GULA RY INFORMATION DIST UTI SYSTEM (RIDS)

ACCESSION NBR: 8708120010 DOC. DATE: 87/08/05 NOTARIZED: NO DOCKET #
FACIL: 50-397 WPPSS Nuclear Project, Unit 2, Washington Public Powe 05000397

AUTH. NAME	AUTHOR AFFILIATION
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POWERS, C. M.	Washington Public Power Supply System

RECIP. NAME RECIPIENT AFFILIATION

SUBJECT: LER 87-022-00: on 870706, reactor scrammed on low reactor water level due to loss of single operating reactor feedwater (turbine) pump. Caused by inadequate procedures. Procedures revised & training will be provided. W/870805 ltr.

DISTRIBUTION CODE: IE22D COPIES RECEIVED: LTR 1 ENCL 1 SIZE: 7
TITLE: 50.73 Licensee Event Report (LER), Incident Rpt, etc.

NOTES:

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	NRR/PMAS/ILRB	1	1	REG FILE 02	1	1
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EXTERNAL:	EG&G GROH, M	5	5	H ST LOBBY WARD	1	1
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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 6

TITLE (4) Reactor Scram on Low Reactor Water Level caused by Plant Procedures which did not prevent a Power Breaker trip which caused a Reactor Feedwater Pump trip

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 7	0 6	8 7	8 7	— 0 2 2 —	0 0	0 8	0 5	8 7		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 0 3 2

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
Steven L. Washington, Compliance Engineer	5 0 9 3 7 7 - 2 0 8 0

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
B	E A	B K R	W 1 2 0	Y					
B	E A	2 7	W 1 2 0	Y					

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)

On July 6, 1987, at 0546 hours a Reactor Scram on low water level occurred due to the loss of the single (one of two) operating Reactor Feedwater (Turbine) Pump (RFW-P-1B). The event occurred when Plant operators transferred power on Plant Bus SM-2 (4160V) from the Startup Power Supply to the Normal Power Supply. Immediately after closing the Normal Power Supply Breaker N1-2 the Breaker tripped open leaving Bus SM-2 de-energized. Bus SM-2 through Plant Bus SL-21 (480V) powers both the Main and Auxilary Control Oil Pumps of the Feedwater Turbine which tripped due to low control oil pressure. The High Pressure Core Spray Diesel Generator (HPCS-DG-3) started on an SM-2 under voltage signal and assumed the load on Critical Bus SM-4 (4160V).

In response to the loss of Reactor Feedwater, Plant Operators manually initiated the Reactor Core Isolation Cooling (RCIC) System. Reactor water level decreased to -46 inches, near the Reactor Water Level-2 setpoint, and at -46 inches a Level-2 Nuclear Steam Supply Shutoff System (NSSSS) isolation occurred, including a Main Steamline Isolation. The RCIC system restored Reactor water level to normal.

The cause of the event was a bent finger in the C Phase Disconnecting Contact Finger Cluster which prevented the N1-2 Breaker from being fully inserted. The jarring motion of the Breaker closing caused the floor tripper to trip the Breaker.

IE22 V1

(Abstract continued on page 2)

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 7	— 0 2 2	— 0 0	0 2 OF 0 6

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

The root cause of the event is inadequate Plant procedures which did not provide operators a positive method of checking breaker position prior to closing the breaker. During the investigation of this event a new and more reliable method of verifying breaker position was developed. This new method has been incorporated into Plant Operator instructions.

There is no safety significance associated with this event since all safety systems functioned as designed.

Plant Conditions

a) Power Level - 32%
b) Plant Mode - 1 - Power Operation

Event Description

On July 6, 1987 at 0546 hours the Plant was operating at 32% on an ascension to 100% power when a Reactor scram occurred due to low reactor water level caused by the trip of the operating Reactor Feedwater Pump (RFW-P-1B). There are two Reactor Feedwater Pumps, but only one operating pump is required at the 32% power level. The event occurred when Plant Operators, while transferring Plant electrical loads from the Startup Power Supply to the Normal Power Supply, closed the Normal Power Supply N1-2 Breaker for Plant Bus SM-2(4160V). Immediately after the N1-2 Breaker was closed, the floor tripper tripped the Breaker open, de-energizing SM-2. Bus SM-2 powers Plant Bus SL-21(480V) which was de-energized. Both the Main and Auxiliary Control Oil Pumps for the RFW-P-1B turbine are powered from Bus SL-21. With the loss of both Control Oil Pumps the RFW-P-1B turbine tripped on low control oil pressure which caused a loss of feedwater flow to the Reactor.

Plant Bus SM-2 also powers Critical Bus SM-4 (4160V) which is the power supply for the High Pressure Core Spray (HPCS) System. An SM-2 bus undervoltage condition caused the High Pressure Core Spray Diesel Generator (HPCS-DG-3) to start and assume the load on Bus SM-4. The HPCS Diesel Generator ran for three hours and eleven minutes.

Following the Reactor Feedwater Pump "B" trip, Reactor water level began decreasing rapidly, and at Reactor Water Level -3(+13 inches) the Reactor scrammed. Plant Operators manually initiated the RCIC System to supply water to the Reactor. Reactor water level continued to decrease until the -46 inch level was reached at which point the Main Steamline Isolation Valves closed on an NSSSS Reactor Water Level -2 (-50 inches) trip. The RCIC System restored Reactor water level to the normal Plant operating level (+36 inches) in 15 minutes. Plant Operators then maneuvered the Plant to "Cold Shutdown" which was reached at 1029 hours on July 7,1987.

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 7	0 2 2	0 0	0 3 OF 0 6

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

The lowest reactor water level reached during the event was -46 inches, which is very close to the Reactor Water Level-2 trip setpoint of -50 inches and, as a result, some Level 2 trips did occur. There is no safety significance associated with this situation, since the Reactor water level never did reach the Level-2 trip setpoint. And, after the event all Reactor Level-2 instrumentation which did not trip was checked using Plant Surveillance Procedures and each instrument "as found" setpoint was within Plant Technical Specification allowable limits.

The following is a summary of Reactor Water Level-2 trips:

1) Level-2 Actuation	Trip	Event Status
Initiate RCIC	N/A	Manually initated prior to reaching Level-2
Initiate HPCS	No	Did not initiate
Isolate NSSSS Groups 1, 2, 3, 4, and 7	Yes	NSSSS Groups isolated
Trip RRC pumps off	Partial	RRC-P-1B tripped to off. RRC-P-1A did not trip but remained on at 15HZ.

The Plant recovery from the SM-2 de-energization was complicated by the pick-up of the SM-2 undervoltage and seal-in relays. By Plant design, the seal-in relay was never supposed to pick-up and no Plant Operator reset capability was provided. The seal-in relay caused the 27X-2 relay to remain energized, which prevented Plant Operators from repowering Bus SM-4 from Bus SM-2. Bus SM-2 was repowered after the event from the Startup Power Supply. Plant electricians reset the seal-in relay, and the HPCS Diesel Generator was secured at 0857 hours.

During the event, position indication on inboard MSIV-V-22D was lost. The problem was traced to the closed position limit switch, which was out of adjustment. The position switch was repaired.

The root cause of the event is inadequate Plant procedures, which did not provide a positive method of verifying that a breaker is fully inserted. Prior to this event, it was believed that the breaker control room light and a visual check of the breaker provided positive indication that a breaker was fully inserted. In the case of the N1-2 Breaker, a bent finger in the C Phase Disconnecting Contact Finger Cluster prevented the Breaker from fully inserting (approximately 1/8" from full insertion). The breaker control room light was on even though the Breaker was not fully inserted. In this position the Breaker floor tripper was very close to tripping, and the jarring motion as the Breaker closed caused the floor tripper to trip the Breaker immediately after it was closed.

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 7	— 0 2 2	— 0 0	0 4 OF 0 6

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

Immediate Corrective Action

Plant operators manually initiated the RCIC System. Plant operators stabilized the Plant and called in System experts to evaluate the Breaker problem.

Further Corrective Action and Evaluation

A task force of Plant Engineers and Electricians performed extensive troubleshooting and testing to determine the cause of the N1-2 Breaker (Westinghouse, DHP-350) trip. The cause was determined to be a bent finger in the C Phase Disconnecting Contact Finger Cluster, which prevented the Breaker from being fully inserted. In this position, approximately 1/8" from full insertion, the floor tripper mechanism is very close to tripping and the jarring motion caused by the Breaker closing caused the floor tripper to release and trip the breaker. Breaker trips by this mechanism were reproduced during post-event Breaker testing. Further analysis determined that the root cause of the event was inadequate plant instructions, in that existing instructions did not provide Plant Operators and Electricians with a positive method of verifying that a breaker is fully inserted. It was believed prior to this event that a visual check of the breaker and verification that the breaker control room light was on was a positive check that the breaker was fully inserted. This investigation found that verifying a clearance in the floor tripper mechanism provides a positive check that the breaker is fully inserted. As a result of this investigation, the following four Corrective Actions have been identified.

1) Plant Operator instructions have been revised to include a check of the floor tripper clearance to positively verify full insertion of the breaker.

2) Electrical Breaker Preventive Maintenance Procedures will be revised to include a check to ensure that the Disconnecting Contact Finger Clusters are properly aligned when the breaker is positioned in its cubical prior to being racked in.

3) Training for Plant Operators and Electricians will be provided on this new method of verifying breaker insertion.

4) The floor tripper clearance on comparable Plant Breakers was verified. Two other Plant Breakers were found which required adjustment of the floor tripper linkage. These adjustments have been made.

The Reactor Water Level-2 trip setpoints for the HPCS Reactor Water Level and ATWS RRC-P-1A Reactor Water Level instruments were verified using Plant Technical Specification Surveillance Procedures. All "as-found" trip setpoints for these instruments were within Technical Specification allowable limits.

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 7	— 0 2 2	— 0 0	0 5 OF 0 6

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

Plant Engineers and Electricians also investigated the cause of the SM-2 undervoltage relay and seal-in relay pick-ups. By engineering design, the seal-in relay was (Westinghouse, CV-2) set to pick-up at .2 amps, and by design, the maximum current in the seal-in circuit is .12 amps; therefore, the seal-in relay was never supposed to pick-up. Bench tests in the Plant Electrical Shop showed that with a sustained .12 amp current, the seal-in relay would move very slowly until it did pick-up. Since the seal-in relay was never supposed to actually pick-up, no Plant operator reset capability was provided. With the seal-in relay energized, the 27X-2 relay remained energized, which prevented Plant Operators from realigning the power supply for SM-4 back to SM-2. Plant Electricians were able to reset the seal-in relay by de-energizing and then re-energizing the SM-2 undervoltage relay circuit. A Plant Modification Package was implemented which changed the seal-in relay amperage required from .2 amps to 2 amps. This will prevent the seal-in relay from picking up again in this situation. Other similar plant relays were also modified as described above. Further, an Engineering evaluation will be performed to determine if the undervoltage relay can be modified to eliminate the seal-in feature.

An Engineering evaluation will be performed to determine the feasibility of separating the power supplies for the Main and Auxiliary Feedwater Turbine Control Oil Pumps.

Safety Significance

There is no safety significance associated with this event. All safety systems functioned as designed. There is no safety significance associated with the partial Reactor Water Level-2 actuations, since Reactor water level did not reach the Level-2 trip setpoint and all instrumentation which did not trip was verified to be within Plant Technical Specification allowable trip limits. The SM-2 undervoltage and seal-in relays did not prevent the HPCS Diesel Generator from performing its safety function of assuming Critical Bus SM-4 loads. This event posed no threat to the health and safety of the public or Plant personnel.

Similar Events

84-006

EIIS Information

Text Reference	EIIS Reference System	EIIS Reference Component
Reactor Feedwater Pump	SJ	P
Startup Power Supply (TRS)	EA	
Normal Power Supply (N1-2 Breaker)	EA	
High Pressure Core Spray Diesel Generator (HPCS-DG-3)	BG	DG

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 7	0 2 2	0 0	0 6 OF 0 6

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

EIIS Information

Text Reference	EIIS Reference System	Component
Plant Bus SM2 (4160V)	EA	BU
Plant Bus SL21 (480V)	EC	BU
Plant Bus SM4 (4160V)	EB	BU
Reactor Core Isolation Cooling (RCIC)	CEA	
Nuclear Steam Supply Shutoff System (NSSSS)	BD	
Reactor Feedwater Turbine Main and Auxiliary Control Oil Pumps	JB	PC
High Pressure Core Spray (HPCS)	BG	
Main Steamline Isolation Valves (MSIVs)	SB	ISV
Reactor Recirculation System Pumps (RRC-P-1A and 1B)	AD	P

WASHINGTON PUBLIC POWER SUPPLY SYSTEM

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

Docket No. 50-397

August 5, 1987

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

Subject: NUCLEAR PLANT NO. 2
LICENSEE EVENT REPORT NO. 87-022

Dear Sir:

Transmitted herewith is Licensee Event Report No. 87-022 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,

C M Powers

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

CMP:ac

Enclosure:
Licensee Event Report No. 87-022

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)

IE22
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