. The qualified life of the pressure transmitters of concern would be correspondingly reduced. Since these transmitters are required for the operation of the safety-related SGTS system, this poses yet another potential mode of failure for this system.

3.4 Section 5.3 on Page 14 of the Attachment describes the ratings of the six fire dampers as 285°F. Please remember that the rating of these dampers at the time of the 10CFR21 report and until August 1993 was not 285°F but merely ≈165°F. The table contained in Section 5.2.1 on page 13 of the Attachment indicates that the original fire damper rating would have been exceeded prior to the water accumulation limit. Please also recall that the PP&L Safety Evaluation performed per 10CFR50.59 for the replacement of the fusible links for these dampers concluded that the SGTS was designed for the conditions resulting from fuel pool boiling and that no new failure mode would be introduced. That conclusion was incorrect. PP&L failed to satisfy the requirements of 10CFR50.59 by not obtaining NRC approval for this Unreviewed Safety Question.

3.5 Section 5.6 on Page 15 of the Attachment again states that the temperature of the SGTS room will not reach its limit, 104°F, before the plenum flooding limit occurs. Might it still be a secondary limiting parameter as discussed in 3.2 above?

3.6 The evaluations performed by PP&L on the Standby Gas Treatment System performance in a fuel pool boiling event are silent with respect to the ability of the system to maintain the required 0.25" W.G. vacuum within secondary containment. The boiling spent fuel pool introduces a pressurizing effect which is essentially equivalent to greater inleakage with respect to SGTS operation. The system can probably handle it, but this pressurization effect should be evaluated.

3.7 Although this analysis demonstrates that the SGTS system will fail, even for the very narrow interpretation of the licensing basis that has been argued by both PP&L and the NRC, i.e., fuel pool boiling as a result of a seismic event only, it is no surprise that the 10CFR100 limits are not exceeded, even with this failure. We have recognized from the beginning that this event was of relatively small concern compared to the event addressed by our 10CFR21 report, the boiling spent fuel pools as a result of a LOCA or a LOCA/LOOP, which is also within the licensing basis for this plant.

4.0 <u>Comments on Enclosure 4, Letter from R. G. Byram, PP&L to C. L. Miller, NRC, "*Response to Request for Additional Information Concerning Loss of Spent Fuel Pool Cooling*", PLA-4134 dated May 5, 1994</u>

4.1 The response on page 5 of 25 to NRC question 1b includes an evaluation of the skimmer surge tank level as a function of time. This response states that "*the low level pump trip for NPSH pump protection occurs at the 11% level*". It should be noted that this setpoint was established based on normal system operation with the fuel pool bulk temperature $\leq 125°F$ [2]. The skimmer surge tank level required for adequate NPSH at 177°F will be greater than 11%. This response is misleading in that it implies that operators have 82 hours to restore the service water system when adequate NPSH will be lost in less than 82 hours.

4.2 In the response to NRC question 2, PP&L states that emergency procedure EP-PS-102 "*will be revised to reflect*" the manual actions necessary to prevent fuel pool boiling from adversely impacting the performance of safety related equipment in the reactor building. Please remember that the concern about the environmental consequences from fuel pool boiling were initially identified by a PP&L engineer in 1983 [3] and documented again in March 1992, yet the SSES procedures still do not fully address these necessary actions. The reason stated this time by PP&L for the delay in implementing this procedure change is the need to complete a COTTAP analysis.

4.3 In the response to NRC question 3, PP&L states that airborne radiation doses were not addressed in their determination of operator doses for manual ESW makeup actions for consistency with the results reported in FSAR Chapter 18. Please recall what else PP&L has said on this subject:

> "*Our documented FSAR shielding study, which is based on a Reg Guide 1.3 source term, conservatively concludes that the resulting contained source would severely limit access. Thus, the accident unit's Reactor Building was considered inaccessible. No airborne calculation was included in this shielding study since this would only add to the conclusion of inaccessibility for the accident unit's Reactor Building.*" [4]

Last year, PP&L said their FSAR Chapter 18 analysis neglected airborne radiation since the contained sources alone resulted in the reactor building being considered inaccessible. This year, PP&L is neglecting airborne radiation for alleged consistency with this earlier analysis, but the new results allow access into the reactor building. Consistency or not, the operator entering the reactor building will encounter airborne radiation. PP&L's own time-motion study featured the operator fully dressed out <u>and wearing SCBA</u>.

Neglecting airborne radiation for operator access calculations is in direct and flagrant violation of NUREG-0737 Item II.B.2, Reg Guide 1.3, and the NRC's own finding at SSES in 1984 by Mr. John White concerning the location of sample panels on the refueling floor.

This position by PP&L, which has been echoed by the NRC in meeting conversations, is based on the statement in Item II.B.2 that "Radiation from leakage *from systems outside of containment* (italics added) need not be considered for this analysis." This has been unjustifiably extrapolated to mean that leakage *from containment* need not be considered. Leakage *from systems outside of containment* is definitely <u>not</u> the same as leakage *from containment*!

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[2] PP&L Calculation SEA-JNPE-162 Rev.2, "*Fuel Pool Skimmer Surge Tank Level Setpoint Evaluation*", *Bechtel Calculation 153-008 Rev. 1, "Fuel Pool Skimmer Surge Tank 1T208 and 2T208"*, and FSAR 9.1.3.2

[3] PP&L Engineering Work Request 830658 dated March 30, 1983

[4] Letter PLA-3978 from R. G. Byram, PP&L, to C. L. Miller, NRC, "*Request for Additional Information on the Effects of a Loss of Spent Fuel Pool Cooling Event Following a Loss of Coolant Accident*", May 24, 1993

Leakage from systems outside of containment means leakage from systems external to the containment, such as the RHR and core spray systems which are filled with water and operating post-LOCA. This interpretation is borne out by another statement in Item II.B.2, Paragraph (2), third sentence, "Leakage measurement and reduction is treated under Item III.D.1.1, 'Integrity of Systems Outside Containment Likely to Contain Radioactive Material for PWRs and BWRs'" This section of the NUREG confirms our interpretation of "systems outside of containment" and lists RHR as one of the example systems. It also describes the required program to reduce the leakage of these systems. Therefore, it is logical to assume for the accident that they do not leak.

But it is not logical to extrapolate this assumption to the containment, which is known to leak. Leakage from containment definitely <u>is</u> required to be considered by Item II.B.2, as indicated by the first sentence in the item which states that the post-accident release of radioactivity should be assumed to be as described in Regulatory Guide 1.3. Regulatory Guide 1.3 requires that "the primary containment should be assumed to leak at the leak rate incorporated in the technical specifications for the duration of the accident." Such leakage would very definitely create high airborne radiation in the reactor building. To neglect airborne radiation is not only a violation of this requirement, it is ludicrous.

When airborne radiation is considered, operator exposure to perform the actions required for fuel pool makeup would be significantly higher than the 7.27 Rem dose calculated by PP&L which did not considering airborne radiation and which already exceeds the 5 Rem operator dose allowable.

4.4 On page 2 of the cover letter, PP&L states that "*Emergency Service Water valves that support make-up to the Spent Fuel Pool have been determined to be accessible from the affected unit (i.e. < 5 Rem) even if a Regulatory Guide 1.3 source term is assumed.*" On page 8 of 25 in the Attachment to this letter, PP&L reports the operator mission dose to be 7.27 Rem at 24 hours post-LOCA and 4.8 Rem at 40 hours post-LOCA. PP&L states that "*if conditions warrant, operator access to provide ESW make-up to the spent fuel pool should be delayed until at least until 40 hours.*" The skimmer surge tank level tables provided on pages 5 and 11 do not indicate any data points between 24 hours and 82 hours following loss of spent fuel pool cooling. As identified in comment 4.1 above, adequate NPSH is not assured for 82 hours. Therefore, if ESW makeup is to be delayed beyond 24 hours (since doses at 24 hours would exceed 5 Rem), an evaluation should be performed to provide an engineering basis for the maximum delay. It is less than 82 hours.

4.5 The last paragraph on page 8 of 25 refers to the flexibility in the SSES design which permits makeup to one spent fuel pool to eventually cascade over into the other spent fuel pool via the cask storage pit. PP&L has referred to this feature many times. This feature will perform as advertised if and only if the drains in the cask storage pit are closed and intact. Since these drains are not required to be tested, monitored, verified or even closed once the gates between the fuel pools and cask storage pit are installed, does PP&L's continued reliance on this feature constitute a commitment to maintain these drains closed and to implement administrative controls necessary to ensure their leak tightness? Additionally, has PP&L ever performed any testing which verifies that their multiple spillover mode of supplying water from one pool to the other will actually work?

4.6 The response on page 11 of 25 to NRC question 3 includes an evaluation of the skimmer surge tank level as a function of time. This response states that "*the low level pump trip for NPSH pump protection occurs at the 11% water level.*" It should be noted that this setpoint was established based on normal system operation with the fuel pool bulk temperature $\leq$ 125°F.

The skimmer surge tank level required for adequate NPSH at 177°F will be greater than 11%. This response is misleading in that it states that operators have 82 hours to restore the service water system when adequate NPSH will be lost in less than 82 hours.

4.7 With respect to NRC question 4 and its response, we wish to clarify that this scenario (convection through the cask storage pit allowing one unit's fuel pool to be cooled by the adjacent unit's Fuel Pool Cooling System) requires that the gates between both fuel pools and the cask storage pit be removed. This scenario is not a perturbation of the cask storage pit cascade feature discussed in comment 4.5 above.

4.8 The response to NRC question 4 refers to test procedure TP-135(235)-011. It is our understanding that the system alignment during this test is for the Fuel Pool Cooling System with the operating Service Water System to take suction from the operating unit's fuel pool skimmer surge tank and to return the cooled discharge flow to the outage unit's fuel pool, promoting the cross-flow through the cask storage pit [5]. This may be an incorrect impression, but we request that the NRC determine the actual system configuration for this test since it can significantly affect the results.

4.9 The response to NRC question 5 still does not make a definitive statement to the effect that the watertight doors for which they take credit in their flooding analysis have been either designed or tested to prevent leakage in the unseating direction. Their response is obviously intended to indicate otherwise, it still appears that they have inferred this from various sources.

4.10 The second response to NRC question 6 again states that the SGTS room temperature will not exceed the 104°F before the SGTS capability is lost due to condensation in the recirculation plenum. However, as discussed in our Items 3.2 and 3.5 above, this does not mean that room temperature is not a design limit that is violated, just that another limit is violated first.

4.11 The response to NRC question 7 reports that a seismic event with a single failure with both units at power *"will lead to one of the two fuel pools boiling or a unit being unable to be placed into shutdown cooling."* We are having trouble understanding how a postulated single failure (one of seven single failure opportunities in the RHR Fuel Pool Cooling assist subsystem or failure of one diesel generator) in conjunction with a seismic event results in probabilities less than 10^{-6} such that this event can be summarily dismissed.

4.11 The response to NRC question 7 reports that a seismic event with a single failure with both units at power and the fuel pools crosstied will not result in fuel pool boiling or inability to achieve safe shutdown of both units. Since the response to design bases events is enhanced if the fuel pools are crosstied, why is PP&L reluctant to operate with the pools permanently crosstied, and why then is the scenario they describe applicable if they are not in fact operating this way?

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[5] Appendix C to Nuclear Safety Assurance Group Report 4-90 Rev. 0, *"Outage Planning Information"*, October 17, 1990, Section 3.12 of PP&L Operating Procedure OP-135(235)-001, *"Fuel Pool Cooling System"*, and Section 6.2.A of PP&L Engineering Report SEA-ME-281 Rev. 1, *"Power Uprate Impact Review - Fuel Pool Cooling and Cleanup System"*, April 27, 1992