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

June 23, 2025

Mr. 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. 372 AND 366 REGARDING REVISION OF THE FUEL HANDLING ACCIDENT ANALYSIS, DELETION OF TECHNICAL SPECIFICATION 3.9.4, AND REVISION OF TECHNICAL SPECIFICATION 3.3.6 (EPID L-2024-LLA-0117)

Dear Mr. Erb:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 372 to Renewed Facility Operating License No. DPR-77, and Amendment No. 366 to Renewed Facility Operating License No. DPR-79, for the Sequoyah Nuclear Plant (Sequoyah), Units 1 and 2, respectively. These amendments are in response to your application dated August 28, 2024, as supplemented by letters dated January 13, 2025, and March 17, 2025.

The amendments revise the Sequoyah fuel handling accident analysis, delete Sequoyah, Units 1 and 2, Technical Specification (TS) 3.9.4, "Containment Penetrations," and modify TS 3.3.6, "Containment Ventilation Isolation Instrumentation."

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

Sincerely,

/RA/

Kimberly Green, 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. 372 to DPR-77
2. Amendment No. 366 to DPR-79
3. Safety Evaluation

cc: Listserv

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. 372
Renewed License No. DPR-77

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated August 28, 2024, as supplemented by letters dated January 13, 2025, and March 17, 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. 372 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 its date of issuance and shall be implemented within 120 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

DAVID WRONA

Digitally signed by DAVID WRONA
Date: 2025.06.23 15:43:37 -04'00'

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

Attachment:
Changes to the Renewed Facility Operating License and Technical Specifications

Date of Issuance: June 23, 2025

ATTACHMENT TO LICENSE AMENDMENT NO. 372

SEQUOYAH NUCLEAR PLANT, UNIT 1

RENEWED FACILITY OPERATING LICENSE NO. DPR-77

DOCKET NO. 50-327

Replace page 3 of the Renewed Facility Operating License with the attached page 3. The revised page 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.

Remove	Insert
3.3.6-1	3.3.6-1
3.3.6-2	3.3.6-2
3.3.6-3	3.3.6-3
3.3.6-4	3.3.6-4
3.3.6-5	3.3.6-5
3.3.6-6	--
3.9-4-1	3.9.4-1
3.9.4-2	--

(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. 372 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;

3.3 INSTRUMENTATION

3.3.6 Containment Ventilation Isolation Instrumentation

LCO 3.3.6 The Containment Ventilation Isolation instrumentation for each Function in Table 3.3.6-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.6-1.

ACTIONS

--NOTE--

Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ------------NOTE------------ Only applicable in MODE 1, 2, 3, or 4. -------------------------------- One or more Functions with one or more manual or automatic actuation trains inoperable. <u>OR</u> One required radiation monitoring channel inoperable.	A.1 Enter applicable Conditions and Required Actions of LCO 3.6.3, "Containment Isolation Valves," for containment purge supply and exhaust isolation valves made inoperable by isolation instrumentation.	Immediately

SURVEILLANCE REQUIREMENTS

--NOTE--

Refer to Table 3.3.6-1 to determine which SRs apply for each Containment Ventilation Isolation Function.

SURVEILLANCE		FREQUENCY
SR 3.3.6.1	Perform CHANNEL CHECK.	In accordance with the Surveillance Frequency Control Program
SR 3.3.6.2	-------------------------------NOTE------------------------------ This Surveillance is only applicable to the actuation logic of the ESFAS Instrumentation. --- Perform ACTUATION LOGIC TEST.	In accordance with the Surveillance Frequency Control Program
SR 3.3.6.3	-------------------------------NOTE------------------------------ This Surveillance is only applicable to the master relays of the ESFAS Instrumentation. --- Perform MASTER RELAY TEST.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.6.4	Perform COT.	In accordance with the Surveillance Frequency Control Program
SR 3.3.6.5	Perform SLAVE RELAY TEST.	In accordance with the Surveillance Frequency Control Program
SR 3.3.6.6	------------------------------NOTE------------------------------ Verification of setpoint is not required. -- Perform TADOT.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.6.7	Perform CHANNEL CALIBRATION.	In accordance with the Surveillance Frequency Control Program
SR 3.3.6.8	-----------------------------NOTE------------------------------ Radiation detectors are excluded from response time testing. --- Verify ESF RESPONSE TIME is within limits.	In accordance with the Surveillance Frequency Control Program

Table 3.3.6-1 (page 1 of 1)
Containment Ventilation Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	TRIP SETPOINT
1. Manual Initiation	1,2,3,4	2	SR 3.3.6.6	NA
2. Automatic Actuation				NA
a. Logic	1,2,3,4	2 trains	SR 3.3.6.2	NA
b. Relays	1,2,3,4	2 trains	SR 3.3.6.3 SR 3.3.6.5	NA
3. Containment Purge Air Radiation Monitor	1,2,3,4	1	SR 3.3.6.1 SR 3.3.6.4 SR 3.3.6.7 SR 3.3.6.8	≤ 8.5 x 10^{-3} μCi/cc
4. Safety Injection	Refer to LCO 3.3.2, "ESFAS Instrumentation," Function 1, for all initiation functions and requirements.			

3.9 REFUELING OPERATIONS

3.9.4 Deleted

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. 366
Renewed License No. DPR-79

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated August 28, 2024, as supplemented by letters dated January 13, 2025, and March 17, 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 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. 366 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 its date of issuance and shall be implemented within 120 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

DAVID WRONA

Digitally signed by DAVID WRONA
Date: 2025.06.23 15:45:16 -04'00'

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

Attachment:
Changes to the Renewed Facility Operating License and Technical Specifications

Date of Issuance: June 23, 2025

ATTACHMENT TO LICENSE AMENDMENT NO. 366

SEQUOYAH NUCLEAR PLANT, UNIT 2

RENEWED FACILITY OPERATING LICENSE NO. DPR-79

DOCKET NO. 50-328

Replace page 3 of the Renewed Facility Operating License with the attached page 3. The revised page 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.

Remove	Insert
3.3.6-1	3.3.6-1
3.3.6-2	3.3.6-2
3.3.6-3	3.3.6-3
3.3.6-4	3.3.6-4
3.3.6-5	3.3.6-5
3.3.6-6	--
3.9-4-1	3.9.4-1
3.9.4-2	--

(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. 366 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;

3.3 INSTRUMENTATION

3.3.6 Containment Ventilation Isolation Instrumentation

LCO 3.3.6 The Containment Ventilation Isolation instrumentation for each Function in Table 3.3.6-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.6-1.

ACTIONS

---NOTE---

Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ------------NOTE------------ Only applicable in MODE 1, 2, 3, or 4. -------------------------------- One or more Functions with one or more manual or automatic actuation trains inoperable. <u>OR</u> One required radiation monitoring channel inoperable.	A.1 Enter applicable Conditions and Required Actions of LCO 3.6.3, "Containment Isolation Valves," for containment purge supply and exhaust isolation valves made inoperable by isolation instrumentation.	Immediately

SURVEILLANCE REQUIREMENTS

---NOTE---
Refer to Table 3.3.6-1 to determine which SRs apply for each Containment Ventilation Isolation Function.

SURVEILLANCE		FREQUENCY
SR 3.3.6.1	Perform CHANNEL CHECK.	In accordance with the Surveillance Frequency Control Program
SR 3.3.6.2	-------------------------------NOTE------------------------------ This Surveillance is only applicable to the actuation logic of the ESFAS Instrumentation. -- Perform ACTUATION LOGIC TEST.	 In accordance with the Surveillance Frequency Control Program
SR 3.3.6.3	-------------------------------NOTE------------------------------ This Surveillance is only applicable to the master relays of the ESFAS Instrumentation. -- Perform MASTER RELAY TEST.	 In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.6.4	Perform COT.	In accordance with the Surveillance Frequency Control Program
SR 3.3.6.5	Perform SLAVE RELAY TEST.	In accordance with the Surveillance Frequency Control Program
SR 3.3.6.6	-------------------------------NOTE------------------------------ Verification of setpoint is not required. --- Perform TADOT.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.6.7	Perform CHANNEL CALIBRATION.	In accordance with the Surveillance Frequency Control Program
SR 3.3.6.8	-----------------------------NOTE------------------------------ Radiation detectors are excluded from response time testing. --- Verify ESF RESPONSE TIME is within limits.	In accordance with the Surveillance Frequency Control Program

Table 3.3.6-1 (page 1 of 1)
Containment Ventilation Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	TRIP SETPOINT
1. Manual Initiation	1,2,3,4	2	SR 3.3.6.6	NA
2. Automatic Actuation				NA
a. Logic	1,2,3,4	2 trains	SR 3.3.6.2	NA
b. Relays	1,2,3,4	2 trains	SR 3.3.6.3 SR 3.3.6.5	NA
3. Containment Purge Air Radiation Monitor	1,2,3,4	1	SR 3.3.6.1 SR 3.3.6.4 SR 3.3.6.7 SR 3.3.6.8	≤ 8.5 x 10^{-3} μCi/cc
4. Safety Injection	Refer to LCO 3.3.2, "ESFAS Instrumentation," Function 1, for all initiation functions and requirements.			

3.9 REFUELING OPERATIONS

3.9.4 Deleted

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 372

RENEWED FACILITY OPERATING LICENSE NO. DPR-77

AND AMENDMENT NO. 366

RENEWED FACILITY OPERATING LICENSE NO. DPR-79

TENNESSEE VALLEY AUTHORITY

SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-327 AND 50-328

1.0 INTRODUCTION

By application dated August 28, 2024 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML24247A185 (package), as supplemented by letters dated January 13, 2025 (ML25013A334), and March 17, 2025 (ML25076A261), Tennessee Valley Authority (TVA or the licensee) requested changes to the technical specifications (TSs) for Sequoyah Nuclear Plant (Sequoyah or SQN), Units 1 and 2. The proposed changes would revise the fuel handling accident (FHA) analysis to no longer credit containment penetration closure. The proposed changes also would revise the SQN, Units 1 and 2, TSs to delete TS 3.9.4, “Containment Penetrations,” it its entirety, and modify TS 3.3.6, “Containment Ventilation Isolation Instrumentation.”

The supplements dated January 13, 2025, and March 17, 2025, provided additional information that clarified the license amendment request (LAR), did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC or the Commission) staff’s original proposed no significant hazards consideration determination as published in the *Federal Register* on October 29, 2024 (89 FR 85996).

2.0 REGULATORY EVALUATION

2.1 Background

The FHA is described in section 15.5.6 of the Sequoyah Nuclear Plant Updated Final Safety Analysis Report (UFSAR) (ML23349A014). In the UFSAR, the licensee analyzed two scenarios—an event occurring in the auxiliary building (AB) and an event occurring inside the primary containment. These analyses use the methodology from Regulatory Guide (RG) 1.183, Revision 0, and use an alternative source term (AST).

As noted in the UFSAR, the FHA inside the AB assumes, in part, the following: it occurs 100 hours after the plant shuts down; all fuel rods in one assembly rupture; the highest powered assembly in the core region is damaged; all of the gap activity in the damaged rods is released to the spent fuel pool; no credit is taken for natural decay either due to holdup in the AB or after release to the atmosphere; and the activity released from the pool is all assumed to be released to the environment over a 2-hour period.

An FHA inside the primary containment has several possible release paths, including the reactor building purge ventilation (RBPV) system, when operating, which would discharge through the shield building (SB) vent, and any open containment penetrations and hatches. A higher atmospheric dispersion factor will result in a higher dose, and according to the UFSAR, the AB vent has the highest atmospheric dispersion factor of all other potential release points; therefore, the analysis for an FHA inside the containment does not result in a greater dose than an FHA in the AB.

The RBPV system is designed, in part, to limit releases of radioactivity to the environment. The purge system consists of supply and exhaust fans, cleanup filters, and among other components, containment isolation valves. The supply fans, exhaust fans, and air cleanup filter assemblies for each unit are connected and controlled in two 50 percent capacity trains. The controls are designed to produce simultaneous starting and stopping of the matching supply and exhaust equipment. The controls are also designed to result in an automatic shutdown and isolation upon receipt of containment ventilation isolation (CVI) signal.

The RBPV system will isolate in the event radioisotopes are released from the fuel rod assembly during an FHA inside containment. The FHA analysis takes no credit for the cleanup operation of the containment purge exhaust system to mitigate the accident. Rather, the system is assumed to be isolated on a CVI signal by the purge line radiation monitors and the associated containment isolation valves on high radiation in the exhaust air stream. The containment purge system will also be isolated upon the actuation of a CVI signal whenever the primary containment is being purged during normal operation.

Sequoyah, Units 1 and 2, TS 3.3.6 contain conditions for the operation of the CVI instrumentation.

2.2 Requested Changes

The licensee proposed the following revisions to the FHA analysis:

- no longer credit the containment penetration closure,
- update the assumption in hours of delay after shutdown,
- eliminate the tritium source term associated with a fuel assembly containing tritium-producing burnable absorber rods (TPBARs),
- no longer analyze an FHA inside containment, and
- update the atmospheric dispersion factors.

The licensee proposed the following revisions to the SQN, Units 1 and 2, TSs:

- delete TS 3.3.6, ACTION B,
- revise the FREQUENCY for Surveillance Requirement (SR) 3.3.6.4 to remove reference to “movement of irradiated fuel,” and delete the logical connector “AND”,

- revise the FREQUENCY for SR 3.3.6.6 to remove reference to “movement of irradiated fuel” and delete the logical connector “AND”,
- delete Table 3.3.6-1, SPECIFIED CONDITION (a), and
- delete TS 3.9.4

2.3 <u>Regulatory Requirements and Guidance</u>

Under Title 10 of the *Code of Federal Regulations* (10 CFR) 50.90, “Application for amendment of license, construction permit, or early site permit,” whenever a holder of a license wishes to amend the license, including TSs in the license, an application for amendment must be filed, fully describing the changes desired.

Under 10 CFR 50.92(a), determinations on whether to grant an applied-for license amendment are to be guided by the considerations that govern the issuance of initial licenses to the extent applicable and appropriate. Both the common standards for licenses in 10 CFR 50.40(a), and those specifically for issuance of operating licenses in 10 CFR 50.57(a)(3), provide that there must be reasonable assurance that the activities at issue will not endanger the health and safety of the public, and that the applicant will comply with the Commission’s regulations.

Section 182a of the Atomic Energy Act requires applicants for nuclear power plant operating licenses to include TSs as part of the license. The Commission’s regulatory requirements related to the content of TSs are set forth in 10 CFR 50.36, “Technical Specifications,” which require, in pertinent part, that the TSs include: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions of operation (LCOs); (3) SRs; (4) design features; and (5) administrative controls.

Section 50.36(a)(1) states, that “Each applicant for a license authorizing operation of a . . . utilization facility shall include in his application proposed technical specifications with the requirements of this section. A summary statement of the bases or reasons for such specifications . . . shall also be included in the application, but shall not become part of the technical specifications.”

As stated in 10 CFR 50.36(b), each license authorizing operation of a production or utilization facility will include technical specifications. The TSs will be derived from the analyses and evaluation included in the safety analysis report, and amendments thereto.

Under 10 CFR 50.36(c)(2), the technical specifications will include LCOs, which are the lowest functional capability or performance level of equipment required for safe operation of the facility. When LCOs are not met, the licensee shall shut down the reactor or follow any remedial action permitted by the TSs until the LCOs can be met. Additionally, an LCO must be established for each item that meets one or more of the following four criteria in 10 CFR 50.36(c)(2)(ii):

- *Criterion 1*. Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
- *Criterion 2*. A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
- *Criterion 3*. A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that

either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

- *Criterion 4*. A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

Under 10 CFR 50.36(c)(3), TSs will include SRs, which 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 LCOs will be met.

Under paragraph (a) of 10 CFR 50.34, "Contents of applications; technical information," each applicant for an operating license is required to provide an analysis and evaluation of the design and performance of structures, systems, and components of the facility with the objective of assessing the risk to public health and safety resulting from operation of the facility. Applicants are also required by 10 CFR 50.34 to provide an analysis of the proposed site. These analyses, called design basis radiological consequence analyses, include evaluations of the radiological consequences at prescribed distances from the facility and at prescribed locations within the facility.

The regulation at 10 CFR 50.34(a)(3) cites Appendix A, "General Design Criteria [GDC] for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," which establishes the minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have been issued by the Commission. The principal design criteria establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety. The NRC staff considered the following GDC as part of its review:

GDC 19, "Control room," states, in part:

> A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents . . . Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

According to section 3.1.2, "Overall Requirements," of the SQN UFSAR, the plant was designed to meet the intent of the "Proposed General Design Criteria for Nuclear Power Plant Construction Permits," published in July 1967. The SQN construction permit was issued in May 1970. The SQN UFSAR addresses the NRC GDC published as Appendix A to 10 CFR Part 50. As described therein, the plant is provided with a main control room within the control building. The control room is designed and equipped to minimize the possibility of events such as fire, high radiation levels, excessive temperature, etc., which might preclude occupancy. Sufficient shielding, distance, and containment integrity are provided to assure that under postulated accident conditions control room personnel shall not be subjected to radiation doses which would exceed 5 rem (roentgen equivalent man) to the whole body, or its equivalent to any part of the body, including doses received during both ingress and egress. Control room ventilation is provided by a system having a large percentage of recirculated air. After an accident, makeup air will automatically be routed through a system of HEPA and charcoal filters.

Under 10 CFR 50.67, a licensee can revise the accident source term that is assumed in design basis radiological consequence analyses through a license amendment. The application for such an amendment shall contain an evaluation of the consequences (i.e., the radiological doses at the exclusion area boundary (EAB), low population zone (LPZ) and the control room) of the design basis accidents that would be impacted by the amendment. The application can request either a full or selective implementation, where a selective implementation involves reevaluation of a limited subset of design basis radiological analyses as described in section 1.2.2, Revision 0, RG 1.183. In October 2003, the NRC issued amendments approving the selective implementation of the AST for the FHA at SQN, Units 1 and 2 (ML033030206).

The regulation at 10 CFR 50.67(b)(2) requires that the licensee's analysis demonstrates with reasonable assurance that:

(i) An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 0.25 Sv [Sievert] (25 rem) total effective dose equivalent (TEDE).
(ii) An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), would not receive a radiation dose in excess of 0.25 Sv (25 rem) TEDE.
(iii) Adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) TEDE for the duration of the accident.

The regulation at 10 CFR Part 100, "Reactor Site Criteria," Section 100.11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance," provides criteria for evaluating the radiological aspects of the proposed site based on distances from the site.

Safety Guide 23, "Onsite Meteorological Programs," February 17, 1972, also referred to as RG 1.23, Revision 0, describes (what was then considered) a suitable onsite meteorological program to provide meteorological data needed to estimate potential radiation doses to the public resulting from actual routine or accidental releases of radioactive materials to the atmosphere, or to evaluate the potential dose to the public from hypothetical reactor accidents.

RG 1.23, "Meteorological Monitoring Programs for Nuclear Power Plants," provides current guidance to licensees concerning criteria for an onsite meteorological measurements program that NRC staff considers acceptable for the collection of basic meteorological data needed to support plant licensing and operation.

RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," identifies acceptable methods for (1) calculating atmospheric relative concentration (X/Q) values, (2) determining X/Q values on a directional basis, (3) determining X/Q values on an overall site basis, and (4) choosing X/Q values to be used in evaluations of the types of events described in Regulatory Guides 1.3 and 1.4.

RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," provides guidance to licensees on acceptable applications of alternative source terms (ASTs). It establishes an acceptable AST and identifies the significant attributes of other ASTs that may be found acceptable by the NRC staff. It also includes an appendix that contains assumptions acceptable to the NRC staff for evaluating the radiological consequences of an FHA at light-water reactors.

RG 1.194, “Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants,” provides guidance on determining X/Q values in support of design basis control room radiological habitability assessments at nuclear power plants in support of applications for licenses and license amendment requests.

NRC staff guidance for the safety review of construction permits or operating license applications (including requests for amendments) under 10 CFR Part 50 is provided in NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition” (SRP). The NRC staff considered the following SRP sections during its review of this LAR:

- Section 2.3.3, “Onsite Meteorological Measurements Program,” Revision 3, March 2007.
- Section 2.3.4, “Short-Term Atmospheric Dispersion Estimates for Accident Releases,” Revision 3, March 2007.
- Section 6.4, “Control Room Habitability System,” Revision 3, March 2007.
- Section 15.0.1, “Radiological Consequence Analyses Using Alternative Source Terms,” Revision 0, July 2000.
- Chapter 16.0, “Technical Specifications,” Revision 3, March 2010. As described therein, as part of the regulatory standardization effort, the NRC staff has prepared Standard Technical Specifications (STS) for each of the LWR nuclear designs. The SQN units are Westinghouse four-loop design reactors. Accordingly, the NRC staff’s review includes consideration of whether the proposed changes are consistent with NUREG-1431[1], as modified by NRC approved travelers.

NUREG/CR-2260, “Technical Basis for Regulatory Guide 1.145, “Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants,” describes a parametric study that was performed to examine the consequences on licensing activities related to the substantial changes to the methodology for atmospheric dispersion analyses in RG 1.145 as compared to the previous methodology in RGs 1.3 and 1.4 used by the staff in the review of construction permit applications until adequate site meteorological (Met) data were obtained.

NUREG/CR-2858, “PAVAN: An Atmospheric Dispersion Program for Evaluating Design-Basis Accidental Releases of Radioactive Materials from Nuclear Power Plants,” provides a user's guide for the NRC computer program, PAVAN, used by NRC staff to estimate downwind ground-level air concentrations for potential accidental releases of radioactive material from nuclear facilities. Such an assessment is required by 10 CFR Part 100 and 10 CFR Part 50. The computer program implements the guidance provided in RG 1.145.

NUREG/CR-6331, Revision 1, “Atmospheric Relative Concentrations in Building Wakes,” May 1997, documents the ARCON96 computer code developed for the NRC staff for potential use in control room habitability assessments. The ARCON96 code uses hourly meteorological data and methods for estimating dispersion in the vicinity of buildings to calculate relative

[1] U.S. Nuclear Regulatory Commission, “Standard Technical Specifications, Westinghouse Plants,” NUREG-1431, Volume 1, “Specifications,” Revision 5, and Volume 2, “Bases,” Revision 5, September 2021 (ML21259A155 and ML21259A159, respectively).

concentrations at control room air intakes that would be exceeded no more than five percent of the time.

Regulatory Issue Summary (RIS) 2006-04, “Experience with Implementation of Alternative Source Terms,” updates reactor licensees on experience with implementation of ASTs in design basis accident radiological analyses of currently licensed light-water reactors.

3.0 TECHNICAL EVALUATION

The NRC staff evaluated the licensee’s application, as supplemented, to determine whether the proposed changes are consistent with the regulations, guidance, and plant-specific design and licensing basis information discussed in section 2.3 of this safety evaluation.

3.1 Evaluation of Revision to Fuel Handling Accident

The LAR proposed the following changes to the FHA analysis:

- no longer credit the containment penetration closure
- no longer analyze an FHA inside containment,
- update the assumption in hours of delay after shutdown
- eliminate the tritium source term associated with a fuel assembly containing TPBARs, and
- update the atmospheric dispersion factors.

The licensee stated that all input and assumptions used in the previous FHA analysis, which was evaluated by the NRC staff as part of a TS change and selective implementation of an alternate source term (ML033030206 and ML030160157, respectively) remain the same, except for the proposed changes described above. The licensee has not proposed any deviations or departure from guidance provided in RG 1.183, Revision 0.

To make its findings and draw conclusions in accordance with 10 CFR 50.92, the NRC staff evaluated the licensee's revised FHA against the radiological dose acceptance criteria specified for the FHA in table 6 of RG 1.183, Revision 0. The staff also confirmed that other input and assumptions remained unchanged.

3.1.1 Evaluation of No Longer Analyzing an FHA Inside Containment and No Longer Crediting Containment Penetration Closure

The licensee had previously analyzed an FHA inside containment; however, it proposes to no longer do so and instead will credit a bounding analysis of an FHA outside containment. This change will enable the licensee to no longer credit containment penetration closure as part of its FHA analysis.

The licensee explained that the FHA inside containment assumed fission products from the spent fuel pool are exhausted and released through the SB vent, until the purge system is isolated upon detection of increased radiation levels in containment, and are then released through the AB vent. The FHA outside containment assumes the release is through the AB vent. The SB vent has a lower X/Q than the AB vent. Therefore, the licensee concluded that because the AB vent X/Q is higher, and because the source term and transport parameters are the same for both vent paths, the FHA outside containment will always be bounding.

The licensee also stated that a majority of other containment building penetrations do not release to the environment. Releases from these penetrations would be through the SB vent via the containment purge and emergency gas treatment system. As mentioned above, releases through the SB vent are bounded by releases through the AB vent. The licensee further stated that all other containment penetrations to the outside were determined to be either physically below the control room intake, or farther away than the AB vent and not in a more dominant wind sector. As a result, the licensee concluded that the AB vent bounds a release from any existing containment building penetration that opens to the environment.

The licensee stated in section 3.2.1 of enclosure 1 of the LAR that the containment building equipment hatch and airlock doors open into the AB. The licensee asserted that if an FHA were to occur inside containment with these doors open, the releases would flow into the AB, and that this is the same scenario as the FHA outside containment, which it stated is the bounding case. Therefore, the licensee concluded that closure of the containment building equipment hatch and airlock doors during movement of irradiated fuel assemblies is no longer a primary success path to mitigate an FHA.

The licensee analyzed various release point-receptor combinations for the onsite X/Q to determine the limiting (i.e., overall highest) control room X/Q. In this analysis the licensee considered the following release points: Unit 1 SB vent, Unit 2 SB vent, and AB vent. The receptors were the normal control room intake and the main control room emergency intake. The licensee provided the results of this analysis in table 3.1.1-3 of enclosure 1 of the LAR. NRC staff reviewed these results and verified that the AB vent release point, in combination with the normal control room intake, resulted in the highest X/Q. Consistent with the guidance in RG 1.183, Revision 0, Appendix B, Regulatory Position 4, the licensee assumed that the release of fission products to the environment occurs over a 2-hour period and the licensee did not credit holdup or dilution of the released activity within the auxiliary building.

Since the licensee takes no credit for reduction of the source term associated with transport, i.e., filtration, holdup, or dispersion within the AB and a higher X/Q value at the AB vent release point results in less dispersion and higher doses at the dose receptors, the NRC staff finds the licensee's assertion that the FHA outside containment (which releases to the AB vent) to be bounding is reasonable. Since the licensee's results are based on an evaluation of the FHA outside containment and this analysis bounds the previously evaluated FHA inside containment, the NRC staff finds the elimination of the analysis of an FHA inside containment acceptable. Additionally, since an FHA with containment penetrations open is effectively the same accident scenario as an FHA outside containment, the NRC staff finds it acceptable that the licensee no longer credits containment penetration closure as part of its FHA analysis.

3.1.2 Evaluation of Delay After Shutdown

In its analysis and in accordance with its licensing basis, the licensee accounts for radioactive decay of the fission product inventory in the gap of the damaged fuel assembly during the interval between shutdown and the commencement of fuel handling activities, i.e., the time typically assumed for FHA analyses. TS 3.9.8, “Decay Time,” requires that the reactor be subcritical for at least 100 hours prior to commencing core alterations and, thus, controls the earliest time fuel handling can begin. However, the licensee used a delay of 70 hours following shutdown in its revised FHA analysis. In section 3.3.1 of enclosure 1 of the LAR, the licensee explained that its decision to use the 70-hour decay time, instead of the 100-hour time reflected in TS 3.9.8, was made to establish a technical basis to support a potential future amendment of TS 3.9.8, but is not requested as part of this LAR.

The radiological source term decreases as more time elapses after reactor shutdown. Therefore, the 70-hour decay time assumption is conservative because it represents a shorter time since shutdown resulting in a larger radiological source term. The larger radiological source term will result in calculated radiological doses that are higher and, thus, more conservative than would result if the 100 hour decay time from TS 3.9.8 were used in the analysis. Therefore, the NRC finds the licensee's assumption of 70 hours of decay to be acceptable, but recognizes that, in accordance with TS LCO 3.9.8, 100 hours will remain the earliest that irradiated fuel can be moved at SQN, Units 1 and 2.

3.1.3 Evaluation of Elimination of Tritium Source Term

The licensee stated that it does not currently have any TPBARs installed in the SQN reactors, nor does it plan to install any in the future. Therefore, it proposed to eliminate tritium from the source term associated with a fuel assembly containing TPBARs. Additionally, to support eliminating the tritium source term from the analysis, the licensee referenced an email dated October 12, 2011 (ML11285A203), which acknowledges that it must request and receive NRC approval before introducing TPBARs into either reactor unit.

Since the operating licenses do not permit introducing TPBARs at SQN, Units 1 and 2, the NRC staff finds that it is acceptable for the licensee to remove the tritium source term from the revised FHA.

3.1.4 Evaluation of Meteorological Measurements and Atmospheric Dispersion Factor Estimates

The licensee updated the main control room (MCR) (i.e., onsite) and offsite accident-related dispersion factors (X/Q values) using meteorological (Met) data from 2004 – 2013. The licensee stated that it developed its Met monitoring program consistent with the guidance in RG 1.23, Revision 1; that the wind direction is consistent with section 5.3.1 of American National Standards Institute/American Nuclear Society (ANSI/ANS) 3.11, "Determining Meteorological Information at Nuclear Facilities"; and that the number of wind speed categories reflects the guidance of RIS 2006-04, "Experience with Implementation of Alternative Source Terms." The licensee's offsite and onsite atmospheric dispersion modeling analyses used the PAVAN and the ARCON96 codes, respectively. The NRC staff reviewed applicable portions of the LAR, checked the licensee's dispersion modeling inputs, and compared its confirmatory modeling results against those of the licensee. Additional information specific to the Met data and each of the modeling analyses are discussed in their respective subsections below.

3.1.4.1 Meteorological Measurements and Dispersion Model Input Data

The SQN UFSAR indicates that the Met monitoring program commenced in April 1971, including use of a 91-meter (m) (300-ft) Met tower. System configuration and instrumentation has changed somewhat over time. Nevertheless, measurements relevant to atmospheric dispersion modeling include wind speed, wind direction, and temperature at nominal heights of 10, 46, and 91 m (33, 150, and 300 ft) above ground level. Vertical temperature differences (or delta-Ts) between the 46- and 10-m levels and the 91- and 10-m levels (i.e., upper minus lower) are determined to characterize atmospheric stability. Measurements are made on a continuous basis.

For this LAR, the licensee used Met data covering a 10-year period of record (POR) from January 1, 2004, through December 31, 2013. The same POR is used for both the offsite and onsite dispersion modeling analyses. However, the input data format varies depending on the dispersion model.

Met input for the offsite PAVAN model runs consisted of 10-year composite joint frequency distributions (JFDs) of hourly-averaged wind speeds and wind directions by seven classes of atmospheric stability (designated A though G). Met input data for the onsite ARCON96 model runs consisted of ten individual files of hourly wind speed, wind direction, and stability class values, one for each year of the 10-year POR from which a composite of that model's output results was determined internally by the code.

The current version of the SQN UFSAR indicates that wind speed and wind direction measurements are determined with an ultrasonic wind sensor. Based on Enclosure 1 of the LAR (ML24247A175), scalar averaging was used to determine hourly wind speed values. Scalar averaging is essentially equivalent to arithmetic averaging. Hourly wind directions, although a vector quantity, were determined as unit vector averages (i.e., unit vectors are only weighted by a 1.0 m/sec or 1 mph wind speed based on the units of measure as opposed to the concurrent wind speed if full vector averaging is used). The NRC staff recognized and determined that this approach is consistent with section 5.3.1 of ANSI/ANS-3.11 (i.e., "Determining Meteorological Information at Nuclear Facilities").

The number of wind speed categories represented in the JFDs meet the intent of the guidance in table 3 of Revision 1 to RG 1.23. This guidance provides enhanced resolution in the lower wind speed categories and the overall (i.e., greater) number of wind speed classes as compared to those specified in table 1 of Safety Guide 23. The NRC staff finds that this approach is conservative and acceptable. Further, as indicated in Enclosure 1 of the LAR, this reflects the guidance in RIS-2006-04. This RIS is the basis for table 3 in RG 1.23.

The licensee used delta-T values between the nominal 46-m minus 10-m measurement levels to determine stability class. The NRC staff finds that this is an appropriate assumption because FHA releases were assumed to be at ground level due to structural wake effects (as opposed to elevated releases) per the respective modeling guidance. The listings of hourly temperature data, originally provided as Enclosure 6 of the LAR (ML24248A092), are in degrees Fahrenheit. However, stability class designations in table 1 of Revision 1 to RG 1.23 are given as degrees Centigrade per 100 m. After appropriate conversion of the given temperature measurement units and extrapolation to the necessary 100-m vertical distance interval, a spot check of the resulting stability class determinations showed them to be reasonably acceptable.

The POR used by the licensee for the dispersion analyses (i.e., from January 1, 2004, through December 31, 2013) was older than the recommended 10-year limit in the last paragraph of Section B (Discussion) of Revision 1 to RG 1.23. To evaluate the long-term representativeness of the 10-year POR, the NRC staff made a cursory check of several JFDs appearing in more recent annual radioactive effluent release reports (ARERRs) - that is, 2019, 2020, and 2023 (i.e., ML20115E357, ML21117A320, and ML24114A048, respectively). The staff recognizes that year-to-year variations will be present in the Met data and resulting JFDs. After accounting for the fact that these ARERRs list the JFDs on a quarterly rather than an annual basis and that the wind speed class breakdown is more similar to that in Safety Guide 23, the summaries still suggest that the POR used for this LAR reasonably represent current dispersion conditions at the plant site.

Finally, the data recovery for the 10-year POR, the JFDs in Enclosure 5 of the LAR and in the March 17, 2025, supplement, indicate that the joint (concurrent) recovery of the wind and stability model input parameters was well above the 90 percent criterion in Regulatory Position 5 of RG 1.23. Therefore, the NRC staff finds the Met data acceptable.

3.1.4.2 Evaluation of Offsite Atmospheric Dispersion Factors Modeling Analysis

The licensee performed a dispersion modeling analysis to estimate accident-related dispersion factors (X/Q values) at the EAB and the outer boundary of the LPZ. Results from the dispersion analysis provide direct inputs to dose calculations at these offsite receptor locations to which the public has access and are associated with potential releases due to an FHA. In turn, the FHA dose assessment is necessary to support revisions to several technical specifications, discussed elsewhere in this safety evaluation, as proposed in the LAR.

The licensee used the NRC-endorsed PAVAN dispersion model to calculate these X/Q values. The PAVAN model implements the guidance in RG 1.145, the associated model user's guidance in NUREG/CR-2858, and the technical basis document for RG 1.145 in NUREG/CR-2260. As indicated in the LAR, this model was not integrated with another code so additional documentation of consistency checks related to their adaptation was not necessary.

PAVAN model inputs are specifically listed in table 3.1.1-4 of Enclosure 1 to the LAR submittal. The NRC staff reviewed these inputs and other documentation related to the offsite dispersion modeling analysis also provided with the LAR. The information reviewed included:

- Enclosure 1, "Description and Assessment of Proposed Changes (ML24247A175);
- Enclosure 3, "Calculation of Atmospheric Dispersion Factors Exclusion Area Boundary & Low Population Zone for SQN Units 1 & 2" (ML24247A171);
- Enclosure 5, "Joint Frequency Distributions of Wind Speed and Direction 2004 to 2013 SQN Units 1 & 2" (ML24247A173); and
- Enclosure 6, "ARCON96 Files, Hourly Meteorological Data for 2002-2013, MET Data Files, and PAVAN Files" (ML24248A092).

Enclosure 6 of the LAR included three pairs of PAVAN model input and output files with designators of "e1", "e2", and "e3" as part of the file names. These designators represent EAB receptor distances of 556, 600, and 509 m, respectively. They correspond to three release zones evaluated by the licensee in order to determine the most conservative release point (highest X/Qs) for the EAB as described in Enclosure 1 under the heading "Offsite X/Q's". The NRC staff finds this to be an acceptably conservative approach. Distances to the outer boundary of the LPZ were the same (i.e., 4828 m) in all three pairs of these PAVAN input/output files. The NRC staff confirmed all distances to the EAB and LPZ were consistent with those given in the UFSAR.

The PAVAN model runs were configured to only account for enhanced building wake effects on plume dispersion. The building dimensions input to the model, that is, the containment building height (40.8 m) and the containment building minimum cross-sectional area (1632 square-meters) were cross-checked against the UFSAR and found to be acceptable. The NRC staff considers the approach of accounting for building wake effects in the dispersion analysis for FHA releases to be appropriate based on where the potential releases are expected to occur.

The NRC staff determined that the PAVAN model inputs listed in table 3.1.1-4 of Enclosure 1 to the LAR submittal were acceptable as discussed above. However, the staff also checked the model input files (provided in Enclosure 6 of the LAR) for proper formatting against the model users guidance in NUREG/CR-2858 and identified an error common to all three input files. As a result, a request for additional information (RAI) describing this matter was transmitted to the licensee on January 21, 2025 (ML25022A059).

The licensee confirmed the formatting error among its response of March 17, 2025, stating that "[t]his issue has been documented in the TVA corrective action program." The licensee's resolution included correcting the formatting error, determining the corrected X/Q values, revising the FHA analysis with the corrected X/Q values as necessary, and revising other calculations that used the X/Q values as necessary. The licensee also stated that "[t]he new X/Q analysis determined the values used as direct input in the FHA analysis were not impacted," and that "the FHA analysis provided in the original submittal remains valid."

The licensee's March 17, 2025, response also provided information that either supersedes or reiterates portions of the LAR submittal. In summary:

- Enclosure 2 of the RAI response replaces Enclosure 3 of the LAR submittal.
- Attachment 2 to Enclosure 2 of the RAI response contains the same JFDs provided in Enclosure 5 of the LAR submittal.
- Attachments 3 and 4 to Enclosure 2 of the RAI response replace the PAVAN output files provided in Enclosure 6 of the LAR submittal.
- The PAVAN input files in Enclosure 6 of the LAR submittal are superseded by the new X/Q analysis provided in Enclosure 2 of the RAI response.

Regarding these last two bullet items, the licensee indicated that the model output files are a direct function of the input files and that the input files were not resubmitted noting that Attachments 3 and 4 to Enclosure 2 of the RAI response each begin with input file data. The NRC staff agrees with this observation and considers it acceptable.

Further, the licensee mentioned two other minor differences between the PAVAN model output submitted in Enclosure 2 of the RAI response as compared to the model input and output files provided in Enclosure 3 of the original LAR submittal. Again, in summary:

- Only two model output files are included versus the three pairs of model input and output files in the original LAR submittal. The three pairs corresponded to each of the three EAB distances with each model run containing the same LPZ distance. The new dispersion analysis determines X/Qs for two EAB distances in one model run, and the second model run contains the third EAB distance and the LPZ distance.
- The revised model runs contain only twelve wind speed categories versus thirteen in the original model runs. The 13th category was originally input as "0", having no JFD input data, and was therefore removed from the two revised model runs.

The NRC staff reviewed these changes and considers them acceptable.

The NRC staff's confirmatory PAVAN model runs led to the RAI and included the relevant adjustments to evaluate any potential implications of the input file formatting error. The staff then compared its offsite X/Q output results for the EAB and LPZ against the corresponding X/Q values listed in the following:

- the revised output files of Attachments 3 and 4 to Enclosure 2 of the RAI response;
- the output presented in tables 2 through 5 of Enclosure 2 of the RAI response; and
- more importantly, the controlling (overall highest) offsite values at the EAB and LPZ as listed in table 3.1.1-1 and the highest X/Q values for the three EAB distances and the single LPZ distance as listed in table 3.1.1-5 of Enclosure 1 to the original LAR.

Finally, the NRC staff confirmed that the highest offsite X/Q at the EAB was associated with Release Zone 3 and a downwind distance of 509 m and that the higher of the sector-specific or the 5 percent overall site X/Qs from each of the model runs was selected consistent with Regulatory Position C.4 in RG 1.145. Therefore, the NRC staff concludes that the offsite X/Q dispersion modeling analysis for this LAR is acceptable.

3.1.4.3 Evaluation of Onsite Atmospheric Dispersion Factors Modeling Analysis

The licensee also performed a dispersion modeling analysis to estimate accident-related dispersion factors (X/Qs) at onsite receptor locations (i.e., the normal and emergency air intakes to the MCR). The MCR is common to SQN Units 1 and 2. Like the dispersion analysis at the offsite EAB and outer boundary of the LPZ, the results from this onsite dispersion modeling provide direct inputs to the dose calculations except that they are used to evaluate the dose criteria for control room habitability. As before, the calculation of potential accident-related releases due to an FHA are necessary to support revisions to several technical specifications, discussed elsewhere in this SE and as proposed in this LAR.

As described in Enclosure 1 of the LAR under the heading “Control Room X/Q’s,” “[t]he intakes to the Technical Support Center are the same as the MCR as it is part of the Main Control Room Habitability Zone.” Therefore, a separate onsite dispersion modeling analysis for the Technical Support Center is not necessary.

The licensee used the NRC-endorsed ARCON96 dispersion model to calculate these X/Q values. The ARCON96 model implements the guidance in RG 1.194 and the associated model user’s guidance in NUREG/CR-6331. As indicated in the LAR, this model was not integrated with another code so additional documentation of consistency checks related to their adaptation was not necessary.

ARCON96 model inputs are specifically listed in table 3.1.1-2 of Enclosure 1 to the LAR submittal for three potential release locations (i.e., the AB and SB vents for Units 1 and 2). The NRC staff reviewed these inputs and other documentation related to the onsite dispersion modeling analysis also provided with the original LAR submittal. The information reviewed included:

- Enclosure 1, “Description and Assessment of the Proposed Changes” (ML24247A175);
- Enclosure 2, “Calculation of Atmospheric Dispersion Factors - Control Room” (ML24247A174);
- Enclosure 4, “Marked-up Equipment Layout Drawings for SQN Units 1 & 2” (ML24247A172); and
- Enclosure 6, which contains the ARCON96 files, hourly meteorological data for 2002-2013, MET data files, and PAVAN files (ML24248A092).

Enclosure 6 included about a dozen pairs of ARCON96 model input and output files. As a set, they represent potential accident releases from several receptor and release point pairs. These model runs also assumed different building dimensions to calculate the varying effects of these different dimensions on the degree of atmospheric dispersion with an aim towards identifying the contributing release point(s) to the normal and emergency air intake receptors. These iterative model run files indicated that they were made in early August 2019.

In addition, Enclosure 6 included two ARCON96 output files designated as “TVA_083” and “TVA_187.” The filename labels “083” and “187” correspond to the respective directions from the normal air intake to the nearest release point on the AB vent as shown in figure 3-3 of Enclosure 2 to the LAR. These two dispersion model runs indicated that they were made in November 2014. All the model runs (i.e., from 2019 and 2014) utilized the same 10-year POR of hourly Met data from 2004 to 2013.

From among the 2019 iterative set of model runs and the 2014 pair of model runs, the overall highest X/Qs at the normal and emergency air intakes are associated with the “TVA_083” and “TVA_187” model runs. These X/Q values were due to potential releases from the AB vent and are listed in Enclosure 2 in several places (e.g., see table 5-1). Of these two values, the controlling (overall highest) X/Q was associated with the “TVA_187” model run (i.e., at the normal air intake). Enclosure 1 also includes a broader summary of the 2019 and 2014 accident-related modeling results (e.g., see table 3.1.1-3) listing the highest X/Qs by source and receptor.

The NRC staff observed that the ARCON96 model inputs listed in table 3.1.1-2 of Enclosure 1 to the LAR submittal appeared to be acceptable and notes that the inputs are echoed as part of the model output. However, the staff’s review of the “TVA_083” and “TVA_187” output files identified several discrepancies with portions of the documentation in Enclosure 2, dated December 2014, that the staff found are inconsistent with certain ARCON96 model input parameters and/or within that documentation itself and/or the ARCON96 modeling guidance in NUREG/CR-6331. As a result, an RAI describing this matter was included in the NRC staff’s January 21, 2025, request to the licensee (ML25022A059). The specific discrepancies are detailed in the referenced RAI.

The NRC staff acknowledged in its RAI that individually, some, or all of the identified items may just be inconsistencies in the indicated documentation. However, much of Enclosure 2 and the controlling model runs are 10 years old. As a result, the staff stated its concern that one or more of these discrepancies between the documentation and certain inputs to the ARCON96 dispersion modeling could result in the X/Qs and downstream dose calculations possibly being affected. The staff also noted that the RAI may not represent a complete list of the disconnects.

In its response of March 17, 2025, to this RAI, the licensee stated that “[t]his issue has been documented in the TVA corrective action program” (ML25076A261). The licensee further stated that, “TVA has reviewed the noted items and determined they are all editorial in nature. Therefore, there is no impact to the revised FHA dose assessment submitted as part of the LAR.”

Based on the above response and the NRC staff’s check of the ARCON96 model inputs in table 3.1.1-2 of Enclosure 1 and related documentation in the original LAR submittal that led to this RAI, the NRC staff finds these model inputs to be acceptable. Further, the staff compared its

ARCON96 confirmatory modeling results for the “083” and “187” scenarios to the corresponding licensee’s X/Qs listed in:

- table 3.1.1-3 of Enclosure 1 to the original LAR submittal; and
- sections 1.0, 5.0, and table 5-1 in Enclosure 2 of the March 17, 2025, RAI response;

and found them to be consistent for all averaging intervals. Therefore, the NRC staff finds that the onsite X/Q dispersion modeling analysis for this LAR is acceptable.

3.1.5 Confirmation of Input and Assumptions

3.1.5.1 Evaluation of Gap Release Fractions

Regulatory Guide 1.183 provides guidance on AST implementation. Footnote 11 in section 3.2 of RG 1.183, Revision 0, notes that the non-loss-of-coolant accident (LOCA) fuel rod gap fractions listed in table 3 of the RG have been found acceptable for LWR fuel with peak rod-average burnup of 62 gigawatt-days per metric ton of uranium (GWd/MTU), provided that the maximum linear heat generation rate (LHGR) does not exceed 6.3 kilowatt per foot at burnups greater than 54 GWD/MTU.

The licensee utilizes RG 1.183, Revision 0, table 3 fuel-cladding gap release fractions, as shown in table 3.1.1-1 of Enclosure 1 of the LAR, although the LAR does not list the release fraction for alkali metals. Even though table 3 of RG 1.183 specifies a gap release fraction for alkali metals of 12 percent, FHA analyses typically assume that no alkali metals are released because particulates have essentially an infinite partition factor in water, which is consistent with Regulatory Position 3 of Appendix B of RG 1.183, Revision 0. Additionally, in response to NRC Request for Confirmation of Information Question 1 (ML25013A334), the licensee confirmed that it will meet the LHGR and burnup range of applicability stated in RG 1.183, Revision 0, footnote 11.

Since the licensee uses non-LOCA gap release fractions consistent with RG 1.183, Revision 0, and will ensure that it will meet the range of applicability of those gap release fractions, as defined in RG 1.183, Revision 0, footnote 11, the NRC staff finds the gap release fractions employed for the revised FHA to be acceptable.

3.1.5.2 Evaluation of Fraction of Fission Products Retained in Water

The fission product inventory in the gap of the damaged fuel rods is released to the overlaying water in either the reactor cavity or spent fuel pool and retained in this water depending on the radionuclide chemical form and speciation. To ascertain the radiological source term that escapes the water column and is thus available for environmental transport the conservative assumptions regarding chemical form and speciation must be made. Following the guidance in RG 1.183, Revision 0, Appendix B, Regulatory Position 1.3, the licensee assumes: (1) that the chemical form of radioiodine released from the fuel to the overlaying water consists of 95 percent cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide, (2) the CsI released from the fuel completely dissociates in the overlaying water, and (3) because of the low pH in the overlaying water, the CsI re-evolves and releases elemental iodine. This results in a final iodine distribution of 99.85 percent elemental iodine and 0.15 percent organic iodine.

As corrected by item 8 of RIS-2006-05, RG 1.183, Revision 0, Appendix B, Regulatory Position 2, should read as follows:

> If the depth of water above the damaged fuel is 23 feet or greater, the decontamination Factors (DF) for the elemental and organic species are 285 and 1, respectively, giving an overall effective decontamination factor of 200 (i.e., 99.5% of the total iodine released from the damaged rods is retained by the water). This difference in decontamination factors for elemental (99.85%) and organic iodine (0.15%) species results in the iodine above the water being composed of 70% elemental and 30% organic species.

The minimum water levels in the refueling cavity and the spent fuel pool are 23 feet by TS 3.9.7 and TS 3.7.13, respectively. These TS ensure that the water level above the damaged fuel assembly is sufficient to support the analytical assumptions regarding the decontamination factors and iodine speciation in RG.1.183, Appendix B.

In accordance with RG 1.183, Revision 0, Appendix B, Regulatory Position 3, the licensee did not credit decontamination for the noble gas constituents and assumed that 100 percent of the noble gas activity is released from the water covering the fuel. Additionally, the licensee assumed that the release to the overlaying water and the chemical redistribution of iodine occurs instantaneously. Therefore, the NRC staff confirms that the licensee's assumptions to determine the release of the source term are unchanged and remain consistent with the guidance in RG 1.183, Revision 0.

3.1.6 Evaluation of Dose Analysis Results

Table 3.1.2-1 provides the results of the licensee's dose calculations. The NRC staff compared these results to the dose acceptance criteria for the FHA in table 6 of RG 1.183, Revision 0. For the EAB, the licensee calculated a dose of 3.72 rem TEDE which is within the acceptance criterion of 6.3 rem TEDE. For the LPZ, the licensee calculated a dose of 0.32 rem TEDE which is within the acceptance criterion of 6.3 rem TEDE. For the control room, the licensee calculated a dose of 0.59 rem TEDE which is within the acceptance criterion of 5 rem TEDE.

The NRC staff performed confirmatory analyses using Version 5.0.3 of the RADionuclide Transport, Removal, and Dose (RADTRAD) computer code and obtained results similar to the licensee's. Therefore, the NRC staff finds the licensee's results to be acceptable.

3.2 Evaluation of Changes to Technical Specifications

The NRC staff reviewed the acceptability of the proposed changes to the TSs by evaluating whether, among other things, the changes provide reasonable assurance that public health and safety will be protected. The NRC staff also verified that the proposed changes to the TSs will continue to ensure that the LCOs will be met, or that the TS is no longer required.

3.2.1 Evaluation of Proposed Modification of TS 3.3.6

LCO 3.3.6 requires the CVI instrumentation for each function in TS table 3.3.6-1 to be OPERABLE as specified in that table. The licensee proposed to remove TS ACTION B and SPECIFIED CONDITION (a) in TS table 3.3.6-1, which are applicable only during movement of irradiated fuel assemblies within containment. These actions were based on the requirement to automatically isolate containment in the event of an FHA during shutdown. The licensee also proposed to delete the frequency of “Within 100 hours prior to the start of movement of irradiated fuel” and the logical connector “AND” from SRs 3.3.6.4 and 3.3.6.6.

As discussed in section 3.1 above, the revised FHA analysis does not credit containment penetration closure to limit releases from the FHA occurring inside containment. Additionally, in the revised FHA, the licensee assumed only a 70-hour delay after shutdown before core alterations begin, which will result in a larger source term and, thus larger doses.

The NRC staff reviewed the proposed deletion of ACTION B and SPECIFIED CONDITION (a) and finds that because containment isolation is not credited to limit releases from containment due to an FHA inside containment, requiring containment ventilation isolation instrumentation for each function in TS table 3.3.6-1 to be OPERABLE while moving irradiated fuel assemblies within containment does not meet any of the four criteria of 10 CFR 50.36(c)(2)(ii). Therefore, the staff finds it acceptable to delete TS 3.3.6 ACTION B and SPECIFIED CONDITION (a) in table 3.3.6-1. The staff also finds that since the revised FHA assumed only a 70-hour delay prior to commencing core alterations, the proposed removal of “Within 100 hours prior to the start of movement of irradiated fuel” and the logical connector “AND” from the FREQUENCY for SRs 3.3.6.4 and 3.3.6.6 is acceptable. The staff notes the licensee will still be required to perform a Channel Operational Test (COT) (SR 3.3.6.4) and a Trip Actuating Device Operational Test (TADOT) (SR 3.3.6.6) for the CVI instrumentation in accordance with the plant’s existing Surveillance Frequency Control Program thus ensuring that LCO 3.3.6 will continue to be met. Therefore, the staff finds that SR 3.3.6.4 and SR 3.3.6.6, as revised, will continue to meet 10 CFR 50.36(c)(3).

3.2.2 Evaluation of Proposed Deletion of TS 3.9.4

TS 3.9.4 requires containment penetrations, including the equipment hatch, airlock doors, and penetrations providing direct access to the outside atmosphere to be closed or capable of being closed during movement of irradiated fuel assemblies within containment. This requirement was based on the previous FHA analysis which credited containment isolation capability to limit dose releases. The licensee has revised the FHA, and the NRC staff evaluation of the revised FHA is discussed in section 3.1 of this safety evaluation. The licensee discussed the containment building penetrations and the paths the releases would take. As noted in the LAR, the containment building equipment hatch and airlock doors open into the AB. If an FHA were to occur inside containment with the equipment hatch or airlock doors open, the releases would flow into the AB. Based on the revised FHA, no path would produce a release larger than the release from the FHA occurring outside containment, i.e., in the AB, which is the bounding case for the FHA. Because of this, the revised FHA analysis does not credit containment isolation to mitigate an FHA inside containment.

The NRC staff reviewed the proposed deletion of TS 3.9.4 and finds that requiring closure of the containment building equipment hatch and containment airlock doors and penetrations during movement of irradiated fuel assemblies does not meet any of the four criteria of 10 CFR 50.36(c)(2)(ii). Therefore, the NRC staff finds it acceptable to delete TS 3.9.4, “Containment Penetrations.”

3.3 Technical Conclusion

The NRC staff reviewed the licensee’s revised FHA analysis and the proposed changes to TS 3.3.6 and TS 3.9.4 and determined that, based on the revised FHA, the proposed changes are consistent with 10 CFR 50.36(c)(2) and (c)(3), and therefore, are acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Tennessee State official was notified of the proposed issuance of the amendment on April 15, 2025. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves 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 previously issued a proposed finding that the amendment involves no significant hazards consideration published in the *Federal Register* on October 29, 2024 (89 FR 85996), and 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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

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Date: June 23, 2025

SUBJECT: SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2 – ISSUANCE OF AMENDMENT NOS. 372 AND 366 REGARDING REVISION OF THE FUEL HANDLING ACCIDENT ANALYSIS, DELETION OF TECHNICAL SPECIFICATION 3.9.4, AND REVISION OF TECHNICAL SPECIFICATION 3.3.6 (EPID L-2024-LLA-0117) DATED JUNE 23, 2025

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