10 CFR 50.55a

aps

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102-07263-MLL/DCE
May 20, 2016

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

Dear Sirs:

Subject: **Palo Verde Nuclear Generating Station**
Unit 3
Docket No. STN 50-530
Response to Request for Additional Information Regarding Relief Request 54, Alternative to Flaw Removal

By letter number 102-07125, dated October 22, 2015 (Agencywide Documents Access and Management System [ADAMS] Accession Nos. ML15300A213, ML15300A214, ML15300A218), Arizona Public Service Company (APS) requested the U.S. Nuclear Regulatory Commission (NRC) to authorize relief from Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components*, of the *American Society of Mechanical Engineers Boiler and Pressure Vessel Code* (ASME Code) for the specific repair/replacement activity identified in Relief Request 54 for Palo Verde Nuclear Generating Station (PVNGS), Unit 3. Relief Request 54 describes the repair method for the reactor coolant pump 2A instrument nozzle and requests relief from the flaw removal and successive examinations required by ASME Code, Section XI.

The NRC staff determined that additional information is required regarding the relief request for PVNGS Unit 3 and provided a formal request for additional information (RAI) by NRC correspondence dated April 13, 2016 (ADAMS Accession No. ML16105A457). The enclosure to this letter provides the APS response to the NRC RAI.

No commitments are being made by this letter. Should you need further information regarding this submittal, please contact Michael D. Dilorenzo, Regulatory Affairs Department Leader, at (623) 393-3495.

A047
NRR

Sincerely,

JJC/DCE

Enclosure: Response to Request for Additional Information

cc:	M. L. Dapas	NRC Region IV Regional Administrator
	S. P. Lingam	NRC NRR Project Manager for PVNGS
	M. M. Watford	NRC NRR Project Manager
	C. A. Peabody	NRC Senior Resident Inspector for PVNGS

Enclosure

Response to Request for Additional Information

Summary Description

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NRC RAI – 1

The enclosure to the licensee's submittal dated October 22, 2015 included WCAP-18051-NP, Revision 0, "Palo Verde Nuclear Generation station Unit 3 Reactor Coolant Pump 2A Suction Safe End Instrument Nozzle Half-Nozzle Repair Evaluation" to support its proposed alternative. Page 1-1 of WCAP-18051-NP states that the purpose of this report is to demonstrate the acceptability of the half-nozzle repair for the flawed reactor coolant pump suction safe end instrument nozzle at Palo Verde, Unit 3 based on assessments detailed in the report. One of the assessments listed is a stress corrosion cracking assessment. However, the report does not address stress corrosion cracking of the carbon steel safe end. Please provide a stress corrosion cracking assessment including a discussion on past plant reactor coolant chemistry and future expected plant chemistry.

APS Response

The NRC Safety Evaluation for WCAP-15973-P (Reference 1) states the Westinghouse Owners Group (WOG) stress corrosion assessment may be used as the bases for concluding that existing flaws in the weld metal will not grow by stress corrosion in the base metal if appropriate plant chemistry reviews are conducted. A review of carbon steel stress corrosion, based on plant chemistry, was conducted for the RCP 2A suction safe end instrument nozzle consistent with Reference 1.

As described in the NRC Safety Evaluation for Reference 1, CE-designed reactor licensees maintain reactor coolant system (RCS) chemistry by use of the chemical and volume control system. This includes the use of ion exchangers to purify the reactor coolant. The critical parameters that need to be controlled in the RCS environment to prevent conditions favorable to

propagation of stress corrosion cracking in carbon steel are dissolved oxygen, halide (fluoride and chloride) and sulfate contaminants. The PVNGS design and licensing basis contains operational limits for RCS impurities, established in accordance with the Technical Requirements Manual (TRM) Section 3.4.101, *RCS Chemistry,* and Electric Power Research Institute (EPRI) document 3002000505, *PWR Primary Water Chemistry Guidelines*. The controls are implemented by site chemistry procedures.

Site chemistry procedures implement control parameters with appropriate action levels for RCS impurities. For example, RCS hydrogen is a control parameter with Action Level 1 (restore parameter within 7 days) if outside the range of 25 – 50 cc/kg, an Action Level 2 (restore parameter within 24 hours followed by orderly plant shutdown to less than 250 degrees Fahrenheit, if not restored), if less than 15 cc/kg, and an Action Level 3 (immediate initiation of an orderly plant shutdown to less than 250 degrees), if less than 5 cc/kg. Chemistry procedures do not allow critical reactor operation with the RCS hydrogen concentration less than 15 cc/kg without immediate corrective action.

APS reviewed RCS chemistry records for the two operating cycles prior to repair of the PVNGS Unit 3 reactor coolant pump 2A suction pressure instrument nozzle (operating cycles 17 and 18), with the following results:

- Contaminant concentrations in the reactor coolant have been maintained at levels below 150 ppb for halide ions, and 150 ppb for sulfate ions. There were no action level entries for chloride, fluoride, and sulfate for the period under review.
- Oxygen levels have been maintained below 10 ppb (typically 0 ppb) during power operation and below 100 ppb during plant startups (RCS temperature >250°F). There is no oxygen limit when the RCS temperature is below 250°F.
- RCS hydrogen overpressure was sufficient to produce RCS hydrogen concentration of > 15 cc/kg prior to criticality (hard hold point) and was maintained in a range of 25 to 50 cc/kg in Modes 1 and 2.

Because of the administrative controls which implement the PVNGS design and licensing basis for RCS chemistry, APS has determined the RCS chemistry regimen will prevent flaws in the carbon steel base metal material from growing due to stress corrosion cracking for the remainder of the Unit 3 licensed operating life, consistent with Reference 1.

<u>NRC RAI – 2</u>

Section 5 of WCAP-18051-NP describes the Loose Parts Evaluation that was performed to support RR 54. While Section 5 describes the evaluation and the results of the evaluation, it does not provide a reference to the document that contains the actual loose parts evaluation. In support of RR 53, the licensee submitted TR-FSE-15-2, Revision 1, "Palo Verde Nuclear Generation Station Unit 3 Evaluation of Potential Loose Part – Reactor Coolant Pump Instrument Nozzle Weld Fragment." The NRC staff notes that this document has not been referenced as part of RR 54. In addition, the evaluation in TRFSE-15-2, Revision 1 only addressees the current 18 month operating cycle. Please provide the loose parts evaluation described in Section 5 of WCAP-18051.

APS Response

Westinghouse document TR-FSE-15-3, Revision 0, *Palo Verde Nuclear Generating Station Unit 3 Evaluation of Potential Loose Part for Life of Plant Operation - Reactor Coolant Pump Instrument Nozzle Weld Fragment*, provided the supporting analysis for the loose parts evaluation summarized in the WCAP-18051-NP (Reference 2). Document TR-FSE-15-3 is not considered proprietary and is provided as an attachment to this enclosure.

<u>References</u>

1. NRC Letter, *Final Safety Evaluation for Topical Report WCAP-15973-P, Revision 1, "Low-Alloy Steel Component Corrosion Analysis Supporting Small-Diameter Alloy 600/690 Nozzle Repair/Replacement Program"* (ADAM Accession No. ML050180528), dated January 12, 2005

2. Westinghouse Document WCAP-18051-NP, Revision 0, *Palo Verde Nuclear Generating Station Unit 3 Reactor Coolant Pump 2A Suction Safe End Instrumentation Nozzle Half-Nozzle Repair Evaluation,* dated October 2015

Attachment to this Enclosure

Westinghouse Document TR-FSE-15-3, Revision 0, *Palo Verde Nuclear Generating Station Unit 3 Evaluation of Potential Loose Part for Life of Plant Operation - Reactor Coolant Pump Instrument Nozzle Weld Fragment*

Attachment

Westinghouse Document TR-FSE-15-3, Revision 0,
Palo Verde Nuclear Generating Station Unit 3 Evaluation of Potential Loose Part for Life of Plant Operation - Reactor Coolant Pump Instrument Nozzle Weld Fragment

TR-FSE-15-3, Rev. 0

Palo Verde Nuclear Generating Station Unit 3

Evaluation of Potential Loose Part for Life of Plant Operation - Reactor Coolant Pump Instrument Nozzle Weld Fragment

This document has been prepared and approved in accordance with Westinghouse Procedure WEC-6.1.

Author: Amanda J. Maguire*

Reviewer: Frank Ferraraccio*

Manager: Dan Sadlon for Tyler R. Upton*

**** Electronically approved records are authenticated in the electronic document management system.***

SUMMARY

During the 3R18 Palo Verde Nuclear Generating Station (PVNGS) Unit 3 refueling outage, leakage from a pressure instrument nozzle on the CE-KSB Type 101 Reactor Coolant Pump (RCP) 2A safe-end was identified. The Arizona Public Service (APS) repair strategy includes performing a half-nozzle repair. This repair involves removing a portion of the existing nozzle, inserting a replacement nozzle design in the same location, and then replacing the original pressure boundary partial penetration weld on the inside wetted surface with a weld located on the outside surface.

Because the repair process involves removing the external portion of the existing RCP nozzle and leaving a small nozzle remnant inside the existing penetration, APS has asked Westinghouse to address the possibility that fragments of the existing partial penetration weld could come loose inside the reactor coolant system (RCS) through the current planned end of plant life, which is 60 years. Westinghouse has concluded that continuation of primary water stress corrosion cracking (PWSCC) processes in the remnant Alloy 600 nozzle and J-weld is unlikely to result in liberation of loose material from the remaining in-place nozzle structure. However, this report conservatively addresses the possibility that one or more fragments of the existing partial penetration weld separates from the nozzle and weld butter and becomes a loose part inside the RCS. Westinghouse is aware of no prior industry experience where a half-nozzle repair has led to loose weld fragments.

This evaluation concluded that the postulated loose part or parts will have no adverse impact on the RCS and connected systems, structures, and components (SSCs). The SSCs continue to be capable of satisfying their design functions.

1.0 Introduction

During the 3R18 PVNGS Unit 3 refueling outage, APS identified signs of leakage (i.e., boric acid) stemming from a pressure instrument nozzle located on the suction nozzle safe-end of the 2A CE-KSB Type 101 RCP. Refer to Figure 1.

Figure 1

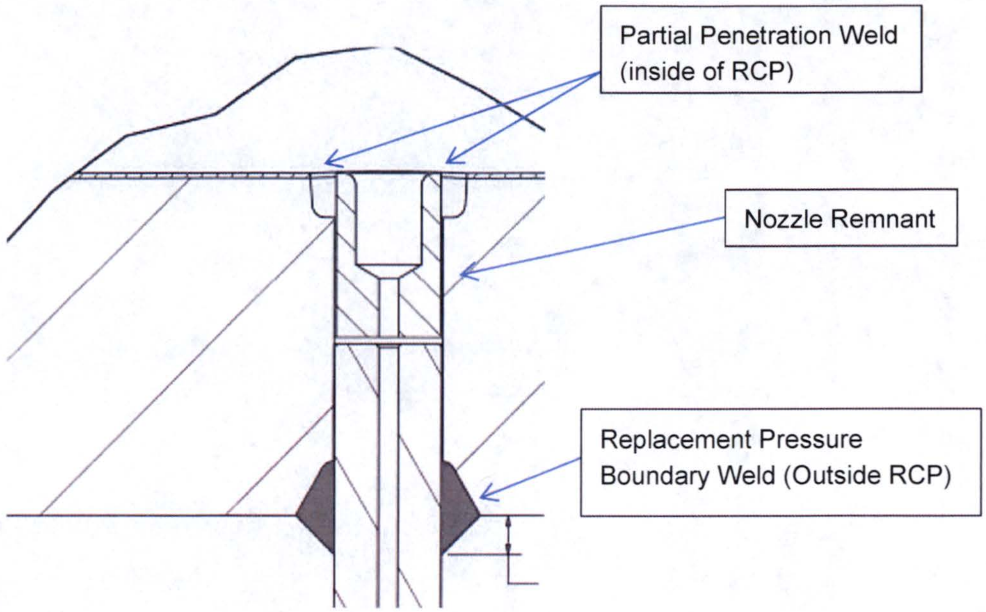
Pressure Instrument Nozzle – Showing Leakage

The APS repair effort performed a half-nozzle repair to replace the 1-inch instrument nozzle. The modification will replace the external portion of the instrument nozzle and leave in place a short segment of the original nozzle and nozzle weld at the inner surface of the pump suction safe-end (see Figure 2).

Because the repair process involved leaving a small remnant of the nozzle inside the penetration, APS asked Westinghouse to address the possibility that portions of the partial penetration weld holding the RCP nozzle remnant in place could further degrade and become a potential loose part inside the RCS before the current planned end of plant life, which is 60 years.

Figure 2

Half-Nozzle Repair Illustration



2.0 Characterization of the Postulated Loose Part

To date, the degradation mechanism responsible for the leak has not been specifically identified. The topic was discussed between APS and Westinghouse and it was agreed upon that the likely mechanism is PWSCC of the susceptible Inconel 600 nozzle and weld materials. Non-destructive examinations (NDE) were performed by APS to describe the flaw. The examinations consisted of:

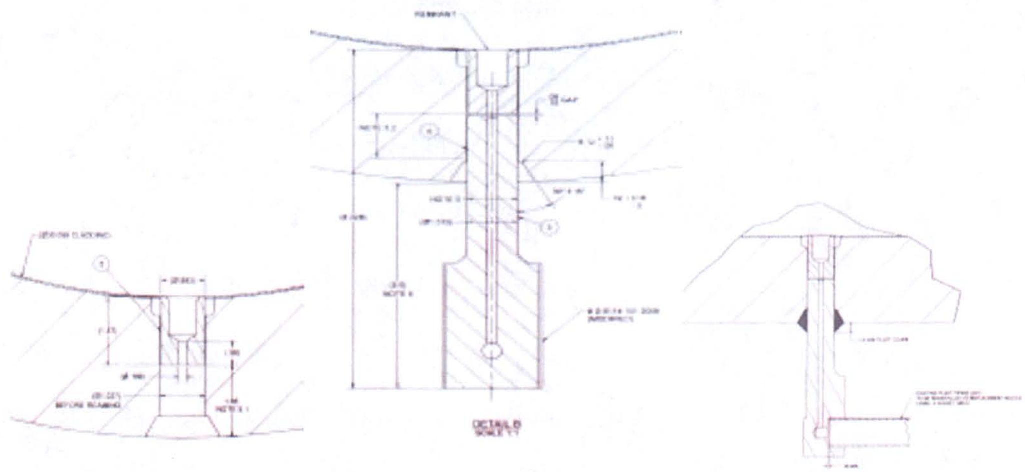
- injection of liquid penetrant at low pressure from the safe-end outside diameter (OD) into the annulus between the bore and instrument nozzle;
- visual inspection using a boroscope of the nozzle inner diameter (ID) and partial penetration weld inside the safe-end; and
- ultrasonic testing (UT) across the nozzle remnant length.

Based on those inspections, APS identified no circumferential cracks in the nozzle (from UT inspection) and no external visually discernable degradation on the surface of the partial penetration weld or the nozzle inside diameter. Thus, it was reasoned that one or more part-through-wall longitudinal cracks likely exist in the nozzle and/or the weld that caused the evident leakage in Figure 1. This weld degradation is consistent with the orientation previously observed by APS for this type of degradation mechanism (i.e., PWSCC) in instrument nozzles in the RCS piping. The likelihood of further degradation following the half-nozzle repair is described in the following discussion (Reference 8.14).

The repair installed on the PVNGS Unit 3 suction nozzle safe-end of the 2A RCP in April 2015 was a half-nozzle repair of the design shown in Figure 3 (Reference 8.11). A photograph of the original tap and nozzle is shown in Figure 1.

Figure 3

Half-Nozzle Repair Design



After the repair, the pressure boundary is moved to the weld on the outer surface of the casing. As a result of this condition, it is assumed that any leakage path from the original wetted surface of the J-weld or the Alloy 600 nozzle is retained within the original Alloy 600 nozzle and filler metal 82 (type Alloy 600 weld) J-weld that are left in place.

For the half-nozzle repair, as previously described, to create a loose part, it would require continued degradation at the remaining portion of the original Alloy 600 nozzle and at the nozzle-to-casing J-weld wetted surface. Material degradation in the nozzle was identified by APS to occur by the process of PWSCC. Other mechanisms have not been observed to produce failures in full-nozzle configurations. However, for consideration of the repair of the partial nozzle configuration, it is appropriate to consider all potential mechanisms (i.e., including those mechanisms that might affect the half-nozzle assembly differently than they can affect the full-nozzle geometry).

Potential materials degradation mechanisms in light water reactors for degradation of the reactor coolant pressure boundary (RCPB) nozzles, as identified in Reference 8.12, are:

- Embrittlement - including thermal and irradiation embrittlement.
- Corrosion and Wastage - including flow assisted corrosion and wear.
- Fatigue including low cycle fatigue and thermal fatigue.
- Stress Corrosion Cracking (SCC) - including intergranular SCC, irradiation assisted SCC, transgranular SCC, low temperature cracking, and PWSCC.

There are no known instances of embrittlement of Alloy 600 type materials at reactor coolant temperatures; so the potential for further damage to the remnant Alloy 600 nozzle and weld by embrittlement can be ruled out.

General corrosion and wastage have not been observed to occur in such nozzles nor in Alloy 600 applications in light water reactors. This is due to the very good corrosion resistance associated with the relatively high chromium content of the alloys used in these configurations. However, given the potential for leakage of pressurized water reactor (PWR) coolant water past or through the original weld, there exists the potential for corrosion interaction between the PWR borated water and the materials it would come into contact with before it reaches the new Alloy 690/filler metal 52 weld pressure boundary. This corrosion interaction is expected to be not only very small but effectively self-limiting. Only small amounts of coolant will pass through the original leakage path and, while there is some potential for this liquid to interact with the low alloy steel of the original RCP casing nozzle; because of the small amount of leakage that can pass through the path and the constrained volume of interaction, after some limited corrosion, this volume would be expected to become an inactive region. Thus, general and local corrosion are not expected to be viable mechanisms for significant further degradation of the half-nozzle repaired structure.

Fatigue can be a contributor to crack growth in a crack specimen (i.e., weld and nozzle remnant), however, given that the nozzle remnant has been separated from the instrument pipe, it no longer carries an external operating cyclic load. In addition, the thermal fatigue due to operating cycles (i.e., heat-up and cooldown) at the relative shallow depth location of the weld is expected to contribute negligible crack growth as compared to that of SCC. Therefore, fatigue in the weld and nozzle remnant is not expected to be a significant contributor to their degradation.

Of the potential mechanisms identified for material degradation, only SCC remains to potentially provide an active degradation mechanism that might give rise to the production of loose parts. The production of loose parts could be via either the release of small sections from the remnant nozzle and weld, or via the full release of the remnant nozzle section by fracture and release of the J-weld.

PWSCC is known to occur under the simultaneous action of environment and tensile stresses. These stresses can be composed of weld related residual stresses and operational stresses. The development of leakage in the original Alloy 600 nozzle/J-weld configuration indicates that some measure of PWSCC has occurred in either the weld or the tubing. However, a significant level of combined stresses is needed to drive crack propagation in either the tubing or the weld. In the following discussion, it is shown that such crack propagation is unlikely, and that there is no significant potential for PWSCC to produce the discrete and highly opened cracks that would be required to allow the release of material from the nozzle into the coolant flow stream.

A single crack propagating through either the weld or the nozzle is very unlikely to produce cracking that could result in the release of free sections of material. PWSCC cracking development results in a tortuous crack path that is mostly observed as a very "tight" array of cracking. It does result in leakage from constraining structures, but does not readily provide for the release of material from constrained sections. Cracking in type 82/182 weld metal, while occurring generally more often than in comparable wrought alloy, is influenced by the dendritic structure of the weld resulting in an even more branched fracture morphology than in the corresponding wrought alloy. Figure 4 shows

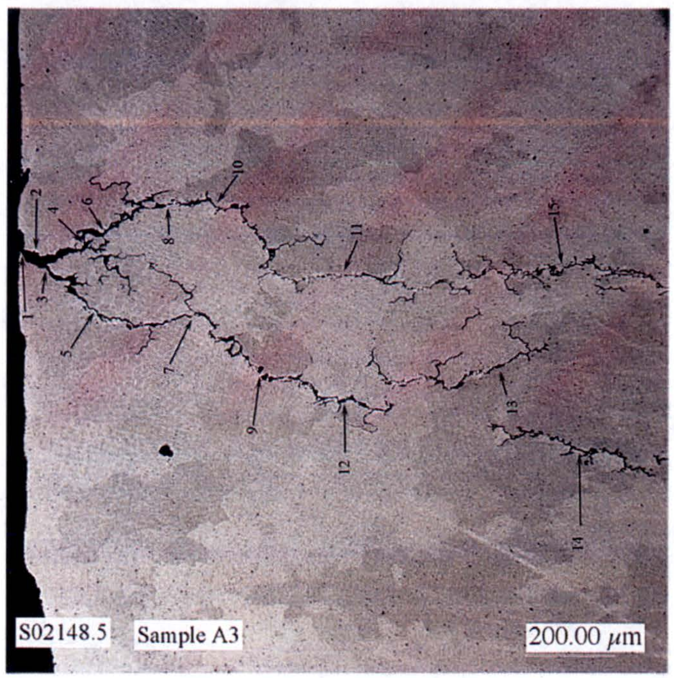
the tortuous nature of such a crack. Loss of material sections from such cracking is not observed in practice or during laboratory testing of Alloy 600 or its weld metals.

PWSCC in a single weld can result in multiple crack nucleations. Short, approximately parallel, cracks can form separately and, oriented similarly, perpendicular to the primary tensile stress direction. On the surface of a J-weld, the traces of these cracks would be oriented generally around the circumferential direction. Link up of the crack segments can occur to form long and, potentially 360 degree, cracks around the circumference of the weld. However, the tortuous three-dimensional (3-D) morphology of the cracks into the depth of the weld will mitigate against the formation of a sufficiently open, approximately planar, crack that would be required to permit the release of the retained nozzle section as a loose part. Thus, continuation of the PWSCC processes in the remnant Alloy 600 filler metal 82 J-weld is unlikely to result in liberation of the remaining nozzle section as a loose part.

Figure 4

Cross-section of a PWSCC Crack in Filler Metal 82/182 Weld

(from Schuster et al., Reference 8.13)



The potential for loss of significantly sized sections of material from the weld or the base metal is also considered to be small. Single cracks, as a result of bifurcated propagation and ductile tearing of ligaments, could result in non-adhered sections of material that would be contained within a local network of cracks. However, the surrounding material would generally constrain the material from release into the coolant stream. In some cases, where surface connected cracks might release material into the coolant stream, geometric considerations lead to the conclusion that any such particles would be small. Thus, continuation of the PWSCC processes in the remnant Alloy 600

nozzle and J-weld is unlikely to result in liberation of loose material from the remaining in-place nozzle structure.

Although the discussion concludes that it is improbable for the half-nozzle repair to lead to a loose part being introduced to the RCS, Westinghouse has conservatively addressed the possibility that one or more fragments of the existing partial penetration weld separates from the nozzle and weld butter and becomes a loose part inside the RCS. For the purposes of this evaluation, the loose part has conservatively been defined to be a relatively large weld fragment weighing approximately 0.1 pounds and having cross-sectional dimensions no greater than the partial penetration weld depth (approximately 0.9 inches) and a length equal to one-quarter of the circumference of the instrument nozzle (approximately 0.8 inches). The weld filler material (i.e., down to the butter layer) is an Inconel alloy (Alloy 182) that is compatible with the ASME SB-166 (UNS N06600) instrument nozzle material (References 8.9 and 8.10).

Although arbitrary, the dimensional characteristics of the weld fragments were judged to be bounding in terms of mass. The larger the mass of the part, the greater its potential to damage a SSC. Other smaller sizes and shapes of the weld fragments are possible and, with this taken into consideration, various weld fragment sizes and shapes have been postulated in the individual evaluations contained in this report, as applicable.

Additionally, it is noted that the postulated weld fragment would be native to the RCS and therefore, compatible with RCS chemistry.

3.0 Assumptions

The description of the loose part provided in Section 2.0 is based on an assumption that a weld fragment(s) is generated. This is an improbable occurrence, thus the approach is conservative.

All other principle assumptions made for various analyses/evaluations are identified in the individual sections that follow.

4.0 Postulated Flow Path through the Reactor Coolant System

The following defines the postulated flow path of the weld fragments, and thus the portions and subcomponents of the RCS that need to be assessed.

The weld fragments will only enter the flow stream while the RCS is in operation. During normal operation, flow through the RCS would carry the weld fragments through the suction of the 2A RCP. The weld fragments could impact components of the RCP and then be passed through the pump discharge into the cold leg. Flow in the cold leg would carry the weld fragments down the pipe. Three cold leg resistance temperature detector (RTD) thermowells are present in the flow path on the vertical half of the pipe perimeter. The weld fragments would likely be carried past the RTD thermowells. However, it is possible that the weld fragments could impact one of the thermowells.

At the end of the cold leg, the weld fragments would impact the core barrel, where flow and gravity would carry the weld fragments down the downcomer. At the bottom of the downcomer, the weld fragments would likely impact the flowskirt and could become trapped in the gap between the flowskirt and the reactor vessel (RV), or travel through the gap between the flowskirt and the RV. This would depend on the size and orientation of the fragment when it hits the flowskirt.

If the weld fragments pass through the gap between the flowskirt and the RV, or through one of the holes in the flowskirt, they would enter the reactor vessel lower plenum. In the lower plenum there is a relatively lower flow stream velocity, and the loose weld fragments could settle at the bottom of the vessel. Alternatively, the turbulent flow in this area may push the weld fragments along the bottom of the reactor vessel or lift it to where they would impact the lower internals, the lower core support plate, or the bottom of the fuel assemblies (i.e., the lower end fitting). The flow could also carry the weld fragments towards the gaps around the core and the core bypass flow paths. The starting postulated size of the weld fragment would be too large to pass through the fuel or the bypass flow gaps. Only smaller weld fragments would be able to pass through the fuel assemblies or bypass gaps.

Larger weld fragments would therefore remain in the lower reactor vessel plenum. It could be postulated that the lower plenum turbulence may cause the weld fragments to fracture into smaller pieces. In this case, the weld fragments could only pass through the core bypass path or the fuel assembly debris capture grid once divided into small enough pieces.

Other piping connected to the 2A cold leg includes the charging and safety injection lines. Both of these systems deliver to the RCS and therefore, flow would not carry the weld fragments out of the RCS through these lines. Larger weld fragments (i.e., those that cannot get past the fuel) could not be carried to the pressurizer because the pressurizer main spray lines are located on the 1A and 1B cold legs.

Similarly, weld fragments small enough to pass through the core could circulate around the RCS, through the hot leg and steam generators (SGs), to the 1A and 1B loops and into the pressurizer spray system. Weld fragments could also travel to systems connected to the RCS, such as the emergency core cooling system (ECCS), through the shutdown cooling system (SDCS) suction lines or chemical and volume control system (CVCS) through the letdown line. Only small weld fragments could reach the hot side of the RCS or connecting systems.

In the short periods of operation as the plant transitions to off-normal conditions, such as during startup or shutdown, an even less likely scenario is that the weld fragments could become loose during these off-normal conditions (i.e., when the RCS is operating with less than four pumps, such as when the 2B RCP is operating while 2A is idle). Because this condition represents a very infrequent mode of operation, these various off-normal conditions are not specifically addressed herein.

Based upon this predicted potential flow path, Sections 5.0 and 6.0 of this report specifically address the consequences that the weld fragments could have on the RCPs, piping, vessel structure and internals, fuel assemblies, control element assembly (CEA) operability, pressurizer, SGs, and connected systems.

5.0 Affected Reactor Coolant System Components

5.1 Reactor Coolant Pumps

This section discusses the evaluation of the effect of the weld fragments on the RCPs provided by Reference 8.1.

The weld fragments, or smaller pieces of a larger weld fragment, are not expected to adversely affect RCP operation. All postulated sizes of weld fragments will likely remain in the flow stream, pass through the impeller, and discharge into the reactor vessel. As the flow propels the weld fragments through the suction pipe and into the impeller, the weld fragments are prevented from entering the plenum above the impeller due to seal injection inducing a positive flow of injection water into the pump casing (via the A-gap between the impeller and diffuser). Furthermore, the radial velocity and momentum of the weld fragments within the flow stream will propel them toward the diffuser, as opposed to making an upward ninety-degree turn as they pass by the A-gap.

Due to the relatively small mass of the weld fragment(s), impact damage upon the impeller and diffuser vanes would be negligible. As a weld fragment passes through the impeller, the flow carries it past the vanes. A direct impact occurs only at the tip of the impeller cone and the leading edges of the impeller and diffuser vanes. All other impacts are postulated to hit the impeller and diffuser at a shallow angle. At most, weld fragment impacts would result in a minor peen mark, if there is a direct impact area, and superficial scratches on all other areas. Any weld fragments would pass directly into the diffuser due to the high exit velocity at the impeller. Once through the diffuser, the weld fragments may cause superficial scratches or minor impact marks on the pump casing cladding before exiting through the cold leg.

Therefore, it is concluded that the weld fragments, or smaller pieces of a larger weld fragment, passing through the RCP would not adversely impact the operation of the RCP.

5.2 RCS Cold Leg Piping

This section discusses the evaluation of the effect of the weld fragments on the RCS cold leg piping, including the tributary nozzles and RTD thermowells, provided by Reference 8.2. This evaluation is limited to the cold leg piping between the 2A RCP and the reactor vessel, and therefore, only the 2A safety injection nozzle, the charging inlet nozzle, and the RTD thermowells are evaluated.

The cold leg piping may be affected by the weld fragments since the weight of the weld fragments and the fluid velocities are great enough to cause the weld fragments to nick or gouge the clad surfaces of the piping. It is highly unlikely that the weld fragments will produce a gouge that extends down to the base metal.

The charging inlet and safety injection tributary nozzles/lines are located in the upper half of the cold leg piping and flow into the RCS. Therefore, the weld fragments do not pose a problem because they cannot enter the nozzles due to the high velocity of the flow during operating conditions. During low or no-flow conditions, the weld fragments will settle to the bottom of the

piping. Settled weld fragments would eventually make their way to the RV when higher flow rates are reached. Additionally, the charging and safety injection lines are either discharging or stagnant, eliminating the possibility of the weld fragments entering and traveling through the two respective systems.

Effects due to projectile impact on the thermowell have been previously evaluated by Westinghouse (References 8.7 and 8.8). Based on a comparison to prior evaluations, it is concluded that an impact from the limiting weld fragment of the size and mass described in Section 2.0 will not cause plastic instability in the thermowell. Hence, the pressure boundary will be maintained after such an impact. However, there remains a possibility that a thermowell could be bent or dented. If an impact occurs, monitoring data and/or alarms may indicate that a RTD has been rendered inoperable. If it is confirmed that a safety-related RTD has become inoperable, then continued plant operation is subject to technical specification requirements. If a RTD does become inoperable after startup with no associated pressure boundary breech occurring at the thermowell, it is conceivably possible that the thermowell could have sustained some damage from an impact. Performance of a visual inspection of the thermowells at the next outage would be advisable.

5.3 Reactor Vessel Structure

This section evaluates the potential consequences of the weld fragments on the structural integrity of the surveillance capsule holder, flowskirt, in-core instrumentation (ICI) nozzles, and RV structure in general, as provided in Reference 8.2.

Surveillance Capsule Holder

The weld fragments may be carried by the reactor coolant flow from the cold leg into the downcomer, and impact a reactor vessel surveillance capsule holder (RVSCH) support bracket. The RVSCH support system consists of pairs of brackets welded to either side of the RVSCH and the RV wall at several elevations.

The effect of such an impact was addressed previously for a loose bolt (Reference 8.3). Review of Reference 8.3 confirmed that the approximately 0.1 pound weight of the limiting weld fragment is less than the weight of the bolt, and therefore, the conclusions of the previous evaluation are also applicable to the weld fragment being evaluated herein. The worst case scenario previously considered was that a loose bolt could strike a RVSCH and damage an intermediate bracket system so as to render one of the two bracket sections incapable of supporting the holder. The bracket section on the other side of the holder would remain intact and maintain support for the holder at that elevation. Because the damaged bracket system will continue to provide support from the remaining section, there should be no issues with the removal of the capsule from the holder at a later date.

Flowskirt

The weld fragments would be carried by the coolant flow and, in a worst case scenario, impact the flowskirt cylinder at one of its supports. The impact forces would generate stresses in the flowskirt cylinder and its support. The effect of such an impact was previously addressed for the bolt evaluated in Reference 8.3. Since the approximately 0.1 pound weight of the weld fragment is less than the weight of the bolt, the conclusions of the previously evaluation can also be applied to the loose weld fragments.

The previous stress evaluation for the loose bolt shows that the stress due to the impact exceeds the yield strength. In the worst case scenario, one of the nine (9) supports is damaged and incapable of supporting the flowskirt but the other eight (8) supports remain intact. Additionally, the current primary stresses and fatigue usage factors on the supports during operating conditions are negligibly small. There could be a localized plastic deformation on one of the nine supports or the flow baffle; however, the flowskirt assembly will remain intact.

ICI Nozzles

The ICI nozzles are welded to the RV bottom head. It is assumed that the weld fragments are either in the downcomer between the RV and core support barrel (CSB) or at the bottom of the RV. In either scenario, the weld fragments would be lifted, swept by the coolant flow, and impact one or more ICI nozzles. The impact forces generate stresses in the ICI nozzles and weld that, when added to stresses due to other design and operating load conditions, may result in stresses exceeding ASME Code stress criteria. The effect of such an impact was previously addressed for the loose bolt evaluated in Reference 8.3. Since the approximately 0.1 pound weight of the postulated weld fragment is less than the weight of the bolt, the conclusions of the previous evaluation can also be applied to the weld fragments as a loose part.

The cited ICI nozzle/weld stress evaluation including the weld fragment impact forces considered normal operating conditions of the RCS. In addition to a weld fragment impact force, the stresses previously evaluated include, as applicable, pressure, flow loads, thermal loads, pump induced mechanical excitation of the reactor vessel, operating basis earthquake (OBE), and safe shutdown earthquake (SSE). These loads were retained along with the impact loads, but they are negligible compared to the weld fragment impact load and therefore, do not impact the conclusions of the analysis.

The evaluated case yields the maximum loads and stresses on the ICI nozzle. The ASME Code stress criteria are not satisfied at the ICI nozzle weld. The stresses are evaluated on an elastic basis. However, the ASME Code, Appendix F provides stress criteria for elastic analyses that approach the material ultimate strength (Su) and allows for some plastic deformation. Since these criteria are also exceeded, there is a reasonable expectation that application of elastic/plastic analyses would also demonstrate localized failure or possibly marginal acceptance. Exceeding ASME Code limits at the ICI weld may result in crack initiation and/or leakage.

Since the limiting size weld fragment weighs less than the bolt previously analyzed, the stresses due to impact will be significantly less and will likely meet ASME Code, Appendix F allowable values.

The previous velocities considered are sufficient to lift the weld fragments and sweep them away from the ICI nozzle, thereby preventing the weld fragments from being wedged at the ICI nozzle. Therefore only one impact of the ICI nozzle would occur and the impact would not contribute to the fatigue usage.

RV Structure

The RV may be affected by weld fragments since the weight of the weld fragments and the fluid velocities are great enough to cause these loose parts to nick or gouge the clad surfaces of the RV. It is highly unlikely that the weld fragments will produce a gouge that extends down to the base metal.

The potential damage caused by the weld fragments would have minimal and acceptable effects on the interior cladding of the RV, the flowskirt, RVSCHs, and ICI nozzle. Postulated damage would not preclude continued plant operation.

5.4 Reactor Vessel Internals

This section discusses the evaluation of the consequences of the weld fragments, either being in or passing through the reactor vessel internals (RVI), as provided by Reference 8.4. The weld fragments, being relatively small, could be carried by the flow into various portions of the RVI.

Evaluation of Weld Fragments Impacting the Core Support Barrel

Since the RCP suction nozzle is located in a RCS cold leg, the weld fragments would be carried by the RCS flow and impact the wall of the core support barrel when they exit the cold leg. The CSB is a large robust structure fabricated from 3-inch plate at the elevation of the cold leg where the weld fragments would impact. Therefore, the largest weld fragment, which is assumed to weigh approximately 0.1 pounds, is judged to impart minimal damage to the CSB. This judgment is further supported by industry experience with safety injection thermal sleeves, which are significantly heavier, having impacted the core barrel at other Combustion Engineering (CE) designed plants after coming loose.

After impacting the CSB, the weld fragments would be carried by the flow down the downcomer between the core barrel and the reactor vessel. Any impact with the core barrel during the traverse of the weld fragments in the downcomer would be less severe than the initial impact at the inlet nozzle location.

Evaluation of Small Weld Fragments

The possibility of small weld fragments being carried throughout the RCS and affecting the RVI is evaluated in the following sub-sections.

Core Support Barrel Alignment Keys and Keyways

The reactor internals alignment keys are part of the CSB assembly and provide the alignment system for the reactor internals, the RV, and the reactor vessel closure head. The alignment system consists of precise gaps between the alignment keys and their respective mating keys slots (i.e., keyways) in the interface components. There are four alignment keys at 90 degrees equally spaced azimuthally that are shrunk-fit into the CSB flange and retained in position by two radial dowel pins at each key location. The keyways are subjected to a small amount of inlet leakage flow into the RV head region that would have a tendency to keep those areas flushed of small particles if they were transported to that location. It is highly unlikely that the weld fragments could reach the alignment keys, which are located in the RV closure head region of the vessel, and if they did, and were not flushed away, the weld fragments still would not affect the alignment key function.

Hold Down Ring

The hold down ring is compressed between the top surface of the CSB flange and the underside of the upper guide structure (UGS) flange. The function of the hold down ring is to prevent movement of the reactor internals during plant operation. In order to perform that function, the hold down ring is compressed, causing the ring to rotate. This exerts a preload on the interface surfaces of the UGS and the CSB. Since the interface surfaces of the hold down ring are in compression, there is no possibility that weld fragments can enter the interface during RCS operation. Therefore, the weld fragments could not affect the hold down ring function.

Core Barrel Flange to Reactor Vessel Seating Surface Annulus

It is highly unlikely that weld fragments will move into this annular space during service since there is only a small amount of inlet coolant leakage flow to transport the fragments through the CSB alignment keys and into this region of the RV head.

Annular Space between the Core Shroud and Core Support Barrel Inside Diameter

It is possible that the weld fragments could enter the annular space between the core shroud and CSB inside diameter. There is a small amount of flow in this annular space to cool the backside of the core shroud and inside diameter of the CSB in this region. However, the weld fragments would tend to settle out in a low flow area and would not have an adverse effect on any of the large components in this region.

Snubbers

There are six (6) snubbers spaced 60 degrees apart between the lower end of the CSB and the RV. These components have a tongue and groove arrangement with a small gap on each side. Weld fragments, if transported to these gaps, would be immediately flushed out, due to the high velocity flow. Therefore, the weld fragments would have no effect on the function of the snubbers.

The Upper Guide Structure Support Barrel Assembly and CEA Shroud Assembly

There are no close fits in this region that would be impaired by the presence of weld fragments. Weld fragments, if deposited on the upper surface of the support plate of the UGS support barrel assembly, would most likely remain in position due to the low velocity flows in that region. However, if transported by the reactor coolant flow, the weld fragments would not impair the functions of the UGS support barrel assembly and CEA shroud assembly.

Other RVI Components

1. The interface between the guide post of the fuel assembly upper end fitting and the UGS tubes in the UGS support barrel assembly is a precise interface, for both the standard fuel design and Next Generation Fuel (NGF). The coolant flow exiting from the fuel assembly guide tubes will tend to flush this annular space of the weld fragments, but even if it did not, there will be no loss of function at the fuel-to-UGS tube interface.
2. At the periphery of the fuel alignment plate there are four keyways spaced 90 degrees apart that form a precise interface gap with the shims on the guide lugs. Weld fragments small enough to fit into these gaps would most likely be flushed out by the coolant flow, but even if they remained, they would cause no loss of function to this interface.

5.5 Fuel

This section summarizes the evaluation of the potential impact of the weld fragments on fuel performance (Reference 8.5).

Since the weld fragment mitigation features are essentially the same for the CE16STD GUARDIAN™[1] grid design and the CE16NGF GUARDIAN grid design, the evaluation is applicable to PVNGS Unit 3 cores containing either fuel product.

The weld fragments as described in Section 2.0 have been evaluated. Additionally, it has been postulated that some cladding material above or adjacent to the weld may also break off with the weld fragment. The composition, size, and shape of the weld fragment are not precisely known, so a mixture of Inconel 600 and stainless steel has been assumed specifically for the evaluation of the fuel. Given that the weld fragments evaluated below are assumed to be a mixture of Inconel alloys and stainless steel, there will be no metallurgical concerns with the presence of these materials within the reactor core region.

<u>Passage of parts through the ICI guide path</u>

The funnel on the fuel assembly lower end fitting (LEF) in non-ICI locations represents a very small percentage of the flow, so it is unlikely that the weld fragments would enter the funnel. If the weld fragments enter the funnel and are larger than the through-hole diameter, they would be caught and would fall out at the end of cycle or become wedged. If an unidentified wedged piece of weld fragment was in an assembly that is moved to an ICI location, the ICI could not be inserted prior to operation.

A weld fragment smaller than the minimum entrance diameter and larger than the exit hole would be retained in the guide tube during operation and may fall out at the end of the cycle or remain in the instrument tube. In this unlikely case, the weld fragment could interfere with the insertion of an ICI in a subsequent cycle. A weld fragment smaller than the exit hole would enter the flow stream above the fuel assemblies.

At ICI locations, there is a very small gap between the LEF funnel and the ICI nozzle, so only very small weld fragments could enter this gap. A second location where the weld fragments could enter an ICI location is at the interface of the instrument nozzles. However, the tight radial clearance within the instrument nozzles would likely capture any weld fragment that may enter at that location. Therefore, there are no operational consequences of weld fragments entering the ICI flow paths. There is some risk of the ICI binding during withdrawal, but this risk is very small given that a weld fragment of a very specific size would have to be wedged between the ICI and wall.

<u>Passage to and through the Lower End Fitting and GUARDIAN grid</u>

The LEF contains flow holes each with a diameter that may or may not prevent weld fragments from passing. However, the largest circular size that can pass through either the CE16STD or CE16NGF fuel GUARDIAN grid is considerably smaller than what can pass through the LEF flow holes.

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Although weld fragments would likely be held against the GUARDIAN grid and/or LEF for both CE16STD and CE16NGF fuel designs, and would evenly distribute at one axial plane, it is conservatively assumed that all of the weld fragments are caught in the LEF/GUARDIAN grid of one assembly (out of 241) for the purposes of evaluating flow starvation upstream of the beginning of the heated length to negatively impact departure from nucleate boiling (DNB). Based upon DNB test results for a four foot heated length, blockage at the inlet greater than the weld fragment size had no measurable impact on DNB performance at nominal conditions. Hence, the weld fragment size is bounded by these results.

For the weld fragment to impact fuel performance due to fretting wear, a small piece of weld fragment must pass through the GUARDIAN grid or around the LEF. For weld fragments to cause fretting wear, they need to be long enough so that the one end is trapped in a grid feature and the other end is free to vibrate due to coolant flow with a hammering or rubbing action on the cladding. Such a weld fragment that was able to pass through this region of the fuel assembly would likely not be of a configuration conducive to cause fragment fretting. However, the material of the weld fragment is harder than the ZIRLO®[2] cladding, so there exists some risk that a small number of fragment fretting failures could occur.

The operating history of the GUARDIAN grid has been excellent. Only one confirmed fretting-related leaker is known to have occurred (Waterford-3 Cycle 19) out of over 7200 16x16 assemblies. If the weld fragment is small and light enough to get through or around the LEF and GUARDIAN grid, it would either be carried through the grids and end fittings and exit the fuel assembly, or be captured in the grids above the GUARDIAN grid. It is unlikely that weld fragments caught in the mid grids above the GUARDIAN grid would be sufficiently long enough to cause fretting failure, but there is always some risk. Although a small risk, weld fragments in the RCS can result in a leaking rod. Therefore, continuous monitoring of the coolant activity to check for the presence of any new grid-to-rod fretting (GTRF) leakers is recommended.

<u>Bypass Flow</u>

The flow area corresponding to the holes in the lower support structure cylinder bypass flow region may result in large enough flow velocities to lift the weld fragments and transport them through the shroud cooling water passages. The much smaller velocities further downstream and the tight turns and small annular gaps would inhibit weld fragment passage beyond this region. Within the core support barrel and core shroud annulus, the flow area will result in very low flow velocities; hence, the weld fragments would be expected to settle to the bottom of the bypass region. Therefore, it is concluded that there is low probability that the postulated weld fragments will pass from the lower reactor vessel environs through the core barrel/core shroud annulus and into the outlet regions of the reactor vessel.

5.6 CEA Operability

This section discusses the evaluation of the effect of the weld fragments on CEA operability provided by Reference 8.5.

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The only viable path for small weld fragments to make it through the GUARDIAN grid and enter the CEA guide path is if they are in the immediate vicinity of the bleed hole and the cooling hole in an outer guide tube. Although extremely unlikely, any weld fragments that might enter the CEA guide path would likely either drop to a benign location at the bottom of the guide tube or be swept up the guide tube and into the CEA shroud. Weld fragments would not be expected to impede the operation of the CEA by being wedged between the CEA and the guide tube based on the small size required to enter the CEA guide path. In the very unlikely event that a loose weld fragment did impede the motion of the CEA, it is most likely to occur when the CEA finger is within the reduced clearance region of the guide tube dashpot. In this event, maneuvering of the CEA is expected to clear the obstruction, based on prior instances of obstructions in the dashpot regions.

6.0 Remaining SSCs in the RCS and Connected and Auxiliary Systems

The following subsections address the portions and major components of the RCS that will be affected by weld fragments that are sufficiently small to pass through the fuel and core bypass, and have access to downstream systems connected to the RCS (i.e., not otherwise addressed in Section 5.0).

Other systems connected to the 2A cold leg includes the charging and safety injection lines. Both of these systems deliver to the RCS and therefore, flow would not carry the initial weld fragments out of the RCS through these lines. As the weld fragments pass through the 2A RCP, the cold leg, and into the RV lower plenum, the fragments could break into much smaller pieces, which will ultimately pass into and remain in circulation within the RCS until they are drawn into the design filtration system of the CVCS or settle elsewhere in an auxiliary system. The following descriptions review the possibility of such effects on the individual auxiliary systems.

6.1 Remaining RCS Components

Upper RVI

All RVI (both upper and lower) are evaluated in Section 5.4.

RCS Hot Legs

The impact of weld fragments small enough to pass through the fuel on the RCS hot legs is bounded by the evaluation for the cold leg, documented in Section 5.2, as well as by the review documented in Reference 8.6.

Pressurizer

During normal operation, the pressurizer receives a continuous bypass spray flow from the cold leg and a corresponding continuous flow from the pressurizer to the hot leg. The smaller weld fragments would most likely remain in the main RCS flow path if they pass through the core. It is possible that if the pressurizer main spray is cycled and weld fragments are in the appropriate cold leg, that they could be drawn into the pressurizer spray line. The larger postulated weld fragments could not be carried to the pressurizer because the pressurizer main spray lines are located on the

1A and 1B cold legs. Only the weld fragments small enough to pass through the fuel assemblies could possibly reach the pressurizer.

There will be no consequence on the main spray piping and valves, or on main spray performance. The main spray valves are 3-inch full-port globe valves and, as such, have no likelihood of blockage due to the small weld fragments. The warm-up/bypass valves are ¾-inch globe valves understood to be throttled to a fairly open position. Even in the throttled position, the small weld fragments have a low likelihood of blockage in the valve.

<u>Pressurizer Spray Nozzle</u>

The pressurizer spray nozzle outlet is a hollow core design attached to the 3-inch nominal RCS spray piping. The flow path through the nozzle is large enough that the weld fragments would flow through the nozzle and into the pressurizer.

<u>Pressurizer Heaters</u>

The pressurizer heaters consist of cylindrical heating elements inserted in the bottom of the pressurizer and are supported by two support plates inside the lower region of the pressurizer. Should any weld fragments make it into the pressurizer, they would settle to the lower region of the pressurizer, settling onto the support plates or lower head. These weld fragments would either remain in place with no consequence or be swept back into circulation through the pressurizer surge line during a normal surge/swell. There is a small gap between the pressurizer heaters and the two horizontal support plates. As previously described, since the likelihood of weld fragments entering the pressurizer is very low, it is considered further unlikely that weld fragments would settle directly within this gap. The downflow out of the pressurizer, which would occur during a spray event, would tend to draw any weld fragments around the top support plate, making it considerably less likely that any weld fragments would land on the lower support plate. Thus, it is considered highly unlikely that any weld fragment would obstruct the gap and thus, have an appreciable effect on thermal growth or heat transfer efficiency of the heaters.

<u>Pressurizer Surge Line</u>

The pressurizer surge line connects to hot leg 1. The surge line diameter is sufficient that the smaller weld fragments would not affect flow along the surge line. The surge line material is stainless steel, which is the same material as the RCS piping cladding. Therefore, there would be no consequence to the surge line piping due to the presence of the weld fragments in the coolant. Weld fragments entering through the spray line may settle in the lower head and not leave the pressurizer.

<u>Steam Generators</u>

During normal operation, the weld fragments could only reach the SG if they were small enough to pass through the fuel assemblies.

Reference 8.7 evaluated loose bolts, nuts, and washers of significantly more size and mass than any weld fragment that could pass through the fuel assemblies, and concluded they would not adversely impact the function of the SGs operationally or as part of the RCS pressure boundary. Therefore, the potential weld fragments in the system would not adversely impact the SGs from performing their design function.

6.2 Connected and Auxiliary Systems

The following subsections address the systems connected to the RCS. The conclusions of this section are based upon the prior evaluation of similar debris evaluated in Reference 8.7.

Safety Injection, Containment Spray, and Shutdown Cooling Systems

The safety injection system (SIS) and containment spray system (CSS), including the ECCS pumps and safety injection tanks (SITs), inject borated water into the RCS in the event of a loss of coolant accident (LOCA). This provides cooling to limit core damage and fission product release, and ensures adequate shutdown margin. The SIS also provides continuous long-term, post-accident cooling of the core by recirculation of borated water that collects in the containment sump.

The shutdown cooling system (SDCS) is used in conjunction with the main steam and main or auxiliary feedwater systems to reduce the temperature of the RCS in post-shutdown periods from normal operating temperature to the refueling temperature.

The piping connected to the 2A cold leg includes one SIS injection nozzle. The SIS delivers flow to the RCS at this location and therefore, flow would not carry the weld fragments out of the RCS through these lines.

The shutdown cooling suction lines are connected to the hot legs of the RCS and this is the only credible flow path for the weld fragment to enter the SIS, CSS, or SDCS. Only weld fragments small enough to pass through the fuel assemblies could reach the shutdown cooling suction line on the hot leg. Weld fragments of this size would not adversely impact the ability of the SIS, CSS, or SDCS to fulfill their design functions.

Chemical and Volume Control System

The CVCS controls the purity, volume, and boric acid content of the reactor coolant. The coolant purity level in the RCS is controlled by continuous purification of a bypass stream of reactor coolant. Water removed from the RCS is cooled in the regenerative heat exchanger. From there, the coolant flows to the letdown heat exchanger and then through a filter and demineralizer where corrosion and fission products are removed. The filtered coolant is then sprayed into the volume control tank and returned by the charging pumps to the regenerative heat exchanger where it is heated prior to return to the RCS.

The letdown system components are not explicitly evaluated, as the purification system is fulfilling its design function. This is due, in part, to the fact that removal of system debris is within the design basis of the letdown system components. The larger weld fragments cannot pass the fuel assemblies GUARDIAN grid; therefore, the larger weld fragments will not be introduced into the letdown system, which connects to the RCS in the suction leg connected to the bottom of the cold side of the steam generator outlet.

In the event that the weld fragments break apart to the point of being able to pass through the fuel it is most likely that the weld fragments will be small enough and the flow sufficient that the weld fragments would flow past the letdown nozzle and not be introduced into the letdown system. In the

event that they are introduced into the letdown system, the weld fragments would pass through the letdown system to the system filters where they would be retained.

Regarding the charging function of the CVCS, the charging pumps only draw inventory from the volume control tank (VCT) and the refueling water storage tank (RWST). The weld fragments will not enter either of these suction sources.

Seal injection water supplied to the RCPs is drawn from the VCT by the charging pumps. Consequently, per the explanation in the prior paragraph, the weld fragments cannot migrate into the RCP seal packages by seal injection.

It is unlikely the weld fragments will enter the CVCS. However, if they do, they would be retained in the system filter and not adversely impact the ability of the CVCS to fulfill its design functions.

<u>Spent Fuel Pool</u>

The spent fuel pool (SFP) is isolated from the refueling cavity and the RCS during normal operation. If the weld fragments came loose during normal operation, they would travel through the RCS and could reach the fuel. It is possible that weld fragments could become caught in the fuel assembly GUARDIAN grid.

During the following refueling cycle, the weld fragments could transfer with the fuel assemblies to the SFP. If the weld fragments entered the SFP during refueling operations (i.e., if they were to fall from the GUARDIAN Grid), they would settle to the floor of the pool and remain there. The weld fragments on the SFP floor would not migrate into the spent pool cooling system due to the relatively high location of the cooling system suction inlet above the pool bottom surface.

Based on this evaluation, the potential presence of the weld fragments in the RCS does not adversely impact the capability of the pool cooling system to fulfill its design function.

7.0 Conclusions

During the 3R18 PVNGS Unit 3 refueling outage, APS identified signs of leakage (i.e., boric acid) coming from a 1-inch pressure instrument nozzle on the 2A loop RCP. The APS repair strategy included performing a half-nozzle repair, which would involve removing a portion of the existing nozzle, inserting a replacement nozzle design into the same location, and then replacing the original pressure boundary partial penetration weld (on the inside wetted surface) with a weld located on the outside surface of the pump safe-end.

Because the repair process involves leaving a small remnant of the nozzle inside the existing penetration, APS asked Westinghouse to address the possibility that fragments of the existing partial penetration weld could come loose inside the RCS through the current planned end of plant life, which is 60 years. Westinghouse and APS postulated, based on NDE performed to describe the flaws, that the crack(s) on the nozzle and/or weld are part-through-wall in the axial direction with no evidence of circumferential cracks. This is consistent with the orientation previously observed by APS for this type of degradation mechanism (i.e., PWSCC) in instrument nozzles in the hot leg.

The remnant Inconel Alloy 600 instrument nozzle (approximately 1.5 inches in length) is recessed inside the safe-end bore. It remains constrained by a relatively tight radial clearance between the bore and the nozzle. For the half-nozzle repair to create a loose part, it would require continued degradation at the remaining portion of the original Alloy 600 nozzle and at the nozzle-to-casting J-weld wetted surface. Based on the discussion in Section 2.0, only PWSCC was identified as a potential active mechanism for material degradation that could potentially give rise to the production of loose parts. However, based on the tortuous, tight array of cracking created by PWSCC, as well as the fact that any non-adhered sections of material would be constrained from release by the surrounding material, it has been determined that continuation of PWSCC processes in the remnant Alloy 600 nozzle and J-weld is unlikely to result in liberation of loose material from the remaining in-place nozzle structure.

However, although Westinghouse has concluded it is very unlikely that a loose part will be released from the Alloy 600 nozzle and/or J-weld, this report conservatively addresses the possibility that one or more fragments of the existing partial penetration weld separates from the nozzle and weld butter and becomes a loose part inside the RCS. Based on this assumption, a conservatively sized fragment of weld was assumed to weigh approximately 0.1 pounds and have dimensions no greater than the partial penetration weld thickness at its cross-section, and a length of one-quarter of the circumference around the instrument nozzle.

Westinghouse evaluated the structural and functional impacts of the loose weld fragment(s) on affected SSCs. Engineering judgments were applied and prior PVNGS loose parts evaluation results were taken into consideration. The evaluation considered that although the aforementioned fragment represents one possible form of the loose part, it is possible that smaller fragments of different sizes, shapes, and weights could be released, or created. Additional smaller fragments are possible, for example, if a weld fragment were to make contact with a high-velocity RCP impeller blade, or perhaps make high-speed contact with the core support barrel.

The evaluation concluded that the postulated loose parts will have no adverse impact on the RCS and connected SSCs through the current planned end of plant life. The evaluation addressed potential impacts to various SSCs where the loose parts might travel. This included the RCPs, the main coolant piping, the reactor vessel and its internals, the fuel, the pressurizer, steam generators, as well as other systems attached to the RCS, including the spent fuel pool. It was determined that all impacted SSCs would continue to be capable of satisfying their design functions.

8.0 References

8.1 LTR-KUAE-15-016, Revision 0, "Loose Parts Evaluation – Suction Nozzle Pressure Tap Reactor Coolant Pump, Palo Verde Unit 3, S/N 1111-2A," April 2015.

8.2 LTR-MRCDA-15-35, Revision 1, "Transmittal of MRCDA-I Input for the Analysis of Loose RCP Suction Nozzle Weld Fragments at PVNGS Unit 3 Considering Continued Operation," July 2015.

8.3 LTR-MRCDA-10-79, Revision 0, "Structural Evaluation of the Palo Verde Unit 1 Reactor Vessel Surveillance Holders due to a Loose Bolt Impact," May 2010.

8.4 LTR-RIDA-15-73, Revision 0 "Palo Verde Nuclear Generating Station Unit 3 Evaluation of Possible Reactor Coolant Pump Suction Nozzle Weld Fragment Loose Parts on the Reactor Vessel Internals," April 2015.

8.5 CE-15-207, Revision 0, "PVNGS RCP Suction Nozzle Remnant Loose Part Evaluation for the Fuel and Core Components," April 2015.

8.6 LTR-RC-14-55, Revision 0, "Palo Verde Nuclear Generating Station Unit 3 Review of Bounding Evaluations for Loose Weld Material," December 2014.

8.7 DAR-SEE-II-10-3, Revision 0, "Palo Verde Nuclear Generating Station Unit 1 Evaluation of Missing Bolt Shanks, Nuts and Washers," May 2010.

8.8 DAR-SEE-II-8-12, Revision 0, "Disposition of Postulated Foreign Material in Palo Verde Nuclear Generating Station Units 1, 2, 3 Nuclear Steam Supply System Originating from Reactor Coolant Pump Wedge Assemblies," December 2008.

8.9 LTR-KUAE-15-017, Revision 1.0, "Documentation of References applicable to Palo Verde Unit 3 Pressure Tap Weld Details," April 2015.

8.10 STD-009-0009, Revision 2, "Coolant Pumps Weld Joint Identification and Fabrication Requirements."

8.11 E-14473-220-001, Revision 0, "Pump Casing – A Pressure Tap Nozzle Modification Assembly."

8.12 "EPRI MRP-156 Pressurized Water Reactor Issue Management Table PWR-IMT Consequence of Failure," EPRI Palo Alto, California (2005).

8.13 "Status of the International Cooperative Program for Inspection of Nickel Alloy Components-PINC," G. J. Schuster et al in "Proceedings of 6th International Conference on NDE in Relation to Structural Integrity for Nuclear and Pressurized Components," (2007).

8.14 MCOE-LTR-15-84, Revision 0, "Consideration for Loose Parts after Drain Line Half Nozzle Repair at Palo Verde Unit 3," July 28, 2015.