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FACIL: 50-397 WPPSS Nuclear Project, Unit 2, Washington Public Powe 05000397

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SUBJECT: LER 87-002-00: on 870322, reactor manually scrammed due to reactor feedwater pumps trip & rapidly decreasing reactor water level. Caused by fuse failure. Transient & subsequent recovery actions will be simplified. W/870421 ltr.

DISTRIBUTION CODE: IE22D COPIES RECEIVED: LTR 1 ENCL 1 SIZE: 8
TITLE: 50.73 Licensee Event Report (LER), Incident Rpt, etc.

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NRC Form 366 (9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/85

LICENSEE EVENT REPORT (LER)

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TITLE (4)
Manual Reactor Scram Due to Failure of Feedwater Level Control System Component

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 3	2 2	8 7	8 7	0 0 2	0 0	0 4	2 1	8 7		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1
POWER LEVEL (10): 0 7 1

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
X	J B	F U	B 0 4 5	N					

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 March 22, 1987 at 1940 hours, the reactor was manually scrammed as a result of an observed trip of both Reactor Feedwater (RFW) pumps and rapidly decreasing reactor water level. The RFW pumps tripped because of low suction pressure condition as a result of rapid acceleration of the pumps in response to an erroneous feedwater-steam flow mismatch signal. The erroneous signal was caused by a blown 0.25 ampere fuse in a summation circuit of the Feedwater Level Control System, resulting in a zero feedwater flow input. The cause of the fuse failure is unknown and, based upon extensive investigation, is not determinable.

Following the loss of feedwater flow, water level decreased to Level-2 (-50") and all required ESF and electrical alignment actuations occurred. During the recovery process, a valve alignment error occurred which complicated recovery operations and resulted in the flooding of the Main Steam Lines (MSLs). The automatic actions that occur on a Reactor Level-2 initiation present a complicated scenario. The transient and subsequent recovery actions required can be simplified by specific training and design changes. Sufficient piping system inspections, component testing and data analysis occurred to ensure that no detrimental affects were caused by the flooding of the MSLs.

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

Plant Conditions

Power Level - 71%
Plant Mode - 1 - Power Operation

Event

On March 22, 1987 at 1940 hours, the reactor was manually scrammed as a result of an observed trip of both Reactor Feedwater (RFW) pumps and rapidly decreasing reactor water level. The RFW pumps tripped because of low suction pressure as a result of the rapid acceleration of the pumps. The RFW pumps accelerated in response to an erroneous feedwater-steam flow mismatch signal. The erroneous signal was caused by a blown 0.25 ampere fuse in a Bailey-Type 752 summer card in the Feedwater Level Control (FWLC) System.

The effect of the fuse failure was to cause the summed feedwater flow signal input, into the RFW level controller circuit, to indicate less than zero feedwater flow. Accordingly, the summer output at zero volts indicated loss of feedwater flow, resulting in maximum feedwater flow demand. The turbine-driven feed pump speed transient resulted in a sudden decrease in RFW pump suction pressure below the RFW pumps instantaneous trip setpoint. As a result, both RFW pumps tripped.

Following the trip of both RFW pumps and subsequent loss of feedwater flow, vessel level rapidly dropped to Level-2 (-50"). Both High Pressure Core Spray (HPCS) and Reactor Core Isolation Cooling (RCIC) systems initiated at Level-2 to restore vessel level. Vessel level restoration was terminated at Level-8 (+54.5") by automatic closure of HPCS-V-4 and RCIC-V-45, both of which were by design. The initial level reduction (Level-2) caused actuation of Nuclear Steam Shutoff Supply System (NSSSS) Groups I through VIII, an ATWS Reactor Recirculation (RRC) Pump trip and the start of the ECCS support systems needed for HPCS and RCIC operation. All automatic ESF functions occurred as designed.

The Operators were presented with a complicated scram, which was exacerbated by failing to properly align the RFW system. The improper alignment led to the flooding of the Main Steam Lines (MSLS) and, based on data evaluation, indicated reactor vessel level was at or above the MSL inlet (108") three times (at 1957, 2014 and 2229 hours) during the event. While the MSLS were flooded, MSRV-4D was manually actuated for pressure control. As discussed later in this LER, an analysis of the effects of the MSL flooding transient showed they were within plant design margins.

In addition, evaluation of the prioritization applied to the various recovery actions indicates a training program deficiency. The combination of the multitude of actions necessary, coupled with training-based perceptions that secondary actions were of primary concern, further complicated the recovery process. The following is intended to provide an appreciation for the complexities of the recovery process and discusses the errors made.

(1) Level was returned to near normal by HPCS and RCIC which isolated on Level-8. Normal vessel letdown and pressure control were lost due to the NSSSS Group isolations. This required manual vessel pressure control using the Mainsteam Safety Relief Valves (MSRVs) if necessary, RCIC for vessel makeup and restoration of the Reactor Water Cleanup (RWCU) system to support vessel letdown control. The reactor pressure control efforts and cooling water system related isolations presented difficulties in establishing and maintaining RWCU operational. Again these evolutions, although successful, occupied a significant amount of shift resources and attention during the recovery effort.

(2) An erroneous report from an equipment operator in the RCIC Pump room that the unit had tripped on overspeed caused the control room operators to rely solely on the SRVs for pressure control and consider the condensate/feedwater system for the principle source of reactor level makeup. As a result, the crew considered RCIC unusable without spending time to troubleshoot. The condensate/feedwater system was selected as the desirable source of makeup as it provided automatic level control and would allow resources to be applied to other recovery efforts. Reliance on RCIC would not have required the reactor pressure reduction necessary to enable the condensate system to feed the reactor. Subsequent investigative efforts established the operability of RCIC.

(3) Significant crew attention and resources were devoted to recovery of containment cooling. As designed, at Level-2 the normal cooling water supply to containment isolates resulting in the inevitable increase in containment pressure. Without successful and rapid restoration of containment cooling, a drywell isolation due to high pressure would be unavoidable. This restoration involves resetting the isolation circuit upon level restoration, starting both the Plant Service Water (TSW) and Reactor Closed Cooling Pumps, and opening the cooling water containment isolation valves. Containment cooling was successfully recovered prior to a high drywell pressure condition.

(4) The operating crew failed to complete the valving sequence required to establish shutdown level control provided by successful alignment of the startup level control valve (RFW-LCV-10). As a result the RFW heater block valves were inadvertently left open. This, in part, resulted in a vessel overfill and the MSL flooding condition. Sufficient training had been provided to the operating crews on the procedure for placing the LCV in service. A recent error in the revision of a procedure that removed the specific operating steps required to place the LCV in service contributed to the error.

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(5) In conjunction with the valve alignment error, manual relief valve actuation during pressure control caused the flooding of the MSLs. MSRV actuation was performed to reduce pressure to allow the level control function to be assumed automatically by the condensate system via RFW-LCV-10. The current high level condition and misaligned RFW heater block valves led to the unthrottled filling of the reactor vessel.

Immediate Corrective Action

The plant operators acted promptly to maneuver the plant to a safe shutdown condition.

Further Evaluation and Corrective Action

A. Further Evaluation

1) The subject fuse was examined to determine the failure mode. The fuse wire was examined at 1700X magnification along with a spectograph of the chemical constituents. No specific cause for the failure was determined. The summer card was functionally tested using a replacement 0.25 amp fuse and responded correctly to various input combinations. The remaining three summer card 0.25 ampere fuses were examined and no degradation was observed. An engineering judgement was made to increase the fuse size in all Bailey 752 applications to one ampere. The size increase will minimize susceptibility to fuse degradation caused by inrush currents during energization and the mechanical shock during card or fuse handling. The size will also provide sufficient power supply protection.

2) An engineering evaluation was performed of the MSL flooding transient. The focus of the evaluation was on the MSLs, MSRVs and the effect of water carryover into the MSLs during the recovery operations. The evaluation considered the effects of the following potentially-damaging phenomena that could be associated with such a transient:

- Thermal expansion/contraction effects of cooler water entering the MSLs, and subsequent potential effects on piping, supports and MSIVs.

- Potential for hydrodynamic effects.

- Effect on MSRVs and associated piping.

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As a result of the engineering evaluation, the following has been determined:

- Filling of the MSLs with water to the MSIVs was analyzed previously as part of the original design basis of the plant. The passage of water down the lines with open MSIVs was evaluated for a previous scram recovery (reference LER 86-25-01). In that evaluation, it was shown that thermal effects on MSL piping were acceptable if the temperature difference between the water flowing and the pipe temperature was less that 144°F. For this transient, the temperature differential was no greater than 81°F and considered acceptable.

- Hydrodynamic effects on the MSLs were not found to be a concern. The MSLs were initially filled with the MSIVs closed. The MSIVs were opened with pressure conditions similar on both sides. The MSL piping inside the drywell and outside to the Main Turbine was physically inspected and no sign of support components damage or abnormal displacement was found.

- The water passage through MSRV-4D was also evaluated for potential hydrodynamic affects. Had significant loads occurred during this event, damage to the MSRV discharge line supports would have occurred. A physical inspection of all MSRVs suspected of passing water and their associated piping, including visual examination of snubbers (included stroke testing of three snubbers), spring cans, and bolting showed no sign of damage. Inspection of the exterior of MSRV-4D, which should have seen the highest load, showed no sign of unusual stressing of bolts or joints, and no sign of abnormal displacement. Based on these examinations, the MSRV discharge line loads experienced during the event were considered within the design capability of the piping/support system. In addition, no valve leakage has been observed following plant restart.

  No damage is anticipated to have occurred internally to the MSRV-4D valve. However, disassembly of the valve is scheduled for the current refueling outage to further substantiate that no undetected damage has occured.

In summary, while this was an undesirable transient, effects of the transient were within plant design margins. This was confirmed by physical walkdown, with special attention paid to the SRVs and associated piping.

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3) In accordance with the requirements of Technical Specifications Sections 3.5.1(f) and 6.9.2 (ECCS Injections), the following special. data is provided:

- Total accumulated initiation cycles to date = 4
- Current usage factor value remains well below 0.70

B. Corrective Actions

1) PPM 3.3.1, "Reactor Scram", was revised (by means of a temporary deviation) to include instructions for placing the RFW Startup Flow Control Valve (RFW-LCV-10) in service. In addition, a visual aid for aligning the RFW System onto RFW-LCV-10 has been re-posted on the Control Room panel.

2) The load shedding of TSW on Level-2 (-50") will be evaluated to shed only upon a loss of off-site power. In addition an engineering evaluation will be performed to consider changing the Level-2 trips and isolation for the RCC pumps and valves to Level-1. These design changes would reduce the complexity of the Level-2 automatic trips.

3) An engineering evaluation will be performed to evaluate the potential benefit of causing an automatic trip of the Condensate Booster Pumps on Level-8.

4) An examination of the corrective actions during the recovery process indicate a need to provide specific training relative to the actions required to respond and recover from a Level-2 occurrence. The Plant Technical, Operations and Training Departments will collectively ensure that an effective training plan is formulated and conducted to improve management of the transient. An assessment will also be made of the need to provide recovery directions in the event of a flooded MSL.

5) The procedures for placing RFW-LCV-10 in service will be reviewed and remphasized during the next Operator Requalification Cycle.

6) The RFW-LCV-10 valve configuration has been redesigned and is scheduled for implementation during R-2. In consideration of this, the Technical and Operations Departments will investigate potential valve alignment configurations that can be used during normal operation to simplify valve alignment following a scram.

7) As a precautionary measure, MSRV-4D will be disassembled and inspected during the R-2 outage.

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Safety Significance

Reactor Protection System, ESF and electrical realignment actuations responded in accordance with Plant design. Safety-related equipment (and other important equipment) operated as designed and plant integrity was not affected as evidenced by physical walkdowns and data analysis. This event, therefore, posed no threat to the safety of plant personnel or the public.

Similar Events

None

EIIS Information

Text Reference	EIIS Reference System	EIIS Reference Component
Fuse	JB	Summer Card
Reactor Feedwater Control System (RFW)	JB	-----------
High Pressure Core Spray System (HPCS)	BG	-----------
Reactor Core Isolation Cooling System (RCIC)	BN	-----------
Plant Service Water System (TSW)	KG	-----------
Reactor Water Cleanup System (RWCU)	CE	-----------
Reactor Closed Cooling Water System (RCC)	CC	-----------
Main Steam System (MS)	SB	-----------

WASHINGTON PUBLIC POWER SUPPLY SYSTEM

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

Docket No. 50-397

April 21, 1987

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

Subject: NUCLEAR PLANT NO. 2
LICENSEE EVENT REPORT NO. 87-002

Dear Sir:

Transmitted herewith is Licensee Event Report No. 87-002 for WNP-2 Plant. This report is submitted in response to the report requirements of 10CFR50.73 and discusses the item 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:db

Enclosure:
Licensee Event Report No. 87-002

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

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