NRC Form 366 (9-83)

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

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Palo Verde Unit 1	0 5 0 0 0 5 2 8	1 OF 0 4

TITLE (4)
Reactor Trip

EVENT DATE (5) MONTH	DAY	YEAR	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	REPORT DATE (7) MONTH	DAY	YEAR	OTHER FACILITIES INVOLVED (8) FACILITY NAMES	DOCKET NUMBER(S)
0 6	1 4	8 5	8 5	0 1 9	0 0	0 7	1 5	8 5		0 5 0 0 0
										0 5 0 0 0

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

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	
William F. Quinn, Manager - Nuclear Licensing (Extension 4087)	6 0 2	9 4 3 - 7 2 0 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
X	S J	F C U	L 1 7 0	N					
X	K A	F C U	F 1 3 0	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 6/14/85, Unit 1 was in Mode 1 at 19% power with the main generator on-line. The "B" main feedwater pump was running in manual, supplying feedwater to the steam generators and a reactor operator was in the process of starting the "A" Main Feedwater Pump per procedure 41OP-1FT01. The "A" Main Feedwater Pump Mini-Flow Control Valve did not properly control the flow, allowing excessive condensate flow back to the condenser. This, in conjunction with improper operation of Condensate Pump Mini-Flow Control Valves, caused a low suction pressure trip of the "B" Main Feedwater Pump at 1155. The reactor operators manually tripped the turbine-generator, started auxiliary feedwater, and began manually reducing reactor power by insertion of CEA's. At 1156 the reactor tripped on high pressurizer pressure. Following the reactor trip, the "C" Condensate Pump tripped on low recirculation flow because it's Mini-Flow Control Valve did not open sufficiently. A slight overcooling occurred after the trip due to excessive steam demand created by the flow through the Main Steam Line Drains and Auxiliary Steam Loads.

Cause of the trip was improper operation of the "A" Main Feedwater Pump Mini-Flow Control Valve.

The Main Feedwater Pump and Condensate Pump Mini-Flow Control Valves are being repaired to correct their operation. The Main Steam Line Drain Valve operation is being evaluated to determine if it is necessary that valves open immediately following a trip. In addition, flow rates are being evaluated for conformance to design.

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FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
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TEXT (If more space is required, use additional NRC Form 366A's) (17)

On 6/14/85 Unit 1 was in Mode 1 at 19% power with the Main Generator on-line. The "B" Feedwater Pump, "A" and "C" Condensate Pumps were in service. The FWCS (Feedwater Control System) was in automatic with the "B" Feedwater Pump Control in manual. The Reactor Operator was in the process of starting up the "A" Main Feedwater Pump per Procedure 41OP-1FT01.

There were no systems or components inoperable at the start of the event that contributed to the event.

At 1138 the "A" Main Feedwater Pump Turbine was reset in accordance with the procedure. Resetting the "A" Feedwater Pump Turbine and the subsequent opening of its Mini-Flow Control Valve to the condenser caused a small upset in feedwater flows and S/G (Steam Generator) levels due to excessive condensate flow through the "A" Feedwater Pump Mini-Flow Control. At 1140 the "A" Main Feedwater Pump Turbine was tripped per procedure. At 1154 the Reactor Operator again reset the "A" Main Feedwater Pump Turbine. The Reactor Operator manually increased the "A" Main Feedwater Pump speed to 100 RPM attempting to establish speed control and increase pump discharge pressure. The effect of this action was to increase flow through the improperly controlling Mini-Flow Valve to the condenser which reduced the condensate available to the "B" Main Feedwater Pump causing its discharge pressure to decrease. The Reactor Operator manually increased the speed of the "B" Main Feedwater Pump to increase its discharge pressure and maintain S/G levels. The combined effect of diverting condensate flow to the condenser through the "A" Main Feedwater Pump Mini-Flow and increasing the "B" Main Feedwater Pump speed caused the "B" pump to lose suction. Immediately after increasing the speed of the "B" Main Feedwater Pump, the Reactor Operator observed its discharge pressure and both S/G levels drop sharply. The "A" Main Feedwater Pump was not providing flow to the S/G's due to its low speed and it received a hydraulic control pressure trip 8 seconds after the "B" Main Feedwater Pump tripped. At this time there was no feedwater being supplied to the S/G's.

The Assistant Shift Supervisor, assuming the role of CRS, (Control Room Supervisor) directed one Reactor Operator to manually trip the turbine-generator and start the non-essential Auxiliary Feedwater Pump, AFN-P01, while the other Reactor Operator manually inserted CEA's to reduce reactor power in an attempt to reduce reactor power and steam flow sufficiently to avoid a reactor trip. At 1156 the reactor tripped on high pressurizer pressure.

After the "A" and "B" Main Feedwater Pumps tripped, condensate pump discharge pressure increased to shut off head pressure due to the Main Feedwater Pump Mini-Flow Control Valves being closed and S/G pressures being higher than the condensate pump discharge pressures. The "C" Condensate Pump Mini-Flow Control Valve did not open sufficiently to clear its low flow alarm condition. After a preset 20 second time delay the "C" Condensate Pump tripped at 1157.

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

The Reactor Protection System functioned according to design. Engineered Safety Feature Systems were not actuated. The Auxiliary Feedwater System was manually started by the Reactor Operator and used to provide feed to the S/G's. Plant response to the trip was normal except for a slight overcooling after the trip. Operator actions for this trip were appropriate and effective and no procedural deficiencies were identified that contributed to the event.

The overcooling experienced was primarily a result of heat removal caused by excessive steam flow through the Main Steam Line Drains and the Auxiliary Steam Supply crossconnect to Unit 2. Unit 2 auxiliary steam and small heat loads were isolated but were not considered to be excessive even though contributing to the total post-trip heat load.

The Reactor Trip First Out Annunicator did not function correctly, however its malfunction did not affect correct diagnosis of the trip cause or proper response to it.

When the Reactor Operator attempted to feed the #2 S/G through the Downcomer By-pass Valve, SG-HV-1145, the valve did not open. The "B" Essential Auxiliary Feed Pump was started to supply feed to the #2 S/G as needed to maintain normal levels. SG-HV-1145 was manually backed off its close-seat, the thermal overload reset and thereafter operated satisfactorily.

The cause of this trip was a failure of the "A" Main Feedwater Pump Mini-Flow Control Valve to properly control flow thereby causing loss of suction pressure available to the operating "B" Main Feedwater Pump which then tripped on low suction pressure. Loss of feedwater to the S/G's caused the reactor to trip on High Pressurizer Pressure due to inadequate heat removal from the Reactor Coolant System.

The Unit was operating at 19% power, conducting extensive equipment and system testing as part of the Power Ascension Testing Program. The 19% plateau is the first step in this program where sufficient steam and fluid flows exist to allow the Main Feedwater Pump and associated Mini-Flow Control Valves to be fully tested. This testing is continuing with the objective of meeting the Main Feedwater Pump minimum flow requirements while not causing pump trips due to short term fluctuations of Main Feedwater Flows or Main Feedwater Pump Mini-Flow Control Valve operation.

Extensive troubleshooting was done to determine the cause of the "C" Condensate Pump trip. While operating the pump on cleanup recirculation it was noted that its Mini-Flow Control Valve was not opening fast enough to prevent the pump from tripping due to low recirculation flow. Further analysis of the Post Trip Review data and the Condensate Pump Control Circuitry confirmed that the "C" Condensate Pump tripped due to low recirculation flow <u>after</u> the "B" Main Feedwater Pump trip and reactor trip. Work is continuing to assure correct operation of all Condensate Pump and Main Feedwater Pump Mini-Flow Control Valves.

The steam flow rates through the Main Steamline Drains are being evaluated to determine if they conform to design or if they are excessive and can be reduced. In addition, automatic operation of Main Steamline Drain Valves is being addressed.

Testing was performed on the Reactor Trip First Out Annunciation. This testing confirmed improper operation of this annunciation. Proper operation was restored prior to restart of the reactor. A work request has been written to assure proper operation of #2 S/G Downcomer Bypass Valve, SG-HV-1145, which tripped on thermal overload while attempting to feed the #2 S/G following the reactor trip.

Testing was performed to assure proper operation of the RPS due to conflicting reports of various trips and pre-trips annunciated by operators and engineers. RPS High Pressurizer Pressure Pre-trip and trip setpoints were verified to be correct; and proper operation of the RPS from the cabinet to and including the Rx Trip Switchgear was confirmed. In addition, CPC sensor out-of-range setpoints were verified to be correct.

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

ANPP-33032-EEVB/GEC
July 15, 1985

Subject: Palo Verde Nuclear Generating Station (PVNGS)
Unit 1
Docket No. STN 50-528, License No. NPF-41
Licensee Event Report - Reactor Trip
File: 85-056-026; G.1.01.10

Dear Sirs:

Attached please find Licensee Event Report (LER) No. 85-019-00 prepared and submitted pursuant to 10 CFR 50.73. This LER addresses a reactor trip resulting from improper operation of the "A" Main Feedwater Pump Mini-Flow Control Valve. In accordance with 10 CFR 50.73(d), we are herewith forwarding a copy of the LER to the Regional Administrator of the Region V Office.

If you have any questions or concerns, please contact me.

Very truly yours,

E E Van Brunt /

E. E. Van Brunt, Jr.
Executive Vice President
Project Director

EEVB/GEC/slh
Attachments

cc: J. B. Martin (all w/a)
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A. L. Hon
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A. C. Gehr
INPO Records Center

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