Tennessee Valley Authority, Sequoyah Nuclear Plant, P.O. Box 2000, Soddy Daisy, Tennessee 37384

June 5, 2019

10 CFR 50.4
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: 1. TVA letter to NRC, "10 CFR 50.59 and 10 CFR 72.48 Changes, Tests, and Experiments Summary Report; Commitment Summary Report; and Update to Fire Protection Report," dated November 28, 2017

In accordance with 10 CFR 50.59(d)(2) and 10 CFR 72.48(d)(2), enclosed 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 previous submittal, Reference 1, through May 6, 2019.

Since last reported in Reference 1, SQN has not revised a regulatory commitment, in accordance with the Nuclear Energy Institute's "Guidelines for Managing NRC Commitment Changes," as endorsed in Nuclear Regulatory Commission (NRC) Regulatory Issue Summary 2000-17, wherein notification to NRC has not been accomplished by separate submittals.

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

Respectfully,

for Matthew Rasmussen
Site Vice President
Sequoyah Nuclear Plant

Enclosure 10 CFR 50.59, and 10 CFR 72.48 Changes, Tests, and Experiments Summary Report

cc (Enclosure):

NRC Regional Administrator – Region II
NRC Senior Resident Inspector – Sequoyah Nuclear Plant

ENCLOSURE

SEQUOYAH NUCLEAR PLANT

10 CFR 50.59 AND 10 CFR 72.48

CHANGES, TESTS, AND EXPERIMENTS SUMMARY REPORT

Design Change Notice (DCN)	DESCRIPTION	SAFETY ANALYSIS
DCN 23216, Revision A	Following an Appendix R fire in the Control Building, Operators are to abandon the Main Control Room (MCR) and report to the various auxiliary control stations. Prior to leaving the MCR, operators are directed to attempt to open level control valves LCV-62-135 and -136 using MCR switches; however, this action cannot be assured to be successful due to postulated failures of the MCR wiring during a Control Building fire. Additionally, if the Centrifugal Charging Pump Injection Tank (CCPIT) valve flowpath and normal charging flowpath are open, the pressurizer could go water solid. To resolve all these concerns, Unit operators are directed to place the transfer switches in the AUX position, transferring control to 6.9kV Shutdown Board and stop both Centrifugal Charging Pumps (CCPs) from the 6.9kV Shutdown Board and then pull the control power fuses when the MCR is exited within a required 5 minute time limit. The fuses are required to be pulled because the pump could be automatically started following a Loss of Offsite Power (LOOP) and Diesel Generator re-energizing the Shutdown Board. The fuses must be reinstalled when the Operator restarts the credited pump to provide Reactor Coolant System (RCS) makeup and Reactor Coolant Pump (RCP) seal cooling required within 13 minutes. Stopping the pump action cannot be reliably completed within 5 minutes and may not be successful if a blackout signal occurs since the shutdown board handswitches have no lockout position and spring return to the auto position, resulting in an automatic pump restart. Currently to cope with this situation, Sequoyah Nuclear Plant (SQN) has to provide fire watches every hour to ensure that no fire is present. This modification will relieve the need for fire watches by de-terminating the leads on the AUTO function of the Aux "C" handswitches on the 6.9kV Shutdown Boards associated with the CCPs. This will ensure that, in the case of a MCR abandonment, a LOOP will not cause an automatic restart. The handswitches will now spring-return to center location with no auto-control function, and operators will manually turn on and off the pumps through the "C" handswitch. This will eliminate the (Abnormal Operating Procedure) AOP-C.04 requirement to pull and reinstall control power fuses; thus, enabling the operator to complete the Operator Manual Action (OMA) within the required time.	The change being made to the CCPs, MD-AFWPs, and pressurizer (PZR) heaters removes an automatic function which is causing additional operator burden during MCR abandonment for an Appendix R fire. The automatic function is already being defeated by operator pulling the control power fuses because the automatic actuation of these components causes Fire Safe Shutdown (FSSD) mitigation strategy problems. This modification allows the operator to maintain control of the FSSD mitigation strategy by using the handswitches located on the 6.9kV Shutdown Board. The design functions of the pumps and heaters will remain the same. This modification does not affect the normal plant operation from the MCR, only during MCR abandonment. There will be no increase in the likelihood of accidents, malfunctions, or consequences due to this change; therefore, this activity does not require Nuclear Regulatory Commission (NRC) approval. As addressed inside the Fire Protection Program Change Regulatory Review (FPPCRR), the modification is an acceptable change to the Fire Protection System.

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	It was noted that the Motor Driven (MD) Auxiliary Feedwater Pumps (AFWPs) and the Pressurizer (PZR) Heaters had the same auto-starting problem. This modification will de-terminate the AUTO function leads on the MD-AFWPs to prevent restart and eliminate the steam generator (S/G) over fill concern. A new handswitch will replace the current PZR heater handswitch. This new handswitch will still have ON-OFF-AUTO functions, but will no longer spring-return from the OFF position, only the ON position will spring return. This will prevent over-pressurization to the RCS and heater burn-up. Again, these changes will eliminate the AOP-C.04 requirement to pull and reinstall control power fuses; thus, reducing operator burden and increasing operator margin in completing the OMAs within the required times.	
DCN 23396, Revision A	Revision 0: This DCN issues design output from Calculation SQS20110 Revision 24. Calculation 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 DCN are taken from existing issued design documents and do not result in changes to system pressures, temperatures, conditions, or requirements. Pressurized Water Reactor Owner's Group (PWROG) DW-06-014 documents guidance to calculate nitrogen expansion in the cold leg accumulators using a different factor in order to prevent nitrogen injection into the system. As a result of this slightly changed calculation, setpoints B93, B99, O07, and O08 in the Emergency Operating Setpoints Calculation SQS20110 increased. PWROG DW-04-009 recommends enhancements to E-3 (S/G Tube Rupture [SGTR]) to allow terminating 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 re-initiation after safety injection is terminated. The recommended change requires a new pressurizer	One revision to an Operator TCA is proposed, and one new TCA is proposed. The changes are necessary due to a revised analysis that includes consideration of the potential failure of one RHR block valve to close. The revised analysis indicates the possibility of pump damage due to vortexing or air ingestions when the water level in the Refueling Water Storage Tank (RWST) gets below the low-low level setpoint. The revised/new TCA are put in place in order to preclude the possibility of pump damage. The change required a full 10 CFR 50.59 review due to accepting an interruption in ECCS flow, resulting in a significant change in Peak Clad Temperature (PCT) during a SBLOCA accident. Additionally, there is a change to EOP ES-1.3, Transfer to RHR Containment Sump. This change deletes a procedural step to verify the water level in the containment sump prior to the manual

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	level setpoint which would be used as a part of another alternate criterion for terminating depressurization. As a result, setpoints D14 and D15 in Calculation SQS20110 were replaced. PWROG DW-10-007 documents a step in Abnormal Response Guideline ARG-2 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. Therefore, the setpoint S91 was created in Calculation SQS20110. The source of information in setpoints B05 and B06 (i.e. pump curves) had been updated but not incorporated into this calculation. Setpoints B05 and B06 are used for two purposes: to determine if RCS pressure is above the Safety Injection (SI) pump shutoff head prior to stopping SI pumps in Emergency Subprocedure ES-1.1 (SI termination) and to determine if SI pumps should be stopped to prevent deadheading in Emergency Subprocedure ES-1.3 (when isolating the miniflow path). In ES-1.1, a conservatively high setpoint value is desired. However, in ES-1.3, a conservatively low value is desired to prevent SI pump damage due to deadheading. Therefore, the information in setpoints B05 and B06 was updated and separated into injection and recirculation setpoints. Setpoint M12 was created in Calculation SQS20110 to align Emergency Subprocedure ES-0.1 with current Technical Specification (T/S) Limiting Condition for Operation (LCO) 3.4.1.2 (RCS Loops-Mode 3). This T/S requires at least two Reactor Coolant Pump (RCP) loops to be operable with S/G level greater than or equal to 21 % (narrow range). Currently, ES-0.1 specifies a minimum S/G level of 10% narrow range, which comes from Setpoint M02 in Calculation SQS20110. The creation of the M12 setpoint allows Operations to align procedures with the current T/S values. Revision 24 of Calculation SQS20110 also adds an existing OMA to the list of Time Critical Operator Actions (TCAs). This new TCA is to trip the RCPs after a small break loss of coolant accident (SBLOCA) when the RCP trip criteria has been met. This addition is based on guidance and evaluations in DW-11-004, WCAP 9584, Westinghouse Owners Group (WOG) Emergency Response Guidelines (ERG) Revision 3, and Owner's Group Report OG-117.	transfer of the CSP suction from the RWST to the Sump. This change screens in due to potential adverse effects of not validating that sufficient water level exists in the Containment Sump prior to the transfer. The change is acceptable due to referenced analysis that demonstrates that the containment sump level will be sufficient with the RWST level at the low-low level that initiates the actions to swap the containment spray pump suction. Revision 1 adds the 5-minute TCA to trip the RCPs after the RCP trip criteria has been met for a SBLOCA. The 5 minute TCA originally screened out for 10 CFR 50.59 Evaluation because the action to trip the RCPs was a beginning action in applicable EOPs such as E-1, Loss of Reactor or Secondary Cooling (i.e. addition of the time limit was not expected to alter any operator actions or completion times). Since a TCA did not exist before, the 5-minute TCA should have been screened in as an adverse change to assumed operator actions and response times as defined in Nuclear Energy Institute (NEI) Guideline NEI 96-07, Section 4.2.1.2. Subsequently, it was identified that the RCP trip action may have to be performed locally. Although not specifically

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	The applicability of setpoint U92 was expanded to include loss of sump recirculation. Revision 1: Adds the scope from Post Issuance Change (PIC) 23529, Revisions 25 and 26 of Calculation SQS20110. These changes all have to do with procedure changes necessary to bring SQN into compliance with NRC Order EA-12-049, events involving an Extended Loss of A/C Power (ELAP). ELAP events are beyond the plant's design basis, and there is no SAR design functions assigned to ELAP actions. The following changes were made: existing setpoints B93, B99, O07, and O08 were revised. The following setpoints were created: O11 Minimum S/G pressure which prevents injection of accumulator nitrogen into the RCS, plus allowances for instrument channel accuracy and nitrogen heatup O12 Minimum S/G pressure for operation of the Turbine-Driven AFW Pump, plus allowances for normal channel accuracy O13 S/G pressure corresponding to RCS saturation pressure at 350°F plus allowances for normal channel accuracy O89 S/G pressure corresponding to the maximum allowable discharge pressure of the FLEX Intermediate Pressure (IP) pump with allowance for instrument accuracy S90 Target feed flow based on removal of decay heat using alternate low pressure feedwater pump T11 Containment design pressure, minus 20% margin U09 Condensate Storage Tank (CST) low level for alternate makeup during an ELAP in plant specific units, plus allowances for normal channel accuracy, or Turbine Driven Auxiliary Feedwater (TDAFW) pump suction pressure V07 Station Blackout (SBO) coping time V08 Time after reactor trip when RCS boration is required using FSI-8, considering time necessary for staging and deployment of alternate boration equipment V89 Time after event initiation for establishing containment cooling X06 Minimum DC bus voltage for vital instruments and control systems	discussed in the FSAR, various credible malfunctions could result in RCPs failing to trip when required. This was previously identified in Condition Report 1115920 and was addressed by adding procedural guidance on locally tripping RCPs. The responses to the 10 CFR 50.59 Evaluation questions indicate that NRC prior approval is not needed.

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	Additionally, setpoint U04 (spent fuel pit level - 725.5') was updated based on new spent fuel pit indication provided by DCN 23195. Revision 2: Adds the scope of PIC 23696, which is Revision 27 of Calculation SQS20110. The changes are to the details of Operator TCAs. Calculation NDQ0063980038 (Refueling Water Storage Tank [RWST] & Residual Heat Removal [RHR] Containment Sump Safety and Operational Limits, RWST Setpoint Required Accuracy & Large Break Loss of Coolant Accident [LBLOCA] / Small Break Loss of Coolant Accident Sump Minimum Levels) was revised due to CR 934663, which was written (in part) due to the apparent lack of analysis of the worst case single failure on the equipment and operator actions needed to accomplish the task of swapping the suction of the Emergency Core Cooling System (ECCS) and Containment Spray Pumps (CSPs) from the RWST to the Containment Sump. The change to Calculation NDQ0063980038 required that Calculation SQS20110 also be revised. The change to Calculation SQS20110 (Emergency & Abnormal Operating Procedure Setpoints) is to the Operator TCAs. TCAs are defined as those operator actions which must be performed to ensure that the acceptance criteria for a SAR accident is met. There are two TCAs affected by the revision to Calculation SQS20110. The first is a change to an existing TCA. There has been a longstanding TCA "Shutdown the CSP at RWST low-low level after Large Break Loss of Coolant Accident (LBLOCA)." This TCA has been revised to state: "Shutdown all pumps taking suction from the RWST at RWST low-low level after SBLOCA or LBLOCA." The revised TCA actually aligns the TCA with Operation's procedures. The change to the TCA was necessary due to the revised analysis (Calculation NDQ0063980038) which brings into Calculation NDQ0063980038 consideration of the worst case single failure during the actions for swapping ECCS/CSP suction from the RWST to the Containment Sump. The reason that the TCA is being revised is to avoid possible pump damage from potential vortexing/air ingestion given the worst case single failure.	

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	The worst case single failure is that one Residual Heat Removal (RHR) block valve (FCV-74-3 or FCV-74-21) will not close during the automatic sump swapover sequence (SQN Design Criteria for Safety Injection System SQN-DC-V-27.3). Previously, stopping pumps other than the CSP was not considered to be a TCA, because the analysis did not consider the failure of one RHR block valve. The failure of an RHR block valve to close causes a much higher rate of flow out of the RWST than was previously acknowledged. The higher flow rate causes the RWST level to drop at a rate such that the existing 8-minute TCA to swap the suction of the Centrifugal Charging Pumps (CCPs) / Safety Injection Pumps (SIPs) to the sump may not be completed prior to the onset of vortexing or air ingestion, in the worst case. As this can happen in a SBLOCA, when the CCPs and SIPs are required for ECCS injection, the TCA required the revision as discussed above. The other change to Calculation SQS20110 is the creation of a new TCA. The new TCA is: restart the CCPs and SIPs following completion of their suction source alignment to the RHR pump discharge IF the CCPs and/or SIPs are stopped during the sump swapover sequence in a SBLOCA. Note that SIP restart is contingent on sufficiently low RCS pressure. This new action was also created due to the revised analysis in Calculation NDQ0063980038 regarding the failure of an RHR block valve to close. With this failure, the existing TCA, to complete the swapover for the CCPs and SIPs to the sump within 8 minutes, is not sufficient to maintain the suction to these pumps. An AREVA analysis (32-9244483-000 - Evaluation of SQN Small Break Loss of Coolant Accident Interruption in ECCS Injection) has been prepared specifically to determine the acceptability of stopping the CCPs/SIPs when the RWST level gets to the low-low setpoint during a SBLOCA, for up to 5 minutes. In a SBLOCA, stopping the CCPs and SIPs means that there is no ECCS injection into the RCS until the CCPs/SIPs suction source is correctly aligned, and the pumps restarted. The AREVA analysis demonstrates that all acceptance criteria are met (i.e., the Peak Clad Temperature remains below 2200°F, and no core damage will occur) if there is no ECCS injection for up to 5 minutes in a SBLOCA at sump swapover. There is an increase in Peak Clad Temperature (PCT) from the previous analysis, however.	

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	The following is excerpted from the conclusions of AREVA Analysis 32-9244483-000: In conclusion, although the limiting break changes from one with a 9.76-inch equivalent diameter (PCT of 1470°F) to one with a 3.00-inch equivalent diameter (PCT of 1621°F), the acceptance criteria of 10 CFR 50.46(b) (1)-(4) are met with a 5 minute interruption in the ECCS flow starting at 15 minutes after break initiation. • The calculated maximum fuel element cladding temperature is 1621°F, less than the 2200°F limit of the criterion. • The maximum local cladding oxidation is 0.95%, less than the 17% limit of the criterion. • The maximum core-wide oxidation is 0.0345%, less than the 1% limit of the criterion. • No hot rod rupture is predicted; therefore core cooling is not degraded from blockage of flow channels. The coolable geometry requirements of the criterion are met. Due to the fact that the revised analysis accepts an interruption in ECCS flow, leading to a significant change in calculated PCT, the revised TCA is considered an adverse change and is therefore screened in. In conjunction with PIC 23696, Emergency Subprocedure ES-1.3 is being revised to modify the guidance for realigning CSP suction to the containment sump. Realignment of CSP suction to the sump is designated as a TCA for a LBLOCA with a time limit of 5 minutes to restore spray flow (from the time that spray flow is interrupted by stopping pumps for the realignment). The current ES-1.3 step which realigns spray pump suction requires containment sump level to be greater than 18% or 22% for adverse containment conditions (setpoints T08 and T08A from Emergency Operating Instruction (EOI) setpoint Calculation SQS20110) prior to restarting CSPs with suction from the sump. This setpoint used in this step is based on test results summarized in Calculation NDQ0063980038 which demonstrate the susceptibility of the sump to vortexing for various flow conditions. This sump level verification requirement (contained in Substeps 21.g and 21.m) is unnecessary for any design basis LOCA scenario since the design analysis (Calculation NDQ0063980038) demonstrates that adequate water level would exist. However, the existing sump level verification could	

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	delay restoration of spray flow during a SBLOCA if worst-case instrument errors exist in the negative direction (resulting in a lower indicated sump level than actual). If the sump level setpoint in the existing step is not met, Operators would be procedurally prevented from restarting spray pumps until sufficient ice melt has occurred to raise the indicated sump level above the setpoint (CR 1152907). Deletion of this unnecessary level verification is appropriate to prevent a possible delay in spray flow restoration during a design basis SBLOCA and to save time (to provide additional margin with respect to the 5 minute time limit). In the event of a beyond design basis condition which causes inadequate sump level, the worst anticipated consequence is vortexing; test results (contained in the TVA Water Systems Development Branch Report WR28-3-45-127 - Vortexing Propensity of the RHR Sump at SQN) support the conclusion that this phenomenon would not be expected to cause any equipment damage or other intolerable effects. This change is being screened in due to the fact that it removes a procedural barrier which protects the spray pumps against operation with inadequate sump level. Since the spray pumps and ECCS pumps share a common suction point from the sump, the ECCS pumps are also potentially affected by this change. Even though Revision 0 of the screening review determined that the procedure was not adversely affected, this revision screens in the addition of the 5-minute TCA. Since a TCA did not exist before, the 5-minute TCA should have been screened in as an adverse change to assumed operator actions and response times as defined in NEI 96-07, Section 4.2.1.2.	
DCN 23639, Revision A	This 10 CFR 50.59 screening review will address the installation of a new digital Distributed Control System (DCS) that will replace the existing obsolete analog control systems. The analog control components will be replaced with a digital control system manufactured by Invensys Systems Inc. (formerly the Foxboro Company), called Foxboro Intelligent Automation (I/A) Series System, for "Intelligent Automation." The new system will not only correct the obsolescence issue but will also eliminate a multitude of single point failures which will improve the reliability of the major control systems on SQN Unit 2. The use of redundant power supplies, redundant signal processors, and redundant signal paths make the system more reliable from a hardware standpoint. The division of software processing tasks across multiple control processors maintains functional diversity within the software. This DCN will replace	The new digital DCS system replaces existing analog components for balance of plant (BOP) control systems and reduces many A single point failure (SPF) vulnerabilities existing in the current analog system. System reliability is improved through the use of automatic signal selection from multiple control signal inputs. The new system provides redundant inputs, redundant processors, networks, and power supplies. The new

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	balance of plant analog controls associated with pressurizer pressure and level, rod control, pressurizer Power Operated Relief Valve (PORV) controls, Volume Control Tank (VCT) level control, Low-Temperature Over Pressure Protection (LTOP), Boric Acid Blender (BAB) control, Steam Generators (S/Gs) Atmospheric Relief valve pressure controllers, Steam Dump pressure and temperature controllers, Emergency Core Cooling System (ECCS) Cold Leg Accumulator (CLA) Nitrogen Vent controller, Residual Heat Removal (RHR) heat exchanger flow controllers, S/G blowdown flow controller, Hotwell (HW) level dumpback and make-up controllers, Generator hydrogen heat exchanger temperature controller, and Main Turbine Oil Tank (MTOT) temperature control systems and makes other changes to eliminate many existing single point failures present in the existing analog system. DCS OVERVIEW AND DESIGN DESCRIPTION The basic components of the DCS are dual, redundant, fault-tolerant processor pairs, redundant power supplies with diverse power sources, redundant communications networks, and redundant operator workstations. Redundant field bus modules (FBMs) are utilized for critical inputs and outputs. The system is designed such that the control system functions most critical to safe and reliable plant operations are not affected by the failure of a single device or component. A control group includes a control processor (CP) pair and its associated input/output (I/O) FBMs. The DCS numbers each control group uniquely. The group's number is used to reference the control group and processor pair. A control group is generally referenced by CP## (example CP06) where ## is the control group's unique number. The control groups are segmented to maintain independence between redundant control functions and to limit control system failure impacts to the plant. A segmentation analysis has been performed to ensure the worst case DCS failures are bounded by the existing Updated Final Safety Analysis Report (UFSAR) analyses. This DCN adds 10 control groups to the 4 existing control groups installed for digital feedwater control. The following table lists the primary functions of each of the new DCS control groups.	system is designated as "Quality Related" and is designed to meet Quality Related requirements. The reliability of the DCS is superior to the old analog system. The modification does not negatively impact any system, structure, or component (SSC) that is important to safety nor does it adversely impact the consequences or the frequency of a malfunction. The new DCS does not create a new type of malfunction or accident. The new DCS reduces the likelihood of failures and their consequences by providing a more reliable and redundant control system. In addition, this modification provides the capability to reduce manual operator actions and adds greater opportunity for assessment, monitoring, and response. The upgrade to DCS results in overall improvement in the plant and the ability to function with individual devices out of service. The DCS provides for use of additional input signals for control. The DCS will continue to maintain function with the loss of a single input for control loops with multiple inputs. In the case of failure of a single input, the last good value prior to the failure will be used. The DCS will provide an alarm on the DCS Visual Display Unit (VDU) for failure of any input. The DCS is powered from redundant power sources, one being from battery

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	Group Control Group Primary Functions/ Controls 05 Miscellaneous Nuclear Steam Supply System (NSSS); RHR A Flow; CLA N_2 Flow; S/G blowdown; C-7 Interlock 06 Rod Control; Turbine Runback Control Logic 07 Steam Generator 1 Atmospheric Relief Valve (ARV); HW Level; RHR B Flow; MTOT Oil Cooling 08 Steam Generator 2 ARV; RHR Bypass Flow 09 Steam Generator 3 ARV; Main Generator H2 Cooling 10 Steam Generator 4 ARV; Condenser Vacuum Pump; No. 3 & 7 Heater Drain Tanks 11 Condenser Steam Dump 12 Pressurizer A (Pressure, Level, Charging, Letdown, Loop 1 Spray, Cold Overpressure Mitigation System [COMS]) 13 Pressurizer B and Chemical & Volume Control System (CVCS) (Pressure, Level, Charging, Letdown, Loop 2 Spray, COMS) 14 Boric Acid Blender The upgraded control system consists of Foxboro I/A Series Automation hardware and software that incorporate all the functions of the existing systems. The new control system provides for additional features and control schemes to improve system performance and reliability, improve operator interface, and simplify system and component maintenance. While the existing system is comprised of many discrete components, the upgraded control system will incorporate the function of various existing signal processors and modifiers into a redundant fault tolerant control system. The system hardware and logic are designed such that the failure of any single input or component will not result in a perturbation to a unit critical control system. Internal system communication is dual redundant. One channel is in service at a time, with the other in backup. In the presence of failure of an active channel, the modules will switch communication into the backup channel and signal a diagnostic message. The two internal communication networks of interest are (1) the I/O channel between I/O modules and their CPs and (2) the control network between control stations and workstations. The I/O channel is implemented using the Standard Microcontroller	backed vital power boards, thus for loss of any single power source, the DCS will continue to maintain control. The signal outputs to plant control devices, such as valves, use redundant field bus modules (FBMs) such that should one FBM fail the other FBM maintains control of the device. Reliability data based on the operational history of Foxboro I/A systems in service throughout a wide range of industrial applications is extremely high. A SPF analysis was performed to evaluate the failure modes and effects of the new components including software failures. Many of the SPFs existing in the analog system were eliminated by DCS implementation. Because two redundant processors in a control group have the same software in common, the software is considered a SPF. As previously discussed, the software is designed to limit the likelihood of software errors that could result in common cause failure. The SPF analysis concluded the software SPF is acceptable based on thorough testing of the software, software control by an approved quality assurance program, and an extensive operating history. Functional diversity and independence are provided through the segmentation of control functions in different control groups and processor pairs. For example, the S/G

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	System Serial Control Bus protocol (from the Institute of Electrical & Electronics Engineers [IEEE] Std. 1118) over copper or fiber optic media. The control network is implemented using switched ethernet technology with fiber optic media (the IEEE Standard 802.3 family of standards - a collection of standards for ethernet networks). The process inputs to the new DCS will be validated by software; faulty inputs are automatically bypassed by the DCS if any validation criteria are not met. It will also be possible to manually bypass any process input from the software to support maintenance activities. The primary operator interfaces for the new DCS will be the MCR Visual Display Units (VDUs) and associated keypads and pointing devices, selected indicators, and associated manual/auto (M/A) handstations. The new VDUs and M/A handstations will provide for manual control similar to the existing controllers. However, the actual automatic control algorithms and outputs are part of the redundant I/A system itself, with the VDUs serving as "dumb terminals." That is, failure of the VDUs will have no effect on the operation of the DCS. Manual control capability (integrated control and display) is provided by the operator workstation central processing units (CPUs). With installation of the DCS, failure of a single input will initiate appropriate alarms, but the DCS will automatically kick out a failed or outside deviation limit signal. This channel can be bypassed via the Operator Work Station (OWS) or Engineering Work Station (EWS). Failure of 2 inputs (in a 2 out of 3 scheme outside Eagle 21) will cause control to shift to manual (MAN) and bring in corresponding MCR annunciators and computer alarms. In each case, plant perturbation will be minimized or eliminated. Communications to/from the CPs is deterministic, with the master CP acting as bus master and polling each channel. CPs execute in a deterministic manner, with the execution period (the "Basic Processing Cycle") selected to meet plant system stability requirements. Control algorithms are implemented using standard, precompiled software blocks, described below under "I/A Software." Field values may be communicated across the control network directly for display and/or historical recording.	ARV controls are segmented such that each S/G ARV is on a separate DCS processor pair. Segmentation of control functions also limits the impact of software common cause failures. A segmentation analysis has been prepared to further analyze the separation of signal processing and control. The segmentation analysis concluded that the design of the DCS does not introduce control system failures which could result in events not bounded by the UFSAR safety analysis or an event not analyzed in the UFSAR.

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	Field values are also involved in control calculations with calculation results written to field outputs and/or displayed or recorded. CPs operate in pairs, with one serving as a master and the other as a backup. The master controls both communication channels, polls field devices, and executes the control algorithms. The backup module receives field data, computes the same control algorithm as the master, and offers its calculated results for comparison to the master. Results of the two modules' calculations are compared in a bit-wise manner; agreement permits the master to write its outputs. In the unlikely event of a bit-wise disagreement, the modules emit a message to start diagnostic action and initiate self-diagnostics. The system is hardened against common mode failure causes as follows: • Environmental factors: devices intended for field installation are designed and qualified for continuous operation in high temperatures up to 65°C. Workstations and network switches are installed within controlled environments Auxiliary Instrument Room/Main Control Room (AIR/MCR) that support their continued operation. • Signal noise immunity: current and voltage signals are measured with input circuits that have large common mode rejection and differential noise immunity. • Electromagnetic Interference/Radio Frequency Interference (EMI/RFI) Compatibility: the system is capable of operating in industrial environments without special design measures. The use of fiber optic media supporting communications between the Cable Spreading Room and the AIR and other locations significantly reduces the potential failures experienced in previous generations of digital control systems. • Malicious intrusion: sensitive entry points to the system are protected both physically and through limiting inputs to allowable values and emitting diagnostic messages in the presence of out of range conditions. I/A HARDWARE The system's hardware has been designed to operate in rugged industrial environments in a wide variety of facilities: Nuclear and fossil power plants,	

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	petrochemical refineries worldwide, and basic material processing plants such as blast furnaces and foundries. • I/O modules: I/O modules interact with electrical signals (voltage, amperage, resistance, milli-volt) or data signals formed according to common industrial or customized protocols. I/O modules are mounted on passive backplane devices (termed "baseplates") that interconnect power, communications, and field signals to designated positions. Module replacement may be accomplished within several minutes. Baseplates accept redundant 24 VDC power feeds; they accept and transmit with a single CP Pair. The I/O modules and associated baseplates and power supplies are designed to operate in environments up to 65°C. • Controllers are termed "Control Processors" or CPs. The Field CP selected for this project performs the following functions: • I/O Communication: deterministic communications to the redundant I/O channel are controlled by each control processor. The master control processor initiates and acknowledges communication; the backup control processor monitors the communication. • Control Computation: control algorithms are executed by a special purpose processor that accepts control configuration and executes control blocks in their predetermined order. • Control Network Communication: communication to and from other control stations/workstations on the control network is supported through a communications controller. • Control Network Communications: the control network is composed of redundant managed ethernet switches, communicating at fast and gigabit ethernet speeds. The communications medium is primarily fiber optic, reducing the potential for noise to degrade communications quality. Network fault detection is integrated with other diagnostics. I/A SOFTWARE Software is divided between "System Software" and "Application Software." System software is developed in Invensys or is procured from third parties and is integrated into the delivered hardware modules. System software may be installed into memory	

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	devices within each hardware module (Programmable Read-Only Memory - PROM) or may be placed on storage media and downloaded as a part of system operation. System software configuration is controlled by a version scheme: hardware module revision level includes the PROM version; system software is released according to a version/level classification scheme. System software also includes utility programs that configure the control network, control algorithms, graphics, historian, alarms, and reports. Software configuration is performed in a manner that is verifiable against the approved system design. Starting with the design, e.g. the control algorithms, a configuration is produced that implements the design by interconnecting previously tested components and characterizing them appropriately with verified plant data. The output of the utility programs is a set of binary files that represent the system's design. These files are termed "application software" and are in one-for-one correspondence with the system's design. The files are installed (downloaded) into the appropriate modules and the system then operates as designed. The files may also be interpreted into human readable form, permitting installation verification at any time. The design process has been accomplished so as to incorporate good practices for software verification and validation. External Communication The new system uses a 1 GB switched ethernet backbone for process control. It deploys standard 100 MB/1 GB switched ethernet (Gigabit Ethernet IEEE 803.w) technology in a redundant/fault tolerant configuration to provide a secure fiber optic network with highly reliable message delivery between equipment stations. The DCS is connected via network to the plant computer. Firewalls between the DCS and plant computer systems limit the volume of data traffic and ensure that ICS Network events, such as data storms, do not impact the DCS control functions. There is no digital communication between the DCS and the protection systems. The Eagle 21 protection system provides many of the analog process control inputs to the	

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	DCS through qualified isolators. This communication is only one way from Eagle 21 to the DCS via analog signals. This DCN upgrades the DCS-ICS interface equipment including firewalls, and the data acquisition system/intrusion detection system (DAQ-IDS) server. Upgrade of the hardware is to improve network capacity and reliability. The design function of the DCS-ICS equipment is not changed. A new power circuit is added to 2-R-154 to supply power for the increased electrical load.	
ECP SQN-18-240, Revision 0 and 1	The proposed activity is a design change which increases the allowable total instrument uncertainty for the Reactor Vessel Level Indication System (RVLIS) Upper Plenum Range instrument channels for Units 1 and 2. This modification is necessary to resolve Condition Report (CR) 1465649, which documents the fact that the Unit 2 Train B RVLIS upper plenum level transmitter failed the static pressure check during periodic off-line calibration in the Unit 2 Cycle 22 Refueling (U2R22) Outage. This portion of the calibration procedure checks for a shift in RVLIS upper range transmitter output when varying pressure is applied to both sides of the sensor bellows and prescribes a maximum allowable shift in the output signal. The test results indicate that, most likely, there is a small amount of trapped air in the sensing (capillary) lines associated with the upper range instrument. Removal of the trapped air is challenging and has not been accomplished. Based on input from the RVLIS vendor (Westinghouse) and a review of previous test data, air bubbles in the sensing lines can migrate and come out of solution in different portions of the capillary tubing. Calculations by TVA Engineering have shown that the potential errors introduced by the observed condition are relatively small and would result in a minor increase in total allowed error. To address this specific condition on Unit 2 and to alleviate future calibration problems on both trains and on both units, the allowable total instrument uncertainty for RVLIS upper range has been recalculated to include the air entrapment static pressure effect, with a change in the total allowed error from ± 10% of span (equivalent to ± 5.6% of indicated level) to ± 10.7% of span (equivalent to ± 6% of indicated level). In the process of revising and reviewing RVLIS calculations and design documents, the following problems were identified which are corrected in this design change:	The proposed change alters the allowable instrument error for RVLIS upper range and modifies associated EOP setpoints; this change cannot increase the frequency of an accident evaluated in the USFAR. The impact of the increased RVLIS upper range error and related EOP setpoint changes has been evaluated; this change does not result in a more than minimal increase in the likelihood of a malfunction in decay heat removal or natural circulation capability, or any other UFSAR-described function. The proposed change does not impact the radiological consequences of any accident, including a design basis steam line break or S/G tube rupture. RVLIS is not relied upon to mitigate any malfunctions to SSCs important to safety, as described in the UFSAR; therefore, there is no increase in the radiological consequences of any malfunction. This RVLIS change cannot create the possibility for any different type of accident

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	• CR 1470143 identified an error in in the calculation of Emergency Operating Procedure (EOP) setpoint J01 (RVLIS Upper Range Value Indicating Upper Head Region is Full). Specifically, the current value of this setpoint was calculated by subtracting the current instrument error (10%) from the calculated RVLIS reading with the reactor vessel full. The instrument error correction which is subtracted in this calculation should have been adjusted to account for the span of the upper range rather than subtracting the uncorrected total error percentage. • CR 1471124 identified legacy design errors in which RVLIS design documents used an incorrect value for the RVLIS upper range span. The affected documents determined instrument uncertainty values and EOP setpoints for RVLIS upper range using an assumed measurement span of 64% to 120% (which is consistent with the range of the associated indicators in the MCR). However, based on sensing line tap elevations, the RVLIS upper range transmitter is only designed to indicate up to 104% and cannot provide a valid signal above this level. Therefore, the transmitter span used in the affected calculations should be 64-104%, consistent with the maximum measured level. The recommended methodology for calculating EOP setpoints, including the method of applying instrument uncertainty is provided by the Westinghouse Owners Group (WOG). As a result of the change to the allowed instrument uncertainty AND the correction of the errors in the existing calculations, this design change will require a revision to Calculation SQS20110 to modify the following EOP setpoints: J01 The RVLIS upper range value corresponding to reactor vessel head full with instrument error subtracted. Changing from current value of 104% to proposed new value of 98%. Used in inventory status tree to initiate yellow path (optional) EOP FR-1.3, Voids in Reactor Vessel; used in Emergency Subprocedure ES-0.3 (Natural Circulation Cooldown with Steam Void in Vessel with RVLIS) to indicate steam void which requires raising RCS pressure to collapse; used in numerous EOPs to initiate actions to mitigate adverse effects when restarting RCPs with void.	nor any unanalyzed combination of accidents. The proposed change does not create the possibility of a malfunction to RVLIS nor any other SSC important to safety with a different result. This RVLIS change does not impact any design basis limits for the RCS, reactor fuel clad, or containment. This change has no impact on evaluation methodologies described in the UFSAR. Therefore, this activity may be implemented per plant procedures without obtaining a License Amendment.

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	J99 (equivalent to Westinghouse Owner's Group setpoint K.03) - The RVLIS upper range value corresponding to top of the hot legs with instrument error added. Changing from current value of 78% to proposed new value of 75%. Used in Emergency Contingency Action Procedure ECA-3.2 (SGTR with LOCA-Saturated Recovery) as a limit on void growth. K04 RVLIS upper range value corresponding to top of the hot legs with no uncertainty applied. Changing from current value of 71% to proposed new value of 69%. Used in Emergency Operating Procedure EOP-0.3 (Natural Circulation Cooldown with Steam Void in Vessel with RVLIS) as a limit on void growth. Numerous EOPs and three Abnormal Operating Procedures (AOPs) which use these setpoints will be revised to incorporate the new setpoint values. RVLIS calibration procedures will be revised to modify the acceptance criteria for the static pressure check. Also, the as-found and as-left acceptance criteria for the calibration of some RVLIS upper range components will be tightened to correct scaling errors. The identified change to setpoint J01 will also require a change to the Safety Parameter Display System (SPDS) Inventory Status Tree display to reflect the new RVLIS setpoint value. This change will be made in accordance with the software change process (Procedure NPG-SPP-12.7, Computer Software Control).	
ECP SQN-19-308, Revision 0	The scope of this change is to remove Emergency Diesel Generator (EDG or D/G) 2A-A's Voltage Overshoot Reduction Device (VORD) from service by abandoning terminal board (TB1) and removing terminal board (TB2). The VORD was damaged due to a ground fault incident and is currently designated as a degraded non-conforming issue, documented in CR 1041370. The VORD is an obsolete part and reverse-engineering attempts have been unsuccessful in matching the existing critical characteristics. During the previous two performances of 2-SI-OPS-082-026.A (Loss of Offsite Power with Safety Injection D/G 2A-A Test) with the VORD out of service, EDG 2A-A did not meet the numerical values stated in Regulatory Guide 1.9, Revision 1 (Application & Testing of Safety-Related Diesel Generators in Nuclear Power Plants). UFSAR, Section	The EDGs original design did not contain VORDs, and the minimal overshoot does not impact load sequencing or downstream equipment. It is concluded that; this change cannot increase the frequency of an accident evaluated in the UFSAR, does not increase the likelihood of malfunction of SSCs, and does not result in an increase of consequences by malfunction of an SSC important to safety

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	8.3.1.2.1 (Standby Alternating Current (AC) Power System), currently states that the EDG voltage is restored to within 10 percent of nominal within 60 percent of each load sequence time interval.	or for an accident previously evaluated. The removal of the VORD cannot create the possibility for an accident of a different type or malfunction of an SSC important to safety with a different result than evaluated in the UFSAR. Fission product barrier design basis limits are not exceeded or altered as the function of the EDG is not impacted by the VORD removal.

Temporary Modification (TMod)	DESCRIPTION	SAFETY ANALYSIS
TMod SQN-2-2018-092-001, Revision 0	Sequoyah has two (2) Thermo Fisher Gamma-Metrics Source & Intermediate Range neutron flux detectors, Channel I (N31 & N35) and Channel II (N32 & N36). The detectors are Dual Chamber Unguarded Fission Chamber detectors, which contain two identical fission chambers. One fission chamber provides a signal to Source Range pre-amplifier A3 and the other fission chamber provides a signal to Source and Intermediate Range pre-amplifier A4. The output signals of the two pre-amplifiers are summed together to produce the overall Source Range signal which is represented over the logarithmic range of 10^0 to 10^6 counts per second (cps). The output of pre-amplifier A4 is also used to develop the logarithmic Intermediate Range signal (N35/Channel I and N36/Channel II) of 10^{-8} to 200% Rated Thermal Power (RTP). Troubleshooting has identified that the fission chamber circuit that feeds pre-amplifier A3 in Wide Range Amplifier Assembly Channel I is the source of noise/signal spikes, which are adversely affecting the operation of Source Range Channel N31.	It is shown that operation on one fission chamber instead of two will not impact any Safety Evaluation, will not impact the ability to provide Operations with counts per second (cps) indication, nor will it impact the ability to perform the intended reactor trip. The conclusion of the evaluation is that the proposed activity may be implemented under 10 CFR 50.59, without required prior NRC review or approval.

Temporary Modification (TMod)	DESCRIPTION	SAFETY ANALYSIS
	Proposed Activity Description: This activity is a Temporary Modification that involves disconnecting the fission chamber input from Source Range Detector N31 pre-amplifier A3 in Wide Range Amplifier Assembly Channel I. This will prevent any noise/signal spikes associated with the A3 fission chamber/cabling from providing an input to Source Range indication for detector N31. The remaining fission chamber is not impacted and will continue to provide a signal to pre-amplifier A4 in Wide Range Amplifier Assembly Channel I, which also provides the signal for N35 Intermediate Range Channel I. The connection on Cable 2NM1383 to pre-amplifier A3 in Wide Range Amplifier Assembly Channel I will be disconnected, and left disconnected for the duration of this activity.	

SAR	DESCRIPTION	SAFETY ANALYSIS
FSAR Section 13.9.2, Revision 0	Commitment 32.C is to be removed from UFSAR Section 13.9.2 commitments. Following discussions with the Nuclear Regulatory Commission (NRC), the Tennessee Valley Authority (TVA) added the statement to the UFSAR Section 13.9.2 and now removes the statement under 10 CFR 50.59. The change removes a commitment statement to "Revise the Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS) procedures to require volumetric inspections be used to confirm the absence of flaws greater than the maximum allowable flaw depth from the flaw tolerance analysis. Industry efforts are continuing for demonstrations of ultrasonic inspection capabilities on CASS. In the absence of successful completion of an industry performance demonstration program on applicable CASS components, the best available inspection methods will be employed for these inspections. Possible methods for performing ultrasonic examinations of CASS piping are described in the American Society of Mechanical Engineers (ASME) Code Case N-824." TVA's statement that "volumetric inspections will be used to confirm the absence of flaws greater than the maximum allowable flaw depths" is in excess of NRC guidance contained in the Generic Aging Lessons Learned (GALL) Revision 2, Aging Management Program (AMP) XI.M12 -Thermal Aging Embrittlement of CASS.	TVA has now completed the flaw tolerance analysis for susceptible CASS components in accordance with the methodology described in Electric Power Research Institute's (EPRI) Materials Reliability Program Report Number MRP-362, Revision 1 and ASME Section XI approved Code Case N-838. The analysis found that for the maximum predicted fatigue crack growth for Sequoyah Units 1 and 2 piping and elbow components, the postulated flaw will not grow beyond the tolerable flaw size after 60-years of crack growth. The results of the probablistic fracture mechanics analysis indicate that the susceptible CASS components for Sequoyah Units 1 and 2 are very flaw tolerant.

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	A volumetric inspection to confirm the absence of flaws is not required since the GALL Revision 2 guidance states either qualified inspections or a flaw tolerance evaluation is required. In addition, there are currently no available qualified techniques for a volumetric inspection. Also, "best available" inspection techniques do not provide a high level of confidence that a flaw greater than that assumed in the Sequoyah Nuclear Plant (SQN) flaw tolerance analysis would not exist. Such an inspection would also be at a significant expense of radiation dose and provide limited value due to the uncertain results.	A volumetric inspection to confirm the absence of flaws is not required since the GALL Revision 2 guidance states either qualified inspections or a flaw tolerance evaluation is required. In addition, there are currently no available qualified techniques for a volumetric inspection. Also, "best available" inspection techniques do not provide a high level of confidence that a flaw greater than that assumed in the SQN flaw tolerance analysis would not exist. Such an inspection would also be at a significant expense of radiation dose and provide limited value. Therefore, this Evaluation has determined that the proposed change may be implemented without obtaining a License Amendment.

DOCUMENT NUMBER/72.48 EVALUATION TRACKING NUMBER	DESCRIPTION	SAFETY ANALYSIS
None		