ACCELERATED DISTRIBUTION DEMONSTRATION SYSTEM

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:8907240014 DOC.DATE: 89/07/17 NOTARIZED: NO DOCKET #
FACIL:50-397 WPPSS Nuclear Project, Unit 2, Washington Public Powe 05000397
AUTH.NAME AUTHOR AFFILIATION
ARBUCKLE,J.D. Washington Public Power Supply System
POWERS,C.M. Niagara Mohawk Power Corp.
RECIP.NAME RECIPIENT AFFILIATION

SUBJECT: LER 89-025-00:on 890617,ESF actuations during excess flow check valve testing due to procedural inadequacies.
W/8 ltr.

DISTRIBUTION CODE: IE22T COPIES RECEIVED:LTR 1 ENCL 1 SIZE: 9
TITLE: 50.73/50.9 Licensee Event Report (LER), Incident Rpt, etc.

NOTES:

	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL
	PD5 LA	1	1	PD5 PD	1	1
	SAMWORTH,R	1	1			
INTERNAL:	ACRS MICHELSON	1	1	ACRS MOELLER	2	2
	ACRS WYLIE	1	1	AEOD/DOA	1	1
	AEOD/DSP/TPAB	1	1	AEOD/ROAB/DSP	2	2
	DEDRO	1	1	IRM/DCTS/DAB	1	1
	NRR/DEST/ADE 8H	1	1	NRR/DEST/ADS 7E	1	0
	NRR/DEST/CEB 8H	1	1	NRR/DEST/ESB 8D	1	1
	NRR/DEST/ICSB 7	1	1	NRR/DEST/MEB 9H	1	1
	NRR/DEST/MTB 9H	1	1	NRR/DEST/PSB 8D	1	1
	NRR/DEST/RSB 8E	1	1	NRR/DEST/SGB 8D	1	1
	NRR/DLPQ/HFB 10	1	1	NRR/DLPQ/PEB 10	1	1
	NRR/DOEA/EAB 11	1	1	~~NRR/DREP/RPB 10~~	2	2
	NUDOCS-ABSTRACT	1	1	REG FILE 02	1	1
	RES/DSIR/EIB	1	1	~~RES/DSR/PRAB~~	1	1
	RGN5 FILE 01	1	1			
EXTERNAL:	EG&G WILLIAMS,S	4	4	FORD BLDG HOY,A	1	1
	L ST LOBBY WARD	1	1	LPDR	1	1
	NRC PDR	1	1	NSIC MAYS,G	1	1
	NSIC MURPHY,G.A	1	1			

Cert No P132528119

R I D S / A D D S

R I D S / A D D S

NOTE TO ALL "RIDS" RECIPIENTS:

PLEASE HELP US TO REDUCE WASTE! CONTACT THE DOCUMENT CONTROL DESK, ROOM P1-37 (EXT. 20079) TO ELIMINATE YOUR NAME FROM DISTRIBUTION LISTS FOR DOCUMENTS YOU DON'T NEED!

FULL TEXT CONVERSION REQUIRED
TOTAL NUMBER OF COPIES REQUIRED: LTTR 43 ENCL 42

WASHINGTON PUBLIC POWER SUPPLY SYSTEM

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

Docket No. 50-397

July 17, 1989

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

Subject: NUCLEAR PLANT NO. 2
LICENSEE EVENT REPORT NO. 89-025

Dear Sir:

Transmitted herewith is Licensee Event Report No. 89-025 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,

JW Baker

C.M. Powers (M/D 927M)
WNP-2 Plant Manager

CMP:lg

Enclosure:
Licensee Event Report No. 89-025

cc: Mr. John B. Martin, NRC - Region V
Mr. C.J. Bosted, NRC Site (M/D 901A)
INPO Records Center - Atlanta, GA
Ms. Dottie Sherman, ANI
Mr. D.L. Williams, BPA (M/D 399)

Cert No
P132528119

IE22
1/1

NRC Form 366 (9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/88

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 0 8

TITLE (4) Engineered Safety Feature (ESF) Actuations During Excess Flow Check Valve Testing Due to Procedural Inadequacies

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 6	1 7	8 9	8 9	— 0 2 5 —	0 0	0 7	1 7	8 9		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 - 2 1 1 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)

During the performance of the Plant procedure for Excess Flow Check Valve Testing, three separate but related events occurred which caused Engineered Safety Feature (ESF) isolations and actuations. These events are combined into one LER in accordance with the guidance procided in NUREG 1022 (Supplement No. 1) as "similar events that are part of the same activity or test program." At the time of the events the Plant was in a shutdown condition for the annual maintenance and refueling outage. The events are described as follows:

1) On June 17, 1989 at 1222 hours, the inboard Residual Heat Removal (RHR) Shutdown Cooling Supply Valve (RHR-V-9) automatically isolated during excess flow check valve testing. The root cause of this event is procedural inadequacy in that the procedure did not caution against increasing pressure to a value that would cause the RPV high pressure isolation actuation. Plant Operators were in the process of increasing reactor pressure by raising vessel level with water from the Control Rod Drive (CRD) System. When RPV pressure reach 122 psig, a Reactor Recirculation (RRC) System pressure switch actuated and, by design, RHR-V-9 closed which isolated RHR Shutdown Cooling.

2) On June 18, 1989 at 1045 hours, during the performance of excess flow check valve testing, a Reactor Protection System (RPS) "A" half-scram and a High Pressure Core Spray (HPCS) Diesel Generator (DG-3) start occurred due to a pressure transient which affected several reactor vessel level instruments. The pressure transient was caused when a Plant Instrument and Control (I&C) Technician inadvertently opened the wrong valve during performance of the procedure. The root cause of this event is procedural inadequacy in that the procedure did not list the corresponding drain

Abstract (cont'd)

valves required to test specific excess flow check valves, and instead relied upon the ability of the field technicians to trace the instrument tubing from the instrumentation to the drain valves. As a result, the I&C Technicians opened the drain valve associated with the opposite side (high) of instrument MS-LT-27 instead of the low side. This caused a lowering in the pressure to instruments connected to two separate containment penetrations and resulted in the isolations and actuations.

3) On June 18, 1989 at 1814 hours, the inboard Residual Heat Removal (RHR) Shutdown Cooling Supply Valve (RHR-V-9) again isolated on a Reactor Pressure Vessel (RPV) high pressure signal during excess flow check valve testing. The root cause of this event is procedural inadequacy in that the procedure did not explicitly state the impact of isolating instrument MS-PT-51A, nor did it provide a clear caution statement directing that the Control Room Operators be alerted that MS-PT-51A was isolated and pressure indication impaired. In accordance with the procedure, Plant I&C Technicians had valved out MS-PT-51A in order to test Excess Flow Check Valve PI-EFC-X114. When the instrument was valved out of service, the pressure at the transmitter was trapped and the indication in the Control Room stayed at that pressure. Because the indicated-pressure remained constant, Plant Operators had no reason to vent the reactor vessel pressure during the test. As a result, eventually reactor pressure reached a point high enough to trip a Reactor Recirculation (RRC) pressure switch which, by design, closed RHR-V-9 and isolated RHR Shutdown Cooling.

Immediate corrective action consisted of resetting the isolations and actuations and restoring systems to service, and (after the third event) suspending excess flow check valve testing until a management team could investigate and develop a plan to prevent further problems. Further corrective actions consist of revising the procedure to identify the consequences of each instrument removed from service, and performing an evaluation of RHR Shutdown Cooling isolations which have occurred.

These events posed no threat to the health and safety of either the public or Plant personnel.

Plant Conditions

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

Event Description

During the performance of Plant Procedure (PPM) 7.4.6.3.4.1, "Excess Flow Check Valve Testing," three separate but related events occurred which caused Engineered Safety Feature (ESF) isolations and actuations. These events are combined into one LER in accordance with the guidance provided in NUREG 1022 (Supplement No. 1) as "similar events that are part of the same activity or test program." At the time of the events the Plant was in a shutdown condition for the annual maintenance and refueling outage. The events are described as follows:

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Washington Nuclear Plant - Unit 2	0 5 0 0 0 3 9 7	8 9	— 0 2 5 —	0 0	0 3 OF 0 8

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

1) On June 17, 1989 at 1222 hours, the inboard Residual Heat Removal (RHR) Shutdown Cooling Supply Valve (RHR-V-9) automatically isolated on a Reactor Pressure Vessel (RPV) high pressure signal during excess flow check valve testing. The valve is a Nuclear Steam Supply Shutoff System (Containment Isolation) valve. Closure of RHR-V-9 isolated RHR Shutdown Cooling Loop A which was in service at the time of the event.

As part of the testing, the procedure required that reactor vessel pressure be maintained at 100 psig minimum pressure. To meet this requirement, Plant Operators were in the process of increasing reactor pressure by raising vessel level with water from the Control Rod Drive (CRD) System.

When reactor pressure reached 122 psig as indicated on Main Steam Pressure Indicator MS-PI-9 in the Control Room, a Reactor Recirculation (RRC) System pressure switch (RRC-PS-18B) actuated and, by design, RHR-V-9 closed which isolated RHR Shutdown Cooling.

2) On June 18, 1989 at 1045 hours, during the performance of excess flow check valve testing, a Reactor Protection System (RPS) "A" half-scram and a High Pressure Core Spray (HPCS) Diesel Generator (DG-3) start occurred due to a pressure transient which affected several reactor vessel level instruments.

The pressure transient was caused when a Plant Instrument and Control (I&C) Technician inadvertently opened the wrong valve during performance of the procedure.

Excess flow check valve testing is accomplished by isolating all instruments utilizing the containment penetration associated with the excess flow check valve being tested. When the instrumentation is isolated, the drain valve for the penetration is opened and both drain flow and Control Room excess check flow valve position indications are observed. The Test Director in the Control Room verifies indication of valve closure and the I&C technician verifies that the flow of water to the drain has stopped. A closed indication and significant flow decrease indicate a successful test.

At the time of the event, Excess Flow Check Valve PI-EFC-X72A was being tested. The I&C Technicians had valved out the instruments associated with containment penetration X72A. These instruments were MS-LT-27 and RFW-DPT-17. Instrument MS-LT-27 is a differential pressure instrument having two sensing lines (high and low), and two drain lines associated with two different containment penetrations. The procedure required that the I&C technician open the drain valve associated with the side of MS-LT-27 connected to penetration X72A. However, the procedure did not list the valves necessary to test specific excess flow check valves, nor did it differentiate between high and low side.

As a result, the I&C Technicians opened the drain valve associated with the opposite side (high) of MS-LT-27 instead of the low side. The drain valve that was opened is associated with penetration X110 and is connected to other level instrumentation. When the valve was opened, it caused a lowering in the pressure to the instruments connected to that penetration. Penetration X110 is also the variable leg of instrument MS-LIS-24C which tripped at the Reactor Level-3 setpoint and caused the RPS, Division 1, half-trip.

In addition, two other instruments (MS-LIS-31B and MS-LIS-31D) connected to penetration X110 share another penetration (X112: reference leg) with MS-LIS-24C and MS-LIS-100B which were affected by the event. The pressure disturbance on penetration X110 was transmitted through the bellows of MS-LIS-24C and MS-LIS-100B to penetration X112. As a result, all of the instruments connected to penetration X112 were affected by this event. Instruments MS-LIS-31B and MS-LIS-31D were affected enough to trip at their Reactor Level-2 (-50") setpoint which caused the HPCS Diesel to start (and attain rated speed) and the HPCS injection valve (HPCS-V-4) to open. As a precautionary measure, the HPCS pump (HPCS-P-1) motor had previously been placed out of service and, as a result, no HPCS injection occurred.

3) On June 18, 1989 at 1814 hours, the inboard Residual Heat Removal (RHR) Shutdown Cooling Supply Valve (RHR-V-9) again isolated on a Reactor Pressure Vessel (RPV) high pressure signal during excess flow check valve testing. As during the previous event (June 17, 1989), closure of RHR-V-9 isolated RHR Shutdown Cooling. During this event, RHR Shutdown Cooling, Loop B, was in service.

In accordance with the procedure, Plant I&C Technicians valved out the instruments associated with containment penetration in order to test Excess Flow Check Valve PI-EFC-X114. One of the instruments connected to penetration X114 is MS-PT-51A which provides the reactor steam dome pressure to instruments MS-LR/PR-623A and MS-PI-9 in the main Control Room. When MS-PT-51A was valved out of service, the pressure at the transmitter was trapped and the indication in the Control Room stayed at that pressure until the transmitter was valved back into service after the event.

Control Room Operators were controlling reactor pressure by blowing down excess water provided by the Control Rod Drive (CRD) System whenever the pressure in the vessel approached 110 psig. Because the Control Room - indicated pressure remained constant, Plant Operators had no reason to vent the reactor vessel pressure. As a result, eventually the reactor pressure reached a point high enough to trip RRC-PS-18 which closed RHR-V-9 and isolated RHR Shutdown Cooling.

<u>Immediate Corrective Action</u>

1) June 17, 1989 (1222 Hours)

- Control Room Operators lowered RPV pressure, reset the RHR-V-9 isolation and opened the valve. At 1301 hours, RHR Shutdown Cooling was restored.

2) June 18, 1989 (1045 Hours)

- Control Room Operators reset the RPS "A" half-scram and HPCS actuation signals. By 1108 hours, the HPCS Diesel Generator was secured and restored to standby status.

3) June 18, 1989 (1814 Hours)

- Control Room Operators lowered RPV pressure, reset the RHR-V-9 isolation and opened the valve. At 1845 hours, RHR Shutdown Cooling was restored.
- As directed by the Assistant Plant Manager, excess flow check valve testing was suspended and a management team assigned to investigate and develop a plan to prevent further problems.

Further Evaluation and Corrective Action

A. Further Evaluation

1. All three events are reportable under 10CFR50.73(a)(2)(iv) as "an event or condition that resulted in manual or automatic actuation of any Engineered Safety Feature (ESF)."

2. There were no structures, systems or components that were inoperable at the start of these events that contributed to the events.

3. The root causes of these events are as follows:

   a) June 17, 1989 Event

   - Procedural Inadequacy: The procedure governing the excess flow check valve surveillance testing did not caution against increasing pressure to a value that would cause the RPV high pressure isolation actuation. The procedure specified a minimum pressure (100 psig) but did not identify a range of allowable pressures. In addition, Control Room Operators were aware of the Technical Specification Trip Setpoint of 125 psig but did not realize that, due to instrument inaccuracies and head corrections, the trip could occur at indicated pressures below 125 psig. A review of the administratively-acceptable trip value for RRC-PS-18B indicated an acceptable trip value as low as 114 psig. Had Plant Operators been provided with either a pressure band within which to perform the test, or a caution against increasing pressure above a specified value, this event would not have occurred.

b) June 18, 1989 (1045 Hours) Event

- Procedural Inadequacy: The procedure did not list the corresponding drain valves required to test specific excess flow check valves, and instead relied upon the ability of the field technicians to trace the instrument tubing from the instrumentation to the drain valves. Had the drain valves been positively identified by tag number in the procedure, the need to trace tubing to the correct valve would have been eliminated and this event would not have occurred.

c) June 18, 1989 (1814 Hours) Event

- Procedural Inadequacy: The procedure did not explicitly state the impact of isolating MS-PT-51A, nor did it provide a clear caution statement directing that the Control Room Operators be alerted that MS-PT-51A was isolated and pressure indication impaired. While the Test Director was aware that MS-PI-9 was being used by the Control Room Operators to monitor reactor pressure, it was not obvious that MS-PI-9 .input comes from MS-PT-51A. At the point where MS-PT-51A is isolated in the procedure, there is a caution to, "Inform the Control Room Operator that pressure indication for reactor vessel leakage test is affected PPM 7.4.0.5.38." Because there were no leakage tests in progress at the time, the Test Director did not specifically inform the Control Room Supervisor of this instrument isolation. A contributing factor was that a verbal agreement had been established whereby the Test Director would inform the Control Room Supervisor whenever the technicians were moving to the next valve. In the case of PI-EFC-X114, this was not done. However, had the caution statement specifically stated that pressure indication on MS-PI-9 would not be available during testing of PI-EFC-X114 (and to alert the control room), this event would not have occurred.

B. Further Corrective Action

1. June 17, 1989 Event

- The procedure was modified to inform the Shift Manager that high reactor pressure (above 125 psig) would cause an RHR Shutdown Cooling isolation and a recommendation to control pressure at less than 125 psig was included. In addition, the limit for the minimum pressure required was reduced from 100 psig to 85 psig.

- In addition, the procedure was modified to provide a pressure band (85-110 psig) to avoid the possibility of a trip at an indicated valve below 125 psig.

2. June 18, 1989 (1045 Hours) Event

   - The procedure was modified to require that the Test Director visually verify that the correct drain valve has been located prior to opening to test the excess flow check valve. This action was taken until such time that formal revision to the procedure could be made.

3. June 18, 1989 (1814 Hours) Event

   - As a result of the management review previously discussed, additional changes to the procedure were made which included requiring that instruments valved out-of-service be caution tagged to indicate status.

4. The procedure (PPM 7.4.6.3.4.1) will be revised prior to the next maintenance and refueling outage to identify the consequences of each instrument removed from service, and to also include specific valve identification.

5. An overall evaluation of the RHR Shutdown Cooling isolations which have occurred is currently being performed by the Plant Technical and Nuclear Safety Assurance Groups. The generic issues associated with those RHR Shutdown Cooling isolations which occurred during the recent maintenance and refueling outage will be addressed in this evaluation. The evaluation is nearing completion and an internal formal report will be issued.

Safety Significance

There is no safety significance associated with these events. For the loss of RHR Shutdown Cooling events, Shutdown Cooling was isolated for 39 minutes (June 17, 1989) and 31 minutes (June 18, 1989) respectively, well within the time frame allowed by the Technical Specifications.

Regarding the HPCS Diesel start, no Plant condition warranting the ESF actuation actually existed and the HPCS actuations occurred as designed.

Accordingly, these events posed no threat to the health and safety of either the public or Plant personnel.

Similar Events

There have been several events associated with the loss of RHR Shutdown Cooling; however, none with the same root cause (i.e., procedural inadequacies during excess flow check valve testing).

NRC Form 366A
(9-83)

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/88

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Washington Nuclear Plant - Unit 2	0 5 0 0 0 3 9 7	8 9	— 0 2 5 —	0 0	0 8 OF 0 8

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

EIIS Information

Text Reference	EIIS Reference System	Component
Residual Heat Removal (RHR) System	BO	---
RHR-V-9	BO	ISV
Reactor Pressure Vessel (RPV)	NH	RPV
Nuclear Steam Supply Shutoff System	BD	---
Control Rod Drive (CRD) System	AA	---
MS-PI-9	SB	PI
Reactor Recirculation (RRC) System	AD	---
RRC-PS-18B	AD	PS
Reactor Protection System (RPS)	JM	---
High Pressure Core Spray (HPCS)	BG	---
HPCS Diesel Generator	EK	DG
Excess Flow Check Valve PI-EFC-X72A	NH	V
Containment Penetration X72A	NH	PEN
MS-LT-27	SB	LT
RFW-DPT-17	JB	PDT
Containment Penetration X100	NH	PEN
MS-LIS-24C	SB	LIS
MS-LIS-31B	SB	LIS
MS-LIS-31D	SB	LIS
Containment Penetration X112	NH	PEN
MS-LIS-100B	SB	LIS
HPCS-V-4	BG	INV
HPCS-P-1	BG	P
PI-EFC-X114	NH	V
Containment Penetration X114	NH	PEN
MS-PT-51A	SB	PT
MS-LR/PR-623A	SB	LR/PR