APS

A subsidiary of Pinnacle West Capital Corporation

Palo Verde Nuclear
Generating Station

10CFR50.73

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192-01147-DMS/SAB/DJS
July 30, 2004

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

Dear Sirs:

Subject: Palo Verde Nuclear Generating Station (PVNGS)
Unit 3
Docket No. STN 50-530
License No. NPF-74
Licensee Event Report 2004-002-00

Attached please find Licensee Event Report (LER) 50-530/2004-002 -00 that has been prepared and submitted pursuant to 10CFR50.73. This LER reports an automatic reactor trip on Low DNBR following a main turbine control system malfunction.

In accordance with 10CFR50.4, a copy of this LER is being forwarded to the NRC Region IV Office 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.

The corrective actions described in this LER are not necessary to maintain compliance with regulations. Arizona Public Service Company makes no commitments in this letter.

Sincerely,

DMS/SAB/DJS/kg

Attachment

cc: B. S. Mallett — NRC Region IV Regional Administrator
M. B. Fields — NRC NRR Project Manager
N. L. Salgado — NRC Senior Resident Inspector for PVNGS

JE22

NRC FORM 366 (7-2001)

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104 EXPIRES 7-31-2004

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.

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

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4. TITLE

Main Turbine Control System Malfunction Results in Automatic Reactor Trip on Low DNBR

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
									None	05000
06	07	2004	2004	– 002	– 00	07	29	2004	FACILITY NAME	DOCKET NUMBER
									None	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)	X	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)		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)		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

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 June 7, 2004 at approximately 1458 hours (MST), Unit 3 was operating in Mode 1, Power Operations, at approximately 99 percent power when the main turbine intercept valves fast closed and the control valves ramped closed. The reactor power cutback and steam bypass control systems (RPCB and SBCS) responded to the large decrease in steam flow by inserting group 4 and 5 control rods into the core and opening steam bypass valves to reduce reactor power and maintain reactor coolant temperature. Approximately 10 seconds after the RPCB occurred, a low DNBR reactor trip was initiated by the core protection calculators (CPCs) as a result of rodded radial peaking factor for the control rod configuration.

The cause of the main turbine valves closure is believed to be a control system malfunction in the speed control circuit. The cause of this malfunction has not been determined and the system is currently under a monitoring program/investigation program. The cause of the CPC reactor trip is still under investigation.

Subsequent to this event, a similar event occurred when Unit 2 experienced a main turbine trip, RPCB, and subsequent reactor trip on low DNBR on July 14, 2004.

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

1. REPORTING REQUIREMENT(S):

This LER (50-530/2004-002-00) is being submitted pursuant to 10 CFR 50.73(a)(2)(iv)(A), to report a reactor protection system (RPS) (EIIS: JC) initiated reactor trip, while critical on June 7, 2004 at approximately 1458 hours Mountain Standard Time (MST).

On June 7, 2004 at 1757 hours MST, APS made notification of the event to the Nuclear Regulatory Commission (NRC) via the emergency notification system (ENS# 40795).

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

The main turbine (EIIS: TA) is equipped with an electro-hydraulic-control (EHC) system that combines the principles of solid-state electronics and high-pressure hydraulics (EIIS: TG) to control steam flow through the turbine. The control system has three major subsystems: speed control unit, load control unit, and valve flow control units.

The steam bypass control system (SBCS) (EIIS: JI) controls the positioning of the turbine bypass valves, through which steam is bypassed around the turbine into the unit condenser, with exception of two valves which dump steam to atmosphere. These two valves are the last to open and first to close during steam bypass operation.

The reactor power cutback system (RPCS) (EIIS: JD) is a control system designed to accommodate loss of load or loss of one main feed pump events by providing a "step" reduction in reactor power. The step reduction in reactor power is accomplished by the simultaneous dropping of one or more pre-selected groups of full length regulating control element assemblies (CEAs) (EIIS: AA) into the core.

3. INITIAL PLANT CONDITIONS:

On June 7, 2004, at approximately 1458 MST, Palo Verde Unit 3 was in Mode 1 (POWER OPERATION), operating at approximately 99 percent power. There were no major structures, systems, or components that were inoperable at the start of the event that contributed to the event.

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4. EVENT DESCRIPTION:

On June 7, 2004 at approximately 1458 hours the Unit 3 main turbine EHC system experienced a malfunction that caused the main turbine intercept valves to fast close and the control valves to stroke closed. Approximately 4 seconds into the event SBCS quick opened its valves and RPCB signals were generated. All eight SBCS valves quick opened and control rods for regulating groups 4 and 5 fully inserted into the core. Some moisture separator reheater (MSR) relief valves lifted and reset. Approximately 1 second after the closure of the main turbine intercept valves, three of the intercept valves started to reopen resulting in approximately 180 megawatt increase in generator output. The other three intercept valves did not indicate reopening. The primary and secondary relief valves did not lift.

Reactor power decreased as expected and leveled off at approximately 57 percent then started to increase. Approximately 10 seconds after the RPCB, three channels (A, B, D) of the core protection calculators (CPC) (EIIS: JC) generated a low DNBR reactor trip. Channel C was inoperable and in bypass at the time of the trip.

All control rods inserted into the core. The feedwater control (FWCS) (EIIS: JB) and SBCS systems operated to maintain steam generator level and reactor coolant system (RCS) heat removal. Pressurizer level and pressure control systems (EIIS: AB) operated to maintain level and pressure.

The control room staff entered the standard post trip procedure and classified the event as an uncomplicated reactor trip. No other safety system actuations occurred and none were required. The operators elected to take manual control of the FWCS downcomer valves on both steam generators to facilitate a faster restoration of pressurizer level and pressure to normal values. RCS Pressure and level control were somewhat impacted by planned work on NGN-L12 (no power to some pressurizer heaters) and planned work on a charging pump.

Off site power remained available throughout the event as did normal heat removal through the condenser. The NRC operations center was called and notified of the event at 2057 hours EST (reference ENS 40795).

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

5. ASSESSMENT OF SAFETY CONSEQUENCES:

The reactor trip did not result in a transient more severe than those already analyzed in the updated Final Safety Evaluation Report Chapters 6 and 15. The primary system and secondary pressure boundary limits were not approached and no violations of the specified acceptable fuel design limits (SAFDL) occurred.

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

The condition would not have prevented the fulfillment of any safety function and did not result in a safety system functional failure as defined by 10CFR50.73(a)(2)(v).

6. CAUSE OF THE EVENT:

An evaluation of the EHC malfunction and subsequent reactor trip is being conducted in accordance with the PVNGS corrective action program. It appeared that the EHC speed control circuit experienced a problem since that is the circuit most likely to cause a fast closure of the Intercept Valves along with a closure of the Control Valves. The only problem found during initial trouble-shooting was a loose amphenol connector at the back-up speed probe located in the turbine front standard. The conclusion at this time is the back-up speed circuit (loose amphenol connector) is a probable cause of the CV/CIV closure, however cannot at this time be definitively called the root cause.

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The cause of the reactor trip has been attributed to a diminished DNBR margin as a result of the rodded radial peaking factor for the control rod configuration that existed following the RPCB. The cause of this decrease in margin remains under investigation.

If APS determines that the cause(s) are something other than a fault with the speed control system or a diminished DNBR margin induced CPC reactor trip, then a supplement to this report will be submitted.

7. CORRECTIVE ACTIONS:

Although the root cause of the EHC malfunction has not been identified, the following actions were taken:

- Replaced 2 analog load control boards
- Replaced 4 analog speed control boards
- Replaced 1 logic speed control board
- Tightened the backup speed probe amphenol connector
- Replaced the primary and backup speed probes

Currently, a data acquisition recorder is attached to the turbine EHC control cabinet in an attempt to capture data that will help determine the cause of failure.

APS is evaluating whether the Radial Peaking Factors and Rod Shadowing Factors for Control Rod Groups 4 and 5 (inputs to the CPC DNBR calculation) should be reduced to alleviate plant trips post RPCB events.

8. PREVIOUS SIMILAR EVENTS:

Subsequent to this Unit 3 event a similar event occurred when Unit 2 experienced a loss of the main turbine from approximately 100 percent power and a subsequent RPCB on July 14, 2004. In that event a reactor trip also occurred on low DNBR. That event will be submitted in another LER.