PDR

November 27, 1992

Mr. Thomas T. Martin
Regional Administrator, Region I
United States Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406-1415

SUBJECT: SUSQUEHANNA STEAM ELECTRIC STATION
DOCKET NO. 50-387
LICENSE NO. NPF-14
10CFR21 REPORT OF SUBSTANTIAL SAFETY HAZARD

Dear Mr. Martin:

Pursuant to the requirements of 10CFR21, Reporting of Defects and Noncompliance, this letter is submitted to report a "substantial safety hazard" that exists in the design of the Susquehanna Steam Electric Station (SSES) located near Berwick, Pennsylvania. This report is being made by Mr. David A. Lochbaum who, through July of this year, worked as a contract engineer in Pennsylvania Power & Light Company's (the licensee) Nuclear Plant Engineering Section, and Mr. Donald C. Prevatte who is currently, and until the end of this year, working as a contract engineer in PP&L's Nuclear Plant Engineering Section.

The substantial safety hazard is as follows: The SSES design for a loss of normal spent fuel pool cooling fails to meet numerous regulatory requirements. As a result, for a design basis accident, there is the potential for meltdown of irradiated fuel outside primary containment and the failure of all safety-related systems in the reactor building.

For an operating plant, 10CFR50.72 requires licensees to report in one hour any instance of the plant (a) being in an unanalyzed condition that significantly compromises plant safety, (b) in a condition that is outside the design basis of the plant, or (c) in a condition not covered by the plant's operating and emergency procedures. It also requires that reports shall be made within four hours of any condition that alone could have prevented the fulfillment of the safety function of structures or systems needed to (a) shut down the reactor and maintain safe shutdown, (b) remove residual heat, (c) control radioactive release, or (d) mitigate the accident. All of these conditions exist at SSES for the design basis accident (DBA) loss-of-coolant accident (LOCA) or LOCA with a loss-of-offsite-power (LOOP) as a result of the heatup of the spent fuel pool which mechanistically follows these accidents.

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On October 9, 1992, after seven months of attempts to convince PP&L's management to address these concerns as required, the signatories to this letter declared to PP&L management our intent to report these concerns to the NRC ourselves unless they were properly handled by PP&L. In response to our declaration and the actions it initiated, Pennsylvania Power & Light Company submitted Licensee Event Report (LER) 92-016-00 to the Nuclear Regulatory Commission on November 17, 1992. Although PP&L's report acknowledged that concerns had been raised, it dismissed them as having minimal safety significance. The LER is incomplete, inaccurate, unbalanced and misleading in its presentation of our concerns, the pertinent technical and licensing information, and its conclusions. The purpose of this letter is to inform the NRC that we still consider these concerns to be a "substantial safety hazard" which should have been reported by PP&L under 10CFR50.72.

The focus of our concerns is the inability to remove decay heat from the spent fuel pools for the various design events which mechanistically incapacitate the normal fuel pool cooling system and the resultant effects from loss of normal cooling on the safety-related systems and components in the reactor buildings.

The heart of the PP&L position stated in LER 92-016-00 is a legalistic argument that the licensing basis of SSES does not require the loss of normal fuel pool cooling to be considered concurrently with other design basis events such as LOCA or LOCA/LOOP. We agree that loss of normal spent fuel pool cooling is not required to be postulated concurrently, but when it follows mechanistically as a result of the design basis events as it does at SSES, it must be considered.

PP&L cites in the LER as support for its position FSAR Section 9.1.3 and Appendix 9A which it contends contain the only design basis requirements for the fuel pool cooling failure which must be considered - basically, failure due to a seismic event. We contend that there are other conditions within the SSES licensing basis as described throughout the FSAR which will mechanistically cause failure of the non-safety related normal fuel pool cooling system, such as hydrodynamic loads associated with a LOCA, environmental conditions associated with a LOCA, LOOP, failure of the non-safety related service water system, and random, single failures. In 1988, PP&L introduced another failure mode when it implemented procedures to manually de-energize non-1E loads in the reactor building following a LOCA without a LOOP.

We contend that as with all other systems described in the FSAR, the design and operation of the fuel pool cooling system cannot be taken out of context of its mechanistic relationships with the other systems, events, and licensing bases without review and approval by the NRC. The current design and operation of SSES for a loss of normal spent fuel pool cooling, even for failure

due to the seismic event indicated by PP&L to be within the licensing basis, clearly do not meet the design or licensing basis requirements if the effects on safety related structures, systems and components in the reactor buildings are considered.

The design basis accident for SSES is the LOCA with a concurrent LOOP. For this event, it must be assumed that the normal fuel pool cooling system will fail as described above. Therefore, the removal of decay heat from the spent fuel pools must be accomplished by the design basis method (and only safety related method available) described in Section 9.1.3 of the FSAR; allowing the fuel pool to boil and providing makeup from the safety related emergency service water (ESW) system. However, at the present time there are no design provisions, analyses or procedures which adequately define how, within the applicable regulatory requirements, this function will be accomplished. The effects of the boiling spent fuel pools on safety related equipment in the reactor buildings are also unanalyzed. These deficiencies exist even for the loss of spent fuel pool cooling event described in FSAR Appendix 9A.

When these concerns are addressed within the context of the regulatory requirements, it appears that the necessary steps to provide makeup water from the ESW system following a design basis LOCA cannot be performed due to very high radiation in the areas where valves must be manually operated. Additionally, the current EQ temperature limits of virtually all of the safety related equipment in the reactor building will probably be exceeded by a large margin due to the heat and moisture put into the reactor building atmosphere by the boiling spent fuel pools.

To appreciate the significance of these concerns, the magnitude of the potential effects for the design basis accidents must be considered:

1) The currently calculated radiation levels at some of the ESW valves which must be manipulated are in the thousands of R/hour, not including the associated airborne dose which may be in the hundreds of R/hr.

2) The boiling fuel pools will add approximately 20 million BTU/hr of latent heat to the reactor building atmosphere which is not currently accounted for in the calculations. The total heat load in the reactor building that is currently accounted for is only 5.2 million BTU/hr, and even at this heat load there are a number of areas where the accident temperatures slightly exceed the EQ temperatures.

3) At the design makeup rate from ESW to the fuel pools, 5.2 million gallons of water are introduced into the reactor buildings (both the accident and non-accident units will be affected) either through evaporation/condensation or

spillover of the pools. None of the possible detrimental effects of this water on the safety related structures, systems and components in the reactor buildings have been analyzed. In fact, PP&L's own recent engineering evaluation for these concerns determined that the standby gas treatment system would isolate due to high inlet temperature.

In addition to these most significant safety concerns, there are also related concerns of lesser safety significance which nonetheless constitute "substantial safety hazards". These include the following:

4) Fuel pool instrumentation for monitoring the cooling of the fuel pool post-LOCA (a safety related function) is not environmentally qualified, and the readouts are located in an area which is not accessible to the operators post-LOCA.

5) The design heat loads and the calculated times to boil for the spent fuel pools have not been updated to reflect changes that have been made in the fuel design, fuel cycle length, and refueling procedures.

Following our October 9, 1992 declaration of intent to report to the NRC on these concerns, there ensued large scale efforts within the PP&L Nuclear Department to analyze the concerns and define the actions needed to be taken. This activity produced an engineering report, NE-92-002 (attached). This report described extensive modifications and procedure changes required for SSES to cope with a loss of normal fuel pool cooling event.

Although the report addressed many of our concerns, it did not adequately address all of them, and some of the proposed solutions are either technically inadequate and/or they do not meet regulatory requirements. In general, the proposed solutions are not acceptable for the following reasons:

1) They place heavy reliance on non-safety related equipment and functions.

2) They place heavy reliance on plant modifications which have not yet been implemented or even designed.

3) They place heavy reliance on procedure changes which have not yet been made.

4) They place heavy reliance on analyses which have not yet been performed.

5) They place heavy reliance on operator and EOF personnel training which has not yet been developed or performed.

6) They place heavy reliance on operator actions following a LOCA when there is already heavy dependence on operator actions and monitoring, and these additional actions must be performed under extremely adverse environmental conditions.

7) They rely on assessments of operator accessibility to the reactor building which in turn are based on assumptions of core damage which are unreviewed by the NRC and are substantially less than the assumptions required by NUREG-0737 and the SSES licensing basis reflected in Chapter 18 of the FSAR. Additionally, the accessibility position taken in the report with respect to airborne radioactivity contributions is inconsistent with the requirements in NUREG-0737, 10CFR50 Appendix J, actual SSES Appendix J test results, and the design of other plant systems (e.g. secondary containment and the standby gas treatment system). For NRC mandated DBA conditions, as stated in FSAR Chapter 18, the reactor building is inaccessible for days following a LOCA.

8) They rely on probability arguments which may be acceptable in an Individual Plant Evaluation and in a justification for interim operation, but which are not acceptable substitutes for compliance with regulatory requirements, unless they are reviewed and approved by the NRC. These have not been.

9) In some areas, the report's conclusions are inconsistent with the facts presented. For example, the report concluded that Zone III venting is acceptable, whereas the supporting documentation indicates that the 10CFR100 and 10CFR50 Appendix A Criterion 19 allowables for offsite and control room doses respectively are exceeded.

It should also be considered that the conclusions in this report represent PP&L's vision of systems, equipment and procedures in the future, not as they exist today. Although essentially none of the technical information from the report is contained in their LER, this information along with their legalistic arguments discussed earlier, has provided the underlying bases for their determinations of operability and reportability. But the law requires determinations of operability and reportability to be based on the plant conditions as they exist at the time of discovery as discussed in considerable detail in NUREG-1022.

In addition to these technical concerns, we also must point out the conditions adverse to quality that PP&L's handling of this case (and other recent safety concerns) demonstrates in violation of 10CFR50 Appendix B. Since our concerns were first discovered and reported in March of this year, there has been a programmatic failure by PP&L to properly evaluate these concerns. PP&L repeatedly attempted to improperly dismiss these concerns or indefinitely defer their evaluation by methods including

classifying them as design basis document issues (in clear violation of the guidelines expressed in NUREG-1397), selectively applying regulatory requirements to permit favorable conclusions, claiming that the NRC had already reviewed and approved the design deficiencies based on the FSAR/SER text, and even claiming that an informal, undocumented agreement had been made with the NRC at the time of initial licensing of the plant. Our experience and our knowledge of the difficulties encountered by other engineers with nuclear safety concerns for SSES indicates that PP&L's program for handling nuclear safety issues is itself cause for concern.

While PP&L cites data to support their contention that their discrepancy management system is effective, most of their data points represent relatively minor discrepancies which are easy to resolve. However, for large problems with extensive or uncertain resolution such as in this case, the system lacks the ability to assure proper evaluation and subsequent implementation. In these cases, PP&L's treatment violates their own administrative procedures controlling discrepancy management.

In the nuclear power industry, organizations such as PP&L and individuals such as ourselves have legal and ethical responsibilities. PP&L has not fulfilled its responsibilities in this case and has forced us to fulfill ours by submitting this letter.

The more detailed technical descriptions for these concerns and the history of their treatment are contained in numerous letters, memos, and documents. A listing of pertinent documents is contained in Attachment 1 to this letter, with copies of these documents provided as the remaining attachments to this letter.

We expect that you may require additional information from us regarding this matter. We will make every effort to support your requirements in a timely manner. We can be reached at the addresses and telephone numbers listed below. We would also greatly appreciate being kept informed of your actions regarding this matter.

Thank you for your consideration.

Sincerely,

David A. Lochbaum

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(908) 754-3577

Donald C. Prevatte

7924 Woodsbluff Run
Fogelsville, PA 18051
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Attachment 1 List of Attachments

No.	Attachment
1	List of Attachments
2	PP&L Memo from Dave Lochbaum and Don Prevatte to Mark Mjaatvedt, "Susquehanna Steam Electric Station Spent Fuel Pool Boiling Issues", March 19, 1992 (ET-0149)
3	PP&L Engineering Discrepancy Report, "Loss of Spent Fuel Pool Cooling Event Design Discrepancies", Originated April 16, 1992 and Dispositioned October 6, 1992 (EDR G20020)
4	PP&L Operability Statement, "EDR #G20020 Loss of Spent Fuel Pool Cooling Event Design Discrepancies", April 23, 1992
5	PP&L Memo from Dave Lochbaum and Don Prevatte to Joe Zola, "Supplemental Information for EDR G20020 on Boiling Spent Fuel Pool", June 22, 1992 (ET-0471)
6	PP&L Draft Screening Worksheet prepared by Art White, "EDR No. G20020", July 1, 1992
7	Handout, "EDR G20020 References", July 15, 1992
8	White Paper prepared by David A. Lochbaum and Donald C. Prevatte, "Safety Consequences of a Boiling Spent Fuel Pool at the Susquehanna Steam Electric Station", July 27, 1992
9	PP&L Memo from G. D. Miller to G. T. Jones, "Fuel Pool Cooling Deficiencies", August 18, 1992 (ET-0586)
10	PP&L Memo from D. C. Prevatte to G. T. Jones, "Fuel Pool Cooling Deficiencies", August 20, 1992 (ET-0587)
11	PP&L Memo from A. Dyszel to T. C. Dalpiaz, "U2 RI05 Fuel Pool Decay Heat Evaluation", August 21, 1992 (PLI-72230)
12	PP&L Memo from J. M. Kenny to G. T. Jones and C. A. Myers, "EDR on Fuel Pool Cooling", August 25, 1992
13	PP&L Memo from George T. Jones to Glenn D. Miller, "Fuel Pool Cooling EDR's G20020, G00005", August 27, 1992 (PLI-72267)
14	PP&L Memo from Glenn D. Miller to George T. Jones, "Fuel Pool Cooling EDRs G20020, G00005", August 31, 1992 (PLI-72297)
15	PP&L Memo from Kevin W. Brinckman to George T. Jones, "Review of Fuel Pool Cooling", September 1, 1992 (PLI-72288)

Attachment 1 List of Attachments (continued)

No.	Attachment
16	PP&L Memo from J. R. Miltenberger to G. T. Jones, "Spent Fuel Pool Cooling", September 9, 1992 (PLI-72367)
17	PP&L Letter from James E. Agnew to David A. Lochbaum, "EDR G20020, Spent Fuel Pool Design Discrepancies", October 7, 1992 (ET-0785)
18	PP&L Memo from G. D. Miller to G. D. Miller, "Assignment of EDR", October 7, 1992 (ET-0780)
19	Letter from David A. Lochbaum and Donald C. Prevatte to George T. Jones, "Reportability of Boiling Spent Fuel Pool Concerns", October 9, 1992
20	PP&L Memo from D. A. Lochbaum and D. C. Prevatte to George T. Jones, "EDR System Concerns", October 13, 1992 (PLI-72365)
21	PP&L Memo from George T. Jones to G. D. Miller, "Spent Fuel Pool Issue", October 14, 1992 (PLI-72640)
22	PP&L Memo from George T. Jones to G. D. Miller, J. S. Stefanko and M. W. Simpson, "Spent Fuel Pool Cooling Issue", October 14, 1992 (PLI-72641)
23	PP&L Memo from George T. Jones to All Nuclear Engineering Managers and Supervisors, "Engineering Discrepancy (EDR) Program", October 14, 1992
24	Letter from David A. Lochbaum and Donald C. Prevatte to George T. Jones, "Disagreement with Screening, Reportability and Operability Evaluations for EDR G20020", October 14, 1992
25	Memo from Charles A. Myers to George T. Jones, "Fuel Pool Cooling Issues - Reportability / Operability", October 20, 1992
26	PP&L Memo from Glenn D. Miller to George T. Jones, "Evaluation of EDR G20020 - Spent Fuel Cooling Issue", October 21, 1992 (PLI-72711)
27	PP&L Memo from David A. Lochbaum and Donald C. Prevatte to George T. Jones, "Evaluation of EDR G20020 Reportability/Operability", October 26, 1992 (PLI-72739)
28	PP&L Memo from David A. Lochbaum and Donald C. Prevatte to George T. Jones, "Response to Evaluation of EDR G20020", October 28, 1992 (PLI-72751)

Attachment 1 List of Attachments (continued)

No.	Attachment
29	PP&L Memo from Glenn D. Miller to George T. Jones, "Evaluation of EDR G20020 - Spent Fuel Pool Cooling Issue", October 29, 1992 (PLI-72763)
30	PP&L Engineering Report, "Loss of Fuel Pool Cooling Event Evaluation for EDR #G20020", October 29, 1992 (NE-92-002 Rev. 0)
31	PP&L Memo from Glenn D. Miller to George T. Jones, "Revised Evaluation of EDR G20020 - Spent Fuel Pool Cooling Issue", October 29, 1992 (PLI-72764)
32	PP&L Memo from David A. Lochbaum and Donald C. Prevatte to George T. Jones, "Position on EDR G20020 and Planned Actions", November 2, 1992 (PLI-72783)
33	PP&L Memo from David G. Kostelnik and Mark R. Mjaatvedt to George T. Jones, "Comments on PLI-72783 Regarding EDR G20020", November 11, 1992 (PLI-72857)
34	PP&L Letter from H. G. Stanley to the U.S. Nuclear Regulatory Commission, "Licensee Event Report 92-016-00", November 17, 1992 (PLAS-546)
35	PP&L Safety Evaluation Summary, "Procedure EO-IP-055", 1988 (SER No. 88-127)

Attachment 2

PP&L Memo from Dave Lochbaum and Don Prevatte to Mark Mjaatvedt, "Susquehanna Steam Electric Station Spent Fuel Pool Boiling Issues", March 19, 1992 (ET-0149)

Note: This memo documents the discovery of the problems with the loss of normal spent fuel pool cooling event and the reporting of these problems to a supervisor in the PP&L Nuclear Plant Engineering Section. Approximately four weeks later, the authors of this memo were directed to initiate an Engineering Discrepancy Report on the concerns. PP&L's decision to generate an EDR on these concerns may have been driven by schedule interests - the authors, as preparer and technical reviewer of reactor building heat load calculations to support the PP&L Power Uprate Project, would not sign off on the calculations until these concerns were addressed. Upon generation of the EDR, the authors signed off the calculations conditionally with a note that the results might be affected by the disposition of the EDR. PP&L needed these calculations issued in order to submit their engineering report on power uprate to the NRC in June 1992.

PP&L

M E M O R A N D U M

TO: Mark Mjaatvedt DATE: March 19, 1992

FROM: Dave Lochbaum
Don Prevatte

JOB: Power Uprate NUMBER: ET-0149 COPIES: Distribution

FILE: P88-1 REPLY: No

SUBJECT: SUSQUEHANNA STEAM ELECTRIC STATION SPENT FUEL POOL BOILING ISSUES

Potential problems resulting from a boiling spent fuel pool have been uncovered during the preparation of the engineering evaluation of the fuel pool cooling system and the reactor building heat load calculations at power uprate conditions. These problems, their brief history and recommendations to resolve these issues are presented in the attachment to this letter.

The consequences of these problems will be made incrementally worse by power uprate and therefore must be addressed before implementation of power uprate. More importantly, however, these problems affect the current operation of SSES and must be evaluated and resolved as soon as possible.

If additional information is needed, please contact Dave Lochbaum at ETN 220-7768 or Don Prevatte at ETN 220-7781.

David A. Lochbaum

Donald C. Prevatte

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Take Care PP&L

Attachment : Boiling Spent Fuel Pool Issues

I. REACTOR BUILDING HEAT LOADS

Problem:

Reactor building design heat loads do not account for the boiling spent fuel pool event.

History:

The calcs of reactor building pre-uprate and uprate heat loads for Zone I, II and III under normal and accident conditions (calcs M-RAF-052, -053, -054) assume the spent fuel pool temperature remains at 125°F for all cases. This assumption relies upon use of the service water system to remove heat from the fuel pool heat exchangers post-LOCA and the fuel pool cooling assist mode of RHR to remove heat from the fuel pool post-LOOP. Neither of these operating modes is safety related and therefore may not be available.

The design provision for the loss of fuel pool cooling event is to permit the fuel pool to boil and use ESW to maintain the level in the pool above the top of the fuel. ESW provides redundant seismic Category I makeup lines to each of the two spent fuel pools.

If the spent fuel pool is permitted to boil, the heat loads in the reactor building, particularly in Zone III, increase significantly. These higher heat loads have not been considered in reactor building analyses to date. The equipment qualification of safety-related equipment in the reactor building may therefore be adversely affected if the heat loads from a boiling spent fuel pool are considered.

Recommendation:

An EDR was prepared on this condition. The options available to resolve this problem include:

1) Analyzing the reactor building heat loads for the boiling spent fuel pool case and update associated analyses for equipment qualification.

2) Providing design capability to maintain spent fuel pool temperature $\leq$ 125°F using safety-related equipment such that existing reactor building heat load analyses are adequate.

Attachment : Boiling Spent Fuel Pool Issues

II. FUEL POOL TIME-TO-BOIL AND RADIOLOGICAL RELEASE ANALYSES

First Problem:

The analytical 25 hour time-to-boil for the spent fuel pool is nonconservative for the maximum normal heat load in the spent fuel pool.

History:

Bechtel calc 200-0048 Rev. 1, "Boiling Spent Fuel Pool" dated May 7, 1982, determined time-to-boil using the equation:

$$\text{Time-to-Boil} = (m * C * \Delta T) / Q, \quad \text{where}$$

m = mass of water in fuel pool, lb

C = specific heat, BTU/lb-°F

ΔT = difference between final pool temperature (212°F) and initial pool temperature (125°F), °F

Q = fuel pool decay heat load, BTU/hr

This calc used a decay heat load of 9.79×10^6 BTU/hr for Unit 1 and 7.92×10^6 BTU/hr for Unit 2 to determine times-to-boil of 25.15 hours and 31.087 hours respectively.

FSAR 9.1.3.1 establishes the maximum normal heat load as that heat load resulting from 2840 assemblies discharged to the fuel pool by a routine refueling schedule. FSAR Tables 9.1-2a and 9.1-2b report the maximum normal heat load for Units 1 and 2 as 12.6×10^6 BTU/hr. These values were determined in Bechtel calc 153-9 Rev. 1, "Fuel Pool Decay Energy and Temperature".

The spent fuel pool decay heat values used in the boiling spent fuel pool calc and for the FSAR discussion were based upon assumptions for cycle operating lengths, fuel exposures, and reactor power level. SSES has subsequently operated differently than had been assumed such that the decay heat loads in the filled spent fuel pools may exceed 9.79×10^6 BTU/hr, resulting in a shorter than analyzed time-to-boil.

Calc NFE-B-NA-053 Rev. 0, "Decay Heat from a Full Spent Fuel Pool (ASB9-2 Method)", determined decay heat from a filled spent fuel pool using actual fuel operating history through 1991 and assumptions which bound operation after power uprate. This recent calc reported a maximum normal heat load of

≈17x10^6 BTU/hr. The methodology use in this calculation is conservative and may over predict actual decay heat loads by ≈20%.

Calc M-FPC-009 was drafted to determine the spent fuel pool time-to-boil and required ESW makeup rate for power uprate. Preliminary results from this calc indicate the fuel pool boils 19.4 hours after loss of fuel pool cooling for the design heat removal capacity of the fuel pool cooling system (13.2x10^6 BTU/hr). This calc determined a time-to-boil of ≈15.5 hours for the ≈17x10^6 BTU/hr heat load calculated for the power uprate case.

Recommendation:

The basis for the time-to-boil analysis should not be the maximum normal heat load, since this value is subject to assumptions of reactor operation which are extremely difficult to predict. Two options are proposed:

1) The time-to-boil analysis for the loss of normal spent fuel pool cooling case should use the design capacity of the fuel pool cooling system since this value bounds any normal heat load stored in the fuel pool. If the normal heat load in the fuel pool exceeded 13.2x10^6 BTU/hr, then modifications to the fuel pool cooling system would be necessary to enable the system to maintain pool temperature less than 125°F.

2) The time-to-boil analysis for the loss of normal spent fuel pool cooling case should use a range of spent fuel pool decay heat loads up to at least the design capacity of the fuel pool cooling system. This method bounds any maximum normal heat load for the fuel pool while limiting overly conservative times in the years while the fuel pool is partially filled. Basically, this method provides time-to-boil as a function of decay heat load in the spent fuel pool. This relation can be used for more realistic time-to-boil for current conditions if actual decay heat load in the spent fuel pool is known.

Attachment : Boiling Spent Fuel Pool Issues

Second Problem:

The analytical 25 hour time-to-boil for the spent fuel pool does not account for the emergency heat load in the spent fuel pool.

History:

FSAR 9.1.3.1 establishes the emergency heat load for the spent fuel pool as that heat load following a full core offload which completely fills the fuel pool. The FSAR specifies the emergency heat load to be 32.6×10^6 BTU/hr.

Bechtel calc 200-0048 Rev. 1, "Boiling Spent Fuel Pool" dated May 7, 1982, determined time-to-boil for the maximum normal heat load case only. The actual decay heat load in the spent fuel pool exceeds the maximum normal heat load during every refueling outage at SSES in which the core is fully offloaded.

SSES currently imposes administrative controls during refueling outages when the core is fully offloaded into the spent fuel pool to reduce the potential for loss of fuel pool cooling. Decay heat is removed from the spent fuel pool during these periods by RHR shutdown cooling (when the fuel pool to reactor cavity gates are removed) and by cross-tieing the operating unit's fuel pool cooling system to the outage unit's fuel pool. However, a seismic event in this configuration could cause loss of fuel pool cooling at a time when the time-to-boil is significantly less than 25 hours.

Calc M-FPC-009 was drafted to determine the spent fuel pool time-to-boil and required ESW makeup rate for power uprate. Preliminary results from this calc indicate the fuel pool boils 7.9 hours after loss of fuel pool cooling for a decay heat load of 36.2×10^6 BTU/hr, which is the currently analyzed emergency heat load.

Recommendation:

The time-to-boil analysis should be expanded to include decay heat loads up to at least the design capacity of the RHR fuel pool cooling assist mode. Operating procedures, off-normal procedures and SSES outage management policies should be reviewed and revised as necessary to ensure that appropriate controls are implemented when the fuel pool decay heat load exceeds the capacity of the fuel pool cooling system and proper responses are taken in event fuel pool cooling is lost.

Attachment : Boiling Spent Fuel Pool Issues

Third Problem:

The radiological release analysis for a boiling spent fuel pool uses nonconservative evaporation rates.

History:

Bechtel calc 200-0048 Rev. 1, "Boiling Spent Fuel Pool" dated May 7, 1982, determined the evaporation rate from a boiling spent fuel pool using the equation:

$$\text{Evap Rate} = Q / (h_g - h_f), \quad \text{where}$$

Q = fuel pool decay heat load, BTU/hr

h_g = enthalpy of vapor at boiling, BTU/lb

h_f = enthalpy of water at boiling, BTU/lb

This calc used a decay heat load of 9.79×10^6 BTU/hr to determine evaporation rate. As reported above, the maximum normal heat load specified in FSAR Table 9.1-2a is 12.6×10^6 BTU/hr and the emergency heat load specified in FSAR 9.1.3.1 is 32.6×10^6 BTU/hr. When the decay heat load in the spent fuel pool exceeds 9.79×10^6 BTU/hr, the evaporation rate from the boiling pool will exceed the rate assumed in the radiological release analysis.

SSES currently applies the 25 hour time-to-boil determined by calc 200-0048 as the criterion in deciding when to permit common RHR work during an outage. Therefore, when decay heat loads are less than 9.79×10^6 BTU/hr, the time-to-boil is longer than 25 hours and the radiological release in event of loss of fuel pool cooling is bounded by the results from calc 200-0048.

The 25 hour criterion for common RHR work prevents this work from beginning prior to ≈Day 18-21 each outage. Since core offloading typically starts on Day 5 and is completed by Day 10 or 11, this means that for at least 7 days, the decay heat load in the spent fuel pool is significantly higher than the heat load used to derive the evaporation rate used in the radiological release analysis. The consequences from a loss of fuel pool cooling may be offset by a longer time-to-boil if the fuel pool to reactor cavity gate is removed and the fuel pools are cross-tied, but credit for the additional water inventory available cannot be taken without administrative controls and a time-to-boil analysis for this configuration.

Attachment : Boiling Spent Fuel Pool Issues

Recommendation:

The radiological release analysis should use appropriate evaporation rates for the decay heat loads used in the associated time-to-boil analysis. The method and results from these analyses should be clearly stated and conveyed to SSES to ensure that adequate administrative controls are implemented during normal operation and in refueling to ensure the radiological release analysis results bound actual plant conditions for all operating configurations.

Fourth Problem:

The radiological release analysis for a boiling spent fuel pool uses nonconservative activity terms.

History:

Bechtel calc 200-0048 Rev. 1, "Boiling Spent Fuel Pool" dated May 7, 1982, determined the radiological release consequences from a boiling spent fuel pool. This calc assumed 12 month operating cycles and 184 bundle equilibrium reload sizes to determine the activity terms for failed fuel in the fuel pool. SSES currently has 18 month operating cycles with ≈230 bundle reloads which will increase to ≈254 bundles after power uprate. Since calc 200-0048 implies that most of the activity results from the most recent discharge batch, the effect of increasing the discharge size from 184 bundles assumed in the calc to 230 and 254 bundles would appear to be nonconservative with respect to the radiological release analysis.

Recommendation:

The basis for the radiological release analysis should not be the projected operating conditions, since these conditions are subject to assumptions which are extremely difficult to predict. The radiological release analysis should assume conditions which will bound future actual operating conditions. For example, a reload batch size of 320 bundles was assumed in calc NFE-B-NA-053, "Decay Heat from a Full Spent Fuel Pool (ASB9-2 Method)", because this size represents the maximum reload batch size possible under core design criteria.

Attachment : Boiling Spent Fuel Pool Issues

III. ESW MAKEUP TO THE SPENT FUEL POOL

First Problem:

The impact of the ESW makeup water to the spent fuel on equipment in the reactor building has not been evaluated.

History:

The ESW system and the ultimate heat sink are designed to provide adequate makeup to the spent fuel pool for 30 days following loss of normal spent fuel pool cooling. Based on the original ESW makeup flow of 60 gpm to each fuel pool, the spray pond inventory allocates 5 million gallons of water for this purpose. However, the consequences of this quantity of water on equipment in the reactor building has not been evaluated.

EDR G00005 was written in 1990 to address discrepancies between the spent fuel pool discussion in FSAR Chapter 9 and actual SSES operation. This EDR also questioned the ESW makeup flow to the spent fuel pool since the 60 gpm flow rate had not been demonstrated to be achievable.

Calc M-FPC-009, "Spent Fuel Pool Boiling Analysis", was drafted to determine the ESW makeup flow required for the design heat removal capacity of the fuel pool cooling system (13.2×10^6 BTU/hr) and for the heat removal capacity of the RHR fuel pool cooling mode (32.6×10^6 BTU/hr). These ESW makeup flows were determined to be 31.8 gpm and 67.5 gpm respectively. The interim disposition to EDR G00005 pointed out that the maximum ESW makeup flow case occurs when all of the reactor core is offloaded to the spent fuel pool, so the higher ESW makeup flow rate could be obtained by the reduced ESW system flow required when there is no fuel in the reactor.

When the ESW makeup flow to the fuel pool exactly matches the boil-off rate from the fuel pool, that quantity of water vapor must also either exit the building via the standby gas treatment system, bring the building to 100% humidity or condense somewhere within the reactor building. When the ESW makeup flow to the fuel pool exceeds the boil-off rate, there will also be overflow once the skimmer surge tank fills and level control is lost.

Eventually in either case, the quantity of water added via the ESW system ends up going through the standby gas treatment system or as water in the reactor building. The consequences of up to 2.5 million gallons of water in each reactor building

could include flooding of the ECCS pumps rooms, inoperability of the ECCS pump room coolers, inoperability of safety related equipment due to higher than analyzed humidity and degradation of the standby gas treatment system due to moisture loading.

Recommendation:

A comprehensive evaluation for the boiling spent fuel pool event needs to be performed which accounts for the water present in the reactor building due to boil-off and overflow from the spent fuel pool. This evaluation must address the effects of this water on the operability of systems and components in the reactor building.

Second Problem:

The manual valve manipulations required to provide ESW makeup flow to a boiling spent fuel pool may not be permitted under post-LOCA conditions.

History:

Off-normal operating procedure ON-135-001, "Loss of Fuel Pool Cooling/Coolant Inventory", requires the operator to manually open the valves in ESW makeup line to fuel pool if all other means of adding water to the fuel pool are lost. The procedure calls for the valves to be left open until the desired water level is obtained. Since these valves are in the reactor building, it may be impossible for them to be manually operated as directed under all conditions including post-LOCA. In addition, even if the throttle valve is initially adjusted so that the ESW makeup flow to the fuel pool exactly matches the boil-off rate, the subsequent exponential decline in fuel pool decay heat load would require the throttle valve to be periodically adjusted to reduce the ESW makeup flow unless the fuel pool is permitted to overflow.

Recommendation:

The required operation of the ESW makeup flow valves should be evaluated from the perspective of accessibility and usage over the entire 30 day period of the boiling spent fuel pool event to ensure that all necessary valve manipulations can be made.

Attachment : Boiling Spent Fuel Pool Issues

Third Problem:

The instrumentation available to the operator post-LOCA may not provide adequate indication of spent fuel pool temperature and level to allow proper response to a loss of fuel pool cooling event.

History:

Off-normal operating procedure ON-135-001, "Loss of Fuel Pool Cooling/Coolant Inventory", requires the operator to manually open the valves in ESW makeup line to fuel pool if all other means of adding water to the fuel pool are lost. The operator enters this procedure upon annunciation of low level in the spent fuel pool or high temperature in the fuel pool cooling system. Each spent fuel pool has temperature indication (TE-15333) and level indication (LT-15332). Each skimmer surge tank has level indication (LT-15312). The skimmer surge tank piping to the fuel pool heat exchangers has temperature indication (TE-15313) and each fuel pool heat exchanger outlet piping has temperature indication (TE-15316A,B,C).

The level and temperature instruments providing these alarms may not be qualified for all conditions, such as post-LOCA, in which they would be required to function. In addition, these instruments may not be powered from class 1E sources such that they would be available post-LOOP when the fuel pool heat exchangers would be without service water.

Recommendation:

The spent fuel pool temperature and level instrumentation, as a minimum, should be verified to be or made to be qualified for all reactor building environmental conditions and required accident conditions.

Attachment 3

PP&L Engineering Discrepancy Report, "Loss of Spent Fuel Pool Cooling Event Design Discrepancies", Originated April 16, 1992 and Dispositioned October 6, 1992 (EDR G20020)

RECEIVED APR 21 1992 Eng. Discrepancy Mgmt.

PP&L

SUSQUEHANNA STEAM ELECTRIC STATION
ENGINEERING DISCREPANCY REPORT
FILE: R42-15

1. EDR No. 6-20020
2. REV. No. 0
3. PAGE 1 of 6

DISCREPANCY ITEM/LOCATION/SUBJECT (Identification): Loss of Spent Fuel Pool Cooling Event Design Discrepancies

5.a PRIORITY: S
5.b IMPLEM. DUE DATE: 11/19/93 01-7 5/2/94 UL-6 Closed
6. SYSTEM No: 134/234 154/254
7. UNIT No: 1 & 2
8. CONTRACTOR/PI/SECTION: Power Uprate

POTENTIAL ENGINEERING DISCREPANCY (Identification): As described in Section 9.1.3 of the Safety Evaluation Report for SSES Units 1 & 2 (NUREG-0776), the design provision for the loss of spent fuel pool cooling event is to permit the fuel pool to boil and maintain its water level above the fuel through makeup from the emergency service water (ESW) system. This design provision is necessary because

ORIGINATOR: David A. Lochbaum / 4-16-92 / DAVID A. LOCHBAUM / 321
Full Signature / Date / Full Printed Name / Cost Area

10. (Validation):
SUPERVISOR: Mark R. Mjaatvedt / 4-20-92 / MARK R. MJAATVEDT — DISC. VALID YES X NO ___
Full Signature / Date / Full Printed Name

11. (Verification):
EDMG VERIFIER: G. D. Miller / 7/10/92 / G.D. Miller — DISC. VALID YES ✓ NO ___ ORIGINATOR NOTIFIED: ___
Full Signature / Date / Full Printed Name

12. INITIAL ASSESSMENT (Screening):
EDMG SUPERVISOR: J. Elmer / 10/6/92 — SAFETY CONCERN: YES* ___ NO* ✓
Full Signature / Date
LICENSING SUPERVISOR: N/A / ___ — TSAS: YES* ___ NO* ✓ PRESUMPTION OF OPERABILITY: ___
Full Signature / Date

13. REPORTABILITY/OPERABILITY DETERMINATION:
EDMG EVALUATOR: Joe Zola / 9/16/92 / JOE ZOLA
Full Signature / Date / Full Printed Name
EDMG SUPERVISOR: J. Elmer / 10/6/92
Full Signature / Date
LICENSING SUPERVISOR: N/A / ___
Full Signature / Date
TSAS: YES* ___ NO* ✓
REPORTABLE: YES* ___ NO* ✓
SOOR No: ___ N/A ✓
ORIGINATOR NOTIFIED: ✓

14. EVALUATION (Disposition):
See attached.

EDMG EVALUATOR: J. Elmer / 10/6/92 / J. EAGLEW
Full Signature / Date / Full Printed Name

15. DOCUMENTS GENERATED: ___

16. DISCREPANCY REVIEW COMPLETE: ___ / ___
SUPERVISING ENGINEER, ENGR PROJECTS — Full Signature / Date

FORM EPM-QA-122A, Rev. 3
Page 1 of 1

* Documentation of bases required. Use EDR Continuation Sheet.

9. Potential Engineering Discrepancy (continued)

the fuel pool cooling system used for normal operation and the RHR fuel pool cooling assist mode used for abnormal heat loads are not designed to satisfy seismic category I and single failure criteria. The following discrepancies for the loss of spent fuel pool cooling event were discovered during the system evaluations for power uprate:

A. Reactor building design heat loads do not account for the boiling spent fuel pool event. The current calculations for reactor building Zone I, II and III heat loads assume a spent fuel pool temperature of 125°F for all cases. The reactor building heat load analyses and attendant temperature analyses upon which equipment qualification environmental parameters are based do not account for the additional heat load from boiling spent fuel pool(s). The additional heat load could be as high as 26.4×10^6 BTU/hr compared to the current maximum reactor building heat load of 5.5×10^6 BTU/hr (Unit 1 LOCA case). Therefore, the design environmental conditions of safety related equipment in the reactor building may be exceeded if the heat load from the boiling spent fuel pool(s) is considered (See Note 1 below). MKM 4/20

B. The impact of the ESW makeup water to the spent fuel pool on equipment in the reactor building has not been evaluated. The ESW system and the ultimate heat sink are designed to provide adequate makeup to the spent fuel pool for 30 days following loss of normal spent fuel pool cooling. Based on the original design ESW makeup flow of 60 gpm to each fuel pool, 5 million gallons of the spray pond inventory is allocated for this purpose. The water added to the spent fuel pool via the ESW system boils off and exits through the standby gas treatment system or condenses in the reactor building, or the water overflows the pool. The consequences of up to 2.5 million gallons of water in each reactor building could include flooding of ECCS pump rooms, inoperability of ECCS pump room coolers, emergency switchgear and load center room coolers and/or other safety related equipment due to higher than analyzed temperature and humidity conditions, and degradation of the standby gas treatment system due to moisture loading. The standby gas treatment system is designed for 100% relative humidity conditions in the reactor building, but a system design calculation

9. Potential Engineering Discrepancy (continued)

(M-SGT-015) which determined that water buildup in the ductwork before the inlet HEPA filter would not degrade system performance does not consider the potential collapse/failure of the ductwork from the weight of this water.

C. The manual valve manipulations required to provide ESW makeup flow to a boiling spent fuel pool may not be possible. The off-normal operating procedure (ON-135-001) requires the operator to manually open the valves in the ESW makeup line to the fuel pool if all other means of adding water to the fuel pool are lost. The procedure calls for the valves to be left open until the desired water level is obtained. Since these valves are in the reactor building, it may be impossible for them to be manually operated as directed under all conditions including post-LOCA without unacceptable risk to the operator from the high radiation levels in the building and potentially high temperature and humidity conditions. The maximum gamma dose rates reported for EQ purposes for Unit 1 reactor building elevations 749'-1" to 818'-1" ranged between 140 and 360 R/hr (C-1815 Sh 7-10). In addition, even if the throttle valve is initially adjusted so that the ESW makeup flow to the fuel pool exactly matches the boil-off rate, the subsequent exponential reduction in fuel pool decay heat load would require the throttle valve to be periodically adjusted to lower the ESW makeup flow unless the fuel pool is permitted to overflow.

D. The instrumentation available to the operator post-LOCA does not provide adequate indication of spent fuel pool temperature and level to allow proper response to a loss of fuel pool cooling event. The off-normal operating procedure ON-135-001, "Loss of Fuel Pool Cooling/Coolant Inventory", requires the operator to manually open the valves in ESW makeup line to the fuel pool if all other means of adding water are lost. The operator enters this procedure upon annunciation of low level in the spent fuel pool or high temperature in the fuel pool cooling system. Each spent fuel pool has temperature indication (TE-15333) and level indication (LT-15332). Each fuel pool skimmer surge tank has level indication (LT-15312). The skimmer surge tank piping to the fuel pool heat exchangers has temperature indication

9. Potential Engineering Discrepancy (continued)

(TE-15313) and each fuel pool heat exchanger has temperature indication (TE-15316A,B,C) in its outlet piping. The level and temperature instruments providing these alarms may not be qualified for the temperature and humidity conditions, such as post-LOCA, in which they would be required to function. In addition, these instruments are not powered from class 1E sources such that they would be available post-LOOP when the fuel pool heat exchangers would be without service water.

E. The analytical 25 hour time-to-boil for the spent fuel pool is nonconservative for the maximum normal heat load in the spent fuel pool. The original design calculation (200-0048) used a decay heat load of 9.79×10^6 BTU/hr for Unit 1 and 7.92×10^6 BTU/hr for Unit 2 to determine times-to-boil of 25.15 hours and 31.087 hours respectively.

FSAR 9.1.3.1 establishes the maximum normal heat load as that heat load resulting from 2840 assemblies discharged to the fuel pool by a routine refueling schedule. FSAR Tables 9.1-2a and 9.1-2b report the maximum normal heat load for Units 1 and 2 as 12.6×10^6 BTU/hr.

The spent fuel pool decay heat values used in the boiling spent fuel pool calculation and for the FSAR discussion were based upon assumptions for cycle operating lengths, fuel exposures, and reactor power level. SSES has subsequently operated differently than had been assumed such that the decay heat loads in the filled spent fuel pools may exceed 9.79×10^6 BTU/hr, resulting in a shorter than analyzed time-to-boil.

A recent calculation prepared for power uprate (NFE-B-NA-053) determined decay heat from a filled spent fuel pool using actual fuel operating history through 1991 and assumptions which bound operation after power uprate. This calculation reported a maximum normal heat load of $\approx 17 \times 10^6$ BTU/hr.

Another recent calculation (M-FPC-009) determined the spent fuel pool time-to-boil and required ESW makeup rate for power uprate. Preliminary results from this calculation indicate the fuel pool boils 19.4 hours

9. Potential Engineering Discrepancy (continued)

after loss of fuel pool cooling for the design heat removal capacity of the fuel pool cooling system (13.2×10^6 BTU/hr). This calc determined a time-to-boil of ≈15.5 hours for the $\approx 17 \times 10^6$ BTU/hr heat load calculated for the power uprate case.

F. The analytical 25 hour time-to-boil for the spent fuel pool does not account for the emergency heat load in the spent fuel pool. FSAR 9.1.3.1 establishes the emergency heat load for the spent fuel pool as that heat load following a full core offload which completely fills the fuel pool. The FSAR specifies the emergency heat load to be 32.6×10^6 BTU/hr.

The original design calculation (200-0048) determined time-to-boil for the maximum normal heat load case only. The actual decay heat load in the spent fuel pool exceeds the maximum normal heat load during every refueling outage at SSES in which the core is fully offloaded.

SSES currently imposes administrative controls during refueling outages when the core is fully offloaded into the spent fuel pool to reduce the potential for loss of fuel pool cooling. Decay heat is removed from the spent fuel pool during these periods by RHR shutdown cooling (when the fuel pool to reactor cavity gates are removed) and by cross-tieing the operating unit's fuel pool cooling system to the outage unit's fuel pool. However, a seismic event in this configuration could cause loss of fuel pool cooling at a time when the time-to-boil is significantly less than 25 hours, which is not reflected in the off-normal operating procedure (ON-135-001).

Another recent calculation (M-FPC-009) determined the spent fuel pool time-to-boil and required ESW makeup rate for power uprate. Preliminary results from this calculation indicate the fuel pool boils 7.9 hours after loss of fuel pool cooling for a decay heat load of 36.2×10^6 BTU/hr, which is the currently analyzed emergency heat load.

9. Potential Engineering Discrepancy (continued)

G. The radiological release analysis for a boiling spent fuel pool uses nonconservative evaporation rates. The original design calculation (200-0048) used a decay heat load of 9.79×10^6 BTU/hr to determine evaporation rate. As reported above, the maximum normal heat load specified in FSAR Table 9.1-2a is 12.6×10^6 BTU/hr and the emergency heat load specified in FSAR 9.1.3.1 is 32.6×10^6 BTU/hr. When the decay heat load in the spent fuel pool exceeds 9.79×10^6 BTU/hr, the evaporation rate from the boiling pool will exceed the rate assumed in the radiological release analysis.

H. The radiological release analysis for a boiling spent fuel pool uses nonconservative activity terms. The original design calculation (200-0048) assumed 12 month operating cycles and 184 bundle equilibrium reload sizes to determine the activity terms for failed fuel in the fuel pool. SSES currently has 18 month operating cycles with ≈230 bundle reloads which will increase to ≈254 bundles after power uprate. Since the calculation implied that most of the activity results from the most recent discharge batch, the effect of increasing the discharge size from 184 bundles assumed in the calc to 230 and 254 bundles would appear to be nonconservative with respect to the radiological release analysis.

I. The analysis for maximum time prior to makeup to a boiling spent fuel pool is based upon nonconservative assumptions. The original design calculation (175-14) determined the time using evaporation of the entire fuel pool water inventory. The maximum time should be based upon a minimum fuel pool water level which is sufficiently above the top of the fuel to provide the shielding required to allow corrective operator actions.

Note 1: The existing COTTAP analysis (Calc M-RAF-024, Rev. 0) accounts for the sensible heat load from a boiling spent fuel pool at 212°F. No consideration was given to the latent heat load generated by the boiling pool, due to the limitations of the COTTAP code at that time. MRMyantvedt 4/20/92

EDR G20020
Loss of Spent Fuel Pool Cooling Event Design Discrepancies

A. Reactor Building Design Heat Loads Do Not Account for the Boiling Spent Fuel Pool Event

REQUIREMENT: 10 CFR 50.49 requires that electrical equipment must be qualified to the temperature "*for the most severe design basis accidents.*"

10 CFR 50 Appendix A General Design Criterion 4 states that "*structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents.*"

CONCERN: Secondary containment design analyses are required to account for all heat loads in the reactor building including from the boiling spent fuel pool. The existing design reactor building heat load calcs consider sensible heat from the boiling pool, but neglect latent heat. These calcs indicate little margin to design temperatures in many rooms for a maximum heat load in the reactor building of approximately 5.5×10^6 BTU/hr. The total design heat load from the spent fuel pools is 26.4×10^6 BTU/hr, which would add at least approximately 20.9×10^6 BTU/hr to the existing maximum heat load.

This concern affects the present operation of SSES because:

1) the boiling spent fuel pool is a current design bases event,

2) the fuel pools will boil following loss of fuel pool cooling with their existing decay heat loads,

3) the boiling spent fuel pool event has not been fully considered in reactor building heat load calcs, and

4) the potential consequences from the boiling spent fuel pool event will significantly and adversely affect the safety of SSES due to room temperatures in the reactor building exceeding design EQ values.

B. The Impact of the ESW Makeup Water to the Spent Fuel Pool on Equipment in the Reactor Building Has Not Been Evaluated

REQUIREMENTS: 10 CFR 50 Appendix A General Design Criterion 4 states that "*structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents.*"

Standard Review Plan (NUREG-0800) 3.4.1 states that the review of "*plant flood protection includes all structures, systems and components (SSC) whose failure could prevent safe shutdown of the plant or result on uncontrolled release of significant radioactivity...*" and that this review "*also includes consideration of flooding from internal sources.*"

CONCERN: FSAR 9.1.3.3 states the design ESW makeup function "*is based on replenishing the boil-off from the MNHL in each fuel pool for 30 days following the loss of the FPCCS capacity.*" The ultimate heat sink and ESW are designed to provide 1.5 million gallons of water to each fuel pool over the 30 day period. In the LOCA-LOOP condition, the reactor building HVAC system in Zone I; II and III isolation mode recirculates refueling floor air throughout all three zones. The water added to the fuel pools ends up in the reactor building following boil-off and overflow. The effects of this water on the structures, systems and components in the reactor buildings have not been included in design analyses. The potential for common mode failures of multiple ECCS and safety-related systems such as the standby gas treatment system exists.

EDR G20020
Loss of Spent Fuel Pool Cooling Event Design Discrepancies

This concern affects the present operation of SSES because:

1) the boiling spent fuel pool is a current design bases event,

2) the fuel pools will boil following loss of fuel pool cooling with their existing decay heat loads,

3) the boiling spent fuel pool event has not been fully considered in EQ and flooding effects calcs, and

4) the potential consequences from the boiling spent fuel pool event will significantly and adversely affect the safety of SSES due to common mode equipment failures due to water/humidity.

C. The Manual Valve Manipulations Required to Provide ESW Makeup Flow to a Boiling Spent Fuel Pool May Not Be Possible

REQUIREMENTS: 10 CFR 20.1 requires licensees to "*make every reasonable effort to maintain radiation exposures, and releases of radioactive materials in effluents to unrestricted areas, as low as is reasonably achievable.*"

10 CFR 50 Appendix A General Design Criterion 19 requires suitable design features to limit control room radiation exposure to 5 rem. GDC 19 also requires design features for equipment outside the control room to permit operation in accordance with suitable procedures.

10 CFR 50.47(b)(11) states that licensees assure that "*means for controlling radiological exposures, in an emergency, are established for emergency workers. The means for controlling radiological exposures shall include exposure guidelines consistent with EPA Emergency Worker and Lifesaving Activity Protective Action Guides.*"

NDI-6.4.3 specifies that the whole body dose for life saving actions "*shall not exceed 75 rem*" and the whole body dose for entry into a hazardous area to protect facilities or equipment "*shall not exceed 25 rem.*"

CONCERN: The ESW system is required to provide makeup to the pools following loss of fuel pool cooling. Either a seismic event or loss of offsite power can lead to loss of fuel pool cooling. Both conditions are assumed to occur concurrent with a LOCA in the DBA for containment analyses. However, the post-LOCA design EQ dose rates in the reactor building areas where the manual valves are located are 140-360 R/hr and will prevent these valves from being accessed without excessive radiation exposure to the operator. In addition, the reactor building temperature, humidity and emergency lighting conditions would not be conducive to the location and manipulation of manual valves which are used infrequently.

10 CFR 20's ALARA provision requires plant design to minimize radiation exposure. Application of the emergency dose guidelines to this manual valve operation is contrary to the intent of 10 CFR 20.1 and 10 CFR 50 App A GDC 19.

EDR G20020
Loss of Spent Fuel Pool Cooling Event Design Discrepancies

This concern affects the present operation of SSES because:

1) the boiling spent fuel pool is a current design bases event,

2) the fuel pools will boil following loss of fuel pool cooling with their existing decay heat loads,

3) the boiling spent fuel pool event analysis depends on makeup from ESW to prevent uncovering irradiated fuel and subsequent fuel damage from overheating, and

4) the potential consequences from the boiling spent fuel pool event will significantly increase if adequate makeup cannot be established, or

5) personnel will receive unnecessary radiation exposures which exceed 10 CFR 20.1/GDC 19 requirements and probably exceed 10 CFR 50.47 guidelines in order to align the makeup path.

D. The Instrumentation Available to the Operator Post-LOCA Does Not Provide Adequate Indication of Spent Fuel Pool Temperature and Level to Allow Proper Response to a Loss of Fuel Pool Cooling Event

REQUIREMENTS: 10 CFR 50 Appendix A General Design Criterion 63 states that "*appropriate systems shall be provided in fuel storage and radioactive waste systems and associated handling areas (1) to detect conditions that may result in loss of residual heat removal capability and excessive radiation levels and (2) to initiate appropriate safety actions.*"

Regulatory Guide 1.97 defines accident-monitoring instrumentation to include "*those variables to be monitored that provide the primary information required to permit the control room operators to take the specified manually controlled actions for which no automatic control is provided and that are required for safety systems to accomplish their safety function for design basis accident events.*"

Standard Review Plan (NUREG-0800) 9.1.3 states that the review of the spent fuel pool cooling and cleanup system design includes "*the instrumentation provided for initiating appropriate safety actions.*"

Standard Review Plan (NUREG-0800) 7.1 states that "*information systems important to safety include those systems which provide information for manual initiation and control of safety systems, to indicate that plant safety functions are being accomplished, and to provide information from which appropriate actions can be taken to mitigate the consequences of anticipated operational occurrences and accidents.*"

CONCERN: The ESW system is required to provide makeup to the pools following loss of fuel pool cooling. A loss of offsite power can result in loss of fuel pool cooling. The loss of offsite power will also disable the fuel pool temperature and level instruments monitored by the operator and used to initiate the safety action of providing ESW makeup to the boiling spent fuel pool.

EDR G20020
Loss of Spent Fuel Pool Cooling Event Design Discrepancies

This concern affects the present operation of SSES because:

1) the boiling spent fuel pool is a current design bases event,

2) the fuel pools will boil following loss of fuel pool cooling with their existing decay heat loads,

3) the boiling spent fuel pool event analysis depends on makeup from ESW to prevent uncovering irradiated fuel and subsequent fuel damage from overheating, and

4) the potential consequences from the boiling spent fuel pool event will significantly increase if adequate makeup cannot be established and lack of monitoring could prevent the required safety action from being initiated properly.

EDR G20020
Loss of Spent Fuel Pool Cooling Event Design Discrepancies

E. The Analytical 25 Hour Time-to-Boil for the Spent Fuel Pool is Nonconservative for the Maximum Normal Heat Load in the Spent Fuel Pool

REQUIREMENTS: FSAR Appendix 9A states "*conservative results showed that the pools would not boil until at least 25 hours after the loss of cooling.*"

FSAR Table 9A-2 states the total decay heat loads in the Unit 1 and Unit 2 fuel pools assumed in the loss of spent fuel pool cooling analysis are "*9.79*" and "*7.92*" BTU/hr x 10^6.

CONCERN: The maximum normal heat load in the spent fuel pool is presently higher than 9.79×10^6 BTU/hr and will also increase as a result of power uprate. The fuel pool will boil in less than 25 hours for any fuel pool heat load greater than 9.79×10^6 BTU/hr. (See Figure 1 from Calc M-FPC-009 attached).

This concern does <u>not</u> affect the present operation of SSES because the existing decay heat loads in the fuel pools are less than 9.79×10^6 BTU/hr.

NOTE: The original determination of maximum normal heat load relied on assumed reactor operating parameters such as fuel type, fuel discharge average exposure, and operating cycle length. These parameters have changed since the original calculation and will probably continue to change as fuel design and fuel management evolves. An approach to bound all such variables would consider the maximum normal heat load in the spent fuel pool to be equal to the design capacity of the fuel pool cooling system (13.2x106 BTU/hr). This approach would bound all heat loads capable of being handled by the fuel pool cooling system without depending upon predictions of fuel and core designs.

F. The Analytical 25 Hour Time-to-Boil for the Spent Fuel Pool Does Not Account for the Emergency Heat Load in the Spent Fuel Pool

REQUIREMENTS: Standard Review Plan (NUREG-0800) 9.1.3 states that the review of the spent fuel pool cooling and cleanup system design are reviewed to determine that "*a seismic Category I makeup system and an appropriate backup method to add coolant to the spent fuel pool are provided*" and that "*engineering judgement ... used to determine that the makeup capacities and the time required to make associated hookups are consistent with heatup times or expected leakage.*"

SSES Safety Evaluation Report (NUREG-0776) 9.1.3 states "*makeup from the Seismic Category I emergency service water systems would keep the fuel covered during loss of spent fuel pool cooling accidents.*"

FSAR 9.1.3.1 states that "*during an emergency heat load (EHL) condition, one RHR pump and heat exchanger are available for fuel pool cooling.*"

CONCERN: The emergency heat load condition requires an RHR loop to remove decay heat from the spent fuel pool. A single failure of the valve in the RHR line to the fuel pool, even without a concurrent seismic event or loss of offsite power, could initiate a loss of fuel pool cooling in which the time-to-boil would be significantly less than the 25 hours assumed in the radiological release analysis and in plant operating procedures. This potential exists presently during every refueling outage when the full core is offloaded to the spent fuel pool.

This concern affects the present operation of SSES during refueling outages because:

1) the boiling spent fuel pool is a current design bases event,

2) the fuel pools will boil in as little as 12 hours following loss of fuel pool cooling with the existing decay heat loads in the pools during refueling, and

3) the spent fuel pool boiling analysis assumes a minimum time to boil of 25 hours.

G. The Radiological Release Analysis for a Boiling Spent Fuel Pool Uses Nonconservative Evaporation Rates

REQUIREMENT: FSAR Appendix 9A reported that the radiological release analysis for the boiling spent fuel pools event were within the thyroid dose guidelines of 10 CFR 100 and the 1.5 rem thyroid dose requirement of Reg Guide 1.29.

CONCERN: The design calculation which performed the radiological release analysis for the boiling spent fuel pools event determined the evaporation rate from the pools based upon maximum normal heat loads of 9.79 and 7.92 x 10^6 BTU/hr. As discussed in Item (E) above, the present maximum normal heat load exceeds 9.79x10^6 BTU/hr and will increase after power uprate. Therefore, the actual rate at which water evaporates from the boiling spent fuel pool is higher than analyzed which introduces nonconservatism into the offsite dose calculation.

This concern does <u>not</u> affect the present operation of SSES (except during refueling outages as noted in Item F above) because the existing decay heat loads in the fuel pools are less than 9.79x10^6 BTU/hr.

H. The Radiological Release Analysis for a Boiling Spent Fuel Pool Uses Nonconservative Activity Terms

REQUIREMENT: FSAR Appendix 9A reported that the radiological release analysis for the boiling spent fuel pools event were within the thyroid dose guidelines of 10 CFR 100 and the 1.5 rem thyroid dose requirement of Reg Guide 1.29.

CONCERN: The design calculation which performed the radiological release analysis for the boiling spent fuel pools event determined the source terms in the spent fuel pools based upon assumptions for fuel design and cycle operation. SSES has been operated with different fuel types and longer cycles than assumed in the analysis which introduces nonconservatism into the offsite dose calculation.

In addition, the conclusions reported in FSAR Appendix 9A regarding the thyroid doses from FSAR Table 9A-1 are not valid for all cases. FSAR Table 9A-1 only addresses offsite doses from activity released from the two boiling spent fuel pools. Since the boiling spent fuel pools can occur as a result of the LOCA-LOOP with SSE DBA, these thyroid doses should be added to the doses resulting from the LOCA.

This concern affects the present operation of SSES because:

1) the boiling spent fuel pool is a current design bases event,

2) the fuel pools will boil following loss of fuel pool cooling with their existing decay heat loads,

3) the potential consequences from the boiling spent fuel pool event may significantly increase due to higher source term activity associated with 9x9 fuel, larger discharge batch sizes, and higher bundle exposures, and

4) the offsite dose resulting from the boiling spent fuel pool is not considered in the total offsite dose resulting from the DBA LOCA-LOOP.

EDR G20020
Loss of Spent Fuel Pool Cooling Event Design Discrepancies

I. The Analysis for Maximum Time Prior to Makeup to a Boiling Spent Fuel Pool is Based Upon Nonconservative Assumptions

REQUIREMENT: Calc 175-14 determined the maximum time available before makeup to a boiling spent fuel pool is required.

CONCERN: The time determined by this design calculation is based upon how long it would take to completely evaporate the entire spent fuel pool water inventory. Allowing the entire spent fuel pool to evaporate prior to makeup would have severe and unanalyzed consequences:

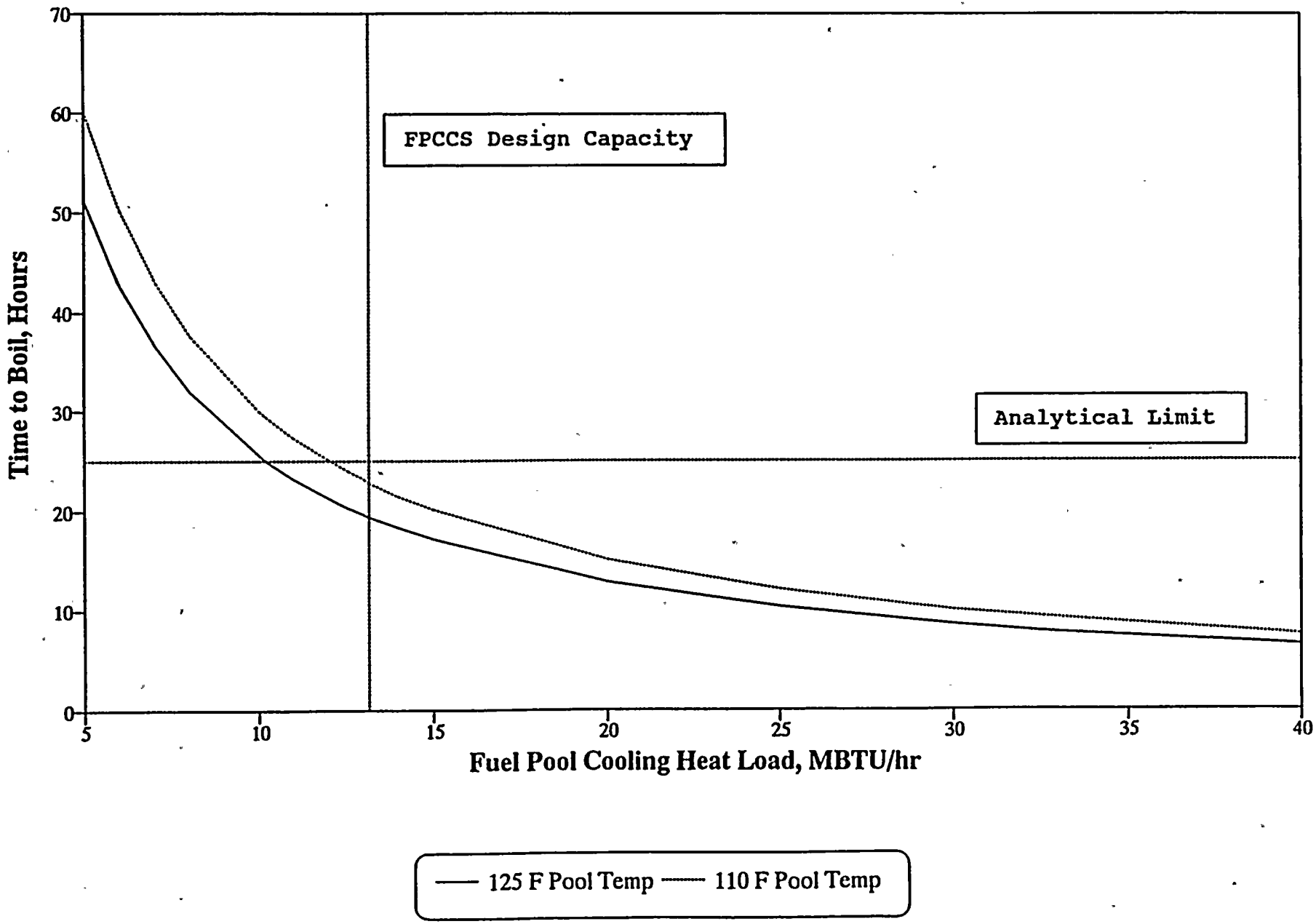
a) reactor building radiation doses would significantly increase,

b) offsite radiological doses would significantly increase due to skyshine, and

c) fuel integrity of the irradiated fuel would be challenged as it was uncovered.

This concern does <u>not</u> appear to affect the present operation of SSES because no document or procedure is known to use the results of this calc. However, an exhaustive search was not performed.

Figure 1
Time to Boil vs. FPC Heat Load



Attachment 6

PP&L Draft Screening Worksheet prepared by Art White, "EDR No. G20020", July 1, 1992.

Note: The first nine text pages of this draft evaluation of EDR G20020 prepared by an engineer within the PP&L Engineering Discrepancy Management Group were taken almost verbatim from the authors' memo dated June 22, 1992 (Attachment 5). The final three pages of 'analysis' for EDR G20020 provide ample evidence of PP&L's reliance upon probability arguments, use of realistic instead of design conditions, and oversimplification of issues while assessing the safety significance of concerns.

SCREENING WORKSHEET

DRAFT

UNIT 1 & 2

EDR No.G20020

SUSQUEHANNA STEAM ELECTRIC STATION

PENNSYLVANIA POWER & LIGHT COMPANY

PREPARED BY	DATE	REVIEWED BY	DATE

DISCREPANCY ITEM/LOCATION/SUBJECT:

Loss of Spent Fuel Cooling Event Design Discrepancies

DESCRIPTION OF CONDITION

The regulatory requirements for cooling the spent fuel pool are based upon: 10 CFR Appendix A Design Criterion 61 which states that the fuel storage system shall be designed " to prevent significant reduction in fuel storage coolant inventory under accident conditions" and Standard Review Plan (NUREG-800) 9.1.3 for the spent fuel pool cooling and cleanup system which states that the "safety function to be performed by the system in all cases remains the same; that is, the spent assemblies must be cooled and must remain covered with water during all storage conditions.

The SSES design utilizes non-seismic, non-Class IE powered fuel pool cooling and cleanup systems for cooling the fuel pools. In the event of a loss of spent fuel pool cooling, the design provision at SSES is to allow the fuel pools to boil with adequate makeup provided to maintain the water level in the pools above the fuel. The SSES design requirements are based upon:

FSAR Appendix 9A which states that it is assumed "a seismic event causes the loss of cooling to both spent fuel pools" and that "if cooling is not restored before the pool boils, then makeup water from the Category I Emergency Service Water System can be added to the pool to keep the fuel covered at all times," and

FSAR 6.2.1.1.1(a) states that "The LOCA scenario used for containment functional design includes the worst single failure (which leads to maximum coincident containment pressure and temperature), postulated to occur simultaneously with loss of offsite power and a safe shutdown earthquake (SSE)."

Since an analyzed design basis accident (DBA) at SSES is a LOCA with a concurrent LOOP and SSE, and either a seismic event or a loss of offsite power will result in a loss of spent fuel pool cooling, the consequences of this DBA include boiling spent fuel pools. The SSES design was (NUREG-0776) 9.1.3 which states "makeup from the Seismic Category I emergency service water systems would keep the fuel covered during loss os spent fuel pool cooling accidents."

The following design discrepancies for the loss of spent fuel pool event:

A. Reactor Building Design Heat Loads Do Not Account for the Boiling Spent Fuel Pool Event

Requirement: 10 CFR 50.49 requires that electrical equipment must be qualified to the temperature "for the most severe design basis accidents."

> 10 CFR 50 Appendix A General Design Criterion 4 states that "structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents."

Concern: Secondary containment design analyses are required to account for all heat loads in the reactor building including from the heat load calcs consider sensible heat from the boiling pool, but neglect latent heat. These calcs indicate little margin to design temperatures in many rooms for a maximum heat load in the reactor building of approximately 5.5E6 BTU/hr. The total design heat load from the spent fuel pools is 26.4E6 BTU/hr, which would add at least approximately 20.9E6 BTU/hr to the existing maximum heat load.

This concern affects the present operation of SSES because:

1) the boiling spent fuel pool is a current design bases event,

2) the fuel pools will boil following loss of fuel pool cooling with their existing decay heat loads,

3) the boiling spent fuel pool event has not been fully considered in reactor building heat load calcs, and

4) the potential consequences from the boiling spent fuel pool event will significantly and adversely affect the safety of SSES due to room temperatures in the reactor building exceeding design EQ values.

B. The Impact of the ESW Makeup Water to the Spent Fuel Pool on Equipment in the Reactor Building has not been evaluated

Requirements: 10 CFR 50 Appendix A General Design Criterion 4 states that "structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents."

Standard Review Plan (NUREG-0800) 3.4.1 states that the

review of "plant flood protection includes all structures, systems and components (SSC) whose failure could prevent safe shutdown of the plant or result in uncontrolled release of significant radioactivity...." and that this review "also includes consideration of flooding from internal sources."

Concern: FSAR 9.1.3.3 states the design ESW makeup function "is based on replenishing the boil-off from the MNHL in each fuel pool for 30 days following the loss of the FPCCS capacity." The ultimate heat sink and ESW are designed to provide 1.5 million gallons of water to each fuel pool over the 30 day period. In the LOCA-LOOP condition, the reactor building HVAC system in Zone I, II and III isolation mode recirculates refueling floor air throughout all three zones. The water added to the fuel pools ends up in the reactor building following boil-off and overflow. The effects of this water on the structures, systems and components in the reactor buildings have not been included in design analyses. The potential for common mode failures of multiple ECCS and safety-related systems such as the standby gas treatment system exists.

This concern affects the present operation of SSES because:

1) the boiling spent fuel pool is a current design bases event,

2) the fuel pools will boil following loss of fuel pool cooling with their existing decay heat loads,

3) the boiling spent fuel pool event has not been fully considered in EQ and flooding effects calcs, and

4) the potential consequences from the boiling spent fuel pool event will significantly and adversely affect the safety of SSES due to common mode equipment failures due to water/humidity.

C. The manual valve manipulations required to provide ESW makeup flow to a boiling spent fuel pool may not be possible.

Requirements: 10 CFR 20.1 requires licensees to "make every reasonable effort to maintain radiation exposures, and releases of radioactive materials in effluents to unrestricted areas, as low as is reasonably achievable."

10CFR 50 Appendix A General Design Criterion 19 requires suitable design features to limit control room radiation exposure to 5 rem. GDC 19 also requires design features for equipment outside the control room to permit operation in accordance with suitable procedures.

10 CFR 50.47 (b) (11) states that licensees assure that

"means for controlling radiological exposures, in an emergency, are established for emergency workers. The means for controlling radiological exposures shall include exposure guidelines consistent with EPA Emergency Worker and Lifesaving Activity Protective Action Guides."

NDI-6.4.3 specifies that the whole body dose for life saving actions "shall not exceed 75 rem" and the whole body dose for entry into a hazardous area to protect facilities or equipment "shall not exceed 25 rem."

Concern: The ESW system is required to provide makeup to the pools following loss of fuel pool cooling. Either a seismic event or loss of offsite power can lead to loss of fuel pool cooling. Both conditions are assumed to occur concurrent with a LOCA in the DBA for containment analyses. However, the post-LOCA design EQ dose rates in the reactor building areas where the manual valves are located are 140-360R/hr and will prevent these valves from being accessed without excessive radiation exposure to the operator. In addition, the reactor building temperature, humidity and emergency lighting conditions would not be conducive to the location and manipulation of manual valves which are used infrequently.

10CFR 20's ALARA provision requires plant design to minimize radiation exposure. Application of the emergency dose guidelines to this manual valve operation is contrary to the intent of 10 CFR 20.1 and 10 CFR 50 App A GDC 19.

This concern affects the present operation of SSES because:

1) the boiling spent fuel pool is a current design bases event,

2) the fuel pools will boil following loss of fuel pool cooling with their existing decay heat loads,

3) the boiling spent fuel pool event analysis depends on makeup from ESW to prevent uncovering irradiated fuel and subsequent fuel damage from overheating, and

4) the potential consequences from the boiling spent fuel pool event will significantly increase if adequate makeup cannot be established, or

5) personnel will receive unnecessary radiation exposures which exceed 10 CFR 20.1/GDC 19 requirements and probably exceed 10 CFR 50.47 guidelines in order to align the makeup path.

D. The instrumentation available to the Operator Post-LOCA does not provide adequate indication of spent fuel pool temperature and level to allow proper response to a loss of fuel pool cooling event

Requirements: 10 CFR Appendix A General Design Criterion 63 states that "appropriate systems shall be provided in fuel storage and radioactive waste systems and associated handling areas (1) to detect conditions that may result in loss of residual heat removal capability and excessive radiation levels and (2) to initiate appropriate safety actions."

Regulatory Guide 1.97 defines accident-monitoring instrumentation to include "those variables to be monitored that provide the primary information required to permit the control room operators to take the specified manually controlled actions for which no automatic control is provided and that are required for safety systems to accomplish their safety function for design basis accident events."

Standard Review Plan (NUREG-0800) 9.1.3 states that the review of the spent fuel pool cooling and cleanup system design includes "the instrumentation provided for initiating appropriate safety actions."

Standard Review Plan (NUREG-0800) 7.1 states that "information systems important to safety include those systems which provide information for actual initiation and control of safety systems, to indicate that plant safety functions are being accomplished, and to provide information from which appropriate actions can be taken to mitigate the consequences of anticipated operational occurrences and accidents."

Concern: The ESW system is required to provide makeup to the pools following loss of fuel pool cooling. A loss of offsite power can result in loss of fuel pool cooling. The loss of offsite power will also disable the fuel pool temperature and level instruments monitored by the operator and used to initiate the safety action of providing ESW makeup to the boiling spent fuel pool.

This concern affects the present operation of SSES because:

1) the boiling spent fuel pool is a current design bases event,

2) the fuel pools will boil following loss of fuel pool cooling with their existing decay heat loads,

3) the boiling spent fuel pool event analysis depends on makeup from ESW to prevent uncovering irradiated fuel and subsequent fuel damage from overheating, and

4) the potential consequences from the boiling spent fuel pool event will significantly increase if adequate makeup cannot be established and lack of monitoring could prevent the required

safety action from being initiated properly.

E. The analytical 25 hour time-to-boil for the spent fuel pool is nonconservative for the maximum normal heat load in the spent fuel pool.

Requirements: FSAR Appendix 9A states "conservative results showed that the pools would not boil until at least 25 hours after the loss of cooling."

> FSAR Table 9A-2 states the total decay heat loads in the Unit 1 and Unit 2 fuel pools assumed in the loss of spent fuel pool cooling analysis are " 9.79E6 BTU/hr and 7.92E6 BTU/hr.

Concern: The maximum normal heat load in the spent fuel pool is presently higher than 9.79E6 BTU/hr and will also increase as a result of power uprate. The fuel pool will boil in less than 25 hours for any fuel pool heat load greater than 9.79E6 BTU/hr. (See Figure 1 from Calc M-FPC-009 attached).

This concern does not affect the present operation of SSES because the existing decay heat loads in the fuel pools are less than 9.79E6 BTU/hr.

Note: The original determination of maximum normal heat load relied on assumed reactor operating parameters such as fuel type, fuel discharge average exposure, and operating cycle length. These parameters have changed since the original calculation and will probably continue to change as fuel design and fuel management evolves. An approach to bound all such variables would consider the maximum normal heat load in the spent fuel pool to be equal to the design capacity of the fuel pool cooling system (13.2E6 BTU/hr). This approach would bound all heat loads capable of being handled by the fuel pool cooling system without depending upon predictions of fuel and core designs.

F. The analytical 25 hour time-to-boil for the spent fuel pool does not account for the emergency heat load in the spent fuel pool.

Requirements: Standard Review Plan (NUREG-0800) 9.1.3 states that the review of the spent fuel pool cooling and cleanup system design are reviewed to determine that " a seismic Category I makeup system and an appropriate backup method to add coolant to the spent fuel pool are provided" and that "engineering judgement. . .used to determine that the makeup capacities and the time required to make associated hookups are consistent with heatup times or expected leakage."

SSES Safety Evaluation Report (NUREG-0776) 9.1.3 states "makeup from the Seismic Category I emergency service water systems would keep the fuel covered during loss of spent fuel pool cooling

accidents."

FSAR 9.1.3.1 states that "during an emergency heat load (EHL) condition, one RHR pump and heat exchanger are available for fuel pool cooling."

Concern: The emergency heat load condition requires an RHR loop to remove decay heat from the spent fuel pool. A single failure of the valve in the RHR line to the fuel pool, even without a concurrent seismic event or loss of offsite power, could initiate a loss of fuel pool cooling in which the time-to-boil would be significantly less than the 25 hours assumed in the radiological release analysis and in plant operating procedures. This potential exists presently during every refueling outage when the full core is offloaded to the spent fuel pool.

This concern affects the present operation of SSES during refueling outages because:

1) the boiling spent fuel pool is a current design bases event,

2) the fuel pools will boil in as little as 12 hours following loss of fuel pool cooling with the existing decay heat loads in the pools during refueling, and

3) the spent fuel pool boiling analysis assumes a minimum time to boil of 25 hours.

G. The Radiological Release analysis for a boiling spent fuel pool uses nonconservative evaporation rates

Requirement: FSAR Appendix 9A reported that the radiological release analysis for the boiling spent fuel pools event were within the thyroid dose guidelines of 10 CFR 100 and the 1.5 rem thyroid dose requirement of Reg Guide 1.29.

Concern: The design calculation which performed the radiological release analysis for the boiling spent fuel pools event determined the evaporation rate from the pools based upon maximum normal heat loads of 9.79E6 BTU/hr and will increase after power uprate. Therefore, the actual rate at which water evaporates from the boiling spent fuel pool is higher than analyzed which introduces nonconservatism into the offsite dose calculation.

This concern does not affect the present operation of SSES (except during refueling outages as noted in Item F above) because the existing decay heat loads in the fuel pools are less than 9.79E6 BTU/hr.

H. The radiological release analysis for a boiling spent fuel pool uses nonconservative activity terms.

Requirement: FSAR Appendix 9A reported that the radiological release analysis for the boiling spent fuel pools event were within the thyroid dose guidelines of 10 CFR 100 and the 1.5 rem thyroid dose requirement of Reg Guide 1.29.

Concern: The design calculation which performed the radiological release analysis for the boiling spent fuel pools event determined the source terms in the spent fuel pools based upon assumptions for fuel design and cycle operation. SSES has been operated with different fuel types and longer cycles than assumed in the analysis which introduces nonconservatism into the offsite dose calculation.

In addition, the conclusions reported in FSAR Appendix 9A regarding the thyroid doses from FSAR Table 9A-1 are not valid for all cases. FSAR Table 9A-1 only addresses offsite doses from activity released from the two boiling spent fuel pools. Since the boiling spent fuel pools can occur as a result of the LOCA-LOOP with SSE DBA, these thyroid doses should be added to the doses resulting from the LOCA.

This concern affects the present operation of SSES because:

1) the boiling spent fuel pool is a current design bases event,

2) the fuel pools will boil following loss of fuel pool cooling with their existing decay heat loads,

3) the potential consequences from the boiling spent fuel pool event may significantly increase due to higher source term activity associated with 9X9 fuel, larger discharge batch sizes, and higher bundle exposures, and

4) the offsite dose resulting from the boiling spent fuel pool is not considered in the total offsite dose resulting from the DBA LOCA-LOOP.

I. The analysis for maximum time prior to makeup to a boiling spent fuel pool is based upon nonconservative assumptions

Requirement: Calc 175-14 determined the maximum time available before makeup to a boiling spent fuel pool is required.

Concern: The time determined by this design calculation is based upon how long it would take to completely evaporate the entire spent fuel pool water inventory. Allowing the entire spent fuel pool to evaporate prior to makeup would have severe and unanalyzed consequences:

1) reactor building radiation doses would significantly increase,

2) offsite radiological doses would significantly increase due to skyshine, and

3) fuel integrity of the irradiated fuel would be challenged as it was uncovered.

This concern does not appear to affect the present operation of SSES because no document or procedure is known to use the results of this calc. However, an exhaustive search was not performed.

ANALYSIS

I. DOES THE ENGINEERING DISCREPANCY APPEAR TO CREATE A HIGH CALCULATED ACCIDENT SEQUENCE FREQUENCY?

BASIS: No, the postulated concern is based on postulating a DBA LOCA, a LOOP and an SSE all simultaneously. The probability of such an event approaches zero, it is so vanishingly small.

II. DOES THE ENGINEERING DISCREPANCY APPEAR TO INVALIDATE DEFINED STAGE IN "DEFENSE-IN-DEPTH" AGAINST AN ACCIDENT SEQUENCE, WHETHER EQUIPMENT OR PROCEDURE RELATED?

BASIS: No, the postulated concern takes no credit for manual action. From a realistic point-of-view, there is no basis to assume that fuel damage will occur to the extent that manual actions can be taken to line up ESW and RHR in the spent fuel area.

III. DOES THE ENGINEERING DISCREPANCY APPEAR TO ADVERSELY IMPACT A SYSTEM OR COMPONENT EXPLICITLY LISTED IN THE TECHNICAL SPECIFICATIONS?

YES	TECHNICAL SPECIFICATION SECTION(S)
NO X	

BASIS: This discrepancy has no basis in fact, and takes no credit for expected operator action.

IV. DOES THE DISCREPANCY APPEAR TO COMPROMISE THE CAPABILITY OF A SYSTEM OR COMPONENT TO PERFORM ITS SAFETY RELATED FUNCTION AS DESCRIBED IN THE SAFETY ANALYSIS REPORT?

YES	SAR SECTION(S)
NO X	

BASIS: This concern has no effect on any safety related function as described in the Safety Analysis Report. The EDR's basis appears to be an invalid application of design philosophies to (realistic) post accident manual actions.

V. DOES THE DISCREPANCY APPEAR TO ADVERSELY IMPACT ANY APPLICABLE LICENSING COMMITMENTS?

YES	REFERENCE
NO X	

BASIS: The discrepancy does not appear to adversely impact any applicable licensing commitments. In fact, the SER specifically addresses the Spent Fuel Cooling Function, and its ESW makeup and RHR cooling function.

VI. SAFETY SIGNIFICANCE ASSESSMENT

Address plant-specific features which affect the safety significance of the concern. Provide a realistic assessment of the actual safety consequences and implications of the concern.

SAFETY SIGNIFICANCE SUMMARY*	
NONE	X
MINIMAL	
MODERATE	
CONSIDERABLE	

BASIS: There is no safety significance to this EDR since it has no basis in fact if one does not accept the premise that no action can or will be taken by operations personnel to stop a boiling pool from boiling, or to inhibit it from boiling in the first place. Basically, it is a misapplication of plant design parameters, such as postulated fuel melt, to post accident operator actions.

*This is an initial assessment. The screening function is to be considered a continuous process. A re-evaluation of the screening status (not necessarily formal, except when determined to be "significant") should take place by referencing this procedure at each stage of EDR processing (e.g. EDR implementation) to determine if the issue is now a "safety concern" and is subject to Reportability and/or Operability determinations.

Attachment 7

Handout, "EDR G20020 References", July 15, 1992

Note: This handout was prepared by the authors and distributed during a meeting on July 15, 1992 to discuss EDR G20020. The handout summarizes the documents researched by the authors while preparing the EDR and subsequently defending its merits.

EDR G20020 References

Design Bases and Related Issues

FSAR 6.2.1.1.1 states that the "*LOCA scenario used for containment functional design includes the worst single failure (which leads to maximum coincident containment pressure and temperature), postulated to occur simultaneously with loss of offsite power and a safe shutdown earthquake (SSE).*"

FSAR Appendix 9A states that "*it was assumed that a seismic event causes the loss of cooling to both spent fuel pools.*"

FSAR Appendix 9A states that "*if cooling is not restored before the pool boils, then makeup water from the Category I Emergency Service Water System can be added to the pool to keep the fuel covered at all times.*"

SSES Safety Evaluation Report 9.1.3 states that "*makeup from the seismic Category I emergency service water systems would keep the fuel covered during loss of spent fuel pool cooling accidents.*"

Bechtel Spec M-192 for the High Density Spent Fuel Storage Racks (June 1977) states that the "*seller shall perform analysis to determine the makeup flow rate required to maintain the pool water level under conditions of maximum heat load, none of the cooling systems available and pool water boiling.*"

Letter PLI-7457 from A. M. Male to R. J. Shovlin (July 1979) states that "*the spent fuel pool cooling system is designed to maintain temperature at or below 125°F. The system is further backed up by the Seismic Category I Appendix B qualified emergency systems which have sufficient capacity to handle this load. If all of these redundant systems are somehow unavailable, it will still take more than one day before boiling begins. This is more than sufficient time for onsite personnel to provide from many alternate water sources enough make up water to keep the pool from boiling.*"

Technical Report NPE-84-002 (December 1983) states that "*SSES is designed to accept and mitigate a loss of coolant accident (LOCA) concurrent with a complete loss of offsite power (LOOP)*" and "*the assumption was made, in the design of SSES, that the LOCA and LOOP would occur simultaneously, and the simultaneous occurrence of LOCA and LOOP becomes the design basis event.*"

EDR G20020 References

EWR MIS 86-0637 determined that the RHR fuel pool cooling assist mode lines could be deleted from the ISI program since the "*present design uses the ESW makeup line as the ultimate heat removal source*" with this source being "*sufficient to cover the maximum boiloff of a full core offload.*"

EWR MIS 85-0740 stated that "*the RHR assist mode to fuel pool cooling is a non-safety function and therefore may be deleted from the ISI program boundaries*" and this mode is "*non-safety and adequate cooling is still available form boiling and ESW makeup.*"

NSAG 4-90 (September 1990) reported that in the RHR system design "*the fuel pool cooling assist and the shutdown cooling modes share a common suction line. Therefore, the system can not operate in both modes at the same time.*"

EPRI Report NP-2301 (March 1982) reported that 27% of loss of offsite power events at nuclear plants had been caused by weather related problems. This report also stated that in 5% of all the loss of offsite power events at nuclear plants, the duration exceeded 24 hours.

NSAC Report 182 (March 1992) reported 21 loss of offsite power events lasting longer than one hour at nuclear plants between 1980 and 1991, with the longest event lasting 18:58.

Telecon from Michael Rose (PP&L) to Mort Renslo (Bechtel) of November 9, 1981 states that "*according to Bechtel's Civil and Structural Design Criteria for the Susquehanna Steam Electric Station...This criteria states Fuel Pool Structure shall be designed for water boiling during accident condition.*"

A. **Reactor Building Design Heat Loads Do Not Account for the Boiling Spent Fuel Pool Event**

FSAR 6.2.2.1(d) states that the safety design bases for the containment removal system is that the system "*shall maintain operation during those environmental conditions imposed by the LOCA.*"

EWR 830658 (March 1983) noted "*the initial boiling rate corresponds to ≈3000 cfm of 100% water vapor at one atm. Is the equipment which will be exposed to this atmosphere qualified for it?*"

SEA-ME-099 (December 1987) analyzed reactor building temperatures for LOCA, LOCA/LOOP and LOCA/false LOCA cases assuming spent fuel temperatures remained at 125°F, but listed as a nonconservatism that fuel pool heatup in the LOCA/LOOP case would result in higher heat loads from the RHR systems, fuel pool walls and fuel pool surface.

B. **The Impact of the ESW Makeup Water to the Spent Fuel Pool on Equipment in the Reactor Building Has Not Been Evaluated**

FSAR 6.3.1.1.3 states that separation barriers for ECCS "*shall be constructed between the functional groups as required to assure that environmental disturbances such as fire, pipe rupture, falling objects, etc., affecting one functional groups will not affect the remaining groups. In addition, separation barriers shall be provided as required to assure that such disturbances do not affect both RCIC and HPCI.*"

FSAR 9.1.3.3 states that "*the design makeup rate from each ESW loop is based on replenishing the boil-off from the MNHL in each fuel pool for 30 days following the loss of FPCCS capacity.*"

EWR 830658 (March 1983) noted "*condensation may be expected from this evaporation which will run down to lower levels of the R.B. Will this cause loss of essential equipment, particularly electrical? Has an evaluation been performed?*"

Minutes from Bechtel meeting on HVAC systems (February 1980) states that original requirement for SGTS was "*to handle fumes from a boiling fuel pool,*" but that SGTS will not be able to handle this mixture since the room will become too hot. "*This requirement will be deleted from the FSAR.*"

C. **The Manual Valve Manipulations Required to Provide ESW Makeup Flow to a Boiling Spent Fuel Pool May Not Be Possible**

FSAR 9.1.3.2 states that "*the manual supply valves in these emergency makeup lines are accessible apart from the refueling floor.*"

FSAR 18.1.20 (NUREG-0737 Item II.B.2) states that "*each licensee shall provide for adequate access to vital areas and protection of safety equipment by design changes, increased permanent or temporary shielding, or postaccident procedural controls. The design review shall determine which types of corrective actions are needed for vital areas throughout the facility.*"

FSAR 18.1.20.3.3.4.1 defines vital areas as those "*which will or may require occupancy to permit an operator to aid in the mitigation of or recovery from an accident.*"

FSAR 18.1.20.3.2.1 states that "*a review was made to determine which systems could be required to operate and/or be expected to contain highly radioactive materials following a postulated accident where substantial core damage has occurred.*"

FSAR 18.1.20.3.2.5 states "*exposures for areas not continuously occupied (frequent and infrequent occupancy) must be determined case by case, that is, multiply the task duration by the area dose rate at the time of exposure.*"

FSAR 18.1.20.3.3.3 states that "*GDC 19 is also used to govern design bases for the maximum permissible dosage to personnel performing any task required post-accident. These requirements translate roughly into the objectives to be met in the post-accident review as given below.*

Radiation Exposure Guidelines

Occupancy	*Dose Rate Objectives*	*Dose Objective*
Continuous	*15 mR/hr*	*5 Rem for duration*
Frequent	*100 mR/hr*	*5 Rem for all activities*
Infrequent	*500 mR/hr*	*5 Rem per activity*
Accessway	*5 R/hr*	*Included in above doses*"

FSAR 18.1.20.3.4.3 states that the review results "*show that the reactor building will be generally inaccessible for several days after the accident due to contained radiation sources.*"

FSAR Figure 18.1-4 shows Room I-105 where ESW valves 153500/153501 are located to be in Rad Zone VIII with dose rates over 5000 R/hr.

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FSAR Figure 18.1-6 shows Room I-514 where ESW valves 153090A&B and 153091A&B are located to be in Rad Zone V with dose rates between 5 and 50 R/hr.

NDI-6.4.3 establishes the whole body dose for life saving to be 75 Rem, with a dose limit of 25 Rem for less urgent measures to protect equipment.

D. The Instrumentation Available to the Operator Post-LOCA Does Not Provide Adequate Indication of Spent Fuel Pool Temperature and Level to Allow Proper Response to a Loss of Fuel Pool Cooling Event

E. **The Analytical 25 Hour Time-to-Boil for the Spent Fuel Pool is Nonconservative for the Maximum Normal Heat Load in the Spent Fuel Pool**

FSAR 9.1.3.1 states that "*the pool will begin to boil 25 hours after loss of cooling.*"

FSAR Appendix 9A states that "*the conservative results showed that the pools would not boil until at least 25 hours after the loss of cooling.*"

FSAR Table 9A-2 reports the total decay heat load to be 9.79×10^6 BTU/hr in the Unit 1 SFP and 7.92×10^6 BTU/hr in the Unit 2 SFP for the boiling spent fuel pool analysis.

Bechtel Calc 200-0048 (July 1977) determined a 25 hour time to boil for the Unit 1 SFP and a 31 hour time to boil for the Unit 2 SFP based upon 12 month operating cycles and 184 bundle reload sizes.

PP&L Calc NFE-B-NA-053 (February 1992) determined a spent fuel pool maximum normal heat load of $\approx 14.6 \times 10^6$ BTU/hr and emergency heat load of $\approx 30 \times 10^6$ BTU/hr using actual SSES operating history through 1991 and projected operation until the pool is filled.

NSAG Report 4-90 states that "*Appendix 9A of the FSAR states that at least 25 hours would be required to boil the spent fuel pool under worst case loading.*"

F. The Analytical 25 Hour Time-to-Boil for the Spent Fuel Pool Does Not Account for the Emergency Heat Load in the Spent Fuel Pool

G. The Radiological Release Analysis for a Boiling Spent Fuel Pool Uses Nonconservative Evaporation Rates

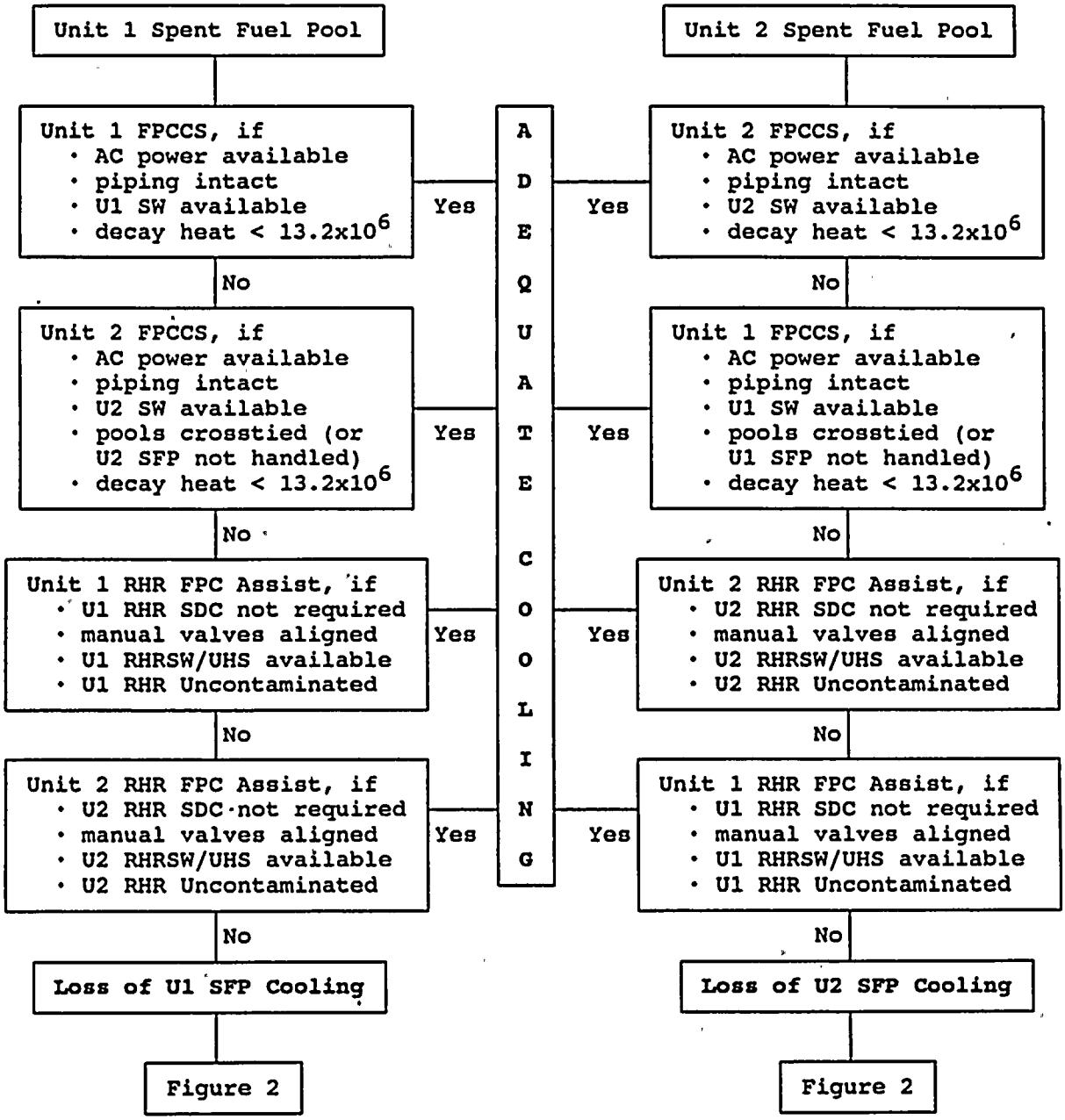
H. The Radiological Release Analysis for a Boiling Spent Fuel Pool Uses Nonconservative Activity Terms

EDR G20020 References

I. **The Analysis for Maximum Time Prior to Makeup to a Boiling Spent Fuel Pool is Based Upon Nonconservative Assumptions**

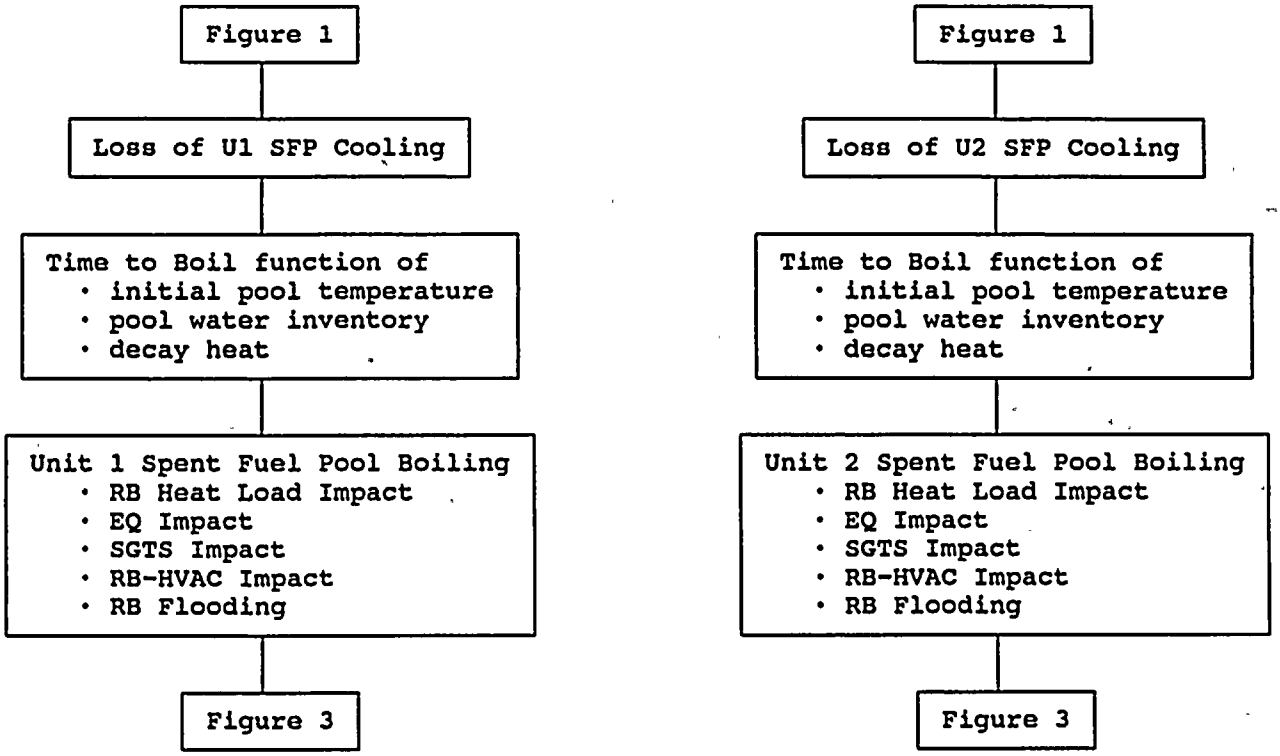
NSAG Report 13-84 (December 1984) reported that water level in the spent fuel pool dropping to within five inches of the top of the irradiated fuel "*would cause radiation levels on the 818′ elevation of the reactor building in excess of 100,000 rem/hour.*"

Figure 1
Fuel Pool Cooling



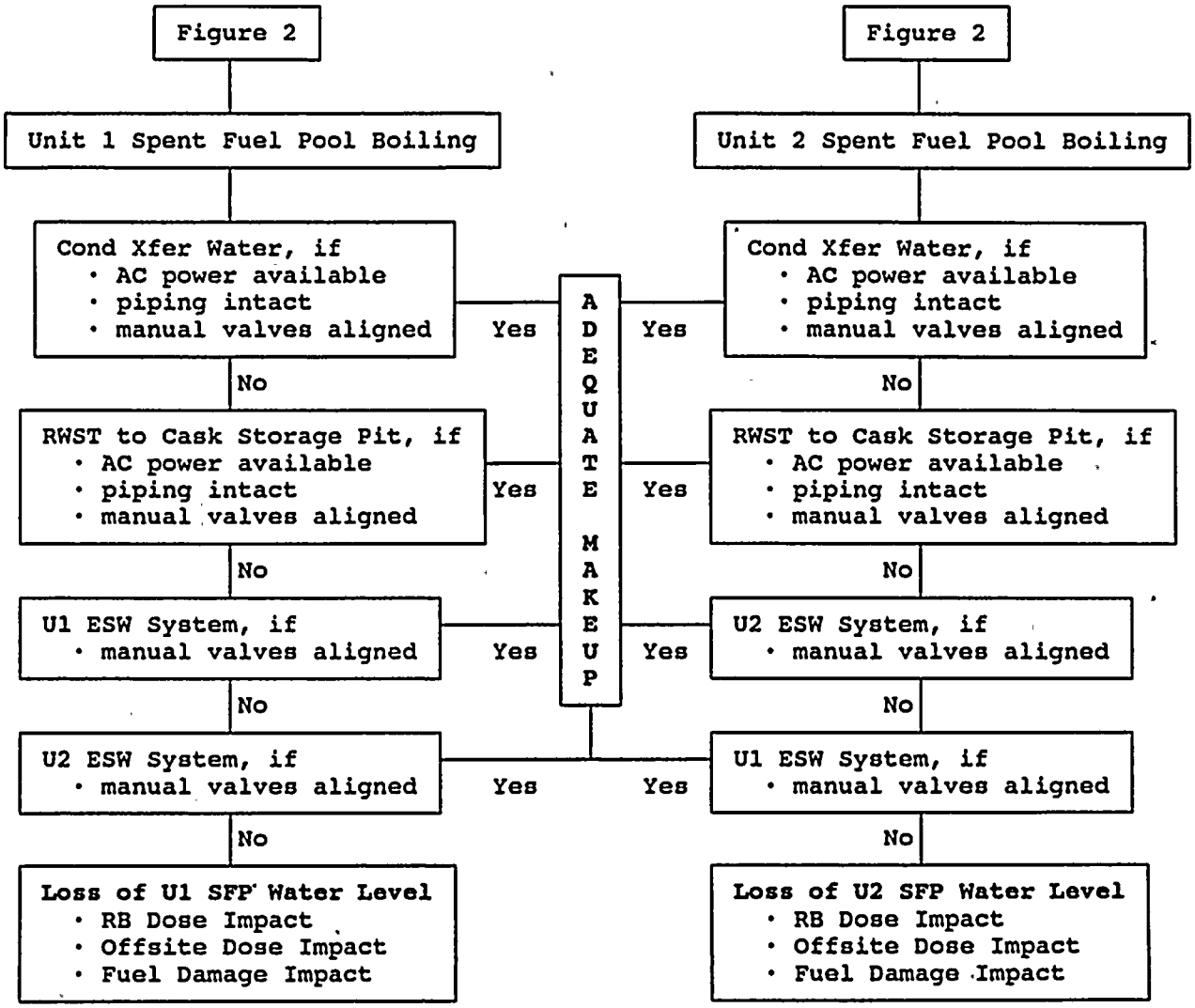
June 25, 1992

Figure 2
Fuel Pool Heatup



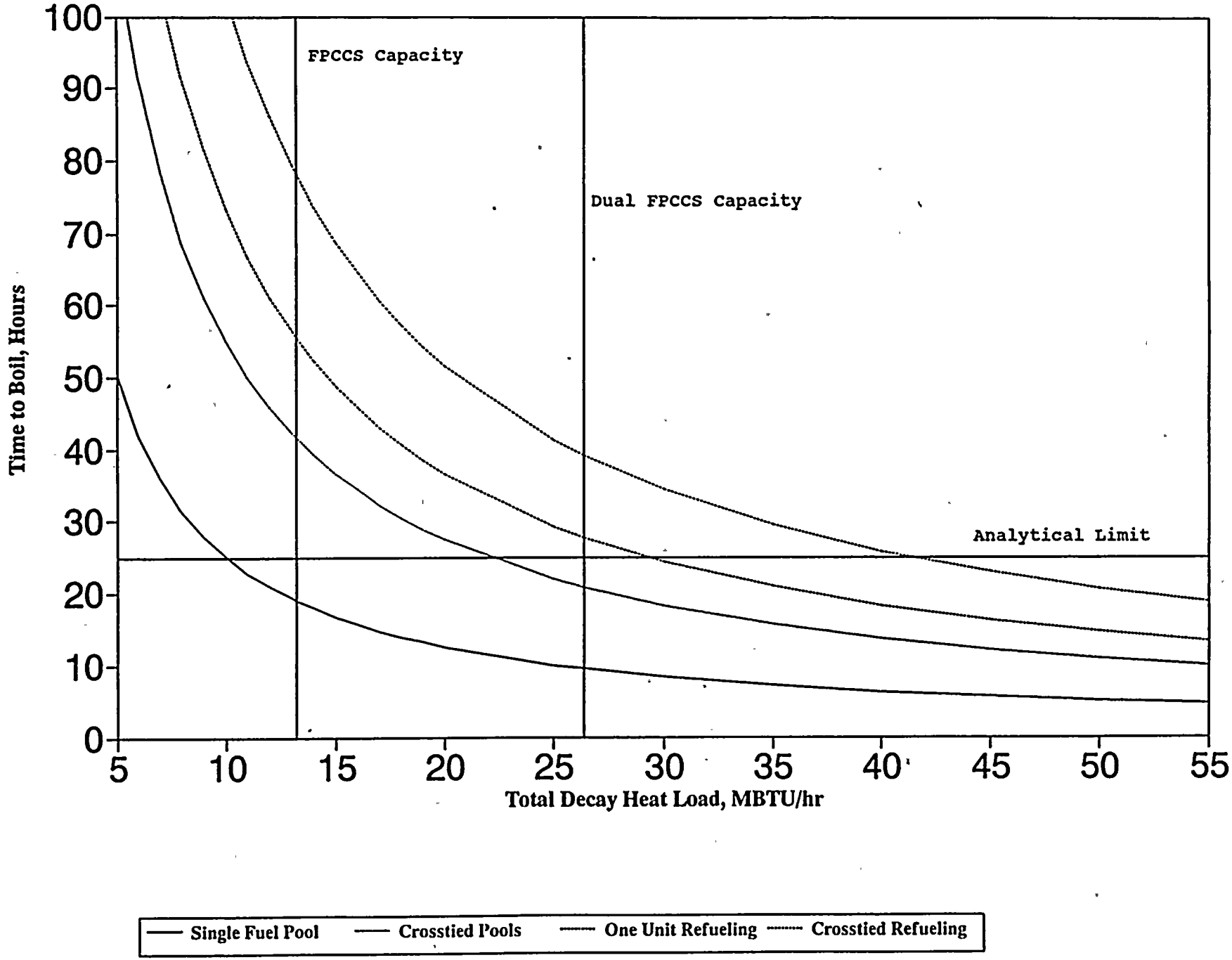
June 25, 1992

Figure 3
Fuel Pool Boiling & Makeup



June 25, 1992

Loss of Spent Fuel Pool Cooling



Attachment 8

White Paper prepared by David A. Lochbaum and Donald C. Prevatte, "Safety Consequences of a Boiling Spent Fuel Pool at the Susquehanna Steam Electric Station", July 27, 1992

Note: This paper was handed to the PP&L Manager of Nuclear Plant Engineering in a meeting requested by the authors. This paper was prepared when the authors became convinced that the PP&L Engineering Discrepancy Management Group and the PP&L Supervisor, Engineering Projects were unable to properly evaluate EDR G20020. The timing of this paper was dictated by the end of Mr. Lochbaum's contract at PP&L.