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Palo Verde Nuclear
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March 1, 2010

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

Dear Sirs:

Subject: Palo Verde Nuclear Generating Station (PVNGS)
Units 1, 2, and 3
Docket Nos. STN 50-528, 50-529 and 50-530
Response to January 14, 2010, Request for Additional Information Regarding Time Limited Aging Analyses (TLAA) for the Review of the PVNGS License Renewal Application, and License Renewal Application Amendment No. 10

By letter dated January 14, 2010, the NRC issued a request for additional information (RAI) related to the PVNGS license renewal application (LRA). On February 10, 2010, Lisa Regner, NRC Project Manager for PVNGS license renewal, agreed to extend the RAI response due date to March 1, 2010. Enclosure 1 contains APS's response to the January 14, 2010, RAI. Enclosure 2 contains PVNGS LRA updates to reflect changes made as a result of the RAI responses.

APS makes no commitments in this letter. Should you need further information regarding this submittal, please contact Russell A. Stroud, Licensing Section Leader, at (623) 393-5111.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on 03/01/2010
(date)

Sincerely,

JHH/RAS/GAM

A001
NRR

Enclosures:

1. Response to January 14, 2010, Request for Additional Information Regarding Time Limited Aging Analyses (TLAA) for the Review of the PVNGS License Renewal Application

2. Palo Verde Nuclear Generating Station License Renewal Application Amendment No. 10

cc:	E. E. Collins Jr.	NRC Region IV Regional Administrator
	J. R. Hall	NRC NRR Project Manager
	R. I. Treadway	NRC Senior Resident Inspector for PVNGS
	L. M. Regner	NRC License Renewal Project Manager
	G. A. Pick	NRC Region IV (electronic)

ENCLOSURE 1

Response to January 14, 2010, Request for Additional Information Regarding Time Limited Aging Analyses (TLAA) for the Review of the PVNGS License Renewal Application

Section 4.3.2.4 *Pressurizer and Pressurizer Nozzles*

NRC RAI 4.3.2.4-1

Page 4.3-39 of the license renewal application (LRA) discusses the effect of Combustion Engineering Infobulletin 88-09. The applicant states that the revised 40-year design basis cumulative usage factor at the worst location (pressurizer bottom head support skirt) is 0.7223. However, the applicant did not discuss the cumulative usage factor at the worst location for the 60-year plant life and whether the fatigue analysis of the bottom head support skirt is a time-limited aging analysis (TLAA). (a) Provide the cumulative usage factor for the worst location at the pressurizer for the period of extended operation or justify how the bottom head support skirt satisfies the allowable cumulative usage factor of the American Society of Mechanical Engineers (ASME) Code, Section III, at the end of the 60-year plant life. (b) Discuss whether the fatigue analysis of the pressurizer bottom head support skirt is a TLAA.

APS Response to RAI 4.3.2.4-1

The LRA (page 4.3-39) states that the pressurizer bottom head support skirt is the location *worst affected by the Infobulletin 88-09 reanalysis,* not that it is the highest usage factor location as determined by the code analysis of the pressurizer.

Response (a)

In Units 1 and 2, the highest-CUF locations for a fatigue TLAA are the short heater sleeve plugs (Table 4.3-7, line 15), with a CUF of 1.0 for the stated basis. In Unit 3, the highest-CUF location for a fatigue TLAA is the spray nozzle and safe end with overlay repair (Table 4.3-7, line 20), with a 40-year calculated CUF of 0.9923.

These values indicate that 60-year usage factors, calculated on the same bases, would exceed the code allowable of 1.0. 10 CFR 54.21(c)(1)(i) validation of these TLAAs would not be possible on this basis. Disposition of these TLAAs for the period of extended operation, therefore, depends on 10 CFR 54.21(c)(1)(iii) aging management using the enhanced fatigue management program.

Response (b)

All fatigue analyses of the pressurizer and its subcomponents are TLAAs, except those already extended to a 60-year design life under analyses for the current licensing basis (see 10 CFR 54.3(a), Criterion 3), and those are not the basis for a safety determination (Criterion 4). The fatigue analysis of the bottom head support skirt is, therefore, a TLAA.

NRC RAI 4.3.2.4-2

Page 4.3-39 of the LRA discusses two flaws detected in the pressurizer support skirt forging weld. (a) The applicant stated that the fatigue crack growth analysis predicted growth from the as-found size of 0.59 inch to 0.6921 inch over the design life. Discuss how many years are assumed for the design life and are assumed in the fatigue crack growth calculation. (b) If the 0.6921 inch flaw size is not calculated for the 60-year plant life, calculate the final crack size at the end of 60 years or justify the structural integrity of the support skirt forging weld to the end of the 60 years.

APS Response to RAI 4.3.2.4-2

Response (a)

The fatigue crack growth analysis has shown that no repair is required for continued operation over the service life of the vessel (Section 3.0 of the crack growth analysis), which, at the time of the analysis (1993), was understood to be 40 years.

Response (b)

A 60-year analysis is not required under the license renewal rule if the aging effect is managed, as described in the "Disposition" subheading. The disposition subheading describes aging management separately for this analysis because it calculates crack growth rather than fatigue usage factor.

NRC RAI 4.3.2.4-3

Page 4.3-39 of the LRA states that power uprate and steam generator replacement have no effect on the design reports for any of the three units. (a) Reference the design reports associated with pressurizer and pressurizer nozzles that have been reviewed to determine the impact of power uprate and steam generator replacement. Describe them briefly in the context of Section 4.3.2.4. (b) Clarify whether the loadings on the pressurizer and pressurizer nozzles are affected by the power uprate and steam generator replacement.

APS Response to RAI 4.3.2.4-3

Response (a)

The power uprate and replacement steam generator evaluation reports are in the following submittals:

1. APS Letter No. 102-04641 to NRC, dated December 21, 2001, "Palo Verde Nuclear Generating Station (PVNGS) Unit 2, Docket No. STN 50-529, Request for a License Amendment to Support Replacement of Steam Generators and Uprated Power Operations" (Agencywide Document Access and Management System [ADAMS] Accession No. ML013650419).

2. APS Letter No. 102-05116 to NRC, dated July 9, 2004, "Palo Verde Nuclear Generating Station (PVNGS) Unit 1 and 3, Docket Nos. STN 50-528, STN 50-529, and STN 50-530, Request for a License Amendment to Support Replacement of Steam Generators and Uprated Power Operations in Units 1 and 3, and Associated Administrative Changes for Unit 2" (ADAMS Accession No. ML042010289).

The power uprate licensing reports are in Attachment 4 to Ref. 1 and Attachment 6 to Ref. 2. The reports did not explicitly cite or revise the analyses of record. Instead, these reports reviewed the supporting design transients, as documented in the code design specifications for nuclear steam supply system structures, systems, and components (SSCs). The Units 1 and 3 report (Attachment 4 to Ref. 2, Section 3, "Design Transients") cites the Unit 2 report Section 3 "unchanged." The following conclusion regarding the Unit 2 SSC design transients review is in Section 3.1.4 of Attachment 6 to Ref. 1:

> In conclusion, the original design transients are more limiting than the corresponding limiting calculated transients associated with PUR. Hence, the original SSC design specifications remain bounding and bound the new operating conditions associated with PUR.

The analyses of record for the pressurizer were—

3. Combustion Engineering Report CENC-1336. J. Halliday et al. *Analytical Report for Arizona Unit 1 Pressurizer.*

4. Combustion Engineering Report CENC-1395. J. Halliday et al. *Analytical Report for Arizona Unit 2 Pressurizer.*

5. Combustion Engineering Report CENC-1490. J. Halliday et al. *Analytical Report for Arizona Unit 3 Pressurizer.*

—plus, in each case, a number of amendments and addenda (then current), are either attached or incorporated by reference.

<u>Response (b)</u>

The power uprate and replacement steam generator evaluations (Attachment 4 to Ref. 1 and Attachment 6 to Ref. 2) demonstrated that the loadings on the pressurizer remain within the conditions assumed for the analysis of record: "The design requirements for

this component remain bounding.... Therefore, the AOR [analysis of record] remains bounding" (Attachment 4 to Ref. 1 §5.6, cited by Attachment 6 to Ref. 2 §5.6). In other words, loads on the pressurizer and its nozzles remained less than or equal to those used for the analyses of record, and, therefore, no changes to the analyses of record were necessary.

NRC RAI 4.3.2.4-4

Page 4.3-39 of the LRA, last paragraph, states that the original stress and fatigue analysis of the surge nozzle have been superseded by the reanalysis for a compressive overlay, which included the thermal stratification and insurge-outsurge effects. Submit the reanalysis. Alternatively, reference the reanalysis and describe the analysis input, method, results, and acceptance criteria in detail, demonstrating that the structural integrity of the surge nozzle will be maintained to the end of 60 years.

APS Response to RAI 4.3.2.4-4

The surge nozzle compressive overlay reanalysis is:

> SIA, Inc. Calculation PV-23Q-315. "Section III Code Evaluation of Pressurizer Surge Nozzle with Weld Overlay Repair." Rev. 0. Contains vendor proprietary information, May 18, 2007.

SIA has released the design CUFs as non-proprietary. However the document contains vendor proprietary information, including methods.

The results of this evaluation are summarized under "Surge Nozzle" on LRA Page 4.3-43:

> *Surge Nozzle:* APS evaluated effects of the weld overlay repairs on the pressurizer surge nozzle. The worst-case projected usage factor for a 60 year lifetime, that is, for 1.5 times 40 year cycles, is 1.4402616 (U40=0.9602) at the end of the overlay on the outside surface of the nozzle. However, the surge nozzle will be monitored for fatigue usage, and the fatigue CUF will not exceed the code limit of 1.0 so long as the number of applied load cycles does not exceed the number specified by the design specification for this nozzle, and used in the analysis. The analysis includes effects of thermal stratification and insurge-outsurge.

The disposition of this TLAA for the period of extended operation is, therefore, aging management, in accordance with 10 CFR 54.21(c)(1)(iii).

Please see also the responses to RAIs 4.3.2.4-11 and 4.3.2.4-14.

NRC RAI 4.3.2.4-5

Page 4.3-40 of the LRA discusses the fatigue crack growth in the original heater sleeve attachment welds. (a) Discuss the postulated initial cracks in the original sleeve-to-innerwall attachment welds. (b) Discuss the projected final flaw size for the postulated cracks at the end of 60 years. (c) Discuss the allowable flaw size. (d) Discuss the results and describe the methodology used in the "subsequent" report and "code design" reports that are mentioned in the second paragraph on page 4.3-40. (e) Provide the references of these reports.

APS Response to RAI 4.3.2.4-5

Response to Request (a)

The postulated initial crack size in the crack growth analyses was a flaw size of 0.6 inches.

Response to Request (b)

The projected final flaw size at the end of 60 years is 1.16 inches.

Response to Request (c)

The fracture mechanics analysis permits an allowable flaw size greater than 1.2 inch.

Response to Request (d)

The fatigue crack growth analysis described in these paragraphs of the LRA was performed in support of the temporary mechanical nozzle seal assembly (MNSA) repairs to three Unit 3 pressurizer heater sleeves, and was performed for a 60-year period (Ref. 4). As a result of the replacement of all heater sleeves with the half nozzle method, the fatigue crack growth analysis of the remnants, performed in support of the MNSA repairs, has been superseded by analyses performed to support the heater sleeve replacement modification and associated Relief Request 29.

- Reference 5 is the "subsequent report" originally referred to in this paragraph of the LRA. It reports results of the fatigue crack growth analysis supporting Relief Request 29 (Refs. 6 and 7). This request for relief from ASME Section XI, IWA-3300, IWA-4310, IWB-2420, IWB-3242.4, and IWB 3610 inspection requirements for the pressurizer heater sleeve and Alloy 82/182 remnants was approved by an NRC letter and safety evaluation of November 5, 2004 (Ref. 10).

- Reference 9 and its supporting fatigue crack growth calculation (Ref. 8) support the modification package that installed the replacement heater sleeves by demonstrating acceptable fatigue crack growth without removing sleeve

remnants. This report and calculation were included by reference in the ASME Section III pressurizer code design reports of all three units (Refs. 1, 2, and 3).

The results of these analyses are essentially identical (see responses above to RAI 4.3.2.4-5 (a), (b), and (c)). These superseding analyses are applicable to all three units.

A subsequent evaluation of effects of the Unit 3 lower head overheating events determined that the calculated increase in the growth of the postulated defect would be a negligible $4.44X10^{-5}$ inches (Ref. 11); and this evaluation was, therefore, not included in the Unit 3 code design report. (This evaluation applied only to the brief period of time for which the Unit 3 pressurizer lower head was subject to overheating, and is, therefore, not a TLAA.)

The currently-applicable analyses (Refs. 5 and 8) used finite-element models to calculate stress intensity factors for postulated flaws for various operating conditions, and calculated the crack growth per cycle for the significant contributors to crack growth, using the methodology of ASME Section XI (1992) for a water environment. The analyses then determined the maximum permissible crack size for which the most-limiting allowed stress intensity would not be exceeded, based on the assumption that the applied stress intensity factor is proportional to the square root of the crack dimension. The crack growth was calculated as the sum of the products of lifetime design cycles times their respective crack growth increments per cycle.

These analyses assume a 60-year design life, and are, therefore, not TLAAs, by 10 CFR 54.3(a), Criterion (3).

The currently-applicable analyses (Refs. 5, 8, and 9), and supporting calculations, were incorporated by reference, unchanged, in the three pressurizer code design reports.

The pressurizer code design reports use standard ASME III Subsection NB methods. The code design reports are References 1, 2, and 3.

LRA Changes

Amendment 10 will correct the heading at the top of LRA page 4.3-40, and two paragraphs below it, to read:

Absence of TLAAs in the Analysis of Thermal Fatigue Crack Growth in Original Heater Sleeve Attachment Welds, in Support of ~~MNSA~~ ASME Section XI Inspection Relief, and of Heater Sleeve Repairs ~~of Unit 3 Heater Sleeves~~

There are currently no mechanical nozzle seal assemblies (MNSAs) in use at PVNGS. Three MNSAs were used as a temporary means of sealing U~~u~~nit 3 pressurizer heater sleeves. ~~However the MNSAs were replaced with half-nozzle repairs during unit 3 refueling outage 3R11.~~

A supporting Westinghouse ~~linear elastic fracture mechanics~~ fatigue crack growth analysis for the Unit 3 MNSA repairs, for postulated cracks in the original sleeve-to-inner-wall attachment welds, was based on a 60-year design life, and was therefore not a TLAA. ~~Although t~~These MNSAs were ~~replaced,~~ removed and all heater sleeves in all three units were replaced, using half-nozzle repairs. The Westinghouse ~~this~~ analysis in support of this temporary MNSA modification ~~is still applicable, and is cited~~ was superseded by:

- A~~a~~ subsequent report and fatigue crack growth analysis in support of relief from ASME Section XI inspection requirements, ~~incorporated by reference in the code design reports,~~ because the area of the postulated initial cracks - at the original attachment J-welds - has not been removed.

- A subsequent report and fatigue crack growth analysis in support of the half-nozzle repair of all 36 heater sleeves in each of Units 1, 2, and 3. This report and analysis are included by reference in the Unit 1, 2, and 3 pressurizer code design reports.

The analyses are consistent. Both report a projected final flaw size at 60 years of 1.16 inch, based on an initial flaw size of 0.6 inch, and cite supporting fracture mechanics analyses permitting an allowable flaw size of 1.2 inch. Both of these currently-applicable fatigue crack growth analyses apply to all three units. Both were performed for a 60-year operating life, and are therefore not TLAAs.

Response to Request (e)

References

The ASME III code design reports for the pressurizer are:

1. Combustion Engineering Report CENC-1336. J. Halliday et al. *Analytical Report for Arizona Unit 1 Pressurizer.*

2. Combustion Engineering Report CENC-1395. J. Halliday et al. *Analytical Report for Arizona Unit 2 Pressurizer.*

3. Combustion Engineering Report CENC-1490. J. Halliday et al. *Analytical Report for Arizona Unit 3 Pressurizer.*

—plus, in each case, a number of amendments and addenda that are either attached or incorporated by reference.

Other references for the pressurizer heater sleeve repairs:

4. Westinghouse Calculation Note CN-CI-03-31. "Thermal Fatigue Crack Growth Analysis in Support of MNSA Repair of Palo Verde Units 1, 2, & 3 Pressurizer Heater Sleeves." Rev. 1. Westinghouse Proprietary Class 2. April 25, 2003.

5. SIA Report SIR-04-045. *Technical Report Supporting the Palo Verde Pressurizer Heater Sleeve Mid-Wall Repair.* Rev. 0. Palo Alto, CA: Structural Integrity Associates, Inc., June 2004. (A copy is attached to Ref. 6)

6. APS Letter No. 102-05112 to NRC, "Palo Verde Nuclear Generating Station (PVNGS) Units 1, 2 and 3, Docket No. STN 50-528, 50-529 and 50-530, 10 CFR 50.55a Alternative Repair Requests for the PVNGS Pressurizers: Relief Requests 28 and 29." June 15, 2004. (ADAMS Accession No. ML041750296)

   With *Enclosure 1, Relief Request 28, "Ambient Temperature Temper Bead Welding For Pressurizer Half-Sleeve replacement,"*

   —and *Enclosure 2, Relief Request 29, "Remnant Sleeve(s) Flaw Evaluation."* (Including Ref. 5 as Attachment 1).

7. APS Letter No. 102-05141 to NRC, "Palo Verde Nuclear Generating Station (PVNGS) Units 1, 2 and 3, Docket No. STN 50-528, 50-529 and 50-530, Response to Request for Additional Information – Relief Requests 28 and 29." August 24, 2004. (ADAMS Accession No. ML042450041)

   With enclosed *Response to the Request for Additional Information - Relief Requests 28 and 29,*

   —and attached reformatted SIA Technical Report SIR-04-045, Revision 0 (Ref. 5).

8. SIA Calculation PV-04Q-318. "Fatigue Crack Growth Analysis of Palo Verde Pressurizer Heater Sleeve Revised Outside Diameter Repair." Rev. 0. October 1, 2004. For a Unit 3 addendum see Ref. 11. Incorporated by reference into Refs. 1, 2, and 3. Support for Ref. 9.

9. SIA, Inc. Report SIR-04-011. *Fracture Mechanics Evaluation of Potential Remnant Crack in Palo Verde Pressurizer Heater Sleeve Nozzles.* Rev 1. October 1, 2004. Incorporated by reference into Refs. 1, 2, and 3. Concluding report. Cites Ref.8.

10. NRC Letter to APS, "Palo Verde Nuclear Generating Station, Units 1, 2, and 3, Relief Request No. 29, RE: Remnant Sleeve(s) Flaw Evaluation (TAC Nos. MC3606, MC3607, and MC3608)." November 5, 2004. (ADAMS Accession No. ML043130170)

With enclosed *Safety Evaluation by the Office of Nuclear Reactor Regulation, Inservice Inspection Program Relief Request No. 29, 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.*

11. SIA, Inc. Letter Report SIR-05-232 and CJF-05-001. "Failed Heater Element Event Crack Growth." Unit 3 addendum to SIA Calculation PV-04Q-318 Rev. 0 (Ref. 8). July 29, 2005. Additional growth for 75 cycles experienced is $4.44X10^{-5}$ inches.

NRC RAI 4.3.2.4-6

Page 4.3-40 of the LRA, first paragraph, states that for the half nozzle repair method, the original sleeve-to-inner wall attachment welds (i.e., J-groove welds) are analyzed for 60 years and, therefore, are not a TLAA. However, other welds and parts were used for the half nozzle repair. For example, welds were used to join the half nozzle to the weld pad. Discuss why the fatigue analysis of the half nozzles and associated new attachment welds, and weld pads are not considered as a TLAA and not discussed in this subsection.

APS Response to RAI 4.3.2.4-6

The fatigue analyses of the pressurizer heater sleeve half nozzle repairs for all three units were evaluated for a period of 60 years. These analyses are, therefore, not TLAAs, by 10 CFR 54.3(a) Criterion 3. The 60-year usage factors calculated by these repair analyses can be found in LRA Table 4.3-7, Item 16 for Unit 2, and Item 17 for Units 1 and 3.

(Note: LRA Table 4.3-7 Item 16 is an analysis intended to be applicable to all three units, as indicated by this Item 16. However this repair was installed only in the Unit 2 pressurizer. Item 16 is, therefore, a design basis analysis for Unit 2 only. Item 17 is the equivalent analysis for Units 1 and 3.)

NRC RAI 4.3.2.4-7

In the LRA, page 4.3-40, 5th paragraph states, "The analysis of the weld pads does not explicitly supersede the results of the fatigue analysis with the tapped anchor holes. Therefore, both fatigue analysis results apply." (a) Discuss whether the anchor holes and weld pads on the pressurizer bottom are a TLAA. (b) Clarify the above two statements in the context of TLAA. (c) Clarify whether the cumulative usage factors for the anchor bolts and weld pads for 60 years are within the allowable cumulative usage factor of 1.0.

APS Response to RAI 4.3.2.4-7

Response (a)

The anchor hole analysis was for a 40-year design basis set of cycles and is, therefore, a TLAA (Table 4.3-7, Item 12). The fatigue analysis of the weld pads was for a 60-year design basis set of cycles and is, therefore, not a TLAA (Table 4.3-7, Item 17).

(Note: Table 4.3-7 Item 16 is an analysis intended to be applicable to all three units, as indicated by this Item 16. However this repair was installed only in the Unit 2 pressurizer. Item 16 is, therefore, a design basis analysis for Unit 2 only. Item 17 is the equivalent analysis for Units 1 and 3.)

Response (b)

The statement, "The analysis of the weld pads does not explicitly supersede the results of the fatigue analysis with the tapped anchor holes. Therefore, both fatigue analysis results apply," means that the fatigue analysis that evaluated effects of the anchor holes was not included in the otherwise-superseding analysis of the weld pads that overlaid them, and, therefore, that the results of both analyses are applicable to the safety determination of the pressurizer pressure boundary. Because the analyses affect adjacent but different locations, the calculated usage factors are not cumulative. That is, the worst-case usage factor is the worst of the two, not the sum of the two, and either analysis may, therefore, be limiting.

Response (c)

The 40-year Unit 3 anchor hole analysis resulted in a maximum CUF of 0.443 [Table 4.3-7 Item 12], which, projected to the end of the period of extended operation, would be 0.6645. The 60-year Units 1 and 3 weld pad analysis resulted in a maximum CUF of 0.551 [Table 4.3-7 Item 17]. Both 60-year results applicable to Unit 3 are, therefore, less than 1.0.

NRC RAI 4.3.2.4-8

In the LRA, page 4.3-42, 5th paragraph states that because the pressurizer heaters were subjected to a high temperature of 779 degrees F for 3,700 hours; the evaluation of the creep effects is not a TLAA. Explain why the evaluation of the creep effect is not a TLAA when heaters are subject to a temperature of 779 degrees F for 3,700 hours. Reference technical paper(s) or report(s) to support your technical basis.

APS Response to RAI 4.3.2.4-8

The evaluation of the creep effect is not a TLAA because the effect was terminated and does not continue into the period of extended operation. The evaluation of the creep

effect is, therefore, not a TLAA because its assumptions are not "...time-limited [as] defined by the current operating term..." (10 CFR 54.3(a) Criterion 3), and the evaluation and its conclusions will, therefore, not change with a change in the licensed operating period.

Please see the response to RAI 4.3.2.4-9 for effects of this temperature excursion on the code fatigue analysis of the pressurizer lower head, which *is* a TLAA.

The evaluation of the creep effects is described in Enclosure 1 of—

> APS Letter No. 102-05301 to APS, dated June 28, 2005, "Revision 1 to 10 CFR 50.55a(3)(i) Request for Alternatives to 10 CFR 50.55a(c) Requirement to Comply with ASME Section III, Subsection NB-1120, 'Temperature Limits,' for a Portion of the PVNGS Unit 3 Pressurizer that was Subjected to Heating Above Code Parameters (ISI Relief Request No. 33)" (ADAMS Accession No. ML051890145).
>
> With Enclosure 1, *Revision 1 to 10 CFR 50.55a(a)(3)(i) Request for Alternative to 10 CFR 50.55a(c) Requirement to Comply with ASME Section III, Subsection NB-1120, "Temperature Limits" for a Portion of the PVNGS Unit 3 Pressurizer.*

NRC RAI 4.3.2.4-9

In the LRA, page 4.3-42, states, "However, overheating did affect the code fatigue analysis." Clarify the intent of this statement in the context of the TLAA for the pressurizer heaters.

APS Response to RAI 4.3.2.4-9

The 2005 overheating event did not affect the currently-installed heaters or their code fatigue analyses, because the affected heaters were replaced, but did affect the code fatigue analysis of the Unit 3 pressurizer lower head.

The effects of the overheating on the code fatigue analysis were evaluated for a 60-year design life, as summarized in LRA Table 4.3-7, Item 18. Since this addendum to the design report was performed for a 60-year design life, PVNGS did not classify it as a TLAA. However the affected code design report is a TLAA.

NRC RAI 4.3.2.4-10

In the LRA, page 4.3-42, last paragraph and page 4.3-43, first paragraph, discuss fatigue analysis revisions due to pressurizer nozzle overlays. The applicant states that the fatigue crack growth analyses do not support safety determinations for a defined

design lifetime and are, therefore, not TLAAs. The applicant stated further, "However, the revised fatigue analyses of the adjacent materials affected by the overlays are time-dependent and are TLAAs unless successfully projected to the end of the period of extended operation." (a) Clarify whether the cumulative fatigue usage factor calculation and fatigue crack growth calculation were calculated for the adjacent materials to the end of 60 years. (b) Submit the revised fatigue analyses for the adjacent materials or describe in detail the analysis input, methodology, acceptance criteria, and result. (c) Identify all the pressurizer nozzles in all three units that have been weld overlaid and identify "the adjacent materials" affected by the weld overlays. (d) The applicant states that the adjacent materials analyses are TLAA. Discuss the actions that will be taken as a result of the TLAA determination.

APS Response to RAI 4.3.2.4-10

Response (a)

Fatigue Crack Growth: ASME Section XI IWB-3640 crack growth analyses address potential growth of cracks in the susceptible dissimilar weld metal overlay region, but no fatigue crack growth analysis was performed for materials *adjacent to* the overlay. The fatigue crack growth analysis pertains to original materials *under* the overlay.

As stated in the LRA, page 4.3-42, "The ... fatigue crack growth analyses ... assume 1.5 times the design basis number of events assumed for 40 years...." In effect the analyses assume a 60-year design life.

The flaw growth analyses projected flaw sizes either to the end of 60-year design life or to the next scheduled inservice inspection, or determined the maximum time permitted between inservice inspections, as required to support the safety determination. In none of these cases did the flaw evaluation support a safety determination for a time period defined by the current licensed operating term (40 years), and, therefore, the flaw evaluation was not a TLAA in any of these cases.

Cumulative Usage Factor: ASME Section III fatigue analyses were performed only for regions adjacent to the overlay not affected by cracks or assumed cracks. Cumulative usage factors were calculated adjacent to the overlay on a similar (i.e., 60-year) basis. Therefore, the cumulative usage factor analysis that met the ASME code fatigue usage factor limit of 1.0 for 60 years is not a TLAA. Calculations for fatigue in the surge and spray nozzles that did not meet the 60-year life, and qualified the location for only 40 years, are TLAAs. See (d) below.

Response (b)

The revised fatigue analyses of adjacent materials are cited below. These analyses contain vendor proprietary information.

1. SIA, Inc. Calculation PV-23Q-315. "Section III Code Evaluation of Pressurizer Surge Nozzle with Weld Overlay Repair." Rev. 0. Contains vendor proprietary information. May 18, 2007. SIA has released the design CUFs as non-proprietary.

2. SIA, Inc. Calculation PV-23Q-325. "Section III Code Evaluation of Pressurizer Spray Nozzle with Weld Overlay Repair." Rev. 0. Contains vendor proprietary information. May 18, 2007. SIA has released the design CUFs as non-proprietary.

3. SIA, Inc. Calculation PV-23Q-335. "Section III Code Evaluation of Pressurizer Safety Nozzle with Weld Overlay Repair." Rev. 0. Contains vendor proprietary information. May 18, 2007. SIA has released the design CUFs as non-proprietary.

Response (c)

See the succeeding paragraphs on LRA page 4.3-43, which describe results of the analyses of the affected pressurizer surge, spray, and safety valve nozzles; and LRA Table 4.3-7 Lines 19, 20, and 21, which show applicability of these analyses to all three units. The adjacent materials are those determined by the analyses to have effects on their design stresses due to the overlays. The analyses identify the highest usage factor locations in detail, and these are briefly described in these LRA paragraphs.

Response (d)

No action is required for the safety valve nozzles, which were successfully analyzed for a 60-year life. See LRA Table 4.3-7 Line 21. This analysis is, therefore, not a TLAA, by 10 CFR 54.3(a), Criterion 3.

Two calculations for fatigue in the surge and spray nozzles for a 60-year life did not render the desired resuts, and demonstrated an acceptable fatigue usage for only the design basis number of events assumed for 40 years. See LRA Table 4.3-7 Lines 19 and 20, and Note 6. Fatigue in these locations will, therefore, be managed for the period of extended operation, as stated in the "Disposition."

NRC RAI 4.3.2.4-11

In the LRA, page 4.3-43, the second and third paragraphs discuss the fatigue usage factors for the surge and spray nozzles for a 60-year life as being 1.44 and 1.49, respectively. The applicant states that the surge nozzle is monitored for fatigue usage and the fatigue usage factor will not exceed the code limit of 1.0 as long as the number of applied load cycles does not exceed the number specified by the design specification for this nozzle and used in the analyses. (a) Clarify whether the spray nozzle will be monitored for fatigue usage. (b) Discuss why a plastic analysis was not performed in

accordance with NB-3228 of the ASME Code, Section III when the cumulative usage factors for surge and spray piping exceed 1.0 which were calculated by the elastic analysis of NB-3222. Section 4.3.2.9 discussed a plastic analysis performed by Combustion Engineering on the surge line that lowered the cumulative usage factor.

APS Response to RAI 4.3.2.4-11

Response (a)

The spray nozzle will be monitored. See LRA Table 4.3-4 Line 17.

Response (b)

The fatigue management program will maintain the usage factor of the surge and spray nozzles (not "piping," in this case) at less than 1.0 for the period of extended operation, or ensure that other acceptable actions maintain the basis of the safety determination. A plastic analysis was, therefore, not required.

LRA Section 4.3.2.9 (and Section 4.3.4) reported the plastic analysis of the surge line because it has been performed in support of the safety determination for the current licensed operating term, and is, therefore, the basis for a TLAA.

NRC RAI 4.3.2.4-12

Table 4.3-7 of the LRA summarizes the results of fatigue usage factors.
Discuss the actions that will be taken if an item requires the TLAA as shown in the "TLAA" column.

APS Response to RAI 4.3.2.4-12

The column in LRA Table 4.3-7 titled "TLAA?" does not indicate that an item *may require* a TLAA, but indicates whether the listed analysis *is* a TLAA requiring disposition. The dispositions are briefly described under the "Disposition" subheading.

NRC RAI 4.3.2.4-13

In the LRA, Table 4.3-7, item 2, provides no cumulative usage factors for the pressurizer support skirt forging weld. It is stated that crack growth of the flaw in the skirt forging weld is less than the critical size for the assumed number of lifetime cycles. Clarify whether the "lifetime" cycles implies that the transient cycles used in the fatigue analysis are for 40 years or 60 years of plant life. If crack growth of the flaw in the skirt forging weld is calculated based on the 40 year cycles, perform a crack growth calculation using the 60 year cycles or justify why the 40 year crack growth results are acceptable for the period of extended operation.

APS Response to RAI 4.3.2.4-13

LRA Table 4.3-7, item 2, provides no cumulative usage factors for the pressurizer support skirt forging weld because item 2 is the fatigue crack growth analysis of flaws in this location, in Unit 2 only, and this analysis did not calculate a usage factor. The pertinent highest adjacent calculated fatigue usage factor is described in item 1, "Bottom Head and Support Skirt – Worst Location, Inside the Support Knuckle." (The reference in Table 4.3-7, item 1, to "item 17 below" is a typographical error. It should refer to "item 18 below." This has been corrected in LRA Amendment 10 in Enclosure 2.)

The Class 1 analyst determines the locations for which fatigue usage factors are calculated, consistent with code guidance. No usage factor was calculated for the forging weld. The fatigue crack growth analysis of these forging weld defects is not a code fatigue analysis and did not calculate a fatigue usage factor.

The response to RAI 4.3.2.4-2 describes how fatigue crack growth effects evaluated by the 40-year crack growth analysis will be managed, to ensure that they will remain acceptable for the period of extended operation.

NRC RAI 4.3.2.4-14

On page 4.3-47 of the LRA, under the Disposition section, the applicant identified the Metal Fatigue of Reactor Coolant Pressure Boundary Program as the program to manage the crack in the support skirt forging weld. There are many pressurizer components listed in Table 4.3-7 that have been identified as requiring TLAA and should be required to be monitored by the metal fatigue aging management program. However, these pressurizer components are not identified in the Disposition section on page 4.3-47. Explain why the affected components in Table 4.3-7 are not included in the Disposition Section on page 4.3-47.

APS Response to RAI 4.3.2.4-14

The "Disposition" at the end of each LRA Chapter 4 section pertains to all TLAAs in the section, except those for which individual dispositions are described. Calculated fatigue usage factors in the pressurizer will be managed by the Metal Fatigue of Reactor Coolant Pressure Boundary Program, in accordance with 10 CFR 54.21(c)(1)(iii).

Disposition of the fatigue crack growth analysis of the indications in the Unit 2 support skirt forging weld was described separately because fatigue crack growth analyses do not result in calculated fatigue usage factors. Please see the response to RAI 4.3.2.4-13. Because no fatigue usage factor was calculated by this analysis, aging management of this fatigue crack growth analysis is restricted to cycle counting.

NRC RAI 4.3.2.4-15

In the LRA, page 4.3-47, "Fatigue Analysis," states that the Metal Fatigue of Reactor Coolant Pressure Boundary program will ensure that the fatigue usage factors based on those transient events that will remain within the ASME Code limit of 1.0 for the period of extended operation or will ensure that appropriate reevaluation or other of corrective actions before the design basis number of these events is exceeded. Identify the exact pressurizer components that will be reevaluated in the context of Section 4.3.2.4.

APS Response to RAI 4.3.2.4-15

A subcomponent will be reevaluated or reanalyzed upon determination by the aging management program that (1) fatigue in the subcomponent has reached an action limit that requires mitigation of fatigue effects, and (2) that reevaluation or reanalysis is the preferred mitigation (among other possible corrective actions, including repair, replacement, or a fatigue crack growth analysis plus augmented inspection). The exact pressurizer components that will be reevaluated will therefore be determined by the state of fatigue effects in the pressurizer at that time, as tracked by the fatigue management program, and upon review of the affected Class 1 pressurizer analyses.

See LRA Section 4.3.1 and Appendix B3.1 for a further description of the fatigue management program, including locations to be monitored and the bases for action limits.

Section 4.3.2.7 *ASME III Class 1 Piping and Piping Nozzles*

NRC RAI 4.3.2.7-1

On page 4.3-56 of the LRA, the applicant found that reactor coolant system (RCS) piping, nozzles, RTD thermowells, and other Class 1 piping satisfy the current licensing basis design number of transients under power uprate and steam generator replacement. However, the applicant did not discuss whether the piping components have been analyzed for 60 years. (a) Clarify whether these piping components satisfy the allowable cumulative usage factor of 1.0 using a projected number of transients at the end of 60 years. (b) If this calculation has not been performed, perform a fatigue usage factor analysis for 60 years, or justify how the cumulative fatigue usage factors for all Class 1 piping and associated components satisfy the allowable of 1.0 per the ASME Code, Section III at the end of 60 years.

APS Response to RAI 4.3.2.7-1

Response (a)

The RCS piping, nozzle, RTD thermowell, and other Class 1 piping component analyses that were not based on a 60-year life are TLAAs.

Response (b)

The disposition of these TLAAs depends on the fatigue management program to ensure that the 40 year design numbers of transients will not be exceeded during 60 years of operation without appropriate corrective actions. These TLAAs will, therefore, be managed for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

NRC RAI 4.3.2.7-2

In the LRA, page 4.3-56, last paragraph, states that the Metal Fatigue of Reactor Coolant Pressure Boundary Program will calculate stress-based fatigue in the chemical and volume control system (CVCS) charging nozzle. Describe the analysis in detail, such as the analysis input, analytical procedures and method, acceptance criteria, and results.

APS Response to RAI 4.3.2.7-2

This statement refers to the aging management method which will be used for the charging nozzle. The calculation has not yet been performed.

The future Stress-Based Fatigue monitoring program is discussed in the Disposition of LRA Section 4.3.2.7, and in LRA Section 4.3.1, as follows:

> Stress-based fatigue (SBF) monitoring will compute a "real time" stress history for a given component from actual temperature, pressure, and flow histories. SBF is intended for those high-fatigue components where a more refined approach is necessary to show long-term structural acceptability. SBF monitoring depends on "global-to-local" correlation or "transfer" functions which calculate local transient pressures and temperatures from data collected by the limited number of plant instruments, and from them, local stresses and fatigue usage.

Stress-Based Fatigue monitoring method (SBF) is an enhancement to the PVNGS Fatigue Management Program. The analysis details, such as the analysis input, analytical procedures and method, and acceptance criteria have not yet been finalized. The development of the stress-based fatigue monitoring method is a licensing commitment. For stress-based fatigue monitoring, APS has committed to the use of a fatigue monitoring software program that incorporates a three-dimensional, six-element model meeting ASME III NB-3200 requirements, and committed to the implementation of this method for SBF monitoring at least two years prior to the period of extended operation. See updated Commitment 39 in Table A4-1 in LRA Amendment No. 9, submitted in APS letter no. 102-06134, dated February 19, 2010.

NRC RAI 4.3.2.7-3

Page 4.3-57 of the LRA discusses the reduced wall thickness in the RCS hot leg and cold leg piping. (a) Clarify why the fatigue usage factor for the RCS hot and cold leg piping with reduced wall thickness was not calculated for 60 years. (b) List all Alloy 82/182 welds in the RCS hot leg and cold leg piping. (c) Discuss whether there are any indications/flaws detected in the Alloy 82/182 welds that remained in service. (d) If there are flaws in the Alloy 82/182 welds, perform a fatigue crack growth analysis for the 60-year plant life, or justify why flaw evaluations are not need to demonstrate the structural integrity of the affected welds at the end of 60 years.

APS Response to RAI 4.3.2.7-3

Response (a)

Fatigue usage factors in the RCS hot and cold leg piping with reduced wall thickness were not projected to 60 years because the disposition of the TLAA depends on the fatigue management program to ensure that the 40 year design numbers of transients will not be exceeded during 60 years of operation without appropriate corrective actions. These TLAAs will, therefore, be managed for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

Response (b)

There are no Alloy 82/182 welds in the main loop RCS hot and cold leg piping. See response to RAI 4.3.2.7-4 for branch, instrument, and RTD nozzle connections.

Response (c)

There are no Alloy 82/182 welds in RCS hot leg and cold leg main loop piping.

Response (d)

See response to RAI 4.3.2.7-3 (c).

NRC RAI 4.3.2.7-4

In the LRA, page 4.3-57, last paragraph, states that the original RCS hot legs contained a total of 27 Alloy 600 small-bore nozzles in each unit. The applicant replaced seven pressure differential transmitter (PDT) nozzles and one sample nozzle in Unit 2 with full nozzles during 1992. On page 4.3-58 of the LRA, the applicant states that the remaining hot leg smallbore nozzles were replaced with the Alloy 690 half-nozzle design. However, it is not clear whether these remaining nozzles are located in Unit 1, 2, or 3 and it is not clear why small-bore nozzles in Units 1 and 3 are not discussed in this section. (a) Provide a table, similar to Table 4.3-7, with the following information: list all

27 small-bore nozzles for each unit, identify the type of the nozzle (e.g., Resistance Temperature Detector (RTD), PDT) or systems, identify the repair method for each nozzle, identify whether a fatigue analysis was performed for 60 years for each nozzle, and specify whether a TLAA is needed. (b) If a nozzle is not analyzed for 60 years, perform a fatigue analysis for 60 years, or justify why a fatigue analysis is not needed to demonstrate that that small-bore nozzle satisfy the ASME Code Section III allowable usage factor of 1.0 at the end of 60 years. (c) Discuss whether cold leg piping contains small-bore Alloy 600 nozzles, whether they were replaced with Alloy 690 nozzles, and whether their fatigue usage factors were analyzed for 60 years.

APS Response to RAI 4.3.2.7-4

The statement on page 4.3-58 of the LRA that "the remaining hot leg small-bore nozzles were replaced with the Alloy 690 half-nozzle design" refers to all three units.

Response (a)

Table RAI 4.3.2.7-4(a), History of Palo Verde Alloy 600 Small-Bore Hot Leg Nozzle Repairs

Unit	Number & Type of Nozzle	Repair Type	Maximum CUF*
1	9 Pressure and Sampling	Half-Nozzle Repair	Fatigue Waiver
	8 Spare RTD	Welded Plugs	0.051
	10 In-service RTD	Three-quarter Nozzle Repair	0.0105
2	8 Pressure and Sampling	Full Nozzle Repair	0.863
	1 Pressure and Sampling	Half-Nozzle Repair	Fatigue Waiver
	8 Spare RTD	Welded Plugs	0.051
	10 In-service RTD	Three-quarter Nozzle Repair	0.0105
3	9 Pressure and Sampling	Half-Nozzle Repair	Fatigue Waiver
	8 Spare RTD	Welded Plugs	0.051
	10 In-service RTD	Three-quarter Nozzle Repair	0.0105

* Fatigue analysis is 40-year analysis, and is a TLAA.

Response (b)

These TLAAs will be managed for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii). The disposition of these TLAAs depends on the fatigue management program to ensure that the 40 year design number of transients will not be exceeded during 60 years of operation without appropriate corrective actions.

Response (c)

The RCS cold leg piping still contains 12 small-bore Alloy 600 nozzles (the RTD nozzles). The maximum CUF associated with the cold leg RTD Nozzles is 0.0591. The fatigue analysis was performed using the 40 year design numbers of transients. These TLAAs will, therefore, be managed for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii). The fatigue management program will ensure that the 40 year design numbers of transients will not be exceeded during 60 years of operation without appropriate corrective actions.

NRC RAI 4.3.2.7-5

In the 4th paragraph on page 4.3-58 of the LRA, the applicant discusses that the PDT and sampling half-nozzle repairs do not need fatigue analysis of NB-3222.4(d) of the ASME Section III. However, in the 6th paragraph, the welded plugs for the RTD nozzles repairs were analyzed for fatigue per NB-3222.4(e). Explain why a fatigue analysis does not need to be performed for the half nozzle repair, but one is required for the weld plug repair.

APS Response to RAI 4.3.2.7-5

The Class 1 main loop piping fatigue analysis was performed to ASME Section III, Subsection NB, 1974 edition with addenda through Summer 1974.

The PDT and sampling half-nozzle repairs satisfied the fatigue waiver evaluation of ASME Section III, Subsubparagraph NB-3222.4(d). Per Subsubparagraph NB-3222.4(a), "If the specified operation of the component meets all of the conditions of NB-3222.4(d), no analysis for cyclic operation is required...."

The NB-3222.4(d) fatigue waiver option for the welded plugs was not pursued.

NRC RAI 4.3.2.7-6

In the LRA, page 4.3-59, "Redesigned Reactor Coolant System Thermowells," states that the thermowells modification did not affect the previous conclusion concerning fatigue of the thermowells and that there is no safety determination based on the plant life for these high-cycle loads and therefore no TLAA. (a) Explain why this issue is not a

TLAA because the thermowells experience high-cycle fatigue which is time-dependent. (b) Perform a fatigue analysis of the thermowells for 60 years, or justify why a fatigue usage factor analysis for 60 years is not needed to demonstrate that the new thermowell design satisfy the allowable cumulative usage factor of 1.0.

APS Response to RAI 4.3.2.7-6

Response (a)

The failure mechanism of the thermowells was high-cycle fatigue caused by resonance between the thermowell natural frequency and the vortex shedding frequencies. The analysis and testing of the redesigned thermowell determined that the new design was not susceptible to this failure mechanism.

Response (b)

This determination did not consider the plant life; therefore, the evaluation for high-cycle fatigue is not a TLAA, in accordance with 10 CFR 54.3(a) Criterion 3.

NRC RAI 4.3.2.7-7

In the LRA, pages 4.3-59 and 4.3-60, "Removal of Reactor System Safety Injection Nozzle Thermal Sleeves," conclude that the modification did not affect the previous conclusion concerning fatigue of the safety injection nozzles. However, the applicant did not discuss whether the fatigue analysis of the safety injection nozzles was based on 40-year or 60- year transient cycles. (a) Perform a fatigue analysis for 60 years, or justify why a fatigue analysis of the safety injection nozzles for the end of 60 years is not needed to demonstrate that the cumulative usage factors of the subject nozzles at the end of plant life satisfies the ASME Code Section III allowable of 1.0. (b) Discuss whether this issue is a TLAA.

APS Response to RAI 4.3.2.7-7

The fatigue analysis of the safety injection nozzles was performed using the 40 year design numbers of transient cycles, therefore it is a TLAA.

Response (a)

The safety injection nozzles TLAA will be managed for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii). The disposition of the fatigue analysis of the safety injection nozzles depends on the fatigue management program to ensure that the 40 year design numbers of transients will not be exceeded during 60 years of operation without appropriate corrective actions.

Response (b)

Fatigue analysis of the safety injection nozzles is a TLAA.

NRC RAI 4.3.2.7-8

In the LRA, page 4.3-60, "Flow Stratification Thermal Gradient in the Auxiliary Spray Line and Tee," states that the cumulative fatigue usage factor, including the effects of thermal gradient in the auxiliary spray line meets ASME Section III for a 40-year plant life. (a) Perform a fatigue analysis for 60 years, or justify why a fatigue analysis is not needed for 60 years to demonstrate that the cumulative usage factor of the auxiliary spray line satisfies a 60-year plant life. (b) Discuss whether this issue is a TLAA.

APS Response to RAI 4.3.2.7-8

Response (a)

The auxiliary spray line and tee fatigue analysis is a TLAA and will be managed for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii). The disposition of the fatigue analysis of the auxiliary spray line and tee depends on the fatigue management program to ensure that the 40 year design numbers of transients will not be exceeded during 60 years of operation without appropriate corrective actions.

Response (b)

Fatigue analysis of the auxiliary spray line and tee is a TLAA.

NRC RAI 4.3.2.7-9

LRA page 4.3-61, first paragraph. Explain why fracture mechanics analyses of the hot leg surge and shutdown cooling nozzles overlaid by the weld repair are not a TLAA.

APS Response to RAI 4.3.2.7-9

The fatigue crack growth analyses to support the hot leg surge and shutdown cooling nozzle weld overlay repairs calculate the crack propagation in order to demonstrate that a postulated crack will not exceed the acceptance criterion of the analysis during the inspection interval. The inspection interval is less than the plant life; therefore, the fatigue crack growth analyses are not TLAAs, in accordance with 10 CFR 54.3(a) Criterion 3.

NRC RAI 4.3.2.7-10

In the LRA, page 4.3-61, Disposition Section, states, "The Metal Fatigue of Reactor Coolant Pressure Boundary program will continue to confirm that this is so, or that appropriate reevaluation or other corrective action is initiated if an action limit is reached." (a) Specify the exact piping components and systems that will be monitored under the Metal Fatigue program in the context of Section 4.3.2.7. (b) Discuss how often the actions (e.g., monitoring the transient cycles and review the records) under the Metal Fatigue program will be performed. Section 4.3.1 discusses that a FatiguePro computer software program is used at Palo Verde to monitor the transient cycles; however, it is not clear how often the monitoring is performed by the applicant and when corrective actions are taken.

APS Response to RAI 4.3.2.7-10

Response (a)

The piping components that will be monitored are listed in LRA Table 4.3-4. LRA Section 4.3.1.5, page 4.3-22, states the following:

> "The scope of the bounding set of monitored locations is sufficient to ensure that fatigue in any other locations of concern, not included in the set, is within the same system, or within a system affected by the same transients.

Response (b)

As stated in LRA Section 4.3.1.5, page 4.3-23, "The PVNGS fatigue management program currently ... requires this evaluation at least once per fuel cycle." This schedule is expected to apply to the enhanced fatigue management program for the period of extended operation.

Section 4.3.2.9 *Bulletin 88-11 Revised Fatigue Analysis of the Pressurizer Surge Line for Thermal Cycling and Stratification*

NRC RAI 4.3.2.9-1

In the LRA, page 4.3-64, last paragraph, states that the cumulative usage factor for the surge line is 1.65 using the elastic analysis. Combustion Engineering performed a plastic analysis which reduced the limiting cumulative usage factor to 0.937. (a) Submit both the elastic and plastic analyses. Alternatively, describe the analyses in detail, including methodology, input, acceptance criteria, and results. (b) Clarify whether the fatigue usage factor analysis is based on a 40-year period or 60-year period. (c) On Page 4.3-65 of the LRA, the applicant states that the Metal Fatigue of Reactor Coolant Pressure Boundary Aging Management Program will be used to monitor the surge line. Discuss whether the Metal Fatigue aging management program will initiate actions

based on the elastic analysis result (CUF of 1.65) or plastic analysis result (CUF of 0.937).

APS Response to RAI 4.3.2.9-1

Response (a)

Palo Verde calculation 13-MC-ZZ-595 performed fatigue evaluations for the pressurizer surge line, including the IEB 88-11 additional thermal cycling and stratification effects. This analysis incorporated results of CE Owners Group (CEOG) calculation MISC-ME-C-115 and report CEN-387-P.

These analyses found that the highest CUF in the surge line is in the elbow below the pressurizer. The 1.65 CUF is a result of a preliminary shakedown analysis. As stated in the scope of calculation MISC-ME-C-115, "The fatigue evaluation program developed to analyze the shakedown analysis results is used only to rank each point in the elbow. The output from this program in no way represents the fatigue usage for the actual transients listed...." The generalized NB-3228.4 shakedown analysis resulted in a maximum CUF (in the surge line elbow below the pressurizer) of 1.65, as shown in calculation MISC-ME-C-115. The same analysis reanalyzed this highest-ranking 1.65 CUF location, and calculated an actual CUF of 0.778. As stated in the conclusion of calculation MISC-ME-C-115, "Using the transients analyzed in the shakedown analysis as a method to rank locations in the limiting surge line elbow, the location of greatest usage is analyzed for its actual usage.... The point of highest actual usage is, U=0.778."

In response to NRC Bulletin 88-11, CEOG performed the evaluation reported in CEN-387-P on pressurizer surge line thermal stratification. CEN-387-P reported a plant-specific analysis CUF of 0.937 at this location, also using the methods of NB-3228.4. The acceptance criterion is a calculated 40-year CUF ≤ 1.0. The result is the CUF of 0.937 reported by CEN-387-P. This plant-specific analysis is the analysis of record for this component at Palo Verde.

Response (b)

The CEN-387-P analysis of record calculated a 40-year CUF of 0.937 at the limiting location in the surge line.

Response (c)

CEN-387-P is the analysis of record for effects of insurge-outsurge and thermal stratification at this limiting location. If cycle counting were used as the fatigue management method, the Metal Fatigue aging management program would, therefore, initiate actions based on the analysis of record result (CUF of 0.937). However, this is a NUREG/CR-6260 sample location, and when a conservative estimate of the multiplier for effects of the reactor coolant environment is used, the calculated CUF becomes

several times the code acceptance criterion of 1.0. Fatigue in this location will, therefore, be managed by stress-based fatigue monitoring, and the action limits for this location will, therefore, depend on calculated actual fatigue usage, not on a 40-year (or 60-year) value determined by the analysis of record. See LRA Section 4.3.4 for the NUREG/CR-6260 sample locations, and LRA Table 4.3-4 Item 24 for monitoring of this location.

The first sentence of the last paragraph on LRA page 4.3-64 misidentified the 1.65 CUF as the result of "The elastic analysis...." This has been revised in Amendment 10 in Enclosure 2 to read, "~~The elastic~~ A preliminary shakedown analysis produced...."

The related paragraphs in LRA Section 4.3.4, page 4.3-79, also misidentify the 1.65 CUF as the result of an elastic analysis, and include an unnecessary sentence. In LRA Amendment 10 in Enclosure 2, this sentence has been deleted and clarification has been inserted into the intended comparison, to read as follows:

> Pressurizer Surge Line (Hot Leg) Elbow (Location 4): Combustion Engineering (CE) performed a fatigue evaluation of surge lines in various CE Owners Group (CEOG) plants, with thermal stratification loading. The analysis assumed the design basis number of 500 heatup transients. ~~The elastic~~ A preliminary shakedown analysis produced a cumulative usage factor of 1.65 in the comparable (and more limiting) surge line pressurizer elbow. To decrease the CUF below the ASME fatigue limit of 1.0, CE then performed a plastic analysis resulting in a limiting CUF of 0.937 in the pressurizer elbow. APS confirmed this bounding analysis as the fatigue analysis of record for this component at PVNGS. See Section 4.3.2.9.
>
> To evaluate effects of the reactor coolant environment, APS re-evaluated the CUF in the pressurizer surge line hot leg elbow using design basis transient cycles and ASME Subsection NB-3200 6-component stress tensors. The simplified elastic-plastic analysis produced a CUF of 1.9396, which is above the ASME code allowable fatigue limit of 1.0. ~~This result is consistent with the CUF of 1.65 in the pressurizer elbow from the CE elastic analysis.~~ The CE plastic analysis described in Section 4.3.2.9 that calculated a CUF of 0.937 is more precise than the APS reevaluation; however, the APS hot leg elbow reevaluation will....

NRC RAI 4.3.2.9-2

In the LRA, page 4.3-65, first paragraph, discusses the surge line elbow and risk-informed inservice inspection program. (a) Confirm that the surge line elbow is a component that requires a nondestructive examination to be performed under the inservice inspection program. (b) Discuss how often the surge line elbow will be inspected in each of the 10-year inservice inspection intervals through the 6th inservice

inspection interval. (c) Discuss the nondestructive examination method that will be used for each inspection.

APS Response to RAI 4.3.2.9-2

Response (a)

The surge line elbow is subject to the ASME Section XI inspection program.

Subsequent to the initial LRA submittal, no relief request has been filed to permit use of a risk-informed inservice inspection program for the current, third inspection interval. LRA, page 4.3-65, first paragraph, has been revised as shown below and in Amendment No. 10 in Enclosure 2:

> PVNGS augmented its ASME Section XI, ISI program to include inspections of the surge line elbow, which were performed to address NRC Bulletin 88-11 concerns. PVNGS subsequently proposed the alternative RI-ISI in Relief Request 32 for the third period of the second ISI interval [Ref. 23]. The RI-ISI application ~~is~~ was based on the EPRI RI-ISI program, which explicitly considered NRC Bulletin 88-11 concerns in its application. The NRC Bulletin 88-11 concerns ~~are~~ were therefore addressed by the PVNGS RI-ISI program. However the program was approved only for the third period of the second ISI interval, and no relief request has been filed for the current, third interval.

Response (b)

ASME Section XI requires that the surge line elbow in each of the Palo Verde units be VT-2 examined each refueling outage. The current ASME Section XI program does not extend through the sixth inservice inspection interval. The inservice inspection program is revised to the requirements of 10 CFR 50.55a each inspection interval.

Response (c)

The current ASME Section XI nondestructive examination requirement for the surge line elbow is a VT-2 examination for leakage.

NRC RAI 4.3.2.9-3

In the LRA, page 4.3-65, Disposition section, states that the surge line is subject to stress-based fatigue monitoring under the Metal Fatigue of Reactor Coolant Pressure Boundary program. Discuss how often the record of the worst case cumulative usage factors will be reviewed and evaluated by the applicant.

APS Response to RAI 4.3.2.9-3

As stated in LRA Section 4.3.1.5, page 4.3-23, "The PVNGS fatigue management program currently ... requires this evaluation at least once per fuel cycle." This schedule is expected to apply to the enhanced fatigue management program for the period of extended operation.

Section 4.3.2.15 *Absence of TLAAs in Fatigue Crack Growth Assessments and Fracture Mechanics Stability Analyses for the Leak-Before-Break (LBB) Elimination of Dynamic Effects of Primary Loop Piping Failures*

NRC RAI 4.3.2.15-1

In Section 3.6.3 of the Standard Review Plan (NUREG-0800), paragraph III.10 states, "The reviewer should determine that the candidate piping does not have a history of fatigue cracking or failure. An evaluation to ensure that the potential for pipe rupture due to thermal and mechanical induced fatigue is unlikely should be performed." The technical basis for the leak-before-break (LBB) approval for Palo Verde is provided in a Combustion Engineering (CE) topical report entitled "Leak Before Break Evaluation of the Main Coolant Loop Piping of a CE Reactor Coolant System," provided as an attachment to a letter dated June 14, 1983 (also, CE report "Leak Before Break Evaluation of the Main Coolant Loop Piping of a CE Reactor Coolant System," Revision 1, provided as an attachment to a letter dated December 23, 1983). Section 3 of this report describes fatigue calculations to demonstrate the acceptability of fatigue crack growth for various postulated flaws, which demonstrates that fatigue is not an active degradation mechanism of concern. One of these calculations, for a relatively small flaw of 1 inch in depth and 8 inches in length, demonstrates that the crack will not penetrate through wall for a very large number of cycles, principally heatup and cool-down cycles. Although the technical basis report for approval of LBB for Palo Verde incorporates fatigue crack growth calculations that have a time-basis (40 years) and/or consider numbers of cycles in the calculations, the first paragraph on page 4.3-72 of the LRA states that the LBB fatigue crack analyses are not TLAAs because the postulated fatigue cracks grow slowly and the fatigue evaluation does not depend on the design life. Provide an assessment of the acceptability of the fatigue-based aspects of the Palo Verde LBB approval, consistent with the requirements of 10 CFR 54.21(c).

APS Response to RAI 4.3.2.15-1

The acceptability of the Palo Verde LBB evaluation depends on a fatigue crack growth analysis and on a fracture mechanics stability analysis. The fracture mechanics stability analysis is not time-dependent and, therefore, remains applicable for the period of extended operation. The fatigue crack growth analysis is time-dependent, but is valid for a design life well beyond the period of extended operation.

The existing Palo Verde fatigue crack growth evaluation demonstrates that initially-postulated cracks larger than those required by the LBB rule will remain within allowable sizes for an order of magnitude longer than the 40-year current licensed operating period. Since the safety determination supported by this evaluation does not depend on the design life, and, therefore, does not meet Criteria 3 and 5 of the 10 CFR 54.3(a) TLAA definition, PVNGS did not classify this fatigue crack growth evaluation as a TLAA.

Since the safety determination supported by the existing Palo Verde fatigue crack growth evaluation is valid for several times a 60-year design life, it is valid for at least the period of extended operation. Even if the existing Palo Verde fatigue crack growth evaluation were classified as a TLAA it would, therefore, remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

The fatigue crack growth evaluation of the Palo Verde LBB evaluation examined the rate of growth for four sample crack sizes and geometries.

> The CE main loop pipes are fabricated and inspected in accordance with NB 2532 and NB 5000 of the ASME Section III Code which ... allows indications no longer than three inches and nor deeper than 10% of the pipe wall thickness.... In order to conservatively evaluate crack growth and extension, a variety of crack sizes much larger than those expected to exist are considered in the subsequent analyses.
>
> [Ref. 2 §3.a p. 5]

For the three larger crack sizes postulated (0.5 x 39 inches, 1.0 x 34 inches, and 0.35 x 45.5 inches), the evaluation calculated that through-wall leaks would occur at 21, 4, and 38 years, respectively; determined that preferential growth would be in the radial direction; and determined that the rate of growth between a through-wall detectable leak and a critical crack size would be acceptable for defects much larger than any anticipated actual initial defect (Ref. 2 §3.b pages 6-7, and Figure 2-4).

The startup-shutdown transient was found to be the greatest contributor to usage factor on the main loop piping, and a cyclic stress bounding this startup-shutdown transient was applied to the fatigue crack growth analysis of a smaller one-inch deep by 8 to 18-inch long crack, still enveloping any defects meeting ASME III initial inspection criteria. This analysis similarly demonstrated preferential growth in the radial direction, and demonstrated that defects larger than the ASME III inspection acceptance criteria would not grow through the pipe wall in several design lifetimes, at least ten lifetimes, more than 400 years, for the 1 x 8-inch crack size (Ref. 2 §3.b page 7, and Figure 5).

The basis for the safety determination is therefore not that the crack will remain within an acceptable size within a 40-year design lifetime, but (1) that the rate of growth for any anticipated crack size is acceptable, even following wall penetration and detectable leakage, and (2) that initial defects somewhat larger than the ASME III initial inspection criteria will not even grow through the pipe wall in several 40-year design lifetimes.

The frequency of load application was determined by assuming a uniform distribution of a typical 40-year set of CE design basis loading events over a 40-year life (Ref. 2 §3.b page 6, and Table 1). This is a conservative rate assumption. LRA Table 4.3-3 demonstrates that the rates of accumulation of transient cycles have remained less than those assumed as the basis for the LBB evaluation, with a few exceptions that have no significant effect on the bases for the LBB fatigue crack growth evaluation. (Compare the LRA Table 4.3-3 "Limiting Number of Events" to the "Projected to 40 Years" values.)

Therefore, neither the basis of the fatigue crack growth analysis, nor the basis for the safety determination, nor the conclusion of the safety determination (Ref. 2 §7 pages 27-28), will change with an increase in the licensed operating period.

The Palo Verde fatigue crack growth analyses, therefore, do not "...involve time-limited assumptions *defined by the current operating term,* for example, 40 years" [10 CFR 54.3(a) Criterion 3, emphasis added]. They are *time-dependent,* but for an indefinite period, and are, therefore, not *time-limited.*

The Metal Fatigue aging management program is not implemented to monitor the transient cycles to confirm that the transient cycles used in the fatigue crack growth analyses for the LBB piping exceed the actual transient cycles; because the existing LBB fatigue crack growth evaluation is valid for the period of extended operation, as described above. Amendment 10, therefore, revises the LRA text, in three locations, as marked below:

> To ensure that the analytical bases of ~~the leak-before-break (LBB) fatigue crack propagation analysis and of~~ the high-energy line break (HELB) locations are maintained.

This statement has been revised in the following LRA sections, as shown in LRA Amendment 10 in Enclosure 2:

> LRA Section 4.3.1.5, page 4.3-24, under "Cycle Count Corrective Actions," Item 1), third bullet.
>
> LRA Appendix A Section A2.1, page A-22, under *"Cycle Count Action Limit and Corrective Actions,"* Item 1c.
>
> LRA Appendix B Section B3.1, page B-116, under "Cycle Count Action Limit and Corrective Actions," Item 1c.

<u>References</u>

The Combustion Engineering LBB analyses applicable to Palo Verde, and the associated NRC SE, are:

1. Combustion Engineering Letter LD-83-053. A. E. Scherer; to Darrell G. Eisenhut, NRC. Docket No. STN 50-470F. "Basis for Design of Plant Without Pipe Whip Restraints." June 14, 1983. (ADAMS Accession No. 8306200254)

   With enclosed *Basis for Design of Plant Without Pipe Whip Restraints for RCS Main Loop Piping.*

2. Combustion Engineering Letter LD-83-108. A. E. Scherer; to Darrell G. Eisenhut, NRC. Docket No. STN 50-470F. "Basis for Design of Plant Without Pipe Whip Restraints." December 23, 1983. (ADAMS Accession No. 8312300117)

   With enclosed *Leak Before Break Evaluation of the Main Loop Piping of a CE Reactor Coolant System.* Revision 1, November 1983.

3. NRC letter to A. E. Scherer, Combustion Engineering, "CESSAR System 80 Safety Evaluation Report (NUREG-0852) Removal of Pipe Whip Restraints." October 11, 1984. (ADAMS Accession No. 8410240182)

NRC RAI 4.3.2.15-2

Page 4.3-71 of the LRA. (a) List the piping systems in each of three units that have been approved for LBB application. (b) List references of the LBB analyses.

APS Response to RAI 4.3.2.15-2

Response (a)

As stated in LRA Section 4.3.2.15, the LBB exemption from dynamic effects of pipe ruptures (jets and pipe whip) applies to "...the main reactor coolant loops," meaning the hot, cold, and crossover legs.

Response (b)

The Combustion Engineering LBB analyses applicable to Palo Verde, and the associated NRC Safety Evaluations, are listed under the response to RAI 4.3.2.15-1.

NRC RAI 4.3.2.15-3

Nickel-based Alloy 600/82/182 material in the pressurized water reactor environment has been shown to be susceptible to primary water stress corrosion cracking (PWSCC). (a) Identify any Alloy 82/182 weld metal and Alloy 600 components used in the LBB approved piping for both units. (b) If LBB piping contains Alloy 600/82/182 material, discuss any measures (such as weld overlays or mechanical stress improvement) that have been or will be implemented to reduce the susceptibility of PWSCC in the LBB

piping components. (c) Discuss the inspection history and future inspection frequency of the Alloy 82/182 dissimilar metal butt welds.

APS Response to RAI 4.3.2.15-3

No Alloy 82/182 weld metal or Alloy 600 components remain in the main reactor coolant loops within the scope of the LBB analysis, except the branch, instrument, and RTD nozzle connections shown in Table RAI 4.3.2.15-3 below.

Table RAI 4.3.2.15-3
Alloy 82/182 Weld Metal and Alloy 600 Components in the Main Reactor Coolant Loops (Branch, Instrument, and RTD Nozzle Connections) within the Scope of the LBB Analysis

RC Loop Nozzle to	Description	Failure History	Inspection	Mitigating Strategy
SDC line 1 & 2	Alloy 82/182 weld	None	100% volumetric once in the next 5 years, if no additional indications/growth, continue with the existing Code examination program for unflawed condition or approved alternative. Bare metal visual examination once every three (3) refuel outages (RFO) when volumetric exams are not performed.	Full Structural Overlay: Unit 1 fall 2008 Unit 2 spring 2008 Unit 3 spring 2009
Pressurizer Surge line	Alloy 82/182 weld	None	100% volumetric once in the next 5 years, if no additional indications/growth, continue with the existing Code examination program for unflawed condition or approved alternative. Bare metal visual examination once every three (3) refuel outages (RFO) when volumetric exams are not performed.	Full Structural Overlay: Unit 1 spring 2007 Unit 2 spring 2008 Unit 3 fall 2007
Pressurizer Spray lines 1A & 1B	Alloy 82/182 weld	None	Bare metal visual examination once every three (3) RFO.	Potential for future structural weld overlay or mechanical stress improvement
Safety Injection lines 1A, 1B, 2A, & 1B	Alloy 82/182 weld	None	100% volumetric every 6 yrs & bare visual examination once every three (3) RFO when volumetric exams are not performed.	None
Drain Line 1A, 1B, & 2A	Alloy 82/182 weld	None	Bare metal visual examination once every three (3) RFO.	None
Letdown Line 2B	Alloy 82/182 weld	None	Bare metal visual examination once every three (3) RFO.	None
Charging Line 2A	Alloy 82/182 weld	None	Bare metal visual examination once every three (3) RFO.	None
12 Cold Leg RTD Nozzles	Alloy 600	None	Bare metal visuals	None
8 RCP Instrument Taps	Alloy 600	None	Bare metal visuals	None

NRC RAI 4.3.2.15-4

(a) Discuss the inspection history and results of the LBB-approved piping. (b) If indications or flaws remain in inservice LBB piping, discuss how the indications and flaws will be monitored to the end of the period of extended operation. (c) Discuss future inspection schedules for each of the LBB pipes (other than existing indications and flaws).

APS Response to RAI 4.3.2.15-4

Response (a)

Welds and piping that are part of the LBB piping have been examined under the ASME Section XI Rules for inservice inspection starting in the first interval and will be examined under the rules of ASME Section XI in future intervals.

There is no direct tie between assumptions of the LBB evaluation and the ASME Section XI program. The piping in question is part of the overall population of welds subject to examination.

Response (b)

No rejectable indications have been found.

Response (c)

Piping will continue to be examined in accordance with ASME Section XI. Surface and volumetric exams are completed on 25% of the welds spread out over each ten year interval, and a visual exam is completed every outage.

Please see also the response to RAI 4.3.2.15-3.

NRC RAI 4.3.2.15-5

Discuss whether the LBB analyses have been re-evaluated to determine the impact of operating conditions due to system modifications such as power uprates or steam generator modifications on the LBB analyses for the period of extended operation.

APS Response to RAI 4.3.2.15-5

As stated on LRA page 4.3-73, "Effects of Power Uprate and Steam Generator Replacement have been evaluated and resulted in no change to the conclusion of the LBB analysis." Since the LBB analyses have been reviewed for effects of power uprate and steam generator replacement with no effect on the conclusion, and since they are

not TLAAs (see the response to RAI 4.3.2.15-1), they are, therefore, not affected by the increase in operating life for the period of extended operation.

The NRC license amendment review of LBB for power uprate and steam generator replacement describes these evaluations and also provides a description of the original LBB basis. See Section 3.4.1 of the safety evaluation attached to the following reference:

> US NRC Letter. Mel B. Fields, Senior Project Manager, Plant Licensing Branch IV, Division of Operating Reactor Licensing, Office of Nuclear Reactor Regulation; to James M. Levine, Executive Vice President, Generation, APS. "Palo Verde Nuclear Generating Station, Units 1, 2, and 3, Issuance of Amendments RE: Replacement of Steam Generators and Uprated Power Operations, and Associated Administrative Changes (TAC Nos. MC3777, MC3778, and MC3779)." November 16, 2005. (ADAMS Accession No. ML053130275)
>
> With enclosed *Amendment No. 157 to* [Unit 1] *Facility Operating License No. NPF-41, Amendment No. 157 to* [Unit 2] *Facility Operating License No. NPF-51,* and *Amendment No. 157 to* [Unit 3] *Facility Operating License No. NPF-74,* and supporting Technical Specification changes.
>
> With enclosed *Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 157 to Facility Operating License No. NPF-41, Amendment No. 157 to Facility Operating License No. NPF-51, and Amendment No. 157 to Facility Operating License No. NPF-74, 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.*

See also the response to RAI 4.3.2.15-1.

Section 4.7.4 ***Fatigue Crack Growth and Fracture Mechanics Stability Analyses of Half-Nozzle Repairs to Alloy 600 Material in Reactor Coolant Hot Legs: Absence of a TLAA for Supporting Corrosion Analyses***

NRC RAI 4.7.4-1

In the LRA, page 4.7-8, second paragraph, states that the applicant made a commitment to monitor the 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 for the second, third, and fourth 10-year inspection intervals. Discuss whether the same commitment will be implemented in the fifth and sixth inspection intervals. If not, provide justification.

APS Response to RAI 4.7.4-1

LRA Appendix A, Table A4-1, Commitment 46, documents the APS commitment to continue to monitor the cold shutdown conditions, via the current tracking method for the period of extended operation, that is, for the fifth and sixth inspection intervals.

NRC RAI 4.7.4-2

In the LRA, page 4.7-8, "Extension to All Hot Leg Small-bore Nozzles," states that 63 previously repaired small-bore hot leg nozzles in all three units were added to Relief Request 31. The applicant states further that there are a total of 27 small diameter hot leg penetrations per unit. If there are 27 small-bore nozzles in each unit, the total number of smallbore nozzles in all three units should be 81. It is not clear whether the exact number of smallbore nozzles on the hot leg piping is 63 or 81. (a) Provide the exact number of small-bore nozzles in the hot leg piping in each unit, the number of small-bore nozzles that have been repaired in each unit, and the number of small-bore nozzles that have not been repaired. (b) Discuss whether any small-bore nozzles in hot leg piping that are not covered under Relief Request 31. Confirm that the small bore nozzles in hot leg piping that are not covered under Relief Request 31, were analyzed for TLAA.

APS Response to RAI 4.7.4-2

Relief Request 31, Revision 0, addressed 10 nozzles replaced in Unit 2 during the spring of 2005.

Relief Request 31, Revision 1, added the following previously-repaired PVNGS Alloy 600 small-bore reactor coolant system (RCS) hot leg nozzles as follows:

- Unit 1 - 27 nozzles,
- Unit 2 - 9 nozzles, and
- Unit 3 - 27 nozzles (63 total).

The last 63 nozzles repaired under Relief Request 31, Revision 1 were replaced under the Palo Verde Alloy 600 replacement program, from approximately October 1999 to April 2003.

Response (a)

There are 27 small-diameter hot leg penetrations per unit. All 81 have been repaired:

8	Unit 2 nozzles repaired in 1991 via a full nozzle repair (see below)
10	Unit 2 nozzles repaired in 2005 under Relief Request 31, Revision 0
9	Unit 2 nozzles repaired under Relief Request 31, Revision 1
27	Unit 1 nozzles repaired under Relief Request 31, Revision 1
27	Unit 3 nozzles repaired under Relief Request 31, Revision 1
81	Total

A detailed list is provided in the response to RAI 4.3.2.7-4.

Response (b)

The eight Unit 2 nozzles repaired in 1991 via a full nozzle repair are not covered by this relief request, and the fatigue crack growth analysis and corrosion analysis are, therefore, not applicable. The TLAAs associated with the design of the eight Unit 2 full nozzle repairs are addressed in LRA Section 4.3.2.7.

NRC Appendix A

Appendix A---Updated Final Safety Analysis
The staff reviewed Appendix A3 of the License Renewal Application and did not find any discussion regarding leak-before-break (LBB) evaluations of the RCS piping as is discussed in Section 4.3.2.15. Justify why the TLAA of the LBB evaluation of the RCS piping is not included in Section A3 of Appendix A of the License Renewal Application.

APS Response to Appendix A

The LBB evaluation of the RCS piping is not included in Section A3 of Appendix A of the License Renewal Application because the LBB evaluation does not include, and is not supported by, a TLAA, and no change to the licensing basis is required for the period of extended operation. Please see the response to RAI 4.3.2.15-1.

ENCLOSURE 2

Palo Verde Nuclear Generating Station
License Renewal Application Amendment No. 10

Palo Verde Nuclear Generating Station
License Renewal Application
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Source: RAI 4.3.2.15-1 Response

LRA Section 4.3.1.5 (page 4.3-24), under "Cycle Count Corrective Actions," Item 1), the third bullet is revised to read (deleted text is struck out):

- To ensure that the analytical bases of ~~the leak-before-break (LBB) fatigue crack propagation analysis and of~~ the high-energy line break (HELB) locations are maintained.

Source: RAI 4.3.2.4-5 Response

LRA Section 4.3.2.4 (page 4.3-40): The subheading at the top of LRA page 4.3-40, and the two paragraphs below it, are revised to read (deleted text is struck out, new text is underlined):

Absence of TLAAs in the Analysis of Thermal Fatigue Crack Growth in Original Heater Sleeve Attachment Welds, in Support of ~~MNSA~~ ASME Section XI Inspection Relief, and of Heater Sleeve Repairs ~~of Unit 3 Heater Sleeves~~

There are currently no mechanical nozzle seal assemblies (MNSAs) in use at PVNGS. Three MNSAs were used as a temporary means of sealing U~~u~~nit 3 pressurizer heater sleeves. ~~However the MNSAs were replaced with half-nozzle repairs during unit 3 refueling outage 3R11.~~

A supporting Westinghouse ~~linear elastic fracture mechanics~~ fatigue crack growth analysis for the Unit 3 MNSA repairs, for postulated cracks in the original sleeve-to-inner-wall attachment welds, was based on a 60-year design life, and was therefore not a TLAA. ~~Although t~~These MNSAs were ~~replaced,~~ removed and all heater sleeves in all three units were replaced, using half-nozzle repairs. The Westinghouse ~~this~~ analysis in support of this temporary MNSA modification ~~is still applicable, and is cited~~ was superseded by:

- A~~a~~ subsequent report and fatigue crack growth analysis in support of relief from ASME Section XI inspection requirements, ~~incorporated by reference in the code design reports,~~ because the area of the postulated initial cracks - at the original attachment J-welds - has not been removed.

- A subsequent report and fatigue crack growth analysis in support of the half-nozzle repair of all 36 heater sleeves in each of Units 1, 2, and 3. This report and analysis are included by reference in the Unit 1, 2, and 3 pressurizer code design reports.

The analyses are consistent. Both report a projected final flaw size at 60 years of 1.16 inch, based on an initial flaw size of 0.6 inch, and cite supporting fracture mechanics analyses permitting an allowable flaw size of 1.2 inch. Both of these

currently-applicable fatigue crack growth analyses apply to all three units. Both were performed for a 60-year operating life, and are therefore not TLAAs.

Source: RAI 4.3.2.4-13 Response

Section 4.3.2.4 Table 4.3-7 item 1 (page 4.3-44) is revised to change the reference "item 17 below" to read "item 18 below" (deleted text is struck out, new text is underlined):

Table 4.3-7 - Summary of PVNGS Pressurizer ASME III Class 1 Analyses and Fatigue Usage Factors

Component Analysis (Date Replacement or Reanalysis Initiated)	**TLAA? Basis if not.**	**Maximum CUF** (U_{40}, Unless Noted Otherwise)[1]		
		Unit 1	**Unit 2**	**Unit 3**
1. **Bottom Head and Support Skirt - Worst Location, Inside the Support Knuckle** (1993 reanalysis in response to CE Infobulletin 88-09) See also item ~~17~~ 18 below for a reanalysis of the Unit 3 bottom head.	**Yes**	0.7223	0.7223	0.7223

Source: RAI 4.3.2.9-1 Response

LRA Section 4.3.2.9 (page 4.3-64), last paragraph, first sentence is revised to read (deleted text is struck out, new text is underlined):

"~~The elastic~~ A preliminary shakedown analysis produced...."

Source: RAI 4.3.2.9-2 Response

Section 4.3.2.9 (page 4.3-65) is revised to update descriptions of the risk-informed inservice inspection (RI-ISI) program (deleted text is struck out, new text is underlined):

Effect of NRC Bulletin 88-11 on Risk Informed Inservice Inspection (RI-ISI) Program, Relief Request 32.

PVNGS augmented its ASME Section XI, ISI program to include inspections of the surge line elbow, which were performed to address NRC Bulletin 88-11 concerns.

PVNGS subsequently proposed the alternative RI-ISI in Relief Request 32 for the third period of the second ISI interval [Ref. 23]. The RI-ISI application ~~is~~ was based on the EPRI RI-ISI program, which explicitly considered NRC Bulletin 88-11 concerns in its application. The NRC Bulletin 88-11 concerns ~~are~~ were therefore addressed by the PVNGS RI-ISI program. However the program was approved only for the third period of the second ISI interval, and no relief request has been filed for the current, third interval.

Source: RAI 4.3.2.9-1 Response

LRA Section 4.3.4 (page 4.3-79), last two paragraphs, is revised, consistent with the change to Section 4.3.2.9, to read (deleted text is struck out, new text is underlined, except that the underlined run-in subhead is original):

Pressurizer Surge Line (Hot Leg) Elbow (Location 4): Combustion Engineering (CE) performed a fatigue evaluation of surge lines in various CE Owners Group (CEOG) plants, with thermal stratification loading. The analysis assumed the design basis number of 500 heatup transients. ~~The elastic~~ A preliminary shakedown analysis produced a cumulative usage factor of 1.65 in the comparable (and more limiting) surge line pressurizer elbow. To decrease the CUF below the ASME fatigue limit of 1.0, CE then performed a plastic analysis resulting in a limiting CUF of 0.937 in the pressurizer elbow. APS confirmed this bounding analysis as the fatigue analysis of record for this component at PVNGS. See Section 4.3.2.9.

To evaluate effects of the reactor coolant environment, APS re-evaluated the CUF in the pressurizer surge line hot leg elbow using design basis transient cycles and ASME Subsection NB-3200 6-component stress tensors. The simplified elastic-plastic analysis produced a CUF of 1.9396, which is above the ASME code allowable fatigue limit of 1.0. ~~This result is consistent with the CUF of 1.65 in the pressurizer elbow from the CE elastic analysis.~~ The CE plastic analysis described in Section 4.3.2.9 that calculated a CUF of 0.937 is more precise than the APS reevaluation; however, the APS hot leg elbow reevaluation will....

Source: RAI 4.3.2.15-1 Response and AMP Audit Question

LRA Section A2.1 (page A-22), under *"Cycle Count Action Limit and Corrective Actions,"* Item 1c is revised to read (deleted text is struck out):

c. To ensure that the analytical bases of ~~the leak-before-break (LBB) fatigue crack propagation analysis and of~~ the high-energy line break (HELB) locations are maintained.

Source: RAI 4.3.2.15-1 Response

LRA Section B3.1 (page B-116), under "Cycle Count Action Limit and Corrective Actions," Item 1c is revised to read (deleted text is struck out):

c. To ensure that the analytical bases of ~~the leak-before-break (LBB) fatigue crack propagation analysis and of~~ the high-energy line break (HELB) locations are maintained.