SAFETY CONSEQUENCES OF A BOILING SPENT FUEL POOL AT THE SUSQUEHANNA STEAM ELECTRIC STATION

July 27, 1992

Prepared by: David A. Lochbaum

Donald C. Prevatte

EXECUTIVE SUMMARY

Engineering Discrepancy Report (EDR) G20020 was written in April 1992 after nine potential problems associated with the boiling spent fuel pool event were identified during system evaluations to support the power uprate project for PP&L's Susquehanna Steam Electric Station. The major concerns raised in EDR G20020 are:

1. Regulations require that instrumentation shall be provided for the fuel storage systems to detect conditions that may result in loss of heat removal capability and to initiate appropriate safety actions.

   Contrary to this requirement, the water level and temperature instrumentation for the spent fuel pools do not satisfy Class 1E criteria and are not included in the equipment qualification program. These instruments will fail following a loss of offsite power and may fail following a loss of coolant accident. The ultimate consequence of such failure could be an irradiated fuel meltdown outside primary containment.

2. Regulations require that nuclear power plant designs limit personnel radiation exposures to $\leq$ 5 Rem per individual for control room occupation and actions required to mitigate or recover from an accident.

   Contrary to this requirement, the manual ESW valve manipulations required to provide makeup to a boiling spent fuel pool following a loss of coolant accident could require a radiation exposure significantly higher than 5 Rem. The ultimate consequence could be significant radiation overexposure or inability to provide ESW makeup and an irradiated fuel meltdown outside primary containment.

3. Regulations require that structures, systems and components important to safety be designed to accommodate the effects of the environmental conditions associated with postulated accidents.

   Contrary to this requirement, the effects of ESW makeup water to a boiling spent fuel pool have not been considered in the SSES design. The effects include flooding, high temperature, and high humidity. The ultimate consequences could include failure of multiple ECCS and other safety related systems.

4. Regulations require that electrical equipment be qualified to the temperature for the most severe design basis accidents.

   Contrary to this requirement, the SSES reactor building temperature analyses used in equipment qualification evaluations do not account for the heat load from a boiling spent fuel pool. The ultimate consequences could include failure of multiple ECCS and other safety related systems.

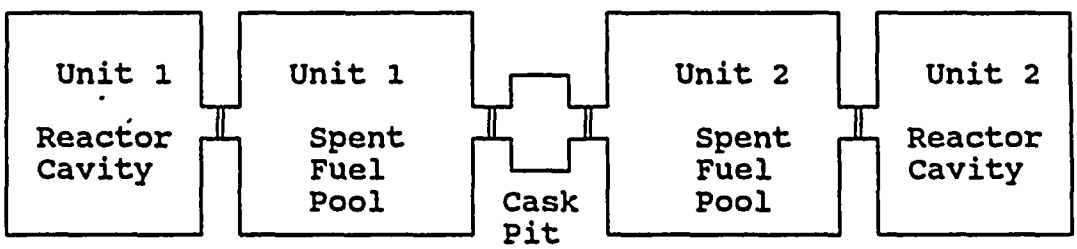
Safety Consequences of a Boiling Spent Fuel Pool

SYSTEM DESCRIPTION

Each of the two operating nuclear power plants at the Pennsylvania Power and Light (PP&L) Company's Susquehanna Steam Electric Station (SSES) has a spent fuel pool. Each spent fuel pool is designed to store up to 2,840 irradiated fuel bundles discharged from the reactor core after approximately four and a half years of operation. As of July 1992, the Unit 1 spent fuel pool contained 1400 irradiated fuel bundles and the Unit 2 spent fuel pool held 1004 irradiated fuel bundles.

The irradiated fuel bundles stored in the spent fuel pools generate heat from the nuclear decay of fission products. The amount of heat generation exponentially decreases with time as a function of the half life of the fission products.

The spent fuel pools are located in a common refueling area within the secondary containment structure. Each spent fuel pool is connected to a reactor cavity and to the other spent fuel pool. The reactor cavity is the area above the reactor pressure vessel which is flooded during a refueling outage after removing the drywell shield blocks, drywell head and reactor pressure vessel head to permit fuel transfer between the reactor core and the spent fuel pool. These connections are normally isolated, except during refueling outages, using gates.



Each spent fuel pool has a fuel pool cooling and cleanup system (FPCCS) which circulates water from the fuel pool through a heat exchanger and demineralizer to maintain proper fuel pool water chemistry and to keep its temperature $\leq$ 125°F. The FPCCS has a design capacity of 13.2×10^6 BTU/hr. As of July 1992, the decay heat load in the Unit 2 spent fuel pool was $\approx 2.1 \times 10^6$ BTU/hr while the decay heat load in the Unit 2 spent fuel pool was $\approx 2.97 \times 10^6$ BTU/hr. Heat from the FPCCS heat exchangers is transferred to the service water (SW) system which in turn dissipates the energy to the atmosphere via the cooling tower. The FPCCS and the SW system are non-safety related systems which are not designed to satisfy seismic, Class 1E power, equipment qualification and single failure criteria. The FPCCS is designed such that it cannot fail in a way which drains water from the spent fuel pool.

If the FPCCS is unavailable, the fuel pool cooling assist mode of the residual heat removal (RHR) system is designed to circulate water from the spent fuel pool through a heat exchanger to keep the fuel pool from boiling. The fuel pool cooling assist mode of RHR is manually initiated by opening valves in the reactor building. The fuel pool cooling assist mode of RHR has a capacity of 32.6×10^6 BTU/hr. Heat from the RHR heat exchanger is transferred to the RHR service water (RHRSW) which in turn dissipates the energy to the atmosphere via the spray pond. The fuel pool cooling assist mode of RHR is a non-safety related function which is not designed to satisfy seismic, Class 1E power, and single failure criteria. The fuel pool cooling assist mode of RHR is designed such that it cannot fail in a way which drains water from the spent fuel pool.

If both the FPCCS and the fuel pool cooling assist mode of RHR are unavailable, the spent fuel pool water will boil unless cooling is re-established. The time required to reach boiling is a function of the decay heat load in the spent fuel pool, the initial temperature of the water, and the volume of water available. The volume of water available is primarily dependent upon the presence or absence of the gates between the spent fuel pools and the reactor cavity. The emergency service water (ESW) system is designed to provide makeup to the boiling spent fuel pool to compensate for water lost through boil-off and evaporation. The ESW makeup supply is manually initiated by opening three valves in the reactor building. The ESW system uses water from the spray pond. The ESW system and the spray pond are safety related systems which are designed to satisfy seismic, Class 1E power, and single failure criteria as applicable. The design provision at SSES is for the ESW system to provide adequate makeup to a boiling spent fuel pool if cooling is lost.

The reactor building heating, ventilating and air conditioning (RB-HVAC) system circulates tempered air through each reactor building and the refueling zone during normal operation. The RB-HVAC system maintains these areas at a slight negative pressure relative to the outside environment to prevent leakage of potentially airborne radioactivity to the atmosphere. The exhaust from the potentially contaminated areas is filtered to remove radioactive materials. In an emergency, the supply and exhaust lines are isolated and the RB-HVAC system recirculates air throughout the reactor building affected by the emergency and the refueling zone. During a loss of offsite power (LOOP), the supply and exhaust lines are isolated and the RB-HVAC system recirculates air throughout the both reactor buildings and the refueling zone. The RB-HVAC system in recirculation mode does not provide any cooling function, so the reactor building and refueling zone air temperatures increase based upon piping, lighting, transmission and equipment heat loads.

The standby gas treatment system (SGTS) is designed to maintain the secondary containment at a negative pressure relative to the outside environment in an emergency. The SGTS takes suction on the recirculation plenum of the RB-HVAC system and processes this air through a filter train to remove radioactive materials. The SGTS is normally in standby except during testing. The SGTS is designed to satisfy seismic, Class 1E power, and single failure criteria.

The emergency core cooling systems (ECCS) and the reactor core isolation cooling (RCIC) system are located in the lower elevations of each reactor building. These systems provide water to the reactor pressure vessel during transients and accidents. These systems are normally in standby except during testing. The ECCS are designed to satisfy seismic, Class 1E power, and single failure criteria.

BOILING SPENT FUEL POOL DESIGN ANALYSIS

SSES Final Safety Analysis Report (FSAR) Appendix 9A reports the results of an analysis performed to quantify the radiological consequences of a loss of spent fuel pool cooling event. The analysis assumed the initiating event was an earthquake which resulted in the failure of the FPCCS on both units. The analysis concluded that the secondary containment design with SGTS operation kept offsite doses to a small fraction of 10 CFR 100 limits even with conservative assumptions of initial fuel failures in the spent fuel pools.

CONTAINMENT DESIGN ANALYSES

SSES Final Safety Analysis Report (FSAR) Chapter 6 reports the results of analyses performed to demonstrate the capability of the safety related systems to mitigate the consequences of postulated accidents such that the containment design parameters are not exceeded. The postulated accidents included main steam line breaks and loss-of-coolant accidents (LOCAs) with and without concurrent loss of offsite power. A design basis accident (DBA) for SSES is defined as a LOCA with a simultaneous LOOP and safe shutdown earthquake and the worst case single failure which results in the maximum containment pressure and temperature conditions. SSES FSAR Chapter 6 indicates margin to containment design parameters for the analyzed postulated accidents.

Reactor building room temperatures following postulated accidents were analyzed for equipment qualification. A procedure to manually shed all the non-Class 1E power loads in the reactor building ≈24 hours after a LOCA without a LOOP was developed to prevent room temperatures from exceeding equipment qualification limitations.

CONCERNS OVER BOILING SPENT FUEL POOL EVENT

Engineering Discrepancy Report (EDR) G20020 was written in April 1992 after nine potential problems associated with the boiling spent fuel pool event were identified during system evaluations to support the power uprate project. The four major concerns raised in EDR G20020 are:

1. **Inadequate Instrumentation**

   A. Regulatory 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) 9.1.3 for the spent fuel pool cooling and cleanup system states that the "*safety function to be performed by the system in all cases remains the same; that is, the spent fuel assemblies must be cooled and must remain covered with water during all storage conditions.*"

   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.*"

B. Concerns

The ESW system is required to provide makeup water to the spent fuel pools following loss of fuel pool cooling to keep the irradiated fuel covered thus preventing fuel damage from overheating. A loss of offsite power or LOCA can result in loss of fuel pool cooling since the FPCCS and the SW system are not normally supplied by Class 1E power. The loss of offsite power will also disable the spent 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. The post-LOCA environment in the reactor building may disable the spent fuel pool temperature and level instruments since they are not covered under the equipment qualification program. Therefore, the existing spent fuel pool temperature and level instrumentation is inadequate to ensure the required safety action of providing adequate makeup to a boiling spent fuel pool is properly initiated and monitored under all postulated accident conditions.

If the spent fuel pool is permitted to boil without adequate makeup, its water level will drop. A study by the PP&L Nuclear Safety Assurance Group (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.*" At that dose rate, an individual on the refueling floor would receive a lethal radiation exposure in approximately 16 seconds. This severe condition is just the beginning of the adverse consequences of spent fuel pool boiling without adequate makeup. At this point, the radiation source term results in offsite doses exceeding 10 CFR 100 limits and in dose rates within the reactor building that prevent any personnel access. The situation progresses ultimately to uncovering irradiated fuel bundles in the spent fuel pool and fuel damage from overheating. The situation has the potential for a substantial meltdown of irradiated fuel outside the primary containment.

2. Manual ESW Valve Operation

A. Regulatory Requirements and Licensing Commitments

10 CFR 20.1 requires licensees to "*make every reasonable effort to maintain radiation exposures ... 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.*"

SSES FSAR 18.1.20 in response to 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.*"

SSES 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.*"

SSES 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.*"

SSES 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.*"

SSES 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 - all activities*
Infrequent	*500 mR/hr*	*5 Rem per activity*
Accessway	*5 R/hr*	*Incl in above doses"*

SSES 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.*"

SSES 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. SSES 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. These valves must be manually opened to initiate ESW makeup to the spent fuel pools in the loss of fuel pool cooling event.

PP&L administrative procedure 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.*" 10 CFR 20's ALARA provision requires plant design to minimize radiation exposure. Application of the emergency dose guidelines to a design which requires manual valve operation is contrary to the intent of 10 CFR 20.1 and 10 CFR 50 App A GDC 19.

B. <u>Concerns</u>

The ESW system is required to provide makeup to the spent fuel 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 dose rates in the reactor building areas where the manual valves are located are 5 to 5,000+ 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. Therefore, the manual ESW valve manipulations required for makeup to boiling spent fuel pools may not be accomplished for all postulated accident conditions.

In addition, since the boiling spent fuel pool analysis reported in SSES FSAR Appendix 9A assumed a seismic event initiated the loss of fuel pool cooling, the intentional shedding of non-Class 1E power loads in the reactor building following a LOCA without a LOOP represents either the creation of a new kind of accident or the increased probability of a previously analyzed accident.

3. **Effects of ESW Makeup Water on Reactor Building Systems**

A. Regulatory Requirements, Licensing Commitments and Design Bases

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.*"

SSES 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.*"

SSES 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.*"

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.*"

An internal PP&L engineering work request (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?" The response to these questions was "*This is an inappropriate format to ask questions. Comments were requested and none received. Furthermore, no budget exists with which to fund the engineering time required to respond to these questions.*"

B. Concerns

The ultimate heat sink and ESW are designed to provide 1.5 million gallons of water to each spent 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 spent fuel pools ends up in the reactor building following boil-off and overflow. The effects of this water on the safety-related structures, systems and components in the reactor buildings have not been included in design analyses. The ECCS and RCIC room coolers are known not to be designed for latent heat effects. Dampers in the SGTS and RB-HVAC system close when the entering air temperature exceeds 165°F, while the boiling spent fuel pool was calculated to produce air temperatures of ≈180°F. The potential for common mode failures of multiple ECCS and safety-related systems such as the standby gas treatment system exists. Failure of one or more of these safety-related systems could increase the consequences of postulated accidents.

4. Reactor Building Heat Loads

A. Regulatory Requirements, Licensing Commitments and Design Bases

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.*"

An internal PP&L engineering work request (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?*" The response

to this question was "*This is an inappropriate format to ask questions. Comments were requested and none received. Furthermore, no budget exists with which to fund the engineering time required to respond to these questions.*"

A PP&L engineering report (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. 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 equipment qualification temperature limits 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. Even the current heat loads in the spent fuel pools could increase the maximum heat load in the reactor building by ≈50%.

The remaining five concerns raised in EDR G20020 involved nonconservatisms in analyses for the boiling spent fuel pool event.

DISCUSSION OF OPPOSING VIEWPOINT

The discussions and meetings which have occurred since EDR G20020 was initiated have yielded one primary argument against the issues raised in EDR G20020 having nuclear safety significance. This argument is that the licensing bases LOCA/LOOP accident for SSES does not assume a boiling spent fuel pool resulting from the event. In order for this assumption to be valid, spent fuel pool cooling must either not be lost or must be restored prior to boiling. There are several faults in this assumption:

1) Since the FPCCS and the SW system are non-safety related systems, their components are not included in the equipment qualification program and may not survive the pressure, temperature, humidity and radiation environment in the reactor building following a postulated accident. Therefore, the

FPCCS which is definitely lost following a LOCA/LOOP may also be lost following a LOCA without a LOOP.

2) Since the fuel pool cooling assist mode of RHR is a non-safety related function, its components are not included in the equipment qualification program and therefore may not survive the pressure, temperature, humidity and radiation environment in the reactor building following a postulated accident. In addition, the fuel pool cooling assist mode of RHR has not been utilized since the initial startup testing program and its valves were removed from the inservice inspection program several years ago and the valves may have experienced failures which have not yet been detected which would prevent their successful operation. Therefore, the fuel pool cooling assist mode of RHR may be lost following a LOCA/LOOP and a LOCA without a LOOP.

3) The fuel pool cooling assist mode of RHR requires the manual opening of valves in the reactor building which may be inaccessible following a postulated accident due to radiation levels.

4) For the LOCA/LOOP case, it has been argued that the SSES design implicitly assumes restoration of offsite power typically within 24 hours and essentially always within 48 hours after event initiation. SSES FSAR Chapter 8 reports PP&L grid experience in support of these restoration times. However, no documentation was found which states that PP&L has defined the LOOP duration for design bases events. As EDR G20020 and EDR G00005 both address, the spent fuel pool may begin boiling in less than 24 hours. In any case, the reactor building temperature analyses for equipment qualification purposes presently counter any such credit for restoration of offsite power since non-Class 1E power loads in the reactor building may be shed ≈24 hours after offsite power is restored in order to satisfy room temperature limitations.

EDR G20020 identified concerns with the SSES design provisions for the boiling spent fuel pool event. The SSES design, coupled with current operating procedures, would have significant nuclear safety consequences if a loss of spent fuel pool cooling occurred. Therefore, these concerns must be resolved for SSES. In addition, many of these concerns are applicable to other BWRs and possibly even PWRs in the United States. Therefore, these concerns must be reported to INPO/NRC in order for the adverse condition to be remedied throughout the industry.

Attachment 9

PP&L Memo from G. D. Miller to G. T. Jones, "Fuel Pool Cooling Deficiencies", August 18, 1992 (ET-0586)

Note: This memo by the PP&L Supervisor, Engineering Projects provides an indication of how PP&L narrowed their scope of evaluation for the concerns in EDR G20020 to just the design of the fuel pool cooling and cleanup system. With the exception of the instrumentation for the fuel pools, the design of the fuel pool cooling and cleanup system has not been challenged in EDR G20020 and its subsequent supporting documents. The concerns are that the effects of boiling spent fuel pools on <u>other</u> systems and components in the reactor building have not been adequately analyzed.

M E M O R A N D U M

DATE: 8/18/92

TO: G. T. Jones A6-2

FROM: G. D. Miller A6-3

JOB: Engineering Technology

NUMBER: ET-0586

COPIES: Distribution
Corres. File A6-2
Engr Tech File A6-3

FILE: A45-1A

REPLY: Not applicable

SUBJECT: Fuel Pool Cooling Deficiencies

The following actions are being taken in response to concerns raised about the EDR program and the fuel pool cooling issues described in EDR G20020 and EDR G00005.

EDR Process Changes

The EDR process governing procedure, EPM-QA-122, will be revised as follows:

1. The appeal process described in paragraph 5.4 will be revised with the first step of the appeal changed to the Engineering Review Committee. Subsequent appeal may be to either the Manager - Nuclear Engineering, Superintendent - SSES, or Manager - NSAG.

2. The management review described in paragraph 5.14 will be revised to include:

   a. Periodic review of all invalid EDRs by the Engineering Review Committee, and

   b. Periodic review of all EDRs open greater than six months by the Engineering Review Committee.

Fuel Pool Cooling Deficiency Resolution

Engineering Technology has responsibility to resolve both EDRs (EDR G00005 was previously assigned to System Engineering). These EDRs are assigned to Mark Mjaatvedt and are being worked full time by Michael Crowthers. The EDR

M E M O R A N D U M
(CONTINUATION SHEET)

evaluations (safety significance, operability/reportability) are being done by Jim Agnew and Joe Zola.

Our initial evaluation of these EDRs has concluded that the safety significance is minimal. This is based primarily on our understanding that the design of the fuel pool cooling system was specifically reviewed and approved by the NRC with full knowledge of the fact that the FPCCS was not a safety related system and that fuel pool boiling could be expected to occur under a specific combination of hypothetical conditions. However, it is not yet clear to what extent the NRC (or the industry) considered the long-term effects of the fuel pool boiling condition. The record on this subject is confusing and further complicated by changes to the fuel design and actual outage practices which have not been accounted for in the FSAR analysis. We believe that the design and procedural features which exist today provide a reasonable level of assurance that the actual safety consequences are minimized. However, procedure enhancements and additional operator training are clearly required as part of the resolution of these concerns. This evaluation will be fully documented as part of the revised EDR package.

A final evaluation of this concern is predicated on completion of a historical review of all available documentation. Thus, our plan for resolution includes:

1. Complete investigation of historical design and licensing information, including requests for information from the original design organizations (GE and Bechtel),

2. Establish the fuel pool and fuel pool cooling system design basis based on the derived design basis and current operating practice,

3. Review fuel pool designs of other boiling water reactors,

4. Complete a new analysis of the fuel pool and fuel pool cooling system based on the established design basis,

5. Prepare a point-by-point comparative description of our current operating practice and design basis against the original FSAR analysis (including the individual issues identified in the EDR),

6. List and assess each deviation from the original analysis as described in item 5,

7. Prepare an operability evaluation accounting for each deviation as additional information becomes available,

8. Re-evaluate all issues for reportability (ongoing),

9. Prepare recommendations to resolve each issue described in item 4.

M E M O R A N D U M
(CONTINUATION SHEET)

We plan to keep the originators of these concerns informed of our progress as we work our way through this effort. A formal plan including schedule for the above activities is under development.

Independent Review by Systems Analysis

Systems Analysis (Kevin Brinckman) is in the process of conducting an independent design review of these issues. Additionally, I have requested their review of this issue from an IPE perspective when resources become available.

Assessment of EDR Process

As a followup activity I plan to request an assessment of the EDR process from an independent organization. This assessment will focus specifically on the validation and verification steps of the process. This should be conducted by NSAG. NQA has once in the past conducted an audit of the process. They do not agree that discrepancies are not necessarily deficiencies. They define any discrepancy to be a condition adverse to quality, whereas our program recognizes the potential for discrepancies in documentation which do not constitute actual deficient conditions. Other programmatic audits have taken place on the EDR process, but none have examined the philosophy or criteria used to determine the validity of engineering issues.

Attachment 10

PP&L Memo from D. C. Prevatte to G. T. Jones, "Fuel Pool Cooling Deficiencies", August 20, 1992 (ET-0587)

M E M O R A N D U M

TO:	G. T. Jones A6-2	DATE:	August 20, 1992
FROM:	D. C. Prevatte A6-3	FILE:	A45-1A
JOB:	Engineering Technology	cc:	Distribution Corres. File A6-2
NUMBER:	ET-0587 REPLY:		ET File A6-3
SUBJECT:	Fuel Pool Cooling Deficiencies		

This memo is written in response to Mr. G. D. Miller's memo ET-0586 of 8/18/92 concerning the discrepancies associated with the fuel pool cooling system described in EDRs G20020 and G00005.

Mr. Miller's memo states that, "Our initial evaluation of the EDRs has concluded that the safety significance is minimal." I strongly disagree with this evaluation and I hereby request that the safety significance of these EDRs, particularly EDR G20020 be reevaluated for the following reasons:

1. The primary basis given for this conclusion is that "...the design of the fuel pool cooling system was specifically reviewed and approved by the NRC with full knowledge of the fact that the FPCCS was not a safety related system and that fuel pool boiling could be expected to occur under a specific combination of hypothetical conditions."

   I consider this basis to be invalid for the following reasons:

   a. This basis appears to miss most of the main points of EDR G20020. It focuses on the non-safety related FPCCS which is not the concern. The primary concerns are with the NRC mandated (Reg. Guide 1.13), safety-related backup cooling scheme of allowing the fuel pool to boil and providing makeup water from the safety-related ESW system. The concern is the potential inability of the operators to put this scheme into effect because of inaccessibility to the associated valves due to post-LOCA radiation levels in the reactor building, and the potential negative effects of a boiling spent fuel pool on virtually all of the safety-related systems in the reactor building, effects which have not been analyzed.

   b. The "... specific combination of hypothetical conditions ..." referred to in the memo is LOCA/LOOP. This is not some off-the-wall accident scenario as the response seems to imply. This is the standard, universally recognized, NRC mandated design basis accident (DBA).

The conditions of concern are not postulated. They are mechanistic consequences of that DBA.

2. The memo also cites as another basis, confusion concerning what was originally considered in the design, the changes to the fuel design, and outage practices which have not been accounted for in the FSAR analysis. This basis would seem to support the EDRs contentions, not refute them. If indeed there is confusion on these subjects, then, at best, the condition is unanalyzed and therefore by definition in NRC regulations and our procedures, a safety concern.

   Although this information is certainly pertinent to a historical perspective of these concerns, to determination of the magnitude of the heat loads involved, and to formulation of the corrective actions that may be effected, it is not pertinent to the ability of the plant to perform as required for the DBA conditions. The information to make this determination is clear and available today.

3. Another basis cited is that "... the design and procedural features which exist today provide a reasonable level of assurance that the actual safety consequences are minimal." There is no elaboration on what these design and procedural features are. In conversations with Mr. Miller and others who seem to consider the EDRs as having very low safety significance, no design or procedural features have been cited. The only features that have been cited are "heroic action" of the operators, the EOPs, an EOC staff who will understand the concern and do whatever needs to be done, and a low probability of occurrence.

   These are not valid features. Heroic operator action is not a valid basis for the design of a plant, nor are the EOPs (even if they were correct in this area) which address many conditions potentially outside the plant licensing and/or design bases.

   And, contrary to the memo's contention, the EOPs as they stand today are not correct. They currently tell the operator he has a minimum of 25 hours until the fuel pool boils. Under worst case conditions, it may be less than half that time; and with the LOOP conditions, he has no instrumentation to tell him the condition of the fuel pool. Under DBA conditions, the operator is flying blind using nonconservative information.

   Additionally, current EOPs move the plant toward the conditions of concern, not away from them. The current EOPs require deenergizing the non-1E loads in the reactor building

at t=24 hours if reactor building temperatures are as analyzed. This, in effect, imposes a LOOP on the reactor building, thus initiating the conditions of concern.

The knowledge of the EOC staff is also not a valid feature if the conditions of concern are not formally addressed in any official design and/or procedural documents. Although it is claimed that today's staff would understand the concerns, there is no reason to believe this is true since the concerns aren't documented outside the EDRs and there is no training on this eventuality. And ten years from now, if the concerns are not formalized in writing, they will be even less understood.

Additionally, even if the staff does understand, if the conditions are not analyzed, which they are not, the plant could be brought to a condition where recovery is not possible in spite of their full understanding.

Low probability is also not a valid feature. This is discussed in detail further in this memo.

4. The statement in the memo regarding the NRC's "... full knowledge of the fact that the FPCCS was not a safety-related system and that fuel pool boiling could be expected to occur ..." seems to imply that if the NRC approved it as is, that makes it acceptable even if we discover discrepancies that may not have been originally considered. I am aware of no evidence that indicates the NRC approved of our design with the understanding that: (a) the operator would be exposed to unacceptable radiation levels under design basis conditions in effecting the FSAR described fuel pool boil scheme for alternate cooling; and (b) that the boiling fuel pool might create a myriad of unanalyzed conditions in the reactor building that could threaten the operability of many of the safety-related systems in the building.

5. The memo concludes that the safety significance is "minimal." Per procedure EPM-703, Rev. 0, Section 5.3, "...a 'minimal' classification generally signifies a documentation type of discrepancy." In other words, not a real engineering concern, but rather a documentation error that can be resolved by making editorial changes to the documents. Per this procedure, if an EDR's safety significance is classified as "minimal", it does not even have to be evaluated for operability and reportability.

   These EDRs are not in any reasonable evaluation just a documentation discrepancy. They are fundamental engineering concerns raised by two engineers intimately familiar with the systems after exhaustive research. To dismiss these concerns

by classifying them as just a documentation discrepancy is ludicrous. If concerns such as these don't even get to the stage in the process where they are required to be evaluated for operability and reportability, concerns which involve the safety of the operators and potential threat to virtually every safety-related system in the reactor building, then what does it take to trigger operability and reportability evaluations? The threshold appears to be much too high.

6. The memo's reference to "... a specific combination of hypothetical conditions ..." implies a probabalistic argument as to why the safety significance is "minimal." Indeed, in conversations with Mr. Miller and others this argument has been explicitly raised. This argument is not valid with regard to design bases for several reasons.

   First, our design basis conditions of LOCA/LOOP which produce the conditions of concern are mandated by regulation. That, for design purposes, dictates a probability of 1.

   Second, even for LOCA without a LOOP, our current EOPs dictate a self-imposed LOOP on the reactor building at 24 hours, again making the probability for LOCA/LOOP equal 1.

   Third, even for a LOOP or FPCCS system failure without a LOCA, the consequences of fuel pool boil are unanalyzed.

   Fourth, EDR procedure EPM-703, Rev. 0, Section 5.3, has as a caution, capitalized, bold letters and underlined as follows, "The EDMG Evaluator must not put heavy emphasis on the perceived small probability of occurrence or the expected satisfactory outcome of analysis or reanalysis to justify continued operation with the existing discrepancy." Section 5.4 goes on to say, "SAFETY SIGNIFICANCE must be based on the potential adverse consequences of failures, even those of very low probability." Thus, by our procedures, potential consequences should be the dominant factor in evaluating safety significance, not probability. The potential consequences of the concerns raised in these EDRs, and subsequent documentation generated by Mr. Lochbaum and myself, are very grave.

The EDR process at PP&L was developed in response to a 1990 SALP inspection finding that safety significant issues were not being handled in a timely manner. Our EDR procedures are filled with words that reflect this concern; words like quickly, expeditiously, immediately, early, timely. For the step where we are today, the "screening" step, the procedures' (EPM-703, Rev. 0, Section 5.2)

intent is that EDRs be "quickly" screened after a discrepancy enters the EDR process. As of today, the official screening still has not been completed four months after the EDR entered the process, and approximately one month after Mr. Lochbaum and I personally brought these concerns to your attention. Neither the intent of the procedure nor the intent of the NRC are being fulfilled.

The plan outlined in Mr. Miller's memo for the "final evaluation" would appear to further delay the required actions. Although all of the activities in the plan are important to understanding the problems more completely and effecting the most effective solutions, none of them are prerequisites for performing a valid "screening", and most of them are not required to determine operability and reportability. To make these activities prerequisites for a final "screening" evaluation and then for the operability and reportability determinations, is to further delay the process unnecessarily. The information is available to make these determinations today, and they should be made immediately if we are to do what's legitimately required of us.

I have made my own operability and reportability determinations based on extensive research on these concerns. At best, the operability of the fuel pool cooling in the boil and feed mode, the fuel pool instrumentation, and much of the safety-related equipment in the reactor building is unanalyzed with regard to the effects of the boiling fuel pool on this equipment, with strong indications that analysis would show it as inoperable. If this is the case, per 10CFR50.72 and 50.73, it is reportable.

I would welcome any hard, definitive, documentary information indicating that my conclusions are wrong. Both Mr. Lochbaum and I, and for that matter, many others who would like to see different conclusions, have searched for contrary evidence. To the best of my knowledge, none has been found.

This is not to say that the plant should necessarily be shut down. I believe that very credible arguments can be made for a J.I.O. I therefore don't understand why there is such an apparent reluctance in the organization to acknowledge these concerns and move ahead with resolution expeditiously. Although resolution will have a cost, certainly, that cost does not necessarily have to include plant shutdown.

I therefore strongly urge that the formal screening evaluation and the evaluations of the operability and reportability of these concerns proceed without further delay with priority over all other activities in Mr. Miller's plan, and that we expeditiously get on with the process of resolving these concerns.

I sincerely appreciate your continued personal attention in these matters, and I am at your service in addressing these concerns.

Donald C. Prosetti

DISTRIBUTION:

J. E. Agnew	A6-3	G. D. Miller	A6-3
F. G. Butler	A6-3	J. R. Mittenberger	A6-1
R. G. Byram	A6-1	M. R. Mjaatvedt	A6-3
M. H. Crowthers	A6-3	C. A. Myers	A2-4
G. D. Gogates	SSES	J. G. Refling	A6-3
		J. S. Stefanko	A9-3
G. J. Kuczynski	SSES, S&A	T. J. Sweeney	SSES
D. A. Lochbaum	Enercon	J. A. Zola	A6-3

FUELPOOL.DCP/kbw

Attachment 11

PP&L Memo from A. Dyszel to T. C. Dalpiaz, "U2 RI05 Fuel Pool Decay Heat Evaluation", August 21, 1992 (PLI-72230)

Note: This letter transmits interim guidance to the SSES site personnel for use during an upcoming refueling outage. This guidance is necessary because EDR G00005, initiated in September 1990, has not yet been dispositioned and the applicable discussions in FSAR Section 9.1 and Appendix 9A are no longer accurate.

PP&L

August 21, 1992

T. C. Dalpiaz SSES

SUSQUEHANNA STEAM ELECTRIC STATION
U2 R105 FUEL POOL DECAY HEAT EVALUATION
CCN 741087 FILE[redacted]
PLI-72230

References: 1) PLI-67513, "Fuel Pool Cooling Heat Loads - EDR G00005," April 12, 1991.

2) PLI-70395, "U1 R106 Fuel Pool Decay Heat Evaluation," February 7, 1992.

3) ET-0586, "Fuel Pool Cooling Deficiencies," 8/18/92.

This memo provides short term relief for the open EDR, G00005, (Reference 1) with respect to the upcoming U2 5RIO by providing NFE's evaluation of the resulting time constraints for performance of common RHR system outage work, similar to the U1 R106 evaluation (Reference 2). A long term solution to this EDR is required to ensure future successful, timely outages. In short, the U2 5RIO evaluation shows no change in the current outage schedule (i.e., Sept. 28 for common RHR system work) for no fuel failures in Unit 2 Cycle 5. However, future evaluations, which will involve higher heat loads and address the potential for fuel failure(s), will likely affect the outage schedule. Reference 3 provides a description of the long term solution to this problem.

ISSUE

FSAR Section 9.A describes the radiological release results from a loss of fuel pool cooling event. The assumptions used for the FSAR analysis include ¼ core reloads, in core fuel shuffling, and a maximum fuel exposure of 28,500 MWD/MTU. The current operation at Susquehanna SES includes a fuel reload batch size of approximately ⅓ of the core, a maximum fuel discharge exposure of 40,000 MWD/MTU, and a full core offload for each outage. If one or more fuel failures are suspected to have occurred during the operating cycle just prior to an outage or during fuel handling after shutdown, analyses must be performed to assure the radiological release from the postulated loss of fuel pool cooling event are less than those presented in the FSAR. The FSAR analyses are bounding provided that common RHR system outage work is not started until the decay heat level is low enough to prevent fuel pool boiling in less than 25 hours. If no fuel failures are suspected to have occurred during the operating cycle just prior to an outage or during fuel handling after shutdown, analyses must be performed to assure the fuel pool water level can be maintained during a loss of fuel pool cooling event. The fuel pool

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water level can be maintained provided that common RHR system outage work is not started until the ESW makeup flow rate to the spent fuel pool (i.e., 60 GPM) is greater than the spent fuel pool boiling rate during the loss of fuel pool cooling event.

U2 5R10 ISSUE RESOLUTION

To address this issue for the upcoming outage, Nuclear Fuels Engineering has calculated the total decay heat of the fuel in the spent fuel pools for the U2 5R10 consistent with the approach in Reference 2. This decay heat level includes a total of the decay heat from the fuel in the U1 pool, U2 pool, and the full core offload. Figure 1 shows the calculated spent fuel pool decay heat as a function of time after shutdown. The curve labelled "nominal" is a calculation of the decay heat based on the methodology in NUREG-0800. This methodology has been shown to produce higher decay heat levels than the more rigorous methodology in the ANS 5.3-1979 decay heat standard. The curve labelled "maximum" is also based on the methodology in NUREG-0800 but accounts for uncertainty in the reactor power level (1σ) and uncertainty in the decay heat methodology.

Nuclear Fuels Engineering has also performed a calculation to determine the time after the U2C5 shutdown that the spent fuel pool boiling rate during a postulated loss of spent fuel pool cooling event is less than the 60 GPM ESW makeup flow rate. Based on the "maximum" decay heat curve in Figure 1, a spray pond temperature of 90°F, and open fuel pool gates, the fuel pool boiling rate during a postulated loss of fuel pool cooling event is less than the 60 GPM ESW makeup rate subsequent to 14 days after reactor shutdown. Therefore, NFE's evaluation indicates that provided fuel failure does not occur during the remaining U2C5 operation or during fuel handling after shutdown, common RHR system outage work should not start until at least 14 days after reactor shutdown. If a fuel failure occurs, a calculation should be performed to determine if further outage restrictions are necessary. Note that the above calculations have been documented in NFE-8-NA-046, Rev. 2 and independently reviewed in accordance with QA procedures EPM-QA-301.

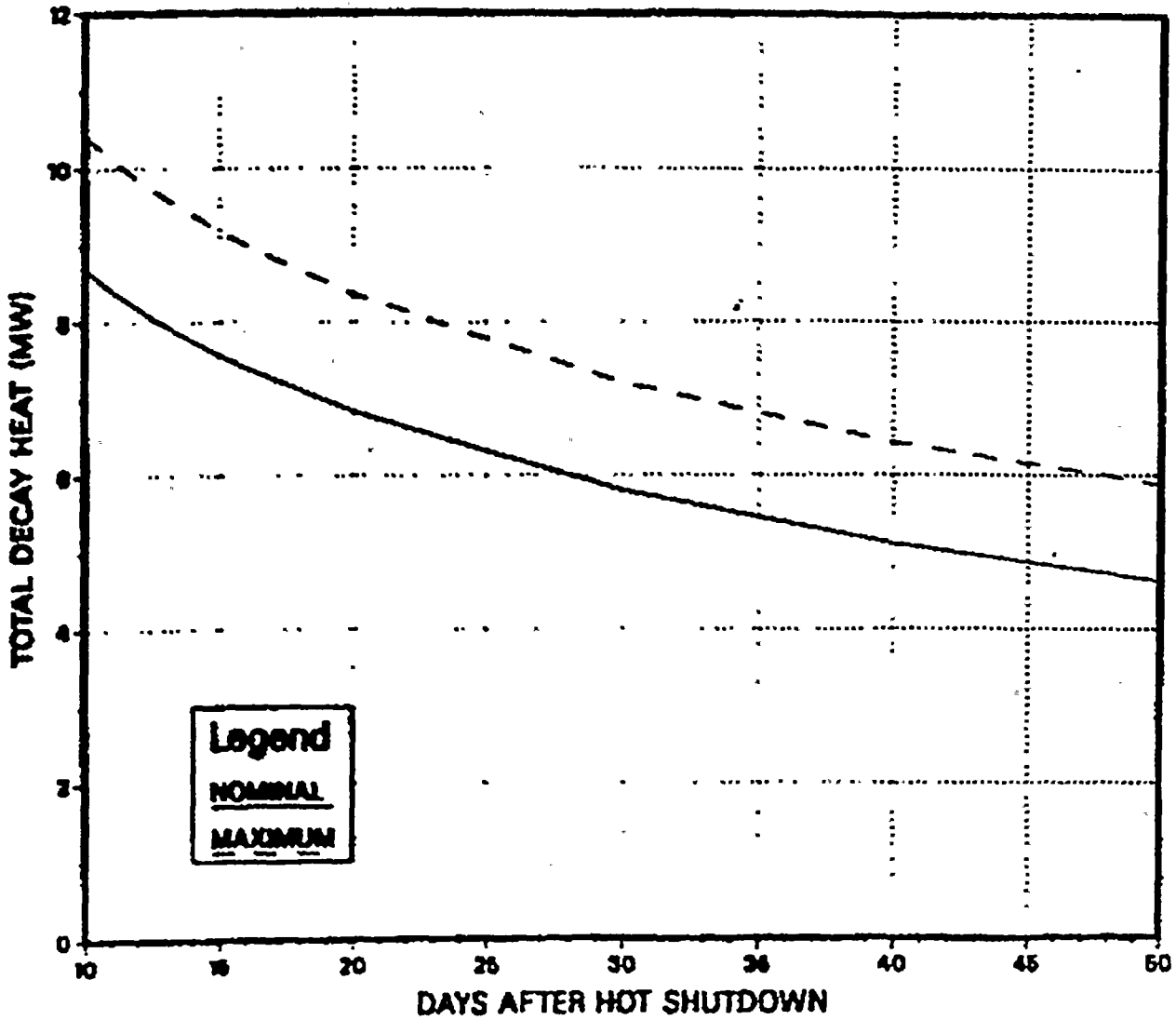
The 14 day restriction on commencement of common RHR system work does not impact the current U2 5R10 outage schedule.

A. Dyszel
Nuclear Fuel Management Project Engineer
Nuclear Fuels Engineering

AD/el
mel371a.ad

cc:	J. E. Agnew	SSES	A. J. Roscioli	A9-3
	K. W. Harwanko	SSES	R. A. Saccone	SSES
	G. T. Jones	A6-2	J. P. Spadaro	A9-3
	J. M. Kulick	A9-3	J. S. Stefanko	A9-3
	C. R. Lehmann	A9-3	J. Zola	A6-3
	G. D. Miller	A6-3	NR File	A6-2

FIGURE 1
U2 5RIO SPENT FUEL POOL DECAY HEAT
(U1 PLUS U2)



Attachment 12

PP&L Memo from J. M. Kenny to G. T. Jones and C. A. Myers, "EDR on Fuel Pool Cooling", August 25, 1992

Note: This 'confidential' memo is the first documented indication that the NRC had been informally notified of the concerns raised in EDR G20020.

PP&L

CONFIDENTIAL

August 25, 1992

G. T. Jones A6-2
C. A. Myers A2-4

EDR ON FUEL POOL COOLING

On August 24, 1992 I briefed both Scott Barber and Jim Raleigh of the NRC on the status of our review of contractor originated fuel pool cooling concerns documented on an EDR. I noted that our current position was there were no immediate concerns with system operability or need for reportability under regulations identified but that our efforts were continuing to address the identified issues. I also noted that George Jones had discussed the concerns with the contractors and was personally involved in resolving the issues.

I had previously briefed Jim Raleigh in July of the fuel pool concerns and reviews being performed. Scott did bring to my attention an open inspector finding concerning the Haddam Neck fuel pool draindown event and subsequent efforts by NSAG on fuel pool issues. He noted there were 28 open items and that we should review these issues for status. I indicated it was my understanding Engineering would be addressing the NSAG open issues on the fuel pool as part of their effort to resolve the open EDR.

J. M. Kenny

cc: J. E. Agnew A6-3
G. D. Miller A6-3
J. R. Miltenberger A6-1
R. R. Sgarro A2-4
H. G. Stanley SSES

JMK:tah
FuelPool.EDR

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Attachment 13

PP&L Memo from George T. Jones to Glenn D. Miller, "Fuel Pool Cooling EDR's G20020, G00005", August 27, 1992 (PLI-72267)

PP&L

RECEIVED SEP 0 9 1992

August 27, 1992

Glenn D. Miller A6-3

FYI

SUSQUEHANNA STEAM ELECTRIC STATION
FUEL POOL COOLING EDR'S G20020, G00005
PLI-72267 FILE A45-1A

Reference: ET-0587, ET-0586

The reference letters show me that there still is a difference of professional opinion relative to the significance of the subject EDR's. This difference is to be resolved. This issue is to be worked expeditiously until we have resolved the outstanding questions.

I wish to point out that our belief that a design has been reviewed and approved by NRC, is not adequate justification for classification of the significance of an issue. The issue must stand on it's own merits.

I am further concerned that our actual configuration and method of operation apparently differs from that described in the FSAR. The FSAR is our licensed bases and any deviation from that description is required to have a thorough and complete documented evaluation on file.

There were twenty-eight open items resulting from NSAG Review of Fuel Pool Cooling. These need to be included in this review.

I wish to have the schedule for resolution of this issue accelerated and the classification of the significance of this issue reevaluated. I am expecting at least daily updates of our progress. ~~Please reply back to me by August 31, 1992.~~

If you have any questions, please call me.

G. T. Jones

George T. Jones

Attachment

cc:	J. E. Agnew	A6-3 w/a	D. C. Prevatte	A6-3 w/a
	F. G. Butler	A6-3 w/a	J. R. Miltenberger	A6-1 w/a
	R. G. Byram	A6-1 w/a	M. R. Mjaatvedt	A6-3 w/a
	M. H. Crowthers	A6-3 w/a	C. A. Meyers	A2-4 w/a
	G. D. Gogates	SSES w/a	J. G. Refling	A9-3 w/a
	G. J. Kuczynski	SSES, S&A w/a	T. J. Sweeney	SSES w/a
	D. A. Lochbaum	Enercon w/a	J. A. Zola	A6-3 w/a
	S. M. Hauseman	A6-2 w/a	Nuc. Rec. Files	A6-2 w/o

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c:\wp51\docs\20020&5.sgt 8/26/92

Attachment 14

PP&L Memo from Glenn D. Miller to George T. Jones, "Fuel Pool Cooling EDRs G20020, G00005", August 31, 1992 (PLI-72297)

PP&L

August 31, 1992

George T. Jones A6-2

SUSQUEHANNA STEAM ELECTRIC STATION
FUEL POOL COOLING EDRs G20020, G00005
PLI-72297 FILE A45-1A

Reference: PLI-72267, ET-0587, ET-0586

In response to your letter PLI-72267 we are continuing to work to resolve the issues in the referenced EDRs. As I explained to you previously I met with Mr. Prevatte and discussed his concerns at length on August 21. We acknowledged our differences and agreed to continue working toward resolution.

I want to reemphasize to you that we are not using the prior review and approval of our system design by the NRC as a basis for the classification of the safety significance of this issue. Statements which I made in ET-0586 relative to the safety significance were intended to summarize the EDMG position on screening. The reference to the NRC is only a statement of the fact that our design philosophy at the time of licensing was reviewed and approved by the NRC. This position in fact constitutes our licensing basis.

The FSAR contains references to analyses regarding the Fuel Pool Cooling and Cleanup System and the Fuel Pool. It does not describe the exact manner in which we operate the plant. Our current fuel design and fuel cycle deviates from the FSAR description. This is the subject of G00005. Our outage practices differ from the description in the FSAR. The current outage practice is the subject of a periodic analysis done by Nuclear Fuels for each refueling outage.

We have reviewed the twenty-eight "open items" from the NSAG review. Twenty-six of the twenty-eight items were resolved to the satisfaction of NSAG. The remaining two items refer to the need for the level and temperature indication to be available in the control room via a PMS (computer display) format and to add reflash capability for alarms from panel OC211. These modifications are on the books but not being actively worked to the best of my knowledge.

EDMG completed a revision to the EDR screening for G20020 on Friday August 28, 1992 and requested comments. The significance was evaluated as minimal. Based on my review of this screening document it is unacceptable as written and I have requested it be revised. We are proceeding to review the issue for reportability regardless of the final significance level from the screening review.

Work on resolving the issue is assigned to Mark Mjaatvedt. Michael Crowthers has been working on this issue since July 20, 1992. I have also assigned Dave Kostelnik as of today. We are planning to involve Bechtel and GE. A schedule is under development.

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Systems Analysis' independent evaluation will be completed this week. We will factor their evaluation into our ongoing work. I will report on their results when available.

I will continue to keep you apprised of our progress on a daily basis.

Glenn D. Miller

cc:	J. E. Agnew	A6-3	D. C. Prevatte	A6-3
	F. G. Butler	A6-3	J. R. Miltenberger	A6-1
	R. G. Byram	A6-1	M. R. Mjaatvedt	A6-3
	M. H. Crowthers	A6-3	C. A. Myers	A2-4
	G. D. Gogates	SSES	J. G. Refling	A6-3
	G. J. Kuczynski	SSES	J. S. Stefanko	A9-3
	[redacted]	[redacted]	T. J. Sweeney	SSES
	S. M. Hauseman	A6-2	J. A. Zola	A6-3
			Nuclear Records	A6-2

Attachment 15

PP&L Memo from Kevin W. Brinckman to George T. Jones, "Review of Fuel Pool Cooling", September 1, 1992 (PLI-72288)

Note: This engineering report was prepared by a PP&L engineer previously not associated with EDR G20020 at the request of the PP&L Manager of Nuclear Plant Engineering to provide him with an independent appraisal of the concerns raised in the EDR. This independent evaluation basically concludes that a LOCA with a loss of normal fuel pool cooling would put the operators "in a position where they would be required to make decisions on removing ECCS equipment from containment/core cooling service to cool the fuel pool" and points out that it would involve unanalyzed conditions. This report also raises, for the first time, the concern that the hydrodynamic loads of the LOCA might damage the non-seismic, non-safety related fuel pool cooling system piping.