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 Initiated by Load Rejection Test From 80% Power

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)
1 0	2 4	8 5	8 5	0 7 1	0 0	1 1	2 5	8 5		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 0 8 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)	X 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)	Special Report
20.405(a)(1)(iv)	50.73(a)(2)(ii)	50.73(a)(2)(viii)(B)	1-SR-85-026
20.405(a)(1)(v)	50.73(a)(2)(iii)	50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER
William F. Quinn, Manager of Nuclear Licensing (ext. 4087)	AREA CODE 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

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)

At 1010 MST on October 24, 1985, Palo Verde Nuclear Generating Station-Unit 1 was operating in Mode 1, with a generator output of 1019 MWe (81% power). As a part of scheduled activities supporting the 80% Load Rejection startup test, a turbine trip was manually initiated.

An unanticipated reactor trip immediately followed the turbine trip due to a sensed, but errant, low steam generator level. The reactor trip was accompanied by a cooldown and subsequent depressurization, which resulted in the receipt of actuation signals for Safety Injection (BQ), Containment Isolation (BD), and Main Steam Isolation (SB). The cooldown was caused by the Steam Bypass Control System (JI)(SBCS) not operating as designed.

The low steam generator level signal concern has been corrected by increasing the time delay on the trip initiating signal to the Plant Protection System (JC) to allow system recovery from the rapid pressure spikes incurred during a large load rejection. Operation of the SBCS has been corrected by effecting hardware and setpoint modifications on the SBCS circuitry.

Special Report 1-SR-85-026 is contained within the text of this LER. This Special Report is submitted as required by Technical Specifications whenever Emergency Core Cooling water is injected into the Reactor Coolant System.

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LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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

At 1036 MST on October 24, 1985, the NRC Operations Center was notified via the Emergency Notification System, of the declaration of a NOTIFICATION OF UNUSUAL EVENT for Unit 1 of the Palo Verde Nuclear Generating Station. The NOTIFICATION OF UNUSUAL EVENT was declared pursuant to Emergency Plan Implementing Procedure - 02, which requires the reporting of a reactor trip which is complicated by concurrent or subsequent events or conditions.

Prior to the events which initiated the reactor trip, the unit was in Mode 1, generator output was 1019 MWe, all control element assemblies (AA) were fully withdrawn with the control element drive mechanism system in the auto sequential mode, and house loads were being powered from the startup transformers (EA). At 1010, generator differential relay (EL) MAN-187-10 was manually actuated to initiate a turbine trip as a part of scheduled activities within the power ascension test for 80% Load Rejection. An unanticipated reactor trip immediately followed the turbine trip due to a low steam generator level sensed in number 2 steam generator.

The cause of the low steam generator level indication has been determined to be a high pressure pulse which was initiated by the rapid closure of the Turbine Throttle Valves (TA). It has been verified that during the transient, actual steam generator water levels were at no time actually below the trip setpoint. Reactor trip signals were received on all four Plant Protection System (JC) level channels for each steam generator.

Concurrent with the Turbine Throttle Valve closure, the Steam Bypass Control System (JI)(SBCS) generated a quick-open demand signal, as required by design, due to a rapidly decreasing steam flow rate. All 8 Steam Bypass Valves quick-opened as a result of this signal. The valves received a second, unexpected, quick-open demand signal after they began to modulate closed. Again, this was due to a rapidly decreasing steam flow rate, but was not expected based on system design. The second signal resulted in all 8 valves returning to the fully open position. The second quick-open of the steam bypass valves caused a Reactor Coolant System (AB)(RCS) cooldown and depressurization, and a resultant actuation of Safety Injection (BQ)(SIAS), Containment Isolation (BD)-(CIAS), and Main Steam Isolation (SB)(MSIS) approximately 30 seconds following the reactor trip.

The plant was stabilized, and the Safety Injection Actuation Signal and Containment Isolation Actuation Signal were reset approximately 36 minutes after initiation of the reactor trip. The Main Steam Isolation Actuation Signal was reset approximately 1 hour after initiation of the reactor trip.

At 1020 the shift supervisor had declared a Notification of Unusual Event and subsequently made the appropriate notifications. The Notification of Unusual Event was declassified and terminated at 1111.

During the event, heat removal was accomplished by the use of one Atmospheric Dump Valve (SB)(ADV) per Steam Generator. Two of the four Reactor Coolant Pumps were manually tripped, as directed by procedure, following the SIAS. Letdown (WI) capability was lost temporarily due to the CIAS. It was restored approximately 20 minutes after the reactor trip. The motor driven essential auxilliary feedwater pump (BA) was used to maintain Steam Generator level.

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

The high pressure pulse, which occurs when the Turbine Throttle Valves are closed, caused an undesireable Plant Protection System (PPS) response to the sensed low steam generator level indications. Investigations have determined that the duration of the Turbine Throttle valve closure induced high pressure pulse was approximately 300 msec.

A modification has been completed which increases the present delay time of 100 msec. for the steam generator low level trip bistable, to a value which should prevent pressure spikes from initiating errant reactor trips. The modification maintains the total response time of the steam generator low level reactor protective instrumentation within the response time specified by Technical Specifications.

Investigation into the cause of the second unanticipated quick-open of the Steam Bypass Valves has traced the problem to a combination of setpoint error in the SBCS and jumper installation that was not in accordance with plant design.

Because of similar card design and the integrated effect of these systems on plant operations, a review of setpoints and jumpers has been conducted on the instrument cards for the following non-safety related systems: Feedwater Control System (JB), Steam Bypass Control System, and the Reactor Regulating System (JD). All setpoints and jumpers which were found to be not in accordance with design have been corrected. Setpoint and jumper verifications are not required for any of the similar cards which are installed in safety related applications. Setpoint and jumper installation on the similar safety related instrument cards are functionally tested periodically during routine surveillance testing.

Correction of the incorrect setpoints and jumpers will prevent the second quick-opening of the steam bypass valves and the resultant RCS depressurization and initiation SIAS, CIAS, and MSIS. An analysis into the effect of the setpoint and jumper changes on previous test results has determined that no retesting is required.

During the recovery, when it was attempted to secure Low Pressure Safety Injection Pump "A" with a SIAS signal present, the pump could not be secured by going to the "override" mode. A sealing relay which locks in the override contacts was found to be defective and has been replaced.

It was also observed that the Main Steam Safety Valves (MSSV)(SB) indicated intermittently in the open position. The intermittent open indications have been traced to flow noise from the open ADV's which are located in close proximity to the MSSV's. The sensitivity and response of the acoustic position indicators for the MSSV's will be evaluated during subsequent testing.

All safety systems performed as designed for the duration of the event including the actuation of the Reactor Protection System, the Safety Injection Signal, the Containment Isolation Signal, and the Main Steam Isolation Signal, with the exception of the Low Pressure Safety Injection sealing relay, as noted previously.

The transient did not result in any challenges to fission product barriers or result in the release of radioactive materials.

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This event resulted in the injection of Emergency Core Cooling System (ECCS) water into the RCS. Technical Specification 3.5.3 requires that a Special Report be prepared pursuant to Technical Specification 6.9.2, and submitted within 90 days whenever an ECCS injection to the RCS occurs. This report satisfies the reporting requirement for this Special Report.

The value of the usage factor for each safety injection nozzle did not reach or exceed a value of 0.70 as a result of this event.

Arizona Nuclear Power Project
P.O. BOX 52034 • PHOENIX, ARIZONA 85072-2034

November 25, 1985
ANPP-34107-EEVB/GEC

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 No. NPF-41
Licensee Event Report - Reactor Trip Initiated
by Load Rejection Test From 80 Percent Power
File: 85-020-404

Dear Sirs:

Attached please find Licensee Event Report (LER) No. 85-071-00 prepared and submitted pursuant to 10 CFR 50.73. This LER addresses a reactor trip initiated by a load rejection test from 80 percent power. 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, please contact me.

Very truly yours,

EE Van Brunt Jr/KH

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

EEVB/GEC/dk
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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