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FACIL:50-397 WPPSS Nuclear Project, Unit 2, Washington Public Powe 05000397
AUTH.NAME AUTHOR AFFILIATION
ARBUCKLE,J.D. Washington Public Power Supply System
BAKER,J.W. Washington Public Power Supply System
RECIP.NAME RECIPIENT AFFILIATION

SUBJECT: LER 91-022-00:on 910905,inboard RHR sys shutdown cooling supply valve automatically isolated on high suction line flow signal.Caused by instrumentation drift.Sys realigned & loop B placed back in svc.W/911007 ltr.

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TITLE: 50.73/50.9 Licensee Event Report (LER), Incident Rpt, etc.

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WASHINGTON PUBLIC POWER SUPPLY SYSTEM

P.O. Box 968 • 3000 George Washington Way • Richland, Washington 99352

Docket No. 50-397

October 7, 1991

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

Subject: NUCLEAR PLANT NO. 2
LICENSEE EVENT REPORT NO. 91-022

Dear Sir:

Transmitted herewith is Licensee Event Report No. 91-022 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.

Very truly yours,

J.W. Baker
WNP-2 Plant Manager

Enclosure:
Licensee Event Report No. 91-022

cc: Mr. John B. Martin, NRC - Region V
Mr. C. Sorensen, NRC Resident Inspector (M/D 901A)
INPO Records Center - Atlanta, GA
Ms. Dottie Sherman, ANI
Mr. D. L. Williams, BPA (M/D 399)
NRC Resident Inspector - walk over copy

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NRC FORM 366 (6-89)

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104
EXPIRES: 4/30/92

ESTIMATED BURDEN PER RESPONSE TO COMPLY WTH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE RECORDS AND REPORTS MANAGEMENT BRANCH (P-530), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
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TITLE (4) Residual Heat Removal (RHR) System Shutdown Cooling Isolation On High Suction Line Flow Due To Differential Pressure Instrumentation Drift

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 NUMBER(S)
0 9	0 5	9 1	9 1	0 2 2	0 0	1 0	0 7	9 1		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 4

POWER LEVEL (10): 0 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)	[X] 50.73(a)(2)(iv)	[ ] 73.71(b)
[ ] 20.405(a)(1)(i)	[ ] 50.36(c)(1)	[ ] 50.73(a)(2)(v)	[ ] 73.71(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	
	AREA CODE	
J. D. Arbuckle, Compliance Engineer	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	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)

[ ] YES *(If yes, complete EXPECTED SUBMISSION DATE)* [X] NO

EXPECTED SUBMISSION DATE (15): MONTH | DAY | YEAR

ABSTRACT *(Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines)* (16)

Abstract

On September 5, 1991 at 1122 hours while the Plant was shut down for an outage, the inboard Residual Heat Removal (RHR) System Shutdown Cooling Supply Valve (RHR-V-9) automatically isolated on a high suction line flow signal while Plant Operators were realigning RHR System shutdown cooling from Loop B to Loop A. Closure of RHR-V-9, a Nuclear Steam Supply Shutoff System (NSSSS) containment isolation valve, isolated and tripped RHR Shutdown Cooling Loops A and B which were in service at the time.

The cause of this event was instrumentation drift pertaining to Differential Pressure Indicating Switch RHR-DPIS-12B (RHR Shutdown Cooling Supply Leak Detection High Flow Indicator). As an immediate corrective action Plant Control Room Operators took appropriate steps to re-align the system and, at 1202 hours, one loop of RHR Shutdown Cooling (Loop B) was placed back into service, well within the two hours allowed for shutdown cooling to be out of service.

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TEXT (If more space is required, use additional NRC Form 366A's) (17)

Further corrective actions consisted of 1) recalibrating pressure switches RHR-DPIS-12A and RHR-DPIS-12B, and 2) revising the RHR- DPIS-12B calibration procedure to require that a channel functional test be performed monthly.

This event posed no threat to the health and safety of either the public or Plant personnel.

Plant Conditions

Power Level - 0%
Plant Mode - 4 (Cold Shutdown)

Event Description

On September 5, 1991 at 1122 hours while the Plant was shut down for an outage, the inboard Residual Heat Removal (RHR) System Shutdown Cooling Supply Valve (RHR-V-9) automatically isolated on a high suction line flow signal during efforts associated with realigning the RHR System. Closure of RHR-V-9, a Nuclear Steam Supply Shutoff System (NSSSS) containment isolation valve, isolated and tripped RHR Shutdown Cooling Loops A and B which were in service at the time.

At the time of the event, Plant Operators were in the process of realigning RHR System shutdown cooling from Loop B to Loop A. At 1120 hours, with Loop B (RHR-P-2B) in service, Plant Operators started Loop A (RHR-P-2A) in preparation for changing the in- service RHR Shutdown Cooling System. However, within two minutes while Plant Operators were in the process of securing RHR-P-2B, RHR-V-9 automatically isolated with both loops of RHR shutdown cooling being in service. By design, the closure of RHR-V-9 also caused RHR-P-2A and RHR-P-2B to trip.

Plant design is such that RHR-V-9 will isolate on Reactor Pressure Vessel (RPV) low level (Level 3), Drywell pressure greater than 1.68 psig, or RHR Shutdown Cooling line excess flow. After verifying that there were no level or pressure transients during the event, Plant Operators concluded that the isolation of RHR-V-9 was most likely due to exceeding the high suction flow setpoint. The reason for the isolation was instrumentation drift pertaining to Differential Pressure Indicating Switch RHR-DPIS-12B (RHR Shutdown Cooling Supply Leak Detection High Flow Indicator).

Immediate Corrective Action

Plant Control Room Operators took appropriate action to re-align the system and, at 1202 hours, one loop of RHR Shutdown Cooling (Loop B) was placed back into service, well within the two hours allowed for shutdown cooling to be out of service.

Further Evaluation and Corrective Action

A. Further Evaluation

1. This event is reportable under 10CFR50.73(a)(2)(iv) as an event or condition that resulted in manual or automatic actuation of an Engineered Safety Feature.

2. There were no structures, systems or component that were inoperable prior to the start of the event that contributed to the event.

3. Two redundant differential pressure switches (RHR-DPIS-12A and RHR-DPIS-12B), one for each trip logic, monitor the RHR shutdown cooling suction line. Circuit logic operation is such that the output trip signal of each sensor initiates a logic trip and closure of either the inboard [RHR-V-9 (RHR-DPIS-12B)] or outboard [RHR-V-8 (RHR-DPIS-12A] isolation valve in the event of an excess flow condition in the RHR suction line.

4. The root cause of this event was instrument drift. On September 6, 1991 Plant Instrument and Control (I&C) Technicians checked the as-found trip settings of RHR- DPIS-12A and RHR-DPIS-12B and determined that the high excess flow trip setting for RHR-DPIS-12A had drifted to 147.5 inches (H2O) and the setting for RHR-DPIS-12B had drifted to 141.0 inches (H2O). The normal setpoint for these instruments is 174 inches (H2O). The inaccuracy of these mechanical instruments is 1.5 percent. As a result, with both loops of RHR in service and a combined flow of approximately 15,000 gpm, the drift value was close enough to the nominal process flow variances for RHR-DPIS-12B to sense a high flow condition in the RHR suction line and, by design, RHR-V-9 automatically isolated.

   A contributing factor was that the calibration procedure for RHR-DPIS-12A and RHR-DPIS-12B is only required by the Plant Technical Specifications to be performed during Operational Modes 1, 2 and 3. Although the instrument drift (which is common to Barton mechanical instruments) has been tracked by the Plant Technical Department since 1987, the calibration surveillance

had not been performed since April, 1991 due to RHR system configuration during the extended outage (a prerequisite of the procedure is to ensure that the RHR system is not performing any core cooling function). However, during non-outage periods, both instruments are calibrated monthly in accordance with Technical Specification surveillance requirements.

B. Further Corrective Action

1. On September 6, 1991 pressure switches RHR-DPIS-12A and RHR-DPIS-12B were recalibrated by Plant I&C Technicians.

2. Plant Procedure (PPM) 7.4.3.2.1.63, "RHR Shutdown Cooling Mode High Flow Isolation - CFT/CC," will be revised to require that a channel functional test be performed monthly during all Operational Modes.

Safety Significance

There is no safety significance with this event. Plant Control Room Operators took appropriate and timely action to re-align the RHR System. In addition, shutdown cooling was isolated for only 40 minutes, well within the time-frame (two hours) allowed by the Technical Specifications. Furthermore, during the event period there was no increase in reactor primary cooling loop temperature, which was 126 degrees at the start of the event. Accordingly, this event posed no threat to the health and safety of either the public or Plant personnel.

Similar Events

There have been several LERs pertaining to the loss of RHR shutdown cooling; however, none with a similar root cause.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

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TEXT (If more space is required, use additional NRC Form 366A's) (17)

EIIS Information

	EIIS Reference System	EIIS Reference Component
Residual Heat Removal (RHR) System	BO	---
RHR-V-9	BO	ISV
Nuclear Steam Supply Shutoff System (NSSSS)	BD	---
RHR-P-2A	BO	P
RHR-P-2B	BO	P
RHR-DPIS-12A	BO	PDIS
RHR-DPIS-12B	BO	PDIS
Reactor Pressure Vessel	NH	RPV