SAFETY EVALUATION COVERSHEET

No. NL 86-005

UNIT # 2

PMR# N/A

PROCEDURE # OP-235-001

OTHER NA

SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2
PENNSYLVANIA POWER & LIGHT COMPANY
ALLENTOWN, PENNSYLVANIA

REV.	DATE	PREPARED BY	ACCEPTED BY*	PORC APPROVED MTG#
0	8-6-86	Timothy J Ball	[illegible] 8/6/86	86-098 8-7-86

*This signature of the Responsible Supervisor indicates acceptance of the safety evaluation and confirms that interfaces with other disciplines, functional groups, etc. have been considered and have been incorporated into the evaluation as necessary. The Responsible Supervisor must be designated on NDI-QA-9.1.1C.

A copy of the APPROVED Safety Evaluation must be forwarded to the Manager, Nuclear Licensing.

NDI-QA-9.1.1A, Rev. 3 (2/85)

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I. System/Procedure/Experiment Identification.

Reference OP-235-001 section 3.10 for configuration of Fuel Pool Cooling Systems Unit 1 & 2

II. Description of Proposed Action.

This safety evaluation addresses the safety significance of the proposed plant configuration if TP-235-004, Fuel Pool Cooling System Heat Removal Capabilities, leads to the decision to remove the "B" Loop of RHR Shutdown Cooling Unit 2 from service and begin outage work. TP-235-004 is the test that will be used to determine

III. Does the proposed action increase the probability of occurrence or the consequences of an accident or malfunction of equipment related to safety, as previously evaluated in the FSAR?

___ YES ✓ NO

Provide a discussion of the basis and criteria used in arriving at the above conclusion.

A complete review of FSAR Section 9.1, Fuel Storage and Handling, Appendix 9A, Analysis for Non-Seismic Spent Fuel Pool Cooling System, and Chapter 15, Accident Analysis was performed leading to the following conclusions. Of the eight (8) accident event analytic categories as defined in Chapter 15 only one category of analyzed events could possibly be affected by the proposed plant configuration. That category is "Decrease in Reactor Coolant Inventory: Reductions in coolant inventory could threaten the fuel as the coolant becomes less able to remove the heat generated in the core." Specifically in section 15.6, Decrease in Reactor Coolant Inventory, are the line break and LOCA accident analysis. The proposed plant configuration can not increase

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II. Description of Proposed Action continued:

if the Unit 2 Fuel Pool Cooling System assisted by the Unit 1 Fuel Pool Cooling System during refueling operations can maintain the Unit 2 & 1 Fuel Storage Pool bulk temperatures ≤ 125°F without Unit 2 RHR system cooling. If the test indicates the Fuel Pool Systems Unit 1 & 2 in this assist configuration can remove the remaining decay heat and maintain fuel pool temperatures then the "B" loop of RHR Shutdown Cooling Unit 2 will be removed from service and outage work begun. The "A" loop of RHR Shutdown Cooling Unit 2 will already be inoperable (outage work begun). This safety evaluation addresses the safety significance of this configuration in which both loops of RHR unit 2 are inoperable and the remaining decay heat is to be removed using the Fuel Pool Cooling Systems in this assist mode of operation per section 3.10 of OP-235-001.

III. continued:

the probability of occurrence or the consequences of any line break as postulated in the FSAR. All the postulated line breaks are at power and address consequences through to safe shutdown. Since the proposed configuration is a refueling configuration, a shutdown configuration, it has to bounded by the existing analysis.

A review of all the other accident analysis of Chapter 15 revealed no other specific event where the proposed configuration increases the probability of occurrence or the consequences of the event. Specifically, Failure of RHR Shutdown cooling is analyzed in section 15.2.9 through cold shutdown as an event leading to increases in reactor pressure that would threaten the reactor coolant pressure boundary therefore is not applicable to the proposed refueling configuration.

FSAR Appendix 9A; Analysis for Non Seismic Spent Fuel Pool Cooling Systems, examines the consequences of a loss of fuel pool cooling. The proposed plant configuration could clearly affect this accident analysis therefore this event was reviewed in detail. The analysis assumes " a seismic event causes the loss of cooling to both spent fuel pools. In addition, in order to maximize both the heat loads and the iodine inventories in the pools, sequential refuelings were postulated. The loss of cooling was assumed during the second refueling, just after the refueling cavity water level was lowered and the seismically qualified RHR system would not be available for cooling the cavity and SFP. The analysis involved an evaluation of the time to pool boiling, the capability to add makeup water if the pool boils, and the thyroid dose consequences at the LPZ boundary due to iodine releases from the boiling pools. The assumptions used in this analysis were consistently chosen to be the worst case design basis assumptions, similar to those in Regulatory Guides for design basis accidents (e.g. Regulatory Guides 1.3, 1.25, etc.). The combination of all of these design basis assumptions occurring at the same time would be extremely unlikely making this accident as analyzed, one of very low probability. Many of the assumptions are considered to be overly conservative."

"A more realistic evaluation of this accident would result in releases of radioactivity, if any, many orders of magnitude below the calculated values. The realistic releases would be well below the Appendix I Technical Specifications, indicating that such an incident is of little or no consequence."

" The conservative results showed that the pools would not boil until at least 25 hours after the loss of cooling. 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".

All the above information as quoted from FSAR Appendix 9A must be kept in mind when reviewing the following conclusions.

The proposed plant configuration does not increase the probability of occurrence of the analyzed event. The proposed plant configuration does not increase the probability of a loss of fuel pool cooling. The events that could lead to a loss of fuel pool cooling like a seismic event (as postulated) or loss of service water, loss of power, loss of fuel pool pumps, etc. are all independent of the proposed plant configuration; the initiating events are independent of fuel pool configuration. One could ask whether operating the Fuel Pool Systems in the assist mode increases the probability of a dual unit loss of fuel pool cooling, but since the analyzed accident assumed a seismic event that causes a loss of both units cooling because "both units are cross-connected and in close proximity" the question becomes irrelevant. The proposed configuration does not increase the probability of a seismic event nor the probability of a loss of both units cooling from a seismic event since the physical ties exist regardless of the proposed operating configuration.

In order to answer whether the consequences of the accident of loss of fuel pool cooling are increased by the proposed configuration one must further examine the assumptions of the existing analysis and compare those assumptions to the proposed plant configuration.

From FSAR Appendix 9A:

"The following assumptions were used to calculate the heat generation and boiling rates in the two spent fuel pools.

1. Each pool contains the maximum fuel inventory of 15 quarter cores. For Unit 1 the fuel has decayed for 10.5 days, the length of time from shutdown until water level in the refueling pool has been lowered and the RHR system

would not be able to cool the refueling and spent fuel pools. For Unit 2 the decay time is 13.5 + 10.5 = 24 days. The 13.5 days is the minimum time to complete a fuel unloading and loading.

The following assumptions were used to calculate the offsite doses for the loss of cooling to the spent fuel pools.

b. During refueling 184 fuel elements (approximately 1/4 core) are removed and transferred to the SFP. Iodine in fuel from past refuelings will be negligible due to long decay times.

c. It is assumed that 1% of the fuel rods in the core are defective and that this 1% is in the 1/4 core transferred to the SFP.

e. The SFP cooling systems are assumed to fail 24 days after shutdown from the first reactor and 10.5 days after shutdown for the second.

i. The activity release rate from the pool depends on the evaporation (boiling rate). "

Comparing these assumptions to the proposed configuration raises the following issues. The first issue is since by the proposed schedule both loops of RHR could be made inoperable by day 8 since shutdown the available decay heat

for boiling would be greater for the proposed configuration than for the analyzed event assuming the loss of fuel pool cooling at that time. The activity release rate from the pool depends on the evaporation (boiling rate) therefore the consequences in the proposed configuration could be increased; however, one must remember that the proposed configuration involves boiling of not just the fuel pools as in the analyzed event but also the Unit 2 reactor well, reactor vessel, dryer/separator pool and cask storage pit. The following calculation will show that the time to boiling (therefore boiling rate) will be no greater than for the analyzed event.

Approximate volumes

Fuel Pools (2)	400,000 gallon each x 2
Reactor Well	400,000 gallon
Dryer/Separator Pool	160,000 gallon
Shipping Cask Pit	75,000 gallon
Reactor Vessel	150,000 gallon
	1,510,000 gallon

Assume heating from 140°F to 212°F for boiling. 140°F is the Technical Specification limit on average reactor coolant temperature in condition 5. It would be a hot or conservative starting temperature for all the volumes concerned.

$$1{,}510{,}000 \text{ gallon } (212-140)\,°F \left(\frac{1 \text{ Btu}}{\text{lb } °F}\right)\left(\frac{ft^3}{7.48 \text{ gallon}}\right)\left(\frac{62.4 \text{ lb}}{ft^3}\right) =$$

$$9.07 \times 10^8 \text{ Btu}$$

Using Reactor Engineering estimates for decay heat on day 8 after shutdown:

$Y = 2.33 - .046X$

Y = Decay Heat at Day X, 10^7 BTU/hr
X = Days since shutdown

$Y = 2.33 - .046(8) = 1.96 \times 10^7$ BTU/hr

$$\frac{9.07 \times 10^8 \text{ Btu}}{1.96 \times 10^7 \text{ Btu/hr}} = \underline{\underline{46 \text{ hrs to boil}}} \text{ after loss of cooling.}$$

The analyzed event showed the pools would not boil until at least 25 hours after the loss of cooling therefore the proposed configuration is more conservative.

The second issue of concern would be that given the pools boil does the fuel configuration of the proposed action increase the expected releases. This can be answered regardless of boiling rate, amount of fuel, and location of fuel. The greatest portion of release will be from the failed fuel. The analyzed event assumed 1% of the fuel rods in the core are defective and that this 1% is in the 1/4 core transferred to the SFP. Offgas radiation in Unit 2 indicates that less than 1 fuel rod has failed (no fuel failure). One rod of 62 rods/bundle and 764 bundles/core is orders of magnitude less than 1% failed fuel therefore expected release rates would be orders of magnitude less.

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For these reasons as given the proposed configuration does not increase the consequences of a loss of fuel pool cooling as previously evaluated in the FSAR.

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IV. Does the proposed action create a possibility for an accident or malfunction of a different type than any evaluated previously in the FSAR?

____ YES, ✓ NO

Provide a discussion of the basis and criteria used in arriving at the above conclusion.

The only accident of consequence in the proposed configuration that can be postulated is one resulting in the inability to remove decay heat. Regardless of the initiating events that accident is loss of fuel pool cooling. One might question whether the proposed configuration could have different consequences, therefore be of a different type that should be considered, but as already shown in the response to question III those different consequences are less significant than the analyzed event. There is no new accident possibility beyond loss of fuel pool cooling as previously analyzed in the FSAR.

V. Does the proposed action reduce the margin of safety as defined in the basis for any Technical Specification?

____ YES, ✓ NO

Provide a discussion of the basis and criteria used in arrive at the above conclusion.

A review of all the basis for the Technical Specifications for refueling operations was performed leading to the following conclusions. The basis for Technical Specification 3/4.9.11 is the only basis of concern. In order to answer the question, "Does the proposed action reduce the margin of safety as defined in the basis for any Technical Specification?", in the proposed configuration, one must answer the question, "Does a proposed action or configuration that places the Unit in an action statement for a Technical Specification reduce the margin of safety as defined in that Technical Specification? The answer to the second question is no, if (1) the proposed action or configuration does not prevent fulfilling the action statement and

II continued:

(2) the proposed action or configuration does not prevent meeting the intent (basis) of that Technical Specification. The basis for 3/4.9.11 Residual Heat Removal and Coolant Circulation is

"The requirement that at least one residual heat removal loop be OPERABLE or that an alternate method capable of decay heat removal be demonstrated and that an alternate method of coolant mixing be in operation ensures that (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 140°F as required during REFUELING, and (2) sufficient coolant circulation would be available through the reactor core to assure accurate temperature indication and to distribute and prevent stratification of the poison in the event it becomes necessary to actuate the standby liquid control system."

The proposed configuration does not prevent fulfilling the action statements of Technical Specification 3/4.9.11. TP-235-004 in itself will demonstrate Fuel Pool Cooling as an alternate method capable of decay heat removal. ON-249-001, Loss of RHR (Shutdown Cooling) System, step 3.3.2 ensures the operability of alternate means of decay heat removal are demonstrated by demonstrating operability of method(s) of adding water to the reactor vessel (1) RWCU return to vessel (2) CRD, (3) Condensate transfer (keep fill / SDC Flush)

(4) Core Spray and demonstrating operability of method(s) of rejecting water from the Reactor Vessel: (1) RWCU suction from vessel (2) RWCU letdown (3) Fuel Pool Cooling (4) Skimmer Surge Tank letdown. Step 3.3.1 of ON-249-001 directs performance of the alternate means of circulation (1) Maintain water level 90 to 100 inches on shutdown indication to establish natural recirculation OR (2) operate RWCU system at its maximum flow rate OR (3) Start Reactor Recirculation System with at least one pump in minimum speed. The proposed configuration does not prevent taking the actions of ON-249-001 steps 3.3.1 and 3.3.2 when both loops of RHR are not available in condition 5 with the vessel head removed.

The proposed configuration does not prevent meeting the intent (basis) of the Technical Specification. Given the action statement has been fulfilled in the proposed configuration as already discussed, sufficient cooling capacity is available and sufficient coolant circulation would be available to meet the intent of the Technical Specification. TP-235-004 in itself will demonstrate in the proposed configuration that average reactor coolant can be maintained below 140°F therefore showing the intent (basis) of the Technical Specification is met. One might ask, "Would a loss of Fuel Pool Cooling in the proposed configuration reduce the margin of safety as defined in the basis for the Technical Specification". The answer is no because the

action statement of the Technical specification can be fulfilled with other alternate means of decay heat removal. This is clearer if one considers the loss of RHR Shutdown Cooling when irradiated fuel is in the reactor vessel and the water level is less than 22 feet above the top of the reactor vessel flange. In this configuration Fuel Pool Cooling is not an alternate means of decay heat removal and more decay heat is available than in the proposed configuration. In the low water level condition, Technical Specification 3/4.9.11 action statements require demonstrating one alternate method capable of decay heat removal for each inoperable RHR shutdown cooling mode loop. Obviously the loss of RHR shutdown cooling is more severe for the low water level case even though the basis for the Technical Specification is the same regardless of water level. The basis for Technical Specification 3/4.9.11 itself states, " With the reactor vessel head removed and 22 feet of water above the reactor vessel flange, a large heat sink is available for core cooling. Thus, in the event a failure of the operating RHR loop, adequate time is provided to initiate alternate methods capable of decay heat removal or emergency procedures to cool the core." For these reasons the margin of safety as defined in the basis for Technical Specification 3/4.9.11 is not reduced for the proposed configuration.

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VI. Does the proposed action involve a change in a Technical Specification?

___ YES, ✓ NO

If "YES", NDI-QA-3.2.1 "Technical Specification Changes" applies. A "YES" answer does not preclude activity up to a point just before it would physically affect the functioning of the plant.

Provide a discussion of the basis and criteria used in arriving at the above conclusion. If appropriate, describe the extent of activity and why it should be allowed to proceed prior to the Technical Specification change.

A review of all Technical Specifications for refueling operations was performed and the proposed configuration does not create the need for a change in a Technical Specification. The proposed configuration with both loops of RHR Shutdown Cooling inoperable and irradiated fuel in the reactor vessel will put Unit 2 in the action statements of Technical Specification 3/4.9.11 but does not require a change to that Technical Specification. Technical Specification 3/4.9.11 is Residual Heat Removal and Coolant Circulation High Water Level.

VII. Does the proposed action create the need to make an application for amendment to the license other than to Appendix A?

___ YES, ✓ NO

Provide a discussion of the basis and criteria used in arriving at the above conclusion.

The proposed configuration does not create a conflict with the license therefore does not create the need to make application for amendment to the license.