10 CFR 50.73

APS

A subsidiary of Pinnacle West Capital Corporation

Palo Verde Nuclear Generating Station

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102-05479-CE/SAB/DJS
April 28, 2006

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 2006-002-00

Attached please find Licensee Event Report (LER) 50-530/2006-002-00 prepared and submitted pursuant to 10 CFR 50.73. This LER reports an automatic Unit 3 Reactor Protection System (RPS) actuation (Reactor Trip) on Low DNBR due to Control Element Assembly Calculator (CEAC) number 1 penalty factor receiving an invalid input signal.

In accordance with 10 CFR 50.4, copies of this LER are being forwarded to the NRC Regional Office, NRC Region IV and the PVNGS Senior Resident Inspector. If you have questions regarding this submittal, please contact James A. Proctor, Section Leader, Regulatory Affairs, at (623) 393-5730.

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,

CE/SAB/DJS/gt

Attachment

cc: B. S. Mallett NRC Region IV Regional Administrator
M. B. Fields NRC NRR Project Manager - (send electronic and paper)
G. G. Warnick NRC Senior Resident Inspector for PVNGS

IE22

NRC FORM 366 (6-2004) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB: NO. 3150-0104 EXPIRES: 06/30/2007

Estimated burden per response to comply with this mandatory 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 and FOIA/Privacy Service Branch (T-5 F52), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects@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 an 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)

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
Palo Verde Nuclear Generating Station (PVNGS) Unit 3	05000530	1 OF 5

4. TITLE
Automatic Reactor Protection System actuation due to an invalid Control Element Assembly Calculator Penalty Factor

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
										05000
03	05	2006	2006	- 002 -	00	04	28	2006	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE
1

10. POWER LEVEL
100

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

- ☐ 20.2201(b)
- ☐ 20.2201(d)
- ☐ 20.2203(a)(1)
- ☐ 20.2203(a)(2)(i)
- ☐ 20.2203(a)(2)(ii)
- ☐ 20.2203(a)(2)(iii)
- ☐ 20.2203(a)(2)(iv)
- ☐ 20.2203(a)(2)(v)
- ☐ 20.2203(a)(2)(vi)
- ☐ 20.2203(a)(3)(i)
- ☐ 20.2203(a)(3)(ii)
- ☐ 20.2203(a)(4)
- ☐ 50.36(c)(1)(i)(A)
- ☐ 50.36(c)(1)(ii)(A)
- ☐ 50.36(c)(2)
- ☐ 50.46(a)(3)(ii)
- ☐ 50.73(a)(2)(i)(A)
- ☐ 50.73(a)(2)(i)(B)
- ☐ 50.73(a)(2)(i)(C)
- ☐ 50.73(a)(2)(ii)(A)
- ☐ 50.73(a)(2)(ii)(B)
- ☐ 50.73(a)(2)(iii)
- ☒ 50.73(a)(2)(iv)(A)
- ☐ 50.73(a)(2)(v)(A)
- ☐ 50.73(a)(2)(v)(B)
- ☐ 50.73(a)(2)(v)(C)
- ☐ 50.73(a)(2)(v)(D)
- ☐ 50.73(a)(2)(vii)
- ☐ 50.73(a)(2)(viii)(A)
- ☐ 50.73(a)(2)(viii)(B)
- ☐ 50.73(a)(2)(ix)(A)
- ☐ 50.73(a)(2)(x)
- ☐ 73.71(a)(4)
- ☐ 73.71(a)(5)
- ☐ OTHER

Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

FACILITY NAME	TELEPHONE NUMBER (Include Area Code)
James A. Proctor, Section Leader, Regulatory Affairs	623-393-5730

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
X	SB	CPU	S204	Y					

14. SUPPLEMENTAL REPORT EXPECTED

☐ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)* ☒ NO

15. EXPECTED SUBMISSION DATE

MONTH	DAY	YEAR

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

On March 5, 2006 at 07:10 Mountain Standard Time (MST) Palo Verde Unit 3 was in Mode 1 (Power Operations), operating at approximately 100 percent power, when an automatic reactor trip occurred on a low Departure from Nucleate Boiling Ratio (DNBR) input. The four Core Protection Calculators (CPC) generated a reactor trip on low DNBR due to an invalid Control Element Assembly Calculator (CEAC) input. CEAC number 1 sensed positional data for Control Element Assembly (CEA) 60 that exceeded the allowed deviation and as a result generated a high penalty factor which was transmitted to all four CPC channels. This high penalty factor resulted in a CPC DNBR trip output, as per design. The reactor was stabilized in Mode 3 (HOT STANDBY), and the Shift Manager classified the event as an uncomplicated reactor trip. No engineered safety feature actuations occurred during the event and none were required.

The cause of the reactor trip was an electronic component failure on a circuit board that caused an invalid (CEA) position deviation to CEAC number 1. The circuit board was replaced and the Unit was restarted.

In the past three years, Palo Verde reported Reactor Protection System (RPS) actuations, but none associated with the same root cause.

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Palo Verde Nuclear Generating Station Unit 3	05000530	2006	-- 002	-- 00	2 OF 5

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

1. REPORTING REQUIREMENT(S):

This LER (50-530/2006-002-00) 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 which occurred on March 5, 2006 at approximately 07:10 Mountain Standard Time (MST).

On March 5, 2006 at 09:12 MST, APS made notification of the event to the Nuclear Regulatory Commission (NRC) via the event notification system.
(Reference: ENS call 42387)

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

The core protection calculator/control element assembly calculator (CPC/CEAC)(EIIS: JC) system monitors pertinent reactor core conditions to provide CEA withdrawal prohibit (CWP) signals to the control element drive mechanism control system (CEDMCS) (EIIS: AA) and provides an accurate, reliable means of initiating a reactor trip. The CPC/CEAC system is an integral part of the plant protective system in that it provides departure from nucleate boiling ratio (DNBR) and local power density (LPD) trip signals to the reactor protection system (RPS) (EIIS: JC). Trip signals are provided to the reactor protection system whenever the minimum DNBR or fuel design limit LPD is approached during reactor operation.

Each CEAC receives reed switch assembly inputs for all control element assemblies (CEAs) (EIIS: AA). The CEACs compare the positions of all CEAs within each CEA subgroup and determine penalty factors based upon CEA deviations within a subgroup. A penalty factor is transmitted via four fiber-optic data links to the CPCs. The CPCs also compute penalties for CEA group out-of-sequence and deviations between subgroup conditions.

The reactor protection system (RPS) provides a rapid and reliable shutdown of the reactor to protect the core and the reactor coolant system pressure boundary from potentially hazardous operating conditions. Shutdown is accomplished by the generation of reactor trip signals. The trip signals open the reactor trip switchgear (RTSG) breakers (EIIS: AA), de-energizing the control element drive mechanism (CEDM) coils (EIIS: AA), allowing all CEAs to drop into the core by the force of gravity.

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

3. INITIAL PLANT CONDITIONS:

On March 5, 2006 Palo Verde Unit 3 was in Mode 1 (Power Operations), operating at approximately 100 percent power. At the start of the event an unexpected alarm was received for Control Element Assembly Calculator (CEAC) number 1 (CEAC 1). No other major structures, systems, or components were inoperable at the start of the event that contributed to the event.

4. EVENT DESCRIPTION:

On March 5, 2006 at approximately 07:10 MST the Unit 3 reactor tripped from 100% power due to Low DNBR trips on all four CPCs. At 07:04 MST, a CEA Sensor CEAC 1 alarm was received. The Control Room staff addressed the alarm response procedure and determined CEA number 60 input to CEAC 1 was indicating 160.6" withdrawn. CEA number 60 indicated 151.1" withdrawn on CEAC 2. The staff entered procedure 72AO-9SB01 "CEAC Inoperable" and the reactor trip occurred before CEAC INOP codes could be placed into the CPCs.

The control room staff entered the emergency operations procedures and diagnosed a reactor trip, the Control Room Supervisor (CRS) entered site procedure 40EP-9EO02 "Reactor Trip". The event was classified by the Shift Manager as an Uncomplicated Reactor Trip. The Shift Manager and Shift Technical Advisor (STA) reviewed site procedure EPIP-99 and no event classification was required. The plant was stabilized in Mode 3.

The NRC Operations Center was notified of the reactor trip in accordance with 10CFR50.72(b)(2) at 09:12 MST. (ENS 42387)

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

5. ASSESSMENT OF SAFETY CONSEQUENCES:

The low DNBR trip is provided to prevent the DNBR in the core from exceeding the fuel design limit in the event of design bases anticipated operational occurrences. The reactor trip occurred when all four channels of CPCs calculated a DNBR value that exceeded the low DNBR trip setpoint. The cause of the reactor trip was a hardware induced CEA position deviation error that resulted in a large penalty factor being generated in control element assembly calculator CEAC1. The CPCs calculated DNBR based on the penalty factor generated in CEAC1. The actual DNBR safety limit was not approached nor exceeded.

Primary and secondary pressure boundary limits were not approached due to the reactor tripping from a steady state condition, followed by a "quick open" of the steam bypass control system (EIIS: JI). The transient did not cause any violation of the specified acceptable fuel design limits. 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. Unit 3 plant performance and plant protection system evaluations were performed to determine plant responses to transients experienced subsequent to the plant trip. The plant performance evaluation included a safety function impact analysis for each of the safety functions and included an assessment of equipment malfunctions, abnormal alarms and/or events observed during the event. The evaluations revealed that the plant responded as required, the reactor trip was uncomplicated, no safety limits were exceeded, and the event was bounded by current safety analyses.

There are no actual safety consequences as a result of this condition, the condition would not have prevented the fulfillment of the safety function, and the condition did not result in a safety system functional failure as defined by 10 CFR50.73 (a) (2) (v).

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

6. CAUSE OF THE EVENT:

The direct cause of the reactor trip was an erroneous position indication signal for CEA #60 as sensed by CEAC #1 due to a faulty CEA Positional Isolation Amplifier (CPIA) board. The root cause of the CPIA board failure is categorized as a probable cause of "random electronic failure" as defined by site procedure 70DP-0EE01 (Equipment Root Cause of Failure Analysis) of the U6 operational amplifier.

7. CORRECTIVE ACTIONS:

On March 6, 2006 repairs were made to the CEAC number 1 by replacing and satisfactorily retesting the faulted circuit board that contained the integrated operational amplifier for CEA number 60.

The Unit was started and synchronized to the grid on March 6, 2006.

Additional detailed equipment root cause of failure analysis is in progress as an Equipment Root Cause of Failure Analysis (ERCFA) Level II investigation (CRDR 2873800). If any conclusions from that investigation differ significantly from the information provided above, a supplement to this LER will be provided.

8. PREVIOUS SIMILAR EVENTS:

In the past three years, Palo Verde reported reactor shutdowns initiated by Reactor Protection System but none associated with the same root cause.