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RECIP.NAME RECIPIENT AFFILIATION

SUBJECT: LER 93-001-02:on 930213,determined that both trains of SPC of RHR system were inoperable due to LOP coincident w/LPCI safety function.Reviews & safety evaluations has been enhanced.W/940603 ltr.

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WASHINGTON PUBLIC POWER SUPPLY SYSTEM

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June 3, 1994
G02-94-131

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. 93-001-02**

References: Letter GO2-93-293, dated November 1, 1993, JV Parrish (SS) to USNRC "Resolution of Residual Heat Removal (RHR) Potential Water Hammer Issue"

Transmitted herewith is Revision 2 to Licensee Event Report (LER) No. 93-001 for the WNP-2 Plant. This report is submitted in response to the report requirements of 10CFR50.73 and incorporates the positions and conclusions of the referenced letter.

In summary, the reference concluded that the accident sequence of a loss of power (LOP) coincident with a loss of coolant accident (LOCA), occurring while an RHR loop is in the Suppression Pool Cooling (SPC) or Suppression Pool Spray (SPS) mode was not in the original WNP-2 design basis. This was supported by General Electric (GE), the plant designer. GE confirmed that with limited use of RHR in SPC/SPS the sequence of events resulting in a RHR water hammer event was not sufficiently credible to be included in the design basis accident analyses. The Supply System considers that despite increased WNP-2 SPC/SPS use beyond that originally expected adherence to limits on SPC/SPS operation will maintain the probability of the postulated scenario as a very low likelihood event. Hence, the increased usage does not warrant a new or changed design basis accident analysis.

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This revision of LER 93-001 is consistent with the referenced resolution and revises the commitment reported in LER 93-001 such that, with adherence to limits on duration of RHR operation in SPC/SPS, an RHR loop in SPC/SPS will not be declared inoperable and entry into a Technical Specification Action Statement will not be required. However, this does not represent a change in the recognized commitment to ensure that two RHR loops are not put in SPC/SPS at the same time during normal operation.

This submittal also revises the root cause and corrective action statements to be consistent with the referenced evaluation.

The reference stated that the Supply System had "adopted procedures to ensure that use of the RHR in the SPC mode will not exceed acceptable time limits" (page one, paragraph four). A clarification of this statement is needed to describe the temporary controls and permanent procedures used. Procedural controls for SPC/SPS mode time limits were approved on February 23, 1994. Prior to that time, a Technical Staff Engineer was monitoring the number of hours per week spent in the SPC/SPS mode of RHR System operation to assure that established time limits were not exceeded.

Should you have any questions or desire additional information, please call me or D.A. Swank at (509) 377-4563

Sincerely,

J. V. Parrish

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

JVP/PLP/my
Enclosure

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NRC Sr. Resident Inspector (Mail Drop 927N, 2 Copies)
INPO Records Center - Atlanta, GA
DL Williams, BPA (Mail Drop 399)

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) 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	9 3	0 0 1	0 2	0 5	1 1	9 4		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 1 0 0

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

[ ] 20.402(b)	[ ] 20.405(C)	[ ] 50.73(a)(2)(iv)	[ ] 77.71(b)
[ ] 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)	[X] 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 (AREA CODE)
P.L. Powell, Licensing Engineer	5 0 9 3 7 7 - 4 2 8 1

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)	[X] NO				

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), and Low Pressure Core Injection (LPCI) safety functions. On December 22, 1992, engineering evaluations concluded that water hammer could fail the train of RHR in SPC/SPS 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) management methods that failed to identify this problem during review of a 1987 NRC Information Notice (IN), and 2) failure to follow procedures which led to inappropriate two train operation of SPC.

Leaking Main Steam Relief Valves (MSRV) were reworked during the 1993 Refueling Outage to minimize leakage. This LER was made required reading for all Control Room supervision. An evaluation performed subsequent to submittal of LER 93-001 revision 0 concluded that, with limits on RHR operation in SPC, RHR operability was not compromised and it was overly conservative to enter Technical Specification Action Statements when an RHR system was put in the SPC mode. Therefore, procedures changed as a result of the initial LER submittal, to declare an RHR system in SPC/SPS inoperable, will be revised such that an RHR system in SPC/SPS will remain operable. This evaluation was provided in Supply System letter G02-93-293 dated November 1, 1993.

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 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/SPS coincident with a LOP-LOCA. WNP-2 engineering personnel were unaware of this potential vulnerability of RHR and 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 to the Startup transformer) to the associated bus of RHR in SPC/SPS causes the corresponding RHR pump to stop, which 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 a Licensing Engineer in support of a reportability evaluation for conditions when both SPC/SPS 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. On January 13, 1993, the 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. This represented a condition that could have prevented the RHR System from performing its SPC, SPS, and DWS safety functions. The low pressure safety function was still capable of being performed by either RHR "C" or the Low Pressure Core Spray (LPCS) System.

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 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/SPS 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/SPS 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 of SPC/SPS 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/SPS 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/SPS mode of RHR to one train operation per the recommended solution used by Susquehanna. However, the train of RHR in SPC/SPS 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/SPS.

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" in the SPC mode. 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 procedure PPM 2.4.2 was changed in 1990 to preclude two train operation of SPC/SPS, 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 where the condition was prohibited by the TS given the fact that the RHR train in SPC or SPS should have been 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/SPS mode of RHR to a LOP-LOCA initially reported in LER 93-001 was the inadequacy of the original design analysis. However, as stated in Supply System letter GO2-93-293 dated November 1, 1993, submitted subsequent to LER 93-001, it was concluded that at the time of the original design the sequence of events potentially resulting in an RHR system water hammer event was not sufficiently credible to be included in the design accident analysis. General Electric (GE), the plant designer, supports this conclusion. Therefore, there was no need to include a capability to withstand the postulated event in the WNP-2 design basis. This root cause (inadequate original design analysis), although considered to be correct at the time LER 93-001 revision 0 was submitted, is no longer considered to be correct. This is based on limiting use of RHR in the SPC/SPS mode. Even though use of RHR in SPC/SPS has increased, with adherence to limits on the use of RHR in the SPC/SPS mode, the increased use does not warrant a new or changed design basis accident analyses.

2. A root cause for 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 for inappropriate two loop operation of SPC in 1991 and 1992 was failure to follow procedures.

C. Further Corrective Action Taken

1. The OER process has required that OER issues be discussed with affected departments (cross-disciplinary review). In June 1993 Plant Procedure PPM 1.10.4, "External Operating Experience Review," was revised to require that primary contributors to resolving the OER be listed on the OER issue form.

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. 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. Presently, the SPC/SPS 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. However, as supported by the resolution of the RHR potential water hammer issue provided in Supply System letter GO2-93-293 dated November 1, 1993, the Supply System intends to change the affected procedures such that the SPC/SPS train in operation will not be declared inoperable and the appropriate Technical Specification action statements for an inoperable RHR loop will not be required to be entered. Assuming one train of SPC/SPS 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 system failure and unavailability of the remaining train of SPC/SPS, does not change the margin of safety.

   Procedure 3.1.10, "Operating Data and Logs," has been modified to track SPC/SPS cumulative hours in operation to a quarterly limit with a guide to maintaining the average less than 15 hours per week. If the quarterly limit is exceeded, an assessment with respect to core damage frequency will be performed.

4. An evaluation was completed to determine the appropriate maintenance activities required to significantly reduce the leaking MSRVs.

5. 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, was revised on March 17, 1993, to indicate that a train of RHR in SPC or SPS must be considered inoperable. The change also included a caution statement against any combination of two train operation of SPC and/or SPS unless directed to do so by Emergency Operating Procedures.

   As discussed above, based on the resolution of the RHR potential water hammer issue provided in Supply System letter GO2-93-293 dated November 1, 1993, the Supply System intends to change the affected procedures such that the SPC/SPS train in operation will not be declared inoperable and the appropriate Technical Specification action statements for an inoperable RHR loop will not be required to be entered. The caution against two train operation of SPC and/or SPS will remain.

2. All Shift Managers and Control Room Supervisors were required to read revision 0 of this LER by March 31, 1993.

3. Thirteen MSRVs were reworked in 1993 to significantly reduce MSRV leakage. Nine MSRVs are scheduled for maintenance during the 1994 outage. Future repairs to MSRVs, if necessary, will be evaluated along with other equipment repairs and plant modifications based on necessity and resource availability.

4. The resolution of the RHR potential water hammer issue was submitted in Supply System letter GO2-93-293 dated November 1, 1993. In summary, it concluded that a LOP/LOCA accident analysis of RHR in the SPC/SPS mode, was not included in the original design basis accident analysis for WNP-2 and, with adherence to limitations on the hours of RHR operation per year in the SPC/SPS mode, the concern remains a low likelihood event and continued operability of RHR in SPC/SPS is justified.

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 recently revised and quantified IPE probabilistic safety assessment model Core Damage Frequency (CDF) value of 1.8E-5. (The CDF value was recently revised, and will be formally submitted to the staff.) Therefore, the increase to the CDF caused by these events 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 Power (LOP) 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 have caused 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 LOP, which could have resulted in the loss of the safety function of the three DGs.

EIIS Information

Text Reference	EIIS Reference System	Component
Containment Atmosphere Control System	BB	--
Control Rod Drive System	AA	--
Emergency Power System for HPCS	EK	--
Plant AC Distribution System	EA	--
Main Steam System	SB	RV
RHR/Containment Spray	BO	--
Suppression Pool System	BT	--
Diesel Building HVAC	VJ	--