TVA

Tennessee Valley Authority, 1101 Market Street, Chattanooga, Tennessee 37402

July 5, 2013

10 CFR 50.59
10 CFR 72.48

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

Sequoyah Nuclear Plant, Units 1 and 2
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**

In accordance with Title 10 of the Code of Federal Regulations (10 CFR) 50.59(d)(2) and 10 CFR 72.48(d)(2), enclosed is the Sequoyah Nuclear Plant, 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 evaluations occurred since the previous submittal dated December 16, 2011.

There are no commitments contained in this letter. If you have any questions concerning this issue please contact Michael McBrearty at (423) 843-7170.

Respectfully,

J. W. Shea
Vice President, Nuclear Licensing

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

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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
20673A	This 50.59 was originally submitted on May 27, 2004. On August 17, 2012, this evaluation was revised to address field changes to the modified steam generator compartment roof design as a corrective action to Problem Evaluation Report (PER) 532450. The evaluation was not affected. The scope of DCN D20673 addresses the modifications to the Shield Building concrete dome, the steel containment vessel (SCV), and the steam generator compartment concrete roofs that are necessary to support removal of the original steam generators (OSGs) and installation of the replacement steam generators (RSGs). To facilitate removal of the OSGs and installation of the RSGs, openings will be cut in the concrete Shield Building dome, the SCV, and the steam generator compartment roofs. The two openings in the Shield Building concrete will be restored by splicing new reinforcing bar to the existing reinforcing bar using Bar-Lock mechanical couplers, Cadwelds and/or welding and pouring new concrete to close the openings. The two openings in the SCV will be restored by welding the cut steel pieces back in place. The four openings in the steam generator compartment roofs will be restored by reinstalling the cut sections of the roof in their respective holes and using through-bolted connection frames to hold the concrete sections in place.	The proposed modifications do not increase the frequency or likelihood of accidents or malfunctions, increase the consequences of an accident or malfunction, or create a new type of accident or malfunction. The SCV and Shield Building are fission product barriers, but they will be restored to meet their design bases, so no design basis limits will be altered or exceeded. The use of Bar-Lock reinforcing bar splices to restore the Shield Building is a different method of reinforcing bar splicing that has not been previously approved and is subject to Nuclear Regulatory Commission (NRC) approval prior to entering Mode 4. NRC approval for use of the Bar-Lock reinforcing bar splices, Topical Report 24370-TR-C-001, was provided via NRC letter dated March 13, 2003. Amendment No. 283 to the Unit 1 Operating License approving the methodology submitted in Topical Report 24370-TR-C-001 was issued by the NRC on April 24, 2003. The use of through-bolted connection frames to restore the roof of the steam generator compartments has been shown analytically to be acceptable. However, the methodology used deviates from that described in the Updated Final Safety Analysis Report (UFSAR), and therefore, requires NRC approval prior to entering Mode 4. NRC approval of the methodology used to analyze the through-bolted connection frames, Topical Report 24370-TR-C-003, was provided via NRC letter dated April 18, 2003. Amendment No. 284 to the Unit 1 Operating License approving the methodology submitted in Topical Report 24370-TR-C-003 was issued by the NRC on April 25, 2003.

Design Change Notice (DCN)	DESCRIPTION	SAFETY ANALYSIS
22471A	The scope of DCN D22471A, Sequoyah Nuclear Plant (SQN) Unit 2 Reactor Building Structural Modifications, addresses the modifications to the Shield Building concrete dome, the SCV, and the steam generator compartment concrete roofs that are necessary to support removal of the OSGs and installation of the RSGs. The four openings in the steam generator compartments roofs will be restored by reinstalling the cut sections of the roof in their respective holes and using through-bolted connection frames to hold the concrete sections in place. Technical justification for the design of the restored steam generator compartment roofs is provided in Technical Report SQN2-SGR-TR2, "Sequoyah Unit 2 Steam Generator Replacement - Steam Generator Compartment Roof Modification Technical Report." The activities implemented by DCN D22471A will restore the openings in the Shield Building concrete by splicing new reinforcing bars to the existing reinforcing bars using Bar-Lock mechanical couplers, Cadwelds and/or welding, and pouring new concrete to close the openings. Technical justification for use of Bar-Lock mechanical couplers for this purpose is provided in Technical Report SQN2-SGR-TR3, "Sequoyah Unit 2 Steam Generator Replacement - Alternate Rebar Splice - Bar-Lock Mechanical Splices Technical Report."	The activities implemented by DCN D22471A will restore the openings in the Shield Building concrete by splicing new reinforcing bars to the existing reinforcing bars using Bar-Lock mechanical couplers, Cadwelds and/or welding, and pouring new concrete to close the openings. The proposed modifications do not increase the frequency or likelihood of accidents or malfunctions, increase the consequences of an accident or malfunction, or create a new type of accident or malfunction. The SCV and Shield Building are fission product barriers, but they will be restored to meet their design basis limits, and therefore, no design basis limits will be altered or exceeded.

Design Change Notice (DCN)	DESCRIPTION	SAFETY ANALYSIS
22478A	In support of the SQN Unit 2 Steam Generator Replacement (SGR) Project, DCN D22478A, “SG Vessel Replacement Modification,” provides for the replacement of the Unit 2 original construction Model 51 Westinghouse steam generators (termed OSGs) with Westinghouse Model 57AG+ RSGs. The primary basis document for evaluating the design and manufacture of the RSGs is AREVA Document No. 77-9142036, “Replacement Steam Generator Report for Tennessee Valley Authority (TVA) Sequoyah Unit 2,” hereafter referred to as the "Unit 2 RSG Report." DCN D22478A includes changes to the UFSAR that will be made to reflect the Unit 2 RSG characteristics, parameters, and other descriptive details.	The assumptions, methods of evaluation, and conclusions of this Unit 2 RSG Report are based on a comparison against the similar report prepared by AREVA's predecessor entity, Framatome ANP (Framatome ANP Document No. 77-5016198-01, “Replacement Steam Generator Report for Tennessee Valley Authority Sequoyah Unit 1.”) The Unit 2 RSG Report describes RSG design and fabrication, and evaluates the installed operation of the RSGs in SQN Unit 2. The contents of the report support the conclusion that the RSGs will support normal and transient plant operation with no adverse effects, and that the existing licensing basis is maintained with the RSGs. Utilizing the Unit 2 RSG Report and other supporting information, the 10 CFR 50.59 Evaluation performed for DCN D22478A concluded that no 10 CFR 50.59(c)(2) criteria exists that would require a License Amendment Request.

Design Change Notice (DCN)	DESCRIPTION	SAFETY ANALYSIS
22566A	DCN 22566A performs modifications on the Motor Operated Valves (MOVs) listed below in order to comply with Generic Letter (GL) 96-05, which was issued by the NRC to establish a periodic verification program to provide confidence in the long term capability of MOVs to perform their design basis safety functions. The modified MOVs are: 1-FCV-63-25 -26 -39 and -40 (Centrifugal Charging Pump Injection Tank (CCPIT) isolation valves) 1-FCV-63-156 and -157 (Safety Injection (SI) Pump Isolation valves) In addition, the DCN revised UFSAR Table 6.3.2-1, "Emergency Core Cooling System Component Parameters," to document the increase in maximum stroke time from 10 to 25 seconds for the CCPIT isolation valves, which is documented and evaluated in AREVA Document No. 51-9155373-000, AREVA letter 11-00689, and TVA Letter S-415.	The evaluation has determined that the proposed changes do not result in the possibility of new accidents or malfunctions, and do not result in the increased frequency of accidents or malfunctions evaluated in the UFSAR. The change does not result in more than minimal increases of the consequences of an accident or malfunction and does not result in an unacceptable departure from methodologies used to establish the design basis and safety analysis. In addition, no design basis limits for fission product barriers are exceeded or altered by this change, and the Technical Specifications (TSs) are not affected. The required safety function of the CCPIT valves is to open to establish high-head Emergency Core Cooling System (ECCS) injection. However, these valves are also required to close electrically to terminate ECCS injection during an inadvertent/spurious SI or a steam generator tube rupture. These two events have time critical actions requiring CCPIT isolation which are specified in calculation SQS20110 Appendix 1A, "Emergency and Abnormal Operating Procedure Setpoints." During a loss-of-coolant accident (LOCA) under shutdown conditions (Mode 4), operator action is relied upon to establish ECCS flow, by aligning one Centrifugal Charging Pump (CCP) to inject into the reactor coolant system (RCS) via the CCPIT flow path. This stroke time increase will not prevent operators from completing these actions within the required time. The SI Pump Isolation valves are potentially subject to pressure-locking conditions, in which the inlet and outlet of the valve disc are depressurized with pressure remaining in the valve bonnet, preventing it from opening. To address the potential for this condition, the DCN revises the pressure locking calculation and required thrust calculations for these valves to demonstrate that sufficient margin exists to ensure operation of the valves. No parts are replaced on these valves.

Design Change Notice (DCN)	DESCRIPTION	SAFETY ANALYSIS
22621B	DCN 22621B modifies the Emergency Gas Treatment System (EGTS) by installing new fuses and relays with contacts placed in the 125 volt direct current (VDC) control circuits for the Shield Building Isolation Valves in the EGTS exhaust damper flow path. This change is made to eliminate the potential for a single transmitter failure causing inadvertent automatic aligning of the stand-by EGTS dampers to the Unit 2 annulus while the normal damper train is in service (i.e., both damper trains aligned to Unit 2 annulus simultaneously). This could establish an excessive vacuum in the annulus, resulting in exceeding allowable containment leakage during an accident and potentially exceeding allowed offsite dose limits. The existing calculation reflected in the UFSAR uses 95 percent filter efficiency with a five second containment purge isolation time. This change documents a filter efficiency of 99 percent and purge isolation time of 5.5 seconds, as allowed by Regulatory Guide 1.52, "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post Accident Engineered Safety-Feature Atmosphere Cleanup Systems in Light Water-Cooled Nuclear Power Plants."	This DCN eliminates the potential for inadvertent aligning of the stand-by EGTS damper train to the Unit 2 annulus as a result of a pressure transmitter failure in the stand-by train. This modification precludes automatic alignment of the stand-by EGTS train while the other train is aligned as a result of a stand-by train transmitter failure or failure of a single pressure sensing device. This modification does not modify the initiating signal for EGTS, nor change the swapover criteria for aligning the stand-by train. The change made under this modification will be that swapover will now require 2-out-of-2 logic from independent pressure sensing devices to verify swapover is required. This modification will not prevent swapover to stand-by damper control, if required, due to an actual failure of the normal train. It maintains the original design basis of EGTS. Dose increases remain well within the regulatory limit with some values actually decreasing (increase in margin). No increase was more than minimal.

Design Change Notice (DCN)	DESCRIPTION	SAFETY ANALYSIS
22621B (Cont)	In addition, in Revision 7 of dose calculation SQN-APS3-067, "Offsite and Control Room Operator Doses Due to a MHA [Maximum Hypothetical] LOCA With a Maximum Allowable Annulus Inleakage," contribution due to ECCS leakage outside of containment following a LOCA is addressed. Revision 7 of SQN-APS3-067 supports amending the current UFSAR Tables 15.5.3-1, 15.5.3-4, and 15.5.3-7 to reflect the changes in the assumed offsite doses, as per the changes in the dose calculation. Also, based on the revision to dose calculation SQN-APS3-100, "Demonstrated Range for Sequoyah Nuclear Plant Radiation Monitors," the UFSAR is revised to reflect new dose ranges for minimum and maximum detectable concentrations resulting from this modification.	

Design Change Notice (DCN)	DESCRIPTION	SAFETY ANALYSIS
22622A	DCN 22622A modifies the EGTS in the 125VDC control circuits for the Shield Building Isolation Valves in the EGTS exhaust damper flow path. This change is made to eliminate the potential for a single transmitter failure causing inadvertent automatic aligning of the stand-by EGTS dampers to the Unit 1 annulus while the normal damper train is in service (i.e., both damper trains aligned to Unit 1 annulus simultaneously). This could establish an excessive vacuum in the annulus, resulting in exceeding allowable containment leakage during an accident and potentially exceeding allowed offsite dose limits. The existing calculation reflected in the UFSAR uses 95 percent filter efficiency with a five second containment purge isolation time. This change documents a filter efficiency of 99 percent and purge isolation time of 5.5 seconds, as allowed by Regulatory Guide 1.52. In addition, in Revision 7 of dose calculation SQN-APS3-067, contribution due to ECCS leakage outside of containment following a LOCA is addressed. Revision 7 of SQN-APS3-067 supports amending the current UFSAR Tables 15.5.3-1, 15.5.3-4, and 15.5.3-7 to reflect the changes in the assumed offsite doses, as per the changes in the dose calculation. Also, based on the revision to dose calculation SQN-APS3-100, the UFSAR is revised to reflect new dose ranges for minimum and maximum detectable concentrations resulting from this modification.	This DCN eliminates the potential for inadvertent aligning of the stand-by EGTS damper train to the Unit 1 annulus as a result of a pressure transmitter failure in the stand-by train. The modification installs new relays with contacts placed in the 125VDC control circuits of the Shield Building Isolation Valves. Addition of the new relay contacts into the isolation valve circuits will create a 2-out-of-2 logic (based on independent pressure measuring devices) to permit automatic alignment of the stand-by EGTS damper train to the annulus. This modification precludes automatic alignment of the stand-by EGTS train while the other train is aligned as a result of a stand-by train transmitter failure or failure of a single pressure sensing device. This modification does not modify the initiating signal for EGTS, nor change the swapover criteria for aligning the stand-by train. The change made under this modification will be that swapover will now require 2-out-of-2 logic from independent pressure sensing devices to verify swapover is required. With the installation of the new relays with contacts in the 125VDC control circuits for the Shield Building Isolation Valves, a failure of the stand-by transmitter will preclude the stand-by train to automatically align while the other train is aligned and in service. Should the normal train fail to properly maintain desired pressure (vacuum) in the annulus (2-out-of-2 logic), the stand-by isolation valves will OPEN and will automatically align the stand-by damper train to maintain proper annulus vacuum. Unless the required logic from the pressure devices is made up, swapover will not occur. Dose increase remain well within the regulatory limit with some values actually decreasing (increase in margin). No increase was more than minimal.

Design Change Notice (DCN)	DESCRIPTION	SAFETY ANALYSIS
22819A	DCN D22819A documents setpoint changes to Design Output Calculation SQS20110, Revision 21, "Emergency and Abnormal Operating Procedure Setpoints," including new Time Critical Operator Actions. Engineering Document Change (EDC) E22819A was being prepared for implementation of Design Output Calculation SQS20110. This is a documentation only calculation change to a design output calculation. This calculation determines the setpoints used in the emergency operating procedures (EOPs), as well as Time Critical Operator Actions. The following changes to SQS20110 are included: Setpoint, [F98], "Temperature that can result in flashing at the RHR pump suction when it is realigned to the Refueling Water Storage Tank (RWST)," is revised to 213 degrees Fahrenheit (°F) from 200°F for sump recirculation, and 238°F at eight percent (RWST) level (from 242°F at 27 percent RWST level). Added two Time Critical Actions (TCAs) and revised one existing TCA. The only situation where these TCAs are applicable is after the occurrence of a LOCA. The applicable LOCA only occurs after the Residual Heat Removal (RHR) system has been aligned for shutdown cooling with the unit in Mode 4.	The described scenario involves the plant actions after a small break LOCA in Mode 4 has already occurred. As such, there are no changes to the frequency of an accident, nor will any new accident be created. The NRC, in TS Change 07-05 (Units 1 and 2 Amendments Nos. 326 and 319), recognized that manual actions are necessary to align the RHR system for ECCS use following a LOCA that may occur in Mode 4, after RHR has been placed in the shutdown cooling mode. Evaluations have been performed that evaluate the actions needed to re-align RHR for ECCS. These evaluations include what the specific actions are, how long they will take to accomplish, and how long a time period is available for these actions to be completed. The evaluation has included examination for any change in the likelihood of occurrence of a malfunction, and for any type of new accident. None were identified. The analysis indicates that the ECCS function and the containment cooling function are successful. Therefore, this change does not result in an increase in the consequences of any accident or malfunction.

Procedure	DESCRIPTION	SAFETY ANALYSIS
FE 42823 and AOP-C.04	Over the course of the Unit 2 Cycle 16 fuel cycle, it became evident that the 2B-B CCP discharge check valve (2-VLV-62-532) was allowing a small amount of back-leakage, which was causing gas to accumulate in the discharge piping of the 2B-B CCP. As water leaks through the discharge check valve, it causes gas to come out of solution, resulting in gas accumulation of approximately 0.03 cubic feet (ft^3) per 12 hour shift. Functional Evaluation (FE) 42823, Revision 5, was written to evaluate this condition, and it contains compensatory actions that the 2B-B CCP preferentially remain in service in order to eliminate the gas accumulation mechanism. Five accident analyses were considered for evaluation of the compensatory measures. The accidents are a small and large break LOCA, a steam generator tube rupture, a faulted steam generator, and a spurious SI. The result of the evaluation is that the compensatory actions are adequate to ensure that the CCP is functional with respect to its design basis requirements. This evaluation is based upon the assumptions made in the accident analysis and that if the 2B-B pump were stopped during an accident, then the gas accumulation in the piping could be stopped simply by swapping from the 2A-A CCP to the 2B-B CCP. Compensatory measures have been established that will enable the 2A and 2B CCPs to remain fully functional. All automatic and manual pump start sequences are unaffected by the required compensatory measures.	The proposed activities are feasible and effective for maintaining the design functions of the CCPs. The compensatory measures do not increase the consequences due to an accident or a malfunction, nor do they increase the likelihood of an accident or malfunction. Therefore, the compensatory measures are acceptable and may be implemented without prior NRC approval.

Procedure	DESCRIPTION	SAFETY ANALYSIS
EA-68-7 R(0), ES-0.1 R(35), and associated FSAR Change	This evaluation addresses procedure changes associated with the need for contingency actions to address RCS inventory control for a postulated Turbine Building fire, resulting in the unavailability of non-essential control air, which causes Chemical Volume Control System (CVCS) letdown isolation valves to fail in the closed position (PER 282069). In addition, PER 160072 identified the generic need for procedure enhancements to address the unavailability of normal and excess letdown. If the CVCS normal and excess letdown flow paths are unavailable due to conditions such as loss of control air, an alternate letdown path is needed to prevent filling the pressurizer solid as a result of continued RCS mass input via RCP seal injection. The change involves use of the reactor vessel head vent valves as a backup RCS letdown path under infrequent/abnormal conditions. This change is evaluated to address using the head vent system in a manner which is inconsistent with the UFSAR description, and the potential to release liquid water and steam to containment.	Based upon the design of the Reactor Head Vent System, the proposed change does not result in a greater likelihood of any accident or malfunction of a structure, system, or component (SSC). This change does not result in more severe radiological consequences from any design basis accident or malfunction. This change is bounded by existing UFSAR accident analyses and the UFSAR descriptions of head vent system malfunctions. Therefore, the described changes are acceptable and may be implemented without prior NRC approval.
0-GO-13 R(72)	This change revises 0-GO-13, “Reactor Coolant System Drain and Fill Operations,” to remove the interlock between flow control valves FCV-63-1, FCV-74-1, and FCV-74-2 by use of a wire jumper. Currently, FCV-63-1 must be fully closed before FCV-74-1 and FCV-74-2 can be opened. This requires the RHR pumps to be stopped during re-alignment of the RHR suction from the RWST to the RCS during refueling cavity flood up and draindown operations. This change allows the heat removal design function of the RHR system to be maintained during re-alignment. This is a change to the control of FCV-74-1 and FCV-74-2 as described in the UFSAR. This jumper will be procedurally installed to aid in RCS drain and fill operations. This procedure is applicable only with the reactor in a shutdown condition with RHR cooling in operation.	This change is applicable only when the RHR system is placed in service for core cooling with the reactor in a cold shutdown or defueled condition. For refueling cavity flood up and draindown operations, the RCS and RWST are designed to communicate via the refueling cavity. Since the change in the interlock logic will be administratively controlled to apply only during these shutdown operations, there is no potential for primary system inventory loss and, as such, the interlock is not required. The ability of the RHR system to perform its ECCS functions during any required conditions will not be affected. The ECCS is required for accident mitigation during loss of coolant, main steam line break (MSLB), and steam generator tube rupture events. This change does not create the possibility of a previously unanalyzed event or impact the consequences of any event described in the UFSAR.

Temporary Alteration Control Form (TACF)	DESCRIPTION	SAFETY ANALYSIS
0-12-011-067	TACF 0-12-011-067, Essential Raw Cooling Water (ERCW) Building Temporary Sump Pumps, Piping, and Power, authorizes the installation and operation of additional sump pumps in the ERCW Pumping Station. This compensatory measure is required, as discussed in the FE for PER 610005, to remove significant amounts of water that could enter the ERCW Pumping Station during a design basis flooding event. There is a newly discovered leak path that could bring flood waters into the building. Small and large temporary sump pumps will be installed in specific ERCW bays. The power source will be from an emergency diesel generator backed board. The discharge will be to the ERCW screen wells.	TACF 0-12-011-067 provides for the installation and operation of additional sump pumps, and the guidance for any Operations procedure changes. In a design basis flooding event, the temporary sump pumps will remove enough water from the ERCW pumping station to the meet design requirements as an active function, instead of a passive function as outlined in the UFSAR. The 10 CFR 50.59 Evaluation performed for TACF 0-12-011-067 demonstrates that the proposed modifications do not increase the frequency or likelihood of accidents or malfunctions, increase the consequences of an accident or malfunction, or create a new type of accident or malfunction.
2-11-003-410	In support of the SQN Unit 2 SGR Project, TACF 2-11-003-410, "Temporary Defeat of Interlocks for Doors A122 and A123," authorizes the temporary defeat of electrical interlocks for and the blocking open of Auxiliary Building doors A122 and A123 as part of creating the personnel access path into the Auxiliary Building to the Unit 2 Containment that will be employed to support the SGR Project These activities do not implement any permanent changes to existing permanent plant SSCs subject to design configuration control management. However, because the compensatory actions directed by TACF 2-11-003-410 involve manual operator actions in lieu of relying upon passive means for establishing the ABSCE barrier, TACF 2-11-003-410 was determined to be adverse to a UFSAR-described design function, and therefore, subject to further review by performing a 10 CFR 50.59 Evaluation.	TACF 2-11-003-410 authorizes the temporary defeat of electrical interlocks associated with the operation of Auxiliary Building doors A122 and A123 and the blocking open of these two doors. This TACF also provides the procedural details for closure of door A123 in the event of 1) occurrence of an accident requiring the function of the ABSCE, 2) a fire event, or 3) as directed by the Control Room. The evaluation performed for TACF 2-11-003-410 demonstrated that appropriate design and implementation measures (i.e., door design, personnel training and staffing, and other committed actions) will be taken to ensure manual closure of Auxiliary Building doors A122 and A123 to re-establish the ABSCE boundary/fire boundary in the event of an accident/fire event. The design and implementation measures for manual closure of the Auxiliary Building doors meet the requirements for effective manual operator actions that can substitute for automatic actions as given in Section 4.3.2 of NEI 96-07, "Guidelines for 10 CFR 50.59 Implementation."

Temporary Alteration Control Form (TACF)	DESCRIPTION	SAFETY ANALYSIS
2-11-004-410	In support of the SQN Unit 2 SGR Project, TACF 2-11-004-410, "Temporary ABSCE Door Installation, Operation, and Removal," authorizes the installation and operation of a temporary ABSCE door on the outside face of the equipment hatch between the Auxiliary Building and the Unit 2 Containment to provide the capability to isolate the Unit 2 SGR activities within the Unit 2 Shield Building and Unit 2 Containment from the Auxiliary Building and Unit 1 during the Unit 2 SGR outage. TACF 2-11-004-410 provides the criteria for testing and closure of the temporary ABSCE door in accordance with the requirements for restoring breaches in the ABSCE boundary that are defined in Technical Instruction O-TI-SXX-000-016.0, "Breaching the Shield Building, ABSCE, or Control Room Boundaries." These activities do not implement any permanent changes to existing permanent plant SSCs subject to design configuration control management. However. because the compensatory actions directed by TACF 2-11-004-410 involve manual operator actions in lieu of relying upon passive means for establishing the ABSCE barrier, TACF 2-11-004-410 was determined to be adverse to a UFSAR-described design function, and therefore, subject to further review by performing a 10 CFR 50.59 Evaluation.	TACF 2-11-004-410 authorizes the installation and operation of a temporary ABSCE door on the outside face of the equipment hatch between the Auxiliary Building and the Unit 2 Containment to provide the capability to isolate the Unit 2 SGR activities within the Unit 2 Shield Building and Unit 2 Containment from the Auxiliary Building and Unit 1 during the Unit 2 SGR outage. TACF 2-11-004-410 provides the criteria for the testing and closure of the temporary ABSCE door, as well as the closure of doors A77 and A157, if required, in order to establish the ABSCE boundary. These criteria are implemented via instructions. The personnel designated to perform the manual actions necessary to re-establish the ABSCE boundary in the event that it is required are to receive training for this task. The 10 CFR 50.59 Evaluation performed for TACF 2-11-004-410 demonstrated that appropriate design and implementation measures (i.e., equipment design and testing, personnel training and staffing, and other committed actions) will be taken to ensure manual closure of the temporary ABSCE door and necessary closure of doors A77 and A157 to re-establish the ABSCE boundary in the event of an accident. The design and implementation measures for manual closure of the building doors meet the requirements for effective manual operator actions that can substitute for automatic actions as given in Section 4.3.2 of NEI 96-07.

Core Operating Limits Report (COLR)	DESCRIPTION	SAFETY ANALYSIS
Unit 2 Cycle 19 COLR	The change to the Core Operating Limits Report (COLR) reflects the transition to the AREVA Advanced W17 HTP (HTP) fuel, which involves a change in the departure from nucleate boiling (DNB) correlation employed in the safety analyses. For the HTP fuel assemblies in the MSLB analysis, the Biasi correlation is applied over the full length of the HTP fuel assembly, replacing the BHTP correlation above the lower most spacer grid and the BWU-N correlation below the lower most spacer grid. The Biasi correlation is approved by NRC for application in the MSLB analysis for all thermal-hydraulic conditions that occur throughout the core during DNB limiting periods of the MSLB accident, including the lowest coolant pressure reached during the transient and the conditions that occur in the region below the lower most spacer grid.	For the HTP fuel assemblies, the Biasi correlation replaces the BHTP and BWU-N correlations in the MSLB analysis. Use of the Biasi correlation does not constitute a departure from a method of evaluation because (1) the Biasi correlation is approved by the NRC specifically for this type of accident and (2) the Biasi correlation was used under the terms, conditions, and limitations of the NRC approval. Accordingly, since the evaluation methodology was previously approved by the NRC for the application, this change does not constitute a departure from a method of evaluation for SQN Unit 2, and NRC approval is not required prior to implementation of the change.

Safety Analysis Report	DESCRIPTION	SAFETY ANALYSIS
TR 3.1.2.2 and TR 3.1.2.4	Technical Requirements Manual (TRM) Technical Requirement (TR) 3.1.2.2, "Flow Paths - Operating," and TR 3.1.2.4, "Charging Pumps - Operating," require the boron injection systems to be operable in Modes 1, 2 and 3 with two charging pumps, and the associated flow paths to the charging pumps, as required for negative reactivity control. TS Limiting Condition for Operation (LCO) 3.5.2, "ECCS - Operating," also requires two ECCS trains, which include the charging pumps, to be operable in Modes 1, 2, and 3. To support Low Temperature Over Pressure Protection (LTOP) System operations, provisions are provided in the TSs that allow the ECCS pumps to be made incapable of injecting in Mode 3 for a limited amount of time or until specified system conditions are exceeded. These provisions; therefore, prevent TS non-compliance when entering into and out of Mode 3 when two charging pumps are required to be operable. This change will revise the TRM and the appropriate UFSAR section to allow operational provisions similar to the TS allowance for temporarily disabling one half of the boron injection function of the CVCS (i.e., one charging pump and associated flow path) to support transition between Modes 3 and 4.	During normal operating conditions, the CVCS provides boron injection via charging flow to the RCS using only one of the two charging pumps. Shared components of the CVCS also provide for accident mitigation. The temporary incapability of having redundant boron injection for normal plant operations was found not to affect normal plant operations during this period. The capability to maintain adequate shutdown margin has not been effected. This change is consistent with the provision in TSs to minimize the chances of a cold over-pressurization event by the ECCS. This protection provision is found in TS LCO 3.5.2. The provision is provided to plants where the LTOP arming temperature is near the Mode 3 boundary temperature of 350°F. TS LCO 3.4.12, "Low Temperature Over Pressure Protection (LTOP) System," requires that certain ECCS pumps be rendered incapable of injecting at or below the LTOP arming temperature. When RCS temperature is at or near the Mode 3 boundary temperature, time is needed to make the pumps incapable of injecting prior to entering the LTOP applicability, and to restore the inoperable pumps to operable status on exiting the LTOP applicability.

Safety Analysis Report	DESCRIPTION	SAFETY ANALYSIS
FSAR Table 5.2.1-1	The number of loading and unloading power changes to which the Unit 1 RSGs were designed has been reduced as a result of DCN D20672A. Additionally, other cyclic/transient limits were affected by the Unit 1 SGR Project. This change to FSAR Table 5.2.1-1 reconciles the cyclic/transient limits and evaluates the change as identified in PER 422244.	The evaluation for this change to the FSAR Table determined that the change does not increase the frequency of the occurrence of an accident; does not increase the likelihood of the occurrence of a malfunction; does not increase the consequences of either an accident or a malfunction; does not create the possibility of an accident of a different type; does not create the possibility of a malfunction with a different result; nor does it exceed or alter any design basis limits for any fission product barriers. Therefore, it is concluded that NRC approval is not required prior to implementation of the FSAR change.
FSAR Change Package No. 24-21	A UFSAR change is proposed to resolve two differences between the facility and the facility as described in the UFSAR. The first difference is that calculations indicate that several control rods may not reach the fully inserted position within the UFSAR described time after the rods are released. Several rods may be "slow to settle" as a result of fuel assembly distortion. The second difference is that a process exists for identifying these rods and evaluating the acceptability of the condition. This process is not described in the UFSAR. The change would revise the UFSAR to (1) add a description of the engineering process for assessing the effects of fuel assembly distortion and (2) add a note on Figure 15.1.5-1 indicating that all acceptance criteria can still be met if a limited number of control rods do not reach the fully inserted position within the indicated time. This proposed change is necessary because it resolves the inconsistency between the implication that all rods must respond as indicated in UFSAR Figure 15.1.5-1, and the fact that several rods can and may perform differently, with all acceptance criteria met.	This evaluation examines the effects of all currently identified slow to settle rods in every applicable UFSAR analyzed event. Based on the results of the evaluation, there is no effect on any previously analyzed event. The evaluation also examines the process by which slow to settle rods are identified and evaluated, and it was determined that the identification and evaluation process is consistent with the facility as described in the UFSAR. Therefore, the proposed changes to the UFSAR to address the possibility of slow to settle rods are acceptable for incorporation without NRC prior approval.

Safety Analysis Report	DESCRIPTION	SAFETY ANALYSIS
TS Bases 3/4.5.3, ECCS - Shutdown	A change is proposed affecting the Bases for TS LCO 3.5.3, "ECCS - Shutdown," for SQN Units 1 and 2, with a corresponding change to UFSAR Section 6.3.2.2. These changes are made due to the difficulty experienced in previous outages in completing leak tests of the RCS Pressure Isolation Valves, particularly the RHR Primary and Secondary check valves, within the one hour time allowed by the LCO action statement provisions. Note 2 under the Action section of LCO 3.5.3 allows the required RHR subsystem to be inoperable for up to 1 hour in order to perform leak testing of these valves. The TS Bases change allows one train of the cold leg injection lines to be out of service for testing during Mode 4. This means that cold leg injection flow would be into two of the four RCS cold legs. A new paragraph is added to the Applicable Safety Analysis section for TS LCO 3.5.3. This paragraph documents that one train of ECCS, injecting into two cold legs, provides sufficient flow for core cooling. Under the LCO section, a paragraph is added to document that either of the two cold leg injection flow paths may be isolated for testing in Mode 4. A new reference has also been added. This reference is from an NRC pre-licensing Safety Evaluation Report (SER) for SQN in which it is concluded that a large break LOCA is not credible in Mode 4.	The positions of the RHR valves, which might be closed for testing, do not affect the frequency of any accident involving the piping systems within the RCS boundary. No additional malfunctions of the valves were identified that would be created by the valves being in a closed position. Core cooling is shown to be adequate, and there is no effect on the remaining fission product barriers. As such, there will not be any change in the consequences of an accident, and no accident limits are exceeded or altered. The dose consequences of the at-power LOCA (with all single failures considered) remain limiting. No new accidents relating to the proposed change have been identified. No different equipment is manipulated, no different alignments are meant to exist, and the components will not be manipulated more frequently than is done under the current TS Bases. Accordingly, there is no possibility for a malfunction with a different result.

Safety Analysis Report	DESCRIPTION	SAFETY ANALYSIS
FSAR Change Package No. 24-44	The proposed change reduces the analytical limit for the trip setpoint for the Power Range Monitor (PRM) high flux, high setpoint reactor trip in the dropped rod analysis (UFSAR 15.2.3, "Rod Cluster Control Assembly [RCCA] Misalignment"). This analytical limit is an assumed value used in the dropped rod safety analyses to conservatively represent the highest reactor power level reached and the power level at which the dropped rod safety analysis must meet DNB acceptance criteria. Under the proposed change the analytical limit for the PRM trip setpoint will be reduced from 116.5 percent of Rated Thermal Power (RTP) to 115 percent RTP. The 116.5 percent RTP analytical limit will not be changed in any other analysis. The TS nominal and allowable values, 109 percent and 111.4 percent RTP, respectively, will not change. No changes will be required in the TSs, COLR, or associated surveillance and monitoring procedures. Therefore, the scope of the proposed activity is limited to the safety analysis evaluation of the PRM reactor trip, as it functions in the dropped rod accident.	The net effect of the proposed change is an earlier termination of one UFSAR accident that otherwise proceeds as previously described, up to the point of the earlier termination. Under the proposed change, the accident is evaluated using the same methods and the same design basis limits as previously described in the UFSAR. The only change to the facility created by the proposed activity is that the portion of the dropped rod evaluation that existed between 115 percent RTP and 116.5 percent RTP has been removed. Such a change cannot increase the frequency or likelihood of existing accidents or malfunctions, and cannot change the consequences of previously analyzed events. As stated, the evaluation methods and design basis limits do not change. All that remained to evaluate was any potential to create a new type of accident or a malfunction with a different result. Because the change to the analytical limit is technically sound and consistent with methods, procedures, and criteria presently described in the UFSAR, the change does not create any new accident initiators or failure mechanisms that could lead to a new type of accident or a malfunction with a different result.

Commitments	DESCRIPTION	SAFETY ANALYSIS
None		

Document Number/72.48 Evaluation Tracking Number	DESCRIPTION	SAFETY ANALYSIS
22621B	Calculation 8082-0225, Revision 004, "Post LOCA & Loss of Offsite Power Responses to Place a Loaded HI-TRAC Cask Into a Safe Condition," is revised for maximum stay times that were previously calculated in order to ensure personnel do not receive more than the 10 CFR 50 Appendix A, General Design Criteria (GDC) 19 limit of 5 Roentgen equivalent man (REM) or equivalent (30 REM beta, 30 REM thyroid) in performing functions necessary to ensure a cask is either placed back into the spent fuel pit or cooling set up so that the spent fuel assemblies will not overheat.	The maximum stay times on the refuel floor increased from the previous values based on the calculated dose in Calculation 8082-0225. There are no new actions required for dry cask operations coincident with a design basis LOCA or loss of offsite power (LOOP). Actions currently incorporated into plant procedures to instruct plant personnel on steps necessary to place the multi-purpose canister (MPC) and/or transfer cask in a safe condition in the event of a LOCA or LOOP during Independent Spent Fuel Storage Installation (ISFSI) operations are not being changed by this activity. Current actions required for maintaining annulus, alternate or supplemental cooling systems operable, and for compliance with the HOLTEC TSs in order to perform HOLTEC Cask Final Safety Analysis Report (CFSAR) design functions are not affected by this change. The conclusion of this evaluation is that the changes to the Calculation do not affect any regulatory or licensing requirements, result in special conditions or limitations to be created or affected, or have any affect on the health and safety of the public or on nuclear safety.