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SUBJECT: Responds to 990422 RAI re request for amend to TS 3.5.3 to extend completion time for low pressure safety injection subsystem.

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APS
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Palo Verde Nuclear
Generating Station

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102-04287-WEI/SAB/GAM
May 21, 1999

U. S. Nuclear Regulatory Commission
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Dear Sirs:

Subject: Palo Verde Nuclear Generating Station (PVNGS)
Units 1, 2, and 3
Docket Nos. STN 50-528/529/530
Response to NRC Request for Additional Information
Regarding Low Pressure Safety Injection Completion Time

In a letter to Arizona Public Service Company (APS) dated April 22, 1999, the NRC requested additional information regarding APS's request for an amendment to Technical Specification 3.5.3 to extend the completion time for a low pressure safety injection subsystem. Enclosed are the responses to the NRC request for additional information.

If you have any questions, please contact Scott A. Bauer at (602) 393-5978. No commitments are being made to the NRC by this letter.

Sincerely,

Will E Ide

WEI/SAB/GAM/rlh

Enclosure

cc: E. W. Merschoff
M. B. Fields
J. H. Moorman
A. V. Godwin [ARRA]

A001

Enclosure

Responses to NRC Request for Additional Information Regarding Low Pressure Safety Injection Completion Time

Responses to NRC Request for Additional Information Regarding Low Pressure Safety Injection Completion Time

NRC Request 1

Discuss what proposed Tier 2 Defense-In-Depth requirements, if any, will be imposed when a low pressure safety injection system is removed from service.

APS Response 1

No additional Tier 2 Defense-In-Depth requirements are proposed as a result of the low pressure safety injection (LPSI) system Technical Specification Completion Time extension.

NRC Request 2

Provide the pertinent information regarding the Tier 3 tools used for the Configuration Risk Management Program; for example, upgrades of matrices.

APS Response 2

Palo Verde Nuclear Generating Station uses a Plant Configuration Risk Indicator Matrix (PCRIM) to assess risk associated with having systems out of service. The control of and requirements for use of the matrix are governed by procedure 30DP-9MT01, "Assessment of Risk When Performing Maintenance." A copy of this procedure was submitted to the NRC with APS letter number 102-04029, dated October 10, 1997.

NRC Request 3

Discuss the available information on any Level 2 and External Event considerations, either quantitative or qualitative.

APS Response 3

Level 2 Information

The Palo Verde PRA does not currently include Level 2 results. Inclusion of the Level 2 results into the PVNGS PRA is being pursued, but is not needed for the proposed LPSI Completion Time extension. Assessment of Large Early Release

associated with the proposed change to the LPSI Completion Time was evaluated in Combustion Engineering Owners Group (CEOG) report number CE NPSD-995, "Joint Applications Report for Low Pressure Safety Injection System AOT Extension," submitted to the NRC with APS letter no. 102-03392, dated June 13, 1995. The following information is excerpted from CE NPSD-995 Section 6.3.5, "Assessment of Large Early Release":

> A review of large early release scenarios for the CE PWRs indicates that early releases arise from a result of the following class of scenarios:
>
> 1. Containment Bypass Events.
>
>   These events include interfacing system LOCAs and steam generator tube ruptures (SGTRs) with a concomitant loss of SG isolation (e.g. stuck open MSSV).
>
> 2. Severe Accidents accompanied by loss of containment isolation.
>
>   These events include any severe accident in conjunction with an initially unisolated containment.
>
> 3. Containment Failure associated with energetic events in the Containment.
>
>   Events causing containment failure include those associated with the High-Pressure Melt Ejection (HPME) phenomena (including direct containment heating (DCH)) and hydrogen conflagrations/detonations.
>
> Of the three release categories, Class 1 tends to represent a large early release potentially direct, unscrubbed fission products, to the environment. Class 2 events encompass a range of releases varying from early to late that may or may not be scrubbed. Class 3 events result in a high-pressure failure of the containment, typically immediately upon or slightly after reactor vessel failure. Detailed Level 2 analysis for the plant condition with one LPSI train inoperable was not performed. However, assessment of the expected change in the large early release fraction was made by assessing the impact of the availability of the LPSI System on the above event categories.
>
> *Containment Bypass Events*
>
> Events contained in this category that may rely on the LPSI for event mitigation include the Large Interfacing System LOCA (i.e. failure of a SDC line). Testing and/or maintenance of containment isolation valves residing in the LPSI System are governed under the plant technical

specifications. Arguments provided in this report are not intended to justify "at power" maintenance of these valves. Thus, no change in the ISLOCA frequency is expected.

ISLOCAs are characterized by continuous and unreplenished loss of RCS inventory and makeup. In these scenarios, core damage ultimately results following the depletion of reactor coolant. Thus, provided that a continuous independent water supply is not available during the accident, the ISLOCA will progress into early core damage regardless of LPSI availability.

Severe Accidents accompanied by Loss of Containment Isolation

Another event contributing to large early fission product releases could occur when an unmitigated large LOCA occurs in conjunction with an initially unisolated containment. Significant fission product releases would not occur unless the containment atmosphere is unscrubbed (that is sprays are inoperable). This latter combination of events is considered of very low probability and would not significantly increase with a decrease in LPSI pump availability.

Containment Failure associated with Energetic events in the Containment

Class 3 events are dominated by RCS transients that occur at high pressure. These events exclude those where LPSI System performance would be called for and therefore LPSI status is not a contributor to this event category. It is therefore concluded that increased unavailability of the LPSI System (as could potentially result as a consequence of an increased AOT) will have a negligible impact on the large early release fraction for CE PWRs.

<u>External Events Information</u>

The LPSI at-power function is to mitigate large LOCA events. The external events of fire, severe weather, and flooding are not considered to be initiators of large LOCA events. The only external events that need to be considered are seismic events.

In letter number 102-03407, dated June 30, 1995, APS submitted to NRC the Individual Plant Examination of External Events (IPEEE) for PVNGS. Section 3 of the IPEEE documents an extensive evaluation of seismic events. Subsection 3.1.3.2, "RLE Response Spectra Conclusions," concluded the following concerning the Review Level Earthquake (RLE):

Based upon the comparison of the RLE spectra and the design basis In-Floor Response Spectra (IFRS), it can be concluded that all seismic category 1 equipment on the IPEEE Safe Shutdown Equipment List, including relays and equipment required for containment performance, was qualified to a higher level of seismic demand than required by IPEEE. The seismic qualification process assures that seismic category 1 equipment, including relays, will perform their intended function during and after the RLE (as well as for the design basis earthquake). This confidence level is greater than would be achieved by meeting only the NP 6041 screening criteria.

Thus a large LOCA is not considered to be a credible consequence of a seismic event for PVNGS, and there is no impact of LPSI availability on External Event Initiated Risk.

<u>NRC Request 4</u>

Provide any recent verification and validation information on the probabilistic risk analysis calculations performed, such as peer reviews and external reviews.

<u>APS Response 4</u>

The updated PVNGS PRA model was documented in APS Engineering Study 13-NS-B67 Revision 0 and approved on October 1, 1998. The update included a complete Level 1 re-quantification and received an APS internal independent review. Three other support studies, 13-NS-B60 At-Power PRA Initiating Events, 13-NS-B62 At-Power Study for Human Reliability Analysis, and 13-NS-B64 Common Cause Failure for Level 1 PRA were reviewed by an outside consultant. ERIN, Engineering and Research, Inc., performed this review and documented it in report number S1179701-473 dated December 10, 1998. These engineering studies and reports are available for review on site.

An independent review of Palo Verde Nuclear Generating Station's (PVNGS) Individual Plant Examination (IPE) report was performed by a team of independent in-house and outside consultants. This review was performed in accordance with Generic Letter 88-20 and is documented in the PVNGS IPE submittal to NRC, APS letter number 161-04750, dated April 28, 1992.

NRC Request 5

Describe the changes made to the probabilistic risk analysis that resulted in a lowering of the incremental conditional core damage probability.

APS Response 5

The PVNGS PRA has combined corrective maintenance (CM) and preventive maintenance (PM) unavailabilities at the component level, rather than at the system level. For the proposed values, the CEOG suggested conservative numbers of 24 hours for CM and 168 hours for PM. Proposed LPSI maintenance duration was based on actual plant data and the assumptions of 24 hours per CM event and one additional PM event of 2/3 AOT duration per cycle per train, resulting in a total yearly unavailability of 157 hours, or 1.79E-02 unavailability probability. The effect on CDF was determined by replacing the CM/PM unavailability of the LPSI pump CM event. With the new value of LPSI pump unavailability, the CDF was re-quantified as 3.93E-05/yr; no significant change from base case.

As stated above in the original submittal, the conservative assumption was made to use 2/3 of the proposed new AOT due to lack of real plant data. With three years of actual maintenance history the following method was used:

PM Mean Duration hours was determined by using the maintenance history tracked for the Maintenance Rule. It is assumed that as a result of the change in AOT that one additional PM event per train per cycle will be scheduled with a duration of 2/3 of the allowed AOT. During the period of 1994 through 1996 there were 317 PM events with a total duration of 909.03 hours. This resulted in 101.0 hours per unit per year. To this amount, 149.33 hours (2/3 of an AOT times 2 trains times 2/3 to account for the 18 month fuel cycle) was added to obtain a proposed total maintenance duration. The proposed maintenance duration was divided by the actual PM events per unit per year plus 2*2/3 (to account for the two new proposed events per cycle). This method is also supported in a CEOG report NPSD-995.

$$\text{PM mean duration} = \{(909.03 / 9) + [2/3*(168)*2/3*2\} / (317 / 9 + 2*2/3)$$
$$= (101 + 149.33) / (35.22 + 4/3) = 6.85 \text{ hours/event.}$$

50-528
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A043

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March 27, 2000

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SUBJECT: PALO VERDE NUCLEAR GENERATING STATION, UNITS 1, 2, AND 3 - EVALUATION OF ALTERNATIVES TO ASME SECTION XI CONTAINMENT INSERVICE INSPECTION REQUIREMENTS (TAC NOS. MA7799, MA7800, MA7801)

Dear Mr. Overbeck:

The staff has reviewed and evaluated the information provided by Arizona Public Service Company (APS) by letter dated December 22, 1999, in support of alternatives to some of the containment inservice inspection requirements contained in American Society of Mechanical Engineers (ASME) Section XI. Based on the information provided in this letter, the staff concludes that, for Relief Requests EE-1, 4, 5, 6, and 7, APS proposed alternatives will provide an acceptable level of quality and safety. Therefore, the proposed alternatives are authorized pursuant to 10 CFR 50.55a(a)(3)(i). For Relief Requests EE-2 and -3, the staff concludes that compliance with the code requirements would result in hardship without a compensating increase in the level of quality and safety, and that APS' proposed alternatives will provide reasonable assurance of containment pressure integrity. Therefore, the proposed alternatives are authorized pursuant to10 CFR 50.55a(a)(3)(ii). The enclosed safety evaluation provides the bases for these conclusions.

Sincerely,

/RA/

Stephen Dembek, Chief, Section 2
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

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

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Enclosure: Safety Evaluation

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

CONTAINMENT INSERVICE INSPECTION RELIEF REQUESTS

ARIZONA PUBLIC SERVICE COMPANY

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

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

1.0 INTRODUCTION

By letter dated December 22, 1999, the Arizona Public Service Company (the licensee) submitted alternatives to some of the containment inservice inspection (ISI) requirements contained in American Society of Mechanical Engineers Boiler and Pressure Vessel (ASME Code) Section XI for the Palo Verde Nuclear Generating Station, Units 1, 2, and 3.

ISI of Class CC (concrete containments), and Class MC (metallic containments) shall be performed in accordance with Section XI of the ASME Code and applicable addenda as required by 10 CFR 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i). Paragraph 50.55a(a)(3) of 10 CFR Part 50 states in part that alternatives to the requirements of paragraph (g) may be used, when authorized by the NRC, if (i) the proposed alternatives would provide an acceptable level of quality and safety or (ii) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

In *Federal Register* Notice No.154, Volume 61, dated August 8, 1996, the NRC amended 10 CFR 50.55a to incorporate by reference the 1992 edition with 1992 addenda of Subsections IWE and IWL of Section XI of the ASME Code. Subsections IWE and IWL provide the requirements for ISI of Class CC and Class MC containments. The effective date for the amended regulation was September 9, 1996, and requires licensees to incorporate the new requirements into their ISI plans and to complete the first containment inspection by September 9, 2001.

The licensee is requesting relief from seven of the ISI requirements contained in the 1992 edition with 1992 addenda of Subsections IWE and IWL of Section XI of the ASME Code. The specifics of the proposed relief requests, and the staff's evaluation of each relief request, are contained in the next section.

2.0 EVALUATION

2.1 Relief Request No. RR-E1

Torque/Tension Testing of Bolted Connections

Code Class	MC (IWE)
Code Reference	ASME Section XI, 1992 Edition, 1992 Addenda IWE-2500, Table IWE-2500-1
Examination Category	E-G
Item Numbers	E 8.20
Component Description	Bolted Connections
Palo Verde Units	1, 2, 3

Requirement:

IWE-2500, Table IWE-2500-1 requires bolt torque-tension tests to be performed on 100 percent of the bolts when the connection has not been disassembled and reassembled during the interval.

2.1.1 Licensee's stated proposed alternative:

The following examinations and tests required by Subsection IWE ensure the structural integrity and leak-tightness of Class MC pressure retaining bolting. Therefore no additional alternative examinations are proposed:

1) Exposed surface of bolted connections shall be visually examined in accordance with the requirements of Table IWE-2500-1, Examination Category E-G, Pressure Retaining Bolting, Item E8.10;

2) Bolted connections shall meet the pressure test requirements of Table IWE-2500-1, Examination Category E-P, All Pressure Retaining Components, Item E9.40; and

3) A general visual examination of the entire containment once each inspection period shall be conducted in accordance with 10CFR50.55a(b)(2)(ix)(E).

2.1.2 Licensee' stated basis for alternative:

10CFR50.55a was amended in the Federal Register (61FR41303) to require the use of the 1992 Edition, 1992 Addenda of Section XI when performing containment examinations. Bolt torque or tension testing is required on bolted connections that have not been disassembled and reassembled during the inspection interval.

Determination of the torque or tension value would require that the bolting be untorqued and then re-torqued or re-tensioned. The performance of the

10CFR50, Appendix J, Type B test itself proves that the bolt torque or tension remains adequate to provide a leak rate that is within acceptable limits. The torque or tension value of bolting only becomes an issue if the leak rate is excessive. Once a bolt is torqued or tensioned, it is not subject to dynamic loading that could cause it to experience significant change.

An in-situ test of an undisturbed connection would not be meaningful. Paint or corrosion on the bolted connection would result in a higher indicated torque and would not be representative of the pre-load on the connection.

Verification of torque or tension values on bolted joints that are proven adequate through Appendix J testing and visual inspection is satisfactory to demonstrate that design function is met. Torque or tension testing is not required on any other ASME Section, Class, 1, 2, or 3 bolted connections or their supports as part of the inservice inspection program.

The requirement for torque testing of containment bolting does not appear in the 1998 Edition of Section XI, Subsection IWE.

2.1.3 Staff Evaluation

The code requires that pressure-retaining bolting that has not been disassembled and reassembled during the inspection interval be torque or tension tested. This examination is used to aid in the determination that a leak-tight seal exists and that the structural integrity of the subject bolted connections is maintained. The licensee proposed to use the existing 10 CFR Part 50, Appendix J, Type B test as an alternative to the code requirement to verify the integrity of penetrations with bolted connections.

The Appendix J, Type B, test provides an adequate method to ensure the leak tightness of the pressure retaining bolting. Therefore, the licensee's proposed alternative is authorized pursuant to 10 CFR 50.55a(a)(3)(i) on the basis that it provides an adequate level of quality and safety.

2.2 Relief Request No: RR-E2

Successive Examination of Containment Repairs

Code Class	MC (IWE)
Code Reference	ASME Section XI, 1992 Edition, 1992 Addenda, IWE-2420(b), IWE-2420(c)
Examination Category	N/A
Item Numbers	N/A
Component Description	Metal (Class MC) Portions of the Containment Building, Containment Liner, Penetrations, Hatches, and Attachments
Palo Verde Units	1, 2, 3

Requirement:

ASME Section XI, 1992 Edition, 1992 Addenda, IWE-2420 (b) states that when component examination results require evaluation of flaws, areas of degradation, or repairs in accordance with IWE-3000, and the component is found to be acceptable for continued service, the areas containing such flaws, degradation, or repairs shall be reexamined during the next inspection period listed in the schedule of the inspection program of IWE-2411 or IWE-2412, in accordance with Table IWE-2500-1, Examination Category E-C (Augmented Examination).

IWE-2420(c) requires that this reexamination continue for at least three consecutive inspection periods.

2.2.1 Licensee's stated proposed alternative:

Relief is sought only from the requirement to reexamine areas that have been repaired. As an alternative PVNGS [Palo Verde] will perform the repair of degraded areas in accordance with an approved Repair/Replacement Program. For degraded areas that are accepted by engineering evaluation, the applicable successive inspection requirements specified in paragraph IWE-2420 will be met.

2.2.2 Licensee's stated basis for alternative:

Pursuant to 10CFR50.55a(a)(3)(ii), relief is requested from the Code requirements stated above on the basis that compliance with this requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality or safety.

The purpose of a repair is to restore the component to an acceptable condition for continued service in accordance with the acceptance standards of IWE-3000. IWA-4150 requires the Owner to conduct an evaluation of the suitability of the repair including consideration of the cause of failure. This requirement for successive examination presupposes that the repair was not suitable. If the repair has restored the component to an acceptable condition, successive examinations are not warranted. If the repair was not suitable, then the repair does not meet Code requirements and the component is not acceptable for continued service. Neither IWB-2420(b), IWC-2420(b), nor IWD-2420(b) require a repair to be subject to successive examination requirements for ASME Class 1, 2, or 3 components respectively. The successive examination of repairs in accordance with IWE-2420(b) constitutes a burden without a compensating increase in quality or safety.

In SECY 96-080, ["Issuance of Final Amendment to 10 CFR 50.55a To Incorporate by Reference ASME BPV Code, Section XI, Division 1, Subsection IWE and Subsection IWL"] response to Comment 3.3 regarding IWE-2420, the NRC stated, "The purpose of IWE-2420(b) is to manage components found to be acceptable for continued service (meaning no repair or replacement at this time) as an Examination Category E-C [Containment Surfaces Requiring Augmented

Examination] component... If the component had been repaired or replaced, then the more frequent examination would not be needed."

The requirement for re-examination of repairs was removed from IWE-2420(b) and (c) in the 1995 Edition,1995 Addenda to ASME Section XI.

2.2.3 Staff Evaluation

When repairs are complete, IWA-4150 requires licensees to evaluate the suitability of the repair. When a repair is required because an item fails, the evaluation will consider the cause of failure to ensure that the repair is suitable. Considering that the failure mechanism is identified and corrected as required and that the repair receives pre-service examinations, as required, the proposed alternative to perform inspections and evaluations following repairs will provide reasonable assurance of structural integrity. Performance of successive examinations presents a hardship on the licensee, due to increased radiation exposure to the personnel conducting the additional examinations, without a compensating increase in quality or safety. Therefore, the licensee's proposed alternative is authorized pursuant to 10 CFR 50.55a(a)(3)(ii) 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. The proposed testing provides reasonable assurance of containment leak tight integrity.

The staff notes that IWB-2420(b), IWC-2420(b), and IWD-2420(b) do not require the successive inspection of repairs for Code Class 1, 2, and 3 components as is required in IWE-2420(b) for Class MC components.

2.3 <u>Relief Request No. RR-E3</u>

Seals and Gaskets

Code Class	MC (IWE)
Code Reference	ASME Section XI, 1992 Edition, 1992 Addenda, IWE-2500,Table IWE-2500-1
Examination Category	E-D, Seals, Gaskets, and Moisture Barriers
Item Numbers	E5.10, Seals and E5.20, Gaskets
Component Description	Seals and Gaskets in the Containment Pressure Boundary
Palo Verde Units	1, 2, 3

Requirement:

ASME Section XI, 1992 Edition, 1992 Addenda, IWE-2500 and Table IWE-2500-1 require seals and gaskets on airlocks, hatches, and other devices that are required to assure containment leak-tight integrity to be visually examined (VT-3) once each interval to assure containment leak-tight integrity.

2.3.1 Licensee's stated proposed alternative:

As an alternative, the leak-tightness of seals and gaskets will be verified using 10CFR50, Appendix J, Type B testing. No additional alternatives to the visual examination, VT-3, of the seals and gasket will be performed.

2.3.2 Licensee's stated basis for alternative:

In accordance with 10CFR50.55a(a)(3)(ii), relief is requested from the Code requirements on the basis that compliance with this requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality or safety.

Seals and gaskets receive a 10CFR50, Appendix J, Type B test. As noted in 10CFR50, Appendix J, the purpose is to measure leakage of containment or penetrations whose design incorporates resilient seals, gaskets, sealant compounds, and electrical penetrations fitted with flexible metal seal assemblies. Physical examination of the seals and gaskets requires disassembling joints, that are proven adequate through Appendix J testing.

For electrical penetrations, disassembly would involve a pre-maintenance Appendix J test, de-termination of cables at electrical penetrations if enough cable slack is not available, disassembly of the joint, removal and examination of the seals and gaskets, reassembly of the joint, re-termination of the cables if necessary, post maintenance testing of the cables, and a post maintenance Appendix J test of the penetration.

For containment hatches, blind flanges, and equipment hatches, the work required would be similar except for the de-termination, re-termination, and testing of cables.

For those penetrations that are routinely disassembled, such as equipment and personnel hatches, a Type B test is required upon reassembly and prior to start-up. Since the Type B test will assure the leak-tight integrity of the connection, the performance of a visual examination would not increase the level of quality or safety.

Seals and gaskets are not included in the definition of the containment pressure-retaining boundary under current Code rules (NE-2110(b)). When the airlocks and hatches containing these materials are tested in accordance with 10CFR50, Appendix J, degradation of the seat or gasket material is revealed by an increase in the leakage rate. In this case, corrective measures would be applied and the component retested. Furthermore, seals and gaskets are specifically excluded from Code rules for Repair and Replacement in IWA-4111(b)(5) (1992 Edition, 1992 Addenda, and 1998 Edition).

The 1995 Edition, 1996 Addenda of Section XI recognizes that disassembly of joints to perform these examinations is not warranted. Note I in Table IWE-2500-1, Examination Category E-D has been modified to state that sealed or gasketed connections need not be disassembled solely for performance of examinations. However, without disassembly, most of the surface of the seals and gaskets would be inaccessible. The requirement to examine seals and gaskets does not appear in the 1998 Edition of ASME Section XI.

2.3.3 Staff Evaluation

The licensee proposes to use the existing 10 CFR Part 50, Appendix J, Type B testing as a verification of seal and gasket integrity, rather than disassembling the subject components for the sole purpose of examination.

Performing the VT-3 examinations on the subject gaskets and seals would require disassembly and reassembly of the mechanical connection for those penetrations that are not routinely disassembled during a refueling outage. The ASME Main Committee and the Board of Nuclear Codes and Standards have also determined that a VT-3 examination of the seals and gaskets is no longer warranted. Both organizations have approved a revision to Subsection IWE to delete the requirement for performing a VT-3 examination of the seals and gaskets. This revision to Subsection IWE was published in the 1998 Edition of the ASME Code, Section XI. Requiring the licensee to disassemble components for the sole purpose of inspecting seals and gaskets would place a significant hardship on the licensee without a compensating increase in quality and safety.

The licensee will verify the leak-tight integrity of seals and gaskets, utilized on penetrations, that are required to assure containment leak-tight integrity, in accordance with the applicable requirements of 10 CFR Part 50, Appendix J. The proposed testing provides reasonable assurance of containment leak-tight integrity. Therefore, the proposed alternative is authorized pursuant to 10 CFR 50.55a(a)(3)(ii) 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.

2.4 Relief Request No. RR-E4

Performance of VT-3 Other Than at End-of-Interval

Code Class	MC (IWE)
Code Reference	ASME Section XI, 1992 Edition, 1992 Addenda Table IWE-2500-1
Examination Category	E-A
Item Numbers	E 1.12
Component Description	Metal Surfaces of the Containment Building
Palo Verde Units	1, 2, 3

Requirement:

ASME Boiler and Pressure Vessel Code, Section XI, 1992 Edition, 1992 Addenda, Table IWE-2500-1 requires that a VT-3 visual examination be performed on 100 percent of the accessible containment surface at the end of the inspection interval.

2.4.1 Licensee's stated proposed alternative:

The VT-3 examination will be performed on accessible surfaces of the containment structure in accordance with Code Case N-601. This Code Case allows the visual examinations to be performed at any time during the interval provided that the requirements for successive inspections stated in IWE-2420 are met.

2.4.2 Licensee's stated basis for alternative:

Pursuant to 10CFR50.55a(a)(3)(i), relief is requested from the Code requirements stated above on the basis that the proposed alternative would provide an acceptable level of quality and safety.

Code Case N-601, "Extent and Frequency of VT-3 Visual Examination for Inservice Inspection of Metal Containments" provides an alternative to the Code requirement to perform 100% of the VT-3 examinations on Item E 1.12 at the end of the interval. It recognizes that it is more important to perform visual examinations on the accessible surfaces of the containment structure during the course of the interval rather than at the end. In this way, the integrity of the containment can be better monitored between the 10CFR50, Appendix J testing and the visual examinations required by Table IWE-2500-1. The successive inspection requirements of IWE-2420 will be maintained.

The proposed alternative examination scheduling is in accordance with Code Case N-601 that has been approved and published by ASME.

The requirements of Code Case N-601 have been incorporated into the 1998 Edition of ASME Section XI, Table IWE-2500-1.

2.4.3 Staff Evaluation

The ASME Code, Table IWE-2500-1, Category E-A, Items E1.12 and E1.20, requires all of the VT-3 visual examinations be performed at the end of the inspection interval.

Performing visual examinations during the course of the inspection interval, as recommended in Code Case N-601, provides a more practical method of performing the inspections than performing all the visual examinations at the end of the interval. In doing this, the integrity of the containment and vent system can be monitored more effectively between the 10 CFR Part 50, Appendix J testing and the visual examination required by Table IWE-2500-1. On this basis, the NRC staff finds that the proposed alternative to use Code Case N-601 provides an adequate

method to perform visual examinations of the containment surface area and vent systems. Therefore, the licensee's proposed alternative is authorized pursuant to 10 CFR 50.55a(a)(3)(i) on the basis that it provides an acceptable level of quality and safety.

2.5 Relief Request No. RR-E5

UT Thickness Measurement of Augmented Exam Areas

Code Class	MC (IWE)
Code Reference	ASME Section XI, 1992 Edition, 1992 Addenda, IWE-2500(c)(3) and IWE-2500(c)(4).
Examination Category	E-C, Containment Surfaces Requiring Augmented Examination
Item Numbers	E4.12
Component Description	Containment Building
Palo Verde Units	1, 2, 3

Requirement:

ASME Section XI, 1992 Edition, 1992 Addenda, Subsection IWE-2500(c)(3) requires that 1-foot-square grids be used when ultrasonic thickness measurements are performed on surfaces requiring augmented examination. IWE-2500(c)(4) requires that the minimum wall thickness within each grid be determined.

2.5.1 Licensee's stated proposed alternative:

The alternative requirements approved by ASME in Code Case N-605 and in the 1998 Edition of ASME Section XI, Subsection IWE will be used when performing UT thickness examinations on areas requiring augmented examination.

2.5.2 Licensee's stated basis for alternative:

Pursuant to 10CFR50.55a(a)(3)(i), relief is requested from the Code requirements stated above on the basis that the proposed alternative would provide an acceptable level of quality and safety.

IWE-2500 (c)(3) and (4) in the 1992 Edition, 1992 Addenda of Section XI require that for surfaces requiring augmented ultrasonic thickness measurement, the surface to be examined is to be marked off into a one-foot square grid and that the minimum thickness in each grid square be marked, recorded, and periodically re-measured. It may be that the area being re-measured is not the area most susceptible to accelerated degradation.

Code Case N-605 and the 1998 Edition of Section XI, Subsection IWE provide an alternative to the one-foot square grid required by IWE-2500(c)(3) in the 1992 Edition with 1992 Addenda of Section XI. The Code Case and 1998 Edition call for setting up a grid system of between 2 and 12 inches and taking measurements at the intersections. The grid size is to be determined by the

Owner. At least 100 intersections must be measured if the augmented examination area is equal to or less than 100 square feet unless the required grid spacing is less than 2 inches. For augmented examination areas greater than 100 square feet, the Code Case and 1998 Edition of Subsection IWE detail a statistical sampling plan for determining the number of intersections to be measured.

If the measurement at an intersection is found to be reduced by more than 10% of the nominal plate thickness, the location of the minimum wall thickness shall be determined and located in each adjoining grid, as required by IWE-2500(c)(4) in the 1992 Edition, 1992 Addenda.

This is similar to the requirements of IWE-2500(c)(4) in the 1992 Edition with 1992 Addenda of Section XI except that under the Code Case and the 1998 Edition, the focus is on areas that exhibit degradation, rather than repeatedly reexamining areas that have not exhibited degradation.

The proposed alternative examination is in accordance with Code Case N-605 that has been approved and published by ASME.

The requirements of Code Case N-605 have been incorporated into the 1998 Edition of ASME Section XI as IWE-2500(b)(3) and (4).

2.5.3 Staff Evaluation

ASME Section XI, 1992 Edition, 1992 Addenda, IWE-2500(c)(3) requires that 1-foot-square grids be used when ultrasonic thickness measurements are performed. The licensee's proposed alternative method is in accordance with Code Case N-605 which is incorporated into the 1998 Edition of ASME Section XI. The alternative is an improvement over the original requirement because the alternative provides more flexibility that results in better detection of the plate thickness degradation. Since the alternative method is an improvement over the code-required method, the licensee's proposed alternative is authorized pursuant to 10 CFR 50.55a(a)(3)(i) on the basis that it provides an acceptable level of quality and safety.

2.6 <u>Relief Request No. RR-E6</u>

Preservice Examination of Reapplied Coatings

Code Class	MC (IWE)
Code Reference	ASME Section XI, 1992 Edition, 1992 Addenda IWE-2200(g)
Examination Category	N/A
Item Numbers	N/A
Component Description	Containment Building
Palo Verde Units	1, 2, 3

Requirement:

ASME Section XI, 1992 Edition, 1992 Addenda, IWE-2200(g) requires that when paint or coatings are reapplied, the condition of the new paint or coating shall be documented in the preservice examination records.

2.6.1 Licensee's stated proposed alternative:

The paint or coatings in the containment will be examined in accordance with established controls per the PVNGS coatings program. If degradation of the coating is identified, additional measures will be applied to determine if the containment pressure boundary is affected.

2.6.2 Licensee's stated basis for alternative:

Paint and coatings are not part of the containment pressure boundary under current Code rules. Because they are not associated with the pressure retaining function of the component, neither paint nor coatings contribute to the structural integrity or leak tightness of the containment (Ref. ASME Section III, NE-2110(b), 1998). Furthermore, the paint and coatings on the containment pressure boundary were not subject to Code rules when they were originally applied and are not subject to ASME XI rules for repair or replacement in accordance with IWA-4111(b)(5). The adequacy of applied coatings is verified through the PVNGS coatings program. Recording the condition of reapplied coatings in the preservice record does not contribute to the containment structural integrity. Should deterioration of the coating in the reapplied area occur, the area would require additional evaluation regardless of the preservice record.

Recording the condition of new paint or coating in the preservice records does not increase the level of quality and safety of the containment.

SECY 96-080, response to Comment 3.2 about IWE-2200(g) states, "In the NRC's opinion, this does not mean that a visual examination must be performed with every application of paint or coating. A visual examination of the topcoat to determine the soundness and the condition of the topcoat should be sufficient." This is currently accomplished through the PVNGS coatings program.

The requirement to perform a preservice examination when paint or coatings are reapplied was removed from the Code in the 1997 Addenda to ASME Section XI.

2.6.3 Staff Evaluation

In the basis for the relief request, the licensee states that it has established the appropriate controls for the coating applications associated with the interior and exterior surfaces of the primary containment structure. These controls are contained in a plant procedure that covers (1) materials to be used, (2) application methods, (3) inspection, (4) personnel qualification, (5) repair, and (6) documentation. The plant procedure is written to comply with the applicable

requirements of Regulatory Guide 1.54, "Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants," ANSI N5.12, "Protective Coatings (Paints) for the Nuclear Industry," ANSI N101.2, "Protective Coatings (Paints) for Light Water Nuclear Reactor Containment Facilities," and ANSI N101.4, "Quality Assurance for Protective Coatings Applied to Nuclear Facilities." The licensee's Protective Coatings Program provides a conservative approach to the inspection and documentation of new coatings and as such, the staff concludes that the proposed alternative provides an acceptable level of quality and safety. Therefore, the licensee's proposed alternative is authorized pursuant to 10 CFR 50.55a(a)(3)(i).

2.7 Relief Request No. RR-E7

VT Prior to Removal of Coatings

Code Class	MC (IWE)
Code Reference	ASME Section XI, 1992 Edition, 1992 Addenda IWE-2200(b)
Examination Category	N/A
Item Numbers	N/A
Component Description	Containment Building
Palo Verde Units	1, 2, 3

Requirement:

ASME Section XI, 1992 Edition, 1992 Addenda, IWE-2500(b) requires that when paint or coatings are to be removed, the paint or coatings shall be visually examined in accordance with Table IWE-2500-1 prior to removal.

2.7.1 Licensee's stated proposed alternative:

The condition of the containment vessel base material will be verified prior to the application of new paint or coating as required by the PVNGS coating program. If degradation is identified, additional measures will be applied to determine if the containment pressure boundary is affected. Repairs to the primary containment boundary, if required, will be conducted in accordance with ASME Section XI Code rules.

2.7.2 Licensee's stated basis for alternative:

Pursuant to 10CFR50.55a(a)(3)(i), relief is requested from the Code requirements stated above on the basis that the proposed alternative would provide an acceptable level of quality and safety.

Paint and coatings are not part of the containment pressure boundary under current Code rules as they are not associated with the pressure retaining function of the component (ASME Section III, Paragraph NE-2110(b), 1998). The interiors of containments are painted to prevent rusting and to facilitate decontamination. Neither paint nor coatings contribute to the structural integrity or leak tightness of the containment.

Furthermore, the paint and coating on the containment pressure boundary were not subject to ASME Code rules when they were originally applied and are not subject to ASME Section XI rules for repair or replacement in accordance with IWA-4111(b)(5).

The 1998 Edition of ASME Section XI does not contain this requirement to inspect coatings prior to their removal.

2.7.3 Staff Evaluation

The purpose of performing the visual examination per IWE-2500(b) is to identify any evidence of base metal degradation prior to removal of the coating or paint. As an alternative to the requirements of IWE-2500(b), the licensee has proposed to inspect the coatings, including paints, using its protective coating program. The licensee informed the staff that the protective coating program at Palo Verde has been written to comply with the applicable requirements of Regulatory Guide 1.54 and ANSI codes such as ANSI N101.4. Section 6 of ANSI N101.4 requires stringent inspection of the entire completed coating work by qualified coating inspection personnel, as well as quality assurance documentation. The Palo Verde Updated Final Safety Analysis Report further discusses compliance of the coating program with Regulatory Guide 1.54. The licensee states that degradation of the base metal would be identified at this time and that corrective actions would be initiated prior to the re-application of the coating or paint. Based upon the licensee's verification of sound base metal prior to application of new coatings, the staff considers the proposed alternative, as stated by the licensee, adequate for protecting the containment surfaces. Therefore, the licensee's proposed alternative is authorized pursuant to 10 CFR 50.55a(a)(3)(i) on the basis that it provides an acceptable level of quality and safety.

3.0 CONCLUSION

Based on the information provided in the relief requests, the staff concludes that, for Relief Requests RR-E1, -E4, -E5, -E6 and -E7, the licensee's proposed alternatives will provide an acceptable level of quality and safety. Therefore, the proposed alternatives are authorized pursuant to 10 CFR 50.55a(a)(3)(i). For Relief Requests RR-E2 and -E3, the staff concludes that compliance with the code requirements would result in hardship without a compensating increase in the level of quality and safety, and that the licensee's proposed alternatives will provide reasonable assurance of containment pressure integrity. Therefore, the proposed alternatives are authorized pursuant to 10 CFR 50.55a(a)(3)(ii).

Principal Contributor: S. B. Kim

Date: March 27, 2000

50-528
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Mr. Gregg R. Overbeck
Senior Vice President, Nuclear
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P. O. Box 52034
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SUBJECT: PALO VERDE NUCLEAR GENERATING STATION, UNITS 1, 2, AND 3 - USE OF MECHANICAL NOZZLE SEAL ASSEMBLIES (TAC NOS. MA7737, MA7738, AND MA7740)

Dear Mr. Overbeck:

In a letter dated September 24, 1999, Arizona Public Service Company (APS) submitted a request to use the mechanical nozzle seal assembly (MNSA) as an alternative repair method pursuant to 10 CFR 50.55a(a)(3)(i) for the Palo Verde Nuclear Generating Station, Units 1, 2, and 3. Specifically, APS requested authorization for alternative use of MNSAs for reactor coolant system hot leg instrumentation and sampling nozzles for an installation period not to exceed two operating cycles. The MNSAs would be installed over those nozzles found to exhibit signs of leakage that are not part of the planned replacement activities for the plant outage.

The staff has completed its review of your request, and our findings are contained in the enclosed safety evaluation. The staff concludes that, pursuant to 10 CFR 50.55a(a)(3)(i), the use of MNSAs as an alternate to an ASME Section XI Code repair on any leaking nozzles of the type describe above, is permissible for a period not to exceed two operating cycles, since it is found to provide an acceptable level of quality and safety.

Sincerely,

/RA/

Stephen Dembek, Chief, Section 2
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

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

Enclosure: Safety Evaluation

cc w/encl: See next page

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO THE MECHANICAL NOZZLE SEAL ASSEMBLY

ARIZONA PUBLIC SERVICE COMPANY

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

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

1.0 INTRODUCTION

By letter dated September 24, 1999, the Arizona Public Service Company (APS, the licensee) submitted a request to use the mechanical nozzle seal assembly (MNSA) as an alternative repair method pursuant to 10 CFR 50.55a(a)(3)(i) for the Palo Verde Nuclear Generating Station (Palo Verde), Units 1, 2, and 3. Specifically, APS requested authorization for alternative use of MNSAs for reactor coolant system (RCS) hot leg instrumentation and sampling nozzles for an installation period not to exceed two operating cycles. The MNSAs would be installed over those nozzles found to exhibit signs of leakage that are not part of the planned replacement activities for the plant outage.

2.0 DISCUSSION

2.1 Background

The potential exists for leaks to occur in RCS hot leg Alloy 600 instrumentation and sampling nozzles due to primary water stress corrosion cracking. These nozzles are welded to the RCS hot leg piping walls with inner diameter J-groove welds. The typical repair of these nozzles utilizes either an internal or external weld repair, or a half nozzle replacement. As an alternative under the provisions of 10 CFR 50.55a(a)(3)(i), the use of an MNSA is proposed as a repair to restore nozzle integrity and prevent leakage of nozzle assemblies for up to two cycles of operation.

Although there are currently no identified nozzle leaks at Palo Verde, the licensee has undertaken a proactive long-term Alloy 600 nozzle replacement plan. The nozzle replacement plan at Palo Verde calls for replacement of all Alloy 600 RCS hot leg instrumentation and sampling nozzles by the completion of the twelfth cycle unit refueling outages. MNSAs would be used if nozzle leaks are identified that would require full-core offload and drain down to facilitate weld repair and replacement. Unplanned replacement of these nozzles could significantly increase plant outage duration for no significant safety benefit in comparison to the use of MNSAs combined with a well-planned nozzle replacement effort.

An MNSA is a mechanical device consisting of a split gasket/flange assembly that is placed around a leaking instrument nozzle. The gasket is made of Grafoil packing, a graphite

compound that is compressed within the assembly to prevent RCS leakage past the nozzle. This assembly is bolted into holes drilled and threaded on the outer surface of the RCS component wall. Another assembly is bolted to the flanges, which serves as the structural attachment of the nozzle to the wall. This assembly serves to carry the loads in lieu of the "J" welds on the Alloy 600 nozzles.

The NRC has approved the use of this MNSA design for similar applications at the San Onofre Nuclear Generating Station, Units 2 and 3 (NRC letters dated February 17, 1998 and January 29, 1999) and at the Waterford Generating Station, Unit 3 (NRC letter dated March 25, 1999).

2.2 Licensing Basis

Section 50.55a to Title 10 of the *Code of Federal Regulations* (10 CFR 50.55a) requires, in part, that all inservice examinations and system pressure tests conducted during the first 10-year interval and subsequent intervals on American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) Class 1, 2, and 3 components must comply with the requirements in the latest edition and addenda of Section XI incorporated by reference in 10 CFR 50.55a(b) on the date 12 months prior to the start of the 10-year interval. By reference to and implementation of paragraphs IWB-3132 or IWB-3142 to Section XI of the ASME Code, 10 CFR 50.55a also requires that existing flaws in ASME Code Class components be removed by mechanical means, or else that the components be repaired or replaced to the extent necessary to meet the acceptance standards in Article IWB-3000 of Section XI to the code. Detection of leaks in the structural portion of an ASME Code Class 1, 2, or 3 component is direct evidence of a flaw in the component.

Paragraph IWA-4170 of Section XI of the ASME Code requires that repairs and the installation of replacements to the reactor coolant pressure boundary be performed and reconciled in accordance with the Owner's Design Specifications and Original Code of Construction for the component or system. The RCS hot legs of the Palo Verde units were designed and constructed to the rules of ASME Section III, 1974 Edition through and including the Summer 1974 Addenda.

Paragraph NB-3671.7 to Section III of the ASME Code, "Sleeve Coupled and Other Patented Joints," requires that ASME Code Class 1 joints be designed to meet the following criteria:

(1) provisions must be made to prevent separation of the joint under all service loading conditions,

(2) the joint must be designed to be accessible for maintenance, removal, and replacement activities, and

(3) the joint must either be designed in accordance with the rules of Section III to the ASME Code, Subarticle NB-3200, or else be evaluated using a prototype of the joint that will be subjected to additional performance tests in order to determine the safety of the joint under simulated service conditions.

3.0 EVALUATION

Section 50.55a(a)(3) of 10 CFR allows licensees to use alternatives to the requirements of the ASME Code when authorized by the Director of the Office of Nuclear Reactor Regulation. The licensee must demonstrate that, pursuant to the requirements of 10 CFR 50.55a(a)(3)(i), the alternatives would provide an acceptable level of quality and safety in lieu of meeting the requirements, or that, pursuant to the requirements of 10 CFR 50.55a(a)(3)(ii), complying with the requirements of 10 CFR 50.55a would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

The licensee requests the use of MNSAs pursuant to 10 CFR 50.55a(a)(3)(i), stating that this alternative provides an acceptable level of quality and safety. In order to determine if the MNSAs provided an acceptable level of quality and safety, the staff compared the MNSA design and operational characteristics to the applicable ASME requirements, reviewed the MNSAs' resistance to corrosion for the intended service period, and evaluated the licensee's commitments associated with the use of the MNSAs. The staff's review is described below.

The MNSAs are designed, fabricated, and constructed using approved ASME Code materials in accordance with the applicable rules of ASME Section III. The MNSAs are designed to prevent separation of the joint under all service loadings. This design is supported by technical analysis and tests that meet the design criteria specified in ASME Section III. Additionally, MNSA installations are accessible for maintenance, removal, and replacement.

The Combustion Engineering (CE) Design Report No. V-PENG-DR- 007, Rev. 1, Addendum to CENC-1500, CENC-1590 and CENC-1642, "Analytical Reports for Arizona Units 1, 2 and 3 Piping," was provided as an attachment to the licensee's September 24, 1999, letter. This addendum demonstrates that stresses under all service conditions do not exceed the code allowables as stated in ASME Section III and that fatigue limits are not exceeded using the conditions in the original Palo Verde design specification.

Modification of the RCS hot leg for MNSA installation has been analyzed in accordance with the Original Construction Code for the Palo Verde Main Loop Piping (ASME Section III, 1974 Edition, Summer 1974 Addenda). The analysis, contained in the addendum referenced above, included the following items and documented the required ASME Section XI reconciliation for the use of a component built to a later edition of the code.

- Fatigue analysis to demonstrate that the code-prescribed cumulative usage factor of 1.0 is not exceeded (NB-3222.4)
- Analysis to demonstrate adequate reinforcement in the wall of the RCS piping for the bolt holes (NB-3643.3)
- Analysis to demonstrate stresses do not exceed the allowables as stated in the code.

The staff concludes that the applicable ASME Code requirements are met by the MNSA design and installation criteria, and that the MNSAs can remain in operation for the period of time requested by the licensee (i.e., not to exceed two cycles).

The licensee provided an evaluation to address potential corrosion of the nozzle bore holes, corrosion of the pipe outside diameter (O.D.) surface, galvanic corrosion, and stress corrosion cracking (SCC) of the MNSA fasteners. The results of this evaluation are summarized as follows:

- Laboratory corrosion data and service experience indicate that any corrosion of the carbon steel in the hot leg Alloy 600 nozzle holes will be minor and will not affect the requested duration of the MNSA repair (i.e., not to exceed two cycles).

- Boric acid corrosion of the materials of construction for the MNSA and the O.D. piping surfaces have been addressed by testing and analysis. With the inspections currently performed, a leaking MNSA would be detected before significant corrosion of the piping occurs.

- There is no history of galvanic corrosion problems in similar applications where carbon steel is in contact with a Grafoil seal. This particular combination is used in other applications where the low alloy (or carbon steel) is frequently inspected (for example, steam generator secondary side manway and hand hole applications). The MNSA application is similar (i.e., Grafoil material is in contact with carbon steel and inspections will be conducted at each refueling outage) and for these reasons significant galvanic corrosion is not expected. In addition, the Grafoil used in the MNSA is Grade GTJ, which has been treated with ammonium phosphate to inhibit corrosion. The corrosion protection provided by this inhibitor is comparable to sacrificial inhibitors such as zinc or aluminum. Testing has shown that GTJ Grafoil significantly reduces the galvanic corrosion process. The licensee also noted that, in the absence of leakage past the Grafoil seal, the annulus will become stagnant and will not allow replenishment of the boric acid or oxygen.

- Testing in pressurized water reactor environments and concentrated boric acid solutions and service experience indicate that A-286 bolts in the MNSA application will operate indefinitely without SCC failures under normal conditions. If the MNSA device leaks, the bolts may be exposed to borated water or steam under conditions in which deposits or slurries will develop. Under these conditions and at stress levels present in the MNSA application, these bolts will operate satisfactorily for more than one fuel cycle. A leaking MNSA will be discovered and repaired as part of the walk-down inspections performed in response to Generic Letter 88-05, *Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants.* These walk-down inspections are performed prior to entering unit outages. Therefore, the existence of leaking MNSA conditions would be limited to one cycle.

Based on the above evaluation of potential corrosion effects, the staff concludes that there are no significant corrosion issues associated with the application of the mechanical nozzle seal assemblies to hot leg piping at Palo Verde. The data indicates that corrosion of the nozzle hole will be acceptable over the requested two-cycle period of use.

The licensee has committed to the following actions should MNSAs need to be utilized in any of the Palo Verde units.

(1) As required by IWA-4820, a VT-1 preservice inspection will be performed on all MNSA installations in accordance with IWB-2200.

(2) During plant startup (Mode 3) after initial MNSA installation and during subsequent plant restarts following outages, the MNSAs will be VT-2 examined (without insulation) for leakage. Additionally, VT-3 exams will be performed to verify general structural and mechanical condition of the MNSAs.

(3) In accordance with ASME Section XI, IWA-4710(b)(5), component connections, piping, and associated valves that are NPS 1 and smaller are exempt from pressure testing. However, to ensure quality of installation and continued operation with the absence of leakage, a pressure test with visual inspection will be performed on each of the installed MNSAs with the insulation removed. The test will be performed as part of plant restart and will be conducted at normal operating pressure with the test temperature determined in accordance with the Palo Verde pressure and temperature limits as stated in the Palo Verde Technical Specifications.

(4) This request for alternative is for up to two cycles of operation. Prior to exceeding two operating cycles, installed MNSAs will be removed and nozzle replacement activities will be implemented as part of the licensee's long-term Alloy 600 nozzle replacement strategy.

(5) APS will verify pipe wall thickness prior to machining MNSA bolt holes to further assure that adequate pipe wall reinforcement exists.

The staff has reviewed the above licensee commitments and conclude that they are sufficient to assure proper installation and operation of the MNSAs for their intended use and duration.

4.0 CONCLUSION

Section 50.55a(a)(3) of Title 10 of the *Code of Federal Regulations* states that alternatives to the requirements of paragraph (g) may be used, when authorized by the NRC, if "(i) The proposed alternatives would provide an acceptable level of quality and safety, or (ii) Compliance with the specified requirements of this section would result in hardship or unusual difficulty without compensating increase in the level of quality and safety." The staff concludes that, pursuant to 10 CFR 50.55a(a)(3)(i), the use of MNSAs as an alternate to an ASME Section XI Code repair on any leaking nozzles of the type describe above, is authorized for a period not to exceed two operating cycles, since it is found to provide an acceptable level of quality and safety.

Principal Contributor: Mel Fields

Date: March 16, 2000

50-528
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

February 14, 2000

Mr. Gregg R. Overbeck
Senior Vice President, Nuclear
Arizona Public Service Company
P. O. Box 52034
Phoenix, AZ 85072-2034

SUBJECT: PALO VERDE NUCLEAR GENERATING STATION, UNITS 1, 2, AND 3 - CLOSEOUT OF GENERIC LETTER 99-02 (TAC NOS. MA5824, MA5825, AND MA5826)

Dear Mr. Overbeck:

On June 3, 1999, the U.S. Nuclear Regulatory Commission (NRC) issued Generic Letter (GL) 99-02, "Laboratory Testing of Nuclear-grade Activated Charcoal," to all holders of operating licenses for nuclear power reactors, except those who have permanently ceased operations and have certified that fuel has been permanently removed from the reactor vessel.

The purpose of the GL was to:

(1) Alert addressees that the NRC has determined that testing nuclear-grade activated charcoal to standards other than American Society for Testing and Materials (ASTM) D3803-1989, "Standard Test Method for Nuclear-Grade Activated Carbon," does not provide assurance for complying with the current licensing basis as it relates to the dose limits of General Design Criterion (GDC) 19 of Appendix A to Part 50 of Title 10 of the *Code of Federal Regulations* (10 CFR) and Subpart A of 10 CFR Part 100.

(2) Request that all addressees determine whether their Technical Specifications (TS) reference ASTM D3803-1989 for charcoal filter laboratory testing. Addressees whose TS do not reference ASTM D3803-1989 should either amend their TS to reference ASTM D3803-1989 or propose an alternative test protocol and provide the information discussed in the requested actions.

(3) Alert addressees to the staff's intent to exercise enforcement discretion under certain conditions.

(4) Request that all addressees send the NRC written responses on the implementation of the actions requested in this GL.

In GL 99-02, the NRC staff established the following four groups of plants:

(1) plants in compliance with their TS that test in accordance with ASTM D3803-1989

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(2) plants in compliance with their TS that test in accordance with a test protocol other than ASTM D3803-1989

(3) plants not in compliance with their TS that test in accordance with ASTM D3803-1989

(4) plants not in compliance with their TS that test in accordance with a test protocol other than ASTM D3803-1989

In response to GL 99-02, you provided a letter dated November 19, 1999, for the Palo Verde Nuclear Generating Station, Units 1, 2, and 3. In this letter you stated that the Palo Verde units were in compliance with their TS, but were using a test protocol other than ASTM D3803-1989. Associated with your response was a request for a license amendment to change your TS to provide for testing as described in the GL. The NRC staff has reviewed your response and has concluded that you have provided the requested information and a TS change request in accordance with the GL. In addition, you state that Palo Verde already performs testing in accordance with ASTM D3803-1989 for purposes of performance monitoring. Therefore, we consider GL 99-02 to be closed for your facilities. The TS change will be reviewed as a separate, plant-specific action under TAC Nos. MA7743, MA7744, and MA7745. We thank you for your prompt and complete response.

If you have any questions regarding this matter, please contact me at (301)415-3062.

Sincerely,

/RA/

Mel B. Fields, Project Manager, Section 2
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

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

December 21, 1999

Mr. Gregg R. Overbeck
Senior Vice President, Nuclear
Arizona Public Service Company
P. O. Box 52034
Phoenix, AZ 85072-2034

SUBJECT: CLOSEOUT OF GENERIC LETTER 96-05 FOR THE PALO VERDE NUCLEAR GENERATING STATION UNITS 1, 2, AND 3 (TAC NOS. M97080, M97081, AND M97082)

Dear Mr. Overbeck:

On September 18, 1996, the NRC issued Generic Letter (GL) 96-05, "Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves," requesting each nuclear power plant licensee to establish a program (or to ensure the effectiveness of its current program) to verify on a periodic basis that safety-related motor-operated valves (MOVs) continue to be capable of performing their safety functions within the current licensing bases of the facility.

On November 14, 1996, Arizona Public Service Company (APS) submitted a 60-day response to GL 96-05 notifying the NRC that it would review the MOV program at the Palo Verde Nuclear Generating Station, Units 1, 2 and 3, to determine whether any changes were appropriate in light of the information in GL 96-05. On March 18, 1997, APS submitted a 180-day response to GL 96-05 stating that it had implemented a program to periodically verify safety-related MOVs at Palo Verde are capable of performing their safety functions. In this letter, APS provided a summary description of its MOV periodic verification program. In a letter dated July 19, 1998, APS updated its commitment to GL 96-05. On June 15, 1999, APS provided a response to a request for additional information regarding GL 96-05 forwarded by the NRC staff on March 17, 1999.

The NRC staff has reviewed APS' submittals and applicable NRC inspection reports for the MOV program at Palo Verde. The staff finds that APS has established an acceptable program to periodically verify the design-basis capability of the safety-related MOVs at Palo Verde through its commitment to all three phases of the Joint Owners Group (JOG) Program on MOV Periodic Verification and the additional actions described in its submittals. As discussed in the enclosed safety evaluation (SE), the staff concludes that APS is adequately addressing the actions requested in GL 96-05. As part of this review, the staff finds that the interim MOV static test program proposed for the Palo Verde units as an alternative to the JOG interim static test interval is acceptable.

The NRC staff may conduct inspections at Palo Verde to verify the implementation of the MOV periodic verification program is in accordance with APS' commitments cited in this SE and in

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the NRC's SE dated October 30, 1997, on the JOG Program on MOV Periodic Verification. Such an inspection could include an evaluation of APS' ranking of MOVs by their risk significance where applied as part of APS' implementation of its commitment to the three phases of the JOG program.

This completes the staff's efforts on TAC Nos. M97080, M97081 and M97082. If you have any questions regarding this matter, please contact me at (301) 415-3062.

Sincerely,

/s/

Mel Fields, Project Manager, Section 2
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. STN 50-528, STN 50-529,
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the NRC's SE dated October 30, 1997, on the JOG Program on MOV Periodic Verification. Such an inspection could include an evaluation of APS' ranking of MOVs by their risk significance where applied as part of APS' implementation of its commitment to the three phases of the JOG program.

This completes the staff's efforts on TAC Nos. M97080, M97081 and M97082. If you have any questions regarding this matter, please contact me at (301) 415-3062.

Sincerely,

Mel Fields, Project Manager, Section 2
Project Directorate IV & Decommissioning
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Docket Nos. STN 50-528, STN 50-529,
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Enclosure: Safety Evaluation

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

GENERIC LETTER 96-05, "PERIODIC VERIFICATION OF

DESIGN-BASIS CAPABILITY OF SAFETY-RELATED MOTOR-OPERATED VALVES,"

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

DOCKET NUMBERS STN 50-528, 50-529 and 50-530

1.0 INTRODUCTION

Many fluid systems at nuclear power plants depend on the successful operation of motor-operated valves (MOVs) in performing their safety functions. Several years ago, MOV operating experience and testing, and research programs sponsored by the nuclear industry and the NRC, revealed weaknesses in a wide range of activities (including design, qualification, testing, and maintenance) associated with the performance of MOVs in nuclear power plants. For example, some engineering analyses used in sizing and setting MOVs did not adequately predict the thrust and torque required to operate valves under their design-basis conditions. In addition, inservice tests of valve stroke time under zero differential-pressure and flow conditions did not ensure that MOVs could perform their safety functions under design-basis conditions.

Upon identification of the weaknesses in MOV performance, significant industry and regulatory activities were initiated to verify the design-basis capability of safety-related MOVs in nuclear power plants. After completion of these activities, nuclear power plant licensees began establishing long-term programs to maintain the design-basis capability of their safety-related MOVs. This safety evaluation (SE) addresses the program developed by Arizona Public Service (APS or the licensee) to periodically verify the design-basis capability of safety-related MOVs at Palo Verde Nuclear Generating Station, Units 1, 2, and 3.

2.0 REGULATORY REQUIREMENTS

The NRC regulations require that MOVs important to safety be treated in a manner that provides assurance of their intended performance. Criterion 1 to Appendix A, "General Design Criteria for Nuclear Power Plants," to Part 50 of Title 10 of the *Code of Federal Regulations* (10 CFR Part 50) states, in part, that structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. The quality assurance program to be applied to safety-related components is described in Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50. In Section 50.55a of 10 CFR Part 50, the NRC requires licensees to establish inservice testing (IST) programs in accordance with Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code.

In response to concerns regarding MOV performance, NRC staff issued Generic Letter (GL) 89-10 (June 28, 1989), "Safety-Related Motor-Operated Valve Testing and Surveillance," which requested that nuclear power plant licensees and construction permit holders ensure the capability of MOVs in safety-related systems to perform their intended functions by reviewing MOV design bases, verifying MOV switch settings initially and periodically, testing MOVs under design-basis conditions where practicable, improving evaluations of MOV failures and necessary corrective action, and trending MOV problems. The staff requested that licensees complete the GL 89-10 program within approximately three refueling outages or 5 years from the issuance of the generic letter. Permit holders were requested to complete the GL 89-10 program before plant startup or in accordance with the above schedule, whichever was later.

The NRC staff issued seven supplements to GL 89-10 that provided additional guidance and information on MOV program scope, design-basis reviews, switch settings, testing, periodic verification, trending, and schedule extensions. GL 89-10 and its supplements provided only limited guidance regarding MOV periodic verification and the measures appropriate to assure preservation of design-basis capability. Consequently, the staff determined that additional guidance on the periodic verification of MOV design-basis capability should be prepared. On September 18, 1996, the NRC staff issued GL 96-05, "Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves," requesting each licensee establish a program, or ensure the effectiveness of its current program, to verify on a periodic basis that safety-related MOVs continue to be capable of performing their safety functions within the current licensing bases of the facility. In GL 96-05, the NRC staff summarized several industry and regulatory activities and programs related to maintaining long-term capability of safety-related MOVs. For example, GL 96-05 discussed non-mandatory ASME Code Case OMN-1, "Alternative Rules for Preservice and Inservice Testing of Certain Electric Motor Operated Valve Assemblies in LWR [Light-Water Reactor] Power Plants, OM Code 1995 Edition; Subsection ISTC," which allows the replacement of ASME Code requirements for MOV quarterly stroke-time testing with exercising of safety-related MOVs at least once per operating cycle and periodic MOV diagnostic testing on a frequency to be determined on the basis of margin and degradation rate. In GL 96-05, the NRC staff stated that the method in OMN-1 meets the intent of the generic letter with certain limitations. The NRC staff also noted in GL 96-05 that licensees remain bound by the requirements in their code of record regarding MOV stroke-time testing, as supplemented by relief requests approved by the NRC staff.

In GL 96-05, licensees were requested to submit the following information to the NRC:

a. within 60 days from the date of GL 96-05, a written response indicating whether or not the licensee would implement the requested actions; and

b. within 180 days from the date of GL 96-05, or upon notification to the NRC of completion of GL 89-10 (whichever is later), a written summary description of the licensee's MOV periodic verification program.

The NRC staff is preparing an SE on the response of each licensee to GL 96-05. The NRC staff intends to rely to a significant extent on an industry initiative to identify valve age-related degradation which could adversely affect the design-basis capability of safety-related MOVs (described in Section 3.0) where a licensee commits to implement that industry program. The NRC staff will conduct inspections to verify the implementation of GL 96-05 programs at nuclear power plants as necessary.

3.0 JOINT OWNERS GROUP PROGRAM ON MOV PERIODIC VERIFICATION

In response to GL 96-05, the Boiling Water Reactor Owners Group (BWROG), Westinghouse Owners Group (WOG), and Combustion Engineering Owners Group (CEOG) jointly developed an MOV periodic verification program to obtain benefits from the sharing of information between licensees. The Joint Owners Group (JOG) Program on MOV Periodic Verification is described by the BWROG in its Licensing Topical Report NEDC-32719, "BWR Owners' Group Program on Motor-Operated Valve (MOV) Periodic Verification," and described by the WOG and the CEOG in their separately submitted Topical Report MPR-1807, "Joint BWR, Westinghouse and Combustion Engineering Owners' Group Program on Motor-Operated Valve (MOV) Periodic Verification." The stated objectives of the JOG program on MOV Periodic Verification are (1) to provide an approach for licensees to use immediately in their GL 96-05 programs; (2) to develop a basis for addressing the potential age-related increase in required thrust or torque under dynamic conditions; and (3) to use the developed basis to confirm, or if necessary to modify, the applied approach. The specific elements of the JOG program are (1) providing an "interim" MOV periodic verification program for applicable licensees to use in response to GL 96-05; (2) conducting a dynamic testing program over the next 5 years to identify potential age-related increases in required thrust or torque to operate gate, globe, and butterfly valves under dynamic conditions; and (3) evaluating the information from the dynamic testing program to confirm or modify the interim program assumptions.

The JOG interim MOV periodic verification program includes (1) continuation of MOV stroke-time testing required by the ASME Code IST program; and (2) performance of MOV static diagnostic testing on a frequency based on functional capability (age-related degradation margin over and above margin for GL 89-10 evaluated parameters) and safety significance. In implementing the interim MOV static diagnostic test program, licensees will rank MOVs within the scope of the JOG program according to their safety significance. The JOG program specifies that licensees need to justify their approach for risk ranking MOVs. In Topical Report NEDC-32264, "Application of Probabilistic Safety Assessment to Generic Letter 89-10 Implementation," the BWROG described a methodology to rank MOVs in GL 89-10 programs with respect to their relative importance to core-damage frequency and other considerations to be added by an expert panel. In an SE dated February 27, 1996, the NRC staff accepted the BWROG methodology for risk ranking MOVs in boiling-water reactor nuclear plants with certain conditions and limitations. In the NRC SE (dated October 30, 1997) on the JOG Program on MOV Periodic Verification, the NRC staff indicated its view that the BWROG methodology for MOV risk ranking is appropriate for use in response to GL 96-05. With respect to Westinghouse-designed pressurized water reactor nuclear plants, the WOG prepared Engineering Report V-EC-1658, "Risk Ranking Approach for Motor-Operated Valves in Response to Generic Letter 96-05." On April 14, 1998, the NRC staff issued an SE accepting with certain conditions and limitations the WOG approach for ranking MOVs based on their risk significance. Licensees not applicable to the BWROG or WOG methodologies need to justify their MOV risk-ranking approach individually.

The objectives of the JOG dynamic test program are to determine degradation trends in dynamic thrust and torque, and to use dynamic test results to adjust the test frequency and method specified in the interim program if warranted. The JOG dynamic testing program includes (1) identification of conditions and features that could potentially lead to MOV degradation; (2) definition and assignment of valves for dynamic testing; (3) testing valves three

times over a 5-year interval with at least a 1-year interval between valve-specific tests according to a standard test specification, (4) evaluation of results of each test; and (5) evaluation of collective test results.

In the last phase of its program, the JOG will evaluate the test results to validate the assumptions in the interim program to establish a long-term MOV periodic verification program to be implemented by licensees. A feedback mechanism will be established to ensure timely sharing of MOV test results among licensees and to prompt individual licensees to adjust their own MOV periodic verification program, as appropriate.

Following consideration of NRC staff comments, the BWROG submitted Licensing Topical Report NEDC-32719 (Revision 2) describing the JOG program on July 30, 1997. Similarly, the CEOG and the WOG submitted Topical Report MPR-1807 (Revision 2) describing the JOG program on August 6 and 12, 1997, respectively. On October 30, 1997, the NRC staff issued an SE accepting the JOG program with certain conditions and limitations as an acceptable industry-wide response to GL 96-05 for valve age-related degradation.

4.0 PALO VERDE GL 96-05 PROGRAM

On November 14, 1996, APS submitted a 60-day response to GL 96-05 notifying the NRC that it would review the MOV program at Palo Verde Nuclear Generating Station, Units 1, 2, and 3, to determine whether any changes were appropriate in light of the information in GL 96-05. On March 18, 1997, APS submitted a 180-day response to GL 96-05 stating that it had implemented a program to periodically verify safety-related MOVs at Palo Verde are capable of performing their safety functions. In this letter, the licensee provided a summary description of its MOV periodic verification program. In a letter dated July 19, 1998, the licensee updated its commitment to GL 96-05. On June 15, 1999, the licensee provided a response to a request for additional information regarding GL 96-05 forwarded by the NRC staff on March 17, 1999. The licensee clarified one aspect of its GL 96-05 program in a telephone conference with the NRC staff on October 6, 1999.

In its letter dated March 18, 1997, the licensee described its MOV periodic verification program, including existing preventive maintenance and static dynamic test programs and the addition of a supplemental dynamic testing program. The licensee stated that the dynamic diagnostic test program was scheduled to begin during the Unit 1 refueling outage scheduled for the spring of 1998. In its letter dated July 19, 1998, the licensee committed to continue its participation in the JOG MOV Periodic Verification Program as a member of the CEOG. The licensee reported that it would implement the program elements described in Topical Report NEDC-32719 (Revision 2) as amended by the NRC SE with an exception regarding interim MOV static test intervals. In its letter dated June 15, 1999, the licensee stated that interim MOV static diagnostic testing is conducted on a two-outage frequency (36 months) with a few justified exceptions, but none beyond 5 years. The licensee also stated that it was in the early stages of developing a formal MOV risk-ranking program.

5.0 NRC STAFF EVALUATION

The NRC staff has reviewed the information provided in the licensee's submittals describing the program to verify periodically the design-basis capability of safety-related MOVs at Palo Verde in response to GL 96-05. NRC Inspection Report 50-528, 529, 530/96-15 (IR 96-15) provided the results of an evaluation of the licensee's program to verify the design-basis capability of safety-related MOVs in response to GL 89-10. The staff closed the review of the licensee's GL 89-10 program in IR 96-15 based on verification of the design-basis capability of safety-related MOVs at Palo Verde. The staff's evaluation of the licensee's response to GL 96-05 is described below.

5.1 MOV Program Scope

In GL 96-05, the NRC staff indicated that all safety-related MOVs covered by the GL 89-10 program should be considered in the development of the MOV periodic verification program. The staff noted that the program should also consider safety-related MOVs that are assumed to be capable of returning to their safety position when placed in a position that prevents their safety system (or train) from performing its safety function; and the system (or train) is not declared inoperable when the MOVs are in their nonsafety position.

In reporting its evaluation of the MOV program at Palo Verde in IRs 95-23 and 96-15, the NRC staff did not identify any concerns with the scope of the licensee's MOV program in response to GL 89-10 and its supplements. In its letter dated November 14, 1996, the licensee did not take exception to the scope of GL 96-05. In its letter dated March 18, 1997, the licensee stated that its periodic verification program applied to MOVs in the scope of GL 89-10. The staff considers the licensee to have made adequate commitments regarding the scope of its MOV program.

5.2 MOV Assumptions and Methodologies

Licensees maintain their assumptions and methodologies used in the development of their MOV programs consistent with the plant configuration throughout the life of the plant (a concept commonly described as a "living program"). For example, the design basis of safety-related MOVs is maintained up to date, including consideration of any plant modifications or power uprate conditions.

In IR 96-15, the NRC staff reviewed the licensee's justification for the assumptions and methodologies used in the MOV program in response to GL 89-10 at Palo Verde. With certain long-term items discussed in the following section, the staff determined that the licensee had adequately justified the assumptions and methodologies used in its MOV program. The licensee's letter dated June 15, 1999, indicated ongoing activities, such as review of motor actuator output, to update its MOV program assumptions and methodologies. The staff considers the licensee to have adequate processes in place to maintain the assumptions and methodologies used in its MOV program, including the design basis of its safety-related MOVs.

5.3 GL 89-10 Long-Term Items

When evaluating the GL 89-10 program at Palo Verde, the NRC staff discussed in IR 96-15 several items of the licensee's MOV program to be addressed over the long term. In its letter dated June 15, 1999, the licensee reported on the status of those long-term GL 89-10 aspects.

In particular, the licensee reported that it had completed the reconciliation of its test data in validating the MOV design-basis performance parameters. The licensee also stated that it had revised its diagnostic test acceptance criteria to include a design-allowable unwedging thrust limit that accounts for test equipment and transducer error. The licensee reported that it applied information from the Electric Power Research Institute (EPRI) Topical Report TR-103229-V1 (dated November 1994), "EPRI MOV Performance Prediction Program - Gate Valve Model Report, Volume 1: Summary Through Appendix C," in support of its use of hydrostatic testing in lieu of dynamic flow tests to determine valve factors for certain valves. The licensee stated that EPRI Topical Report TR-103229-V1 showed close agreement between friction coefficients determined from hydrostatic tests and dynamic flow tests. More detailed and up-to-date information on the use of hydrostatic test data is provided in EPRI Topical Report TR-103244-R1 (dated October 1996), "EPRI MOV Performance Prediction Program-Implementation Guide," and the NRC SE dated March 15, 1996, on the EPRI Topical Report TR-103237-R1, "EPRI MOV Performance Prediction Program." For example, the thrust required to operate a valve can be much lower under hydrostatic conditions than under design-basis differential pressure conditions if the pressure decreases rapidly during the hydrostatic test. The maximum thrust required to open some valves under dynamic conditions has been found to occur later during the stroke than might be evidenced by a hydrostatic test. For the specific application at Palo Verde, the staff found in IR 96-15 that the valve factors derived by the licensee from the hydrostatic tests were reasonable in comparison to valve factors obtained from dynamic tests for similar valve types. Also in GL 89-10, the NRC staff identified pressure locking and thermal binding as potential performance concerns for safety-related MOVs. The NRC staff is reviewing licensee's actions in response to GL 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves," and will issue an SE at the completion of the review.

In IR 96-15, the NRC staff discussed qualitative and quantitative aspects of the licensee's program for trending MOV performance at Palo Verde. The NRC staff concluded that the licensee had adequately developed and implemented a comprehensive program for trending MOV performance. For example, the licensee trends static and dynamic diagnostic test data. In its letter dated July 19, 1998, the licensee stated that MOV as-found test results are evaluated on a case-by-case basis to ensure that the actuator has sufficient margin to remain operable until the next scheduled test. In its letter dated June 15, 1999, the licensee indicated that its MOV trending program monitors the results of static testing to identify long-term trends for incorporation into the MOV thrust and torque setpoints, and maintenance practices, as appropriate.

With the licensee's ongoing MOV activities and trending program, no outstanding issues regarding the licensee's GL 89-10 program remain at Palo Verde.

5.4 JOG Program on MOV Periodic Verification

In its letter dated July 19, 1998, the licensee updated its commitment to the JOG Program on MOV Periodic Verification as described in Topical Report NEDC-32719 (Revision 2) with one exception regarding the interim MOV static test interval. In the first phase of the JOG program, the JOG interim static test program establishes a test frequency ranging from every refueling outage to every six refueling outages (not to exceed 10 years) based on MOV safety significance (risk) and functional capability (margin). In its letter dated July 19, 1998, the licensee stated that the interim MOV static test program at Palo Verde provides for testing of

the MOVs in its GL 96-05 program every two refueling outages (with a few exceptions). In support of its alternate interim MOV test intervals, the licensee evaluates as-found MOV test results to ensure that the actuator has sufficient margin to remain operable until the next scheduled test. In its letter dated June 15, 1999, the licensee reported that its interim MOV static testing continues to be performed every two outages with a few justified and documented exceptions (which do not exceed 5 years). The licensee indicated that establishment of a proceduralized program for extending frequencies was awaiting completion of MOV risk ranking at Palo Verde and the availability of sufficient supporting trend data. The NRC staff finds that the interim MOV static test program proposed by the licensee at Palo Verde as an alternative to the JOG interim static test interval is acceptable, based on (1) the applicability of the alternate static MOV test intervals only to the first phase of the JOG program, (2) the limited extension of the minimal test interval from one to two outages, (3) the reduction of the maximum test interval at Palo Verde from the JOG program, (4) the ongoing evaluation of MOV test data by the licensee, and (5) the reported MOV performance history at Palo Verde.

In its letter dated June 15, 1999, the licensee stated that it is developing a risk ranking approach for the safety-related MOVs at Palo Verde based on improvements to its probabilistic risk assessment model and industry experience. The licensee's MOV risk-ranking approach will include use of an expert panel similar to the WOG methodology. The licensee also stated that the CEOG is considering cross-comparing the MOV risk rankings at CE plants. The staff might evaluate the licensee's ranking of MOVs by their risk significance during a future inspection where applied as part of the licensee's implementation of its commitment to the three phases of the JOG program.

In an SE dated October 30, 1997, the NRC staff accepted the JOG program as an industry-wide response to GL 96-05 with certain conditions and limitations. The JOG program consists of the following three phases: (1) the JOG interim static diagnostic test program; (2) the JOG 5-year dynamic test program; and (3) the JOG long-term periodic test program. The staff considers the licensee's commitment in response to GL 96-05 to include implementation of all three phases of the JOG program at Palo Verde, with the exception described above. The conditions and limitations discussed in the NRC SE dated October 30, 1997, apply to the JOG program at Palo Verde. The staff considers the commitments by the licensee to implement all three phases of the JOG program at Palo Verde to be an acceptable response to GL 96-05 for valve age-related degradation.

The JOG program is intended to address most gate, globe, and butterfly valves used in safety-related applications in the nuclear power plants of participating licensees. The JOG indicates that each licensee is responsible for addressing any MOVs outside the scope of applicability of the JOG program. The NRC staff recognizes that the JOG has selected a broad range of MOVs and conditions for the dynamic testing program, and that significant information will be obtained on the performance and potential degradation of safety-related MOVs during the interim static diagnostic test program and the JOG dynamic test program. As the test results are evaluated, the JOG might include or exclude additional MOVs with respect to the scope of its program. Although the test information from the MOVs in the JOG dynamic test program might not be adequate to establish a long-term periodic verification program for each MOV outside the scope of the JOG program, sufficient information should be obtained from the JOG dynamic test program to identify any immediate safety concern for potential valve age-related degradation during the interim period of the JOG program. Therefore, the NRC staff considers it acceptable for the licensee to apply its interim static diagnostic test program to

GL 96-05 MOVs that currently might be outside the scope of the JOG program with the feedback of information from the JOG dynamic test program to those MOVs. In the NRC SE dated October 30, 1997, the NRC staff specifies that licensees implementing the JOG program must determine any MOVs outside the scope of the JOG program (including service conditions) and justify a separate program for periodic verification of the design-basis capability (including static and dynamic operating requirements) of those MOVs.

5.5 Motor Actuator Output

The JOG program focuses on the potential age-related increase in the thrust or torque required to operate valves under their design-basis conditions. In the NRC SE dated October 30, 1997, on the JOG program, the NRC staff specifies that licensees are responsible for addressing the thrust or torque delivered by the MOV motor actuator and its potential degradation. Although the JOG does not plan to evaluate degradation of motor actuator output, significant information on the output of motor actuators will be obtained through the interim MOV static diagnostic test program and the JOG dynamic test program. Several parameters obtained during MOV static and dynamic diagnostic testing help identify motor actuator output degradation when opening and closing the valve including, as applicable, capability margin, thrust and torque at control switch trip, stem friction coefficient, load sensitive behavior, and motor current.

In IR 96-15, the NRC staff reported that the licensee monitors stem friction coefficient under static and dynamic conditions and compares these results to program assumptions. In its letter dated July 19, 1998, the licensee indicated that as-found test results are used to ensure adequate actuator output capability for safety-related MOVs at Palo Verde to perform their design-basis functions. In its letter dated June 15, 1999, the licensee stated that any MOVs found outside of their expected thrust and torque setpoint range are evaluated in accordance with station procedures to ensure that they will remain operable until the next scheduled static test. Further, the licensee noted that its MOV trending program will monitor the results of static testing to identify long-term trends, and that these trends will be incorporated into the MOV thrust and torque setpoints, and maintenance practices, as appropriate.

In Technical Update 98-01 and its Supplement 1, Limitorque Corporation provided updated guidance for predicting the torque output of its ac-powered motor actuators. In its letter dated June 15, 1999, the licensee reported that this guidance had been incorporated into the MOV setpoint calculations at Palo Verde. Further, the licensee indicated that its GL 89-10 MOVs have been evaluated for continued operability. Any MOV operability concerns that might be identified in the future will be processed in accordance with established regulatory requirements and plant-specific commitments.

In its letter dated July 17, 1998, forwarding Technical Update 98-01, Limitorque indicates that a future technical update will be issued to address the application of dc-powered MOVs. In IR 95-23, the NRC staff reported that the licensee was applying nominal rated starting torque and pullout efficiencies in evaluating the output capability of safety-related dc-powered MOVs at Palo Verde. In its June 15, 1999, letter, the licensee indicated that the guidance in Technical Update 98-01 had been applied to all GL 89-10 MOVs. In the telephone conference with the NRC staff on October 6, 1999, the licensee clarified that the recommendations of Technical Update 98-01 regarding use of pullout efficiency and a 0.9 application factor had been applied to the dc-powered MOVs in its GL 96-05 program. The industry is sponsoring a testing

program to support updated guidance for the application of dc-powered MOVs that should be made available in the near future.

The NRC staff considers the licensee to be establishing sufficient means to monitor MOV motor actuator output and its potential degradation.

6.0 CONCLUSION

The NRC staff finds that the licensee has established an acceptable program to verify periodically the design-basis capability of the safety-related MOVs at Palo Verde through its commitment to all three phases of the JOG Program on MOV Periodic Verification and the additional actions described in its submittals. Therefore, the staff concludes that the licensee is adequately addressing the actions requested in GL 96-05. As part of this review, the staff finds that the interim MOV static test program proposed by the licensee at Palo Verde as an alternative to the JOG interim static test interval is acceptable.

The NRC staff may conduct inspections at Palo Verde to verify the implementation of the MOV periodic verification program is in accordance with the licensee's commitments cited in this SE and in the NRC's SE dated October 30, 1997, on the JOG Program on MOV Periodic Verification. Such an inspection could include an evaluation of the licensee's ranking of MOVs by their risk significance where applied as part of the licensee's implementation of its commitment to the three phases of the JOG program.

Principal Contributors: T. Scarbrough, NRR
S. Tingen, NRR

Date: December 21, 1999

10/29/99

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

October 29, 1999

Mr. Gregg R. Overbeck
Senior Vice President, Nuclear
Arizona Public Service Company
P. O. Box 52034
Phoenix, AZ 85072-2034

SUBJECT: PALO VERDE NUCLEAR GENERATING STATION, UNITS 1, 2, AND 3 - WITHDRAWAL OF EXEMPTION REQUEST/ EXEMPTION FROM FEES/ REFUND REQUEST (TAC NOS. M94139, M94140, AND M94141)

Dear Mr. Overbeck:

I am responding to your December 17, 1998, letter informing the NRC that Arizona Public Service Company (APS) wishes to withdraw its November 27, 1995, request for exemption from 10 CFR 50.55a(f)(4)(i) and (ii) for Inservice Testing (IST) Frequency. APS is also requesting an exemption from 10 CFR Part 170 fees associated with the exemption request in accordance with the provisions of 10 CFR 170.11(b)(1), because APS participated in the Risk-Informed Inservice Testing (RI-IST) pilot program. Your letter further requests a refund of the fees paid to date for the review of the exemption request for the Palo Verde Nuclear Generating Station (Palo Verde), Units 1, 2, and 3 (TACs M94139, M94140, and M94141). As explained below, an exemption is granted in accordance with 170.11(b)(1) of 10 CFR Part 170.

The reasons you provide for withdrawing your November 27, 1995, exemption request include competing work priorities and limited resources. Although APS believes that risk-informed, performance-based regulation will result in overall increases in safety and reduction in costs, your assessment of RI-IST is that safety and cost benefits are marginal at best and that APS resources would best be spent pursuing other risk-informed applications, specifically, the pursuit of allowed outage time extensions and development of enhancements to your program for assessing on-line risk when equipment is removed from service to ensure compliance with the proposed changes to the Maintenance Rule.

To support your request for an exemption from the Part 170 fee requirement you provided the following:

> APS' involvement was a first-of-a-kind effort to support the NRC's planned generic regulatory improvement by providing input for an approach to RI-IST. APS believes information submitted to date by APS was used more for the development of an acceptable framework for this risk-informed application than for approval of the exemption, which we submitted. The development of RI-IST programs provides guidance to power reactor licensees and the NRC staff on an acceptable approach for utilizing risk information to support plant-specific changes to the current licensing basis for in-service testing programs. The program has the potential to optimize the use of industry and NRC resources, and to continue to assure adequate protection of the public health and safety.

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NRC did accept your November 27, 1995, RI-IST exemption request as a pilot plant review as indicated in our March 15, 1996, March 21, 1997, and June 9, 1997, letters requesting additional information to facilitate the staff efforts to develop a regulatory guide and a standard review plan (SRP) chapter that are sufficiently broad in scope to use in transitioning to more risk-informed regulatory decision making.

The staff agrees that the interactions with APS on its RI-IST program have provided valuable information that can be used to develop generic guidance in this area. The NRC has encouraged licensees to submit applications for the RI-IST pilot programs for demonstrating risk-informed methodologies to be used for piping segment and piping structural element selection in systems scheduled for inservice inspections. To provide the permanent approach to RI-IST, the staff intends to utilize the experience gained through the pilot applications in the proposed rulemaking process to modify 10 CFR 50.55a to explicitly endorse RI-IST methodology.

Based on your participation in a first-of-a-kind pilot plant review the exemption from the 10 CFR Part 170 fee requirements for the review of the exemption request, even though it has been withdrawn, is appropriate. This exemption is authorized by law and is granted in accordance with 10 CFR 170.11(b)(1).

The $72,874.00 paid to date for NRC's review effort billed will be refunded. You should receive the refund in approximately 30 days from the date of this letter.

Sincerely,
Original signed by Jesse Funches
Jesse L. Funches
Chief Financial Officer

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

cc: See next page

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NRC did accept your November 27, 1995, RI-IST exemption request as a pilot plant review as indicated in our March 15, 1996, March 21, 1997, and June 9, 1997, letters requesting additional information to facilitate the staff efforts to develop a regulatory guide and a standard review plan (SRP) chapter that are sufficiently broad in scope to use in transitioning to more risk-informed regulatory decision making.

The staff agrees that the interactions with APS on its RI-IST program have provided valuable information that can be used to develop generic guidance in this area. The NRC has encouraged licensees to submit applications for the RI-IST pilot programs for demonstrating risk-informed methodologies to be used for piping segment and piping structural element selection in systems scheduled for inservice inspections. To provide the permanent approach to RI-IST, the staff intends to utilize the experience gained through the pilot applications in the proposed rulemaking process to modify 10 CFR 50.55a to explicitly endorse RI-IST methodology.

Based on your participation in a first-of-a-kind pilot plant review the exemption from the 10 CFR Part 170 fee requirements for the review of the exemption request, even though it has been withdrawn, is appropriate. This exemption is authorized by law and is granted in accordance with 10 CFR 170.11(b)(1).

The $72,874.00 paid to date for NRC's review effort billed will be refunded. You should receive the refund in approximately 30 days from the date of this letter.

Sincerely,

Jesse L. Funches
Chief Financial Officer

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

cc: See next page

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