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 AUTH. NAME AUTHOR AFFILIATION  
 FULLER, R.E. Washington Public Power Supply System  
 PARRISH, J.V. Washington Public Power Supply System  
 RECIP. NAME RECIPIENT AFFILIATION

SUBJECT: LER 93-001-01: on 930113, determined that both trains of SPC  
 inoperable due to inadequacy of original design analysis.  
 Plant Procedure PPM 2.4.2, RHR will be changed by 930301 &  
 leaking MSRVs will be reworked. W/930611 ltr.

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G02-93-149

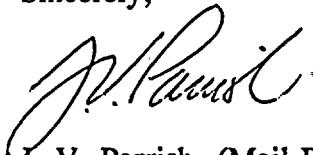
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Washington, D.C. 20555

Subject: **NUCLEAR PLANT WNP-2, OPERATING LICENSE NPF-21  
LICENSEE EVENT REPORT NO. 93-001-01**

Transmitted herewith is Licensee Event Report No. 93-001-01 for the WNP-2 Plant. This report is submitted in response to the report requirements of 10CFR50.73 and incorporates only minor administrative changes.

Sincerely,



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

JVP/REF/jd  
Enclosure

cc: Mr. B. H. Faulkenberry, NRC - Region V  
Mr. R. Barr, NRC Resident Inspector (Mail Drop 901A, 2 Copies)  
INPO Records Center - Atlanta, GA  
Mr. D. L. Williams, BPA (Mail Drop 399)

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# LICENSEE EVENT REPORT (LER)

FACILITY NAME (1) <b>Washington Nuclear Plant - Unit 2</b>	DOCKET NUMBER (2) 0   5   0   0   0   3   9   7	PAGE (3) 1   OF   7
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TITLE (4)  
**INOPERABLE SUPPRESSION POOL COOLING DUE TO POTENTIAL WATERHAMMER**

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	1	1	3	9	3	0	0	1	0	0	1
0	1	1	3	9	3	0	6	1	0	5	0

OPERATING MODE (9) **1** THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)

POWER LEVEL (10) 1   0   0	20.402(b) 20.405(a)(1)(i) 20.405(a)(1)(ii) 20.405(a)(1)(iii) 20.405(a)(1)(iv) 20.405(a)(1)(v)	20.405(C) 50.36(c)(1) 50.36(c)(2) X 50.73(a)(2)(i) 50.73(a)(2)(ii) 50.73(a)(2)(iii)	50.73(a)(2)(iv) X 50.73(a)(2)(v) 50.73(a)(2)(vii) 50.73(a)(2)(viii)(A) 50.73(a)(2)(viii)(B) 50.73(a)(2)(x)	77.71(b) 73.73(c) OTHER (Specify in Abstract below and in Text, NRC Form 366A)
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LICENSEE CONTACT FOR THIS LER (12)	
NAME <b>R. E. Fuller, Licensing Engineer</b>	TELEPHONE NUMBER 5   0   9   3   7   7   -   4   1   4   8

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

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**ABSTRACT (16)**

On January 13, 1993, a Licensing Engineer determined that both trains of Suppression Pool Cooling (SPC) of the Residual Heat Removal (RHR) System (RHR "A" AND "B") were inoperable on August 6, 1990. This represented a condition that could have prevented the RHR System from performing its SPC, Suppression Pool Spray (SPS), and Drywell Spray (DWS) safety functions. On December 22, 1992, engineering evaluations concluded that water hammer could fail the train of RHR in SPC due to a Loss of Power (LOP) coincident with a Loss of Coolant Accident (LOCA). Further review revealed that both trains were again operated concurrently: once in 1991 and twice in 1992. Per the Emergency Plans, these incidents also represented Unusual Events (UE).

A night order was issued immediately to declare any loop of RHR in SPC or SPS mode inoperable and enter the appropriate Technical Specification Action Statements (TSAS).

The root causes included: 1) A design deficiency of the RHR system because it cannot be operated in SPC during normal operation without compromising operability; 2) management methods that failed to identify this problem during review of a 1987 NRC Information Notice (IN), and 3) failure to follow procedures which led to inappropriate two train operation of SPC.

Leaking Main Steam Relief Valves (MSRV) will be reworked during the 1993 Refueling Outage to minimize leakage. Procedure changes will be made to indicate the inoperability of selected modes of RHR during operation of SPC and/or SPS. This LER will be made required reading for all Control Room supervision. An evaluation will be performed to determine the long term solution to the RHR design deficiency.

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION																
FACILITY NAME (1)  Washington Nuclear Plant - Unit 2	DOCKET NUMBER (2)  0   5   0   0   0   3   9   7							LER NUMBER (8)			PAGE (3)					
								Year	Number		Rev. No.					
								93	00	1	01		2	OF	7	
TITLE (4) INOPERABLE SUPPRESSION POOL COOLING DUE TO POTENTIAL WATERHAMMER																

The safety significance of these events was negligible. This condition posed no threat to the health and safety of Plant personnel or the public.

#### Plant Conditions

Power Level - 100%

Plant Mode - 1 (Power Operation)

#### Event Description

On January 13, 1993, a Licensing Engineer determined that both trains of Suppression Pool Cooling (SPC) of the Residual Heat Removal (RHR) System (RHR "A" AND "B") were inoperable for 48 hours beginning on August 6, 1990. Previously, on October 23, 1992, a Problem Evaluation Request (PER) 292-1191 was issued for elevated Suppression Pool air space temperatures caused by leaking Main Steam Relief Valves (MSRV). During evaluation of the PER, a survey of other General Electric (GE) Boiling Water Reactor (BWR) plants revealed a possible water hammer problem with RHR in SPC following a LOP-LOCA. The WNP-2 engineering personnel were unaware of this potential vulnerability of RHR and thus issued PER 292-1243 on November 4, 1992, to document the problem. On December 22, 1992, engineering evaluations concluded that water hammer could fail the train of RHR in SPC or Suppression Pool Spray (SPS) due to a Loss of Power (LOP) coincident with a Loss of Coolant Accident (LOCA). Specifically, any LOP (e.g., failure to transfer from the normal transformer, TR-N, to the Startup transformer, TR-S) to the associated bus of RHR in SPC causes the corresponding RHR pump to stop. The LOP would allow portions of the associated RHR piping and heat exchanger to drain. A LOCA signal coincident with a LOP would result in an automatic start of the pump within 15 seconds following re-energization of the bus. The resulting water hammer from filling of the voided piping could cause failure in portions of the associated RHR piping and/or heat exchanger, resulting in loss of the SPC, SPS, Drywell Spray (DWS), and Low Pressure Coolant Injection (LPCI) capability of the affected train.

Control Room logs were reviewed by the Licensing Engineer in support of a reportability evaluation for conditions when both SPC trains of RHR were inoperable, i.e., any combination of both trains operating in SPC or SPS concurrently or one train operating in SPC or SPS, and the other train out of service. A given train of RHR cannot be operated in more than one mode at a time. The review identified the concurrent operation of both SPC trains. This represented a condition that could have prevented the RHR System from performing its SPC, SPS, and DWS safety functions. The LPCI safety function was still capable of being performed by either RHR "C" or the Low Pressure Core Spray (LPCS) System.

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION								
FACILITY NAME (1)  Washington Nuclear Plant - Unit 2	DOCKET NUMBER (2)  0   5   0   0   0   3   9   7		LER NUMBER (8)			PAGE (3)		
			Year	Number	Rev. No.			
			9   3	0   0   1	0   1	3	OF	7
TITLE (4) INOPERABLE SUPPRESSION POOL COOLING DUE TO POTENTIAL WATERHAMMER								

### Immediate Corrective Action

A night order was issued on December 22, 1992, to declare any loop of RHR in SPC or SPS mode inoperable and enter the appropriate Technical Specification Action Statements (TSAS).

### Further Evaluation and Corrective Action

#### A. Further Evaluation

1. This event is considered reportable per 10CFR 50.73(a)(2)(v)(B) as a condition alone that could have prevented the fulfillment of the safety function of systems needed to remove residual heat. In addition, this condition is reportable per 10CFR 50.73(a)(2)(i)(B) as a condition prohibited by the plant's Technical Specifications (TS) Section 3.6.2.3. This event exceeded the TSAS allowable of 12 hours for the Plant to be in Hot Shutdown and 24 hours to be in Cold Shutdown with both loops of SPC inoperable. NRC was verbally notified of this condition on January 13, 1993, per 10CFR 50.72(b)(2)(iii)(B). This event also satisfied the condition of an Unusual Event (UE) per PPM 13.1.1, Emergency Plans.
2. The NRC issued Inspection and Enforcement Information Notice IN 87-10 on February 11, 1987, describing possible failure of the RHR train in SPC due to water hammer from a LOP-LOCA event. This IN was initiated based on the results of analyses performed at the Susquehanna Nuclear Power Plant on December 11, 1986. The notice indicated that Susquehanna limited operation of SPC to one train at a time in response to this deficiency.

On February 24, 1987, the Supply System's Nuclear Safety Assurance Group (NSAG) initiated an Operating Experience Review (OER 81078E) of IN 87-10. NSAG consulted only with a Shift Manager concerning the IN recommendation to limiting operation of SPC to one train at a time. Engineering personnel or the System Engineer were not contacted. The initial evaluation concluded that there was no Suppression Pool heat-up problem that would necessitate use of both trains concurrently. Although the procedures did not restrict two train operation, the Shift Manager considered the procedures adequate. The OER was closed on March 3, 1987, with no actions. It was still assumed that a train of RHR in SPC was operable.

3. While performing work on the Technical Specification Improvement Program (TSIP) in 1990, a Plant Technical engineer determined that the issue of two train operation had not been adequately addressed in the 1987 OER 81078E. NSAG limited their re-review of the OER to the procedure changes identified in IN 87-10 without consulting Engineering personnel or the System Engineer. Plant procedure PPM 2.4.2, Residual Heat Removal, was changed on October 8, 1990, to limit the SPC mode of RHR to one train operation per the

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION											
FACILITY NAME (1) Washington Nuclear Plant - Unit 2		DOCKET NUMBER (2) 0   5   0   0   0   3   9   7					LER NUMBER (8) Year   Number   Rev. No. 9   3   0   0   1   0   1			PAGE (3) 4 OF 7	
TITLE (4) INOPERABLE SUPPRESSION POOL COOLING DUE TO POTENTIAL WATERHAMMER											

recommended solution used by Susquehanna. However, the train of RHR in SPC was still considered operable. In addition, the 10CFR 50.59 review of the procedure change failed to identify the inoperable condition of the RHR train in SPC.

4. Based on the experience of Operations personnel, both trains of SPC would only have been operated during periods of MSRV testing. A review of Control Room logs since startup for only the periods of MSRV testing revealed three incidents of simultaneous operation of RHR "A" and "B". These incidents occurred from 0103 hours to 0546 hours on September 30, 1991, from 0206 hours to 0810 hours on July 6, 1992, and from 1319 hours to 1542 hours on July 11, 1992. Since the Plant procedure PPM 2.4.2 was changed in 1990 to preclude two train operation of SPC, these incidents occurred because of failure to follow procedures by the Reactor Operators.

The Control Room logs between July 1 and December 20, 1992, were also reviewed for instances where at least one train of RHR was in SPC or SPS and other safety related equipment were inoperable. No occurrence was found that could have compromised the safety function of a system or the condition was prohibited by the TS-given the RHR train in SPC or SPS was inoperable. Also, no other instance was found where a TS Limiting Condition of Operation (LCO) was exceeded.

#### B. Root Cause

1. The root cause of the vulnerability of the SPC mode of RHR to a LOP-LOCA was the inadequacy of the original design analysis.
2. A root cause that led to operating both trains of SPC simultaneously on August 6, 1990, was that management methods did not ensure appropriate technical input during review and closure of the OER on March 20, 1987. A multi-discipline review was not performed to ensure appropriate corrective actions would be implemented to preclude inappropriate operation of RHR.
3. A root cause that led to inappropriate two loop operation of SPC in 1991 and 1992 was failure to follow procedures.

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION														
FACILITY NAME (1) Washington Nuclear Plant - Unit 2		DOCKET NUMBER (2) 0   5   0   0   0   3   9   7							LER NUMBER (8) Year   Number   Rev. No. 9   3   0   0   1   0   1			PAGE (3) 5 OF 7		
TITLE (4) INOPERABLE SUPPRESSION POOL COOLING DUE TO POTENTIAL WATERHAMMER														

C. Further Corrective Action Taken

1. The OER process was enhanced in 1991 to require the reviewer to obtain the appropriate independent and multi-discipline review prior to closure of the report.
2. The process of performing 10CFR 50.59 reviews and safety evaluations has been enhanced since 1990 to further assure changes to Plant operation and configuration are made within the Licensing Basis Documents (LBD). The enhancements include required training of persons preparing and reviewing 10CFR 50.59 reviews and safety evaluations, and procedure changes that provide additional guidance and clarification of the 10CFR 50.59 process. The experience to date indicates a continuing improvement in the quality of the 10CFR 50.59 reviews and safety evaluations as the enhancements take affect.
3. Due to MSRV leakage, one train of SPC is currently required to be operated from 6 to 14 hours per week. The SPC train in operation is declared inoperable and the appropriate TSAS in Sections 3.5.1, Emergency Core Cooling Systems (ECCS) Operating, and 3.6.2.3, Suppression Pool Cooling, are entered. Assuming one train of SPC would be operated a maximum of 15 hours per week, the probability of a severe water hammer from a LOP-LOCA was estimated to be  $2.9E-7$  events per year. This value coupled with the probability of coincident failure and unavailability of the remaining train of SPC results in an acceptably low increase in Core Damage Frequency (CDF).
4. An evaluation was completed to determine the appropriate maintenance activities required to significantly reduce the leaking MSRVs.
5. All Control Room operating crews and supervision have been trained regarding mandatory procedure compliance. The training emphasized that unauthorized departure from a Plant procedure is forbidden, especially in response to a recurring problem, even though the reasons for the problem are well understood. If difficulties are encountered while performing the procedure, it is management's expectations that performance of the procedure should be suspended, the Plant placed in a safe condition, and the conditions evaluated and appropriately resolved.

D. Further Corrective Action

1. Plant procedure PPM 2.4.2, RHR, will be changed by March 1, 1993, to indicate that a train in SPC or SPS must be considered inoperable. The change will also indicate that any combination of two train operation of SPC and/or SPS: results in a loss of the safety function of SPC, SPS, and DWS, is required to be reported to the NRC within four hours, and is forbidden under non-emergency conditions.

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION											
FACILITY NAME (1) Washington Nuclear Plant - Unit 2		DOCKET NUMBER (2) 0   5   0   0   0   3   9   7					LER NUMBER (8) Year   Number   Rev. No. 93   001   01			PAGE (3) 6 OF 7	
TITLE (4) INOPERABLE SUPPRESSION POOL COOLING DUE TO POTENTIAL WATERHAMMER											

2. All Shift Managers and Control Room Supervisors will be required to read this LER by March 31, 1993.
3. The leaking MSRVs will be reworked, up to a maximum number of 12, by July 1, 1993, to significantly reduce MSRV leakage. Long Range Planning provides for rework of an additional 10 valves each outage as the need dictates.
4. Plans will be developed by July 1, 1993, to perform an evaluation that will determine the long term solution to the RHR design deficiency and provide an expected completion date for the evaluation.

#### Safety Significance

The safety significance of these events is negligible. The probability of a LOP-LOCA occurring during the four times both trains of SPC were operating was at least two orders of magnitude less than the estimated CDF for WNP-2 of  $5.4E-5$  events per year. Therefore, the increase to the CDF was insignificant. This condition posed no threat to the health and safety of Plant personnel or the public.

#### Similar Events

One or more of the following original design deficiencies could have caused loss of the safety function of the respective system following a Loss-of-Offsite Power (LOOP) or LOCA.

LER 84-013 documents an original design deficiency where undersized fuses to the fan motors of the Containment Atmosphere Control (CAC) System would have prevented the CAC System from performing its safety function following a LOCA.

LER 92-007 documents an original design deficiency where location of restricting orifices in the RHR return lines to the Suppression Pool could cause the loss of the safety function of the CAC System following a LOCA that required CAC operation.

LER 92-028 documents an original design deficiency where the Diesel Generator (DG) room normal air handling fans (DMA-FN-12, -22, and -32) would not restart following a LOOP, which could result in the loss of the safety function of the three DGs.



LICENSEE EVENT REPORT (LER) TEXT CONTINUATION											
FACILITY NAME (1)  Washington Nuclear Plant - Unit 2	DOCKET NUMBER (2)  0   5   0   0   0   3   9   7	LER NUMBER (8)						PAGE (3)			
		Year		Number		Rev. No.					
		93	001	01	7	OF	7				
TITLE (4) INOPERABLE SUPPRESSION POOL COOLING DUE TO POTENTIAL WATERHAMMER											

### EIIS Information

#### Text Reference

#### EIIS Reference

Containment Atmosphere Control System  
 Control Rod Drive System  
 Emergency Power System for HPCS  
 Plant AC Distribution System  
 Main Steam System  
 RHR/Containment Spray  
 Suppression Pool System  
 Diesel Building HVAC

<u>System</u>	<u>Component</u>
BB	--
AA	--
EK	--
EA	--
SB	RV
BO	--
BT	--
VJ	--