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

January 21, 2025

Adam Heflin
Executive Vice President/
Chief Nuclear Officer
Mail Station 7605
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P.O. Box 52034
Phoenix, AZ 85072-2034

SUBJECT: PALO VERDE NUCLEAR GENERATING STATION UNITS 1, 2, AND 3 REGULATORY AUDIT PLAN IN SUPPORT OF LICENSE AMENDMENT REQUEST TO REVISE TECHNICAL SPECIFICATIONS 3.5.1 AND 3.5.2 SAFETY INJECTION TANK PRESSURE BANDS AND TO USE GOTHIC CODE (EPID L-2024-LLA-0116)

Dear Adam Heflin:

By letter dated August 28, 2024 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML24241A278), Arizona Public Service Company (APS, the licensee) submitted a license amendment request (LAR) for Palo Verde Nuclear Generating Station, Units 1, 2, and 3 (Palo Verde).

The amendments would modify Palo Verde Technical Specification (TS) section 3.5.1, “Safety Injection Tanks (SITs) -Operating” and TS section 3.5.2, “Safety Injection Tanks (SITs) – Shutdown,” and their associated bases. Specifically, the proposed TS changes revise Palo Verde Surveillance Requirements 3.5.1.3 and 3.5.2.3 to increase the upper limit of their SIT pressure bands and to list their pressure requirements in units of pounds per square inch absolute as reflected in the Palo Verde safety analyses, with no instrument uncertainties included, instead of the SIT instrument units of pounds per square inch gauge with instrument uncertainties included.

In addition, the proposed changes would include use of the Generation of Thermal Hydraulic Information for Containments (GOTHIC) code as part of the methodology to perform calculations of the containment pressure and temperature response to various postulated pipe breaks.

The U.S. Nuclear Regulatory Commission (NRC) staff will conduct a virtual regulatory audit to support its review of the request. The audit will be conducted to increase the NRC staff’s understanding of the LAR and identify information that may require docketing to support the NRC staff’s regulatory findings.

In accordance with the enclosed audit plan, the audit team will review documentation and calculations that support the efficient review of the LAR. The team requests the licensee to establish a secure, online portal for remote access to this material. The team may also interview the licensee's subject matter experts using video and teleconferencing if necessary.

If you have any questions, please contact me at 301-415-3329 or by email at William.Orders@nrc.gov.

Sincerely,

/RA/

William Orders, Senior Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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

Enclosure:
Audit Plan

cc: Listserv

REGULATORY AUDIT

IN SUPPORT OF LICENSE AMENDMENT REQUEST TO REVISE TECHNICAL

SPECIFICATIONS 3.5.1 AND 3.5.2 AND TO USE GOTHIC CODE

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

DOCKET NOS: 50-528, 50-529, AND 50-530

EPID: L-2024-LLA-0116

1.0 BACKGROUND

By application dated August 28, 2024 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML24241A278), Arizona Public Service Company (APS, the licensee) submitted a license amendment request (LAR) for Palo Verde Nuclear Generating Station, Units 1, 2, and 3 (Palo Verde) proposing to modify Technical Specifications (TSs), section 3.5.1, “Safety Injection Tank (SITs) – Operating,” and TS section 3.5.2 , “Safety Injection Tanks (SITs) – Shutdown.”

The proposed TS changes would revise Palo Verde Surveillance Requirements (SRs) 3.5.1.3 and 3.5.2.3 to increase the upper limit of their SIT pressure bands, and to list their pressure requirements in units of pounds per square inch absolute (psia) as reflected in the Palo Verde safety analyses, with no instrument uncertainties included, instead of the SIT instrument units of pounds per square inch gauge (psig) with instrument uncertainties included.

In addition, the proposed changes would include use of Generation of Thermal Hydraulic Information for Containments (GOTHIC) code as part of the methodology to perform calculations of the containment pressure and temperature response to the postulated pipe breaks. The currently used code for containment response analysis is the Bechtel Containment Pressure and Temperature Transient Analysis (COPATTA) code.

2.0 REGULATORY AUDIT BASIS

A regulatory audit is a planned activity to examine and evaluate information that provides the technical basis for actions related to a license or regulation, allows the U.S. Nuclear Regulatory Commission (NRC) staff to gain insights to the licensee’s processes and procedures and clarify its understanding of the request, verify the information submitted, and identify additional information that the staff needs to make a licensing or regulatory decision. Information that the NRC staff relies upon to make a safety determination or decision must be submitted on the docket.

3.0 REGULATORY AUDIT SCOPE

The audit team will view the documentation and calculations that provide the technical support for the LAR.

Enclosure

The scope of the NRC staff's audit will focus on the following subjects:

- Confirm the NRC staff's understanding of the LAR and gain a better understanding of the detailed calculations, analyses, and bases on which it is based.
- Identify any information needed to enable the NRC staff's evaluation of whether the proposed changes challenge design-basis functions or adversely affect the capability or capacity of plant equipment to perform design-basis functions.

Information needed to support a licensing decision may already be docketed, which can be determined during the audit. Otherwise, it must be submitted under oath or affirmation. The licensee may supplement the LAR with this information. Alternatively, the licensee may respond to a formal request for additional information (RAI). If needed, RAIs will be developed and transmitted separately in accordance with the Office of Nuclear Reactor Regulation (NRR) Office Instruction LIC-115, "Processing Requests for Additional Information," Revision 1, dated August 5, 2021 (ML21141A238).

4.0 INFORMATION AND OTHER MATERIAL NECESSARY FOR THE REGULATORY AUDIT

The NRC staff may request information and interviews throughout the audit period. The NRC staff will identify the information to be audited (e.g., methodology, process information, and calculations) and the subjects of requested interviews and meetings.

The NRC staff requests that the licensee have the information referenced in this audit plan available and accessible for the NRC staff's review via a web portal within 2 weeks of the date of this audit plan. If additional or supplemental information is requested, it should be available and accessible within 1 week of the date of the NRC staff's notification to the licensee of the new requests. The NRC staff requests that the licensee notify the review team when an audit item is added to its portal by sending an email to the NRC licensing project manager.

The NRC staff acknowledges and will observe appropriate handling and protection of proprietary information made available for the audit. The NRC staff will not remove non-docketed information from the audit site or web portal.

5.0 AUDIT TEAM

The following table identifies the NRC audit team members and their respective organization:

Name	Email	Organization
Ahsan Sallman	Ahsan.Sallman@nrc.gov	NRR Senior Nuclear Engineer Nuclear Systems Performance Branch
Jo Ambrosini	Josephine.Ambrosini@nrc.gov	NRR Nuclear Engineer Nuclear Systems Performance Branch
William Orders	William.Orders@nrc.gov	Project Manager NRR Plant Licensing Branch 4

6.0 LOGISTICS

The audit will be conducted remotely via a secure, online portal, established by the licensee to present supporting documentation and calculations. In addition to the review of documents made available on the portal, the NRC staff may interview the licensee's subject matter experts, virtually (by telephone and videoconference) and, if necessary, in person. The audit will begin within 2 weeks of the date of this audit plan. The NRC staff will establish a schedule of audit meetings as appropriate on mutually agreeable dates and times.

At the meetings, the licensee is to discuss information needs and questions arising from the NRC's review of the application. The NRC project manager will inform the licensee of audit meeting dates when they are established.

7.0 SPECIAL REQUESTS

The following conditions associated with the online portal must be maintained while the NRC staff on the audit team have access to the online portal:

- The online portal will be password-protected. A separate password will be assigned to each member of the NRC staff taking part in the audit.

- The online portal will prevent the NRC participants from printing, saving, downloading, or collecting any information directly from the online portal.

- Conditions of use of the online portal will be displayed on the login screen and will require acknowledgment by each user.

The licensee should provide username and password information directly to the NRC staff on the audit team members listed above. All communications should be coordinated with the NRC project manager. The NRC project manager will inform the licensee via routine communications when the NRC staff no longer needs access to the portal.

No data accessed by the audit team members will be retained by the NRC following the conclusion of the audit.

8.0 DELIVERABLES

The NRC staff will develop any RAIs, if needed, in accordance with NRR Office Instruction LIC-115 and issue such RAIs independent of audit-related correspondence. At the end of the audit, the NRC staff will inform the licensee of the outcome. The NRC staff will issue an audit summary report within 90 days of the audit exit.

9.0 AUDIT ITEM LIST

AUDIT ITEM 1

Regulatory Background:

The following General Design Criteria (GDC) of Appendix A to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, General Design Criteria for Nuclear Power Plants," are applicable:

- GDC 16, "Containment design," as it relates to providing a "[r]eactor containment and associated systems. . . to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require."

- GDC 50, "Containment design basis," as it relates to designing "[t]he reactor containment structure, including access openings, penetrations, and the containment heat removal system so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident."

In LAR enclosure, section 3.1.3, "Use of GOTHIC Code," the licensee states, in part:

> The benchmark evaluation uses GOTHIC Version 8.4 [GOTHIC, "Generation of Thermal Hydraulic Information for Containments," Version 8.4(QA), Electric Power Research Institute, Palo Alto, CA, 2022]. The GOTHIC code is being continuously maintained and updated to include new features and/or correct problems. Therefore, although the analysis models and methods described in the benchmark evaluation were developed using GOTHIC Version 8.4, APS intends to use future versions of GOTHIC as they become available.

Documentation Request

Attachment 5, "Benchmark Evaluation for Use of the GOTHIC Code," of the LAR enclosure presents benchmarking of the proposed GOTHIC, Version 8.4 with the currently used COPATTA code. Since GOTHIC is not an NRC-approved code but accepted for containment response analysis, and the future possible changes in GOTHIC are not controlled by APS, provide documentation describing the method that will be used to justify the use of future GOTHIC versions that will not result in a numerical departure from the proposed containment response analysis.

AUDIT ITEM 2

Regulatory Basis

Same as in AUDIT ITEM 1

Documentation Request

In section 3.0,"Benchmarking of GOTHIC Results," of attachment 5 to the LAR enclosure, the licensee states, in part:

> When the results between GOTHIC and COPATTA differed, GOTHIC parameters were subsequently adjusted and additional runs were performed, to ensure the reason(s) for the differences were understood.

(a). Provide documentation describing which parameters in GOTHIC model were selected and adjusted in the benchmark process of GOTHIC with COPATTA code. Provide reasons and justification for selecting these parameters.

(b). For each parameter adjusted, provide documentation describing what reasons were understood from the differences in the benchmarking results.

AUDIT ITEM 3

Regulatory Basis

Same as in AUDIT ITEM 1

Documentation Request

In section 3.0,"Benchmarking of GOTHIC Results," of attachment 5 to the LAR enclosure, the licensee states, in part:

> The conversion of the COPATTA model of the DEDLS [double-ended discharge leg slot] break LOCA [loss-of-coolant accident] with maximum SI [safety injection] flow to a corresponding GOTHIC model is performed using GOTHIC Version 8.4 [GOTHIC, "Generation of Thermal Hydraulic Information for Containments," Version 8.4(QA), Electric Power Research Institute, Palo Alto, CA, 2022]. However, the analysis models and methods described in this benchmark evaluation are not intended to be restricted to a specific GOTHIC code version.

Provide documentation describing your justification that the analysis models and methods described in the benchmark evaluation are not restricted to the GOTHIC 8.4 version.

AUDIT ITEM 4

Regulatory Basis

GDC 38, "Containment heat removal." as it relates to the system that removes heat from the reactor containment whose system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any LOCA and maintain them at acceptably low levels.

Documentation Request

In LAR enclosure, attachment 6, "Technical Analysis of Changes to the SIT Pressure Bands, section 5.0, "Post-LOCA Containment Pressure and Temperature Response – Impact of

Changes to The SR 3.5.1.3 SIT Pressure Band," the licensee does not provide the impact of the changes on the containment sump temperature response and on the net positive suction head (NPSH) of the pumps that draw water from the sump during the post-LOCA recirculation phase. The licensee is requested to provide documentation of the following information:

(a). Post-LOCA sump temperature response based on the Westinghouse and Framatome fuels.

(b). Transient available NPSH, required NPSH, and minimum NPSH margin for the pumps that draw water from the sump during post-LOCA recirculation phase.

(c). Any containment accident pressure (CAP) above the vapor pressure at the transient sump temperature used in the calculation of transient available NPSH in item (b).

AUDIT ITEM 5

Regulatory Basis

The regulation in 10 CFR 50.46(b)(3), *"*Maximum hydrogen generation,*"* states:

> The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

Documentation Request

In LAR enclosure, attachment 6, section 3.1.1, "CE16STD Large Break LOCA," the licensee states, in part:

> The core wide cladding oxidation (CWO) cases are not limiting with respect to maximum SIT pressure.

Provide documentation explaining why the CWO cases are not limiting with respect to the maximum SIT pressure.

AUDIT ITEM 6

Regulatory Basis:

The regulations in 10 CFR 50.46(b) states, in part:

> (1) *Peak cladding temperature*. The calculated maximum fuel element cladding temperature shall not exceed 2200°F [degrees Fahrenheit].
>
> (2) *Maximum cladding oxidation*. The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.
>
> (3) *Maximum hydrogen generation*. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall

> not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

Documentation Request:

In LAR enclosure, attachment 6, sections 3.1.1, “CE16STD Large Break LOCA,” and 3.1.2, “CE16NGF Large Break LOCA,” the licensee states CE16STD and CE16NGF large break LOCA analysis of record (AOR) limiting peak cladding temperature (PCT) cases has been reevaluated for an increase in its modeled maximum SIT pressure to 675 psia. As stated in these sections, the impact on the PCT for both cases is 2°F resulting in a PCT of 2108°F for CE16STD and PCT of 2132°F for CE16NGF. The licensee is requested to provide documentation of the following:

(a). The large break LOCA re-analysis methodology used to re-evaluate the impact on PCT, peak local oxidation (PLO), and CWO.

(b). Confirm the methodology in response to (a) is the same as in the AOR. Provide justification if a different methodology is used.

(c). Confirm the inputs and assumptions in the re-analysis using the methodology in response to (a) were kept the same as in the AOR with the exception of the change in the SIT pressure. In case the conservatism in any input or assumption was reduced, provide justification.

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DISTRIBUTION
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ADAMS Accession No.: ML25016A345 ***concurrence via email**

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DATE	1/15/2025	1/21/2025	1/ 21/2025

OFFICE	NRR/DORL/LPL4/BC*	NRR/DORL/LPL4/PM*
NAME	TNakanishi	WOrders
DATE	1/21/2025	1/21/2025

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