CATEGORY 1

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RECIP.NAME RECIPIENT AFFILIATION

SUBJECT: LER 95-012-00:on 951126,RT occurred caused by high water level in SG number 1.Sys walkdown performed.Work request written & work on solenoid valves completed on 951202.

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

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Arizona Public Service Company
PALO VERDE NUCLEAR GENERATING STATION
P.O. BOX 52034 • PHOENIX, ARIZONA 85072-2034

192-00954-JML/BAG/BE
December 26, 1995

JAMES M. LEVINE
VICE PRESIDENT
NUCLEAR PRODUCTION

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Station P1-37
Washington, DC 20555-0001

Dear Sirs:

Subject: Palo Verde Nuclear Generating Station (PVNGS)
Unit 1
Docket No. STN 50-528 (License No. NPF-41)
Licensee Event Report 95-012-00

Attached please find Licensee Event Report (LER) 95-012-00 prepared and submitted pursuant to 10CFR50.73. This LER reports a November 26, 1995, reactor trip on high water level in Steam Generator Number 1 and the actuation of the main steam isolation system. In accordance with 10CFR50.73(d), a copy of this LER is being forwarded to the Regional Administrator, NRC Region IV.

If you have any questions, please contact Burton A. Grabo, Section Leader, Nuclear Regulatory Affairs, at (602) 393-6492.

Sincerely,

James M Levine

JML/BAG/BE/pv

Attachment

cc: L. J. Callan (all with attachment)
K. E. Perkins
K. E. Johnston
INPO Records Center

IE22
1/1

020030

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 7

TITLE (4)
Reactor Trip Caused By A High Water Level In Steam Generator Number 1

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	
1 1	2 6	9 5	9 5	- 0 1 2	- 0 0	1 2	? ?	9 5	N/A	

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

OPERATING MODE (9): 1

POWER LEVEL (10): 1 0 0

	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
Burton A. Grabo, Section Leader, Nuclear Regulatory Affairs	AREA CODE 6 0 2 3 9 3 - 6 4 9 2

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 G	V A C B	M 1 3 8	Y					

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 November 26, 1995, at approximately 2100 MST, Palo Verde Unit 1 was in Mode 1 (POWER OPERATION), operating at approximately 100 percent power when the main turbine tripped on low condenser vacuum. A reactor power cutback was initiated upon the main turbine trip; however, due to the lockout of the other six steam bypass control valves (SBCVs) (to the condenser) on low condenser vacuum, only the two atmospheric SBCVs were available to maintain secondary pressure. A reactor trip resulted from a high water level in Steam Generator (SG) Number 1, and an Engineered Safety Feature Actuation System actuation of the Main Steam Isolation System also occurred due to the SG high water level.

The cause of the reactor trip was high steam generator water level in Steam Generator Number 1. A preliminary Personnel Performance Evaluation was conducted and no major Human Performance issues were identified.

On November 28 and 29, 1995, a system walkdown was performed in Units 2 and 3 that identified two vacuum breaker valves leaking on the "A" and "B" condenser shells in Unit 2. These valves were reworked on December 2, 1995.

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

TEXT

1. REPORTING REQUIREMENT:

This LER 528/95-012-00 is being written to report an event that resulted in the automatic actuation of an Engineered Safety Feature (ESF)(JE) including the Reactor Protection System (RPS)(JC) as specified in 10 CFR 50.73(a)(2)(iv).

Specifically, at approximately 2100 MST on November 26, 1995, Palo Verde Unit 1 was in Mode 1 (POWER OPERATION), operating at approximately 100 percent power when a main turbine (TA) trip occurred on low condenser vacuum. An automatic reactor power cutback (JD) was initiated upon the turbine trip; however, due to the low condenser vacuum, the six steam bypass control valves (SBCVs)(JI) to the condenser were locked out and only the two atmospheric SBCVs were available to maintain secondary pressure. Three main steam (SB) safety valves (MSSVs) opened to maintain secondary pressure. An automatic reactor trip occurred when the water level in Steam Generator Number 1 (SG-1)(AB) reached the Reactor Protection System (RPS) trip setpoint for high steam generator water level. Concurrent with the reactor trip, Unit 1 received an Engineered Safety Feature Actuation System (ESFAS)(JE) actuation of the Main Steam Isolation System (MSIS A and MSIS B)(SB) on high steam generator water level for SG-1.

2. EVENT DESCRIPTION:

On November 26, 1995, at approximately 2059 hrs, the shell "A" condenser vacuum breaker inadvertently opened, causing a loss of condenser vacuum in shell "A." Trend data shows that the vacuum breaker valve was opened for approximately 65 seconds before being closed by an APS Area Operator (AO)(utility, nonlicensed). Prior to the main turbine trip, the condenser vacuum in shell "A" decreased for approximately 22 seconds before affecting the "B" and "C" condenser shells (i.e., equalizing ducts were emptied). The AO observed SBCV 1001 going open while he was in the vicinity of the vacuum breaker. This occurred prior to the AO taking any action to isolate the shell "A" vacuum breaker.

At 21:00:17 hrs a main turbine trip occurred due to a low vacuum trip signal. The turbine trip resulted from a trip signal generated by 2 out of 3 pressure sensors on condenser shell "A" (7.5 inches HgA).

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

FACILITY NAME	DOCKET NUMBER	LER NUMBER					PAGE
		YEAR		SEQUENTIAL NUMBER		REVISIO NUMBER	
Palo Verde Unit 1	0 5 0 0 0 5 2 8	9 5	-	0 1 2	-	0 0	0 3 of 0 7

TEXT

Following the main turbine trip, control room operators (utility, licensed) entered the Abnormal Operating procedure, 41AO-1ZZ02, Load Rejection. The procedure objective is to stabilize reactor power and prevent a reactor trip following a loss of the main turbine generator.

At 21:00:19 hrs the SBCS generated a quick open signal that resulted in all eight steam bypass control valves going full open, as designed, to control secondary steam pressure with a simultaneous Reactor Power Cutback (JD) signal occurred which allowed selected control rods (AA) to insert and reduce reactor power to approximately fifty-five percent initially.

At 21:00:32 hrs (approximately 13 seconds later) the SBCS received a condenser interlock signal which closed the six steam bypass valves to the condenser. The two SBCVs to the atmosphere were still available and received a signal to modulate open. The SBCS condenser interlock signal was received after 2 of 3 condenser shells reach 5.0 inches HgA as designed.

Beginning at 21:01:06 hrs and over the next 14 seconds, three Main Steam Safety Valves (MSSVs 579, 554 and 561) lifted within their design pressure setpoints (1250 pounds per square inch absolute (psia) plus or minus 3 percent). At 21:02:36 hrs MSSV 579 reseated at approximately 1184 psia and steam pressures continued to decrease. The control room operators responded to the high secondary pressure by opening two Atmospheric Dump Valves (ADVs)(SB), one to each steam generator. The two ADVs (SG-178A and SG-185A) indicated "open" at 21:03:52 hrs and 21:03:56 hrs respectively. At 21:04:14 hrs the MSSV 554 valve position indicated closed and the MSSV reseated at a steam pressure of 1169 psia.

At 21:04:24 hrs reactor power decreased below 15 percent and the feedwater level control system (JB) proceeded through "cross over." The control system shifted from a three element control (above 15 percent power) to a single element control, the economizer valves closed, and the downcomer valves regulated feedwater flow to the SGs.

At 21:05 hrs the SG-2 water level reached its High Level Override (HLO) setpoint of 88 percent narrow range (control channel). The HLO signal closed the downcomer valve, and feedwater flow to SG-2 stopped. (Note: The HLO is not designed to prevent a reactor trip due to SG high water level.)

TEXT

The control room operators were monitoring the high water levels in the SGs and were aware that MSSV 561 had not reseated, and at approximately 21:06 hrs the Control Room Supervisor (CRS) gave permission to quick close the SG economizer outboard feedwater isolation valves. Based on the CRS statements, this action was taken to eliminate any chance of leakage past the economizer valves.

At 21:07:31 hrs the water level in SG-1 reached its high level trip setpoint (91 percent NR) on the safety channels, simultaneously, the control channel reached the HLO setpoints. The high SG level resulted in a reactor trip and MSIS.

Control Room Operators entered into Emergency Operating Procedures (EOP) and performed Standard Post Trip Actions. Three control element assemblies (CEAs) rod bottom lights did not illuminate immediately following the reactor trip. The operators verified the CEAs had inserted using other indicators including Lower Electric Limits (LEL) lights and the Control Element Assembly Calculator cathode-ray tube (CEAC CRT).

At about 21:12:40 hrs MSSV 561 indicated closed and reseated as steam pressure was lowered by the control room operator to approximately 1145 psia. At approximately 21:14:18 hrs steam pressure was being controlled on the high end of the EOP allowable band of 1140 to 1200 psia when MSSV 561 reopened at a steam pressure of 1196 psia. The control operators used the ADVs to lower steam pressure to approximately 1150 psia and MSSV 561 reseated at 21:15:43 hrs. Steam pressures were maintained at the low end of the band at approximately 1150 psia to prevent MSSV 561 from lifting again. The plant was stabilized in Mode 3, and the Site Shift Manager classified the event as an uncomplicated reactor trip.

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

Prior to the reactor trip signal resulting from high steam generator water level in SG-1, pressurizer pressure peaked at 2360 psia. The peak pressure criteria of 110 percent of design (2750 psia) was never challenged during this reactor coolant system (RCS)(AB) transient. The steam generator pressure peaked at 1282 psia and was well below the pressure criteria of 110 percent of secondary design pressure (1398 psia).

TEXT

Technical Specifications (TS) requires that all MSSVs have a lift setting of plus or minus 3 percent of nameplate set pressure. The TS tolerance for MSSVs with a set pressure of 1250 psi gage (psig) is 1228 to 1303 psia. The subsequent actuation of MSSV 561 approximately at 1196 psia and reseating at 1150 psia is bounded by Updated Final Safety Analysis Report (USFAR) Chapter 15.1 for Increased Heat Removal by the Secondary System.

This Unit 1 reactor trip was classified as a reactor trip on loss of condenser vacuum (LOCV) in the Anticipated Operational Occurrence (AOO) category which is a moderate frequency event. This event did not challenge the shutdown margin criteria. All CEAs inserted as designed. Equipment and systems assumed in the Safety Analysis were functional. Plant response was normal for the situation that occurred.

Scenarios defined in the USFAR Chapter 15 and design assumptions of the RPS will remain bounding for this Unit 1 reactor trip. Scenarios defined in the UFSAR Chapter 6 concerning Loss of Coolant Accidents were not implicated by this transient. The transient did not cause any violation of the Specified Acceptable Fuel Design Limits (SAFDLs).

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 independent investigation of this event is being conducted in accordance with the APS Corrective Action Program. Based on the results of the independent investigation to date, the cause of the reactor trip is that the FWCS was not designed to control high SG water level following a transient of this nature, i.e., loss of condenser vacuum (SALP Cause Code B: Design).

No unusual characteristics of the work location (e.g., noise, heat , poor lighting) directly contributed to this event

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

FACILITY NAME	DOCKET NUMBER	LER NUMBER					PAGE
		YEAR		SEQUENTIAL NUMBER		REVISIO NUMBER	
Palo Verde Unit 1	0 5 0 0 0 5 2 8	9 5	-	0 1 2	-	0 0	0 6 of 0 7

TEXT

5. STRUCTURE, SYSTEM, OR COMPONENT INFORMATION:

As stated earlier in Section 2 the condenser "A" vacuum breaker (1JCDNHV0045A) inadvertently opened. The apparent cause of failure was attributed to component aging of the middle insert gasket (O-ring) within the solenoid. Air operated valve 1JCDNHV0045A is comprised of a Matryx 26072SR80 air-to-open/spring-to-close actuator mounted on an 14 inch Pratt butterfly valve. The actuator is oriented in-line with the piping.

There are no indications that any other 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. No failures that rendered a train of safety system inoperable were involved. All safety system actuations that were required actuated as designed.

6. CORRECTIVE ACTIONS TO PREVENT RECURRENCE:

On November 28 and 29, 1995, a system walkdown of Units 2 and 3 condenser vacuum breakers was completed by Maintenance Engineering (utility, nonlicensed) personnel and AOs. The team identified two vacuum breaker valves in Unit 2 with minor solenoid leakage on the "A" and "B" condenser shells. A work request was written and work on the solenoid valves was completed on December 2, 1995.

Actions required from the above investigations will be track~d by APS' Commitment Action Tracking System (CATS). 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.

7. PREVIOUS SIMILAR EVENTS:

There has been one reactor trip due to SG high level reported pursuant to 10CFR50.73 in the last three years (LER 530/94-007-00). The reactor trip was initiated by a failure of the FWCS-2 master controller power fuse. Corrective actions taken for the previous event would not have prevented this event.

TEXT

8. ADDITIONAL INFORMATION:

MSSV 561 initially responded as designed, but subsequent actuations were less than 3 percent of the valve's nameplate set pressure. This was due to the length of time the valve was open, initially causing localized heating of the valve's internals and spring resulting in a reduced set pressure. MSSV 561 was declared inoperable, and the associated action statement was entered for one MSSV inoperable (maximum power level of 98.2 percent and variable over power trip set point reduced to 108 percent of rated thermal power) until the valve was tested. On November 29, 1995, the MSSV satisfactorily completed testing and was declared operable at 2250 MST.