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

April 10, 2014

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 – REQUEST FOR RELIEF FROM ASME CODE, SECTION XI REQUIREMENTS REGARDING HALF-NOZZLE REPAIR AND FLAW EVALUATION AS AN ALTERNATIVE TO FLAW REMOVAL AND FLAW CHARACTERIZATION FOR FLAW IN BOTTOM-MOUNTED INSTRUMENT NOZZLE PENETRATION NO. 3 (TAC NO. MF3051)

Dear Mr. Edington:

By letter dated November 8, 2013 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML13317A071), and as supplemented by letters dated November 18, 2013 (ADAMS Accession No. ML13323A763), and November 20, 2013 (ADAMS Accession No. ML13325A098), Arizona Public Service Company (APS, the licensee) requested U.S. Nuclear Regulatory Commission (NRC) approval of Relief Request 51 from the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, for a proposed alternative related to axial flaw indications identified in a bottom-mounted instrument (BMI) nozzle for Palo Verde Nuclear Generating Station (PVNGS), Unit 3.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(a)(3)(i), the licensee proposed an ASME Code compliant half-nozzle repair and a flaw evaluation as an alternative to the ASME Code, Section XI requirement for flaw removal (IWA-4421) and flaw characterization (IWA-3300) on the basis that the alternative provides an acceptable level of quality and safety. Relief Request 51 was submitted to support the restart of PVNGS, Unit 3 in November 2013, with continued operation for one fuel cycle.

The NRC staff has reviewed the subject request and concludes, as set forth in the enclosed safety evaluation, that APS has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(3)(i). Therefore, during a conference call on November 21, 2013 (ADAMS Accession No. ML13330A573), the NRC staff verbally authorized the use of Relief Request 51 at PVNGS, Unit 3, effective for the duration of the next operating cycle (U3C18).

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

If you have any questions, please contact the Project Manager Jennivine Rankin at (301) 415-1530 or at Jennivine.Rankin@nrc.gov.

Sincerely,

Michael T. Markley, 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

UNITED STATES
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELIEF REQUEST 51

FLAW EVALUATION AS AN ALTERNATIVE TO REMOVAL AND CHARACTERIZATION

OF THE FLAW IN BOTTOM-MOUNTED INSTRUMENT NOZZLE J-GROOVE WELD

PALO VERDE NUCLEAR GENERATING STATION, UNIT 3

ARIZONA PUBLIC SERVICE COMPANY

DOCKET NO. 50-530

1.0 INTRODUCTION

By letter dated November 8, 2013 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML13317A071), and as supplemented by letters dated November 18, 2013 (ADAMS Accession No. ML13323A763) and November 20, 2013 (ADAMS Accession No. ML13325A098), Arizona Public Service Company (APS, the licensee) submitted Relief Request 51, requesting use of an alternative to American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI requirements related to axial flaw indications identified in a reactor pressure vessel (RPV) bottom-mounted instrument (BMI) nozzle for Palo Verde Nuclear Generating Station (PVNGS), Unit 3.

On October 6, 2013, APS found small white boric acid deposits around the annulus of the RPV BMI nozzle No. 3 for PVNGS, Unit 3, during the planned visual examinations pursuant to ASME Code N-722-1, "Additional Examination for PWR Pressure Retaining Welds in Class 1 Components Fabricated with Alloy 600/82/182 Materials ," in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55a. The licensee attributed the through-wall flaw to primary water stress-corrosion cracking (PWSCC).

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(a)(3)(i), the licensee proposed an ASME Code compliant half-nozzle repair and a flaw evaluation as an alternative to the ASME Code, Section XI requirement for flaw removal (IWA-4421) and flaw characterization (IWA-3300) on the basis that the alternative provides an acceptable level of quality and safety. The request was to support the restart of PVNGS, Unit 3 in November 2013, with continued operation for one operating fuel cycle. During a conference call on November 21, 2013 (ADAMS Accession No. ML13330A573), the U.S. Nuclear Regulatory Commission (NRC) staff verbally authorized the use of Relief Request 51 at PVNGS, Unit 3, effective for the duration of the next operating cycle (U3C18).

Enclosure

2.0 REGULATORY EVALUATION

Inservice inspection (ISI) of the ASME Code Class 1, 2, and 3 components is performed in accordance with Section XI of the ASME Code and applicable addenda as a way to detect anomaly and degradation indications so that structural integrity of these components can be maintained. This is required by 10 CFR 50.55a(g), except where specific relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i). 10 CFR 50.55a(a)(3) states that alternatives to the requirements of paragraph (g) may be used, when authorized by the NRC, if: (i) the proposed alternatives would provide an acceptable level of quality and safety or (ii) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

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 preservice examination requirements, set forth in the ASME Code, Section XI, to the extent practical within the limitations of design, geometry, and materials of construction of the components. The regulations require that inservice examination of components and system pressure tests conducted during the first 10-year interval and subsequent intervals comply with the requirements in the latest edition and addenda of Section XI of the ASME Code, which was incorporated by reference in 10 CFR 50.55a(b) 12 months prior to the start of the 120-month interval, subject to the limitations and modifications listed therein.

The ASME Code of record for PVNGS, Unit 3 for the third 10-year interval ISI program is the 2001 Edition of the ASME Code, Section XI, with 2003 Addenda.

3.0 TECHNICAL EVALUATION

3.1 Licensee's Proposed Alternatives and Bases for PVNGS, Unit 3\ BMI Nozzle Penetration 3

The affected component is RPV BMI Nozzle Penetration No. 3 at PVNGS, Unit 3. The licensee identified the following affected ASME Code requirements in its letter dated November 8, 2013:

> Section XI, Article IWA-4000 provides requirements for repair/replacement activities
>
> > IWA-4421 states, in part:
> >
> > > Defects shall be removed or mitigated in accordance with the following requirements...
> >
> > IWA-4422.1(a) states, in part:
> >
> > > A defect is considered removed when it has been reduced to an acceptable size...

IWA-4422.1(b) states, in part:

> Alternatively, the defect removal area and any remaining portion of the defect may be evaluated and the component accepted in accordance with the appropriate flaw evaluation provisions of Section X1...

Section XI, Article IWA-3000 provides standards for examination evaluation.

IWA-3100(a) states, in part:

> Evaluation shall be made of flaws detected during an inservice examination as required by IWB-3000 for Class 1 pressure retaining components...

IWA-3300(b) states, in part:

> Flaws shall be characterized in accordance with IWA-3310 through IWA-3390, as applicable.

Section XI, Article IWB-3000 provides acceptance standards for Class 1 components

IWB-3420 states:

> Each detected flaw or group of flaws shall be characterized by the rules of IWA-3300 to establish the dimensions of the flaws. These dimensions shall be used in conjunction with the acceptance standards of IWB-3500.

By letter dated November 8, 2013, APS proposed the following alternative to flaw removal for BMI Nozzle Penetration No. 3 and provided the basis for their request:

> A completed half nozzle repair at BMI nozzle #3 using PWSCC resistant material which relocates the pressure boundary weld from inside the reactor vessel to outside the reactor vessel.
>
> The half nozzle repair is an industry standard, ASME Code compliant repair method that adds a PWSCC resistant Alloy 52M weld pad on the outside of the reactor vessel using ambient temperature temper bead welding in accordance with ASME Code Case N-638-4. A new PWSCC resistant Alloy 690 half-nozzle is attached to the weld pad using a partial-penetration J-groove weld. The half-nozzle repair of BMI nozzle penetration #3 will not remove the flaws in the original J-groove weld or Alloy 600 nozzle material near this weld. Crack propagation into the vessel wall can be addressed by analysis since low-alloy base material is not susceptible to PWSCC and flaws will blunt at the interface of the low-alloy base material and the J-groove weld.

Additionally, in its letter dated November 8, 2013, the licensee stated that the ambient temperature temper bead weld pad met the NRC staff code case conditions for approval of ASME Code Case N-638-4, "Similar and Dissimilar Metal Welding Using Ambient Temperature Machine GTAW [Gas Tungsten Arc Welding] Temper Bead Technique," in NRC Regulatory Guide (RG) 1.147, Revision 16, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1," October 2010 (ADAMS Accession No. ML101800536), since:

1. The gas tungsten arc machine welding method for the ambient temperature bead weld pad was successfully demonstrated in advance by [ultrasonic] UT examination using a mockup implanted with fabrication-type flaws.

2. Because of radiological conditions, heat flow calculations were performed in lieu of monitoring preheat and interpass temperatures.

Further details of the half-nozzle repair to BMI Nozzle Penetration No. 3 can be found in Attachment 1 of the licensee's letter dated November 8, 2013.

Additionally, APS proposed the following alternative to flaw characterization for BMI Nozzle Penetration No. 3 and provided the basis for their request:

> Since current NDE [non-destructive examination] procedures are not capable of sizing the extent of crack growth into the PWSCC susceptible weld material, the flaw evaluation postulated a maximum bounding flaw that propagates radially through the J-groove weld, buttering, and clad into the low-alloy base material to a depth conservative with respect to one operating fuel cycle.

Further details of the flaw fracture mechanics evaluation for BMI Nozzle Penetration No. 3 can be found in Attachment 2 to the licensee's letter dated November 8, 2013.

The licensee stated that the duration of the request is for the next Unit 3 operating fuel cycle (U3C18).

3.2 NRC Staff Evaluation

3.2.1 Evaluation of the Half-Nozzle Repair

A portion of the existing Alloy 600 nozzle was cut and 5 inches of the remnant portion remains in the BMI penetration. A new Alloy 690 nozzle was welded to the weld buildup in accordance with the requirements of ASME Code, Section XI. The new reactor coolant pressure boundary for BMI Nozzle Penetration No. 3 is formed by the J-groove weld between the new Alloy 690 outer nozzle and the temper bead weld buildup and by the welds between the nozzle and in-core instrumentation guide tube as shown in Figure 1 of Attachment 1 of the licensee's letter dated November 8, 2013.

The half-nozzle repair on BMI Nozzle Penetration No. 3 at PVNGS, Unit 3, entailed a weld buildup on the RPV bottom head using a temper bead technique in accordance with the ASME

Code Case N-638-4, “Similar and Dissimilar Metal Welding Using Ambient Temperature Machine GTAW [Gas Tungsten Arc Welding] Temper Bead Technique.” This code case was approved by the NRC staff, with two conditions, in RG 1.147, Revision 16. The licensee stated that it complied with all the requirements of ASME Code Case N-638-4 including the two conditions specified in RG 1.147.

Based on its review, the NRC staff concluded that the licensee performed the nozzle repair in accordance with ASME Code, Section XI and the NRC staff-approved ASME Code Case N-638-4 and associated conditions. Therefore, the NRC staff concludes that the weld repair of the BMI Nozzle Penetration No. 3 at PVNGS, Unit 3, is acceptable.

3.2.2 Evaluation of the BMI J-groove Weld Remnant Flaw Evaluation

In order to justify allowing the axial flaws in the J-groove weld and the upper portion of the original BMI nozzle to remain in place instead of removing the defect as required by ASME Code, Section XI, IWA-4421 and not characterizing the flaw as required by IWA-3300, the licensee performed a flaw evaluation to demonstrate the structural integrity of the RPV for one operating cycle.

A typical flaw evaluation includes determination of the initial flaw size, the applied stress intensity factor ($K_{applied}$) values (for linear elastic fracture mechanics (LEFM)) or the applied J-integral values (for elastic plastic fracture mechanics (EPFM)), crack growth, and stability of the final crack size.

3.2.2.1 Initial and Final Flaw Size

The postulated initial flaw, as stated in Attachment 2 of the licensee's letter dated November 8, 2013, is assumed to be a radial crack that has propagated through the J-groove weld and butter and touches the RPV carbon steel material that comprises the pressure boundary. This assumed initial flaw size, which represents the radial cross section of the J-groove weld (the worst possible radial crack that could exist in the weld), is consistent with those in approved applications of similar nature and has become standard industry practice. The licensee's postulated final flaw size assumes a flaw growth of 0.04-inch for one fuel cycle. This final flaw size is conservative when compared with a calculated value of 0.305-inch for 50 years of operation for a similar flaw evaluation performed by AREVA for a relief request dated June 13, 2003, regarding the South Texas Project, Unit 1 BMI nozzle nos. 1 and 46 (NRC safety evaluation (SE) located at ADAMS Accession No. ML032130436). The NRC staff concludes that the licensee's initial flaw size and flaw growth is conservative and the final flaw size is appropriate for the subsequent stress and fracture mechanics analyses.

3.2.2.2 Material Property Values

Section 4.1.1 of Attachment 2 to the licensee's letter dated November 8, 2013, indicated that the minimum yield strength values for the reactor vessel bottom head (RVBH) used for the flaw evaluation at operating temperatures are from the ASME Code, Section III. Additionally, the licensee stated that thermal material properties shown in Table A-1, “Material Properties,” which were used in the cooldown stress analysis in Appendix A to Attachment 2 were also from the ASME Code, Section III.

In Section 4.1.2 of Attachment 2, the licensee stated that the RVBH nil-ductility reference temperature (RT_{NDT}) value of -60 degrees Fahrenheit (°F) is taken from Reference 1 of Attachment 2, titled, "Palo Verde Unit 3 Reactor Vessel Bottom Mounted Instrument Nozzle Modification." This RT_{NDT} value is used to calculate the crack arrest fracture toughness (K_{Ia}), which is used in the subsequent LEFM analysis to determine the acceptability of the flaw per ASME Code. By letter dated November 18, 2013, in response to the NRC staff's request for additional information (RAI) RAI-1, dated November 15, 2013 (ADAMS Accession No. ML13322A016), the licensee confirmed that the RT_{NDT} for the RVBH is from the Certified Material Test Report (CMTR) with an adjustment in accordance with ASME Code, Section III, Article NB-2331. Therefore, the NRC staff concludes that the licensee's response to RAI-1 is acceptable.

For the LEFM analysis, the fracture toughness is based on the K_{Ia} (fracture toughness based on crack arrest) equation in Section 4.1.4 of Attachment 2, which is from ASME Code, Section XI, Appendix A, "Analysis for Flaws," which is applicable to RPVs. Therefore, the NRC staff concludes that the LEFM fracture toughness value is acceptable. For the EPFM analysis, the fracture toughness is based on the J integral-resistance (J-R) curve from RG 1.161, "Evaluation of Reactor Pressure Vessels with Charpy Upper-Shelf Energy [USE] Less Than 50 FT-LB." The NRC staff's acceptance of the J-R curve is evaluated in Section 3.2.2.5 of this SE.

3.2.2.3 <u>Operating and Residual Stresses</u>

Section 4.4 of Attachment 2 of the licensee's letter dated November 8, 2013, indicated that the thermal and pressure stresses used in the current flaw evaluation were considered during normal steady state operating conditions, cooldown (CD) conditions, and a combination of the two. These conditions are acceptable for this flaw evaluation because the CD condition creates tensile stresses on an inner diameter (ID) flaw, which is the location of the flaw under evaluation, while the heatup condition does not.

However, a typical flaw evaluation in accordance with the ASME Code, Section XI, Appendix A also requires consideration of emergency and faulted conditions. The NRC staff's RAI-2 dated November 15, 2013, requested the licensee to address the flaw evaluation under the emergency and faulted conditions. In response to RAI-2, the licensee examined the emergency condition and two faulted conditions and performed a detailed flaw evaluation for the faulted condition that could become limiting: the loss of secondary pressure transient condition. The NRC staff compared the flaw evaluation results for the normal condition and for the limiting faulted condition and found that the normal condition is bounding. Therefore, the NRC staff concludes that the licensee's response to RAI-2 is acceptable.

The NRC staff's RAI-3, RAI-4, and RAI-5 are related to the thermal and structural finite element method (FEM) analyses and results that were used to support the licensee's current flaw evaluation. Resolving these RAIs is essential for the NRC staff to have confidence in the licensee's FEM analyses and results.

Section 4.4 of Attachment 2 of the licensee's submittal stated that the thermal stresses during CD were obtained using a two-dimensional axisymmetric FEM model described in Appendix A to Attachment 2. The NRC staff requested in RAI-3 that the licensee (1) demonstrate that the one-dimensional (i.e., same value at the same RPV bottom head wall thickness) temperature and stresses results are realistic, and (2) explain the physical meaning of the unique shape of the temperature difference-time plot (the right figure in Figure A-2 of the licensee's submittal). The licensee's RAI response dated November 18, 2013, stated that the gap between the BMI nozzle and the RPV bottom head bore is filled with stagnant water, limiting heat transfer between the BMI nozzle and the RPV bottom head wall. Hence, the heat transfer in the RPV bottom head is primarily in the radial direction. The NRC staff concludes this explanation is acceptable because the assumption is consistent with the physical observation as explained by the licensee. Further, the licensee's attribution of the unique shape of the temperature difference-time plot to the big difference between the convection heat transfer coefficients at the inner and outer surfaces of the RPV bottom head is reasonable. Therefore, the NRC staff concludes that the licensee's response to RAI-3 is acceptable.

In addition to thermal and pressure stresses, the licensee's evaluation also considered the residual stresses and the stresses due to crack face pressure. The NRC staff's RAI dated November 15, 2013, RAI-4, and the staff's follow-up RAI (RAI-3 in the staff's November 20, 2013, second round of RAIs, available at ADAMS Accession No. ML13324B123) are related to questions on residual and operating stresses. In the licensee's initial response to RAI-4, a copy of Calculation No. C-7789-00-2, Revision 1 (Rev. 1), "Palo Verde Bottom Head Instrumentation Nozzle Stress Analysis," was provided to the NRC staff to support the combined stresses used in the flaw evaluation. Additionally, in Table 6-1 of Attachment 2 of the licensee submittal dated November 8, 2013, the licensee provided selected computer printout on stresses were also provided to explain the deep stress valley along the RPV bottom head bore close to the postulated J-groove crack tip under the steady state (SS) loading condition. The NRC staff concludes that these stresses, based on a 3-dimensional elastic plastic FEM model, are acceptable because sufficient details have been provided to support the RPV bottom head bore stress distribution. Therefore, the NRC staff concludes that the licensee's initial and follow-up responses to RAI-4 are acceptable.

Table 4-3 of Attachment 2 of the licensee submittal dated November 8, 2013, presents the hoop stresses at different depths of the RPV wall for the SS, CD, and combined effect. In RAI-5, dated November 15, 2013, the NRC staff requested that the licensee provide the separate stress contributions due to pressure, thermal, and residual stresses for each stress value of Table 4-3 and confirm that residual stresses are considered in the associated $K_{applied}$ and applied J integral ($J_{applied}$) calculations. The licensee's response pointed to Calculation No. C-7789-00-2, Rev. 1, discussed previously, for the combined hoop and axial stresses from operating pressure, operating temperature, and residual stresses under the SS condition. The hoop and axial stresses under CD condition represent stresses due to a cooldown rate of 100 °F/hour only. The licensee also confirmed that the residual stresses were included in the subsequent $K_{applied}$ and $J_{applied}$ calculations in the fracture mechanics evaluation. The licensee's RAI response provided all the requested information; therefore, the NRC staff concludes that the licensee's response to RAI-5 is acceptable.

3.2.2.4 Linear Elastic Fracture Mechanics (LEFM) Evaluation

Section 2.1 of Attachment 2 of the licensee's letter dated November 8, 2013, indicated that, with the operating and residual stresses discussed in section 3.2.2.3 above as inputs, the licensee used the $K_{applied}$ solution based on the quarter space nozzle corner flaw LEFM model. This LEFM model was used in 1980 in resolving issues with the boiling-water reactor (BWR) feedwater nozzle cracking, and was formally evaluated and accepted by the NRC in recent years in SEs for a series of topical reports (TRs) related to pressure-temperature limits for BWRs. A general application of this LEFM model is established in an Oak Ridge National Laboratory (ORNL) report ORNL/TM-2010/246, "Stress and Fracture Mechanics Analyses of Boiling Water Reactor and Pressurized Water Reactor Pressure Vessel Nozzles," December 2010 (ADAMS Accession No. ML110060164). As an additional example, this LEFM model was used in the SE dated March 14, 2013, for the BWR Owners' Group (BWROG) TR BWROG-TP-11-023, Rev. 0, "Linear Elastic Fracture Mechanics Evaluation of General Electric Boiling Water Reactor Water Level Instrument Nozzles for Pressure-Temperature Curve Evaluations," November 2011 (ADAMS Accession No. ML113250288). Therefore, the NRC staff concludes that the licensee's use of the quarter space nozzle corner flaw LEFM model in this application is acceptable.

Applying the stresses (operating, residual, and crack face) discussed in Section 3.2.2.4 above to the quarter space nozzle corner flaw model with the final flaw size and the RPV-nozzle geometry documented in Section 4.2 of Attachment 2 of the licensee's submittal, the licensee calculated the $K_{applied}$ values during the SS, CD, and SS+CD conditions and summarized the results in Table 6-1. This table also contains the corresponding K_{Ia} values, calculated margins ($K_{Ia}/K_{applied}$), and the ASME Code, Section XI required fracture toughness margins for comparison. Table 6-1 indicates that the actual margin exceeds the required margin for the CD condition, demonstrating that the LEFM analysis result is acceptable for the CD condition. However, for the SS and SS+CD conditions, the required margin per the ASME code exceeds the actual margin. Therefore, the licensee resorts to EPFM to re-evaluate the postulated J-groove crack under the SS and the combined condition. It should be noted, however, that the NRC staff's evaluation of the licensee's LEFM analysis here is still valuable because the $K_{applied}$ values are a key input to the EPFM analysis.

3.2.2.5 Elastic-Plastic Fracture Mechanics (EPFM) Evaluation

Section 4.1.4 of Attachment 2 of the licensee submittal presents the generic J-R curve used in the EPFM evaluation. This J-R curve is based on the model from Appendix D to NUREG-0744, Vol. 2, Rev. 1, "Resolution of the Task A-11 Reactor Vessel Materials Toughness Safety Issue." This NUREG discusses analytical bases for low upper-shelf energy (USE) RPV material toughness evaluations. Since the J-R model from NUREG-0744 is for a fixed temperature, the NRC staff's RAI-6 requested the licensee use the generic temperature-dependent J-R curve models in RG 1.161. The NRC staff noted that the database underlying the J-R model for RPV base metals in RG 1.161 contains the data used to develop the J-R model in Appendix D to NUREG-0744, Vol. 2, Rev. 1. Also, the database contains not just low USE materials. Therefore, the RG 1.161 J-R model remains the generic model acceptable to the NRC for RPV applications using EPFM. After issuance of a follow-up RAI (RAI-4 in the November 20, 2013 second round RAI letter), the licensee provided a revised EPFM evaluation acceptable to the

NRC staff, which used the generic J-R curve model in RG 1.16, which considers the changing temperature during the SS + CD condition and the safety factors that the NRC staff approved in the SEs related to RPV head penetrations (e.g., ADAMS Accession Nos. ML042890174 for Arkansas Nuclear One, Unit 1, and ML102571569 for Davis-Besse Nuclear Power Station). The EPFM methodology has two sets of criteria: one for the flaw stability assessment and the other for the flaw extension assessment. The revised EPFM results reported in the licensee's response to the follow-up RAI revealed that both criteria are satisfied, demonstrating that the unit can be operated for one operating cycle with the postulated flaw in the J-groove weld of the subject BMI penetration.

4.0 CONCLUSION

As set forth above, the NRC staff concludes that the half-nozzle repair and the use of the flaw evaluation as an alternative to the ASME Code, Section XI requirement for flaw removal and flaw characterization provides an acceptable level of quality and safety for the duration of one operating fuel cycle. This conclusion is based on (1) the weld repair complied with the requirements of ASME Code, Section XI and the staff-approved ASME Code Case N-638-4 with conditions, (2) the flaw evaluation is consistent with the approved methodology for RPV penetrations, and (3) the EPFM results support operation for one fuel cycle with the flaw in the J-groove weld of the subject BMI penetration. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(a)(3)(i), and is in compliance with the ASME Code requirements. Therefore, the NRC staff authorizes the alternative at PVNGS, Unit 3, for the duration of the current operating cycle (U3C18).

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 Contributors: Simon Sheng and Ganesh Cheruvenki

Date: April 10, 2014

If you have any questions, please contact the Project Manager Jennivine Rankin at (301) 415-1530 or at Jennivine.Rankin@nrc.gov.

Sincerely,

/RA/

Michael T. Markley, Chief
Plant Licensing Branch IV-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. STN 50-530

Enclosure:
Safety Evaluation

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