UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

June 23, 2016

Mr. Randall K. Edington
Executive Vice President Nuclear/
Chief Nuclear Officer
Mail Station 7602
Arizona Public Service Company
P.O. Box 52034
Phoenix, AZ 85072-2034

SUBJECT: PALO VERDE NUCLEAR GENERATING STATION, UNIT 3 – RELIEF REQUEST 54 TO APPROVE AN ALTERNATIVE TO FLAW REMOVAL FOR REACTOR COOLANT PUMP 2A SUCTION PRESSURE INSTRUMENT NOZZLE (CAC NO. MF6806)

Dear Mr. Edington:

By letter dated October 22, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15300A218), as supplemented by letter dated May 20, 2016 (ADAMS Accession No. ML16147A092), Arizona Public Service Company (the licensee) submitted Relief Request (RR) 54 for the use of an alternative to certain requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," at Palo Verde Nuclear Generating Station (PVNGS), Unit 3. The request for relief would allow a completed ASME Code-compliant half-nozzle repair and a flaw evaluation as an alternative to the requirements of ASME Code, Section XI, IWA-4421, for flaw removal, and ASME Code, Section XI, IWB-2420, for successive examinations, for the pressure instrument nozzle attached to the safe end of the suction side of reactor coolant pump (RCP) 2A.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) paragraph 50.55a(z)(1), the licensee proposed a completed half-nozzle repair and flaw evaluation as an alternative repair for the degraded pressure instrument nozzle attached to the safe end of the suction side of RCP 2A in lieu of flaw removal in accordance with the ASME Code, Section XI, IWA-4421. In addition, the alternative proposes to forgo successive examinations of the remaining flaw, as required by ASME Code, Section XI, IWB-2420. The licensee requested to use the proposed alternative on the basis that the alternative provides an acceptable level of quality and safety.

The U.S. Nuclear Regulatory Commission (NRC) staff reviewed the licensee's submittal and determined that the proposed repair will restore the primary system pressure boundary and provide reasonable assurance that the structural integrity of the repaired pressure instrument nozzle will be maintained for remainder of the third 10-year inservice inspection (ISI) interval. The NRC staff further concludes that the proposed alternative for the repair of the subject pressure instrument nozzle provides an acceptable level of quality and safety and that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(1). Therefore, the NRC authorizes the use of RR 54 at PVNGS, Unit 3 until the end of the Unit 3 third ISI interval, which is currently scheduled to end on January 10, 2018.

All other ASME Code, Section XI requirements for which relief was not specifically requested and approved in the subject request for relief remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

If you have any questions, please contact the Project Manager, Siva P. Lingam, at 301-415-1564 or via e-mail at Siva.Lingam@nrc.gov.

Sincerely,

Robert J. Pascarelli, Chief
Plant Licensing Branch IV-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. STN 50-530

Enclosure:
Safety Evaluation

cc w/encl: Distribution via ListServ

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELIEF REQUEST 54 REGARDING PROPOSED ALTERNATIVE FOR REPAIR OF

INSTRUMENT NOZZLE AT REACTOR COOLANT PUMP 2A

ARIZONA PUBLIC SERVICE COMPANY

PALO VERDE NUCLEAR GENERATING STATION, UNIT 3

DOCKET NO. 50-530

1.0 INTRODUCTION

By letter dated October 22, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15300A218), as supplemented by letter dated May 20, 2016 (ADAMS Accession No. ML16147A092), Arizona Public Service Company (APS, the licensee) submitted Relief Request (RR) 54 for the use of an alternative to certain requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," at Palo Verde Nuclear Generating Station (PVNGS), Unit 3.

The request for relief would allow a completed ASME Code-compliant half-nozzle repair and a flaw evaluation as an alternative to the requirements of ASME Code, Section XI, IWA-4421, for flaw removal, and ASME Code, Section XI, IWB-2420, for successive examinations, for the pressure instrument nozzle attached to the safe end of the suction side of reactor coolant pump (RCP) 2A.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations*, Part 50 (10 CFR 50), paragraph 50.55a(z)(1), the licensee proposed a completed half-nozzle repair and flaw evaluation as an alternative repair for the degraded pressure instrument nozzle attached to the safe end of the suction side of RCP 2A in lieu of flaw removal in accordance with the ASME Code, Section XI, IWA-4421. In addition, the alternative proposes to forgo successive examinations of the remaining flaw, as required by ASME Code, Section XI, IWB-2420. The licensee requested to use the proposed alternative on the basis that the alternative provides an acceptable level of quality and safety.

2.0 REGULATORY EVALUATION

Pursuant to 10 CFR 50.55a(g)(4), ASME Code Class 1, 2, and 3 components (including supports) shall meet the requirements, except the design and access provisions and the pre-service examination requirements, set forth in the ASME Code, Section XI, "Rules for

Inservice Inspection of Nuclear Power Plant Components," to the extent practical within the limitations of design, geometry, and materials of construction of the components.

Paragraph 10 CFR 50.55a(z), "Alternatives to codes and standards requirements," states, in part, that alternatives to the requirements of 10 CFR 50.55a(g) may be used, when authorized by the U.S. Nuclear Regulatory Commission (NRC), if (1) the proposed alternative would provide an acceptable level of quality and safety or (2) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Based on the above, and subject to the following technical evaluation, the NRC staff finds that regulatory authority exists for the licensee to request and the Commission to authorize the alternative requested by the licensee.

3.0 TECHNICAL EVALUATION

3.1 ASME Code Component Affected

The affected component is ASME Class 1, RCP 2A suction pressure instrument nozzle in the cold leg. The nozzle is attached to the safe end of the RCP 2A. The examination category of the subject nozzle is ASME Code Case N-722-1, Class 1 pressurized-water reactor components containing Alloy 600/82/182. The examination Code Item Number is B15.205, cold-leg instrument connections of Code Case N-722-1. In addition, the ASME Code, Section XI, Table IWB-2500-1, Examination Category B-P applies to the suction pressure instrument nozzle with respect to the pressure test program.

3.2 Applicable Code Edition and Addenda

PVNGS, Unit 3 is currently in its third 10-year inservice inspection (ISI) interval, which ends on January 10, 2018. The Code of record for the third 10-year ISI interval is the ASME Code, Section XI, 2001 Edition through 2003 Addenda.

3.3 Applicable Code Requirements

Article IWA-4421 of the ASME Code, Section XI, requires defects to be removed or mitigated.

Article IWB-2420 of the ASME Code, Section XI, requires that areas containing flaws or relevant indications in components that have been accepted for continued service, shall be reexamined during the next three inspection periods.

3.4 Reason for the Request

During the boric acid program walkdowns at the beginning of the 18^{th} refueling outage (denoted as 3R18) in spring 2015, the licensee detected leakage at the pressure instrument nozzle that is attached to the safe end on the suction side of RCP 2A. The licensee inspected the extent of condition of the remaining Unit 3 reactor coolant system (RCS) cold-leg instrument nozzles as required by ASME Code Case N-722-1 and found no other RCS pressure boundary leakage.

The licensee stated that repair of the original RCP 2A instrument nozzle and associated J-groove weld would require removal of the RCP internals. Removing the RCP internals would allow access to the internal surface of the reactor coolant piping in order to grind out the attachment J-groove weld and repair or replace the remaining nozzle. The licensee further stated that such an activity would result in high radiation exposure to the personnel involved and present the additional risk of introducing foreign material into the RCS and reactor core. Additionally, ultrasonic testing examination of the remnant J-groove weld was not feasible because of its configuration and the restrictive access associated with the small bore of the instrument nozzle internal orifice.

During 3R18, an ASME Code-compliant repair was implemented such that the original RCP 2A instrument nozzle and J-groove weld no longer performed a pressure boundary function. The licensee requested an alternative, in RR 53, to the ASME Code requirements in IWA-4421 for removal of the flaw in the original J-groove weld. On April 24, 2015 (ADAMS Accession No. ML15117A042), the NRC verbally authorized RR 53 at PVNGS, Unit 3 for one fuel cycle to the 19th refueling outage. The final NRC approval of RR 53 and safety evaluation was issued on September 15, 2015 (ADAMS Accession No. ML15238B661). RR 53 did not address successive examinations which, in accordance with IWB-2420(b), would be required during the next three inspection periods.

The licensee proposes the completed ASME Code-compliant half-nozzle repair and a flaw evaluation as an alternative to the IWA-4421 requirements for flaw removal and IWB-2420 successive examination requirements. The current proposed alternative, RR 54, seeks approval of the completed repair for the remainder of the PVNGS, Unit 3 third 10-year ISI interval, which expires on January 10, 2018.

3.5 Proposed Alternative and Basis for Use

In its letter dated October 22, 2015, the licensee stated, in part, that:

> APS is proposing an alternative in accordance with 10 CFR 50.55a(z)(1) that provides an acceptable level of quality and safety. The alternative consists of two main elements:
>
> a. ASME Code-compliant Half-Nozzle Repair
>
> The half-nozzle repair is an industry standard, ASME Code-compliant repair method that attaches a new [primary water stress corrosion cracking (PWSCC)] resistant Alloy 690 half-nozzle using an external Alloy 52M partial penetration J-groove weld. The repair relocated the RCS pressure boundary weld from the instrument nozzle J-groove weld on the interior wall of the RCP suction nozzle safe end to the new exterior J-groove weld on the exterior wall. The half-nozzle repair of the RCP 2A suction nozzle penetration will not remove the flaws in the remnant J-groove weld or remnant Alloy 600 nozzle material near this weld. The half-nozzle repair is shown in Figure 1-1, *RCP Instrumentation Nozzle Repair Schematic*, of Attachments 1 [Westinghouse Report, 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," October 2015] and 2 [Palo Verde Nuclear Generating Station Unit 3 Reactor Coolant Pump 2A Suction Safe End Instrumentation Nozzle Half-Nozzle Repair Evaluation, WCAP-18051-P, Proprietary Version] to this Enclosure [APS letter dated October 22, 2015].

b. Flaw Evaluation

The flaw evaluation postulated a maximum bounding flaw that propagates axially and circumferentially through the J-groove weld and butter into the carbon steel base material to a depth conservative with the remainder of the Unit 3 licensed operating life, which expires on November 25, 2047. The results of the evaluation were found to be acceptable and are summarized in Attachments 1 and 2 to this Enclosure (non-proprietary and proprietary versions, respectively).

- The evaluation demonstrates the acceptability of the half-nozzle repair for the flawed RCP suction safe end instrument nozzle at PVNGS Unit 3. A three-dimensional finite element model was used to evaluate ASME Section III stresses and generate transient stress inputs for the fracture mechanics evaluation. The finite element model conservatively accounts for general corrosion of the nozzle bore for the half-nozzle repair.

  The supporting corrosion analysis assumed percentages of time that the plant operates at lower and higher temperatures to calculate total annual corrosion over a 60 year period. The assumptions are verified to remain valid via an engineering study that is periodically updated. The PVNGS engineering study, 13-MS-B041, *Alloy Steel Corrosion Analysis Supporting Alloy 600/690 Nozzle Repair/Replacement*, is the result of an existing regulatory commitment documented in Reference 7 [NRC letter *Palo Verde Nuclear Generating Station, Units 1, 2, and 3 – 10 CFR 50.55a(a)(3)(i) Alternative Repair Request for Reactor Coolant System Hot Leg Alloy 600 Small-Bore Nozzles (Relief Request 31, Revision 1),* dated August 16, 2005, ADAMS Accession No. ML052550368] for a prior RCS hot leg half-nozzle repair. The study tracks shutdown and start-up conditions for half-nozzle repairs to other primary system components, including RCS hot leg instrument nozzles and the Unit 3 reactor vessel bottom mounted instrument nozzle number 3. This ongoing tracking ensures that the corrosion evaluation remains bounding, and therefore, the allowable bore diameter of the RCP suction safe end instrument penetration (RCS cold leg) is not exceeded. The study is updated with operating cycle information following each refueling outage and is current through the most recent refueling outages (1R18 completed November 2014, 2R18 completed May 2014, and 3R18 completed May 2015).

- Transient stresses and welding residual stresses were calculated using finite element methods and the stresses were used in the fracture mechanics evaluation. The fracture mechanics evaluation was performed in accordance with ASME Section XI and justified continued operation for the remaining life of the plant with a flawed attachment weld present in the RCP suction safe end.

- A loose parts evaluation was performed to evaluate the effect that a postulated loose weld fragment(s) of the instrument nozzle partial penetration weld might have on the RCS and connected systems, structures, and components (SSCs). The loose parts 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. It was determined that all impacted SSCs would continue to be capable of satisfying their design functions.

The above two elements, i.e., the nozzle repair and flaw evaluation, provide an acceptable level of quality and safety in accordance with 10 CFR 50.55a(z)(1).

Attachment 1 of the licensee's October 22, 2015, letter is Westinghouse Report, 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," October 2015; Attachment 2 is the proprietary version of WCAP-18051. The licensee also provided supplemental information to WCAP-18051 in its May 20, 2016, supplemental letter. Below is a summary of the information provided by the licensee in WCAP-18051 and its May 20, 2016, supplemental letter.

During the 3R18 spring 2015 refueling outage at PVNGS, Unit 3, the licensee performed a half-nozzle repair of the RCP suction safe end Alloy 600 instrument nozzle due to leakage in the annulus between the outer surface of the instrument nozzle and the bore on the suction safe end. The Alloy 600 instrument nozzle is attached with a partial penetration weld to the inside of the RCP suction safe end using Alloy 82/182 weld filler metal. The licensee stated that the most likely location of the flaw(s) that caused the leakage is in the PWSCC-susceptible Alloy 82/182 weld and the Alloy 600 instrument nozzle, along their fusion line inside the safe end bore. The half-nozzle repair entailed severing the instrument nozzle on the outside of the RCP suction safe end, followed by machining out a portion of the nozzle inside the bore. The removed portion of the Alloy 600 instrument nozzle was then replaced with a section (half-nozzle) of more PWSCC-resistant Alloy 690 material which was welded to the outside surface of the suction safe end using Alloy 52M weld filler material. A small gap is left between the new Alloy 690 replacement nozzle and the remaining Alloy 600 nozzle which allows reactor coolant to enter the crevice region between the carbon steel (SA-508, Class 1) RCP suction safe end material, Alloy 690 replacement nozzle and the remaining Alloy 600 nozzle material. A schematic of the repair is provided Figure 1-1 below. The licensee stated that half-nozzle repairs have been successfully implemented on 73 Alloy 600 small-bore RCS hot-leg nozzles.

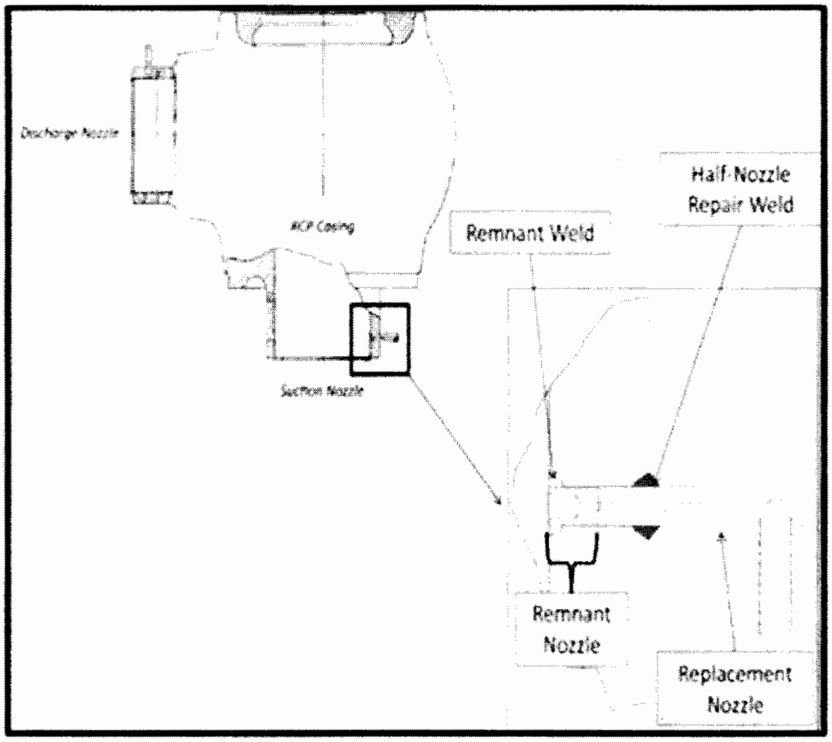

Figure 1-1. RCP 2A Instrumentation Nozzle Repair Schematic

3.5.1 Stress Analysis

The licensee performed a plant-specific stress analysis for the replacement weld and replacement half-nozzle in accordance with the ASME Code, Section III as presented in the Westinghouse report, WCAP-18051-P. The purpose of the stress analysis is twofold. First, the licensee needs to demonstrate that the stresses in the replacement nozzle satisfy the allowable stresses of the ASME Code, Section III. Second, the stress analysis generates loading inputs to be used in the evaluation of potential flaws in the remnant partial penetration J-groove weld. The stress analysis calculated the primary stresses, secondary stress, primary plus secondary stresses, peak stresses, and fatigue usage factors for the existing suction nozzle safe end material, replacement nozzle, and weld.

The licensee stated that the replacement nozzle was procured to the 1998 Edition through the 2000 Addenda of the ASME Code, Section III. The construction code for the existing RCP is the 1974 Edition of the ASME Code with no addenda. Therefore, the existing material is qualified per the construction code, and the new replacement nozzle and attachment weld are qualified to the 1998 Edition through 2000 Addenda of the ASME Code, Section III.

The RCP 2A suction instrument nozzle was a 1-inch nominal pipe size that contained an internal orifice designed to limit RCS leakage to within the capacity of the charging system in the event of a connected instrument line break. The nozzle extends through the approximately

3-inch thick RCP suction safe end pipe wall to connect to its associated instrument piping. A J-groove weld that connects the suction instrument nozzle to the safe end pipe wall inner surface provided the original RCS pressure boundary. The ASME Code, Section III, Class 1 to Class 2 boundary break was originally located at the internal orifice within the suction instrument nozzle.

The cladding in the inside surface of the safe end is SA-240 Type 304, the safe end is SA-508 Class 1, the remnant nozzle is nickel-based SB-166 Alloy 600, and the replacement nozzle is nickel-based SB-166 Alloy N06690. The remnant J-groove weld is nickel-based Alloy 82/182. The new attachment weld is nickel-based Alloy 52M. The thermal properties of water used in the finite element model are obtained as a function of temperature at the normal operation pressure.

The licensee generated a three-dimensional finite element model, including the RCP suction nozzle safe end, the remnant instrument nozzle, partial penetration J-groove weld, replacement nozzle, and the new attachment weld. In the finite element model, the licensee defined various paths to generate temperature and stress profiles which were used in the flaw evaluation.

The licensee included deadweight, various thermal transients, seismic, and accident loads which are combined to analyze the repaired nozzle for normal, upset, faulted, and test conditions. The licensee used a bounding value for the pressure that is higher than the pressure required in Article IWB-5000 of the ASME Code, Section XI, which specifies a maximum value 1.1 times the operating pressure.

The results of the licensee's stress analysis has shown that stresses in the design condition, normal and upset conditions, test conditions, and fault conditions are within the ASME Code allowable limits. In addition, the cumulative fatigue usage factor calculated for the RCP suction nozzle safe end and the replacement nozzle are within the allowable of 1.0 in accordance with the ASME Code, Section III.

3.5.2 Weld Design

The licensee stated that Article NB-3351.4 of the ASME Code, Section III, specifies a Category D weld meeting the requirements of Article NB-4244(d) for attachment of nozzles to vessels using partial penetration welds. The licensee designed the new attachment weld in accordance with Diagram (c) of Figure NB-4244(d)-1. Diagram (c) of Figure NB-4244(d)-1 requires that the J-groove weld depth and fillet weld leg be at least 3/4Tn, where Tn is the replacement instrument nozzle body thickness. Diagram (c) of Figure NB-4244(d)-1 also requires that the total weld size of the groove depth plus fillet leg height be a minimum of 1.5Tn. The licensee stated that the design weld depth is 1/2 inch and the full weld size is 3/4 inches. The licensee stated that the new attachment weld size satisfies the requirements of Diagram (c) of Figure NB-4244(d)-1.

The licensee stated that the Class 2 socket weld connecting the instrumentation nozzle to the downstream piping is qualified by designing the socket weld according to Section NC-3661.2 of the 1974 Edition of the ASME Code, Section III. Because the weld is sized according to design-by rules, it is qualified within the qualification of the existing Class 2 piping.

3.5.3 Vibration Assessment

The licensee moved the new attachment weld from the inside surface of the RCP suction safe end to the outside surface of the safe end as part of the repair. As a result, the pipe and the replacement nozzle may be affected by vibration of the RCP. Therefore, the licensee evaluated the effects of vibration affecting the instrumentation nozzle and attached Class 2 piping. The licensee analyzed the natural frequency of the replacement instrumentation nozzle and attached piping to ensure that the natural frequencies of the replacement nozzle and piping is outside the range of the vibration frequency of the RCP so as to avoid resonant vibration (harmonic excitation) of the nozzle and piping. The licensee stated that based on its analysis, the minimum piping frequency and nozzle frequency are outside of the restricted ranges, which is acceptable to avoid a resonant vibration issue. All other frequencies of the piping and nozzle are well outside of the restricted ranges.

The licensee also performed vibration testing on the replacement nozzle at RCP 2A at Unit 3 that was installed in spring 2015. The licensee found that the maximum vibrational displacement and peak velocity are within the allowable limits.

3.5.4 Flaw Evaluation

The purpose of a flaw evaluation is to show that any flaws in the remnant partial penetration J-groove weld will not grow to an unacceptable size into the safe end of the RCP for the remaining life of Unit 3. The existing J-groove weld joins the original instrument nozzle to the safe end. Therefore, any flaw in the remnant J-groove weld will likely to grow into the safe end base metal in the future. The flaw evaluation is to ensure that after the half-nozzle repair, the structural integrity of the safe end attached to the RCP is maintained, considering the potential flaw(s) in the J-groove weld that may grow into the wall of the safe end.

The licensee calculated the final flaw size based on a hypothetical initial flaw. As the actual flaw size in the original J-groove weld is not available, the licensee assumed the initial flaw size to be the entire radial extent of the J-groove weld, which would expose the RCP suction safe end base metal to the reactor coolant environment. The flaw geometry is considered as two corner cracks emanating from the edge of a hole in plate. The licensee stated that the primary degradation mechanism in ferritic steels for the initial flaw to grow is due to fatigue. The licensee used the fatigue crack growth rate for ferritic steel material in a pressurized-water environment based on the guidelines provided in Article A-4000 of the ASME Code, Section XI. After the final flaw size is calculated, the licensee evaluated the acceptability of the final flaw size inside the wall of the safe end based on the acceptance criteria of the ASME Code, Section XI, Appendix C.

The licensee considered normal, upset, emergency, faulted, and test conditions based on the pressure and thermal transient stresses and welding residual stresses in accordance with the 2001 Edition with 2003 Addenda of the ASME Code, Section XI, which is the current Code of record for PVNGS, Unit 3.

3.5.5 Fatigue Crack Growth

The fatigue crack growth analysis procedure involves postulating an initial flaw at the region of concern and predicting the growth of that flaw caused by an imposed series of loading transients. The input required for a fatigue crack growth analysis is essentially the information necessary to calculate the range of crack tip stress intensity factors, ΔK1, which depends on the crack size and shape, geometry of the structural component where the crack is postulated, and the applied cyclic stresses. The crack growth rate (da/dN) is a function of the applied stress intensity factor range (ΔKI) and the R ratio (Kmin/Kmax) for the transient.

After the fatigue crack growth of the initial flaw into the base metal of the RCP suction safe end has been calculated, the acceptability of the final flaw size is determined based on the flaw size acceptance criteria in Appendix C of the ASME Code, Section XI. The first step in establishing the acceptability of the final flaw size is to determine the failure mode for the operating transients.

The licensee used C-4220 of the ASME Code, Section XI, to determine the failure analysis method to analyze the acceptability of the final flaw size (i.e., whether the final flaw size will affect the structural integrity of the safe end to which the repaired nozzle is attached). The licensee noted that the screening criteria of C-4220 are particularly important when metal temperatures are below the upper shelf of the Charpy Energy curve of the safe end. At temperatures above the upper shelf of the Charpy Energy curve, the evaluation would be based on elastic plastic fracture mechanics (EPFM) because the fracture toughness can be described with elastic plastic parameters at these higher temperatures. At temperatures below the upper shelf of the Charpy Energy curve, the evaluation would be based on linear elastic fracture mechanics (LEFM). The licensee determined that the final flaw in the safe end needs to be analyzed by the LEFM and EPFM.

3.5.6 Linear Elastic Fracture Mechanics

The evaluation procedure and acceptance criteria used to demonstrate structural integrity of ferritic pipe in the LEFM regime is contained in Appendix C, Article C-7000 of the ASME Code, Section XI. A flaw must meet the acceptance criteria for normal, upset, emergency, faulted, and test conditions in order to be acceptable to remain in service. The acceptance criteria for LEFM are based on the crack tip stress intensity factor as follows: $K_I \leq (J_{Ic}E' / 1000)^{0.5}$ which simplifies to $K_I < K_{Ic}$. K_I is the Mode I stress intensity factor. K_{Ic} is the static fracture toughness for crack initiation under plane strain conditions (material resistance to cracking). J_{Ic} is the measure of toughness due to crack extension at the evaluation temperature. $E' = E / (1 - v^2)$. E is the Young's modulus and v is the Poisson's Ratio.

The licensee considered transients during the heatup, cooldown, hydrostatic test, and leak test with a temperature below 200 degrees Fahrenheit (°F). The safety factors from paragraph C-2622 of Appendix C of the ASME Code, Section XI, for axial flaws are included on the stress intensity factors due to primary transient stress intensity factors per Appendix C-7000 of the ASME Code, Section XI. The licensee did not include safety factors applied to the residual stress intensity factors according to Appendix C-7000 of the ASME Code, Section XI.

3.5.7 Elastic Plastic Fracture Mechanics

The licensee performed EPFM evaluation in accordance with Article C-6000 of the ASME Code, Section XI. The licensee also used EPFM evaluation procedures for ferritic components in Appendix K of the ASME Code, Section XI and NRC Regulatory Guide 1.161, "Evaluation of Reactor Pressure Vessels with Charpy Upper-Shelf Energy Less Than 50 Ft-Lb," June 1995 (ADAMS Accession No. ML003740038). Although the original purpose of Appendix K was to evaluate reactor vessels with low upper shelf fracture toughness, the general approaches in paragraph K-4220 and K-4310 are equally applicable to any region where the fracture toughness can be described with elastic plastic parameters. Therefore, the licensee used the general procedures of Appendix K accompanied by Appendix C safety factors applied to the transient stresses for the evaluation of the RCP suction safe end. The safety factors in Appendix C are more conservative than those used in Appendix K and are specific to piping. The licensee stated that suction safe end of the RCP has a 100 percent power normal operating temperature for consideration with the various operating transients. Furthermore, the temperature is considered to be sufficiently high and above the assumed upper shelf temperature of 200 °F, which would thus result in ductile behavior of the material. Therefore, the use of EPFM method is appropriate for the majority of the operating condition transients at high temperatures (above 200 °F).

For EPFM, the acceptance criteria are to be satisfied for each category of transients, namely, Service Load Level A (normal), Level B (upset and test), Level C (emergency) and Level D (faulted) conditions. Appendix C requires two criteria that must be satisfied to demonstrate the ductile stability of the safe end base metal. The first criterion is that the crack driving force must be shown to be less than the material toughness (i.e., $J_{applied} < J_{0.1\ material}$, where $J_{applied}$ is the J-integral value calculated for the postulated flaw under the applicable service level condition and $J_{0.1}$ is the J-integral characteristic of the material resistance to ductile tearing at a crack extension of 0.1 inch).

The second criterion is that the flaw must also be stable under ductile crack growth as follows:

> $\partial J_{applied} / \partial a < dJ_{material} / da$, at $J_{applied} = J_{material}$, where $J_{material}$ is the J-integral resistance to ductile tearing for the material.
>
> $\partial J_{applied} / \partial a$ is the partial derivative (slope) of the applied J-integral curve with respect to flaw depth.
>
> $dJ_{material} / da$ is the slope of the J-R curve. The licensee used NUREG/CR-5729, "Multivariable Modeling of Pressure Vessel and Piping J-R Data," May 1991 (ADAMS Legacy Accession No. 9106120180), to obtain J-R curves for the safe end material.

For small scale yielding, $J_{applied}$ of a crack can be calculated by the LEFM method based on the crack tip stress intensity factor, K_I. However, a plastic zone correction must be performed to account for the plastic deformation at the crack tip similar to the approach in Regulatory Guide 1.161. The plastic deformation ahead of the crack front is then regarded as a failed zone and the crack size is, in effect, increased. The licensee followed Appendix C-7000 of the ASME Code, Section XI, to convert K_I to $J_{applied}$.

3.5.8 Welding Residual Stresses

The licensee performed an analysis to determine the residual stress in the remnant j-groove weld using a finite element model. To calculate the welding residual stress, the licensee used a three-dimensional model, including the safe-end base metal, Alloy 82/182 partial penetration weld, Alloy 600 instrumentation nozzle, Alloy 690 half-nozzle, Alloy 52M outside surface replacement weld, and the cladding. The licensee used the resulting welding residual stresses in the fatigue crack growth calculation and the structural integrity evaluation of the final flaw size for the LEFM calculations.

3.5.9 Primary Stress Limit

In addition to satisfying the fracture criteria, the primary stress limit of the ASME Code, Section III, paragraph NB-3200 must be satisfied. The effects of a local cross-section area reduction that is equivalent to the area of the postulated flaw in the RCP suction safe end attachment weld must be considered by increasing the membrane stresses to reflect the reduced cross section. Membrane stresses in a thinned area of base metal due to the crack can be treated as a local primary membrane stress with an increased allowable stress intensity. The typical sizing is performed on the basis of the primary membrane stress intensity being less than Sm; however, since the reduction in thickness is local, the permissible stress intensity is increased by 50 percent.

The largest flaw size after crack growth for either the axial or circumferential flaw is used in the primary stress limit evaluation to envelop all results. The licensee noted that the final flaw depths include the cladding thickness which must be disregarded in the primary stress limit calculation. Therefore, the cladding thickness is subtracted from the flaw depth to represent the depth into the base metal only. The limiting final flaw depth is acceptable with respect to the primary stress limit.

3.5.10 Results of Flaw Evaluation

The LEFM evaluation results show that the postulated axial and circumferential flaws satisfy the requirements of Appendix C-7000 of the ASME Code, Section III. This shows that the safe end has sufficient fracture toughness to resist unstable crack growth for 40 years of operation. The EPFM evaluation results show that the postulated axial and circumferential flaw at the end of 40 years will not grow in an unstable manner. The licensee determined that crack growth for 40 years was small, and that the axial and circumferential flaws will remain within the allowable limits. The licensee concluded that the half-nozzle repair is acceptable for the 40 years of operation.

3.5.11 Stress Corrosion Cracking

In the licensee's May 20, 2016, letter, APS referenced Westinghouse Report WCAP-15973-P, Revision 1, "Low Alloy Steel Component Corrosion Analysis Supporting Small-Diameter Alloy 600/690 Nozzle Repair/Replacement Program, May 2004," to address the potential growth of flaws, as a result of stress-corrosion cracking, into the carbon steel safe end from the remnant J-groove weld.

The NRC safety evaluation dated January 12, 2005 (ADAMS Accession No. ML050180528), for Topical Report WCAP-15973-P, Revision 1, stated, in part, that, "the Westinghouse Owners Group 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." In its letter dated May 20, 2016, the licensee stated, in part, that

> 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 [WCAP-15973-P, Revision 1].
>
> 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.
>
> APS reviewed RCS chemistry records for the two operating cycles prior to the 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 [parts per billion] 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 [cubic centimeters] cc/[kilograms] 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 [WCAP-15973-P, Revision 1].

3.5.12 Corrosion

The licensee performed a general corrosion assessment of the nozzle bore diameter for the half-nozzle repair. The assessment determined the allowable increase in the diameter of the carbon steel safe end attachment nozzle bore due to corrosion. The licensee then compared the allowable diametrical increase in the bore hole to the expected corrosion growth of the bore hole after 40 years of operation.

Based on corrosion rate of 1.53 mils per year, the licensee calculated the expected corrosion of the safe end attachment nozzle bore to be 0.1224 inches (diametrically, relative to the penetration) after 40 years of operation. The licensee then compared the expected increase in the nozzle bore diameter to allowable increase in nozzle bore diameter. The results show that after 40 years of operation, the expected increase in nozzle bore diameter is less than the allowable nozzle bore diameter and, therefore, is acceptable for the next 40 years of operation.

3.5.13 Loose Parts Evaluation

The licensee evaluated the possibility that fragments of the remnant J-groove weld could come loose inside the RCS through the current planned end of plant life. The licensee postulated that the crack(s) in the remnant nozzle and weld are part through wall in the axial direction with no evidence of circumferential cracks based on previously observed for PWSCC in instrument nozzles in the hot leg. The remnant nozzle is recessed inside the safe-end bore. It remains constrained by a relatively tight radial clearance between the bore and the nozzle. The licensee stated that the hypothetical cracks are likely longitudinal part through wall, and as such the nozzle is able to maintain its structural integrity. Also, the licensee noted that if the remnant weld had several longitudinal radial cracks, it would require at least two other crack planes oriented in the circumferential direction in order to release a piece of any significant size. The licensee stated that because circumferentially-oriented cracks were not identified by the ultrasonic testing (UT), the likelihood for a fragment of the weld to be released is very low.

The licensee evaluated the structural and functional impacts of the postulated loose weld fragment(s) on affected SSCs based on fragments of different sizes, shapes, and weights. This evaluation 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. The licensee stated that the postulated loose parts will have no adverse impact on the RCS and connected SSCs through the current planned end of plant life.

3.6 Duration of Proposed Alternative Relief (as stated by the licensee)

The duration of the request is for the remainder of the Unit 3 third inservice interval, which expires January 10, 2018, based on the flaw evaluations of the effects of the remnant J-groove weld flaw for the remainder of the PVNGS Unit 3 licensed operating life.

4.0 NRC Staff Evaluation

4.1 Stress Analysis

The NRC staff finds that the licensee has analyzed the stresses in the RCP suction nozzle safe end, the remnant nozzle, weld, half-nozzle replacement nozzle, and the half-nozzle repair weld in accordance with the ASME Code, Section III. The NRC staff finds that the stresses of these components satisfy the allowable stresses of the ASME Code, Section III and, therefore, are acceptable.

4.2 Flaw Evaluation

The NRC staff finds that the licensee has evaluated the postulated flaw in the remnant J-groove weld and crack growth into the safe end in accordance with the ASME Code, Section XI, Appendices A, C, and K. The NRC staff further finds that the safe end with a postulated flaw satisfies the requirements of the ASME Code, Section XI, and therefore, is acceptable to remain in service.

4.3 Vibration Assessment

The NRC staff determines that the licensee has analyzed the vibration of RCP, the replacement nozzle, and associated piping. The licensee also had performed vibration testing of the repaired nozzle. The licensee reported that the vibration analysis and testing showed that the vibration frequencies are outside of the resonant vibration frequency. In this case, the replacement nozzle and piping will not be significantly affected by the RCP vibration. The NRC staff finds that the design of the replacement nozzle has considered the vibration loading. The NRC staff finds that the licensee has addressed the vibration issue with respect to the half nozzle repair satisfactorily.

4.4 Weld Design

The NRC staff finds that the licensee has designed the size of the attachment weld in accordance with the ASME Code, Section III.

4.5 Stress-Corrosion Cracking

The basis for the licensee's assessment of the possibility of flaws propagating from the remnant J-groove weld into the carbon steel RCP 2A suction safe end due to stress-corrosion cracking is based on Westinghouse Report WCAP-15973-P, Revision 1. In the NRC safety evalution dated January 12, 2005, the staff concluded that licensees may use the stress-corrosion assessment in WCAP-15973-P, Revision 1 as a basis for determining, for half-nozzle repairs, that flaws in the weld metal will not grow into the ferritic base material, by stress corrosion, if they meet the following requirements:

NRC Requirement 1

> Conduct appropriate plant chemistry reviews and demonstrate that a sufficient level of hydrogen overpressure has been implemented for the RCS, and that the

> contaminant concentrations in the reactor coolant have been typically maintained at levels below 10 ppb for dissolved oxygen, 150 ppb for halide ions, and 150 ppb for sulfate ions.

NRC Requirement 2

> During the outage in which the half-nozzle or MNSA [mechanical nozzle seal assembly] repairs are scheduled to be implemented, licensees adopting the [WCAP-15973-P, Revision 1] stress corrosion crack growth arguments will need to review their plant-specific RCS coolant chemistry histories over the last two operating cycles for their plants, and confirm that these conditions have been met over the last two operating cycles.

The NRC staff reviewed the supplemental information included in the licensee's May 20, 2016, letter and finds that the licensee's stress-corrosion cracking assessment is acceptable because it is based on the methodology described in NRC-approved WCAP-15973-P, Revision 1 and the licensee has fulfilled the two NRC requirements stated above.

4.6 Loose Parts Evaluation

The NRC staff concludes that the loose parts from the remnant nozzle and J-groove weld are not likely because a flaw in the remnant nozzle and J-groove weld are not likely to grow to the size that would cause the loose parts to fall into the RCS flow. In addition, the licensee considered the effects of a conservatively large postulated fragment, weighing 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), being released into the RCS flow. The licensee's evaluation shows that the postulated loose parts will have no adverse impact on the RCS and connected SSCs through the current planned end of plant life. Therefore, the NRC staff has concluded that the licensee has addressed the loose parts issue satisfactorily.

4.7 General Corrosion

Regarding general corrosion assessment, the NRC staff concludes that the overall general corrosion rate was calculated and is consistent with the methodology of WCAP-15973-P, Revision 1, "Low Alloy Steel Component Corrosion Analysis Supporting Small-Diameter Alloy 600/690 Nozzle Repair/Replacement Programs, May 2004." The NRC issued a letter on January 12, 2005, approving the methodology, described in WCAP-15973-P, Revision 1, for assessing the corrosion of carbon and low alloy steels as a result of half nozzle repairs. The NRC staff determined that the methodology described in WCAP-15973 is applicable to the repair described in RR 54. As required by the NRC's conditional approval of the use of WCAP-15973, the licensee calculated the expected corrosion using PVNGS, Unit 3 plant-specific operating data. The licensee is also tracking the time at cold shutdown conditions, which is the limiting transient. The licensee's analysis shows that the nozzle bore diameter will not increase to the maximum allowable diameter in 40 years of plant operation. This is conservative given that the requested duration of the licensee's alternative is from fall 2016 until the end of the third 10-year ISI interval, which is expected to end January 2018 (less than 18 months). The NRC staff concludes that the licensee has appropriately addressed the potential for corrosion of the nozzle bore.

4.8 Summary

The NRC concludes that the licensee's alternative to allow a defect to remain in the remnant J-groove weld and nozzle, and to forego future inspections, is acceptable because (1) the original RCP 2A suction pressure instrument nozzle and weld that contain the defect(s) no longer perform a pressure retaining function; (2) the new half nozzle and pressure retaining weld on the outside diameter on the RCP 2A suction safe-end meet all ASME Code requirements; (3) flaw(s) in the remnant nozzle and J-groove weld have been analyzed to show that the flaw(s) will not grow to an unacceptable size into the carbon steel safe end material; (4) flaw(s) in the remnant nozzle and J-groove weld have been analyzed to show that the flaw(s) will not grow into the carbon steel safe end material due stress-corrosion cracking; (5) the carbon steel nozzle bore has been appropriately evaluated for the effects of general corrosion; and (6) in the unlikely event that portions of the remnant weld or nozzle should fall into the RCS flow, the licensee's evaluation shows that the postulated loose parts will have no adverse impact on the RCS and connected SSCs.

5.0 CONCLUSION

As set forth above, the NRC staff concludes that the proposed repair will restore the primary system pressure boundary and provide reasonable assurance that the structural integrity of the repaired pressure instrument nozzle will be maintained for remainder of the third ISI interval. The NRC staff further concludes that the proposed alternative for the repair of the subject pressure instrument nozzle provides an acceptable level of quality and safety and that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(1). Therefore, the NRC authorizes the use of RR 54 at PVNGS, Unit 3 until the end of the Unit 3 third ISI interval, which is currently scheduled to end on January 10, 2018.

All other ASME Code, Section XI requirements for which relief was not specifically requested and approved in the subject request for relief remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

Principal Contributor: R. Davis

Date: June 23, 2016

All other ASME Code, Section XI requirements for which relief was not specifically requested and approved in the subject request for relief remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

If you have any questions, please contact the Project Manager, Siva P. Lingam, at 301-415-1564 or via e-mail at Siva.Lingam@nrc.gov.

Sincerely,

/RA/

Robert J. Pascarelli, Chief
Plant Licensing Branch IV-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. STN 50-530

Enclosure:
Safety Evaluation

cc w/encl: Distribution via ListServ

DISTRIBUTION:
PUBLIC
LPL4-1 R/F
RidsACRS_MailCTR Resource
RidsNrrDeEpnb Resource
RidsNrrDorlDpr Resource
RidsNrrDorlLpl4-1 Resource
RidsNrrLAJBurkhardt Resource
RidsNrrPMPaloVerde Resource
RidsRgn4MailCenter Resource
JBowen, EDO RIV
RDavis, NRR/DE/EPNB

ADAMS Accession No: ML16172A038 *SE via email

OFFICE	NRR/DORL/LPL4-1/PM	NRR/DORL/LPL4-1/LA	NRR/DE/EPNB/BC*	NRR/DORL/LPL4-1/BC
NAME	SLingam	JBurkhardt	DAlley	RPascarelli
DATE	6/21/2016	6/21/2016	6/15/2016	6/23/2016

OFFICIAL RECORD COPY