



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

March 6, 2025

Delson C. Erb
Vice President, OPS Support
Tennessee Valley Authority
1101 Market Street, LP 4A-C
Chattanooga, TN 37402-2801

SUBJECT: SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENT NOS. 370 AND 364 REGARDING A REVISION TO TECHNICAL SPECIFICATIONS SURVEILLANCE REQUIREMENT 3.4.14.1, "REACTOR COOLANT SYSTEM (RCS) PRESSURE ISOLATION VALVE (PIV) LEAKAGE,"(EPID L-2024-LLA-0156)

Dear Delson Erb:

The U.S. Nuclear Regulatory Commission (NRC, the Commission) has issued the enclosed Amendment Nos. 370 and 364 to Renewed Facility Operating License Nos. DPR-77 and DPR-79, for the Sequoyah Nuclear Plant, Units 1 and 2 (Sequoyah), respectively. The amendments consist of changes to the Sequoyah technical specifications (TSs) in response to the Tennessee Valley Authority application dated November 26, 2024, as supplemented by letters dated February 10, 2025, and February 20, 2025.

The amendments revise Sequoyah TS 3.4.14, "Reactor Coolant System (RCS) Pressure Isolation Valve (PIV) Leakage," Surveillance Requirement 3.4.14.1.

A copy of the Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's monthly *Federal Register* notice.

Sincerely,

/RA - K. Green for/

Perry H. Buckberg, Senior Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-327 and 50-328

Enclosures:

1. Amendment No. 370 to DPR-77
2. Amendment No. 364 to DPR-79
3. Safety Evaluation



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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-327

SEQUOYAH NUCLEAR PLANT, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 370
Renewed License No. DPR-77

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment by the Tennessee Valley Authority (the licensee) dated November 26, 2025, as supplemented by letters dated February 10, 2025, and February 20, 2025, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in Title 10 of the *Code of Federal Regulations* (10 CFR) Chapter I;
 - B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;
 - C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;
 - D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and
 - E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Renewed Facility Operating License No. DPR-77 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 370 are hereby incorporated into the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

David Wrona, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Date of Issuance: March 6, 2025

ATTACHMENT TO LICENSE AMENDMENT NO. 370

SEQUOYAH NUCLEAR PLANT, UNIT 1

RENEWED FACILITY OPERATING LICENSE NO. DPR-77

DOCKET NO. 50-327

Replace page 3 of Renewed Facility Operating License No. DPR 77 with the attached page 3. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

Replace the following pages of the Appendix A, Technical Specifications, with the attached pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

<u>Remove</u>	<u>Insert</u>
3.4.14-2	3.4.14-2
3.4.14-3	-
3.4.14-4	-

- (3) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;
- (4) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and
- (5) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the Sequoyah and Watts Bar Unit 1 Nuclear Plants.

C. This renewed license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The Tennessee Valley Authority is authorized to operate the facility at reactor core power levels not in excess of 3455 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 370 are hereby incorporated into the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) Initial Test Program

The Tennessee Valley Authority shall conduct the post-fuel-loading initial test program (set forth in Section 14 of Tennessee Valley Authority's Final Safety Analysis Report, as amended), without making any major modifications of this program unless modifications have been identified and have received prior NRC approval. Major modifications are defined as:

- a. Elimination of any test identified in Section 14 of TVA's Final Safety Analysis Report as amended as being essential;
- b. Modification of test objectives, methods, or acceptance criteria for any test identified in Section 14 of TVA's Final Safety Analysis Report as amended as being essential;

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time for Condition A not met.	B.1 Be in MODE 3.	6 hours
	<u>AND</u> B.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.14.1 -----NOTE----- Not required to be performed in MODES 3 and 4. ----- Verify leakage from each RCS PIV is equivalent to ≤ 0.5 gpm per nominal inch of valve size up to a maximum of 5 gpm at an RCS pressure ≥ 2215 psig and ≤ 2255 psig.	In accordance with the Inservice Testing Program



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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-328

SEQUOYAH NUCLEAR PLANT, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 364
Renewed License No. DPR-79

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment by the Tennessee Valley Authority (the licensee) dated November 26, 2025, as supplemented by letters dated February 10, 2025, and February 20, 2025, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in Title 10 of the *Code of Federal Regulations* (10 CFR) Chapter I;
 - B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;
 - C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;
 - D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and
 - E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Renewed Facility Operating License No. DPR-79 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 364 are hereby incorporated into the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

David Wrona, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Date of Issuance: March 6, 2025

ATTACHMENT TO LICENSE AMENDMENT NO. 364

SEQUOYAH NUCLEAR PLANT, UNIT 2

RENEWED FACILITY OPERATING LICENSE NO. DPR-79

DOCKET NO. 50-328

Replace page 3 of Renewed Facility Operating License No. DPR 79 with the attached page 3. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

Replace the following pages of the Appendix A, Technical Specifications, with the attached pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

<u>Remove</u>	<u>Insert</u>
3.4.14-2	3.4.14-2
3.4.14-3	-
3.4.14-4	-

- (3) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;
- (4) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and
- (5) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the Sequoyah and Watts Bar Unit 1 Nuclear Plants.

C. This renewed license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The Tennessee Valley Authority is authorized to operate the facility at reactor core power levels not in excess of 3455 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 364 are hereby incorporated into the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) Initial Test Program

The Tennessee Valley Authority shall conduct the post-fuel-loading initial test program (set forth in Section 14 of Tennessee Valley Authority's Final Safety Analysis Report, as amended), without making any major modifications of this program unless modifications have been identified and have received prior NRC approval. Major modifications are defined as:

- a. Elimination of any test identified in Section 14 of TVA's Final Safety Analysis Report as amended as being essential;

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time for Condition A not met.	B.1 Be in MODE 3.	6 hours
	<u>AND</u> B.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.14.1 -----NOTE----- Not required to be performed in MODES 3 and 4. ----- Verify leakage from each RCS PIV is equivalent to ≤ 0.5 gpm per nominal inch of valve size up to a maximum of 5 gpm at an RCS pressure ≥ 2215 psig and ≤ 2255 psig.	In accordance with the Inservice Testing Program



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
AMENDMENT NO. 370 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-77
AMENDMENT NO. 364 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-79
SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2
TENNESSEE VALLEY AUTHORITY
DOCKET NOS. 50-327 AND 50-328

1.0 INTRODUCTION

By application dated November 26, 2024 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML24331A178), as supplemented by letters dated February 10, 2025 (ML25041A301), and February 20, 2025 (ML25051A357), Tennessee Valley Authority (TVA, the licensee) submitted a license amendment request (LAR) for Renewed Facility Operating License Numbers DPR-77 and DPR-79 for Sequoyah Nuclear Plant (Sequoyah or SQN), Units 1 and 2.

The proposed amendments would revise the Technical Specification (TS) 3.4.14, "Reactor Coolant System (RCS) Pressure Isolation Valve (PIV) Leakage," Surveillance Requirement (SR) 3.4.14.1, to only reference the Inservice Testing Program (IST) Program for the Frequency.

The supplements dated February 10, 2025, and February 20, 2025, provided additional information in response to a NRC staff request for additional information (RAI). These supplements did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on December 23, 2024 (89 FR 104572).

2.0 REGULATORY EVALUATION

2.1 System Description

In its LAR, the licensee described the system as:

The RCS PIVs are any two normally closed valves in series which separate the high pressure RCS from an attached low pressure system, such as the residual heat removal (RHR) system, the safety injection (SI) system and the accumulators. During their lives, these valves can produce varying amounts of reactor coolant leakage through either normal operational wear or mechanical

deterioration. The RCS PIV Leakage LCO [limiting conditions for operation] allows RCS high pressure operation when leakage through these valves exists in amounts that do not compromise safety.

2.2 Description of Changes

The licensee proposed to remove content related to performing the surveillance at a frequency in accordance with the Surveillance Frequency Control Program to reflect reliance on the frequency established by the IST Program. Additionally, the licensee proposed to remove the requirement that the surveillance be performed prior to entering MODE 2 when the unit has been in MODE 5 for 7 days or more, if leakage testing has not been performed in the previous 9 months. Further, the licensee proposed removal of the requirement to perform the surveillance within 24 hours following valve actuation due to automatic or manual action or flow through the valve. Lastly, the licensee proposed removal of SR notes that exempt performance already covered by the SR applicability or are no longer required due to elimination of the 24-hour requirement. The proposed changes do not add or remove any RCS PIVs from the TS or American Society of Mechanical Engineers (ASME) OM Code requirements nor do they alter the SR acceptance criteria.

The Sequoyah TS SR 3.4.14.1 will be revised to reflect a frequency of (deleted text noted by strikeout):

Frequency

In accordance with the Inservice Testing Program ~~and in accordance with the Surveillance Frequency Control Program~~

~~AND~~

~~Prior to entering MODE 2 whenever the unit has been in MODE 5 for 7 days or more, if leakage testing has not been performed in the previous 9 months~~

~~AND~~

~~Within 24 hours following valve actuation due to automatic or manual action or flow through the valve~~

2.3 Applicable Regulatory Requirements and Guidance Documents

In Title 10 of the *Code of Federal Regulations* (10 CFR) 50.36, "Technical specifications," the U.S. Nuclear Regulatory Commission (NRC) established its regulatory requirements related to the content of TS. Pursuant to 10 CFR 50.36, TS are required to include items in the following five specific categories related to station operation: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation; (3) SRs; (4) design features; and (5) administrative controls.

Section 50.36(c)(3) of 10 CFR states, "[s]urveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met."

The regulation 10 CFR 50.54(jj) states that "[s]tructures, systems, and components [SSCs] subject to the codes and standards of 10 CFR 50.55a must be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed."

Option B of 10 CFR Part 50, Appendix J, provides a performance-based option for establishing leak test frequencies for containment isolation valves. Although Appendix J does not apply specifically to all PIVs, it provides requirements for establishing a performance-based test program for valve leakage testing.

10 CFR 50.55a

The NRC regulations in 50.55a(f)(4), "Inservice testing standards requirement for operating plants," state, in part, that throughout the service life of a boiling or pressurized water-cooled nuclear power facility, pumps and valves that are within the scope of the ASME *Operation and Maintenance of Nuclear Power Plants*, Division 1, OM Code: Section IST (OM Code) must meet the IST requirements (except design and access provisions) set forth in the ASME OM Code that become effective subsequent to editions and addenda specified in 10 CFR 50.55a(f)(2) and (3) and that are incorporated by reference in 10 CFR 50.55a(a)(1)(iv), to the extent practical within the limitations of design, geometry, and materials of construction of the components.

10 CFR 50.55a(f)(5)(ii), "IST program update: Conflicting IST Code requirements with technical specifications," which states in part that if a revised inservice test program for a facility conflict with the TS for the facility, the licensee must apply for an amendment of the TS to conform the TS to the revised program.

10 CFR 50.55a(z), "Alternatives to codes and standards requirements," states the following:

Alternatives to the requirements of paragraphs (b) through (h) of this section or portions thereof may be used when authorized by the Director, Office of Nuclear Reactor Regulation. A proposed alternative must be submitted and authorized prior to implementation. The applicant or licensee must demonstrate a(n):

- (1) Acceptable level of quality and safety. The proposed alternative would provide an acceptable level of quality and safety; or
- (2) Hardship without a compensating increase in quality and safety. Compliance with the specified requirements of this section would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

10 CFR Part 50, Appendix A, General Design Criteria (GDC)

In the LAR, the licensee stated:

SQN Units 1 and 2 were designed to meet the intent of the "Proposed General Design Criteria (GDC) for Nuclear Power Plant Construction Permits published in July 1967. The SQN construction permit was issued in May 1970. The Updated Final Safety Analysis Report (UFSAR), however, addresses the Nuclear Regulatory Commission (NRC) GDC published as Appendix A to 10 CFR 50 in July 1971.

The applicable GDCs are:

GDC 1, "Quality standards and records," states:

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. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A Quality Assurance Program shall be established and implemented in order to provide adequate assurance that these structures, systems, and components will satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection and testing of structures, systems, and components important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.

GDC 14, "Reactor coolant pressure boundary," states:

The reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

GDC 30, "Quality of reactor coolant pressure boundary," states:

Components which are part of the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.

GDC 32, "Inspection of reactor coolant pressure boundary," states:

Components which are part of the reactor coolant pressure boundary shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leaktight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel.

GDC 54, "Piping systems penetrating containment," states:

Piping systems penetrating primary reactor containment shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems shall be designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.

GDC 55, "Reactor coolant pressure boundary penetrating containment," states:

Each line that is part of the reactor coolant pressure boundary and that penetrates primary reactor containment shall be provided with containment

isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

- (1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or
- (2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or
- (3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or
- (4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

Isolation valves outside containment shall be located as close to containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

Other appropriate requirements to minimize the probability or consequences of an accidental rupture of these lines or of lines connected to them shall be provided as necessary to assure adequate safety. Determination of the appropriateness of these requirements, such as higher quality in design, fabrication, and testing, additional provisions for inservice inspection, protection against more severe natural phenomena, and additional isolation valves and containment, shall include consideration of the population density, use characteristics, and physical characteristics of the site environs.

3.0 TECHNICAL EVALUATION

3.1 Technical Discussion

The proposed changes, which replace RCS PIV testing Frequency with a reference to the IST Program, are summarized as follows:

- Elimination of the requirement to periodically perform the SR in accordance with the SFCP in addition to the frequency specified in the ASME OM Code.
- Elimination of a Frequency of 9 months prior to entering MODE 2 if the unit has been in MODE 5 for 7 days or more.
- Elimination of a Frequency of 24 hours following valve actuation due to automatic or manual action or flow through the valve, as well as an SR Note associated with the Frequency.
- Elimination of SR Notes that exempt performance for valves located in the RHR flow path when in the shutdown cooling mode, exempt repeat testing of RCS PIVs actuated if a repetitive testing loop cannot be avoided, and exempt testing of specific RCS PIVs in the RHR system.

The staff's evaluation for each of these changes is below.

3.1.1 Elimination of the Frequency in accordance with the Surveillance Frequency Control Program

In support of eliminating the frequency requirement in accordance with the SFCP, TVA stated in its LAR:

The proposed change would remove a surveillance frequency requirement for performing SR 3.4.14.1. The current SR relies on the IST Program and the SFCP to establish the associated frequency for these surveillances. The ASME OM Code establishes a testing frequency for PIV testing of every 2 years. Under the SFCP, changes to the performance frequency requires qualitative considerations to be addressed, such as test intervals specified in applicable industry codes and standards. Because an RCS PIV testing frequency is specified in the ASME OM Code, the frequency cannot be extended beyond every 2 years under the SFCP without prior NRC approval. As a result, the appropriate reference for the SR 3.4.14.1 performance frequency is the IST Program and the ASME OM Code.

Further, the NRC regulations give precedence to the ASME Code when there are conflicts between the TS and the Code. It is stated in 10 CFR 50.55a(f)(5)(ii), "IST program update: Conflicting IST Code requirements with technical specifications," that "If a revised inservice test program for a facility conflicts with the technical specifications for the facility, the licensee must apply to the Commission for amendment of the technical specifications to conform the technical specifications to the revised program."

The leakage rate testing performance history of each PIV at SQN Units 1 and 2 is shown in Attachment 4 [of submittal dated November 26, 2024]. The PIV testing performance history shows that the valves have consistently met the associated leakage criteria when tested. This provides confidence that performing SR 3.4.14.1 at frequencies in accordance with the IST program and eliminating the frequency in accordance with the SFCP from the TS will not degrade the ability of the PIVs to perform their safety function.

In the RAI response dated February 10, 2025, regarding the performance history of PIVs at SQN Units 1 and Units 2, TVA stated the following:

The IST Program for SQN reflects a frequency of testing for the accumulator discharge PIVs (Unit 1 valves 63-561 and 63-562 and Unit 2 valve 63-563) of once every two years; however, SQN performs leakage testing of these valves every 18 months during refueling outages. Although PIV surveillance leak rate data trends can suggest an exceedance of the acceptance criteria at a future performance, the leak rate trend alone is not an indicator of a future result. For example, leakage test parameters can vary between tests such as RCS test pressure and temperature. Therefore, the appearance of a leak rate data trend does not affect the proposed changes to SR 3.4.14.1.

SQN does trend PIV leak performance over time, and contingency inspection and repair plans are developed in the event the acceptance criteria is exceeded upon performance of the next surveillance. Additionally, the SQN corrective action program would document and resolve any PIV seat leakage exceeding the

SR 3.4.14.1 acceptance criteria and adjust the preventative maintenance activities as necessary.

Event driven PIV leak rate testing is infrequently performed under the current SQN requirements for SR 3.4.14.1. As a result, there are very few data points in the PIV leak test history for SQN from event driven performances. Therefore, the availability of PIV leak rate data points would be essentially the same with the implementation of the proposed TS changes to SR 3.4.14.1.

The NRC staff evaluated the information provided in the LAR, as supplemented. The NRC staff finds the proposed changes to the SQN SR 3.4.14.1 frequency have no detrimental effect on PIV leakage testing in the SQN IST Program, as the controlling requirement continues to be the ASME OM Code as required by 10 CFR 50.55a(f).

In Enforcement Guidance Memorandum (EGM) 12-001, "Dispositioning Noncompliance with Administrative Controls Technical Specifications Programmatic Requirements that Extend Test Frequencies and Allow Performance of Missed Tests" (ML11258A243), the NRC staff determined that if a licensee finds that the requirements of TS conflict with the requirements of 10 CFR 50.55a, then the licensee must amend their TS to comply with 10 CFR 50.55a. The position related to conflicting requirements between TS and 10 CFR 50.55a is governed by 10 CFR 50.55a(f)(5)(ii), "IST program update: Conflicting IST Code requirements with technical specifications," which requires, in part, that if a revised IST Program conflicts with the TS for the facility, the licensee must apply for an amendment of the TS to conform the TS to the revised IST program. The proposed change is therefore consistent with NRC staff positions and regulations with respect to conflicting requirements.

Therefore, based on the above, the NRC staff finds that the licensee has shown that elimination of the testing frequency in accordance with the SFCP, while still maintaining reliance on the IST Program testing frequency, meets the requirements of 10 CFR 50.36(c)(3), because the quality of systems and components are maintained, facility operation will be within safety limits, and the limiting conditions for operation will be met.

3.1.2 Elimination of 9 Month Testing Frequency

In support of eliminating the 9-month testing frequency, the licensee stated in its LAR:

The SR 3.4.14.1 Frequency requires the SR to be performed, "Prior to entering MODE 2 whenever the unit has been in MODE 5 for 7 days or more, if leakage testing has not been performed in the previous 9 months." This has the effect of requiring performance of the SR every 9 months, but only if there is an outage of sufficient length to perform the test.

Elimination of the more frequent performance of the SR is acceptable because the ASME OM Code, which provides all other aspects of the testing, specifies a longer testing frequency, and the more frequent testing is not warranted given the SQN Units 1 and 2 PIV surveillance performance history provided in Attachment 4. Therefore, leakage testing of the RCS PIVs at the frequency established by the IST Program is satisfactory in demonstrating pressure isolation functional capability and operational readiness.

The NRC staff evaluated the information provided in the LAR. In removing the 9-month frequency, the TS will instead require leakage testing at a frequency in accordance with the IST Program, which reinforces the established requirements of the ASME OM Code that are incorporated by reference in 10 CFR 50.55a(a)(1)(iv). The proposed change to each unit's TS SR Frequency would ensure ASME OM Code testing requirements for the RCS PIVs will be retained in the IST Program, and frequency testing requirements in the TS SR will be removed. The NRC staff notes that this LAR only focuses on the testing interval for leak-tight capability of the RCS PIVs.

Therefore, based on the above, the NRC staff finds that the licensee has shown that the elimination of the 9-month testing frequency, while still maintaining reliance on the IST Program testing frequency, meets the requirements of 10 CFR 50.36(c)(3), because the quality of systems and components are maintained, facility operation will be within safety limits, and the limiting condition for operation will be met.

3.1.3 Elimination of Testing within 24 Hours of Valve Actuation

In support of eliminating testing within 24 hours of valve actuation, the licensee stated in its LAR:

The SR 3.4.14.1 Frequency requires the SR to be performed within 24 hours following valve actuation due to automatic or manual action or flow through the valve. The purpose of the performance is to verify that the PIVs are closed or seated after being actuated. This test is redundant to other ASME OM Code testing (e.g., exercise testing per ASME OM Code Subsection ISTC-3520 and position indication verification per ASME OM Code Subsection ISTC-3700) and is unnecessary.

In addition, there are other readily available indications that a PIV has failed to close or seat, such as low-pressure system level, temperature, or pressure indications or the lifting of relief valves. Performing the PIV leakage testing for this purpose is unnecessary and will result in higher occupational dose exposure.

The NRC staff evaluated the information provided in the LAR. As stated above by the licensee, the TS requirement to perform leakage testing within 24 hours following valve actuation is the same as the licensee's implementation of the testing requirements specified in other ASME OM Code requirements as incorporated by reference in 10 CFR 50.55a. In addition, the licensee indicates that there are other readily available indications to operators that a PIV has failed to close or seat, such as low-pressure system level, temperature, or pressure indications or the lifting of relief valves.

Therefore, based on the above, the NRC staff finds that the licensee has shown that the elimination of the 24 hour testing frequency, while still maintaining reliance on the IST Program testing frequency, meets the requirements of 10 CFR 50.36(c)(3), because the quality of systems and components are maintained, facility operation will be within safety limits, and the limiting condition for operation will be met.

3.1.4 Elimination of SR Note Exemptions

In support of deleting the SR Note Exemptions, the licensee stated in its LAR:

SQN Units 1 and 2 SR 3.4.14.1, Note 2 states, "Not required to be performed on the RCS PIVs located in the RHR flow path when in the shutdown cooling mode of operation." The Applicability of TS 3.4.14 states:

APPLICABILITY: MODES 1, 2, and 3,

 MODE 4, except valves in the residual heat
 removal (RHR) flow path when in, or during the
 transition to or from, the RHR mode of operation

The Applicability already exempts the RCS PIVs associated with the RHR decay heat removal flow path from testing in MODE 4. Further, SR 3.4.14.1, Note 1, states that the testing is not required to be performed in MODES 3 and 4. The RHR decay heat removal system is only used in MODE 4 of the Applicability; therefore, Note 2 is not needed and is eliminated.

SQN Units 1 and 2 SR 3.4.14.1, Note 3 states, "RCS PIVs actuated during performance of this Surveillance are not required to be tested more than once if a repetitive testing loop cannot be avoided." This Note is discussed in the SQN Units 1 and 2 Bases for the 24-hour testing requirement as, "In addition, testing must be performed once after the valve has been opened by flow or exercised to ensure tight reseating. PIVs disturbed in the performance of this Surveillance should also be tested unless documentation shows that an infinite testing loop cannot practically be avoided." Therefore, this Note is no longer needed with the elimination of the 24-hour testing Frequency.

SQN Units 1 and 2 SR 3.4.14.1, Note 4 states, "Not required to be performed for RCS PIVs FCV-74-1 and FCV-74-2 following manual or automatic actuation or flow through the valves." This Note is specific to two flow control valves within the RHR system. With the elimination of the 24-hour testing Frequency requirement this Note is no longer needed.

TVA confirms that the RCS PIVs required to be tested by the SQN Units 1 and 2 TS are included in the SQN IST Program. Additionally, SQN confirms that the RCS PIV leakage testing Frequencies proposed to be removed from the SQN Units 1 and 2 TS are not credited for satisfying any other requirements described in the Updated Final Safety Analysis Report or any commitments for reasons other than being a TS requirement.

The NRC staff evaluated the information provided in the LAR. As discussed above, SR 3.4.14.1 Note 1, states that the testing is not required to be performed in MODES 3 and 4. Since the RHR shutdown cooling mode is only used in MODE 4, Note 2 is not needed. Therefore, the NRC staff finds that deleting SR 3.4.14.1 Note 2 is acceptable

As discussed above, SR 3.4.14.1 Notes 3 and 4 are no longer needed with the elimination of the 24-hour testing frequency. The NRC staff finds that deleting SR 3.4.14.1 Note 3 and Note 4 is acceptable.

Based on the above, the NRC staff finds that the licensee has shown that with the elimination of the Notes, SR 3.4.14.1 continues to meet the requirements of 10 CFR 50.36(c)(3) because the quality of systems and components are maintained, facility operation will be within safety limits, and the limiting condition for operation will be met.

3.1.5 Plant Risk of Eliminating the Fixed and Event Driven Testing Frequencies

As described in the LAR, the licensee proposed to eliminate fixed and event driven PIV test frequencies from the RCS PIV Leakage SR 3.4.14.1. The proposed change retains an RCS PIV testing frequency in accordance with the IST Program. In addition, in the LAR, the licensee confirmed that the RCS PIV leakage testing frequencies proposed to be removed from the SQN, Units 1 and 2 TS are not credited for satisfying any other requirements described in the Updated Final Safety Analysis Report or any commitments for reasons other than being a TS requirement. However, the SQN TS Bases B 3.4.14, "RCS PIV Leakage," References section, contained two references, reference 4: WASH-1400 (NUREG-75/014), Appendix V, October 1975; and reference 5: NUREG-0677, May 1980. The SQN TS Bases state that WASH-1400 identified potential intersystem loss-of-coolant accidents (ISLOCAs) as a significant contributor to plant risk (core melt). NUREG-0677 indicates that testing needs to be performed on an event-based frequency and/or more frequent time-based frequency to assure plant risk due to ISLOCA is acceptable. In a RAI dated January 23, 2025 (ML25024A017), the NRC staff requested the licensee to review these references and explain why the proposed frequencies (which eliminate fixed and event driven RCS PIV test frequencies from the TS) are adequate to assure TS are derived from the analyses and evaluation included in the bases to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met.

In the RAI response dated February 10, 2025, and supplemented on February 20, 2025, the licensee responded to the NRC staff's question. The licensee's response stated in part:

The state-of-the-art of risk assessment has progressed substantially since the "Reactor Safety Study" in 1975 and NUREG-0677 in 1980. TVA has performed a detailed analysis and developed logic models for ISLOCA events as part of the SQN Full Power Internal Events Probabilistic Risk Assessment (FPIE PRA). This analysis uses NSAC-154 (Reference 3 of this RAI response,) "ISLOCA Evaluation Guidelines," and WCAP-17154-P (Reference 4 of this RAI response,) "ISLOCA Risk Model," as guidance for developing the ISLOCA event trees, success criteria, failure probabilities, and fault trees. NSAC-154 is the industry recognized standard for ISLOCA analysis and it provides detailed instructions on the methodology and documentation of this analysis. Following NSAC-154 ensures that the ISLOCA analysis is done consistent with industry practice. The objective of WCAP-17154-P is to develop guidance on modeling the risk contribution from ISLOCAs during power operation. WCAP-17154-P addresses the full scope of ISLOCA modeling, including the development of its initiating event frequency, assessing its recovery, estimating the rupture probability of the low pressure interfacing system, and mitigation of its consequences.

The SQN FPIE PRA Model (including internal flooding) was peer reviewed against the requirements of the American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) PRA standard (Reference 5 of this RAI response) and any Clarifications and Qualifications provided in the NRC

endorsement of the Standard contained in Revision 2 to Regulatory Guide (RG) 1.200 (Reference 6 of this RAI response). A peer review and independent assessment was performed for the SQN Units 1 and 2 Internal Events PRA and are described in Reference 7 of this RAI response.

No ISLOCA Core Damage Frequency (CDF) or Large Early Release Frequency (LERF) cutsets (above the truncation limit of $1E-12$ /per reactor year for CDF and $1E-13$ /per reactor year for LERF) are included in the results of the FPIE PRA for SQN Unit 1 or Unit 2. Based on the results of the SQN PRA, the risk of an ISLOCA is considered a negligible contributor to the risk of core melt and large early release. Within the ISLOCA analysis performed for the SQN FPIE, the frequency of the RCS PIV testing credited was once every 18 months, which is equivalent to the IST frequency. As a result, the yearly periodic leakage testing of the PIVs recommended in NUREG-0677 is no longer required to reduce the probability of an intersystem LOCA to an acceptable level. The references to WASH-1400 and NUREG-0677 are removed from the bases to the SQN Technical Specifications as provided in Enclosure 2 to Reference 8 of this RAI response.

TVA has reviewed the SQN IST Program frequencies as related to the license amendment request to verify conformance with licensing basis requirements with the changes being proposed.

The NRC staff reviewed the information provided in the RAI response. The licensee stated that an ISLOCA model is included in the FPIE PRA, and that the frequency of RPS PIV testing is once every 18 months, which is equivalent to the IST frequency. The licensee further stated that every ISLOCA cutset fell below the truncation limits of the PRA ($1E-12$ for CDF and $1E-13$ for LERF), and therefore are not included in the PRA results. Based on these results, the licensee concluded that the risk of an ISLOCA is a negligible contributor to the risk of core damage and large early release.

The NRC staff has previously reviewed the technical adequacy of the Sequoyah PRA models for "Provide Risk-Informed Extended Completion Times – RITSTF Initiative 4b" (TSTF-505), (ML22210A118).

Based on the above, the NRC staff concludes that the licensee's response to the RAI is acceptable because the licensee demonstrated that the risk of an ISLOCA is a negligible contributor to the risk of core melt and large early release with a testing frequency of 18 months.

The NRC staff also noted that if the licensee were to consider extending the current testing frequency in accordance with applicable code provisions beyond 18 months, the licensee would be responsible for evaluating the new testing frequency and its contribution to risk.

3.1.6 RCS PIVs Leakage Control

TS SR 3.4.14.1, Surveillance, requires that RCS PIV leakage from each RCS PIV to be equivalent to ≤ 0.5 gallons per minute (gpm) per nominal inch of valve size up to a maximum of 5 gpm at an RCS pressure ≥ 2215 pounds per square inch gage (psig) and ≤ 2255 psig.

In the RAI response dated February 10, 2025, related to use of the mechanical agitation to control the PIV leakage, TVA stated the following:

SQN Units 1 and 2 Alternative Request RV-02 (Reference 1 of this RAI Response [ML22074A315]) was approved by the NRC in Reference 2 of this RAI response [ML22166A167]. Alternative Request RV-02 cites the various requirements of SR 3.4.14.1 in the Reason for Request section including the current requirements for frequency of performance, which include those proposed for deletion in the LAR Reference 3 of this RAI response). Further, in the Basis for Proposed Alternative section of RV-02, TVA indicated that, for PIVs which are opened by flow during shutdowns, the required seat leakage tests will be performed, and acceptable results obtained prior to entering Mode 2 or the plant cannot startup. Retesting the seat leakage following flow through the valve, as discussed in Alternative Request RV-02, may not be required with approval of the proposed TS changes in Reference 3 of this RAI response.

Therefore, TVA will submit a revision to Alternative Request RV-02 within 30 days following NRC approval of Reference 3 of this RAI response. The revisions to RV-02 will address approved changes to the SR 3.4.14.1 performance frequency requirements and changes in the Basis for the Proposed Alternative. TVA will not apply either the existing or proposed revised Alternative Request RV-02 once submitted until the revised request is approved by the NRC.

In the RAI response dated February 10, 2025, the licensee discussed the use of ASME OM Code Case OMN-23, "Alternative Rules for Testing Pressure Isolation Valves [PIVs]," which allows extension of the PIV leakage testing interval up to 6 years as accepted in RG 1.192 as incorporated by reference in 10 CFR 50.55a. In particular, TVA stated the following:

TVA has not adopted American Society of Mechanical Engineers (ASME) Operations and Maintenance (OM) Code Case OMN-23 for application in the SQN Units 1 and 2 IST program. ASME OM Code Case OMN-23 provides adequate controls to ensure component performance with leakage test intervals not to exceed 6 years. OMN-23 is an alternative to the two year test frequency specified in ISTC-3630(a). The TS test frequencies are not addressed in the Code Case OMN-23, therefore there is no impact to the proposed license amendment or future application of OMN-23.

The NRC staff finds these responses by TVA with respect to the NRC-authorized Alternative Request RV-02 for SQN Units 1 and 2 and the NRC-accepted ASME OM Code Case OMN-23 in RG 1.192 to provide adequate controls to avoid conflict during the implementation of this LAR and the activities related to PIV leakage in those documents. The NRC staff also note that if the licensee were to consider extending the current testing frequency in accordance with applicable code provisions beyond 18 months, the licensee would be responsible for evaluating the new testing frequency.

3.1.7 Inservice Testing Program

The NRC staff evaluated the information provided in the LAR with respect to the relationship to the IST Program at SQN Units 1 and 2. The NRC staff finds that the proposed changes to each SQN unit's TS SR Frequency will not impact the ASME OM Code testing requirements as incorporated by reference in 10 CFR 50.55a for the RCS PIVs in the IST Program. Accordingly,

the proposed TS SR Frequency change to “Inservice Testing Program,” is acceptable. TS Section 5, Administrative Controls, Section 5.5 Program and Manual, TS Section 5.5.6 “Inservice Testing Program,” provides controls for inservice testing of ASME Code Class 1, 2, and 3 pumps and valves. The NRC staff notes that the licensee continues to be required to meet the requirements in ASME OM Code, Subsection ISTC, paragraph ISTC-3310, “Effects of Valve Repair, Replacement, or Maintenance on Reference Values,” that an inservice test shall be performed when a valve or its control system has been replaced, repaired, or undergone maintenance that could affect the valve’s performance before it can be returned to service. Therefore, the NRC staff finds that the deletion of the corresponding requirement from the TS is acceptable.

3.2 Technical Conclusion

The NRC staff evaluated the information provided in the LAR. The NRC staff finds that the proposed changes to the TS SR Frequency ensures ASME OM Code testing frequency requirements for the RCS PIVs will be retained in the IST Program, that duplicative testing requirements in the TS SR will be removed and that the TS SR Frequency change to Inservice Testing Program is sufficient. The NRC staff also finds that the proposed changes are consistent with NRC staff positions and the regulations with respect to other requirements, specifically, 10 CFR 50.55a(f)(5)(ii), “IST program update: Conflicting IST Code requirements with technical specifications.”

As described above, TVA will not apply Alternative Request RV-02 for mechanical agitation to address PIV leakage at SQN Units 1 and 2 until a revision is submitted and authorized by the NRC. Further, TVA has not currently adopted ASME OM Code Case OMN-23 to extend the PIVs leakage test frequencies. Therefore, there is no impact on this proposed LAR.

In addition, the NRC staff finds the proposed change acceptable because the impact on plant risk due to eliminating the fixed and event driven testing frequencies was considered a negligible contributor to the risk of core melt and large early release.

Therefore, based on the above, the staff finds the licensee’s request acceptable, because the SR remains consistent with the requirement specified in 10 CFR 50.36(c)(3). The NRC staff also finds that the changes proposed in the subject LAR do not adversely affect the licensee’s current compliance with the GDC listed in Section 2.3 of this safety evaluation.

4.0 STATE CONSULTATION

In accordance with the Commission’s regulations, the Tennessee State official was notified of the proposed issuance of the amendments on February 24, 2025. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, originally published in the *Federal Register* on December 23,

2024 (89 FR 104572). There has been no public comment on such finding. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) there is reasonable assurance that such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: B. Rothberg, NRR
D. Scully, NRR
G. Bedi, NRR
M. Schwieg, NRR
T. Scarbrough, NRR
J. Robinson, NRR
C. Ashley, NRR

Date: March 6, 2025

SUBJECT: SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENT NOS. 370 AND 364 REGARDING A REVISION TO TECHNICAL SPECIFICATIONS SURVEILLANCE REQUIREMENT 3.4.14.1, "REACTOR COOLANT SYSTEM (RCS) PRESSURE ISOLATION VALVE (PIV) LEAKAGE," (EPID L-2024-LLA-0156) DATED MARCH 6, 2025

DISTRIBUTION:

PUBLIC
 RidsNrrDorlLpl2-2
 RidsNrrPMSequoyah
 RidsNrrLAABaxter
 RidsACRS_MailCTR
 RidsRgn2MailCenter
 RidsNrrDexEeeb
 RidsNrrDssStsb
 RidsNrrDssScpb
 RidsNrrDraAplb
 CAshley, NRR
 BRothberg, NRR
 DScully, NRR
 GBedi, NRR
 MSchwieg, NRR
 TScarborough, NRR
 JRobinson, NRR

ADAMS Accession No.: ML25027A314

NRR-058

OFFICE	NRR/DORL/LPLII-2/PM	NRR/DORL/LPLII-2/LA	NRR/DEX/EMIB/BC	NRR/DSS/STSB/BC
NAME	PBuckberg	ABaxter (SLent for)	SBailey	SMehta (ARussell for)
DATE	2/25/2025	2/25/2025	2/25/2025	2/26/2025
OFFICE	NRR/DSS/SCPB/BC	NRR/DRA/APLB/BC	OGC – NLO w/edits	NRR/DORL/LPLII-2/BC
NAME	MValentin (BLee for)	EDavidson	MCarpentier	DWrona
DATE	2/26/2025	2/25/2025	3/5/2025	3/6/2025
OFFICE	NRR/DORL/LPLII-2/PM			
NAME	PBuckberg (KGreen for)			
DATE	3/6/2025			

OFFICIAL RECORD COPY