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AUTH.NAME	AUTHOR AFFILIATION
ARBUCKLE,J.D.	Washington Public Power Supply System
BEMIS,P.R.	Washington Public Power Supply System
RECIP.NAME	RECIPIENT AFFILIATION

SUBJECT: LER 97-004-00:on 970327,TS required manual scram of reactor due to unanticipated interaction of sys or components.Plant operators maneuvered plant to safe shutdown condition.W/ 970428 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 • Richland, Washington 99352-0968

April 28, 1997
GO2-97-081

Docket No. 50-397

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

Gentlemen:

Subject: **NUCLEAR PLANT WNP-2, OPERATING LICENSE NPF-21 LICENSEE EVENT REPORT NO. 97-004-00**

Transmitted herewith is Licensee Event Report No. 97-004-00 for WNP-2. This report is submitted pursuant to 10 CFR 50.73 and discusses the items of reportability, corrective action taken, and action to preclude recurrence.

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

Respectfully,

P.R. Bemis
Vice President, Nuclear Operations
Mail Drop PE23

Enclosure

cc: EW Merschoff - NRC RIV
KE Perkins, Jr. - NRC RIV, Walnut Creek Field Office
TG Colburn - NRR
NRC Senior Resident Inspector - 927N
DL Williams - BPA/399
PD Robinson - Winston & Strawn

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LICENSEE EVENT REPORT (LER)

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TITLE (4) TECHNICAL SPECIFICATION REQUIRED MANUAL SCRAM DUE TO ENTRY INTO REGION A OF THE POWER-TO-FLOW MAP

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)
03	27	97	97	- 0 0 4 -	0 0	04	28	97	N/A	0 5 0 0 0
										0 5 0 0 0

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

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR: (11)

__ 20.402(b)	__ 20.405c	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)	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	
J.D. Arbuckle, Licensing Technical Specialist	509	377-4601

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)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
X YES (if yes, complete EXPECTED SUBMISSION DATE)	NO		06	30	97

ABSTRACT (16)

On March 27, 1997 at 0907 hours with the plant in Mode 1, plant operators manually scrammed the reactor as required by the Technical Specifications due to entry into Region A of the power-to-flow map following the planned trip of a single main feedwater pump from approximately 97 percent power during Reactor Recirculation (RRC) and Reactor Feedwater (RFW) System testing. The post-modification testing was being performed to demonstrate that the feedwater level control system and recirculation flow runback feature would prevent a scram following the trip of a single feedwater pump during power operation.

During the test and after the planned trip of pump RFW-P-1B, an expected reactor recirculation pump speed runback to 27 Hz was observed. Following this planned runback, a second unexpected runback to 15 Hz occurred, which placed the reactor into Region A of the power-to-flow map. The second runback was caused by a reactor recirculation pump differential temperature cavitation interlock condition.

Plant operators responded immediately by manually scramming the reactor and maneuvering the plant to a safe shutdown condition in accordance with procedures.

The preliminary root cause of this event is unanticipated interaction of systems or components. A preliminary contributing cause is analysis deficiency. The differential temperature cavitation interlock was not expected. Corrective actions include the establishment of internal and external event evaluation teams to investigate the event.

This event posed no threat to the health and safety of the public or plant personnel. The plant was in the area of increased awareness for a short time. Stable, non-oscillatory core power conditions were observed at all times during the event.

Event Description

On March 27, 1997 at 0907 hours with the plant in Mode 1, plant operators manually scrammed the reactor as required by the Technical Specifications due to entry into Region A of the power-to-flow map following the planned trip of a single main feedwater pump from approximately 97 percent power during during Reactor Recirculation (RRC) System [AD] and Reactor Feedwater (RFW) System [JB] testing.

At the time of the event, post-modification testing of the digital feedwater level control and RRC adjustable speed drive systems was being conducted to demonstrate that the level control system and recirculation flow runback feature would prevent a scram following the trip of a single feedwater pump during power operation.

During the test and after the trip of pump RFW-P-1B [P], a reactor recirculation pump speed runback to 27 Hz was observed as expected. Following this runback, a second unexpected runback to 15 Hz occurred, which placed the reactor into Region A of the power-to-flow map. The second runback to 15 Hz was caused by a reactor recirculation pump differential temperature cavitation interlock condition existing for greater than 15 seconds.

As expected in response to the manual scram, reactor vessel water level decreased to just below +13 inches. In response to the low level condition, reactor feedwater flow increased and level was recovered to above +13 inches. During this evolution and at approximately +18 inches, pump RFW-P-1A [P] speed failed to decrease in response to the increasing level control signal. As a result, reactor vessel water level reached +54 inches (Reactor Vessel Water Level - High: Level 8) and the pump tripped on high level.

Plant operators successfully returned both reactor feedwater pumps to operation and restored level to within normal limits.

Immediate Corrective Action

Plant operators maneuvered the plant to a safe shutdown condition in accordance with procedures.

Further Evaluation

1. Pursuant to 10 CFR 50.73(a)(2)(i)(A) this event is reportable as, "The completion of any nuclear plant shutdown required by the Technical Specifications." Technical Specification 3.4.1, "Recirculation Loops Operating," requires that the reactor mode switch be placed in the shutdown position within 15 minutes of operation in Region A of the Power-to-Flow map.

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This event was originally determined to be reportable as a four-hour Reactor Protection System actuation in accordance with 10 CFR 50.72(b)(2)(ii). The initial notification to the NRC Operations Center was made on March 27, 1997 at 1013 hours (66 minutes after the event). Upon further review, it was determined that the event should have also been reported within one hour as an initiation of a plant shutdown required by the Technical Specifications in accordance with 10 CFR 50.72(b)(1)(i)(A).

On March 27, 1997 at 1308 hours, the NRC Operations Center was notified of the change in reportability criteria to reflect that the event was also reportable within one hour.

2. Plant design includes an interlock to reduce RRC pump drive flow during conditions which could cause cavitation in the RRC system. A low temperature differential between the RRC pump suction and the vessel steam dome indicates a reduction in subcooling of reactor coolant in the reactor vessel annulus region. A reduction in RRC drive flow under these conditions prevents system cavitation and avoids equipment damage from prolonged vibration.

   The reduction in drive flow initiates if the differential temperature is less than or equal to 9.9 degrees Fahrenheit for more than 15 seconds. If this occurs, the control logic automatically reduces drive speed to 15 Hz, and RRC pump flow reduces to 25 percent.

3. Evaluation of plant data validated that a differential temperature condition of less than 9.9 degrees Fahrenheit was present for longer than 15 seconds, and this condition caused the RRC runback to 15 Hz as designed.

4. The digital feedwater level control and adjustable speed drive systems performed as designed to the planned trip of the reactor feedwater pump up to the point of the second runback to 15 Hz. Following the manual scram, the digital feedwater level control system did not respond as designed to control vessel level below the Reactor Vessel Water Level - High (Level 8) setpoint of +54 inches.

<u>Root Cause</u>

The root cause analysis effort is still in progress. The preliminary root cause of this event is unanticipated interaction of systems or components. A preliminary contributing cause is analysis deficiency.

The differential temperature cavitation interlock was not expected. The digital feedwater level control and adjustable speed drive systems were previously modeled in the plant simulator and in simulation programs used by General Electric and the Supply System. There was no previous simulator indication that the cavitation interlock would be initiated due to a reactor feedwater pump trip from 100 percent reactor power.

TEXT (17)

Further Corrective Action

An event evaluation team, composed primarily of Supply System personnel, was established to investigate the event and present the results and conclusions to plant management. Specific areas that will be evaluated by the team include analytical results of testing performed, performance of the on-shift Operations personnel during this event, performance of the digital feedwater level control and adjustable speed drive systems, the adequacy of the test procedure as it relates to the conditions surrounding the scram, and the adequacy of the design related to the installation of the adjustable speed drive and digital feedwater level control systems.

A second independent evaluation team, made up of primarily non-Supply System personnel, will evaluate the event and perform a critical review of the investigation conducted by the Supply System event evaluation team. Specific items that will be evaluated by this team are identical to those of the other team.

At the completion of these evaluations, the results will be provided in a supplement to this report.

Assessment of Safety Consequences

The probability of thermal-hydraulic oscillations is greatly increased if the plant is operating in Region A of the power-to-flow map.

In response to this event, plant operators performed a manual reactor scram to exit Region A well within the 15 minute time-frame required by the Technical Specifications. Subsequent operator actions were prompt and correct to maneuver the plant to a safe shutdown condition.

During the short time the plant was operating in Region A of the power-to-flow map, stable, non-oscillatory core power conditions were observed. Therefore, this event posed no threat to the health and safety of the public or plant personnel.

Similar Events

Licensee Event Report 96-004 reported a problem involving a manual reactor scram due to a reactor water level transient during testing of the digital feedwater system. After performing a step of the test procedure which reduced reactor water level by six inches, level continued to decrease past the intended value.

The cause of this event was determined to be a manufacturer's programming error in the digital feedwater system testing software which caused mismatches to occur in the feed/steam signal, resulting in unwanted level step changes. The software was modified by the vendor to delete the possibility of unwanted step changes. The software change was verified and validated and functionally tested on the plant simulator prior to intallation in the plant.

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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Corrective action taken in response to the previous event would not have been expected to preclude this event. The action was designed to correct a specific software problem that was causing mismatches in the feed/steam signal.