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)
Palo Verde Unit 1	0 5 0 0 0 5 2 8	1 OF 0 4

TITLE (4)
Reactor 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 6	1 4	8 5	8 5	— 0 1 9	— 0 1	0 3	1 4	8 6		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9)	1	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)			
POWER LEVEL (10)	0 1 9	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)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE)	X NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines) (16)

This is a supplement to LER 85-019-00.

On June 14, 1985, Unit 1 was in Mode 1 at 19% power with the "B" main feedwater pump (SU) running in manual, and in the process of starting the "A" main feedwater pump. Upon resetting the "A" main feedwater pump turbine controls (SL), the mini-flow control valve opened and the pump began windmilling, passing water to the condenser. In order to gain control, pump speed was increased to 100 RPM. Also, main feedwater pump "B" speed was increased to make up for the loss of discharge pressure. This combination of events caused a low suction trip of the "B" main feedwater pump and eventually the "A" main feedwater pump. The reactor operators manually reduced reactor power by insertion of CEA's. At 1156, the reactor tripped on high pressurizer pressure.

The trip occurred because the "A" main feedwater pump was not brought up to proper operating speed soon enough to overcome the suction pressure loss caused by a windmilling pump.

To prevent recurrence, the feedwater control system was adjusted and fine tuned which increased response time dramatically.

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FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Palo Verde Unit 1	0 5 0 0 0 5 2 8	8 5	— 0 1 9	— 0 1	0 2 OF 0 4

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

This is a supplement to LER 85-019-00.

On June 14, 1985, Unit 1 was in Mode 1 at 19% power with the Main Generator on-line. The "B" main feedwater pump and the "A" and "C" condensate pumps were in service. The feedwater control system (SU) was in automatic with the "B" main feedwater pump control in manual. The Reactor Operator was in the process of starting up the "A" main feedwater pump per Procedure 410P-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 (SL) was reset in accordance with the procedure. Resetting of the "A" main feedwater pump turbine and the subsequent opening of its mini-flow control valve to the condenser caused flow to go to the condenser. 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 in an attempt to establish speed control and to increase pump discharge pressure. The effect of this action was to increase flow through the mini-flow control valve to the condenser which reduced the condensate available to the "B" main feedwater pump and caused its discharge pressure to decrease. The Reactor Operator manually increased the speed of the "B" main feedwater pump to increase its discharge pressure and to maintain Steam Generator (S/G) levels. The combined effect of diverting condensate flow to the condenser through the "A" main feedwater pump mini-flow control valve and increasing the "B" main feedwater pump speed caused the "B" main feedwater 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 approximately 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 Control Room Supervisor, directed one Reactor Operator to manually trip the turbine-generator and start the non-essential auxiliary feedwater pump while the other Reactor Operator manually inserted CEA's 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 pressure 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 pre-set 20 second time delay, the "C" condensate pump tripped at 1157.

The Reactor Protection System (RPS) (JC) functioned according to design. No Engineered Safety Feature Systems (JE) were actuated. The auxiliary feedwater system was manually started by the Reactor Operator and was used to provide feedwater to the S/G's. Plant response to the trip was normal with the exception of a slight overcooling of the primary system after the trip. Operator actions during this event were appropriate and effective. No procedural deficiencies were identified that contributed to the event.

The primary system overcooling experienced during this event was primarily a result of heat removal caused by excessive steam flow through the main steam line drains and the auxiliary steam supply cross connect to Unit 2. Unit 2 auxiliary steam and small heat loads were isolated and were not considered to be excessive even though they contributed to the total post-trip heat load.

The reactor trip first out annunciator did not function correctly. However, the malfunction of the annunciator did not affect correct diagnosis of the cause of the trip or the proper response to the trip.

When the Reactor Operator attempted to feed the No. 2 S/G through the downcomer bypass valve, the valve did not open. The "B" essential auxiliary feedwater pump was started to supply feedwater to the No. 2 S/G to maintain normal S/G levels. The downcomer bypass valve was manually backed off of its close-seat, the thermal overload was reset, and the valve operated satisfactorily.

The cause of this trip was the "A" main feedwater pump mini-flow control valve passing flow to the condenser thereby causing a 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 reactor was operating at 19% power, conducting extensive equipment and system testing as part of the Power Ascension Testing Program. The 19% power 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. The objective of this testing is to meet 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 conducted to determine the cause of the "C" condensate pump trip. While operating the pump on cleanup recirculation, it was observed 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 after the "B" main feedwater pump trip and reactor trip. The failure of this valve was caused by a loose air hose within the valve controller. The hose was replaced and the valve has operated properly.

Automatic operation of main steam line drain valves is being addressed. A temporary modification has been installed and a permanent plant change is being processed to eliminate automatic operation of the main steam line drain valves. The permanent plant change is forecast to be completed by April, 1986.

Testing was performed on the reactor trip first out annunciator. This testing confirmed the improper operation of this annunciation during the event. Proper operation of this system was restored prior to restart of the reactor. A work request has been written to assure proper operation of the No. 2 S/G downcomer bypass valve which tripped on thermal overload while attempting to provide feedwater to the No. 2 S/G following the reactor trip.

Testing was performed to assure proper operation of the RPS due to conflicting reports from operators and engineers of various trips and pre-trips annunciated. The RPS high pressurizer pressure pre-trip and trip setpoints were verified to be correct. The proper operation of the RPS from the cabinet up to and including the reactor trip switchgear was confirmed. In addition, Core Protection Calculator sensor out-of-range setpoints were verified to be correct.

All equipment operated as expected during this event. Therefore, there were not any safety consequences and the health and safety of the public was not jeopardized.

To prevent recurrence, the feedwater control system was adjusted and fine tuned which increased response time dramatically. No similar events had occurred previously.

Arizona Nuclear Power Project
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March 14, 1986
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U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

Subject: Palo Verde Nuclear Generating Station (PVNGS)
Unit 1
Docket No. STN 50-528 (License NPF-41)
Licensee Event Report - 85-019-01
File: 86-020-404

Dear Sirs:

Attached please find Supplement Number 01 to Licensee Event Report (LER) No. 85-019-00 prepared and submitted pursuant to 10 CFR 50.73. In accordance with 10 CFR 50.73(d), we are herewith forwarding a copy of this report to the Regional Administrator of the Region V Office.

If you have any questions, please contact me.

Very truly yours,

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

EEVB/BJA/rw
Attachment

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

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