TVA TENNESSEE VALLEY AUTHORITY

Sequoyah Nuclear Plant, Post Office Box 2000, Soddy Daisy, Tennessee 37384

May 9, 2022

10 CFR 50.59
10 CFR 72.48
10 CFR 50.71

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555-0001

Sequoyah Nuclear Plant, Units 1 and 2
Renewed Facility Operating License Nos. DPR-77 and DPR-79
NRC Docket Nos. 50-327, 50-328, and 72-034

Subject: **10 CFR 50.59 and 10 CFR 72.48 Changes, Tests, and Experiments Summary Report and Commitment Summary Report**

Reference: TVA letter to NRC, “10 CFR 50.59 and 10 CFR 72.48 Changes, Tests, and Experiments Summary Report and Commitment Summary Report,” dated November 2, 2020.

In accordance with 10 CFR 50.59(d)(2) and 10 CFR 72.48(d)(2), Enclosure 1 is the Sequoyah Nuclear Plant (SQN), Units 1 and 2, Summary Report regarding the implemented changes, tests, and experiments for which evaluations were performed in accordance with 10 CFR 50.59(c) and 10 CFR 72.48(c). The summarized evaluations provided in the enclosure were implemented since the Reference Letter through April 14, 2022.

Since last reported in the Reference Letter, SQN has revised a regulatory commitment in accordance with NEI 99-04, the Nuclear Energy Institute's "Guidelines for Managing NRC [Nuclear Regulatory Commission] Commitment Changes," as endorsed in NRC Regulatory Issue Summary 2000-17. The commitment change summary is provided in the Enclosure 2.

There are no new commitments contained in this letter. If you have any questions concerning this submittal, please contact Mr. Jeffrey Sowa, SQN Licensing Manager at (423) 843-8129.

Respectfully,

Marshall, Thomas B.

Digitally signed by Marshall, Thomas B.
Date: 2022.05.09 12:53:49 -04'00'

Thomas Marshall
Site Vice President
Sequoyah Nuclear Plant

Enclosures
1. 10 CFR 50.59, and 10 CFR 72.48 Changes, Tests, and Experiments Summary Report
2. Commitment Change Report

cc (Enclosures):

NRC Regional Administrator – Region II
NRC Senior Resident Inspector – Sequoyah Nuclear Plant
Director, Division of Fuel Management, Office of Nuclear Material Safety and Safeguards

ENCLOSURE 1

SEQUOYAH NUCLEAR PLANT

10 CFR 50.59 AND 10 CFR 72.48
CHANGES, TESTS, AND EXPERIMENTS SUMMARY REPORT

<table>
<tr><th>DESIGN CHANGES</th><th>DESCRIPTION</th><th>SAFETY ANALYSIS</th></tr>
<tr><td>23396 including PIC 23529 & 23696</td><td>This design change issues design output from Sequoyah Nuclear Plant (SQN) Calculation SQS20110 Revision 24. SQS20110, Emergency and Abnormal Operating Procedure Setpoints, is a “collector calculation” to assemble process parameters, instrument setpoints, and instrument uncertainty values as contained in existing design documents into a single calculation to provide the basis for the SQN plant-specific Emergency Operating Procedure (EOP) setpoints.

The system process parameters provided in this design change are taken from existing issued design documents and do not result in changes to system pressures, temperatures, conditions, or requirements.

The following Pressurized Water Reactor Owner's Group (PWROG) Direct Work documents were used to revise SQS20110:
DW-06-014 - calculate nitrogen expansion in the cold leg accumulators using a different factor
DW-04-009 - enhancements to Steam Generator [S/G] Tube Rupture event to allow terminating Reactor Coolant System (RCS) depressurization using sprays earlier (prior to RCS and ruptured S/G pressures being completely equalized) to prevent the pressurizer level from dropping below the just on span value requiring Safety Injection (SI) re-initiation after SI is terminated
DW-10-007 provides steps in Abnormal Response Guideline that, as written, could result in aligning the suction of the charging pumps to the Volume Control Tank (VCT) even if charging flow is high and the VCT makeup capacity is not sufficient to maintain VCT level
DW-11-004 - guidance and evaluations tripping the reactor coolant pumps (RCPs) after a small break loss of coolant accident (SBLOCA) when the RCP trip criteria</td><td>None of the changes were determined to:
- result in more than a minimal increase in the frequency of the occurrence of an accident because the changes provide for Operator actions after an accident.

- result in more than a minimal increase in the likelihood of the occurrence of a malfunction of a structure, system, and component (SSC) important to safety as the changes do not physically modify SSCs. TCAs were validated to ensure no more than a minimal increase in the likelihood of the occurrence of a malfunction of an SSC important to safety. Stopping and starting certain emergency core cooling system pumps were determined not to increase the likelihood of malfunctions causing the pumps and the support SSCs to not restart when call upon. Evaluations found peak clad temperature during suction swap evolution for a small-break loss of cooling accident, remain within 10 CFR 50.46 limits and no hot fuel pin rupture was predicted. Removal of verification of containment sump level prior to restarting pumps with suction aligned to the sump during LOCA and non-LOCA accidents, would not likely have a minimal increase in the occurrence of a malfunction of an SSC important to safety because analysis indicates sufficient water level is available.

- result in more than a minimal increase in the consequences of an accident or a malfunction of an SSC important to safety, because the current dose analyses remain unchanged for the actions being taken and accidents evaluated.

- create the possibility for an accident of a different type because the changes only affect actions that Operators would take as a response to an accident. No new accidents were identified for these changes.</td></tr>
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DESIGN CHANGES	DESCRIPTION	SAFETY ANALYSIS
	have been met. This results in an existing operator manual action becoming a Time Critical Action (TCA) Other setpoints were modified or created for the following: • Information in setpoints B05 and B06 was updated and separated into injection and recirculation setpoints • Creation of the M12 setpoint allows Operations to align procedures with the current technical specification (TS) values • Setpoint U92 was expanded to include loss of sump recirculation Post Issuance Change (PIC) 23529, created Revisions 25 and 26 of SQS20110. This resulted in modifying UFSAR Section 15.3 (Small Break LOCA) to include discussion that at least 5 minutes are available to manually trip the RCPs from a loss of subcooling or from an SI signal before Peak Clad Temperature (PCT) limits are exceeded. NRC Information Notice 97-60 regarding the increase in the likelihood of occurrence of a malfunction of equipment important to safety with respect to Emergency Core Cooling System (ECCS) Pump operation/swapover is considered in the evaluation. PIC 23696, created Revision 27 of SQS20110. The changes are to the details of Operator TCAs related to verifying containment sump level prior to restarting Containment Spray pumps with suction aligned to the sump. This PIC also creates a new TCA related to the failure of a Residual Heat Removal (RHR) block valve to close. This PIC documents an increase in PCT, and it removes a procedural barrier which protects the Containment Spray Pumps against operation with inadequate sump level.	- create the possibility for a malfunction of an SSC important to safety with a different result because the addition of the TCA to trip the RCP was already an early option in the emergency operating procedure. Design analysis shows that a sufficient water level is in the containment sump; therefore, deleting level verification prior to starting Containment Spray Pumps will not result in a malfunction of an SSC important to safety. - result in a design basis limit for a fission product barrier being exceeded or altered because the peak clad temperature limit of 10 CFR 50.46 remains satisfied. It was concluded that the proposed change may be implemented in accordance with plant procedures without obtaining a License Amendment and without prior NRC approval.
23780 (Implemented on Unit 2 at this time)	This design change permanently removes the A-AUTO, STANDBY feature for the Emergency Gas Treatment System (EGTS) Air Cleanup Subsystem isolation dampers. The control logic is modified to remove the close signal of the isolation	The proposed changes from design change 23780 cannot initiate an accident, do not increase the probability of a malfunction of an SSC, do not involve any changes to a Design Basis Limit for a Fission Product Barrier, or create the

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	dampers in A-AUTO generated for abnormal pressure. The handswitches are modified to remove the A-AUTO, STANDBY position which would open the dampers. Both EGTS air cleanup exhaust/recirculation flow paths will therefore operate in the A-AUTO mode by modification of control room panel handswitches. The resultant design provides for control, detection, and alarm for abnormal annulus differential pressure (dP), but no automatic downstream isolation of control dampers in either EGTS air cleanup flow path. Procedures are also revised to reflect the handswitch control positions due to removing power to auxiliary (swap over) relays. The scope of this proposed modification also includes a change to the Loss of Coolant Accident (LOCA) analysis to include a postulated leakage of 30 cfm past the decay cooling line isolation valves. These valves are used to provide a flow path from the suction of the in-service EGTS filter bank to the suction of the opposite Train's EGTS fan, around the inactive EGTS filter bank, for decay heat cooling purposes when the high-efficiency particulate air (HEPA) filters and adsorbers in the inactive train are loaded with radioactive material.	possibility for an accident of a different type than already analyzed in the UFSAR. In addition, the increase in dose consequences due to the new configuration where a single failure of a controller could result in an EGTS air cleanup exhaust damper in full-open and the inclusion of a 30 cfm leakage past the decay cooling line isolation valves is not more than minimal. This change may be implemented in accordance with plant procedures and does not require a License Amendment Request.
23874	The Fuel Transfer System (FTS) is upgraded with a new control system (analog to digital) with new removable control panels, conveyor car (or transfer car), push arm, sheaves, winch/motor assemblies, and load weighing assemblies. These upgrades enable the FTS to operate at a higher conveyor car speed. A shock arresting system is installed on the support frame assembly to protect the fuel assembly and conveyor car. The new conveyor car/push arm assembly will have remotely replaceable wheels. The new control system will utilize programmable logic controllers with touch-screen human-machine interfaces (HMIs) and variable frequency drives (VFDs). New emergency stop buttons are added. The existing Sound Powered Telephone System will be spared (no longer used).	The existing control system consists of a relay-based motor control system that utilizes programmable limit switches for frame and conveyor car position inputs. The replacement control system consists of a programmable logic controller (PLC)-based platform with touch-screen displays and VFDs. The touch-screen displays were designed to replicate the existing control panel operator interfaces to the extent practical. The HMI has been reviewed and accepted by FTS operating personnel. The operation of the FTS, as described in the UFSAR, remains unchanged. The design functions, interlocks and other features of the FTS are retained and are not adversely impacted by the proposed activity.

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	Based on a review of Nuclear Energy Institute (NEI) 01-01 "Guideline on Licensing Digital Upgrades," some of the control function changes are considered potentially adverse on the ability of the FTS to perform is design function, including the addition of software/firmware and combining interlock functions into one digital device. The introduction of the HMI touch-screen is considered potentially adverse.	Digital device quality assurance measures include: 1) software quality assurance activities as described in the project specific software quality assurance plan, 2) performance of critical digital reviews of the control system, 3) performance of a failure modes and effects analysis (FMEA) of the replacement control system, and 4) extensive acceptance testing. The use of these quality assurance methods provides reasonable assurance that the use of digital controls in the FTS does not increase the probability or the consequences of any Fuel Handling Accident (FHA) described in the UFSAR, nor will it introduce the possibility of a new type of accident not previously considered. The assumptions and conclusions of the existing FHA analyses remain valid. No reductions in the existing margins of safety are created by implementing this modification. The existing shutdown margins during refueling operations are maintained. The existing submergence limits and radiation shielding margins are retained. No changes to the TSs or their Bases are required. In order to ensure that this 50.59 Evaluation sufficiently addresses issues related to digital upgrades, the questions in Appendix A of NEI 01-01 were listed and answered with the corresponding 50.59 Evaluation question. It was concluded that this proposed modification may be implemented in accordance with plant procedures and does not require a License Amendment Request.
SQN-19-336 (Unit 1) SQN-19-337 (Unit 2)	The existing Main Turbine Analog Electrohydraulic (AEH) control system and Moisture Separator Reheater (MSR) controls at SQN are obsolete and are being replaced with a new Digital Electrohydraulic (DEH) control system, including both hardware and software. The DEH System incorporates the control, protective, and monitoring functions of the existing main turbine AEH system and Moisture Separator Reheater (MSR) controls	The change from analog to digital controls for turbine and MSR controls; the use of the new HMI displays for turbine control, MSR control, testing, and maintenance functions; the introduction of new software/firmware; and turbine trip logic changes are considered adverse for performing design functions.

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	with enhancements designed to improve fault tolerance and alleviate a number of single point vulnerabilities. The scope of the design changes also includes the removal of the Turbine Trip Auto-Stop Header which is replaced with a new four-valve trip block assembly connected to the high pressure Electrohydraulic Control System (EHC) Emergency Trip Header. An associated EHC design change, SQN-19-339, resulted in TS change request SQN-TS-19-04 that was submitted to revise the low oil pressure switches trip setpoints in TS Limiting Condition for Operation (LCO) 3.3.1, Table 3.3.1-1 (ADAMS ML17075A229). NRC approved the TS change request on November 12, 2020 (ADAMS ML20262H026).	This evaluation has determined that the EHC Turbine Control and Turbine Protection Systems (TGCPS) will continue to meet its design requirements following the implementation of the proposed modification that converts to a digital control system. The use of the new Turbine Control System (TCS) HMI displays will not adversely impact any turbine control functions or create additional operator burden. The design functions of the existing DCS Nuclear Steam Supply System (NSSS)/Balance of Plant (BOP) and Auxiliary Control applications are not changed or modified. The use of the existing Distributed Control System (DCS) HMI displays will not adversely impact any control functions or create additional operator burden. Since the new DEH System and DCS components are more reliable than the existing components and no new system level failure mode effects are introduced, the proposed modification does not result in more than a minimal increase in the frequency of occurrence of an accident or transient previously evaluated in the SQN UFSAR. The new equipment being installed will not result in any component malfunctions that could increase the potential for a turbine trip or transient. Any malfunction will not result in an increase in the potential for a required protective function to be performed (tripping the turbine). Therefore, the modification does not result in more than a minimal increase in the likelihood of occurrence of a malfunction of a System, Structure, or Component (SSC) important to safety previously evaluated in the UFSAR. Performance requirements associated with core cooling are unaltered such that fuel integrity will be maintained and the UFSAR analysis of radiological consequences remains bounding. The new equipment will not initiate any new

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		accidents. The modification will not impair or prevent the ECCS from mitigating the consequences of any design basis accidents. Therefore, this activity does not result in more than a minimal increase in the consequence of an accident previously evaluated in the UFSAR. Failure or malfunction of the new equipment will not prevent or affect the ability of safety related systems or systems important to safety to respond to the accidents describe in the UFSAR. Therefore, implementation of the proposed modification does not result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the UFSAR. The potential malfunctions of the modified equipment are bounded at a system level in the UFSAR. Therefore, the possibility for an unanalyzed malfunction of an SSC important to safety or an accident of a different type than any previously evaluated in the UFSAR is not created. As described in the UFSAR accident analysis, no malfunction of the TGCPS or the DCS can cause a transient sufficient to damage the fuel barrier or exceed the nuclear limits as required by the safety design basis. Therefore, the possibility for an unanalyzed malfunction of an SSC important to safety that can challenge a fuel barrier or an accident of a different type than any previously evaluated in the UFSAR is not created. The new digital equipment does not necessitate a revision or replacement of any currently used evaluation methodology. The modification does not result in a departure from the method of evaluation described in the UFSAR in establishing the design bases or in the safety analyses.

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		Guidance for evaluation of digital upgrades is contained in NEI 01-01, Guideline on Licensing Digital Upgrades, March 2002. NRC Information Notice (IN) 2010-10 stated the NRC expectation that all the questions in Appendix A of NEI 01-01 should be considered in a 50.59 Evaluation for systems that could cause a plant trip or reactivity transient. In the 10 CFR 50.59 Evaluation for the proposed modification, the questions in Appendix A of NEI 01-01 are addressed. The 50.59 Evaluation concludes that implementation of the proposed modification does not require a TS change and that implementation of the modification does not require a License Amendment Request (LAR), and therefore may proceed without NRC approval.
SQN-20-1909	This Design Change increases the design stroke time limit (both open & close directions) for pressurizer power operated relief valve (PORV) block valves 1,2-FCV-68-332 & 1,2-FCV-68-333. The design stroke time is being increased from 10 seconds to 20 seconds due to low margin between the block valves stroke times and the required action range. This margin is too low to account for any additional potential response time that could occur during the normal testing method of an Operator using a stop watch. The 10-second design stroke time limit is a condition under which the intended function is required to be performed; therefore, the increase in design stroke time is considered adverse and is Evaluated further.	The Pressurizer (PZR) PORV block valves are normally operated in the OPEN position to allow PZR PORV operation. However, the block valves may be operable when closed to isolate the flow path of an inoperable PZR PORV that is capable of being manually cycled (e.g., as in the case of excessive PZR PORV leakage). Therefore, it is possible for a block valve to be required to open in order to provide a flow path to its associated PZR PORV during a Steam Generator Tube Rupture (SGTR) accident. On the other hand, the PZR PORV block valves may be required to close during an Accidental Depressurization of the RCS event due to a PZR PORV being stuck open. The proposed activity increases the design stroke time limit of the PZR PORV block valves from 10 seconds to 20 seconds to perform the above safety functions. It is important to note that the stroke time for the block valves to perform the aforementioned safety functions is not credited or utilized in the Safety Analyses. The only event where the increased stroke time could potentially have an effect is during a SGTR. However, it is shown that sufficient margin exists to accommodate an increased block valve stroke time. The PZR PORV block valves are not being physically altered, RCS

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		system conditions remain unchanged, and no RCS SSCs are affected by this change. No new accident initiating events are created by this change, and there is no increase to the frequency of previously evaluated accidents. The block valves are the only components directly affected by this change. The malfunction to be evaluated is the inability for the block valves to open or close when required to allow PORV operation or isolate a leaking/stuck open PORV, or failure of a block valve to meet the In-Service Testing (IST) stroke time requirement during the TS required quarterly valve strokes. There is no increase in the likelihood of a malfunction due to this change as no new failure modes are introduced by the proposed activity. Hence, the change does not increase the consequences of any accident described in Chapter 15 of the UFSAR. In addition, this change does not create the possibility of an accident not already described by the UFSAR and will not result in the design basis limit for any fission product barrier being exceeded or altered. Therefore, it is acceptable to implement the proposed activity per plant procedures without obtaining a License Amendment.
SQN-21-017	Design Change SQN-21-017 proposes to cut and cap the R-8 Thermocouple in SQN Unit 1 core exit thermocouple nozzle assembly (CETNA) location R-11. The cut is to be just below the current pressure boundary compression fitting. The thermocouple is to physically remain in the reactor vessel but will be out of scan due to the cable being cut. Inside the tube, the stainless steel thermocouple sheath is to be crimped on the exposed/cut end to preclude RCS water from entering the mineral insulation around the thermocouple wire. It is to then be capped with a new compression fitting that will be of the same American Society of Mechanical Engineers (ASME) classification as the original. The remaining tubing is to be restrained to prevent movement and to allow for proper installation of the bullet-nose.	The Incore Thermocouple System must function to aid with the mitigation of the consequences of the following events: 1. Loss of Coolant Accident 2. Main Steamline Break Inside Containment 3. Feedwater Line Break Inside Containment 4. RHR Line Break Inside Containment 5. Chemical and Volume Control System Line Break Inside Containment. The activity does not increase the frequency of occurrence of any of the above described accidents evaluated in the UFSAR because the core exit thermocouples are not initiators of any of the accidents analyzed in the UFSAR. Although the activity does reduce the number of available thermocouples to monitor

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	The TS Bases is to be revised to update the number of available thermocouples for Unit 1 (TS B 3.3.3 - Post Accident Monitoring Instrumentation).	core temperatures, it does not result in more than a minimal increase in the likelihood of a malfunction of the Incore Instrumentation System, as well as any other SSC evaluated in the UFSAR. The reduction of one additional thermocouple also does not increase the expected Main Control Room (MCR) Operator onsite (mission) dose or offsite radiological releases already analyzed in the UFSAR. This activity also does not cause any new accident that was not previously analyzed in the UFSAR as any malfunction of any SSC that would cause results of any UFSAR analyses to change or diverge in such a manner would place this facility in an unanalyzed condition. In addition, this activity will not cause any design basis limit for a fission product barrier being exceeded or altered. This activity may be implemented per plant procedures without obtaining a License Amendment.

DOCUMENT NUMBER/72.48 EVALUATION TRACKING NUMBER	DESCRIPTION	SAFETY ANALYSIS
None		

ENCLOSURE 2

SEQUOYAH NUCLEAR PLANT

COMMITMENT CHANGE REPORT

Commitment Evaluation No./ Commitment Tracking No.	Source Document	Summary of Original Commitment	Summary of Commitment Changes	Basis/Justification for Changes
NCO970024004	TVA letter to NRC dated April 28, 1998	SQN will implement the program elements described in Topical Report MPR-1807, Revision 2. Deviations identified during the implementation of this program will be justified.	For continued compliance with Generic Letter (GL) 96-05 requirements, the SQN Motor Operated Valve (MOV) Program includes, in addition to the current program, an alternate periodic verification program for valves outside the scope of applicability of the Joint Owners' Group (JOG) Program. Valves in JOG Class D status are considered out-of-scope and includes: 2-FCV-070-0133.	In accordance with NRC letter to TVA dated January 3, 2000, TVA is required to develop a long term MOV periodic verification program for MOVs outside the scope of the JOG Program. Flow control valve, 2-FCV-070-0133 has been identified as a JOG Class D valve, making it outside the scope of the JOG Program. Therefore, an alternate program is required to verify periodically the design-basis capability of the safety-related MOV. TVA report JOG Class D Evaluation for Gate Valve 2-FCV-070-0133, Doc No. 3959C by Kalsi Engineering documents the valve degradation aspect and justification for a periodic verification interval by application of Technical Report BWROG-TP-09-033, Generic Methodology for JOG MOV Periodic Verification (PV) Program – Category D MOV Evaluations, Rev. 1, December 22, 2009. The alternate program is identified in the MOV Program implementing procedure.