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

April 4, 2013

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

SUBJECT: PALO VERDE NUCLEAR GENERATING STATION, UNITS 1, 2, AND 3 – REQUEST FOR RELIEF FROM ASME CODE, SECTION XI REQUIREMENTS REGARDING THE REACTOR VESSEL HEAD FLANGE SEAL LEAK DETECTION PIPING (TAC NOS. MF0447, MF0448, AND MF0449)

Dear Mr. Edington:

By letter dated December 19, 2012 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML12361A256), as supplemented by letter dated February 22, 2013 (ADAMS Accession No. ML13063A062), Arizona Public Service Company (APS, the licensee) requested U.S. Nuclear Regulatory Commission (NRC) review and authorization of relief from the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, for pressure testing the reactor vessel head flange seal leak detection piping at Palo Verde Nuclear Generating Station (PVNGS), Units 1, 2, and 3, for the duration of the third 10-year inservice inspection (ISI) interval.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(a)(3)(ii), the licensee requested to use an alternative on the basis that complying with the system leakage test that is required by the ASME Code, Section XI, Table IWC-2500-1, Examination Category C-H would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

The NRC staff reviewed the subject request and concludes, as set forth in the enclosed safety evaluation, that APS has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(3)(ii). Therefore, the NRC staff authorizes use of the proposed alternative until the end of the third 10-year ISI interval at PVNGS, Units 1, 2, and 3, currently scheduled to end for Unit 1 on July 17, 2018, for Unit 2 on March 17, 2017, and for Unit 3 on January 10, 2018.

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

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

Sincerely,

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

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

Enclosure:
Safety Evaluation

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UNITED STATES
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

REACTOR VESSEL HEAD FLANGE SEAL LEAK DETECTION PIPING

SYSTEM LEAKAGE TEST

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

ARIZONA PUBLIC SERVICE COMPANY

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

1.0 INTRODUCTION

By letter dated December 19, 2012 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML12361A256), as supplemented by letter dated February 22, 2013 (ADAMS Accession No. ML13063A062), Arizona Public Service Company (the licensee) submitted "Reactor Pressure Vessel Head Flange Seal Leak Detection Piping - Relief Request No. 49" for U.S. Nuclear Regulatory Commission (NRC) review and authorization. The licensee requested relief from the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code), Section XI, Table IWC-2500-1, Examination Category C-H, at Palo Verde Nuclear Generating Station (PVNGS), Units 1, 2, and 3, for the duration of the third 10-year inservice inspection (ISI) interval.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, paragraph 55a(a)(3)(ii), the licensee has proposed an alternative on the basis that complying with the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Instead of pressurizing the subject line to the system leakage test pressure required by IWC-5221, the licensee proposes to pressurize the line using the static pressure head of the refueling water prior to performing a VT-2 visual examination.

2.0 REGULATORY EVALUATION

Pursuant to 10 CFR 50.55a(g)(4), "Inservice inspection requirements," ASME Code Class 1, 2, and 3 components (including supports) shall meet the requirements, except the design and access provisions and the pre-service examination requirements, set forth in the ASME Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," to the extent practical within the limitations of design, geometry, and materials of construction of the components. The regulations require that inservice examination of components and system pressure tests conducted during the first 10-year inspection interval and subsequent 10-year inspection intervals comply with the requirements in the latest edition and addenda of Section XI of the ASME Code incorporated by reference in 10 CFR 50.55a(b), "Standards approved for

incorporation by reference," 12 months prior to the start of the 120-month inspection interval, subject to the conditions listed therein.

The regulations in 10 CFR 50.55a(a)(3) state, in part, that alternatives to the requirements of 10 CFR 50.55a(g) may be used, when authorized by the NRC, if (i) The proposed alternatives would provide an acceptable level of quality and safety; or (ii) Compliance with the specified requirements of this section would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Based on analysis of the regulatory requirements, the NRC staff concludes that the NRC has the regulatory authority to authorize the licensee's proposed alternative on the basis that compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Accordingly, the staff has reviewed and evaluated the licensee's request pursuant to 10 CFR 50.55a(a)(3)(ii).

3.0 TECHNICAL EVALUATION

3.1 Licensee's Request for Alternative

3.1.1 Components for which Relief is Being Requested

Reactor Pressure Vessel Flange Leak-off Piping, Number RC-N-081-CCBA-1, ASME Code Class 2, Examination Category C-H, Item Number C7.10

3.1.2 ASME Code Requirements

The Code of record for the PVNGS third 10-year inservice inspection (ISI) interval that is scheduled to end for Unit 1 on July 17, 2018, for Unit 2 on March 17, 2017, and for Unit 3 on January 10, 2018, is the 2001 Edition through the 2003 Addenda of the ASME Code, Section XI.

As stated in the licensee's submittal dated December 19, 2012:

> [ASME Code, Section XI, paragraph] IWC-2500, Table IWC-2500-1, [Examination] Code Category C-H, Item Number C7.10 requires that all Class 2 pressure retaining components be subject to a system leakage test [in accordance with IWC-5220] with a Visual, VT-2 examination each inspection period. The system leakage test is performed at the pressure obtained while the subject portion of the system is performing its normal operating function or during a comparable test.
>
> Per IWC-5222(a), the pressure retaining boundary includes the portion of the system required to operate or support the safety function up to and including the first normally closed valve.

3.1.3 Licensee's Reason for Request

As stated in the licensee's submittal dated December 19, 2012:

> The reactor vessel head flange seal leak detection piping is separated from the reactor coolant pressure boundary by one passive metallic seal, which is the first of two O-rings. The pressure tap for the leak detection piping is located on the vessel flange mating surface. A second O-ring is located on the outside of the pressure tap in the vessel flange. Failure of the inner O-ring is the only condition under which this line is pressurized. Therefore, the line is not expected to be pressurized during the system pressure test following a refueling outage.
>
> The configuration of this piping precludes system pressure testing while the vessel head is removed because the pressure tap would have to be plugged. This would require a design modification to install mechanical threads into the pressure tap on the vessel flange. A threaded plug would need to be installed in the flange face to act as a pressure boundary for each test, then removed after the test. The installation of the modification and subsequent use would incur significant [radiological] dose, which would be inconsistent with as low as reasonably achievable (ALARA) [considerations]. This activity would also present a foreign material exclusion issue for the handling of a very small diameter plug that would be required to be installed to complete the system leakage test at pressure.
>
> The configuration also precludes pressurizing the line externally with the head installed. The closure head contains two concentric grooves that hold the inner and outer O-rings. The O-rings are held in place by a series of retainer clips that are housed in recessed cavities in the flange face. If a pressure test were to be performed with the head installed, the inner O-ring would be pressurized in a direction opposite to its design function. This test pressure would result in a net inward force on the inner O-ring that would tend to push it into the recessed cavity that houses the retainer clips. The thin O-ring material would likely be damaged by the inward force.
>
> Purposely failing or not installing the inner O-ring in order to perform a pressure test would require a new O-ring set to be installed. The time and radiation exposure associated with removing and reinstalling the closure head, replacing the outer O-ring and re-cleaning of the vessel flange mating surface prior to head installation would be an undue hardship. In addition, this special test would require a reactor coolant system heat-up / cooldown cycle. Therefore, compliance with the IWC-5222(b) system pressure test requirements result in an unnecessary hardship without a sufficient compensating increase in the level of quality and safety.

3.1.4 Licensee's Proposed Alternative and Basis for Use

As stated in the licensee's submittal dated December 19, 2012:

> In lieu of the requirements of IWC-5222(b), a VT-2 visual examination of the accessible areas will be performed each inspection period on the piping subjected to the static pressure from the head of water when the reactor cavity is filled. This test will be performed within the frequency specified by table IWC-2500-1 for a System Leakage Test (once each inspection period).
>
> If the inner O-ring should leak during the operating cycle it will be identified by an increase in pressure of the leak-off line above ambient pressure. This leak detection piping has a pressure indicator in the Control Room. This high pressure would actuate an alarm in the Control Room, which would be closely monitored by procedurally controlled operator actions allowing identification of any further compensatory actions required. This piping also acts as a leak-off line to collect leakage which would be routed to the Reactor Coolant Drain Tank.
>
> Additionally, the reactor vessel head flange seal leak detection piping would only function as a Class 2 pressure boundary if the inner O-ring fails, thereby pressurizing the line. If any significant leakage does occur in the leak detection piping during this time of pressurization, it would exhibit boric acid accumulation that would be discernible during the VT-2 visual examination to be performed as proposed in this request.

3.2 NRC Staff Evaluation

The subject leak-off line is associated with the reactor pressure vessel (RPV) closure head flange leakage detection system. The RPV closure head flange is designed with two concentric O-rings that act as flange seals to enable the vessel to be pressurized during normal operation, with the inner O-ring acting as the primary pressure seal for the RPV. The outer O-ring and the leak-off line are designed to support identification of leakage should the primary inner O-ring seal leak. Inner O-ring leakage during the operating cycle would be identified by an increase in the leak-off line pressure, actuating an alarm in the Control Room when 1500 pounds per square inch gauge (psig) is reached. This alarm would be monitored by procedurally controlled operator actions, allowing identification of any further required compensatory actions. The subject leak-off line is not pressurized by primary system water during normal reactor operation, and can only be tested at the Code-required pressure using an external pressure source.

The NRC staff recognizes three possible methods of externally pressurizing the subject line to perform the ASME Code-compliant system leakage test: 1) installing a threaded plug in the flange face during the outage to act as a pressure boundary, and removing it after the examination; 2) pressurizing the leak-off line upon entering the refueling outage prior to removing the RPV head; and 3) pressurizing the leak-off line at the end of the outage after installing new O-rings.

The licensee stated in the submittal dated December 19, 2012, that an ASME Code-compliant examination could be performed by a design modification. This would involve installing mechanical threads into the pressure tap on the vessel flange. A threaded plug would need to be installed in the flange face to act as a pressure boundary for each test, then removed after the test. The licensee also stated that handling a small diameter plug would present a foreign material exclusion issue. The NRC staff concludes that the installation of a mechanical plug and the associated foreign material exclusion issue would present a hardship.

By letter dated February 22, 2013, in response to the NRC staff's request for additional information (RAI) dated January 30, 2013 (ADAMS Accession No. ML13030A401), concerning pressurizing the leak-off line with an external source at the beginning of an outage, the licensee stated, in part, that

> This alternative higher pressure test could be performed at either of 2 operating plateaus: Normal operating pressure and temperature in Mode 3 or at approximately 350 [pounds per square inch absolute (psia)] in Mode 5.
>
> Immediately after reactor shutdown for a refueling outage, the [reactor coolant system (RCS)] is stabilized in Mode 3, followed closely by RCS cooldown. To perform a leak-off line test at or near full RCS pressure would require holding the plant at normal operating pressure and temperature for a minimum of 4 extra hours, beyond the normal refueling outage sequence, while a portable skid is installed and operated. At this time, the entire containment would be a Locked High Radiation Area (LHRA) as radiation protection personnel would not have sufficient time to survey and de-post the walkways and various areas.
>
> Once a cooldown starts, the RCS enters Mode 5 approximately 6 hours later. The RCS is stabilized at approximately 350 psia for about 12 hours to keep the reactor coolant pumps (RCPs) running for peroxide injection / crud removal (to minimize personnel dose during the refueling outage as a whole) and to cool down the RCS metal mass. This 350 psia plateau is maintained for about 12 hours. The portable skid could be installed and operated at this lower pressure plateau. However, during the peroxide injection / crud removal process, which also occurs at this pressure level, the radiation levels in the RCP / Steam Generator (SG) bays increase such that radiation protection personnel prohibit entry into the bays. This would prevent a visual inspection of a large portion of the piping, which is routed through bay 2. As a result, performance of a leak-off line alternative test at this lower RCS pressure would require postponing peroxide injection / crud removal process for a minimum of 4 extra hours while a portable skid is installed and operated.

The NRC staff concludes that the actions required to perform a leak test during either of these operating plateaus would present a hardship.

The final method of pressurizing the leak-off line to the required pressure involves pressurizing the line at the end of the refueling outage with the head installed. This would cause the inner O-ring to be put in a condition opposite to the design function, likely causing O-ring damage. As

a result, the head would be required to be removed and the O-ring replaced prior to plant operation. The NRC staff concludes the heavy lift evolution associated with removing the head and replacing the O-rings would present a hardship or unusual difficulty.

In addition, the NRC staff recognizes the three possible methods of externally pressurizing the subject line may also have associated ALARA hardship considerations.

The licensee proposes to pressurize the subject leak-off line each inspection period using the static pressure present when the reactor cavity is filled, then performing a visual, VT-2 examination of the accessible areas of the piping while it is subjected to the static pressure head of approximately 20 pounds per square inch gauge (psig). In its RAI response dated February 22, 2013, the licensee stated that the subject line is insulated and a minimum 4-hour hold time (after test pressure has been reached) will be observed before performing the visual, VT-2 examination. This is to allow for any leakage to become visible through the insulation, in accordance with the hold time requirements of ASME Code, paragraph IWA-5213(a)(3), for insulated components.

In its RAI response dated February 22, 2013, the licensee stated that the leak-off line is schedule 160 seamless, stainless steel (ASME SA-376 or SA-312 Grade 304) and that there has not been experience with degradation due to corrosion, stress corrosion cracking, or fatigue. The licensee also stated that the "leak-off line is flushed during refueling outages to prevent the buildup of contaminants in the stagnant piping."

The NRC staff concludes, based on evaluation of past performance as well as the service conditions, materials present, and the precautions taken to prevent buildup of contaminants, that service-induced degradation is unlikely.

The NRC staff notes that the system leakage test requirements of the ASME Code, IWC-5220 are focused on demonstrating leak tightness rather than structural integrity. The NRC staff's concern is whether the proposed low-test pressure would be sufficient to demonstrate the leak tightness of the leak-off line. If the leak-off line has a large through-wall crack, leakage would be evident under either a high- or low-pressure test condition. However, if the leak-off line has a very small and tight through-wall crack, the leakage may not be immediately evident under the proposed low pressure test condition. The licensee stated in its RAI response dated February 22, 2013, that the leak-off line is pressurized for a significant length of time during each refueling outage, most recently for about 11.5 days during the Unit 2 outage in fall of 2012. The NRC staff concludes that if any significant leakage were to occur in the leak-off line during the time of pressurization during each refueling outage, boric acid accumulation would be discernible during a subsequent visual examination. The NRC staff, therefore, concludes that the proposed low-test pressure, although not as effective as high-test pressure, will provide reasonable assurance of the leak tightness of the subject leak-off line.

The NRC staff concludes, based on evaluation of the service conditions and the materials of construction, that the proposed visual, VT-2 examination of the subject leak-off line after the reactor cavity is filled provides reasonable assurance of structural integrity and leak tightness. The NRC staff also concludes that requiring compliance with the IWC-5220 system leakage test

requirements would result in a hardship or unusual difficulty without a compensating increase in the level of quality and safety.

4.0 CONCLUSION

As set forth above, the NRC staff determines that the proposed alternative, "Reactor Pressure Vessel Head Flange Seal Leak Detection Piping - Relief Request No. 49," provides reasonable assurance of structural integrity and leak tightness, and that complying with the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(a)(3)(ii) and therefore authorizes use of the proposed alternative until the end of the third 10-year ISI interval at PVNGS, Units 1, 2, and 3, currently scheduled to end for Unit 1 on July 17, 2018, for Unit 2 on March 17, 2017, and for Unit 3 on January 10, 2018.

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

Principal Contributor: Jay Wallace, NRR/DE/EPNB

Date: April 4, 2013

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

Sincerely,

/RA/

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

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

Enclosure:
Safety Evaluation

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