10 CFR 50.73

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102-06904-DCM/DFH
June 24, 2014

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 2013-001-01

Enclosed please find Licensee Event Report (LER) supplement 50-530/2013-001-01 that has been prepared and submitted pursuant to 10 CFR 50.73. The original LER reported a degraded principal safety barrier and a condition prohibited by Technical Specifications that resulted from leakage on one reactor vessel bottom-mounted instrument nozzle assembly. This LER supplement provides results of the cause analysis on the affected bottom-mounted instrument nozzle assembly.

In accordance with 10 CFR 50.4, copies of this LER are being forwarded to the Nuclear Regulatory Commission (NRC) Regional Office, NRC Region IV, and the Senior Resident Inspector.

Arizona Public Service Company makes no commitments in this letter. If you have questions regarding this submittal, please contact Mark McGhee, Regulatory Affairs Department Leader, at (623) 393-4972.

Sincerely,

D.C. Mims

DCM/DFH/hsc

Enclosure

cc:	M. L. Dapas	NRC Region IV Regional Administrator
	J. K. Rankin	NRC NRR Project Manager for PVNGS (electronic & hardcopy)
	M. M. Watford	NRC NRR Project Manager (electronic & hardcopy)
	M. A. Brown	NRC Senior Resident Inspector for PVNGS (electronic)

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NRC FORM 366 (10-2010) U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

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

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4. TITLE
Leakage on Reactor Vessel Bottom-Mounted Instrumentation Nozzle 3

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
10	7	2013	2013	- 001 -	01	6	24	2014	FACILITY NAME	DOCKET NUMBER

9. OPERATING MODE: 5

10. POWER LEVEL: 0

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

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

12. LICENSEE CONTACT FOR THIS LER

FACILITY NAME	TELEPHONE NUMBER (Include Area Code)
Mark McGhee, Regulatory Affairs Department Leader	623-393-4972

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	AB	RPV	CE	Y					

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

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

On October 6, 2013, during a scheduled visual examination of the 61 bottom-mounted instrument (BMI) nozzles on the Unit 3 reactor vessel, white residue was identified around the annulus region of BMI nozzle 3. At 0219 on October 7, 2013, engineering personnel determined, based on past and present photographic evidence, that the white residue likely resulted from reactor coolant system (RCS) pressure boundary leakage. The pressure boundary leakage resulted in a condition prohibited by Technical Specification 3.4.14, reportable pursuant to 10 CFR 50.73(a)(2)(i)(B), and a degraded principal safety barrier, reportable pursuant to 10 CFR 50.73(a)(2)(ii)(A). Non-destructive examination (NDE) of BMI nozzle 3 identified axial cracking in a near-surface weld flaw in the J-groove weld that extended into the nozzle tube and ultimately resulted in the RCS leak path. An extent of condition evaluation found no indication of unacceptable flaws or leakage in the remaining 60 BMI nozzle assemblies in Unit 3. Additionally, no evidence was found that indicated leakage from BMI nozzle assemblies in Units 1 or 2.

The cause of the pressure boundary leakage was axial cracking in a sub-surface BMI nozzle J-groove weld flaw that extended into the nozzle to a length exceeding the J-groove weld leg thickness. Corrective actions included completion of an American Society of Mechanical Engineering (ASME) Code approved weld pad/half-nozzle repair and increasing the frequency of BMI visual examinations to every refueling outage.

No similar events have been reported to the NRC by PVNGS in the prior three years.

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All times are Mountain Standard Time and approximate unless otherwise indicated.

1. REPORTING REQUIREMENT(S):

This Licensee Event Report (LER) is being submitted pursuant to 10 CFR 50.73(a)(2)(ii)(A) as a degraded principal safety barrier and 10 CFR 50.73(a)(2)(i)(B) as a condition prohibited by Technical Specification (TS) Limiting Condition for Operation (LCO) 3.4.14, RCS Operational Leakage, due to reactor coolant system pressure boundary leakage.

At approximately 2100 on October 6, 2013, during a scheduled visual examination of reactor coolant system components, white residue was identified around the annulus region of reactor vessel bottom-mounted instrument (BMI) nozzle 3. At 0219 on October 7, 2013, engineering personnel determined that, based on past and present photographic evidence, the white residue was most likely boric acid resulting from reactor coolant system pressure boundary leakage. This condition was reported to the NRC pursuant to 10 CFR 50.72 (b)(3)(ii)(A) via Event Notification 49416 at 0955 on October 7, 2013.

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

A primary function of the reactor coolant system (RCS) (EIIS: AB) is to provide the second of three principle barriers against fission product release to the environment. The RCS is designed, constructed and maintained to minimize the potential for reactor coolant leakage. RCS component joints are made by welding, bolting, rolling, or pressure loading, and valves are provided to isolate connecting systems from the RCS. During plant life the joint and valve interfaces can produce varying amounts of reactor coolant leakage through normal operational wear or mechanical deterioration.

PVNGS TS LCO 3.4.14, *RCS Operational Leakage*, is applicable in Modes 1 through 4 when the RCS is capable of being pressurized and provides limitations for RCS leakage. The LCO specifies the types and amounts of leakage allowable and provides required actions when leakage rates exceed allowable values. RCS pressure boundary leakage is defined as non-isolable leakage through a component body, pipe wall or vessel wall. No pressure boundary leakage is allowed.

The reactor vessel contains penetrations which include two vessel outlet pipes, four vessel inlet pipes and 61 BMI nozzle assemblies. The BMI nozzle assemblies allow the in-core instrumentation (ICI) to enter the bottom of the reactor vessel (see Figure 1) and transition to the fuel assemblies. There are 61 fixed ICI detector assemblies, each comprised of a string of five self-powered rhodium detectors, one background detector and a core exit thermocouple (CET). The rhodium detectors are used to measure local core neutron flux

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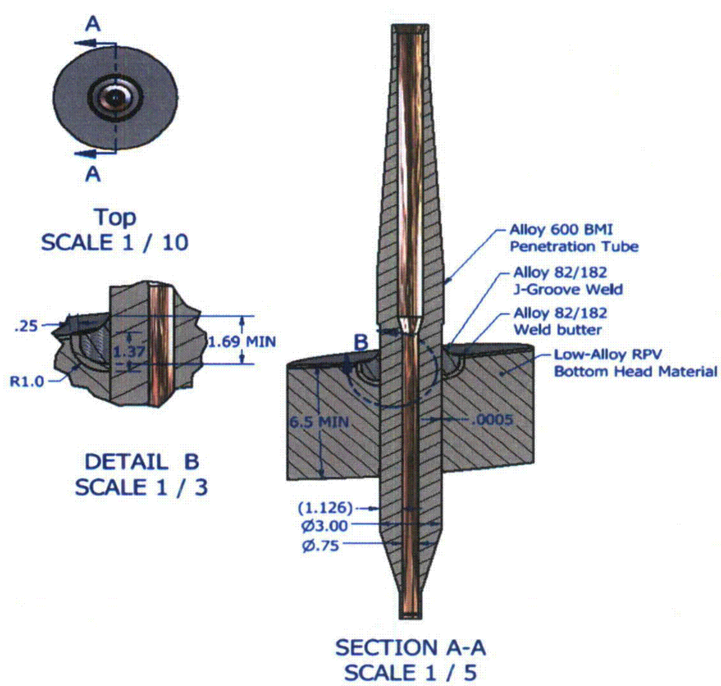
levels to monitor power level, power distribution, flux tilt, flux shape and radial peaking factors. The CETs are used to provide control room indication of coolant temperature at the core exit during operating or accident conditions. The entire ICI assembly is covered with a solid tubular sheath and terminates at a multi-pin connector which mates with a connector on the field cables. The field cables connect to each ICI assembly at a seal table located in the containment building adjacent to the reactor vessel. The seal table houses the 61 seal plugs which allow each ICI element to traverse the RCS pressure boundary with a leak tight connection. The RCS pressure boundary includes the seal plugs and 61 stainless steel tubes which contain the ICI assemblies and connect the seal table to the BMI penetrations on the reactor vessel lower head. During refueling operations the ICI detector assemblies are withdrawn from the reactor vessel to facilitate movement of fuel assemblies. Prior to operation, the ICI detector assemblies are inserted into the reactor vessel and the seal plug configuration and field cable terminations are restored.

Figure 1- Illustration of a BMI Nozzle Assembly Including Location of Welds



The BMI nozzle assemblies in the reactor vessel lower head are fabricated from Alloy 600 and are internally connected to the vessel lower head with J-groove welds made from Alloy 82/182 (see Figure 1). These materials have known susceptibility to primary water stress corrosion cracking (PWSCC). Stresses that make Alloy 600 nozzle tubes and their Alloy 82/182 J-groove attachment welds susceptible to cracking are inherently induced by the welding of the nozzle tube to the inside surface of the reactor vessel lower head during

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vessel fabrication. Three factors: susceptible material, high temperature primary water, and stress must be present for PWSCC to occur. To monitor the integrity of the RCS pressure boundary, the PVNGS in-service inspection program conducts bare metal visual examinations of BMI nozzle assemblies in accordance with 10 CFR 50.55a(g)(6)(ii)(E) and American Society of Mechanical Engineers (ASME) Code Case N-722-1.

3. INITIAL PLANT CONDITIONS:

At approximately 2100 on October 6, 2013, PVNGS Unit 3 was stable in Mode 5 following plant shutdown to commence refueling outage 17 (3R17). Decay heat removal was being performed with the train A Shutdown Cooling System. RCS temperature was 115 degrees Fahrenheit and RCS pressure was 90 pounds per square inch absolute (psia).

There were no structures, systems, or components inoperable that contributed to the event.

4. EVENT DESCRIPTION:

At approximately 2100 on October 6, 2103, during a scheduled visual examination of the reactor vessel BMI nozzle assemblies, white residue was discovered at the annulus region of BMI nozzle 3 (see Figure 2). The BMI visual examinations were performed using remote controlled video and photographic equipment to meet the in-service inspection requirements of 10 CFR 50.55a(g)(6)(ii)(E) and ASME Code Case N-722-1.

Upon discovery, the lead engineer for the PVNGS BMI inspection activity was notified and responded to the site to evaluate the condition. Qualified engineering personnel conducted a review of the photographic images and reviewed photographic records of past visual examinations of the Unit 3 BMI nozzle 3. At 0219 on October 7, 2013, engineering personnel determined the white residue was most likely boric acid resulting from RCS pressure boundary leakage.

At 0955 on October 7, 2013, PVNGS reported the degraded principle safety barrier to the NRC pursuant to 10 CFR 50.72 (b)(3)(ii)(A) via Event Notification 49416.

Operations personnel were notified and performed an Immediate Operability Determination which determined the reactor vessel was capable of supporting heat removal and inventory control functions in Mode 5 and Mode 6. Outage activities to defuel the Unit 3 reactor were continued as planned.

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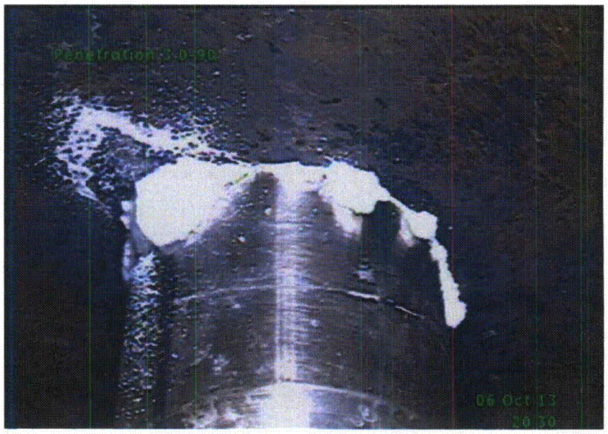
Figure 2 - Photos of Unit 3 BMI Nozzle 3

Year 2010



Year 2013

A sample of the residue was taken and analyzed for chemical and isotopic properties on October 8, 2013. Analysis confirmed the sample contained boron and lithium, with a chemical content consistent with RCS coolant. A gamma isotopic analysis was performed which found the leak to be active at the time of plant shutdown based, in part, on the presence of short-lived radionuclides in the residue. These results confirmed the residue was the result of RCS pressure boundary leakage.

On October 23, 2013, Arizona Public Service (APS) and vendor personnel completed volumetric NDE of BMI nozzle 3. The NDE included two time-of-flight diffraction ultrasonic (UT) examinations in both the axial and circumferential beam directions, a single 45 degree shear examination in the downward-looking direction, a 0 degree examination, and an eddy current examination (ECT) of the inside surface of the nozzle tube. APS, Westinghouse, and Electric Power Research Institution (EPRI) personnel independently reviewed the results of the NDE activities and concluded the source of the boron residue was a group of axially (lengthwise) oriented cracks that likely originated in the J-groove attachment weld of the nozzle tube to the vessel.

In addition to the UT and ECT examinations, a leak test was performed by pressurizing the BMI nozzle 3 annulus area on the bottom of the reactor vessel with helium and inspecting the inside of the vessel for emerging gas bubbles. Gas bubbles were observed rising from a location outside the nozzle tube in the J-groove weld at the tube interface. The leak test confirmed the origin of the leak.

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An additional phased-array UT examination was performed on the low-alloy reactor vessel shell material adjacent to the BMI nozzle 3 annulus area to detect potential loss of vessel material (wastage) resulting from exposure to boric acid. Results of this examination concluded there was no detectable loss of material.

To support the cause evaluation, approximately 0.4 cubic inch of material from the affected region of the BMI nozzle tube and J-groove weld (boat sample) was removed and sent to an off-site laboratory for metallurgical analysis. The boat sample was obtained using electrostatic discharge machining in accordance with ASME Code Section XI Article IWA-4461.

The laboratory analysis of the boat sample concluded the cracking observed in the boat sample is consistent with PWSCC. The laboratory results suggest a weld void became wetted at some point during service, which enabled the concentration of environmental constituents that promoted PWSCC in the weld and nozzle materials. There was no evidence of circumferentially-oriented cracking found in the nozzle material.

PVNGS Operations personnel evaluated parameters in Unit 1 and Unit 2 and determined there were no indications of RCS leakage in either unit. Routine surveillance testing of RCS inventory balance also did not reflect evidence of abnormal system leakage. Monitoring of trend information on containment humidity, containment atmospheric radiation levels and reactor cavity sump level did not indicate the existence of abnormal leakage from the RCS. These trend reviews are periodically performed during the course of unit operations and are expected to detect the development of safety significant RCS leakage problems. The Unit 1 and Unit 2 BMI nozzle assemblies were visually inspected in 2009 and 2010, respectively, with no abnormal indications found. Additionally, the Unit 2 BMI nozzle assemblies were visually inspected at the next available opportunity during the April 2014 refueling outage with no abnormal indications found. The Unit 1 BMI nozzle assemblies will be visually inspected during the next Unit 1 refueling outage in the Fall of 2014.

5. ASSESSMENT OF SAFETY CONSEQUENCES:

In response to operating experience, the nuclear industry addressed the potential for reactor vessel BMI nozzle assembly leakage with requirements to perform periodic visual examinations to detect leakage before safety significant failure can occur. At PVNGS, visual examinations of BMI nozzle assemblies for evidence of leakage and corrosion on adjacent ferritic items have been scheduled every other refueling outage to meet the requirements of 10 CFR 50.55a(g)(6)(ii)(E) (based on EPRI MRP-206, *Materials Reliability Program: Inspection and Evaluation Guidelines for Reactor Vessel Bottom-Mounted Nozzles in U.S. PWR Plants*, and ASME Code Case N-722-1).

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The probabilistic risk assessment performed for the EPRI MRP-206 study concluded that, for plants with 18 and 24 month fuel cycles, a bare metal visual examination every other refueling outage is sufficient to ensure the calculated change in incremental core damage frequency (ICDF) resulting from degraded BMI nozzle assemblies remains at or below 1.0E-06. The risk evaluations summarized in EPRI MRP-206 show that periodic inspections provide reasonable assurance against nozzle ejection and significant head wastage. During the investigation of this event only axial indications were found in Unit 3 BMI nozzle 3 and no loss of reactor vessel lower head shell material was found. Therefore, the Unit 3 BMI nozzle 3 leak is considered to be of low nuclear safety significance.

The Unit 3 BMI nozzle 3 leakage rate was very small with little boron residue accumulation on the lower reactor vessel head and no appreciable boron residue accumulation on the structures beneath the vessel. Ultrasonic phased-array inspection of the reactor vessel low-alloy carbon steel material in the annulus region adjacent to BMI nozzle 3 did not identify the loss of vessel material. The event did not result in the release of radioactive materials to the environment and the event did not adversely affect the safe operation of the plant or health and safety of the public.

The event did not result in a potential for a transient more severe than those analyzed in the Updated Final Safety Evaluation Report chapters 6 and 15. The condition would not have prevented the fulfillment of a safety function; and, the condition did not result in a safety system functional failure as defined by 10 CFR 50.73 (a)(2)(v).

6. CAUSE OF THE EVENT:

The cause of the pressure boundary leakage was axial cracking in a sub-surface BMI nozzle J-groove weld flaw that extended into the nozzle to a length exceeding the J-groove weld leg thickness. The laboratory analysis of the boat sample suggests the weld flaw became wetted at some point during service which enabled concentration of environmental constituents that promoted PWSCC in the weld and nozzle materials. Leakage occurred once the axial cracking extended below the J-groove weld root.

7. CORRECTIVE ACTIONS:

To restore the reactor vessel pressure boundary, a welded pad/half-nozzle repair of BMI nozzle 3 replaced the lower portion of the BMI nozzle assembly and moved the RCS pressure boundary from the J-groove weld inside the reactor vessel to a new J-groove weld outside the reactor vessel. As shown below in Figures 3 and 4, the new J-groove weld joins a new Alloy 690 bottom half-nozzle to an Alloy 52M ambient temperature temper bead weld pad deposited on the outer diameter of the reactor vessel bottom shell. The temper bead

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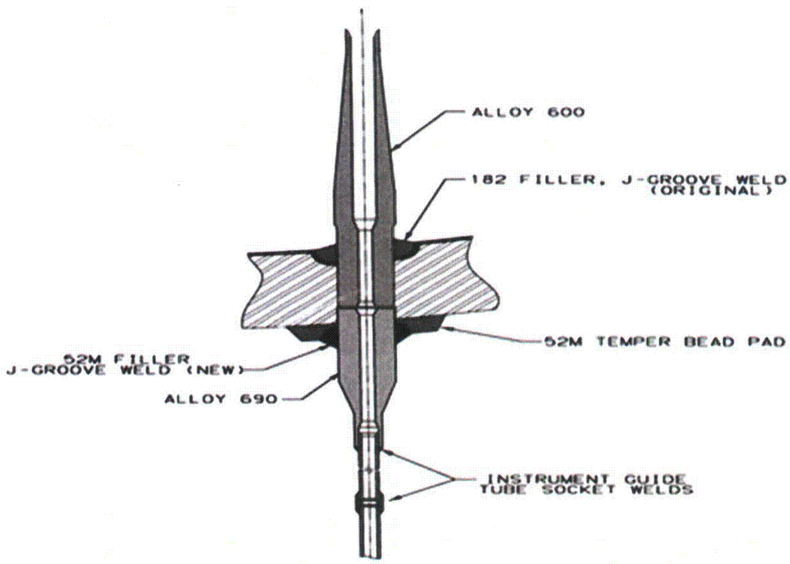
weld pad is addressed in ASME Boiler and Pressure Vessel (B & PV) Code Case N-638-4, which is conditionally accepted in Regulatory Guide 1.147, *Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1*, Revision 16. The temper bead weld pad conforms to the conditions required by Regulatory Guide 1.147 for application of Code Case N-638-4.

Figure 3 - Photo of Repaired Unit 3 BMI Nozzle 3

Figure 4 - Drawing of Repaired Unit 3 BMI Nozzle 3



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To address the extent of condition in Unit 3, visual and UT/ECT examinations were conducted on the remaining 60 Unit 3 BMI nozzle assemblies in the 3R17 outage. These examinations did not identify any additional indications of leakage or unacceptable weld indications.

To establish enhanced monitoring of the BMI nozzle assemblies, the frequency of bare metal visual inspection was changed from every other refueling outage to every refueling outage for all three units. This change was implemented with the next PVNGS refueling outage (Unit 2, spring 2014).

Based on the configuration associated with the corrective action to repair BMI nozzle 3, a Relief Request was submitted to and approved by the NRC to operate Unit 3 with the weld remnant on the remaining upper half of the BMI nozzle assembly in the reactor vessel. PVNGS Relief Request #51 proposed an alternative to the ASME Code requirements of Section XI related to axial flaw indications identified in Unit 3 BMI nozzle 3. Specifically, APS proposed a half-nozzle repair and a flaw evaluation as alternatives to the requirements for flaw removal of IWA-4421 and for flaw characterization of IWA-3300. The flaw evaluation demonstrates that the flaw will remain acceptable over the next operating cycle. In May 2014, APS submitted Relief Request 52 to the NRC, which proposes an alternative to the ASME Code requirements of Section XI related to axial flaw indications identified in a Unit 3 reactor vessel bottom mounted instrument (BMI) nozzle.

8. PREVIOUS SIMILAR EVENTS:

No similar conditions have been reported by PVNGS.