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SUBJECT: LER 89-009-00:on 890712,reactor trip due to partial loss of forced flow.

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TITLE: 50.73/50.9 Licensee Event Report (LER), Incident Rpt, etc.

NOTES:Standardized plant. 05000529

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	NRR/DOEA/EAB 11	1	1	~~NRR/DREP/RPB~~ 10	2	2
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192-00507-JGH/TDS/RKR
August 14, 1989

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

Dear Sirs:

Subject: Palo Verde Nuclear Generating Station (PVNGS)
Unit 2
Docket No. STN 50-529 (License No. NPF-51)
Licensee Event Report 89-009-00
File: 89-020-404

Attached please find Licensee Event Report (LER) No. 89-009-00 prepared and submitted pursuant to 10CFR 50.73. In accordance with 10CFR 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 T. D. Shriver, Compliance Manager at (602) 393-2521.

Very truly yours,

JG Haynes

J. G. Haynes
Vice President
Nuclear Production

JGH/TDS/RKR/kj

Attachment

cc: W. F. Conway (all w/a)
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E. E. Van Brunt, Jr.
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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 2	0 5 0 0 0 5 2 9	1 OF 0 9

TITLE (4)
Reactor Trip Due to Partial Loss of Forced 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 NUMBER(S)
0 7	1 2	8 9	8 9	0 0 9	0 0	0 8	1 4	8 9	N/A	0 5 0 0 0
									N/A	0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 1 0 0

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)	
[ ] 20.405(a)(1)(iv)	[ ] 50.73(a)(2)(ii)	[ ] 50.73(a)(2)(viii)(B)	Special Report
[ ] 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)
Timothy D. Shriver, Compliance Manager	6 0 2 3 9 3 - 2 5 2 1

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
X	E B	F U	G 0 8 0	N					
X	S J	V	P 0 3 2	Y					

SUPPLEMENTAL REPORT EXPECTED (14)

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

EXPECTED SUBMISSION DATE (15): MONTH 1 2 DAY 0 1 YEAR 8 9

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

On July 12, 1989 at approximately 2212 MST Palo Verde Unit 2 was operating at approximately 100 percent power when 2 of the 4 reactor coolant pumps were load shed from their power supply (Bus 2E-NAN-S02), resulting in a reactor trip on calculated low DNBR due to low reactor coolant flow. Immediately following the trip, a Safety Injection Actuation Signal (SIAS) and Containment Isolation Actuation Signal (CIAS) Engineered Safety Features occurred on low Reactor Coolant System (RCS) pressure. Following the event, at approximately 1529 MST on July 13, 1989, a portion of the main feedwater system (MFWS) was overpressurized.

The cause of the load shed was a failed fuse in the bus potential transformer. The cause of the SIAS/CIAS was RCS depressurization due to improper Steam Bypass Control System (SBCS) response and leaking pressurizer spray valves. The cause of the MFWS overpressurization was a failed check valve.

Immediate corrective action taken was to replace the fuse. An independent investigation is being conducted to determine the causes of the incidents which occurred during this event.

This submittal also provides a Special Report in accordance with Technical Specification 3.5.2 ACTION b.

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

I. DESCRIPTION OF WHAT OCCURRED:

A. Initial Conditions:

At approximately 2212 MST on July 12, 1989, Palo Verde Unit 2 was in Mode 1 (POWER OPERATION) at approximately 100 percent power.

B. Reportable Event Description (Including Dates and Approximate Times of Major Occurrences):

Event Classification: Any event or condition that resulted in manual or automatic actuation of any Engineered Safety Feature (ESF)(JE), including the Reactor Protection System (RPS)(JC).

At approximately 2212 MST on July 12, 1989, a reactor (RCT)(AC) trip on low Departure from Nucleate Boiling Ratio (DNBR) occurred due to a partial loss of reactor coolant (AB) flow. The partial loss of flow occurred when two (2) of the four (4) Reactor Coolant Pumps (RCP)(AB) were load shed from their power supply (Bus 2E-NAN-S02)(BU)(EA). Following the reactor trip, at approximately 2213 MST a Safety Injection Actuation Signal (SIAS)(JE) and a Containment Isolation Actuation Signal (CIAS)(JE) occurred when the Reactor Coolant System (RCS)(AB) pressure decreased to approximately 1823 psia (approximately 14 psi below the actuation setpoint). All safety system components actuated as designed. The plant was stabilized at approximately 2322 MST and the event was terminated.

Prior to the event, at approximately 2001 MST on July 12, 1989, the Control Room received a trouble alarm annunciator (ALM)(ANN) for Bus 2E-NAN-S02. Operations personnel (utility, non-licensed) investigated and inspected Bus 2E-NAN-S02. They could not determine the reason for the alarm on Bus 2E-NAN-S02. At approximately 2020 MST a control room annunciator alarm indicated that the unit oscillograph (OSG) had operated. Operations personnel (utility, licensed) responded to the annunciator alarm. The oscillograph indicated that there had been an undervoltage condition on Bus 2E-NAN-S02. Also, the digital fault recorder (XR) printout indicated a disturbance on Bus 2E-NAN-S02. Since no apparent problems were identified with Bus 2E-NAN-S02, unit operations personnel (utility, licensed) continued to investigate the problem. From approximately 2206 MST to approximately 2212 MST the Oscillograph alarmed three more times. Operations personnel were responding to the alarms when at approximately 2212 MST, a load shed actuation occurred on Bus 2E-NAN-S02 even though Bus 2E-NAN-S02 remained energized. The load shed caused the loads on Bus 2E-NAN-S02 to be deenergized.

NRC Form 366A (9-83)

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO 3150-0104
EXPIRES: 8/31/88

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

The RCPs are powered from non-class 1E 13.8 kv Busses 2E-NAN-S01 (BU)(EA) and 2E-NAN-S02. RCPs 1A and 2A are powered from Bus 2E-NAN-S01 and RCPs 1B and 2B are powered from Bus 2E-NAN-S02. Bus 2E-NAN-S01 was being supplied by a Startup Transformer (XFMR)(EA) and Bus 2E-NAN-S02 was being supplied by the Unit 2 Auxiliary Transformer (XFMR)(EA). Since RCP's 1B and 2B were being supplied from Bus 2E-NAN-S02, the RCP's were deenergized and a partial loss of flow occurred resulting in a reactor trip on low DNBR.

Approximately one minute after the reactor trip, Reactor Coolant System pressure dropped lower than normal due to improper Steam Bypass Control System (SBCS)(SG) response and leaking pressurizer spray valves (PZR)(AB)(V). This resulted in concurrent safety injection and containment isolation Engineered Safety Features (ESF) actuations (JE) when the RCS pressure decreased to approximately 1823 psia, which is 14 psi below the low actuation pressure setpoint of 1837 psia and 1 psi above the minimum allowable trip setpoint value of 1822 psia. Immediately following the safety injection, pressurizer level and pressure stabilized. Pressure then began to trend toward steady state Mode 3 (HOT STANDBY) conditions (approximately 2250 psia).

At approximately 2223 MST on July 12, 1989, a Notification of Unusual Event (NUE) was declared. The NUE was declared pursuant to EPIP-02, "Emergency Classification" as a result of the SIAS on low pressurizer pressure.

Operations personnel (utility, licensed and non-licensed) investigated the protective relay targets (RLY) on Bus 2E-NAN-S02 and could find no reason for the protective relay actuation. At approximately 2234 MST on July 12, 1989, the load centers powered from Bus 2E-NAN-S02 were reenergized in accordance with an approved procedure. Eight minutes later at approximately 2242 MST, a load shed signal again deenergized the load centers on Bus 2E-NAN-S02. At approximately 2302 MST, Bus 2E-NAN-S02 was deenergized and taken out of service in order to perform further troubleshooting in accordance with the PVNGS Work Control Program.

At approximately 2322 MST on July 12, 1989 the SIAS/CIAS ESF actuations were secured, plant conditions were stabilized, and the NUE was terminated.

At approximately 0300 MST on July 13, 1989, Protection Relaying and Control (PR&C), the APS group responsible for investigating the oscillograph operated alarm, reset the alarm in accordance with procedure 42AL-2RK1B, "Panel B01B Alarm Response".

Following the event, at approximately 1529 MST on July 13, 1989, the secondary plant was being placed in the long path recirculation

(LPR) mode in accordance with normal operating procedures. Steam generator levels (SG)(AB) were being maintained using the Non-essential Auxiliary Feedwater Pump (P)(BA). Due to back-leakage through a check valve, Auxiliary Feedwater System (AFW)(BA) pressure of approximately 1580 psia was applied to a portion of the Main Feedwater System (MFWS)(SJ). This portion of the Main Feedwater System was rated for at least 1580 psia. When the manual isolation valve (ISV) on the Main Feedwater Pump (MFP) "B" bypass line was opened to establish long path recirculation in accordance with an approved procedure, the suction piping to the Main Feedwater Pumps was overpressurized. The suction piping is rated for 500 psia.

Concurrent with the MFP bypass valve being opened, a low suction pressure trip for Main Feedwater Pumps "A" and "B" was received in the Control Room. Immediately following the low suction pressure trip, operations personnel (utility, non-licensed) observed an abnormal decrease in seventh point feedwater heater (SN) outlet temperature. To prevent thermal shocking of the feedwater heater, operations personnel isolated the Non-essential Auxiliary Feedwater System from the Main Feedwater System. At approximately 1545 MST, the overpressurization event was terminated.

C. Status of structures, systems, or components that were inoperable at the start of the event that contributed to the event:

There were no structures, systems, or components inoperable at the start of the event which contributed to the event.

D. Cause of each component or system failure, if known:

The cause of the Bus 2E-NAN-S02 load shed malfunction described in Section I.B has been determined to be a failed potential transformer (PT) primary fuse on the "C" phase. The cause of the fuse failure is still under investigation and is expected to be completed by November 1, 1989. The cause will be described in a supplement to this report which is expected to be submitted by December 1, 1989.

The cause of the back-leakage through the MFP bypass check valve (SGN-V431) described in Section I.B has been determined to be loose fasteners which allowed the check valve disc to drop and not seat on its seating surface. The fastener locking devices required by the Technical Manual were not installed. The cause for the locking device not being installed cannot be determined since our records show no work has been performed on this valve since initial startup of Unit 2. This valve is a "non-safety related" component.

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

E. Failure mode, mechanism, and effect of each failed component, if known:

The failure of the potential transformer (PT) primary "C" phase fuse resulted in a load shed of Bus 2E-NAN-S02 including deenergization of reactor coolant pumps (RCP) 1B and 2B. This resulted in a reactor trip and turbine trip as described in Section I.B. Shortly before the reactor trip, there was indication of a problem with Bus 2E-NAN-S02 by a trouble alarm annunciation for the bus, an oscillograph operated annunciator alarm, oscillograph indication of an undervoltage condition on the bus, and a digital fault recorder indicating a fault on the bus. Operations personnel (utility, licensed and non-licensed) were attempting to identify the problem when the reactor trip occurred. The root cause of failure of the fuse is under investigation and will be included in a supplement to this report.

The failure of check valve SGN-V431 resulted in overpressurizing a portion of the main feedwater system as described in Section I.B. The cause of the check valve failure is described in Section I.C.

F. For failures of components with multiple functions, list of systems or secondary functions that were also affected:

Not applicable - No component failures had multiple functions which affected other systems or components.

G. For failures that rendered a train of a safety system inoperable, estimated time elapsed from the discovery of the failure until the train was returned to service:

Not applicable - There were no failures that rendered a train of a safety system inoperable.

H. Method of discovery of each component or system failure or procedural error:

The failed potential transformer primary "C" phase fuse was discovered as a result of troubleshooting performed after the event.

The failed check valve SGN-V431 was discovered as a result of troubleshooting performed after the event.

The overpressurization of the MFW pump suction piping was discovered when troubleshooting the feedwater pump low suction pressure trips as described in Section I.B. The bellows in all six low suction pressure switches were found to be deformed due to the pressure transient.

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I. Cause of Event:

1. The cause of the reactor trip discussed in Section I.B was a failed fuse which caused a load shed signal on Bus 2E-NAN-S02 resulting in a partial loss of flow. The "C" phase potential transformer (PT) is connected to Bus 2-NAN-S02. The fuse is located between the PT and the bus. Relay 227-S monitors the bus voltage through the PT. The relay load sheds the bus when the voltage is less than or equal to 77 percent of the bus voltage (13.8 kv). When the fuse failed, the relay saw no voltage from the PT and actuated the load shed relays.

2. The cause of the SIAS/CIAS ESF actuation described in Section I.B. was RCS depressurization due to improper Steam Bypass Control System (SBCS) response and excessive leakage past the pressurizer spray valves.

   a. The SBCS response was caused by the method of calibration used for the Proportional Integral (PI) Controller in the SBCS quick open controllers. This allowed the SBCS valves to remain open longer than was required. PVNGS calibrates the PI controller using the dial settings provided by the Combustion-Engineering (CE) setpoint document. The CE setpoint document also provides calibration curves for optimizing the quick open controller response. However, the CE setpoint document did not clearly indicate that these curves were to be used as part of the PI controller calibration. Therefore, PVNGS calibration procedures only required use of the dial settings and did not require use of the calibration curves to optimize the SBCS response.

   b. The cause of the pressurizer spray valve leakage was due to both valves positioners being out of calibration. PVNGS was aware that these valves were leaking prior to this event. However, because these valves are located in containment and inaccessible during power operation, they were scheduled to be recalibrated during the next plant shutdown. The final root cause analysis is not complete for the positioners. The root cause analysis is expected to be completed by November 1, 1989. The results of the root cause will be included in a supplement to this report which is expected to be submitted by December 1, 1989.

3. The cause of the overpressurization of the Main Feedwater Pump suction piping was due to back-leakage through a check valve as described in Section I.D. This check valve is not safety related and is not currently included in any testing

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or maintenance programs. Therefore there were no procedures for maintenance or testing of these check valves at the time of this event. A separate engineering evaluation is being conducted on this overpressurization event. A copy of this evaluation will be provided to the NRC resident inspector at PVNGS and the Regional Administrator.

J. Safety System Response:

The following automatic and manual safety system responses occurred during this event:

1. Containment Isolation System (automatic)(JM).
2. Low Pressure Safety Injection Trains "A" and "B" (automatic)(BP)
3. High Pressure Safety Injection Trains "A" and "B" (automatic)(BQ)
4. Emergency Diesel Generators Trains "A" and "B" (automatic)(DG)(EK)
5. Essential Spray Pond System Trains "A" and "B" (automatic)(BS)
6. Essential Chilled Water System Trains "A" and "B" (automatic)(KM)
7. Essential Cooling Water System Trains "A" and "B" (automatic)(BI)
8. Condensate Transfer System Trains "A" and "B" (automatic)(KA)
9. Containment Spray Trains "A" and "B" (automatic)(BE)
10. Control Room Essential HVAC Trains "A" and "B" (automatic)(AHU)
11. Control Building Essential Ventilation System Trains "A" and "B" (automatic)(AHU)
12. Auxiliary Building Essential HVAC System Trains "A" and "B" (automatic)(AHU)

K. Failed Component Information:

The failed fuse was manufactured by General Electric. It is a type EJ1, size B, rated at 15.5 kv and 0.5 amps.

The leaking check valve was manufactured by Pacific Valves. It is an 8 inch one-way flow check valve, figure number 58809-7-WE(20).

II. ASSESSMENT OF THE SAFETY CONSEQUENCES AND IMPLICATIONS OF THIS EVENT:

The Palo Verde Updated Final Safety Analysis Report (UFSAR) accident analysis for loss of reactor coolant system (RCS) flow assumes a total loss of offsite power resulting in a coastdown of

all four reactor coolant pumps (RCP's). The accident analysis transient is Departure from Nucleate Boiling Ratio (DNBR) limiting. The reduced RCS flow results in an initial rise in RCS average temperature and a reduction in DNBR. Based on this analysis, a reactor trip on low DNBR mitigates this transient and maintains DNBR above the safety limit. For this event, only two RCPs tripped and coasted down. The Steam Bypass Control System (SBCS) reduced RCS average temperature following the reactor trip. The accident analysis bounds this event. Based on this, DNBR limits were not exceeded.

Depressurization of the RCS resulted in a SIAS. The primary function of the SIAS for this event type is to maintain RCS inventory and maintain shutdown margin. In this event all control element assemblies (CEA)(AA)(ROD) inserted and RCS average temperature decreased to 551°F. Adequate shutdown margin was maintained and pressurizer level remained on scale throughout the event. Therefore adequate RCS inventory was maintained throughout this event.

The check valve SGN-V431 leakage resulted in overpressurization of a portion of the Main Feedwater System pump suction piping. This portion of the Main Feedwater System performs no safety function.

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

III. CORRECTIVE ACTIONS:

A. Immediate:

The failed fuse was replaced in the potential transformer for Bus 2E-NAN-S02.

B. Action to Prevent Recurrence:

1. The failed fuse is being evaluated by General Electric for the root cause of failure determination. Based on the root cause of failure, PVNGS will determine if any additional actions are required to prevent recurrence. Additional actions will be described in a supplement to this report.

2. The SBCS calibration procedure was revised and the SBCS quick open modules were optimized using the nominal setpoint values and the controller performance curve. The SBCS calibration procedure is being reviewed to ensure that all SBCS modules are calibrated using the technique, setpoints and tolerances in the Combustion Engineering setpoint document.

3. The pressurizer spray valves were recalibrated in accordance with an approved procedure to correct the excessive leakage condition. The final root cause determination has not been completed. Additional actions will be described in a supplement to this report.

4. The leaking MFP bypass check valve was repaired. Check valves SGN-V431 and SGN-V432 were pressure tested and no leakage was identified. The overpressurization event was evaluated and Main Feedwater Pump suction piping was walked down. It was determined to be acceptable for continued plant operation.

C. Corrective Actions by Other Units:

Units 1 and 3 will complete the following actions prior to startup from their current outages:

1. Check the optimization of the SBCS quick open modules.

2. Check the calibration of the pressurizer spray valves.

3. Inspect check valves SGN-V431 and SGN-V432.

An independent investigation of this event is also being conducted. Additional actions to prevent recurrence may be developed based upon the results of this independent evaluation. A supplement to this report will be provided to describe additional corrective actions to be taken. The supplement to this report is expected to be submitted by December 1, 1989.

IV. PREVIOUS SIMILAR EVENTS:

There have been no previous similar occurrences reported pursuant to 10CFR50.73.

There have been previous reactor trips reported. However, none of the previous reactor trips were attributable to the same root cause described in Section I.I. Therefore none of the previous corrective actions would have been expected to prevent this event.

V. ADDITIONAL INFORMATION

There have been 5 total accumulated actuation cycles of the Emergency Core Cooling System to date. This report satisfies the requirements of Technical Specification 3.5.2 ACTION b.