September 12, 2006

Mr. James M. Levine
Executive Vice President, Generation
Arizona Public Service Company
P. O. Box 52034
Phoenix, AZ 85072-2034

SUBJECT: PALO VERDE NUCLEAR GENERATING STATION, UNITS 1, 2, AND 3 - RELIEF REQUEST NO. 31, REVISION 1, RE: PROPOSED ALTERNATIVE REPAIR FOR REACTOR COOLANT SYSTEM HOT-LEG ALLOY 600 SMALL-BORE NOZZLES (TAC NOS. MC9159, MC9160, AND MC9161)

Dear Mr. Levine:

By letter dated August 16, 2005, Arizona Public Service Company submitted Relief Request No. 31, Revision 1, requesting relief from certain American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code) requirements at Palo Verde Nuclear Generating Station (Palo Verde), Units 1, 2, and 3. The request for relief would authorize an alternative repair for previously repaired Alloy 600 small-bore reactor coolant system hot-leg nozzles in lieu of the ASME Code, Section XI, requirements for required flaw examinations and successive inspections.

Based on the enclosed safety evaluation, the Nuclear Regulatory Commission staff concludes that the proposed alternative provides an acceptable level of quality and safety. Therefore, pursuant to 50.55a(a)(3)(i) of Title 10 of the *Code of Federal Regulations*, the licensee's alternative repair as stated in Relief Request No. 31, Revision 1, is authorized for Palo Verde Units 1, 2, and 3 for the second, third, and fourth 10-year inservice inspection intervals with the regulatory commitment made by the licensee as listed in the enclosed safety evaluation.

All other requirements of the ASME Code, Section III and XI for which relief has not been specifically requested and approved remain applicable, including third party review by the Authorized Nuclear Inservice Inspector.

Sincerely,

/RA/

David Terao, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. STN 50-528, STN 50-529,
and STN 50-530

Enclosure: Safety Evaluation

cc w/encl: See next page

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cc w/encl: See next page

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

INSERVICE INSPECTION PROGRAM RELIEF REQUEST NO. 31, REVISION 1

ARIZONA PUBLIC SERVICE COMPANY, ET AL.

PALO VERDE NUCLEAR GENERATING STATION, UNITS 1, 2, AND 3

DOCKET NOS. STN 50-528, STN 50-529, AND STN 50-530

1.0 INTRODUCTION

By letter dated August 16, 2005 (Agencywide Documents Access and Mangement System (ADAMS) Accession No. ML052550368), Arizona Public Service Company (APS or the licensee) submitted Relief Request No. 31, Revision 1, requesting relief from certain American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code) requirements at Palo Verde Nuclear Generating Station (Palo Verde or PVNGS), Units 1, 2, and 3. The request for relief would allow an alternative repair for previously repaired Alloy 600 small-bore reactor coolant system (RCS) hot-leg nozzles in lieu of the ASME Code, Section XI, requirements for required flaw examinations and successive inspections.

This request for relief seeks approval for the 27 Unit 1, 9 Unit 2, and 27 Unit 3 RCS remnants of nozzles and/or attachment welds to remain in service without the ASME Code flaw characterizations and successive examinations. Instead, calculations will be used to show that the components will perform their required functions for the remainder of their service life.

2.0 REGULATORY REQUIREMENTS

The inservice inspection (ISI) of the ASME Code Class 1, 2, and 3 components in nuclear plants is to be performed in accordance with the ASME Code, Section XI, and applicable edition and addenda as required by 50.55a(g) of Title 10 of the *Code of Federal Regulations* (10 CFR), except where specific relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i). The regulation at 10 CFR 50.55a(a)(3) states: "Proposed alternatives to the requirements of paragraphs (c), (d), (e), (f), (g), and (h) of this section or portions thereof may be used when authorized by the Director of the Office of Nuclear Reactor Regulation. The applicant shall demonstrate that: (i) The proposed alternatives would provide an acceptable level of quality and safety, or (ii) Compliance with the specified requirements of this section 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 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. The regulations require that ISI 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 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 second 10-year ISI interval for Palo Verde Units 1, 2, and 3 began in July 1998, March 1997, and January 1998, respectively. The ISI Code of record is the 1992 Edition with the 1992 Addenda. The components (including supports) may meet the requirements set forth in subsequent editions and addenda of the ASME Code incorporated by reference in 10 CFR 50.55a(b) subject to the limitations and modifications listed therein and subject to commission approval.

3.0 RELIEF REQUEST NO. 31, REVISION 1, PROPOSED ALTERNATIVE REPAIR FOR REACTOR COOLANT SYSTEM HOT-LEG ALLOY 600 SMALL-BORE NOZZLES

3.1 Code Requirements

Sub-article IWA-4310 of ASME Code, Section XI, 1992 Edition, 1992 Addenda states in part that the “defects shall be removed or reduced in size in accordance with this Paragraph.”

The remaining portion of the flaw left in service requires flaw characterization as stated in IWA-3300 and successive examinations as stated in IWB-2420.

3.2 Licensee’s Code Relief Request and its Proposed Alternative

The licensee proposed alternatives to the required flaw characterization (IWA-3300) and successive inspections (IWB-2420). The licensee did not remove the remnant sleeve or its attachment weld.

In lieu of fully characterizing/sizing the potentially existing cracks, the licensee assumed worst case cracks in the Alloy 600 base and weld material and used the methodology presented in NRC-approved Westinghouse Topical Report (TR) WCAP-15973-P, Revision 01, "Low-Alloy Steel Component Corrosion Analysis Supporting Alloy 600/690 Nozzle Repair Program," to support the request. The licensee reviewed the bases and arguments presented in the TR and determined that the TR bases and arguments can be applied to the previously repaired Unit 1, 2, and 3 hot-leg small-bore nozzles.

The licensee further evaluated the assumptions made to support this relief request using appropriate flaw evaluation rules of ASME Code, Section XI, and determined that the results demonstrate compliance with ASME Code, Section XI, criteria for the expected 40 years of remaining plant life. As a result, the licensee is also requesting relief from the successive inspections required by IWB-2420 for the remainder of plant life (40 year original life plus 20 year life extension).

3.3 Components for which Relief Is Requested

This relief request is retroactive in nature, applying to previously repaired Palo Verde Alloy 600 small-bore RCS hot-leg nozzles classified as ASME Code, Section XI, Class 1, component number B9.32 as follows; twenty-seven (27) Unit 1 nozzles, nine (9) Unit 2 nozzles and twenty seven (27) Unit 3 nozzles. These 63 nozzles were replaced under the Palo Verde Alloy 600 replacement program, from approximately October 1999 to April 2003.

3.4 Licensee's Basis for Proposed Alternative

3.4.1 Introduction

During fabrication of the RCS piping, Alloy 600 small-bore nozzles were welded to the interior of the RCS hot-leg. Industry experience has shown that cracks may develop in the nozzle or in the weld metal joining the nozzles to the reactor coolant pipe and lead to leakage for the reactor coolant fluid. The cracks are caused by primary water stress-corrosion cracking.

The original design of each Palo Verde unit RCS contained a total of twenty-seven (27) Alloy 600 small-bore hot-leg penetrations. These penetrations include pressure taps, sampling line, and resistance temperature detector thermowell nozzles. The 63 nozzles which are the subject of this request were replaced under the Palo Verde Alloy 600 replacement program, from approximately October 1999 to April 2003.

The total removal of all Alloy 600 small-bore nozzle and/or Alloy 82/182 weld material would have required accessing the internal surface of the reactor coolant piping and grinding out the attachment weld and any remaining nozzle. Such an activity would result in high radiation exposure to the personnel involved. Grinding within the pipe would also expose personnel to other safety hazards such as those associated with confined spaces. The analysis in WCAP-15973-P has shown that any cracks in the nozzle or attachment weld and vessel/piping carbon steel base metal will not affect structural integrity or propagate through the reactor coolant pressure boundary; therefore, there is no increase in the level or quality or safety as a result of removing the nozzle or the attaching weld metal.

3.4.2 Licensee's Basis

Pursuant to 10 CFR 50.55a(a)(3)(i), the licensee proposed alternatives to the flaw characterization (IWA-3300) and successive inspections (IWB-2420) requirements of the ASME Code, Section XI. The licensee will not be removing the remnant sleeve or its attachment weld.

In lieu of fully characterizing/sizing the potentially existing cracks, the licensee assumed worst case cracks in the Alloy 600 base and weld material and used the methodology presented in NRC-approved WCAP-15973-P, Revision 01, for determining the following:

1. The overall general/crevice corrosion rate for the internal surfaces of the low-alloy or carbon steel materials that will now be exposed to the reactor coolant and for calculating the amount of time the ferritic portions of the vessel or piping would be acceptable if corrosive wall thinning had occurred.

2. The thermal-fatigue crack-growth life of existing flaws in the Alloy 600 nozzles and/or Alloy 182/82 weld material into the ferritic portion of the vessels or piping.

3. Acceptable bases and arguments for concluding that unacceptable growth of the existing flaw by stress corrosion into the vessel or piping is improbable.

The licensee has reviewed the bases and arguments presented in WCAP-15973-P for the overall general/crevice corrosion rate, thermal-fatigue crack-growth life of existing flaws, and

the bases for concluding that growth of the existing flaw by stress corrosion into the vessels or piping is improbable. The licensee finds that these bases and arguments apply to the replacement of the Units 1, 2, and 3 hot-leg small-bore nozzles. The licensee has evaluated these assumptions using appropriate flaw evaluation rules of Section XI and determined that the results demonstrate compliance with ASME Code, Section XI, criteria for the expected 40 years of plant life. As a result, the licensee is also requesting relief from the successive inspections required by IWB-2420.

The licensee has determined that the proposed alternatives will provide an acceptable level of quality and safety and are within the analysis boundaries provided in WCAP-15973-P, Revision 01, by answering the following requirements set forth by Nuclear Regulatory Commission (NRC) letter dated January 12, 2005, "Final Safety Evaluation For Topical Report WCAP-15973-P, Revision 01."

3.4.2.1 General Corrosion Assessment

Licensees seeking to use the methods of Westinghouse TR WCAP-15973-P, Revision 01, will need to perform the following plant-specific calculations in order to confirm that the ferritic portions of the vessels or piping within the scope of the TR will be acceptable for service throughout the licensed lives of their plants (40 years if the normal licensing basis plant life is used or 60 years if the facility is expected to be approved for extension of the operating license):

<u>NRC Requirement 1</u>

Calculate the minimum acceptable wall thinning thickness for the ferritic vessel or piping that will adjoin to the mechanical nozzle seal assembly (MNSA) repair or half-nozzle repair.

<u>APS Response</u>

> Section 2.4 of the Westinghouse TR, determined the maximum allowable hole size relative to (1) the reduction in the effective weld shear area, and (2) the required area of reinforcement for the nozzle bore holes.
>
> The maximum corroded hole diameter identified in the TR has been verified to apply to Palo Verde Units 1, 2, and 3. As for the second hole size, Palo Verde was used in the TR as one of the limiting hot-leg pipe nozzles.

<u>NRC Requirement 2</u>

Calculate the overall general corrosion rate for the ferritic materials based on the calculational methods in the Westinghouse TR, the general corrosion rates listed in the Westinghouse TR for normal operations, startup conditions (including hot standby conditions), and cold shutdown conditions, and the respective plant-specific times (in-percentage of total plant life) at each of the operating modes.

APS Response

The assumptions in the Westinghouse TR analysis, regarding times at each of the operation modes, are as follows:

- Normal Operation: 88%,
- Startup Condition - 2%,
- Cold Shutdown Condition - 10%

APS has reviewed the operating history for PVNGS Unit 2 from when the first hot-leg nozzle repair was implemented and has determined the percentage of total plant time spent at each of the operating modes as follows:

Mode of Operation	**Unit 1**	**Unit 2**	**Unit 3**
Normal Operations	93.54%	90.22%	90.59%
Startup Conditions	0.93%	1.34%	2.38%
Cold Shutdown Conditions	5.53%	8.44%	7.03%

(PVNGS Technical Specifications Table 1.1-1 defines Cold Shutdown as a cold-leg temperature of <210EF)

Using unit specific percentages and Equation No. 1 of the TR , the calculated corrosion rate (CR) in mills per year (mpy) are shown below :

Unit 1
CR = (0.9354)(0.4 mpy) + (.0093)(19.0 mpy) +(0.0553)(8.0 mpy) = 0.993 mpy

Unit 2
CR = (0.9022)(0.4 mpy) + (.0134)(19.0 mpy) +(0.0844)(8.0 mpy) = 1.291 mpy

Unit 3
CR = (0.9059)(0.4 mpy) + (.0238)(19.0 mpy) +(0.0703)(8.0 mpy) = 1.377 mpy

Thus, the projected corrosion rate for PVNGS Units 1, 2, and 3 do not exceed the Westinghouse TR corrosion rate of 1.53 mpy.

NRC Requirement 3

Track the time at cold shutdown conditions to determine whether this time does not exceed the assumptions made in the analysis. If these assumptions are exceeded, the licensees shall provide a revised analysis to the NRC, and provide a discussion on whether volumetric inspection of the area is required.

APS Response

APS has confirmed, as stated in response to Question 4.1-2 [Condition 2 above] that from the time of half nozzle implementation until March 2005, the percentage of time in Cold Shutdown conditions is less than the Westinghouse TR analysis value for Unit 1, 2, and 3. The percentage of time in Start-up conditions is less than the Westinghouse TR analysis value for Unit 1 and 2. However, for Unit 3, the percentage of time in Start-up conditions is greater than the Westinghouse TR analysis value.

Unit 3 has been in Start-up conditions 2.38% of the time versus the Westinghouse TR analysis value of 2%. The respective Unit 3 corrosion rate is 1.377 mpy. Based on this corrosion rate, the Unit 3 hot-leg piping will reach the allowable diameter 58 years from the date of the repair. Since the Unit 3 repairs began in 2000, the allowable diameter would be reached in 2058.

Unit 3 began operation in 1988. An anticipated plant life of 60 years will be reached in 2048 (40 year original life plus 20 year life extension). Therefore, the maximum allowable diameter will not be exceeded during the life of the plant. As a result, volumetric inspection of the area is not required.

APS has previously committed to tracking the time at cold shutdown to assure the allowable diameter is not exceeded over the life of the plant (see Attachment 2). [Licensee's letter dated August 16, 2005].

NRC Requirement 4

Calculate the amount of general corrosion-based thinning for the vessels or piping over the life of the plant, as based on the overall general corrosion rate calculated in Step 2 and the thickness of the ferritic vessel or piping that will adjoin to the MNSA repair or half-nozzle repair.

APS Response

The Unit specific corrosion rates are used to calculate the amount of general corrosion over a 60-year period.

Unit 1
Corrosion = (0.000993 inch/year) (60 years)
= 0.05958 inch (radially, relative to penetration)
= 0.11916 inch (diametrically, relative to penetration)
Unit 2
Corrosion = (0.001291 inch/year) (60 years)
= 0.07746 inch (radially, relative to penetration)
= 0.15492 inch (diametrically, relative to penetration)
Unit 3
Corrosion = (0.001377 inch/year) (60 years)
= 0.08262 inch (radially, relative to penetration)
= 0.16524 inch (diametrically, relative to penetration)

NRC Requirement 5

Determine whether the vessel or piping is acceptable over the remaining life of the plant by comparing the worst case remaining wall thickness to the minimum acceptable wall thickness for the vessel or pipe. [The reference to "wall thickness" is actually the penetration diameter.]

APS Response

Unit 1
Diameter of penetration in 60 years = (0.11916 inch) + (1.120 inch) = 1.240 inch

Unit 2
Diameter of penetration in 60 years = (0.15492 inch) + (1.120 inch) = 1.275 inch

Unit 3
Diameter of penetration in 60 years = (0.16524 inch) + (1.120 inch) = 1.285 inch

The allowable diameter is 1.280 inch. The calculated Unit 3 diameter is 1.285 inch over a 60 year period. The respective corrosion rate reflects that the allowable diameter would be reached in 58 years from the date of the earliest repair. Since Unit 3 repairs began in 2000, the allowable diameter would be reached in 2058. With commercial operation beginning in 1988 and the anticipated end of plant life of 60 years being reached in 2048, the maximum allowable diameter will not be exceeded during the life of the plant.

3.4.2.2 Thermal-Fatigue Crack Growth Assessment

Licensees seeking to reference this Westinghouse TR for future licensing applications need to demonstrate that:

NRC Requirement 1

The geometry of the leaking penetration is bounded by the corresponding penetration reported in Calculation Report CN-CI-02-71, Revision 01.

APS Response

APS has reconciled the Westinghouse WCAP-15973-P Revision. 01 analysis and the Westinghouse supplemental analysis provided in APS letter 102-05247, dated April 14, 2005, with the non-Westinghouse analysis used to support previous repairs of RCS hot-leg alloy 600 small-bore nozzles. APS has determined that the Westinghouse WCAP analysis is applicable to the previously repaired hot-leg nozzles in Units 1, 2, and 3. The dimensions of the previously repaired nozzles are bounded by the values in Table 2 below.

The geometry of the leaking penetration identified in the referenced calculation (page 17), [Westinghouse Report CN-CI-02-71, Rev.1, "Summary of Fatigue Crack Growth Evaluation Associated with Small Diameter Nozzles in CEOG Plants", dated 3/31/04] and the PVNGS nozzle geometry is compared below.

	Westinghouse Calc	PVNGS
Base metal thickness:	3.75 in	3.75 in
Inside radius to base metal:	21 in	21 in
Cladding thickness:	0.25 in	0.19 in

Westinghouse has provided APS a plant specific calculation using PVNGS geometry. This calculation evaluates the crack growth for the APS specific hot-leg borehole geometry and concludes that the final crack sizes computed for Palo Verde specific dimensions do not impact the conclusions of the original referenced calculation.

Table 1: Hot-Leg Piping Crack Dimensions from CN-CI-02-71 Rev 01
(Borehole Diameter Used is 0.997")

Depth or Length	Initial (in)	Axial Final (in)	Axial Allowable (in)	Circumferential Final (in)	Circumferential Allowable (in)
Depth	0.938	0.984	> 1.3	1.001	> 1.3
Length	0.762	0.791	> 1.1	0.802	> 1.1

Table 2: Hot-Leg Piping Crack Dimensions using PVNGS Dimensions
(Borehole Diameter Used is 1.120")

Depth or Length	Initial (in)	Axial Final (in)	Axial Allowable (in)	Circumferential Final (in)	Circumferential Allowable (in)
Depth	0.950	0.999	> 1.3	1.017	> 1.3
Length	0.762	0.793	> 1.1	0.805	> 1.1

It can be seen by comparing the final crack sizes in Table 2 with those in Table 1 and those reported in References 1 and 2 [see licensee's submittal for references] that the effect of the change in the initial flaw depth from 0.938" to 0.950" and in the borehole diameter from 0.997" to 1.120" on the final crack sizes is very small and considered insignificant. Final crack sizes computed with the Palo Verde specific dimensions do not impact the conclusions made in References 1 and 2. The symbol > used under the maximum allowable crack sizes in the above tables is to be interpreted as the crack sizes which are still stable under the hot-leg applied loading.

<u>NRC Requirement 2</u>

The plant-specific pressure and temperature profiles in the pressurizer water space for the limiting curves (cooldown curves) do not exceed the analyzed profiles shown in Figure 6-2 (a) of Calculation Report CN-CI-02-71, Revision 01, as stated in Section 3.2.3 of the TR Safety Evaluation (SE).

APS Response

APS is not requesting relief from the ASME requirements for the pressurizer in this request.

NRC Requirement 3

The plant-specific Charpy upper-shelf energy (USE) data shows a USE value of at least 70 ft-lb to bound the USE value used in the analysis. If the plant-specific Charpy USE data does not exist and the licensee plans to use Charpy USE data from other plants' pressurizers and hot-leg piping, then justification (e.g., based on statistical or lower bound analysis) has to be provided.

APS Response

The request of USE data is not applicable to PVNGS since elastic-plastic fracture mechanics was not applied to the hot-leg nozzles in the TR. Furthermore, Palo Verde Units 1, 2 and 3 are bounded by the linear elastic fracture mechanics analysis in Calculation Report CN-CI-02-71, Revision 01, since the highest hot-leg pipe RT_{ndt} amongst all Palo Verde units is 20EF versus the 30EF value used in the Westinghouse TR.

3.4.2.3 Stress Corrosion Crack Growth Assessment

Licensees seeking to implement MNSA repairs or half-nozzle replacements may use the Westinghouse Owners Group's (WOG's) stress corrosion assessment as the bases for concluding that existing flaws in the weld metal will not grow by stress corrosion if they meet the following conditions:

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 part per billion (ppb) for dissolved oxygen, 150 ppb for halide ions, and 150 ppb for sulfate ions.

APS Response

A review of plant chemistry records show that the halide /sulfate concentration levels have been maintained below 150 ppb for chloride, fluoride, and sulfate over the two operating cycles prior to the repair. Oxygen levels are maintained below 10 ppb during power operation and below 100 ppb during plant startups (RCS temperature >250EF). There is no oxygen limit when the RCS temperature is below 250EF.

An RCS hydrogen overpressure of > 15 cc/kg is established prior to criticality (hard hold point) and is maintained in a range of 25 to 50 cc/kg in Modes 1 and 2. In Modes 1 and 2, RCS hydrogen is a Control Parameter with Action Level 1 outside the range of 25 - 50 cc/kg, an Action Level 2, less than 15 cc/kg, and an Action Level 3 less than 5 cc/kg. Chemistry administrative control procedures do

> not allow critical reactor operation with the RCS hydrogen concentration less than 15 cc/kg without immediate corrective action. The nominal operating band for RCS hydrogen is 25 to 50 cc/kg.
>
> Thus the conclusion reached in the Westinghouse TR with respect to stress corrosion cracking, applies to PVNGS.

<u>NRC Requirement 2</u>

During the outage in which the half-nozzle or MNSA repairs are scheduled to be implemented, licensees adopting the TR's 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.

<u>APS Response</u>

> The review identified in the response above was completed for the two operating cycles prior to the repair.

3.4.2.4 Other Considerations

The requirements contained in Section 4.0 of the staff's final SE for the TR WCAP-15973-P, Revision 01, must be addressed, along with the following, when this TR is used as the basis for the corrosion and fatigue crack growth evaluation when implementing a half-nozzle or MNSA repair:

<u>NRC Requirement 1</u>

Licensees using the MNSA repairs as a permanent repair shall provide resolution to the NRC concerns addressed in the NRC letter dated December 8, 2003, from H. Berkow to H. Sepp (ADAMS Accession No. ML033440037), concerning the analysis of the pressure boundary components to which the MNSA is attached, and the augmented inservice inspection program.

<u>APS Response</u>

> APS is not currently planning on using a MNSA as a permanent repair.

<u>NRC Requirement 2</u>

Currently, half-nozzle and MNSA repairs are considered alternatives to the ASME Code, Section XI. Therefore, licensees proposing to use the half-nozzle and MNSA repairs shall submit the required information contained in TR WCAP-15973-P, Revision 01, by the conditions of this SE, to the NRC as a relief request in accordance with 10 CFR 50.55a.

<u>APS Response</u>

> This letter provides APS' response to the conditions of the SE as a relief request in accordance with 10 CFR 50.55a.

4.0 REGULATORY COMMITMENTS

The following table identifies the action committed to by APS through Relief Request No. 31, Revision 1.

Regulatory Commitment	Due Date	Tracking #
APS reaffirms its commitment to continue to track the time at cold shutdown conditions against the assumptions made in the corrosion analysis to assure that the allowable bore diameter is not exceeded over the life of the plant. If the analysis assumptions are exceeded, APS shall provide a revised analysis to the NRC and provide a discussion on whether volumetric inspection of the area is required.	Active - ongoing (no due date)	RCTSAI 2782964

5.0 TECHNICAL EVALUATION

By letter dated March 25, 2005, the licensee submitted a relief request for an alternative repair for ten RCS hot-leg Alloy 600 small-bore nozzles at PVNGS Unit 2. The aforementioned request was approved by the NRC via a letter from the NRC to the licensee dated May 5, 2005 (ADAMS Accession No. ML051290123). As part of the previous relief request, the licensee made a commitment to reconcile the WCAP-15973-P, Revision 01, analysis with the non-Westinghouse analysis used to support the previous repairs of 63 RCS hot-leg Alloy 600 nozzles at PVNGS Units 1, 2, and 3 made between approximately October 1999 and April 2003.

The licensee requested that this alternative remain in effect for the remainder of the initial 40-year licensing period plus a 20-year extension. The NRC staff may only authorize alternatives for the duration of an operating license, and since the NRC has not issued a 20-year extension to the operating licenses for the Palo Verde units (nor has the licensee made such a request), the NRC staff cannot consider any request for authorizing an alternative beyond the current 40 years contained in the operating licenses.

The licensee is requesting relief from Sub-article IWA-4310 of ASME Code, Section XI, 1992 Edition, 1992 Addenda which states in part that the “defects shall be removed or reduced in size in accordance with this Paragraph.” The licensee proposed alternatives to the required flaw characterization (IWA-3300) and successive inspections (IWB-2420). The licensee’s alternative is to not fully characterize/size the existing flaws, rather assume a worst case flaw in the Alloy 600 base and weld material and use the methodology presented in WCAP-15973-P, Revision 01, for determining the following:

1. The overall general/crevice corrosion rate for the internal surfaces of the low-alloy or carbon steel materials that will now be exposed to the reactor coolant and for calculating the amount of time the ferritic portions of the vessel or piping would be acceptable if corrosive wall thinning had occurred.

2. The thermal-fatigue crack-growth life of existing flaws in the Alloy 600 nozzles and/or Alloy 82/182 weld material into the ferritic portion of the vessels or piping.

3. Acceptable bases and arguments for concluding that unacceptable growth of the existing flaw by stress corrosion into the vessel or piping is improbable.

In a letter from the NRC to the WOG dated January 12, 2005, the staff provided its final safety evaluation for the TR. The staff's assessment of the TR concluded that the WOG's methods and analysis were generally acceptable and provide sufficient basis to accomplish the above objectives with respect to implementing the half-nozzle repair. The staff found that licensees could use the methods of the TR as a basis, provided several plant specific questions were submitted to the NRC for review in the areas of general corrosion assessment, thermal-fatigue crack growth assessment, stress corrosion cracking growth assessment, and a few other considerations. Through the licensee's Relief Request No. 31, Revision 1, APS provided the required responses to these questions as detailed in Section 3.4 above.

In the area of general corrosion assessment, the NRC staff finds the licensee's responses meet the requirements of the NRC's final safety evaluation for the TR used at PVNGS Units 1 and 2. The overall general corrosion rate was calculated in accordance with the methodology of the TR with plant-specific operating history for PVNGS Units 1 and 2 and determined to be 0.993 mpy and 1.291 mpy, respectively. The amount of general corrosion-based material loss for the piping over a 60-year period was calculated to be 0.11916-inch diametrically, relative to the penetration for Unit 1 and 0.1549 inch diametrically, relative to the penetration for Unit 2.

The licensee's alternative is acceptable for the remaining initial licensing period of each unit as the worst case penetration diameter, 1.240 inch for Unit 1 and 1.275 inch for Unit 2, are within the maximum acceptable penetration diameter for the pipe (1.280 inch) over a 60-year period. Therefore, in the area of general corrosion assessment, the licensee's proposed alternative is acceptable for PVNGS Units 1 and 2 for the remaining initial licensing period.

In the case of Unit 3, the time in startup conditions is 2.38%, which is greater than the 2.00% assumption used in the TR. According to the TR, startup conditions (including hot standby) provide the greatest rate of corrosion (19 mpy). The overall general corrosion rate was calculated in accordance with the methodology of the TR with the plant-specific operating history for PVNGS Unit 3 and determined to be 1.377 mpy. The amount of material loss due to general corrosion for the piping over a 60-year period was calculated to be 0.16524 inch diametrically, relative to the penetration for Unit 3. This will result in a penetration diameter of 1.285 inch, which is greater than the allowable diameter of 1.28 inch.

The staff notes that although the calculated penetration diameter is greater than that allowed, it was calculated over 60 years. Since the repairs began on Unit 3 in 2000, the maximum diameter would be reached in 2058. Since Unit 3 will reach the end of its 40-year operating license in late 2027, it is reasonable to conclude that under present conditions, the maximum allowable diameter will not be exceeded during the remainder of the 40-year initial licensing period. Therefore, in the area of general corrosion assessment, the licensee's proposed alternative is acceptable for PVNGS Unit 3. The licensee has committed to tracking the time at cold shutdown conditions for PVNGS Units 1, 2 and 3.

In the area of thermal-fatigue crack growth assessment, the NRC staff finds the licensee's responses met the requirements of the NRC's final safety evaluation for the TR. While the geometry of the repaired penetration is not bounded by the corresponding penetration reported in Calculation Report CN-CI-02-71, Revision 01, Westinghouse provided the licensee a plant-

specific calculation using PVNGS geometry which shows sufficient margin of final flaw size to the allowable axial and circumferential flaw sizes of the TR. The plant-specific pressure and temperature profiles were not required for this relief request. The plant-specific Charpy USE data was not applicable. Therefore, in the area of thermal-fatigue crack growth assessment, the licensee's proposed alternative is acceptable.

In the area of stress corrosion cracking growth assessment, the NRC staff finds the licensee's responses met the requirements of the NRC's final safety evaluation for the TR. The licensee demonstrated that a sufficient level of hydrogen overpressure is implemented for the RCS, and the contaminant concentrations in the reactor coolant are typically maintained at levels below 10 ppb for dissolved oxygen, 150 ppb for halide ions, and 150 ppb for sulfate ions. The licensee identified that there is no oxygen limit when the RCS temperature is below 250EF and during startup oxygen levels are maintained below 100 ppb. Both conditions are addressed in the final safety evaluation and WCAP-15973-P, Revision 01, with the increased conservative corrosion rates for startup and low temperature oxygenated conditions. If additional laboratory or field data becomes available that invalidates the TR general corrosion rate values for normal operations, startups and cold shutdown conditions, the WOG will add an addendum to the TR that evaluates the impact of the new data of the corrosion rate values for normal operations, startups and cold shutdown conditions and that provides a new overall general corrosion rate assessment for the ferritic components under assessment. As well, the licensee also completed a review of their plant-specific RCS coolant chemistry histories and confirmed that these chemistry conditions were met over the two operating cycles prior to the repairs. Therefore, in the area of stress corrosion cracking growth assessment, the licensee's proposed alternative is acceptable.

In the area of other considerations, the NRC staff finds the licensee's responses met the requirements of the NRC's final safety evaluation for the TR. The licensee provided the information required by the NRC final safety evaluation report of the TR as noted above as a relief request. The information provided was sufficient to meet the requirements for PVNGS Units 1, 2 and 3, to use the TR as a basis for relief from Sub-article IWA-4310, required flaw characterization (IWA-3300) and successive inspections (IWB-2420) of ASME Code, Section XI, 1992 Edition, 1992 Addenda. Therefore, in the area of other considerations, the licensee's proposed alternative is acceptable.

The licensee requested that this alternative remain in effect for the remainder of the initial 40-year licensing period plus a 20-year extension. For the reason discussed earlier in this SE section, the NRC staff cannot consider any request for authorizing an alternative beyond the current 40 years contained in the operating licenses.

The staff performed shorter term calculations for a 30-year interval which indicate that the total percentage of time the plant would have to be in the cold shutdown condition would have to increase above 25% to reach the maximum acceptable penetration diameter for Unit 2, even if the startup time increased to 2%. For Unit 3, assuming an increase from the current 7.03% to 10% shutdown conditions, the startup time would have to increase from the current value of 2.38% to 8.1% to reach the maximum allowable penetration hole diameter in thirty years. Unit 1 is well within the assumed conditions in the TR. Therefore, the staff finds that there is sufficient margin to absorb small changes in cold shutdown and startup condition times and corrosion rates, within the generally approved ISI relief request for the remainder of the second 10-year ISI interval as well as the third and fourth 10-year ISI intervals.

In summary, the staff's review of the licensee's responses in the areas of general corrosion assessment, thermal-fatigue crack growth assessment, stress corrosion cracking growth assessment, and other considerations, supports the licensee's use of the TR as a bases for Relief Request No. 31, Revision 1.

In general, alternatives to ISI requirements are authorized only for the plant's current 10-year ISI interval. Based on the technical evaluation discussed above, the NRC staff concludes that this alternative is acceptable for the remainder of the second 10-year ISI interval as well as the third and fourth 10-year ISI intervals.

Therefore, pursuant to 10 CFR 50.55a(a)(3)(i) the staff finds Relief Request No. 31, Revision 1, will provide an acceptable level of quality and safety for PVNGS Units 1, 2 and 3 for the second, third, and fourth 10-year ISI intervals, with the regulatory commitment listed in Section 4.0 above.

6.0 CONCLUSION

The NRC staff has reviewed the licensee's proposed Relief Request No. 31, Revision 1, as an alternative repair relief request for twenty seven (27) Unit 1, nine (9) Unit 2, and twenty seven (27) Unit 3 Alloy 600 small-bore RCS hot-leg nozzles in lieu of the ASME Code, Section XI, requirements for required flaw examinations and successive inspections, for the components listed above. Based on the NRC staff's review of the licensee's proposed justification, the staff finds that Relief Request No. 31, Revision 1, will provide an acceptable level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(i), the licensee's alternative repair as stated in Relief Request No. 31, Revision 1, is authorized for Palo Verde Nuclear Generating Station, Units 1, 2, and 3 for the second, third, and fourth 10-year ISI intervals with the licensee's previously accepted regulatory commitment listed in Section 4.0 above.

The licensee requested that Relief Request No. 31, Revision 1, remain in effect for the remainder of the initial 40-year licensing period plus a 20-year extension. The NRC staff can only grant relief for the duration of an operating license, and since the NRC has not granted a 20 year extension to the operating licenses for the Palo Verde units (nor has the licensee made such a request), the NRC staff cannot consider any request for relief beyond the current 40 years stated in the operating licenses.

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

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