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SUBJECT: LER 93-018-00:on 930428,discovered design condition that could have impacted plants ability to mitigate accident conditions.Caused by less than adequate change mgt.Abnormal operating procedures will be revised.W/930528 ltr.

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Docket No. 50-397

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: **NUCLEAR PLANT WNP-2, OPERATING LICENSE NPF-21**
LICENSEE EVENT REPORT NO. 93-018

Transmitted herewith is Licensee Event Report No. 93-018 for the WNP-2 Plant. This report is submitted in response to the report requirements of 10CFR50.73 and discusses the items of reportability, corrective action taken, and action taken to preclude recurrence.

Sincerely,

J. V. Parrish (Mail Drop 1023)
Assistant Managing Director, Operations

JVP/KBL/jd
Enclosure

cc: Mr. B. H. Faulkenberry, NRC - Region V
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INPO Records Center - Atlanta, GA
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LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Washington Nuclear Plant - Unit 2	0 5 0 0 0 3 9 7	1 OF 7

TITLE (4)

SPENT FUEL POOL MAKEUP NOT ADEQUATE TO MITIGATE ACCIDENT CONDITIONS

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAMES	DOCKET NUMBERS(S)
0 4	2 8	9 3	9 3	-- 0 1 8	-- 0 0	0 5	2 8	9 3		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9)	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)			
1	20.402(b)	20.405(C)	50.73(a)(2)(iv)	77.71(b)
POWER LEVEL (10) 0 9 3	20.405(a)(1)(i)	50.36(c)(1)	X 50.73(a)(2)(v)	73.73(c)
	20.405(a)(1)(ii)	50.36(c)(2)	50.73(a)(2)(vii)	OTHER (Specify in Abstract below and in Text, NRC Form 366A)
	20.405(a)(1)(iii)	50.73(a)(2)(i)	50.73(a)(2)(viii)(A)	
	20.405(a)(1)(iv)	50.73(a)(2)(ii)	50.73(a)(2)(viii)(B)	
	20.405(a)(1)(v)	50.73(a)(2)(iii)	50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER
Kurt B. Lewis, Licensing Engineer	AREA CODE 5 0 9 3 7 7 - 4 1 4 5

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE)	NO				

ABSTRACT (16)

On April 28, 1993, the WNP-2 Management Review Committee (MRC) concluded its review of a problem associated with the Fuel Pool Cooling (FPC) system and decided it was a reportable event. Previously, on April 23, 1993, a plant support engineer discovered a design condition that could have impacted the plant's ability to mitigate the consequences of an accident. In the event of a Loss Of Coolant Accident (LOCA), the Service Water system (SW) must be able to supply emergency makeup water to the Spent Fuel Pool (SFP). Makeup is normally supplied by the Condensate Storage and Transfer system, a non-safety-related system. Without makeup, water in the pool would have continued to evaporate until the Fuel Pool Cooling system (FPC) pump tripped. This would have resulted in eventual fuel pool boiling. Vapors generated by the elevated Spent Fuel Pool temperature would have condensed, flowed to the bottom of the Reactor building, and caused a loss of the Emergency Core Cooling (ECCS) pumps due to flooding. Service water would not have been immediately available because manual isolation valves SW-V-75AA and BB are presently closed, and access to these valves was analyzed to be restricted during post-LOCA conditions.

Immediate corrective action included performing an analysis to verify the feasibility of continued, safe operation until the next refueling outage, which began on April 30, 1993.

Further corrective actions included performing engineering calculations and revising plant documents to justify continued closure of manual isolation valves SW-V-75AA and BB during normal plant operations.

The root cause for this event was less than adequate change management.

This event posed minimal risk to safe plant operation.

Plant Conditions

Power Level - 93%

Plant Mode - 1 (Power)

Event Description

On April 28, 1993, the WNP-2 Management Review Committee (MRC) concluded its review of a problem associated with the Fuel Pool Cooling (FPC) system and decided it was a reportable event. Previously, on April 23, 1993, a plant support engineer discovered a design condition that could have impacted the plant's ability to mitigate the consequences of an accident. During accident conditions, the SW system is designed to remote-manually supply redundant, emergency makeup to the Spent Fuel Pool from SW loops "A" and/or "B". Each redundant makeup line includes a pair of series-installed isolation valves. The first valve in each pair is a locally operated, manual isolation valve [SW-V-75 AA(BB)]. Both manual valves are located on the 522 elevation of the Reactor Building. The second valve in each pair is a remote-manual, motor operated isolation valve [SW-V-75A(B)]. Remote-manual operation is from the main control room. Currently, all of these valves are normally closed. This is in conflict with the Final Safety Analysis Report (FSAR) which specifies actuation of the SW makeup by no other operator action than remote-manual actuation of the motor operated isolation valves.

During his analysis of the problem, the engineer determined that leaving the manual isolation valves normally closed could preclude a control room operator's ability to remote-manually supply emergency SW makeup to the Spent Fuel pool during a Loss Of Coolant Accident (LOCA). Use of the manual valves could be precluded, because the LOCA could generate sufficient radiation fields in the Reactor Building to restrict accessing the manual valves prior to unacceptably high Spent Fuel Pool temperatures being reached. This inability to supply the fuel pool with makeup could result in pool boiling and the resultant vapor could condense and end up in the ECCS pump rooms. Since the Reactor Building sump pumps are not safety related, no credit can be taken for them during an accident. Thus, the condensed vapor could ultimately flood the ECCS pump rooms and render the ECCS pumps inoperable. Additionally, the operability of the Standby Gas Treatment system (SGT) cannot be assured at Spent Fuel Pool temperatures in excess of 155°F, as no analysis exists for this condition.

Immediate Corrective Action

Immediate corrective action included performing an analysis to verify the feasibility of continued, safe operation. Results of the analysis, based principally on currently low fuel pool temperatures and heat loads, showed that continued operation at power was feasible up to the R8 refueling outage, which began on April 30, 1993.

Further Evaluation and Corrective Action

Further Evaluation

1. On April 28, 1993 at 1019 hours, this event was reported to the NRC as a four-hour reportable event under 10CFR50.72(b)(2)(iii)(D), as "Any event or condition that alone could have prevented the fulfillment of the safety function of structures or systems that are needed to mitigate the consequences of an accident".

2. This event is also being reported under 10CFR50.73(a)(2)(v)(D), as "Any event ... that alone could have prevented ... the safety function of ... systems ... needed to mitigate the consequences of an accident".

3. There were no structures, components, or systems inoperable prior to this event that contributed to this event.

4. Prior to 1979, the original FPC system design basis allowed Spent Fuel Pool boiling during a severe seismic event and provided for safety grade makeup from the SW system. In 1979, the NRC informed the Supply System that boiling was not acceptable and offered several alternative designs. The Supply System subsequently redesigned the FPC system to prevent pool boiling and to ensure that adequate water level is maintained during a seismic event or major plant disturbance. Among the design criteria was the stipulation that "remote-manual, redundant SW system makeup to the fuel pool" be provided (as stated by the FSAR, section 9.1.3.1).

5. To satisfy remote-manual makeup requirements, original design plans requested motors be placed on existing, locally operated, SW-to-SFP manual isolation valves. However, economics, physical constraints, and testing requirements prevented this design from being completed.

6. Rather, Plant Engineering Directive (PED) 215-M-C342, written on August 5, 1982, retagged the manual isolation valves as SW-V-75AA and 75BB and left them in place. Downstream of these valves, the PED directed installation of corresponding remote-manual, motor operated isolation valves SW-V-75A and 75B.

7. Per PED 215-M-C342, no change to the FSAR was anticipated based on leaving the manual valves installed in the system. However, when the final valve configuration was redrawn on SW flow diagrams M524 sheets 1 & 2 (REV 35, November 5, 1982), valves SW-V-75AA and BB were incorrectly depicted as normally closed. The manual isolation valves should have been redrawn as normally open (NO), the position required to support remote-manual operation.

8. Additionally, revision one of the SW operating procedure, issued on August 30, 1983, duplicated this error by aligning these valves as normally closed. Subsequent revisions to this procedure carried this error.

9. On October 31, 1991, Problem Evaluation Request (PER) 291-899 identified this error in valve alignment. The PER stated that the "commitment for remote-manual operation cannot be met". The MRC incorrectly identified the limiting condition for continued FPC system operation without makeup as the rate of pool evaporation, without considering the effect based upon the FPC Skimmer Surge tank low-water-level trip of the FPC pumps. The MRC recommended revising the FSAR to reflect all SW-to SFP isolation valves as normally closed and recommended revising engineering calculations to support manual operator action to open the manual isolation valves.

10. To support the MRC's recommendation to revise engineering calculations, engineering calculated evaporative water losses from the Spent Fuel Pool during a loss of makeup. The calculation conservatively assumed a pool temperature of 155°F and a full offload after cycle 12. However, the calculation incorrectly assumed that one of the two FPC system pumps continued to circulate water from the Spent Fuel Pool, through FPC system heat exchangers, and then back to the Spent Fuel Pool during the loss of makeup. This assumption was incorrect, because the pumps trip on associated low water level in the FPC Skimmer Surge Tanks. Thus, the calculation incorrectly showed that it would take approximately 128 days for the pool's water level to decrease to the top of the fuel. The calculated radiation levels in the Reactor Building at this time would be low enough to allow accessing manual isolation valves SW-V-75AA and BB. Based on this information, a low-priority Request For Technical Services (RFTS) was written to revise the FSAR to reflect all SW-to-SFP isolation valves as NC.

11. On April 23, 1993, a plant support engineer responded to the RFTS by preparing the necessary revisions to the FSAR. The engineer discovered the error in the 1991 engineering calculation while he was completing a related 10CFR50.59 analysis. The engineer immediately wrote a PER documenting the problem. The PER originally requested opening the manual isolation valves, but this request was changed to read "evaluate for other options". Further evaluation was requested, because plant personnel wanted to maintain two barriers between a radioactive system and the environment.

12. Further evaluation was pursued and a Basis For Continued Operation (BCO) was written in the interim to justify continued plant operation with the manual isolation valves left closed. Further evaluation was pursued, because plant personnel wanted to maintain double isolation of the SW-to-SFP emergency makeup lines. Personnel believed this to be desirable, because double isolation provided additional assurance that the fuel pool water does not leak into the Service Water System when the Service Water System is not operating.

13. The BCO, based on low fuel pool temperatures and heat loads, showed that continued operation at power was feasible up to the R8 refueling outage, which began on April 30, 1993. Details of the BCO follow:

    a. The Spent Fuel Pool temperature was assumed to be 155°F during accident analysis (a condition in which evaporation in the Reactor Building has been analyzed with no challenge to the Standby Gas Treatment system); actual pool temperature was well below this value. Accident analysis requires the assumption that the maximum heat load exists in the pool; the actual heat load was low. Recent data from an FPC system outage showed that the Spent Fuel Pool would heat up at 6°F per day. If the Spent Fuel Pool was maintained less than 90°F during reactor operation, even with the FPC Skimmer Surge Tanks at their minimum water level of 26 inches above the FPC system pump trip, the FPC pumps would operate for approximately nine hours, assuming an evaporation rate at a pool temperature of 155°F. After the FPC pumps trip on low Skimmer Surge Tank water level, it would take the relatively low heat load in the pool eleven days to increase the pool's temperature to 155°F. Evaporative losses during this eleven day period would reduce pool water level less than two feet.

    b. Reactor Building radiation levels following a LOCA are based on worst case conditions. Even under these worst case conditions, an operator donned in Self Contained Breathing Apparatus (SCBA) could open both of the manual isolation valves in less than ten minutes. Performing this emergency operation would result in the operator receiving a whole body dose of less than five Rem, the maximum dose allowed by regulatory guides applicable to these manual, operator- action requirements. The entry would have to be made eleven days following the LOCA.

    c. As long as the Spent Fuel Pool is maintained lower than 90°F and the current heat load is maintained, the plant support engineer justified leaving the manual isolation valves closed.

Root Cause

The root cause for maintaining Service Water valves SW-V-75AA and BB normally closed was a less than adequate review of the associated 1982 design change specifications against related FSAR requirements for "remote-manual, redundant SW system makeup to the fuel pool". Therefore, this 1982 event receives an INPO root cause classification of Less Than Adequate (LTA) Change Management. A contributing cause for this event was procedurally specifying Service Water valves SW-V-75AA and BB as normally closed, rather than open. This contributing cause receives an INPO classification of Technical Inaccuracies in Written Procedures.

Further Corrective Action

1. The process for writing and implementing design change packages has improved since the Plant Engineering Directive process that was used to install motor operated valves SW-V-75A and B. For example, the present 10CFR50.59 process and the capability to word search the FSAR has greatly enhanced a person's ability to compare design changes against the FSAR. Additionally, procedural support organizations are now in place to assist in the development or revision of plant procedures. With respect to the Change Management issue, no further corrective action is recommended.

2. On May 13, 1993, operator actions were performed to determine the Reactor Building stay time by entering the building and simulating opening the manual valves and recording the time required. Stay time was determined to be two minutes, thirty seconds.

3. On May 24, 1993, engineering analysis verified that manual isolation valves SW-V-75AA and BB can be opened during post-accident conditions to prevent fuel pool heatup and boiling. From the analysis, it was concluded that the manual isolation valves can be left normally closed, and that if these valves were opened in no more than three minutes during LOCA conditions, an operator would receive a whole body dose of 4.85 REM. Because the probability for a LOCA to actually occur is low (2.17×10^{-4} events/year), and since a LOCA concurrent with a loss of fuel pool makeup is extremely low, the Supply System decided to use the results of this engineering analysis to justify continued closure of manual isolation valves SW-V-75AA & BB.

4. An FSAR change notice will be prepared to include discussion of local-manual, as well as remote-manual operating requirements of the SW-to-SFP emergency makeup water crosstie during a LOCA. This will be completed by September 31, 1993.

5. Instrument Master Data Sheets will be revised per engineering calculation E/I-02-92-1042 to specify a new, lower setpoint for the fuel pool high temperature alarm. This will be completed prior to R8 restart on or about June 15, 1993.

6. Annunciator response procedures 4.626.FPC1 1-2 (4.627FPC2 1-2) "Fuel Pool Temperature High" will be revised to specify a new, lower fuel pool high temperature alarm setpoint and provide the required actions to be taken upon receipt of the alarm. This will be completed prior to R8 restart on or about June 15, 1993.

7. Abnormal operating procedure 4.8.5.1 "Loss of Fuel Pool Cooling" will be revised to include instructions for use of SW-to-SFP emergency makeup during accident conditions. This will be completed prior to R8 restart on or about June 15, 1993.

Safety Significance

The FSAR requires remote-manual, redundant SW emergency makeup to the Spent Fuel Pool. Because related manual isolation valves were left closed, the ability to supply this emergency makeup during accident conditions was challenged. However, due to the low probability for a LOCA and simultaneous loss of the normal source of makeup to the Spent Fuel Pool, this event posed minimal risk to safe plant operation.

Similar Events

There are no similar LER events involving Spent Fuel Pool boiling and loss of accident mitigation capability.

EIIS Information

Text Reference	EIIS Reference System	EIIS Reference Component
Fuel Pool Cooling System (FPC)	OA	---
Service Water System (SW)	BI	---
SW Manual Isolation Valves SW-V-75AA(BB)	BI	---
Condensate Storage & Transfer	KA	---
SW Motor Operated Valves SW-V-75A(B)	BI	---
FPC Skimmer Surge Tank	--	SKR
Standby Gas Treatment System (SGT)	BH	---
FPC Pump	--	P
FPC Heat Exchanger	--	HX