10 CFR 50.55(a)

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102-08091-BJR/MDD
March 30, 2020

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

Dear Sirs:

Subject: **Palo Verde Nuclear Generating Station (PVNGS) Unit 2**
Docket No. STN 50-529
Renewed Operating License Number NPF-51
APS Response to Request for Additional Information – Relief Request 65 Unit 2, COVID-19, Request for Relief from Bottom Mounted Instrumentation Nozzles and a Pressurizer Nozzle to Surge Line Weld Overlay Examination

By letter number 102-08090, dated March 27, 2020 (Agencywide Documents Access and Management System Accession Number ML20088A533), Arizona Public Service Company (APS) submitted relief request 65 in accordance with 10 CFR 50.55a(z)(2).

The Nuclear Regulatory Commission (NRC) staff requested additional information to complete their review with regard to extending scheduled Palo Verde Nuclear Generating Station, Unit 2 inservice inspection (ISI) examinations for the reactor vessel bottom mounted instrumentation (BMI) nozzles and a pressurizer nozzle surge line weld overlay from the currently planned Unit 2 Spring of 2020 refueling outage (2R22) to the next refueling outage (2R23) in the Fall of 2021 due to COVID-19 issues.

A clarifying phone call was held between the NRC staff and APS on March 29, 2020, to discuss the additional information needed. The APS response to the request for additional information is provided in the enclosure to this letter.

No new commitments are being made in this submittal. If you have any questions about this request, please contact Matthew S. Cox, Section Leader, Licensing, at (623) 393-5753.

Sincerely,

Rash, Bruce (Z77439)

Digitally signed by Rash, Bruce (Z77439)
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Enclosure: APS Response to Request for Additional Information – Relief Request 65 – Unit 2, COVID-19, Request for Relief from Bottom Mounted Instrumentation Nozzles and a Pressurizer Nozzle to Surge Line Weld Overlay Examination

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

Enclosure

APS Response to Request for Additional Information

Relief Request 65

Unit 2, COVID-19, Request for Relief from Bottom Mounted Instrumentation Nozzles and a Pressurizer Nozzle to Surge Line Weld Overlay Examination

By letter number 102-08090, dated March 27, 2020 (Agencywide Documents Access and Management System Accession Number ML20088A533), Arizona Public Service Company (APS) reported that it was in Stage 2, *Enhanced,* of the Pinnacle West/APS Pandemic Plan. On March 29, 2020, APS entered Stage 3, *Advanced,* of the Pinnacle West/APS Pandemic Plan, due to an increase in the number of cases in Arizona, a slight increase in absenteeism at Palo Verde due to quarantining and child care, and a positive COVID-19 case at an APS coal generating plant. The following guidelines and restrictions are currently in place at APS, and are unaffected by moving to Stage 3:

1. Employees who do not have a critical need to be at APS facilities must work remotely
2. Employees who must work from an APS facility are to practice strict social distancing
3. Unit 2 Spring refueling outage (2R22) scope has been reduced to limit the number of supporting contract personnel. As a result of removing work from 2R22, the overall number of outage contractors has been reduced by approximately 300.

In the letter, APS requested Nuclear Regulatory Commission (NRC) approval of an alternative to extend scheduled Palo Verde Nuclear Generating Station, Unit 2 inservice inspection (ISI) examinations for the reactor vessel bottom mounted instrumentation (BMI) nozzles and a pressurizer nozzle surge line weld overlay from the currently planned Unit 2 Spring of 2020 refueling outage (2R22) to the next refueling outage (2R23) in the Fall of 2021 due to COVID-19 issues pursuant to a hardship without a compensating increase in quality and safety in accordance with 10 CFR 50.55a(z)(2). The NRC staff requested the following information to complete its review:

<u>PVNGS-RAI-1:</u>

Concerning your leakage monitoring with water inventory balance calculations by the emergency response facilities data acquisition and display system, provide additional detail regarding what actions would be taken per plant procedures once a 0.1 gpm increase in unidentified leakage is detected. That is, provide additional clarification regarding the underlined below:

"In the event that unidentified leakage increases greater than 0.10 gallons per minute (gpm) above the normal, steady state value for a given plant condition during the performance of the RCS water inventory balance, administrative procedures require that the controls and actions for monitoring RCS leakage under the boric acid corrosion control program (BACCP) be implemented. <u>APS would investigate the increased leakage and has the ability to shutdown the unit in a controlled manner prior to a nozzle failure, if unacceptable increased leakage were to occur</u>."

<u>APS Response to RAI-1:</u>

The Reactor Coolant System (RCS) Inventory is calculated by procedure 40ST-9RC02, *ERFDADS (preferred) Calculation of RCS Water Inventory*, with a Surveillance Test (ST) requirement to be performed every 72 hours. The site schedules and performs the ST on a 48-hour interval to proactively identify RCS leakage at an early stage of change. The ST provides steps where the RCS leakage is quantified and compared to the recent history of RCS leakage to better understand if current changes are outside normally expected values.

The steps in procedure 40ST-9RC02, include steps to determine if any procedure action level criteria are exceeded. The action level criteria are provided in the following table.

Action Level	***Action Level Criteria***
Level 1	*Is the rolling average of the last seven performances of Unidentified RCS leak rate greater than 0.1 gpm?*
Level 1	*Are the last nine consecutive Unidentified RCS leak rates greater than baseline mean?*
Level 2	*Are the last two consecutive Unidentified RCS leak rates greater than 0.15 gpm?*
Level 2	*Are two of the last three consecutive Unidentified RCS leak rates greater than [mean Unidentified RCS Leakage + 2 Standard Deviation]?*
Level 3	*Is this Unidentified RCS leak rate greater than 0.3 gpm?*
Level 3	*Is this Unidentified RCS leak rate greater than [mean + 3 Standard Deviation]?*

One of the actions in procedure 40ST-9RC02, states that if ALL of the following conditions exist: (1) An Action Level Criteria is exceeded, (2) Unidentified RCS leakage is greater than 0.07 gpm for three successive surveillance performances, and (3) Unidentified RCS Leakage is trending up to 0.1 gpm, then perform 40OP-9RC03, *RCS Leakage Source Determination*. The 0.1 gpm is consistent with WCAP-16465-NP, *Pressurized Water Reactor Owners Group Standard RCS Leakage Action Levels and Response Guidelines for Pressurized Water Reactors,* and is one-tenth of the Technical Specification (TS) limit for unidentified leakage. Procedure 40OP-9RC03 includes the requirement to identify the leakage source and could include entering containment to identify the source of the leakage. If the source of the leakage is found and isolated, the procedure directs operation personnel to re-perform an RCS leak rate calculation to confirm that the source of leakage has been addressed.

The RCS leakage quantity is reviewed against the TS associated with RCS leakage criteria. Depending on the source identified, a shutdown could be required in accordance with TS Limiting Condition for Operation (LCO) 3.4.14 that has the following specific limits:

- *a. No pressure boundary LEAKAGE;*
- *b. 1 gpm unidentified LEAKAGE;*
- *c. 10 gpm identified LEAKAGE; and*
- *d. 150 gallons per day primary to secondary LEAKAGE through any one steam generator (SG).*

A through-wall leak from either a BMI nozzle or the pressurizer surge line nozzle weld would constitute pressure boundary leakage.

Should any of these limitations be exceeded the appropriate LCO Condition would be entered, and the required actions performed within the specified completion time; including plant shutdown, if required.

PVNGS-RAI-2:

The statement of your requested duration of the alternative appears unclear, "The duration of the request is for the duration of the next Unit 2 operating fuel cycle (2R23), which is Fall 2021." Please clarify that this is intended to say the duration of the request is for Unit 2 operating cycle 23, until refueling outage 2R23 in the Fall of 2021.

APS Response to RAI-2:

The *Duration of Proposed Alternative* is intended to say: "The duration of the request is for one operating cycle (Unit 2 operating cycle 23), until refueling outage 2R23 in the Fall of 2021."

PVNGS-RAI-3:

The staff seeks additional clarification regarding the pressurizer surge line nozzle weld overlay fatigue evaluation discussed on page 9 of the alternative request and identified as Reference 4 to the request. The discussion suggests that a conservative number of heatup and cooldown cycles was assumed in the evaluation. Clarify whether this use of a conservative number of heatup and cooldown cycles was used to also bound other potential thermal cycles during normal power operation, or whether other potential sources of thermal cycling were also explicitly accounted for in the thermal fatigue analysis, and if so how. If the conservative number of heatup and cooldown cycles was used to bound all sources of thermal cycling, please briefly explain how the number of cycles chosen was determined to be bounding for all sources of thermal cycling.

APS Response to RAI-3:

The design of the preemptive weld overlay was performed in accordance with American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC) Section III and Section XI rules utilizing the bounding design interface piping loads, internal pressure, thermal transients, global stratification and residual loads in the sizing of the overlay.

The weld overlay region was analyzed per ASME Section XI assuming 75% of the original pipe wall was cracked over a circumference of 360° and assumed to grow under Primary Water Stress Corrosion Cracking (PWSCC) and Fatigue. These assumed flaws were projected to reach the inside diameter of the overlay in three years.

Evidence of the level of conservatism on the number of transient cycles used in the fracture mechanics evaluation to determine the time to reach the overlay was presented by showing the number of heat-up and cool down cycles in Request Relief 65 for the Unit 2 pressurizer.

Further evidence of cyclic fatigue conservatism is presented by the evaluation for *Environmentally Assisted Fatigue Screening* that was conducted for license extension. The environmental study determined sentinel locations for the environmentally assisted fatigue (EAF) and usage factor (U_{en}) for a number of components. For the pressurizer surge nozzle overlay and the hot leg surge nozzle, the sentinel location for the hot leg surge nozzle contains a U_{en} of 7.58, where pressurizer surge nozzle overlay contains a U_{en} of 2.28. The analysis of the hot leg surge nozzle is bounding for the pressurizer surge nozzle overlay. The analysis for the hot leg surge nozzle results in a design cumulative usage factor of 0.4949, whereas the 60-year usage factor utilizing projected cycle count based on actual plant data for Unit 2 hot leg nozzle is 0.0948. Thermal cycles projected by the end of life for Unit 2 pressurizer surge nozzle overlay are bounded by the design cycles documented in fracture mechanics evaluation.

Based on the above considerations of number of cycles and 60-year life for Level A & B service loads and transients, it can be said that the pressurizer surge line weld overlay with a fatigue life of three years to reach the overlay inside diameter from a 75% cracked pipe is conservative by a factor of four.

The submittal for Relief Request 65 contained Table 1, with a column labeled, *Time to reach Overlay Minimum Allowable Overlay Thickness (2)*. The heading should have read, *Time to reach Overlay or Minimum Allowable Overlay Thickness (2)*. After a closer look at the fatigue evaluation for the Pressurizer surge nozzle overlay, note 2 is not applicable since the evaluation concludes that circumferential flaws "…takes 3 years to reach the overlay…" and "…takes 4 years to reach the overlay…." for the dissimilar metal weld and stainless steel weld, respectively.