APS

A subsidiary of Pinnacle West Capital Corporation

Palo Verde Nuclear
Generating Station

10CFR50.73

David M. Smith
Plant Manager
Nuclear Production

Tel. 623-393-6116
Fax. 623-393-6077
e-mail: DSMITH10@apsc.com

Mail Station 7602
P.O. Box 52034
Phoenix, AZ 85072-2034

192-01135-DMS/RAS
April 1, 2004

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 2004-001-00

Attached please find Licensee Event Report (LER) 50-528/2004-001-00 prepared and submitted pursuant to 10 CFR 50.73. This LER reports the February 3, 2004, discovery of pressure boundary leakage and subsequent manual reactor shutdown required by technical specifications.

The corrective actions described in this LER are not necessary to maintain compliance with regulations and therefore, Arizona Public Service Company makes no commitments to the NRC in this correspondence.

In accordance with 10 CFR 50.4, a copy of this LER is being forwarded to the NRC Region IV Administrator and the Senior Resident Inspector. If you have questions regarding this submittal, please contact Daniel G. Marks, Section Leader, Regulatory Affairs, at (623) 393-6492.

Sincerely,

Terry L. Radtke
for D.M. Smith

IE22

DMS/ras

Attachment

cc: B. S. Mallet, Region IV Administrator (all w/attachment)
N. L. Salgado, Sr. Resident Inspector
M. B. Fields, PVNGS Project Manager

NRC FORM 366 (7-2001)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 7-31-2004
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)		Estimated burden per response to comply with this mandatory information collection request: 50 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records Management Branch (T-6 E6), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to bjs1@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

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4. TITLE

Reactor Shutdown Due to Reactor Coolant System Pressure Boundary Leakage

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
									N/A	05000
02	03	2004	2004	– 001	– 00	04	01	2004	FACILITY NAME	DOCKET NUMBER
									N/A	05000

9. OPERATING MODE	1
10. POWER LEVEL	99

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply)

	20.2201(b)		20.2203(a)(3)(ii)		50.73(a)(2)(ii)(B)		50.73(a)(2)(ix)(A)
	20.2201(d)		20.2203(a)(4)		50.73(a)(2)(iii)		50.73(a)(2)(x)
	20.2203(a)(1)		50.36(c)(1)(i)(A)		50.73(a)(2)(iv)(A)		73.71(a)(4)
	20.2203(a)(2)(i)		50.36(c)(1)(ii)(A)		50.73(a)(2)(v)(A)		73.71(a)(5)
	20.2203(a)(2)(ii)		50.36(c)(2)		50.73(a)(2)(v)(B)		OTHER - Specify in Abstract below or in NRC Form 366A
	20.2203(a)(2)(iii)		50.46(a)(3)(ii)		50.73(a)(2)(v)(C)		
	20.2203(a)(2)(iv)	X	**50.73(a)(2)(i)(A)**		50.73(a)(2)(v)(D)		
	20.2203(a)(2)(v)		50.73(a)(2)(i)(B)		50.73(a)(2)(vii)		
	20.2203(a)(2)(vi)		50.73(a)(2)(i)(C)		50.73(a)(2)(viii)(A)		
	20.2203(a)(3)(i)	X	**50.73(a)(2)(ii)(A)**		50.73(a)(2)(viii)(B)		

12. LICENSEE CONTACT FOR THIS LER

NAME	TELEPHONE NUMBER (Include Area Code)
Daniel G. Marks, Section Leader, Regulatory Affairs	623-393-6492

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU FACTURER	REPORTABLE TO EPIX
B	BP	FCV	D243	YES					

14. SUPPLEMENTAL REPORT EXPECTED			15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
	YES (If yes, complete EXPECTED SUBMISSION DATE)	X NO				

16. ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)

On February 3, 2004 at approximately 1418 MST, Unit 1 operations personnel became aware of a non-isolable reactor coolant system (RCS) pressure boundary leak from a drain valve (SIA-V056) off of a high pressure safety injection line which is connected to the RCS loop 1 hot leg. The source of the estimated 1 to 2 drops/second leak was a crack in a socket weld on the upstream side of the one-inch drain valve.

At the time of discovery, Unit 1 was in Mode 1 (Power Operation) at 99 percent power at normal RCS temperature and pressure. A manual reactor shutdown was commenced at 1535 MST in accordance with Technical Specification and Technical Requirements Manual Limiting Conditions for Operation (LCO) 3.4.14 (RCS Operational Leakage) and TLCO 3.14.103 (Structural Integrity). LCO 3.4.14 requires the plant to be placed in Mode 3 (Hot Shutdown) within 6 hours and Mode 5 (Cold Shutdown) within 36 hours upon identification of RCS pressure boundary leakage. RCS boration was promptly commenced and the reactor was manually shutdown at 1731 MST.

There were no automatic reactor protection system or engineered safety features actuations and no other component failures, testing, or work in progress that contributed to the leak. There was no release of radioactivity to the environment and no impact to the health and safety of the public.

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17. NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)*

1. REPORTING REQUIREMENT(S):

This event is being reported pursuant to 10 CFR 50.73(a)(2)(ii)(A) due to a reactor coolant system pressure boundary leak and 10 CFR 50.73(a)(2)(i)(A) because the leak required a manual reactor shut down in accordance with the Technical Specifications (TS). Initial event notification was made to the NRC headquarters operation officer on February 3, 2004 (reference ENS # 40503).

2. DESCRIPTION OF EVENT RELATED STRUCTURE(S), SYSTEM(S) AND COMPONENT(S):

1PSIAV056 [EIIS: BP, FCV]

SIA-V056 is a one-inch, manually operated, Dresser Industries, globe valve on the train "A" high pressure safety injection (HPSI)[EIIS: BQ] recirculation header drain line. SIA-V056 is also used as a system drain on the safety injection (SI) system shutdown cooling (SDC)[EIIS: BP] suction line. The associated one-inch line (SI-248-BCAA-1) connects to the three-inch diameter HPSI hotleg injection line (SI-248-BCAA-3). The valve also serves as the isolation point for a reactor vessel level monitoring system (RVLMS)[EIIS: JB] connection.

3. INITIAL PLANT CONDITIONS:

On February 3, 2004 at approximately 1418 Mountain Standard Time (MST), Palo Verde Unit 1 was in Mode 1 (POWER OPERATION), operating at approximately 99 percent power at normal operating temperature and pressure. There were no major structures, systems, or components that were inoperable at the start of the event that contributed to the event. There were no failures that rendered a train of a safety system inoperable and no failures of components with multiple functions.

4. CHRONOLOGY OF RELEVANT EVENTS:

On February 3, 2004, during a containment entry to support installation of a temporary modification on the SDC suction piping, a radiation protection technician (other utility personnel) noticed a small amount of water below drain valve SIA-V056. Subsequent inspections by engineering personnel (other utility personnel) revealed that the leak was

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coming from a crack in the socket weld on the upstream side of SIA-V056. Engineers who observed SIA-V056, described the leak as a fine mist of approximately 1 to 2 drops per second.

At approximately 1418 MST, control room personnel (utility-licensed operators) concluded the leak was a non-isolable pressure boundary leak that affected the reactor coolant system (RCS) loop 1 pressure boundary [EIIS: AB] and entered TS Limiting Condition for Operation (LCO) 3.4.14, Condition B (pressure boundary leakage exists) and Technical Requirements Manual (TRM) TLCO 3.4.103 Condition A (structural integrity).

At approximately 1535 MST, after making notification to the energy control center power grid operators, plant operators commenced RCS boration to start the reactor downpower. At 1731 MST, plant operators shutdown the reactor from approximately 20 percent power per applicable procedure, entered Mode 3 (HOT STANDBY) satisfying TS Required Action B.1 of LCO 3.4.14, and commenced standard post trip actions. Heat removal was via normal steaming to the main turbine condenser and the electric grid remained stable throughout the plant shutdown. The cooldown of the plant commenced and Mode 5 (COLD SHUTDOWN) was entered at 1810 MST on February 4, 2004 satisfying Action B.2 of LCO 3.4.14.

On February 6, 2004 at 1215 MST, TRM TLCO 3.4.103 Condition A was exited when SIA-V056 was replaced. On February 8, 2004 at approximately 1145 MST, Unit 1 returned to power operations.

5. ASSESSMENT OF SAFETY CONSEQUENCES:

The manual reactor shutdown was uncomplicated and the plant, with the exceptions noted in section 10, responded as designed. No safety limits were exceeded during the shutdown and the event was bounded by current safety analyses. Primary and secondary pressure boundary limits were not exceeded as a result of the reactor shutdown. The transient did not cause any violation of the safety limits. 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 health and safety of the public.

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6. CAUSE OF THE EVENT:

A preliminary engineering assessment concluded that the pressure boundary leak (failure) was likely the result of a high-cycle fatigue type crack that developed in the upstream socket weld on valve SIA-V056. Metallurgical failure analysis indicates the crack initiated in the root pass of the weld.

As part of the initial evaluation, engineering personnel identified that the SIA-V056 piping support hanger configuration and the associated drain line piping did not match the approved design configuration. Specifically, a tie-back type support had been installed (circa 1989) at the valve's location due to concerns associated with vibration induced fatigue failures of socket weld connections in the RCS. The tie-back support was to have replaced the originally installed support and the modification package work details specified removal of the original support after the tie-back support was installed. However, the modification package instructions appear to have been insufficiently detailed to ensure the original support was removed. Engineering evaluations have preliminarily demonstrated that the socket weld failure is likely attributable to elevated stresses created by the as-found configuration (i.e., the combination of the two supports) while being subjected to higher than normal vibration levels.

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

7. TRANSPORTABILITY:

The SIA-V056 socket weld failure was considered potentially transportable to other portions of the Unit 1 SDC suction line where higher than normal vibration levels could result in elevated stresses and subsequent high cycle fatigue. As a result, engineering personnel inspected the other supports in the train "A" suction line and its branch lines that are subjected to higher vibration levels for compliance with approved designs. The inspections revealed the other supports were installed in accordance with the approved design.

Engineering personnel considered the condition to be potentially transportable to Units 2 and 3, given that the same piping and support designs had been implemented in these

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units. Based on this consideration, the Unit 2 and 3 supports were inspected and found to be in compliance with design configuration. Similarly, Unit 2 and 3 vibration levels were also verified to be significantly lower than those in Unit 1.

8. CORRECTIVE ACTIONS:

The original support (that was to have been removed upon installation of the tie-back support) was removed during the plant shutdown.

The failed weld was corrected by replacing the entire piping spool from the immediate upstream 90 degree elbow to the RVLMS flange plates. All welds on this section of pipe, as well as all welds on the rest of the one-inch pipe, had a two-to-one weld overlay to increase system toughness against sustained higher level vibrations.

The Unit 1 welds on the three-inch and one-inch branch lines were inspected and no additional evidence of cracking was identified.

Engineering personnel increased vibration monitoring of the Unit 1 SDC and branch line piping during power ascension and the increased monitoring will continue for the duration of Unit 1 cycle 11. The vibration data obtained was used to quantify the associated stress levels along the branch piping and component welds to ensure the vibration levels are within acceptable limits for continued operation.

A work order was initiated to remove and repair the sheared tie-back support pipe clamp that was discovered on March 19, 2004, and the condition has been entered into the condition reporting system for further evaluation.

9. PREVIOUS SIMILAR EVENTS:

There have been no RCS pressure boundary leaks in the past three years that have a similar failure mechanism or that should have been prevented by previous implemented corrective actions.

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10. ADDITIONAL INFORMATION:

Subsequent to the reactor shutdown, six control element assembly (CEA)[EIIS: AA] position indication (rod bottom) mimic lights failed to promptly illuminate in the control room and operations personnel verified that the CEAs were in the expected (fully inserted) position by control element assembly calculators. Additionally, after the manual shutdown, high water levels (approximately 73 percent narrow range indication) in steam generator 2 prompted operations personnel to take manual control of each downcomer feedwater control valve to control SG level. These CEA position indication and feedwater control conditions have been documented and are being addressed in accordance with the PVNGS corrective action and work control programs.

On March 19, 2004 at approximately 1600 MST, Unit 1 was operating at approximately 99 percent power, when a containment entry was made to gather routine vibration readings of the train "A" SDC suction valve (SIA-V651) [EIIS: BP, ISV]. During the data gathering, it was observed that a pipe clamp was loose on the tie-back support between the three-inch HPSI long term recirculation line (SI-248-BCAA-3) and the one-inch drain line (SI-248-BCAA-1) to SIA-V056. Further investigation revealed that the pipe clamp had sheared near its weld connection to a vertical stanchion. Engineering personnel had previously inspected the tie-back support in conjunction with corrective actions for the SIA-V056 socket weld failure, and at that time, the support appeared to be intact. Therefore, the tie-back support failure apparently occurred after the return to power following the February 3, 2004, manual shutdown.

Engineering personnel performed stress analyses for the one-inch drain line without crediting the tie-back support and determined the resultant stress met American Society of Mechanical Engineers (ASME) code requirements. Therefore, the degraded tie-back support did not impact the operability of HPSI long term recirculation line or the one-inch drain line. Following removal of the broken support bracket, vibration readings indicated a significant reduction in the one-inch line vibration when compared to the readings taken following the February 3, 2004 manual reactor shutdown. Engineering personnel performed a stress analysis that showed a marked decrease in stress due to the reduced vibration.