LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Palo Verde Unit 1	05000528	1 OF 05

TITLE (4): Reactor Trip Following the Degradation of Main Feedwater Flow

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 NUMBERS
05	30	95	95	- 008 -	00	06	27	95	N/A	05000
									N/A	05000

OPERATING MODE (9): 1

POWER LEVEL (10): 065

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)
Burton A. Grabo, Section Leader, Nuclear Regulatory Affairs	602 393-6492

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)

On May 30, 1995, at approximately 2242 MST, Palo Verde Unit 1 was in Mode 1 (POWER OPERATION) operating at approximately 65 percent power when a reactor trip occurred when Steam Generator Number 2 (SG-2) water level reached the Reactor Protection System trip setpoint for low steam generator water level following the degradation of main feedwater flow. The Feedwater Control System effectively functioned to maintain levels until an Auxiliary Feedwater pump was placed in service to restore levels. Pre-trips for the Auxiliary Feedwater Actuation System (AFAS) were received on SG-2, but level was recovered prior to the setpoint for actuation of the AFAS. The Steam Bypass Control System responded as designed to control secondary system pressure. Required plant equipment and safety systems responded to the event as designed. No other safety actuations occurred and none were required. The Shift Supervisor diagnosed the event as an uncomplicated reactor trip. By approximately 0000 MST on May 31, 1995, the plant was stabilized in Mode 3 (HOT STANDBY).

The reactor trip was initiated by the closure of Feedwater isolation valve (FWIV) 1JSGBUV0137. The valve closed as a result of a loose wire for the A solenoid for the FWIV. The wire bundle was moved during the replacement of the coil for the D solenoid. As corrective action all wire connections on the terminal boards on the Unit 1 FWIVs and Main Steam Isolation Valves were verified to be secure. The initial evaluation of personnel performance has not identified any Human Performance issues during this event.

Previous similar events were reported pursuant to 10 CFR 50.73 in LERs 530/94-005 and 529/92-001.

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 5

TITLE (4)
Reactor Trip Following the Degradation of Main Feedwater Flow

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 NUMBERS
									N/A	0 5 0 0 0
0 5	3 0	9 5	9 5	- 0 0 8 -	0 0	0 6	? ?	9 5	N/A	0 5 0 0 0

OPERATING MODE (9)	1
POWER LEVEL (10)	0 6 5

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)

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LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (AREA CODE)
Burton A. Grabo, Section Leader, Nuclear Regulatory Affairs	6 0 2 3 9 3 - 6 4 9 2

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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
[ ] YES (If yes, complete EXPECTED SUBMISSION DATE)	[X] NO				

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

On May 30, 1995, at approximately 2242 MST, Palo Verde Unit 1 was in Mode 1 (POWER OPERATION) operating at approximately 65 percent power when a reactor trip occurred when Steam Generator Number 2 (SG-2) water level reached the Reactor Protection System trip setpoint for low steam generator water level following the degradation of main feedwater flow. The Feedwater Control System effectively functioned to maintain levels until an Auxiliary Feedwater pump was placed in service to restore levels. Pre-trips for the Auxiliary Feedwater Actuation System (AFAS) were received on SG-2, but level was recovered prior to the setpoint for actuation of the AFAS. The Steam Bypass Control System responded as designed to control secondary system pressure. Required plant equipment and safety systems responded to the event as designed. No other safety actuations occurred and none were required. The Shift Supervisor diagnosed the event as an uncomplicated reactor trip. By approximately 0000 MST on May 31, 1995, the plant was stabilized in Mode 3 (HOT STANDBY).

The reactor trip was initiated by the closure of Feedwater isolation valve (FWIV) 1JSGBUV0137. The valve closed as a result of a loose wire for the A solenoid for the FWIV. The wire bundle was moved during the replacement of the coil for the D solenoid. As corrective action all wire connections on the terminal boards on the Unit 1 FWIVs and Main Steam Isolation Valves were verified to be secure. The initial evaluation of personnel performance has not identified any Human Performance issues during this event.

Previous similar events were reported pursuant to 10 CFR 50.73 in LERs 530/94-005 and 529/92-001.

TEXT

1. REPORTING REQUIREMENT:

This LER 528/95-008 is being written to report an event that resulted in an automatic actuation of the Reactor Protection System (RPS) as specified in 10 CFR 50.73(a)(2)(vii).

Specifically, on May 30, 1995, at approximately 2242 MST, a reactor (AC) trip occurred when Steam Generator Number 2 (SG-2) (AB) water level reached the RPS trip setpoint for low SG water level following the degradation of main feedwater (MFW) (SJ) flow. The Feedwater Control System (FWCS) effectively functioned to maintain SG levels until an Auxiliary Feedwater (AF)(BA) pump was placed in service to restore levels. Pre-trips for the Auxiliary Feedwater Actuation System (AFAS)(JE)(BA) were received on SG-2, but level was recovered prior to the setpoint for actuation of the AFAS. The Steam Bypass Control System (SBCS)(JI) responded as designed to control secondary system pressure. Required plant equipment and safety systems responded to the event as designed. No other safety system actuations occurred and none were required.

2. EVENT DESCRIPTION:

On May 30, 1995, at approximately 2200 MST, Unit 1 was in Mode 1 (POWER OPERATION) at approximately 63 percent power when Electrical Maintenance personnel (utility, nonlicensed) were in the process of replacing the coil on the D solenoid as part of corrective maintenance for Feedwater isolation valve (FWIV) 1JSGUV0137.

At approximately 2242 MST on May 30, 1995, Unit 1 was in Mode 1 at approximately 65 percent power when a reactor (AC) trip occurred when SG-2 water level reached the RPS trip setpoint for low SG water level following the degradation of MFW flow. The FWCS effectively functioned to maintain SG levels until an AF pump was placed in service to restore levels. Pre-trips for the AFAS were received on SG-2, but level was recovered prior to the setpoint for actuation of the AFAS. The SBCS responded as designed to control secondary system pressure. Required plant equipment and safety systems responded to the event as designed. No other safety system actuations occurred and none were required.

Due to the low decay heat load (initial start-up after completing a refueling outage, 1R5), the prescribed feed rate required by the Reactor Trip emergency procedure (41EP-1RO01) was initiating a cooldown.

TEXT

The Shift Supervisor (SS), Control Room Supervisor (CRS) (utility, licensed), and the Shift Technical Advisor (STA) (utility, nonlicensed) discussed the cooldown and decided to deviate from the Reactor Trip procedure concerning feeding both SGs. The secondary Reactor Operator (utility, licensed) was directed to feed only one SG at a time to the required level of 72 percent as read on the SG wide range level instrument. Deviations from emergency procedures are allowed by plant procedures, and this decision probably averted an excessive cooldown that could have caused other Engineered Safety Features (ESF)(JE) to actuate (i.e., Safety Injection Actuation Signal (SIAS) and Containment Isolation Actuation Signal (CIAS)).

The SS diagnosed the event as an uncomplicated reactor trip. By approximately 0000 MST on May 31, 1995, the plant was stabilized in Mode 3 (HOT STANDBY).

The initial investigation has determined that the cause of the FWIV closure was attributed to a loose wire for the A solenoid which resulted in the deenergization of solenoid A and the subsequent closing of FWIV 1JSGBUV0137.

The functional requirement of 1JSGBUV0137 is to provide containment isolation (NH) in Modes 1 through 4 (POWER OPERATION through HOT SHUTDOWN). During power operation, the valve is normally opened and closes upon a Containment Isolation Actuation Signal (CIAS) (BD) to provide containment integrity.

3. ASSESSMENT OF THE SAFETY CONSEQUENCES AND IMPLICATION OF THIS EVENT:

Prior to the reactor trip signal (2242 MST) from low SG-2 level, the reactor coolant system (RCS)(AB) pressure increased to 2270 pounds per square inch absolute (psia). The peak pressure criteria of 110 percent of design (2750 psia) was never challenged during this RCS pressure transient. The steam generator peak pressure of 1170 psia was reached at approximately 2245 MST, and this was also below the 110 percent of design pressure (1397 psia). These low pressure peaks can be attributed to the lower initial operating power 65 percent and low decay heat and the actuation of the SBCS. Four SBCS valves quick opened to relieve the secondary pressure. The transient did not cause any violation of the Specified Acceptable Fuel Design Limits (SAFDLs).

TEXT

This Unit 1 reactor trip can be classified as a Loss of Normal Feedwater which is an infrequent event of the ``decreasing heat removal by the secondary system'' category and is bounded by the limiting event for this category which is the Loss of Condenser Vacuum (LOCV). Finally, equipment and systems, assumed in Safety Analysis, were functional and plant response was normal for the situation that occurred. Scenarios defined in Updated Final Safety Analysis Report (UFSAR) Chapter 15 and design assumptions of the reactor protection system are bounding for this Unit 1 reactor trip. Scenarios defined in UFSAR Chapter 6, concerning Loss of Coolant Accidents (LOCA), were not challenged during this transient.

This event did not result in any challenges to the fission product barriers or result in any releases of radioactive materials. Therefore, there were no adverse safety consequences or implications as a result of this event. This event did not adversely affect the safe operation of the plant or the health and safety of the public.

4. CAUSE OF THE EVENT:

An incident investigation for the Unit 1 reactor trip is being performed in accordance with the APS Corrective Action Program. The investigation to date has concluded that there were no personnel errors associated with the work performed on the D solenoid for FWIV 1JSGBUV0137. The investigation has not determined to date the reason for the wire on the A solenoid for the FWIV being loose. The preliminary investigation determined that this was an isolated event most likely caused by poor work practices from the last time that solenoid A was worked on (SALP Cause Code A: Personnel Error). A review of the work order history for solenoid A showed that work was last performed on this solenoid on October 23, 1993.

No unusual characteristics of the work location (e.g., noise, heat, or poor lighting) directly contributed to this event. There were no apparent procedural errors which contributed to this event.

If the evaluation results differ from this determination or if information is developed which would affect the readers understanding or perception of this event, a supplement to this report will be submitted to describe the final root cause of failure.

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

FACILITY NAME	DOCKET NUMBER	LER NUMBER: YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE
PALO VERDE UNIT 1	0 5 0 0 0 5 2 8	9 5	- 0 0 8 -	0 0	0 5 OF 0 5

TEXT

5. STRUCTURES, SYSTEMS, OR COMPONENTS INFORMATION:

The closure of FWIV 1JSGBUV0137 was attributed to a loose wire on the A solenoid. The preliminary investigation determined that the loose wire was attributed to poor work practices, not a component failure.

There are no indications that any structures, systems, or components were inoperable at the start of the event which contributed to this event. No failures of components with multiple functions were involved. There were no safety system actuations and none were required.

6. CORRECTIVE ACTIONS TO PREVENT RECURRENCE:

The immediate corrective action, for the condition, was to correctly land the wire on the terminal block for the A solenoid and verify that all wire connections on the terminal board were secure. The terminal board for FWIV 1JSGBUV0137 has compression type connections. Terminal boards on the remaining FWIVs and MSIVs in Unit 1 use a ring lug type connection but were inspected to ensure that all wire connections were secure (no discrepancies were identified). An equipment review for Unit 2 did not identify any compression type connections for the Unit 2 FWIVs or MSIVs. Compression type connections have been identified in 7 of the Unit 3 FWIVs and MSIVs. Work requests have been written to verify the Unit 3 connections. No previous cases of loose connections on this type connection have been identified.

To date the Incident Investigation has not identified any Human Performance issues during this event. However, any lessons learned from this event identified by the investigation will be incorporated into the Maintenance and Operations training programs.

7. PREVIOUS SIMILAR EVENTS:

Reactor trips attributed to a Feedwater Control System (FWCS) malfunction have been previously reported in LERs 530/94-005 and 529/92-001. In the previous events, the cause of the specific FWCS component failure was identified and appropriate corrective actions taken. Based on the information available at this time, the cause and specific scenario of the event reported by this LER does not appear to be related to the previous FWCS malfunctions.