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

September 29, 2016

Mr. Joseph W. Shea
Vice President, Nuclear Licensing
Tennessee Valley Authority
1101 Market Street, LP 3R-C
Chattanooga, TN 37402-2801

SUBJECT: SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS TO REVISE TECHNICAL SPECIFICATION FOR ESSENTIAL RAW COOLING WATER SYSTEM ALLOWED COMPLETION TIME (CAC NOS. MF7450 AND MF7451)

Dear Mr. Shea:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 336 to Renewed Facility Operating License (RFOL) No. DPR-77, and Amendment No. 329 to RFOL No. DPR-79, for the Sequoyah Nuclear Plant, Units 1 and 2, respectively. The amendments consist of changes to the RFOLs and Technical Specifications (TSs) in response to your application dated March 11, 2016, as supplemented by letters dated May 31, and July 22, 2016.

The proposed changes revise the Improved Standard TSs to add a new Condition A to TS 3.7.8, Essential Raw Cooling Water (ERCW) System, to extend the allowed completion time to restore ERCW System train to OPERABLE status from 72 hours to 7 days for planned maintenance when the opposite unit is defueled or in Mode 6 following defueled under certain restrictions.

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

Sincerely,

Andrew Hon, 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. 336 to RFOL No. DPR-77
2. Amendment No. 329 to RFOL No. DPR-79
3. Safety Evaluation

cc w/enclosures: Distribution via 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. 336
Renewed License No. DPR-77

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

   A. The application for amendment by the Tennessee Valley Authority (the licensee), dated March 11, 2016, as supplemented by letters dated May 31, and July 22, 2016, 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, as amended; 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. 336 are hereby incorporated into this 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 no later than 30 days from the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Jeanne A. Dion

Jeanne A. Dion, Acting 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: September 29, 2016

ATTACHMENT TO LICENSE AMENDMENT NO. 336

SEQUOYAH NUCLEAR PLANT, UNIT 1

RENEWED FACILITY OPERATING LICENSE NO. DPR-77

DOCKET NO. 50-327

Replace the following page of the Renewed Facility Operating License with the attached revised page. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

Remove	Insert
3	3

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

Remove	Insert
3.7.8-1	3.7.8-1
3.7.8-2	3.7.8-2
- - -	3.7.8-3

(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. 336 are hereby incorporated into this 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;

3.7 PLANT SYSTEMS

3.7.8 Essential Raw Cooling Water (ERCW) System

LCO 3.7.8 Two ERCW System trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
-----------NOTES----------- 1. Only applicable when Unit 2 is defueled or in MODE 6 following defueled with Unit 2 refueling water cavity level ≥ 23 ft. above top of reactor vessel flange. 2. Only applicable when Ultimate Heat Sink temperature is ≤ 79°F ------------------------------- A. One ERCW System train inoperable for planned Shutdown Board maintenance.	A.1 ------------NOTES-------------- 1. Enter applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources - Operating," for emergency diesel generator made inoperable by ERCW System. 2. Enter applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," for residual heat removal loops made inoperable by ERCW System. ------------------------------- Restore ERCW System train to OPERABLE status. AND A.2 Verify Ultimate Heat Sink temperature is ≤ 79°F.	 7 days 1 hour AND Once every 8 hours thereafter

B. One ERCW System train inoperable for reasons other than Condition A.	B.1 --------------NOTES------------ 1. Enter applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources - Operating," for emergency diesel generator made inoperable by ERCW System. 2. Enter applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," for residual heat removal loops made inoperable by ERCW System. ----------------------------------- Restore ERCW System train to OPERABLE status.	72 hours
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 3. <u>AND</u> C.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.8.1	------------------------------NOTE--------------------------- Isolation of ERCW System flow to individual components does not render the ERCW System inoperable. ---	
	Verify each ERCW System manual, power operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.	In accordance with the Surveillance Frequency Control Program
SR 3.7.8.2	Verify each ERCW System automatic valve in the flow path servicing safety related equipment that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.7.8.3	Verify each ERCW System pump starts automatically on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program

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TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-328

SEQUOYAH NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 329
Renewed License No. DPR-79

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

   A. The application for amendment by the Tennessee Valley Authority (the licensee), dated March 11, 2016, as supplemented by letters dated May 31, and July 22, 2016, 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, as amended; 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. 329, are hereby incorporated into the 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 no later than 30 days from the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Jeanne A. Dion, Acting 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: September 29, 2016

ATTACHMENT TO LICENSE AMENDMENT NO. 329

SEQUOYAH NUCLEAR PLANT, UNIT 2

RENEWED FACILITY OPERATING LICENSE NO. DPR-79

DOCKET NO. 50-328

Replace the following page of the Renewed Facility Operating License with the attached revised page. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

Remove	Insert
3	3

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

Remove	Insert
3.7.8-1	3.7.8-1
3.7.8-2	3.7.8-2
- - -	3.7.8-3

(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. 329 are hereby incorporated into this 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;

3.7 PLANT SYSTEMS

3.7.8 Essential Raw Cooling Water (ERCW) System

LCO 3.7.8 Two ERCW System trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
--------------NOTES------------- 1. Only applicable when Unit 1 is defueled or in MODE 6 following defueled with Unit 1 refueling water cavity level ≥ 23 ft. above top of reactor vessel flange. 2. Only applicable when Ultimate Heat Sink temperature is ≤ 79°F ------------------------------------- A. One ERCW System train inoperable for planned Shutdown Board maintenance. .	A.1 --------------NOTES------------ 1. Enter applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources - Operating," for emergency diesel generator made inoperable by ERCW System. 2. Enter applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," for residual heat removal loops made inoperable by ERCW System. ------------------------------------- Restore ERCW System train to OPERABLE status.	 7 days
	AND	
	A.2 Verify Ultimate Heat Sink temperature is ≤ 79°F	1 hour AND Once per 8 hours thereafter

B. One ERCW System train inoperable for reasons other than Condition A.	B.1 --------------NOTES------------ 1. Enter applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources - Operating," for emergency diesel generator made inoperable by ERCW System. 2. Enter applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," for residual heat removal loops made inoperable by ERCW System. ------------------------------------ Restore ERCW System train to OPERABLE status.	72 hours
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 3. <u>AND</u> C.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.7.8.1	------------------------------NOTE--------------------------- Isolation of ERCW System flow to individual components does not render the ERCW System inoperable. --- Verify each ERCW System manual, power operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.	In accordance with the Surveillance Frequency Control Program
SR 3.7.8.2	Verify each ERCW System automatic valve in the flow path servicing safety related equipment that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.7.8.3	Verify each ERCW System pump starts automatically on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program

UNITED STATES
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WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 336 AND 329 TO

RENEWED FACILITY OPERATING LICENSE NOS. DPR-77 AND DPR-79

TENNESSEE VALLEY AUTHORITY

SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-327 AND 50-328

1.0 INTRODUCTION

By letter dated March 11, 2016, (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16071A333), and supplemented by letters dated May 31, 2016 (Accession No. ML16153A071) and July 22, 2016 (Accession No. ML16207A205), Tennessee Valley Authority (TVA, the licensee), requested a change to the Technical Specifications (TSs) to Sequoyah Nuclear Plant (SQN) Units 1 and 2.

The amendment proposes to change TS 3.7.8, "Essential Raw Cooling Water System," to extend the allowed completion time to restore one Essential Raw Cooling Water (ERCW) System train to operable status from 72 hours to 7 days for planned shutdown board maintenance for SQN. These changes are needed to facilitate cleaning and inspection of the 6.9 kilovolt (kV) shutdown boards and associated 480 Volt (V) shutdown boards without requiring a dual unit shutdown.

The SQN ERCW System supports both Unit 1 and Unit 2 and consists of a two-train system (A Train and B Train) with each train having the capability to provide the maximum required cooling water requirement for both units under any credible plant condition. The ERCW System has a total of eight pumps (four pumps per A Train and four pumps per B Train), two traveling water screens per train (four total), two screen wash pumps per train (four total), and two supply strainers per train (four total) located within the ERCW pumping station.

For the SQN ERCW System, each 6.9 kV shutdown board is aligned to power two ERCW pumps. A selector switch determines which of the two associated ERCW pumps the 6.9 kV shutdown board is supporting for operability. Current SQN TSs require two ERCW pumps to be operable per train, with two trains required for each unit. Consequently, this requires all four Emergency Diesel Generator (EDG) supported ERCW pumps to be operable, and thus all four 6.9 kV shutdown boards must be operable to support the ERCW pump TS requirement.

The requested changes are needed because taking one 6.9 kV shutdown board out of service to perform maintenance affects both units. An inoperable 6.9 kV shutdown board renders one of two required ERCW pumps per train inoperable. In this condition the current TS allows 72 hours to return the ERCW pump to an operable status. However, 72 hours is not adequate to safely clean and inspect a shutdown board and perform corrective maintenance. Therefore, the completion time to restore an ERCW System train to operable status needs to be extended from 72 hours to 7 days to facilitate safely cleaning and inspecting a shutdown board and performing corrective maintenance without having to shut down both units unnecessarily. The proposed extended time to restore an inoperable ERCW train while performing planned maintenance on a single shutdown board applies under conditions that allow the single operable pump in the affected train to provide adequate ERCW flow for its essential safety functions. The licensee has analyzed prerequisite valve alignments that must be configured to ensure safe plant operation during the extended completion time for shutdown board maintenance. These prerequisite valve alignments are described in detail in the original License Amendment Request (LAR), and SQN emphasized in its July 22, 2016, supplemental letter that the alignments must be as described in the LAR when maintenance is performed. The valve alignments are described in detail in the SQN Updated Final Safety Analysis Report (UFSAR) Section 9.2.2.2, which is referenced in the proposed TS 3.7.8 Bases.

There will be no physical modification to the plant based on this proposed LAR for the SQN ERCW System.

After the evaluation, the Nuclear Regulatory Commission (NRC) staff found that:

- when the ERCW system is aligned to the prerequisite configuration and the Ultimate Heat Sink (UHS) temperature satisfies the maximum temperature limit, the unaffected ERCW train would be capable of performing the ERCW safety functions during the maintenance period, and the affected train would provide redundant capability when the opposite unit has negligible heat removal requirements. With this additional capability, the ERCW safety function could be satisfied considering a single failure that disables the unaffected train, and

- the licensee adequately described the expected duration of the shutdown board maintenance as well as the equipment protection plans to maintain defense-in-depth during the 7-day completion time. Thus, the extension of the completion time from 72 hours to 7 days is justified.

Based on the regulatory and technical review above, the NRC staff found the licensee's proposed changes to TS 3.7.8 to be acceptable. This details of NRC staff's evaluation is documented below.

2.0 REGULATORY EVALUATION

2.1 System Descriptions

Related to the proposed SQN TS 3.7.8 changes, two SQN systems are described below: The UHS and ERCW System.

2.1.1 Ultimate Heat Sink

The UHS for SQN is the Tennessee River, which provides an effectively infinite heat sink. The UHS is designed to perform two principal safety functions: (1) dissipation of residual and auxiliary heat after reactor shutdown, and (2) dissipation of residual and auxiliary heat after an accident. The UHS achieves these functions through the ERCW System by providing the heat sink function for the ERCW System.

In accordance with TS 3.7.9, the UHS shall be operable in Modes 1, 2, 3, and 4 with TS Surveillance Requirements to 'verify average ERCW supply header water temperature is $\leq$ 87° F' and 'verify water level of UHS is $\geq$ 674 ft. mean seal level USGS [US Geological Survey] datum'. If the UHS function cannot be satisfied, unit shutdown is required in accordance with the associated action statements and completion times.

2.1.2 ERCW System

The ERCW System at SQN is shared between the two units. The ERCW System is a two-train system with each train having the capability to provide the required cooling water for Design-Basis Accident (DBA) mitigation or shutdown and cooldown of one unit while the other unit is safely shutdown to hot standby conditions. These ERCW System trains are sufficiently independent to guarantee the availability of at least one train at any time. Based on these analyses, sharing of the ERCW System by the two nuclear units does not introduce factors that prevent the ERCW System from performing its required function for plant design basis conditions.

Each ERCW train consists of two headers. Each ERCW header consists of a pumping station with one traveling water screen, two pumps, and one supply strainer; piping and valves to deliver the cooling water to components served by the header; and piping and valves for cross-connects between headers in each train. The minimum combined ERCW flow requirements are satisfied by a single operating ERCW pump per header when the ERCW System is aligned in its normal configuration. The pumps associated with a header are powered from its associated 6.9 kV shutdown board, and one pump in each header can be aligned to be repowered by the EDG associated with that shutdown board.

The ERCW System is designed to supply water to the following essential components:

- Component Cooling Heat Exchangers (HXs)
- Containment Spray (CS) HXs
- EDGs HXs
- Emergency makeup for steam generators via the Auxiliary Feedwater (AFW) System
- Emergency makeup for Component Cooling System (CCS)
- Control Building Air-Conditioning Systems
- Auxiliary Building space coolers (for engineered safety features (ESFs) equipment)
- Containment Ventilation System coolers
- Auxiliary Control Air (ACA) compressors
- Reactor Coolant Pump (RCP) motor coolers
- Control Rod Drive (CRD) ventilation coolers

A full description of ERCW system is provided in SQN UFSAR Section 9.2.2.

2.2 Proposed TS Changes

Changes to SQN Unit 1 and Unit 2 TS 3.7.8, ERCW System are proposed to add a new Condition A to TS 3.7.8, "Essential Raw Cooling Water System." Specifically, the changes extend the allowed completion time to restore one inoperable ERCW System train to operable status from 72 hours to 7 days when the train is inoperable for planned maintenance on one shutdown board and conditions have been established to allow the remaining functional portions of the affected train to perform the safety function of the train. These conditions are:

- the opposite unit is defueled or in Mode 6 following defueled with refueling water cavity level ≥ 23 feet above the reactor vessel flange
- UHS temperature is ≤ 79 degrees Fahrenheit (°F)
- the affected ERCW train has been configured in the prerequisite component alignment for shutdown board maintenance
- additional alignment changes are implemented consistent with procedures after the occurrence of a DBA

The first two conditions are notes added to the applicability of the proposed new action. The remaining two conditions are specified in proposed revisions to Section 9.2.2.2 of the SQN UFSAR.

The required action for the proposed Condition A specifies restoration of the affected ERCW train to operable status with a completion time of 7 days. This required action is modified by two notes that require entry into applicable Conditions and Required Actions of Limiting Condition for Operation (LCO) 3.8.1, "AC Sources - Operating," for the EDG and LCO 3.4.6, "RCS Loops - MODE 4," for Residual Heat Removal (RHR) loops made inoperable by the ERCW system condition during shutdown board maintenance. These notes are an exception to LCO 3.0.6 and ensure the proper actions are taken for the affected components in these systems.

2.3 Regulatory Criteria

The proposed license amendment involves a change to the TSs and change to the facility configuration that affects the safety analyses. The NRC staff reviews the changes for compliance with applicable regulations and conformance with associated regulatory guidance.

Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.36 requires that each Operating License issued by the Commission contain TSs that include limiting conditions for operation, which are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When an LCO of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the TSs until the condition can be met.

Guidance for staff review of TSs is contained in Section 16.0, "Technical Specifications," of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light Water-Reactor] Edition." The NRC staff has prepared Standard Technical Specifications (STSs) for each of the LWR nuclear steam supply systems and associated balance–of–plant equipment systems. The guidance specifies that the staff review whether content and format are consistent with the applicable STSs. Where TS provisions depart from the reference TSs, the staff determines whether proposed differences are justified by uniqueness in plant design or other considerations.

The applicable STSs for SQN are contained in NUREG-1431, "Standard Technical Specifications - Westinghouse Plants," Revision 4.0. The completion time allowed by STS 3.7.8, "Service Water System (SWS)," to restore one inoperable train of the SWS to service is 72 hours. The 72 hour completion time is based on the capabilities provided by the operable train and the low probability of a DBA occurring during this time period.

The plant configuration changes associated with maintenance activities can affect the operational safety of the facility. To ensure an adequate level of safety is maintained, 10 CFR 50.65(a)(4) requires licensees to assess and manage the increase in risk that may result from the proposed maintenance activities. The scope of this assessment may be limited to Structure, System, and Components (SSCs) that a risk-informed evaluation process has shown to be significant to public health and safety.

In Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision 3 (ADAMS Accession No. ML113610098), the staff endorsed Nuclear Management and Resources Council (NUMARC) 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision 4A (ADAMS Accession No. ML11116A198), as providing methods acceptable to the staff for implementing the requirements of 10 CFR 50.65.

These guidelines state that the required assessment should consider the following:

- TSs requirements
- The degree of redundancy available for performance of the safety function(s) served by the out-of-service SSC
- The duration of the out-of-service or testing condition
- The likelihood of an initiating event or accident that would require the performance of the affected safety function
- The likelihood that the maintenance activity will significantly increase the frequency of a risk-significant initiating event
- Component and system dependencies that are affected
- Significant performance issues for the in-service redundant SSCs

Normal work controls are adequate to manage nominal increases in risk associated with maintenance activities. Activities to manage more significant increases in risk associated with maintenance activities include measures to increase risk awareness; actions that reduce the duration of maintenance; and actions to minimize the magnitude of the risk increase.

3.0 TECHNICAL EVALUATION

3.1 Evaluation

With either Unit 1 or Unit 2 in Mode 1, 2, 3, or 4, TS 3.7.8 requires two ERCW trains to be operable. The SQN TS Bases state that the operability of the ERCW System ensures that sufficient cooling capacity is available for continued operation of safety related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident conditions within acceptable limits.

To evaluate the capability of an inoperable ERCW train with one ERCW pump in operation, the licensee modeled the ERCW system with one unit in Mode 5, 6, or defueled using a Multiflow Model. Because one unit is in a shutdown condition, several of the shutdown unit's ESF loads are not required and can be isolated. Accordingly, additional conditions were modeled that isolated various ERCW loads on the unit that was shutdown. Isolation of these unnecessary loads with the header crossties open increases the available ERCW supply to the required ERCW loads of the operating unit.

To further develop necessary conditions for accident mitigation using an ERCW train with a single operating pump, the minimum flow rates necessary to remove the design accident heat load were developed in various calculations. The design required flow rate for each safety related ERCW heat load is based on an 87 °F average UHS temperature (i.e., the TS maximum ERCW supply temperature). These various calculations were also used to determine the maximum permissible UHS temperature that would remove the required heat under design-basis conditions at flow rates less than the design required flow rate for these components. For flow rates less than the design required flow rate, a curve fit equation was developed for each of the safety related components. The curve fit equation provided the maximum ERCW temperature that would remove the required heat load with a flow rate of less than the design required ERCW flow rate. The lowest ERCW supply temperature for the ERCW pump outages was used as the maximum UHS TS temperature limit for the proposed extended completion time.

The NRC staff has reviewed the LAR with supplemental letter (ADAMS accession No. ML16153A071) to resolve NRC's Request for Additional Information (RAI). The staff's review and evaluation is shown below and is divided into four subparts including:

1. Determination of acceptability of available ERCW flowrates
2. Limiting UHS temperature determination
3. Equipment environmental qualification issue
4. Generic Letter (GL) 89-13 program
5. TS 3.7.8 proposed completion time of 7 days

3.1.1 Determination of acceptability of available ERCW flowrates

The licensee stated in the LAR that the analysis compared the available flow under specific plant conditions, referred to as cases (e.g., one ERCW pump per train operation) to the required flow. If the available flow rate is less than the required flow rate for 87 °F, then curve fit equations are applied to determine the applicable limitations on the ERCW supply header water temperature.

To determine the available flow for the specific plant condition of concern, the ERCW Flow Balanced Hydraulic Model (Multiflow Model) was configured for several different cases. To determine the available ERCW flow when only one ERCW pump per train is in operation, the Multiflow Model was configured in the various cases that describe alignments that may occur during single unit shutdowns, and would support an extended Completion Time for one ERCW System train to support cleaning shutdown boards. Specifically, this would occur because of the removal of the two ERCW pumps associated with an out of service 6.9 kV shutdown board, with one unit shutdown, and various components with their ERCW supply isolated.

The ERCW flow values determined in the Multiflow hydraulic analysis had a 5 percent uncertainty subtracted from each calculated flow value to account for the measurement and analysis uncertainties. The 5 percent uncertainty was accepted by the NRC in a previously approved license amendment. The 5 percent subtraction bounds the measurement and analysis uncertainties.

In addition to the margin described above, valve leakage of up to 100 gallons per minute (gpm) was determined to not affect the results of the Multiflow analysis by the licensee.

The degraded pump curves that are used in the hydraulic analysis are also the basis for the ERCW pump American Society of Mechanical Engineers (ASME) Section XI acceptance criteria. The ASME Section XI pump test procedures use the hydraulic calculation as the source of the pump test acceptance criteria. The pump test procedure acceptance criteria is the model pump curve, at the appropriate developed head, except where the Section XI allowable ranges are more restrictive.

The four cases in the LAR Tables summarized the analysis determining the restriction necessary when only one ERCW pump per train is operating. If the 5 percent reduced flow rate is less than the component's design required flow rate, then a temperature limit is listed. If no temperature limit is listed, the 5 percent reduced flow rate is greater than the component's design required flow rate. Thus, the current average ERCW supply header water temperature TS limit of 87 °F provides adequate heat removal capability.

The analysis for determining the acceptability of available ERCW flow rates was done by comparing the reconfigured available flow rate to the design required flow rate. If the reconfigured available flow rate is less than the design required flow rate, the curve fit equations are applied to determine the applicable limitation on the ERCW supply temperature.

In the submittal, the licensee used Tables 3.2-2, 3.2-3 and 3.2-4 to present the results of the Multiflow analysis associated with four cases. These tables are organized by component, listing the associated design required flow rates, Multiflow analysis results, and any required temperature limitations less than 87 °F. The first column identifies the component that relies on the ERCW System as a heat sink. The second column is labeled "Required Flow," and lists the ERCW System design required flow rate, in gpm, at the TS limit of 87 °F. The next three columns are duplicated in each ERCW System case. These duplicate columns are grouped by outage designation. The first of the three columns is labeled "Multiflow Results," the second of the three columns is labeled "minus 5%," and the third of the three columns is labeled "Temp Limit." Column 1 of 3 lists the reconfigured available flow rate, in gpm, determined by the Multiflow Model when aligned as identified in LAR Table 3.2-1, and a postulated loss-of-coolant accident (LOCA) is occurring on the designated unit. Column 2 of 3 ("minus 5%") is the value listed in Column 1 of 3 minus 5 percent. Column 3 of 3 ("Temp Limit") lists the reduced ERCW System supply temperature limit required because the Column 2 of 3 flow rate (reconfigured available flow rate minus 5 percent) is less than the component's design required flow rate listed in the "Required Flow" column. This temperature limitation was determined using the reconfigured available flow rate minus 5 percent, and the curve fit equation for the specific component that was previously developed. The last row in Tables 3.2-2 through 3.2-4 lists the maximum temperature limit determined for each ERCW System pump outage case.

In determining the limiting conditions for examining the ERCW System capabilities when only one ERCW pump is in operation, two design basis accidents were reviewed: a LOCA and a Main Steam Line Break (MSLB) inside containment. These accident scenarios were examined to support removing one 6.9 kV shutdown board from service, along with its associated EDG, to perform maintenance on the 6.9 kV shutdown board and its associated 480 V boards.

With both Unit 1 and Unit 2 in Mode 1, 2, 3, or 4, TS 3.7.8 requires two ERCW trains to be operable. The SQN TS Bases state that the OPERABILITY of the ERCW System ensures that sufficient cooling capacity is available for continued operation of safety related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident conditions within acceptable limits. The assumption used is that two ERCW pumps will operate in one train during the hypothetical combined accident and loss of normal power, assuming one of the other train's EDGs fail.

Once the initial conditions of one unit in Mode 5, 6, or defueled was modeled in the Multiflow Model, additional conditions were established that isolated various ERCW loads on the unit that was shutdown. Because one unit is in a shutdown condition, several of the shutdown unit's ESF loads are not required and can be isolated. Isolation of these unnecessary loads with the header crossties open increases the available ERCW supply to the required ERCW loads of the operating unit. LAR Table 3.2-1 provides details of the initial conditions established for the Multiflow Model runs. The results of the calculations for only one ERCW pump in service are contained in LAR Tables. For each outage case, the ERCW Multiflow Model is configured as described in LAR Table 3.2-1

Four outage alignment cases were run. Two of the alignment cases, Outages 1 and 3, were run based on postulating that the operating unit has a LOCA. The other two alignment cases, Outages 2 and 4, were run based on postulating the operating unit has an MSLB inside containment. Outages 1 and 3 were used to determine whether all ESF components would receive sufficient ERCW flow rate to remove the component heat loads. Examples of Outage 1 are further described in the following table below.

One Pump per Loop ERCW Operations Case Description (Outage 1a example)

Normal Power Operations/ ERCW Pump Lineup	Normal Power Operations/ CCW Pump Lineup	ERCW Pre-Condition with 2A loads isolated	Failure with Unit 1 LOCA, Unit 2 Outage	Results
A Train: J-A, Q-A, K-A, R-A Operable	1A-A, 2A-A, 1A2-A	2A shutdown board cleaning A Train: J-A, Q-A Operable K-A, R-A inoperable	1A bus failure A Train: No operable ERCW/CCS pumps	2A shutdown board cleaning plus 1A bus failure A Train: No operable ERCW/CCS pumps
B Train: L-B, N-B, M-B, P-B Operable	1B-B, 2B-B, 2B2-B	B Train: L-B, N-B, M-B, P-B Operable	B Train: L-B, N-B, M-B, P-B Operable	The B CCS, with any one of its three pumps, powered by either the 1B or 2B SDB, fulfills the ERCW/CCS safety function

One Pump per Loop ERCW Operations Case Description (Outage 1b example)

Normal Power Operations/ ERCW Pump Lineup	Normal Power Operations/ CCW Pump Lineup	ERCW Pre-Condition with 2B loads isolated	Failure with Unit 1 LOCA, Unit 2 Outage	Results
A Train: J-A, Q-A, K-A, R-A Operable	1A-A, 2A-A, 1A2-A	A Train: J-A, Q-A Operable	A Train: J-A, Q-A, K-A, R-A Operable	The 1A and 2A CCS loops, with the pumps powered by the 1A and 2A SDB respectively, fulfills the CCS safety function
B Train: L-B, N-B, M-B, P-B Operable	1B-B, 2B-B, 2B2-B	2B shutdown board cleaning B Train: L-B, N-B, Operable. M-B, P-B Inoperable	1B bus failure B Train: No operable ERCW/CCS pumps	2B shutdown board cleaning plus 1B bus failure B Train: No operable ERCW/CCS pumps

LAR Outages 2 and 4 were used to determine if the Lower Containment Vent Coolers (LCCs) received sufficient flow rate to remove the heat load assumed for equipment qualification of certain equipment located in the lower containment compartment.

As stated above by TVA, addition to the margin described above, valve leakage of up to 100 gpm was determined to not affect the results of the Multiflow analysis. The staff sent an RAI, RAI Balance of Plant Branch (SBPB)-3, dated April 28, 2016, inquiring how valve leakage greater than 100 gpm will be tracked for acceptability during the 7 day allowance. In its response, the licensee stated that:

> Based on a number of criteria including valve classification (i.e., ASME Code Class, Appendix J), valve design, valve size, and operating experience, it is unlikely that any realistic boundary valve leakage would challenge the analytical accuracy of the ERCW system hydraulic model. The 100 gpm value referenced [as noted above] is an unrealistically high value for valve leakage. In accordance with TVA's Corrective Action Program (CAP) and the "Operability Determination Process and Limiting Conditions for Operation Tracking" procedure, an Immediate Determination of Operability (IDO) is required for any identified leakage from a process system, such as ERCW. As described in the referenced procedure, if system operability cannot be ensured via the IDO, additional evaluations may be necessary. If at any time during the Operability Determination Process, a reasonable assumption of operability is not supported, appropriate system-specific TS actions would be taken as necessary.

The staff finds this RAI response is acceptable to ensure that the licensee would take appropriate actions if ERCW valve leakage were to compromise operability and reach the 100 gpm threshold.

The licensee stated in the LAR that the UFSAR will be revised to add the following information related to those ERCW loads that need to be isolated and ERCW configuration to support bus cleaning. This information is needed to meet the design function of a single ERCW loop with one operating ERCW pump.

For Unit 1 Train A One Pump Operation:

> (Unit 1 TS 3.7.8 LCO ACTION A entered for 7 days, 2A-A bus cleaning with Unit 2 defueled or in Mode 6 following defueled with Unit 2 refueling cavity level $\geq$ 23 feet above top of reactor vessel flange with UHS $\leq$ 79° F)

ERCW flow is isolated to the following components:

- 2A-A Diesel Generator Heat Exchangers (HXs);
- Unit 2 Containment Spray HX 2A;
- Unit 2 Turbine Driven AFW Pump from the "2A" ERCW Main Supply Header;
- Lower Containment Vent Cooler 2A, CRD Vent Cooler 2A, and RCP 2-1 Motor Cooler;
- Lower Containment Vent Cooler 2C, CRD Vent Cooler 2C, and RCP 2-3 Motor Cooler;
- Upper Containment Vent Cooler 2A;

- Upper Containment Vent Cooler 2C; and
- Incore Instrumentation Room Water Coolers 2A.

The following components are in service:

- Train A ERCW yard header crosstie;
- Train A ERCW 16-inch Auxiliary Building header crosstie; and
- Train A ERCW 6-inch ESF header crosstie

Similar information is included in the LAR for One Pump Operation for Unit 1, Train B; Unit 2, Train A; and Unit 2, Train B. These configurations are described in the LAR as being "prerequisite" alignments for the planned shutdown board maintenance. Thus, the planned maintenance activities described in the LAR cannot be performed without first configuring ERCW as specified in the LAR.

In summary, the NRC staff determined that the licensee's approach for determining available ERCW flowrates was reasonable. Between the combinations of operating units and shutdown units (defueled or in Mode 6 following defueled with the shutdown unit refueling cavity level ≥ 23 feet above top of reactor vessel flange), the ERCW system design function can be met with only one running ERCW pump per train, provided the UHS is ≤ 79 °F, the required loads (as stated above) are isolated, and the piping configuration is established with open cross-ties (as stated above).

3.1.2 Limiting UHS Temperature Determination

The licensee stated in their LAR, to determine the limiting (maximum) UHS temperature, the station's engineering staff applied the methodology previously used to support SQN License Amendments 317/307 [for Unit 1/Unit 2], justifying a maximum UHS temperature limitation of 87 °F. The 87 °F limitation was based on establishing minimum required ERCW flow rates to ERCW supplied ESF components. These minimum required ERCW flow rates are listed in LAR Table 3.2-5 (see below). For the determination of the limiting UHS temperatures, when aligned for single ERCW pump per train operation, the reconfigured flow rates were compared to the design required flow rates. For a component where the reconfigured flow rate was less than the design required flow rate, a curve fit equation was developed to determine the ERCW supply temperature at which the reconfigured flow rate would remove the required heat load from the affected component.

LAR Table 3.2-5 Component Minimum Design Required ERCW Flow Rate

Component	**Required Flow Rate (gpm)****
EDG HX (note: each diesel has two HXs)	522 (1044 total)
CCS HX Train A, LOCA unit	3605
CCS HX Train A, non-LOCA unit	1348
CCS HX 0B1/0B2	3365
CONTAINMENT SPRAY HX	3400
ELECTRIC BOARD (BD) ROOM (RM) - CHILLER (CHR) A	163.9
MAIN CONTROL RM CHR A	95.4
SHUTDOWN BD RM CHR A	380
CENTRIFUGAL CHARGING PUMP (CCP) OIL COOLER (CLR)	23
CCP GEAR OIL CLR	12
CCP RM CLR 1A	34
SAFETY INJECTION SYSTEM (SIS) PUMP (PMP) RM CLR	18
SIS OIL CLR	4.1
EMERGENCY GAS TREATMENT SYSTEM	9
AUXILIARY (AUX) CONTROL AIR	5.1
SFP [Spent Fuel Pool] & THERMAL BARRIER BOOSTER PUMP CLR	28
CCS & AFW CLR	55
BORIC ACID TANK (BAT) & AFW CLR	62
714 PENETRATION (PEN) RM CLR	19
690 PEN RM CLR	12
669 PEN RM CLR	17
PIPE CHASE CLR	29
CONTAINMENT SPRAY PMP RM CLR	10
RHR PMP RM CLR	15
LOW CONTAINMENT COOLERS (LCC) $$	170
** support 87 °F average ERCW supply temperature	
$$ The LCCs only have a safety function to help mitigate a non-LOCA high energy line break inside containment, such as an MSLB. The design flow was determined to be 170 gpm in order to achieve the required thermal performance. However, Technical Requirement Verification 8.6.4.3. requires that 200 gpm be available to each LCC.	

As stated above, the CCS HXs require higher flow rates than other components and need to be further described. A description of the A Train CCS HXs is provided below as an example:

<u>CCS HX Train A, LOCA Unit</u>

The minimum design flow rates for the CCS HXs are the calculated minimum values required to remove the assumed heat load from the CCS following a LOCA. The majority of the heat load on the CCS comes from the RHR HX after sump recirculation begins, which is calculated to be 43 Million British Thermal Unit per hour (MBTU/hr). The other heat inputs comprising an additional 0.8 MBTU/hr following a LOCA are:

Containment Spray Pump (CSP) bearing cooling,
Centrifugal Charging Pump (CCP) seal cooler,
Safety Injection Pump (SIP) seal cooler,
RHR Pump seal cooler,
Seal Water HX,
Various sample coolers.

Therefore, the total heat load on the A train LOCA unit is 43.8 MBTU/hr, which can be accommodated by the 3,605 gpm CCS flow.

<u>CCS HX Train A, non-LOCA unit</u>

The heat load assumed for the non-LOCA unit are:

Reactor Coolant Pump Motors,
RCP Thermal Barriers,
CCP seal coolers,
Letdown Heat Exchanger,
Seal Water HX,
Various sample coolers,
SFP.

The heat load for an on-line unit is ~8-12 MBTU/hr, plus the SFP heat load, which is typically less than 10 MBTU/hr when a refueling outage is not in progress. The assumed heat load is about 26 MBTU/hr, which can be accommodated by the 1,348 gpm CCS flow.

<u>CCS HX 0B1/0B2</u>

For the B-train CCS (CCS HX 0B1/0B2), there is the calculated 43 MBTU/hr assumed heat load from the accident unit's RHR HX, plus the minor heat loads from the seal/bearing coolers on both units CCPs, SIP, RHR Pump, and CSP, for a total heat load of about 43.3 MBTU/hr, which can be accommodated by a 3,365 gpm CCS flow.

Main Control Room (MCR) operator actions are needed to support these CCS HXs design flow rates as described below. Because the majority of the heat load on the CCS HXs occurs when the RHR HXs are placed in service at sump recirculation initiation, the CCS and ERCW alignments are only required to be completed at sump recirculation. The normal ERCW alignment always has flow to the CCS HXs. The CCS pumps are normally running as some heat loads/flow loads on the CCS are present during operation.

On accident initiation on one unit, no operator action is required with regard to ERCW or CCS until sump recirculation is initiated, other than to validate that the proper pumps started upon SI signal receipt or upon blackout restoration. As the pump suction swaps over to the sump recirculation, the ERCW throttling Motor Operated Valve (MOV) at both of the A train CCS HXs is placed in its required position (a total of two ERCW to CCS valves to be manipulated by the Operators). The B train similar MOV is automatically positioned by the SI signal. The ERCW valves to the accident unit CS HXs are opened (a total of four ERCW-CS valves to be manipulated by the Operator). The CCS valves for the accident unit's RHR HXs are opened (a total of two valves). On the B train CCS, the flow to the non-accident unit's RHR HX may need to be adjusted (one valve). The CCS flow to the SFP HXs is removed from the accident unit, if it is supplied by the accident unit (two valves). The SFP cooling load is placed on the non-accident unit, by the non-accident unit's Operators. Therefore, a total of 11 ERCW and CCS valves are manipulated by the control room operators upon sump swapover. All of these valves are located on the same panel in the MCR. The ERCW System and CCS system flow balance is set up so that the above listed manipulations, together with the set of SI-signal driven automatic actuations, would place the systems in the required configuration to support accident mitigation.

To determine the UHS temperature at which the reconfigured flow rate would remove the required heat load, one of three methods was used to develop a "curve fit equation." The curve fit equation provides the relationship between the component's flow rate and the ERCW supply header water temperature (UHS temperature) at which the required heat load was removed.

The first method used energy balance equations to develop a correlation factor that was then applied to the various component flows in order to calculate the available flow margins.

The correlation factor method was used for the electrical board room coolers, MCR chiller, ACA compressors, and the SIS pump oil cooler. Using a previously determined minimum flow rate for 84.5 °F for these components, the required minimum flow rates for 87 °F and 81 °F for the specific Structures, Systems, or Components (SSCs) were determined. The minimum flow rate values at 87 °F, 84.5 °F, and 81 °F were tabulated in Microsoft Excel® (Excel) and an Excel trendgraph was then produced, with a trendline. A second order polynomial curve that fit the trendline was selected. From the polynomial curve, a curve fit equation was developed and used to determine the required maximum supply temperature if the hydraulic analysis shows the HX receives less than the design required flowrate. The results were further reduced in order to ensure that all temperatures computed from the equations are conservative.

The second method used was the manipulation of the PROTO-HX computer code to model the component. In PROTO-HX, changes to the calculation settings were made to provide resulting data for required ERCW flow over various ERCW supply temperatures. The values determined were again tabulated in Excel, and an Excel trendgraph was produced, with a trendline. A second or third order polynomial curve that fit the trendline was selected. From the polynomial curve, a curve fit equation was developed and used to determine the required maximum supply temperature if the hydraulic analysis shows the HX receives less than the design required flowrate. The results were reduced further in order to ensure that all temperatures computed from the equations are conservative.

The third method was the use of values directly from other design inputs that had determined the required flowrate for those components at various temperatures, such as a chiller analysis for ERCW temperature change. The values were tabulated in Excel, and an Excel trendgraph was produced, with a trendline. A second order polynomial curve that fit the trendline was selected. From the polynomial curve, a curve fit equation was developed and used to determine the required maximum supply temperature if the hydraulic analysis shows the HX receives less than the required design flowrate. The results were reduced further in order to ensure that all temperatures computed from the equations are conservative.

The NRC staff determined that the licensee's approach for determining limiting UHS temperature was reasonable. Table 3.2-2, "Train A One Pump Outages 1 and 3," and Section 3.2.5, "Summary of Results," from the LAR indicate that the maximum acceptable UHS temperature is 79 °F to ensure adequate heat removal for EDG 2A2 with Unit 1 in Operating Mode 6.

3.1.3 Equipment environmental qualification (EQ) issue

The licensee stated in their LAR that Multiflow Model runs were prepared assuming an MSLB inside containment. Outage Cases 2 and 4 of LAR Table 3.2-1 provide the initial conditions for the MSLB Multiflow Model runs. Outage Cases 2 and 4 of LAR Table 3.2-4 provide the corresponding ERCW flow results. Outages 2 and 4 in LAR Table 3.2-4 were prepared to ensure the LCCs received the required 200 gpm flow rate with one pump operating on the affected train.

Based on the Multiflow Model results (LAR Table 3.2-4), no additional ERCW supply temperature limitations are required. In the event that LCCs are required on the inoperable train, if necessary, operators will reduce ERCW flow in the accident unit's CS HXs and A train CCS HX to ensure the required flow is available to the Lower Compartment Coolers. Reducing flow to these components is acceptable since this is a long-term action which applies after CS flow is no longer needed (due to break flow terminating as the affected steam generator blows dry) and since the CCS heat load following a steam line break is significantly lower than the post-LOCA heat load. These actions ensure the environmental qualification temperature limitation on instrumentation in lower containment is not exceeded.

The NRC staff finds that the Multiflow calculations including methodologies, assumptions, and results were suitable for modeling ERCW conditions for this LAR and support the Raw Water TS changes for extending the allowed out of service time of 7 days provided the opposite unit was either refueled or in Mode 6 with its refueling water cavity level $\geq$ 23 feet above top of reactor vessel flange and had the associated unit loads isolated as noted above and the UHS temperature $\leq$ 79 °F.

The NRC staff determined that the licensee's approach for determining the EQ was reasonable. The SQN analysis indicate that based on LAR Table 3.2-4, the resultant flow rates to the LCC 1DB (worst case) was 213.8 gpm, which is above the 200 gpm requirement and no additional ERCW supply temperature limitations are required.

3.1.4 Generic Letter 89-13 program

The licensee stated in their LAR that in GL 89-13, "Service Water System Problems Affecting Safety-Related Equipment," the NRC requested that licensees perform the action listed in the GL or equally effective actions to ensure that their Service Water Systems are in compliance and will be maintained in compliance with 10 CFR Part 50, Appendix A, General Design Criteria 44, 45, and 46 and Appendix B, Section XI. By the definition of "service water" found in the GL, "the system or systems that transfer heat from safety related SSCs to the UHS," this system at SQN is the ERCW System. In accordance with GL 89-13, a continuing program is maintained to perform periodic inspections of the ERCW intake structure for biological fouling mechanisms, sediment and corrosion. In addition, a continuing test/inspection program is maintained to verify the heat transfer capability of all safety related HXs included in the GL 89-13 program.

Additional details associated with the licensee's ERCW HXs and GL 89-13 program were provided to the NRC as a part of TVA's LAR to raise the UHS temperature limit and water level limit (ADAMS Accession No. ML063470029). This request was approved by the NRC under License Amendments 317/307 [Unit 1/Unit 2] (ADAMS Accession No. ML072420043).

The NRC staff determined that the licensee's approach for determining the acceptability of GL 89-13 program was reasonable and as part of this LAR approval, there are no changes necessary to the existing GL 89-13 Program.

3.1.5 TS 3.7.8 proposed completion time of 7 days

TVA has evaluated actions that would be required to facilitate the scheduling of maintenance activities on the 6.9 kV and 480 V shutdown boards in the electrical distribution system at SQN. TVA determined that changes to certain SQN TS requirements are needed to facilitate the specific maintenance activities. TVA's objective is to facilitate cleaning and inspection of the 6.9 kV shutdown electrical board and associated 480 V shutdown boards without requiring a dual unit shutdown. Cleaning of one division of 6.9 kV shutdown boards with 480 V shutdown boards are estimated to take approximate 100 hours, which would exceed TS 3.7.8 "Essential Raw Water Cooling Water System," LCO of 72 hours for ERCW being inoperable, reference SQN's response to RAI GMW-004 (ADAMS Accession Nos. ML15176A678 and ML15176A681).

The NRC staff requested TVA to justify why a completion time of 7 days was needed, as opposed to a shorter timeline, such as 100 hours. TVA provided a detailed timeline of maintenance activities, stating that it estimates the activities associated with the 6.9 kV shutdown boards can be completed in approximately 69 hours, with an additional 36 hours for contingencies. This would provide a total of approximately 4.5 days. TVA stated that 7 days will provide margin for unanticipated repair activities in its RAI response dated May 31, 2016 (ADAMS Accession No. ML16153A071).

In RAI SBPB-2, the staff asked for TVA to describe defense-in-depth (DID) strategies to protect operable equipment during the planned maintenance or inadvertent operations that would challenge plant safety. TVA stated that on both the outage unit and the operating unit, Control Room Operations will regularly complete DID Assessments to ensure that protected components or systems are not jeopardized. This may include administrative or physical barriers, or postings labeled "protected equipment."

It should also be noted that this proposed outage of 7 days is consistent with SQN TS 3.8.1, Unit 1 and Unit 2. As part of its Improved STSs (ADAMS Accession No. ML13330A927), the licensee modified TS 3.8.1, "AC Sources – Operating" to allow a completion time of 7 days in the condition of an inoperable diesel.

Based on the information provided by TVA, the NRC staff finds that the proposed TS 3.7.8 completion time of 7 days to complete work activities for the 6.9 kV and 480 V shutdown boards is acceptable, as this will provide approximately 2.5 days that can be utilized for contingencies if major issues are found during bus inspections and cleaning, and adequate DID measures will be taken to ensure protected equipment is not jeopardized during the extended completion time.

4.0 Summary

The staff found that when the ERCW system is aligned to the prerequisite configuration and the UHS temperature satisfies the maximum temperature limit, the unaffected ERCW train would be capable of performing the ERCW safety functions during the maintenance period, and the affected train would provide redundant capability when the opposite unit has negligible heat removal requirements. With this additional capability, the ERCW safety function could be satisfied considering a single failure that disables the unaffected train. The licensee also described equipment protection plans to maintain DID during the planned maintenance. Therefore, the extension of the completion time from 72 hours to 7 days is justified. Per NUREG-800, the staff determined that the completion time of 7 days is a departure from the reference TS, NUREG-1431. However, given the considerations discussed in Section 3.0 and summarized above in this section, the staff also determined that the departure is acceptable. Therefore, NRC staff determined that the TS, as modified by the proposed change, will continue to meet the regulatory requirements of 10 CFR 50.36. Based on the regulatory and technical review above, the NRC staff found the licensee's proposed changes to TS 3.7.8 to be acceptable.

5.0 STATE CONSULTATION

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

6.0 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 or changes surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in amounts, and no significant change in the types of any effluents that may be released offsite, and there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (81 *FR* 21603). 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 amendment.

7.0 CONCLUSION

The NRC staff 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: Nicholas Hobbs
Larry Wheeler

Date: September 29, 2016

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

Sincerely,

/RA/

Andrew Hon, 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. 336 to RFOL No. DPR-77
2. Amendment No. 329 to RFOL No. DPR-79
3. Safety Evaluation

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