Tennessee Valley Authority, Post Office Box 2000, Soddy-Daisy, Tennessee 37384-2000

May 2, 2001

10 CFR 50.71(e)

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

Gentlemen:

In the Matter of)
Tennessee Valley Authority)

Docket Nos. 50-327
50-328

SEQUOYAH NUCLEAR PLANT (SQN) - UNITS 1 AND 2 - 10 CFR 50.59, CHANGES, TESTS AND EXPERIMENTS SUMMARY REPORT

The purpose of this letter is to provide the summary report of the implemented safety evaluations performed in accordance with 10 CFR 50.59(d)(2). The enclosed report covers the period from May 31, 1999, to November 14, 2000.

If you should have any questions, please contact me at (423) 843-7170 or J. D. Smith at (423) 843-6672.

Sincerely,

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IE47

Enclosure

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ENCLOSURE

SEQUOYAH NUCLEAR PLANT

10 CFR 50.59 SUMMARY REPORT

SEQUOYAH NUCLEAR PLANT
CHANGES IN THE FACILITY - MODIFICATIONS
REQUIRING 50.59 EVALUATIONS

DCN	DESCRIPTION	SAFETY ANALYSIS
D20008A & FSAR Change #16-13	The purpose of this modification is to upgrade the filter reliability of the nonsafety-related Auxiliary Building Fuel Handling Exhaust Fan A subsystem. The current configuration of the Auxiliary Building Fuel Handling Exhaust Fan A Subsystem does not include any prefilters or HEPA filters in the exhaust stream to the Auxiliary Building vent stack; however, filter frames do exist from the original design. These filters were originally provided by design and were previously installed (circa 1978) but were subsequently removed after an evaluation by the Radiological Hygiene Branch deemed that the filters could be removed without affecting compliance with 10CFR50 Appendix I. The changes proposed by DCN D-20008-A are to install prefilters and HEPA filters in the Fuel Handling Exhaust Fan A-A suction plenum using the existing filter frames. Fan B has no filter plenum or filters installed since it was converted into the 5th Vital Battery Room. During normal operation either fan may be used. However, during plant outages or periods when maintenance is being performed in the fuel transfer canal / spent fuel pool it will be preferable to use Fan A since it will provide a filtered exhaust path.	There are no TS (TS) impacts as a result of the change proposed by this modification. A FSAR change is required to revise section 9.4.2.2.1 and figure number 9.4.2-5 to reflect the addition of filters to the Fuel Handling Exhaust Fan A-A exhaust path. This modification does not constitute an unreviewed safety question (USQ) because the permanent configuration does not change the function or operation of the Fuel Handling Fan A or Fan B system nor adversely affect the function or operation of any associated safety-related feature.
D20044A & FSAR Change #16-37 & TRM R6	This modification will upgrade the Seismic Monitoring System. This system is not safety related; nor does it have any effect on safety-related systems or equipment. The upgraded seismic instrumentation will improve the accuracy and reliability of the seismic monitoring system; while at the same time greatly simplifying the maintenance and surveillance resources required to support the system. The inclusion of an onboard processing computer and strong motion analysis software facilitates timely evaluation of the recorded event. The addition of the new Strong Motion Accelerograph, featuring a digital recorder, is an equivalent replacement for the existing analog-based Strong Motion Accelerograph. The installation of the upgraded seismic instrumentation components and the removal of the obsolete components is staged to avoid any unacceptable	SAR Section 3.7.4 will be revised to reflect the upgraded configuration of the Seismic Monitoring System and the adoption of EPRI OBE Exceedance Criteria for shutdown logic. The TRM and TRM Bases will be revised to reflect operability conditions and surveillance intervals related to the upgraded system. The upgraded Seismic Monitoring System maintains the SAR commitment to the intent of R.G. 1.12, Revision 1. Therefore, implementation of this modification and corresponding revisions to Section 3.7.4 of the FSAR and TRM Section 3/4.3.3 does not involve an unreviewed safety question (USQ).

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	impact on existing safety-related systems or components. The use of EPRI OBE Exceedance Criteria ensures that the decision for a controlled shutdown is based on actual damage potential of the event, which reduces the shutdown risk associated with unnecessary shutdowns.	
D20058A & FSAR Change #16-65	This design change replaces and upgrades the pressure control valves which supply high pressure nitrogen to the Cold Leg Accumulators (CLA). The new regulators are high flow capacity units which will result in the capability to charge a unit's four CLAs in approximately 1 hour. The previous regulator design required a shift or more to complete CLA charging operations for a unit. The new design also installs a flow restricting orifice in each units supply line to ensure that the supply line flow rate cannot exceed the relieving capacity of the CLA relief valves. The orifices eliminate the need for unit specific regulators which operated as "second stage" regulators downstream of the supply regulators and which performed the flow limit function now accomplished by the new flow restricting orifices.	This change modifies nonsafety-related equipment and piping associated with the nitrogen supply used for charging the safety injection CLAs. This change does not modify or impact any safety related system feature or function. No system, equipment or design parameter comprising the interface with the safety-related end user is adversely impacted. No TS requirement or surveillance is impacted. A safety evaluation (SE) is required to support a Sequoyah (SQN) Final Safety Analysis Report (FSAR) figure revision to show the modified regulator design configuration. No SAR accident analysis or transient analysis consideration is impacted. No new failure mode is introduced by this change. No existing failure mode is adversely impacted by this change. This change does not result in an USQ.
D20071	DCN D20071A replaces the eight existing 120-V AC Vital Inverters with eight new uninterruptible power systems (UPSs) consisting of regulated rectifiers, auctioneering diodes, inverters, static and manual transfer switches, and regulated bypass transformers. The existing Unit 2 Vital Inverters will be reconfigured for use as installed spare UPSs. After the modification, each channel of vital power will have three UPSs: the normal Unit 1 and Unit 2 channelized UPSs, and the spare channelized UPS which can be substituted for either of the normal UPSs for that channel. The function of the UPSs with respect to the vital power system will not change. This modification enhances the present design by (1) replacing the existing 120-V AC Vital Inverters (which have been prone to failure or malfunction) with new UPSs consisting of regulated rectifiers, auctioneering diodes, inverters, static and manual	The vital inverters and the vital instrument power boards are not considered to be an initiator of a design basis accident. These features provide power to instrumentation that support the identification and mitigation of accidents as well as system control functions during normal plant operations. The function of the inverters is not altered by this modification, and this modification will not create the possibility of a new or different accident. The design change is staged and requires that work affecting operability on one channel be completed before work affecting operability on another channel is begun. Work not affecting operability such as equipment installation, cable installation, and testing may be conducted on different channels concurrently. This modification enhances the present design by improving the diverse and redundant options available for plant personnel if confronted with the loss of an inverter. Based on

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	transfer switches, and regulated bypass transformers, (2) providing automatic transfer for a regulated bypass supply for each normal UPS in the event of inverter failure or overload, and (3) providing manual transfer from a normal UPS to a qualified installed spare UPS during maintenance and testing. The addition of a regulated bypass source and the automatic static switch provided additional capability to provide reliable power to the vital instrument power boards and minimize the potential for a unit trip due to inverter failure. The regulated bypass source will be a more reliable supply to the board with better voltage control to support safety-related instrumentation operation in the event of inverter failure or overload. The ability to utilize spare inverters as an alternate source will allow better opportunities to properly maintain the inverters and minimize the potential to require a unit shutdown for an inverter problem. The modified system will meet or exceed the specifications and capabilities of the old system.	this fact, the worst possible failure scenario would be the loss of one channel of vital power, which is adequately covered by TS 3/4.8.2. This modification will not result in any malfunctions of a type worse than those previously analyzed and will not create any new failure modes. SQN has been evaluated per the requirements of the Station Blackout (SBO) rule 10CFR50.63 and USNRC Regulatory Guide 1.155. The plant must cope with the effects of an SBO for four hours. This modification provides additional automatic and manual transfer capabilities and a spare inverter. These changes will provide improved capabilities to tolerate inverter failures and help assure SQN satisfies the SBO requirements.
D20099A & FSAR Change #16-21 & FSAR Change #16-29	DCN D-20099-A provides a design change to the ERCW Strainer Backwash and Flush lines to replace the existing motor-operated backwash valves with manual stainless steel Anchor Darling ball valves and remove the flush lines and flush valves. The existing valves are leaking excessively. They are carbon steel EPG (Energy Products Group) EBV Systems Division ball valves which are no longer being manufactured and the carbon steel has not held-up well in this service.	The safety function of this portion of the system requires a continuous backwash flow of 450 gal/min to be established within three hours of an earthquake, tornado, flood, loss of offsite power, and loss of downstream dam. This flow must be established within 12 hours of a LOCA. The continuous backwash flow is established by manual manipulation (local handswitch) of the backwash valves. The proposed change would establish this condition by manual operation of the backwash valve by manipulation of a manual lever. Implementation of this modification requires changes to FSAR Section 9.2.2 and Figure 9.2.2-5, the ASME Section XI Program Basis Document, and the 10CFR50 Appendix R Program to show the ERCW strainer backwash valves as manually actuated versus electric motor actuated (by local handswitch) and deletion of the flush lines. Since the proposed modifications do not affect any of the automatic functions of the ERCW system, the existing TSs do not require revision to support the proposed change. Implementing the proposed

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		activity does not affect any procedures, processes, or system operation characteristics outlined in the SAR and does not alter the system functional requirements in establishing the required flow condition, within the specified times. Therefore, system safety function is unaffected and the proposed activity is acceptable from a nuclear safety perspective. This change does not involve an USQ.
D20152A & FSAR Change #16-20 & FPR R4	The scope of this modification is to abandon the Main Control Room (MCR) air intake smoke detectors. This will require a revision to the FSAR (chapter 6.4 and 9.4). These sections identify that the MCR intake has smoke detectors but manual operator action is used for isolation and startup of the MCR emergency ventilation system. The Fire Protection Report (Section VIII) is also being revised as a result of the DCN. Reference to the smoke detectors will be removed from both documents. The FSAR and Fire Protection Report currently takes credit for operator action for a control room isolation (CRI) for smoke.	The MCR habitability is not adversely impacted by the removal of the automatic initiation of a CRI by the smoke detectors. The existing analysis takes credit for the operator taking manual action to isolate the air intake and start the emergency system. The Fire Protection Report also takes no credit for the operation of the smoke detectors. Therefore, revision to the FSAR to remove the smoke detectors has no impact on any safety analysis or analysis related to maintaining the habitability of the MCR. This change does not involve an USQ.
D20172A & FSAR Change #16-05	DCN D20172 changed the operating logic for the Diesel Generator (DG) Building Exhaust Fans to ensure that these fans operate any time the diesel generators are running; replaced existing thermostats that control these fans with more accurate temperature switches and changed the high and low setpoints for these switches; and corrected the FSAR to clarify that the standby exhaust fan starts from a low flow signal from the primary fan, not a high temperature signal as previously stated. Previously, the subject fans started when their respective diesel generator started but are allowed to stop if ambient air temperature dropped below the controlling temperature switch setpoint. This arrangement was undesirable because if the exhaust fan stopped and then restarted while the diesel generator was operating in test or exercise mode, slow inlet damper response could result in negative atmospheric pressure in the diesel generator room which could in turn cause the diesel to trip on high crankcase pressure. The crankcase	The only accident analyzed in the SAR that could be potentially impacted by this activity is loss of off-site power to the station auxiliaries. The safety function of the Diesel Generator Building Ventilation System is to maintain an acceptable building environment for the diesel generator system, electrical boards, and equipment. The proposed activity will improve the operating environment for the diesels, and will ensure that the exhaust fans are operating whenever the diesels are running. The credible failure associated with this modification is: The diesel generator room exhaust fans could fail to start and result in an over-temperature condition in the diesel generator building. The proposed activity does not change the basic system design and the new setpoints are bounded by the existing safety limits of 40 and 120 degrees F. This change does not increase the probability of a failure of the subject fans or associated controls. This failure mode does not constitute an USQ because the plant is in the same configuration as analyzed

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	pressure switches are only active during test / exercise mode, thus a diesel trip on high crankcase pressure during accident mode operation is not credible. FSAR figures 8.3.1-2 and 8.3.1-16 are schematic representations of the exhaust fan operation and were revised by this DCN. The setpoints for the temperature switches that control the Diesel Generator Building Exhaust Fans were changed by this DCN to improve the diesel generator operating environment and reduce unnecessary exhaust fan operation. The low temperature setpoint that stops the fans was raised from 48 °F to 68 °F. The exhaust fans pull outside air into the Diesel Generator Building and, during certain times of the year, this movement of cool air across the diesel engines caused the water jacket temperature to drop too low. During the summer months, the 48 degree setpoint may not be reached and the fans would operate continuously or until stopped manually. To minimize switch overlap, the existing thermostats were replaced with new, more accurate temperature switches and the high temperature setpoint was raised from 80 °F to 90 °F. These setpoints appear in FSAR section 9.4.5.2, which was revised by this DCN. The revision to FSAR section 9.4.5.2 also clarified that the standby exhaust fan starts from a low flow signal from the primary fan, not a high temperature signal as previously stated.	and approved in the original design. No new unreviewed safety questions are being introduced by the proposed activity.
D20183	DCN D-20183 is part of the ongoing Thermo-Lag Upgrade Project and documents the acceptability of the duct work modifications (notches) made to the Fuel Handling Exhaust Ventilation duct located in the overhead of floor elevation 714.0 in the Auxiliary Building Unit 1 Penetration Room. This section of ductwork is also a part of the safety-related ABGTS air clean up system. FSAR section 9.4.2.2.1 states two (2) 100 percent capacity fuel handling exhaust fans are provided. During normal plant operation, one fan is in use with the other in standby mode. Both of the fans will be taken out of service during the performance of this modification. During the modification, the Auxiliary Building will be maintained at a	There are no new design basis accidents or anticipated operational transients created by the duct work modifications (notches). There are no new failure modes created. The structure, systems, and components (SSC) being modified by this DCN are safety-related, and there will not be an increase in the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR. The duct work modifications do not create a possibility of an accident or malfunction of a different type than evaluated previously in the SAR. The margin of safety as defined in the basis for the TSs is not reduced. This change is not an USQ.

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	slightly negative water-gauge (WG) pressure by the Auxiliary Building General Exhaust Fan(s) and a temporary damper alignment. The Fuel Handling Exhaust Fans are being taken out of service for ALARA concerns and personnel safety protection while the duct notches are being cut and installed from inside the ductwork. The implementation of this activity will not significantly affect the air flow rates required for normal operation nor will it affect the operation of the safety related ABGTS system.	
D20219A & FSAR Change #16-19	Permanently abandon in place the Unit 1 Additional Equipment Building (AEB) Air Conditioning (AC) Unit by isolating the Raw Cooling Water supply and removing electrical power. To limit erosion and corrosion, 3/4" drain connections will be added to drain the Raw Cooling Water piping. The cooling coil serviced by the Raw Cooling Water system has a pin hole leak. Unlike Unit 2 AEB which has significant heat loads due to ice storage bins, ice machines, and chiller packages that require air conditioning units for cooling the building, the Unit 1 AEB houses the abandoned and nonoperational Upper Head Injection piping/components. The AC unit is not required for cooling of the Unit 1 AEB. Isolation valves for the nonsafety-related RCW system will be configured as closed on the primary drawing and will be tagged as boundary valves for abandoned in place equipment.	FSAR Figure 9.2.7-4 will be revised to document the closed valves and to remove the AC unit. The FSAR Chapter 15 accidents were reviewed and no design basis accidents or anticipated operational transients were determined to potentially be affected by this change. Therefore, the proposed modification does not directly or indirectly impact the nuclear safety margins for operation or shutdown of the plant. This modification does not involve an USQ.
D20225 & FSAR Change #16-57	Plant recorders are nearing the end of their useful life and parts cannot be obtained. The recorders being removed or replaced were classified as Non-safety related, Seismic Category 1L(B), Non-Regulatory Guide 1.97, and not required for safe shutdown of the plant. This modification replaced 14 recorders with 19 and removed one recorder. These recorders are mounted on panels inside the Diesel Generator Building (DGB) Engine and Relay Board rooms, and in the Unit 1 Main Control Room. All inputs on recorders being replaced will be included as inputs on the new recorders. The exception is the 1-RR-90-1A & B recorder where the inputs from monitors other than the	A SE was required for this modification due to removing recorder 0-RR-90-225, "Condensate Demineralizer Liquid Monitor" which required removal of reference to a main control room recorder for 0-RE-90-225 from FSAR Section 11.4. Removal of the recorder will not adversely affect the initiation of the radiation monitors automatic functions. This recorder is not required for the ODCM and is not used in the Emergency Operating Procedures (REP). This recorder was deleted from the plant design by DCN 20225 Stage 13. None of the other stages involve an SAR change. This change, therefore, did not involve an USQ.

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	Auxiliary Building are being placed on the plant computer instead. This modification also split the "Instrument Malfunction" and "Hi Rad" alarms for the Non Auxiliary Building monitors to a separate alarm. All inputs on removed recorders are on the plant computer or trending information is not required.	
D20236	The changes proposed by DCN D-20236-A are to remove the existing Temperature Control Valves located in the Essential Raw Cooling Water (ERCW) supply line to the Main Control Room (MCR), Electric Board Room (EBR), and Shutdown Board Room (SBR) air conditioning condensers. These Temperature Control Valves (TCVs) will be replaced with TCVs designed specifically for this service i.e., stainless steel, self-contained water regulating valves using condenser refrigerant pressure as the only input signal for valve regulation to maintain desired condenser pressure for the associated air conditioning equipment. The TCVs are being procured as ASME Section III, Class 3 and to seismic category 1 requirements for safety-related application. The replacement automatic 3" TCVs proposed by DCN D-20236-A will be sized for a flow range to provide better control of cooling water flow to condenser load. The control air supply instrument isolation valves to the existing controllers and TCVs will be closed and capped and retained as spares.	The changes proposed by DCN D20236A were evaluated for FSAR impacts (Section 9.2 and 9.4) and resulted in only two figure changes to show the relocation of TCV-67-158. There were no FSAR text changes since the functional requirements of the replacement TCV remains unchanged. There are no FSAR Chapter 15 analysis impacts or new credible failure modes created as a result of this proposed modification. TS Sections 3.7.4 and 3.7.7 were also reviewed and determined not to be impacted by DCN D20236A. The changes proposed by DCN D20236A to the MCR, EBR, and SBR air conditioning, ERCW, and Auxiliary Control Air system do not change the function or operation of these systems or their ability to perform in a reliable manner as with the previous design; therefore, the removal of the existing pneumatic TCVs and replacement with self-contained TCVs does not involve an USQ.
D20243A & FSAR Change #16-33	This modification reroutes the safety-related ERCW system supply and return piping to Upper Containment Ventilation (UCVC) Coolers 1A and 1B. Degraded supply and return piping and valves located in the Auxiliary Building and U1 annulus and upper containment will be abandoned or removed. New tie-in locations and isolation valves and other components for new supply piping for UCVCs 1A and 1B in Unit 1 upper and lower containment will be provided from the excess ERCW supply to Lower Containment Ventilation Coolers (LCVCs), Control Rod Drive (CRD) motor coolers and Reactor Coolant	The modifications to the safety-related ERCW system to install and route new supply and return piping to Upper Containment Ventilation (UCVC) Coolers 1A and 1B affects divider barrier containment penetrations. However, the modification involves installation of penetrations which are designed to ensure the integrity and safety function of the divider barrier, as described in the FSAR. The new permanent configuration does not adversely affect the function or operation of the ERCW system, its end users, the divider barrier, or primary and secondary containment penetrations. There are no other systems directly

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	Pump (RCP) motor coolers. New return piping from the UCVCs will also tie in to the RCP motor coolers discharge which also feeds into the LCVC and CRD discharge. Containment Isolation Valves are not required with the new arrangement since the source of water is now inside containment.	affected by this modification. ERCW supply to other systems will not be adversely affected. Therefore, implementing the proposed activity will not result in an USQ.
D20248A & D20249A & FSAR Change #16-30	The modification (D20248A for Unit 1 and D20249A for Unit 2) reroutes discharge cooling water from the Unit 1 or Unit 2 bus heat exchangers to the yard drainage pond via exposed pipe to a surface water drain. This replaces the existing subsurface routing of the discharge piping to the pond. The bus heat exchanger is located adjacent to the west end of the turbine building in line with the projected axis of the main turbine generator unit with the unit base at elevation 706' (ground). The new discharge piping will route from this unit up to an elevation of approximately 734' and then proceed south and east inside the perimeter handrail of the turbine building to a surface water drain located near the control building interface with the turbine building. The design added two new valves inline with the discharge pipe as a consequence of the need to drain/isolate the piping for maintenance activities.	The equipment affected by this change is not safety-related and is isolated from safety related equipment. Design considerations and materials for the pipe reroute are consistent with FSAR requirements. The capacity of the surface drain has been reviewed and determined to be adequate to support the addition of the discharge flow. Failure of the new discharge piping and drain valves was evaluated and found to not impact the possible loss of the Main Generator Bus or the Unit Start Bus. The consequences of equipment failures will remain unchanged by the installation. SQN TSs have been reviewed and the margin of safety was determined to be unaffected by the new routing of the bus discharge. This change does not constitute an USQ.
D20273A & FSAR Change #16-14	The diesel generator building ventilation systems are designed to maintain an acceptable building environment for the protection of the diesel generators, electrical boards and equipment, and for the safety of operating personnel. As part of this system, one battery hood exhaust fan for hydrogen removal is located at elevation 722 in each of the four diesel generator rooms. Based on analysis it has been determined that the battery hood exhaust fans are no longer needed. The existing room ventilation is adequate to maintain the hydrogen concentration in the room as a result of battery charging below the lower explosive limit. This modification documents the abandonment of the diesel generator safety-related battery hood exhaust fan motors, ductwork, dampers, and associated	This modification completely isolates the battery hood exhaust fans, ductwork, dampers, and associated controls from the remainder of the DG ventilation system. FSAR Section 9.4.5 and FSAR Figure 9.4.5-1 were revised to reflect this modification. The ability of the diesel generator building ventilation system to meet its design function is not adversely impacted by the removal of this unnecessary equipment. Specifically, the diesel generator building ventilation system post-modification is capable of maintaining hydrogen concentrations below the lower explosive limit. Therefore, this change does not involve an USQ.

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	controls. This equipment will be isolated from the system and abandoned in place.	
D20276A & T20133A & FSAR Change #16-36	DCN D20276A (Unit 1) and DCN T20133A (Unit 2) respanned the reactor coolant flow transmitters from 0-110% to 0-115% flow. This also changed the scale on the main control room indicators, plant computer, and the milli-amp value for the 90% flow setpoint. This DCN was done due to several loops were indicating near full scale. This change affects main control room (MCR) indicators located on panel M-5. The signals are processed in Eagle 21 racks. There are three protection set loops for each of the four reactor coolant loops. Implementing this modification will a revision to the transmitters span range in FSAR Section 7.2.1.2.6, Reactor Coolant System Design Criteria, and Reactor Protection System Design Criteria.	The reactor coolant flow has not changed from original design. The change will be to indication and not to actual flow or 90% flow trip setpoint. The change in RCS flow instrumentation will not change classification of previously analyzed accidents or transients as defined in FSAR Chapter 15.2.5, Partial Loss of Forced Reactor Coolant Flow. The change in RCS flow instrumentation will not increase challenges to safety systems assumed to function in the accident analysis and will not increase offsite releases as defined in the FSAR Chapter 15.2.5. The logic for the reactor trip is not being altered. Uncertainties are still bounded by the values assumed in the SAR Section 7.2.2.2.2, Reactor Coolant Flow Measurement. No new accident initiators or failures are introduced by the change to RCS flow spans. The respanning of the reactor coolant flow loops does not reduce the margin of safety as defined in the TS sections; 3/4.2.5 DNB Parameters, 3/4.3.1, Reactor Trip System Instrumentation and 3/4.4.1, Reactor Coolant Loops. Therefore, the proposed activity will not result in an USQ.
D20292A & FSAR Change #16-66	DCN D20292 replaces two bypass valves and piping in the bypass line to the temperature control valves for the Unit 2 turbine generator air side and hydrogen side seal oil heat exchanger with stainless steel valves and piping. Install 1-inch piping and valves to the bypass valves in order to allow more precise control of flow rate when manual bypass is required. Replacing these isolation valves, changing the original bypass piping to stainless steel, and adding a second smaller bypass line, will not change the function of either the raw cooling water system or the Unit 2 turbine generator air side and hydrogen side seal oil heat exchanger. This DCN will not change credible failure modes of either the raw cooling water system or the Unit 2 turbine generator air side and hydrogen side seal oil heat exchanger. The possible failure of the valves	The systems affected by this modification will operate in generally the same manner as prior to this modification, and no new accident scenarios will be introduced which will directly or indirectly adversely affect any nuclear related systems, structures, or components. There are no design basis accidents or operational transients in Chapter 15 of the FSAR associated with the proposed modifications. There are no Appendix R components or equipment, or any nuclear safety-related systems or portions of systems affected by the proposed modifications. The raw cooling water system and the Unit 2 turbine generator air side and hydrogen side seal oil heat exchanger do not perform any safety-related function, nor will they compromise the ability of safety-related systems to perform their intended functions or increase challenges to these systems. Therefore,

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	is reduced by selection of valves that are compatible with the conditions in the bypass lines. The proposed modifications will meet all of the design, material, and construction standards applicable to the raw cooling water system and SQNP.	this modification will not affect any design basis accidents or anticipated operational transients. For these reasons, this activity does not constitute an USQ.
D20309 & FSAR Change #16-41	DCN 20309 will abandon in place the duct heaters and their controls. The purpose of the pressurizing fans duct heaters is to ensure that the intake air for the MCR remains above 60 °F. Normal temperature for the MCR is 75 °F. An analysis evaluated the temperature response for the 732 elevation of the control building. The analysis documents that there is sufficient heat load in the control room and other rooms on elevation 732 to maintain the MCR temperature well above 60 °F without the pressurizer fan duct heaters; therefore, the duct heaters are not required.	The FSAR will be updated in Section 6.4.1.5 and 9.4.1.2 to remove discussion of the function of the duct heaters and the duct heater temperature controls. Figure 9.4.1-1 will be revised to remove the duct heater. FSAR Section 9.4.1.3 will be revised to remove the requirement for equipment to operate from 50 to 104 degrees F. The environmental control system for the control building will still meet the design basis requirements without the preheaters. The Auxiliary Power System is not adversely impacted by the load reduction. This change does not impact any safety margin or safety analysis. This change does not involve an USQ.
D20320A & FSAR Change #16-18 & FSAR Change #16-56	This modification adds a new Multi-Purpose Building and Fire Operations Center to the plant site. The construction includes the addition of concrete slabs, metal prefabricated buildings, electrical, potable water, and sewage connections. These buildings will be located on the large open area between the unit 1 reactor building and the intake pumping station. The interface design of these building structures with other plant features including yard drainage, flood control, missile shields, and ERCW piping which runs beneath both buildings as well as the electrical, potable water, and sewage connections will be addressed and documented in this modification. The new facilities and equipment added by this modification are all non safety related and will have no interactions that could adversely impact any plant safety-related or quality-related system, structure or component.	This modification does not modify any system, structure, or component that is safety related or important to safety. The nonsafety related equipment and structures added/modified by this modification were evaluated and do not impact or introduce any interactions with safety-related plant features, electrical, mechanical, or structural. The site drainage plan is not adversely impacted by this change. The 50.59 evaluated the indirect affect of the new structures location over the buried safety-related ERCW supply lines (from the intake to the plant). The loads resulting from the addition of slabs and structures over the ERCW lines are within the allowable design for ground loading as defined in the General Reinforced Concrete Structures Design Criteria and will therefore not challenge or compromise in any way the function of the safety-related ERCW piping. Therefore, this modification does not involve an USQ.
D20348 & D20472 & FSAR Change #16-34	This SE evaluated the installation of pressure reducing orifices downstream of the ECCS throttle valves by DCN 20348 (Unit 1) & DCN 20472 (Unit 2). INPO Operating Experience Report	The activity will not invalidate any accident mitigating assumptions as identified in the accident analysis for SQN. This activity will not increase challenges to the ECCS or other

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	7127 described the potential for flow runout of the Centrifugal Charging Pumps (CCPs) and Safety Injection Pumps (SIPs) due to Emergency Core Cooling System (ECCS) throttle valve plug/seat erosion. The potential flow erosion results from high differential across the valves during a large break Loss of Coolant Accident (LOCA) scenario. NRC Information Notice 97-76, "Cavitation-Induced Erosion During LOCA," addresses the same issue addressed by this OER. SQN Commitment NCO960060001 requires installation of pressure reducing orifices in the Safety Injection hot and cold legs and Centrifugal Charging cold leg injection branch lines to share the pressure drop with the throttle valves, thereby eliminating the erosion concern. Pressure reducing orifices have been designed for installation downstream of the ECCS throttle valves. The orifices are designed to support ECCS long term (100 days) post LOCA recirculation without erosion of the throttle valves or orifices, and ECCS flow requirements are met.	safety systems credited in the SAR for accident analysis mitigation such that safety system performance is degraded below the design basis. This activity does not invalidate any assumptions for the Chapter 15 accident analysis, nor does it change the basis for any TS. Therefore, this activity does not result in an USQ.
D20414 & FSAR Change #16-54	The change installs a cross connection between the discharge side of the Monitor Tank Pumps and the intake to the Portable Demineralizer. The cross connection allows water in the Monitor Tank to be reprocessed directly to the Portable Demineralizer and then routed to the Cask Decontamination Collector Tank without first having to be circulated back to the Floor Drain Collector Tank.	An FSAR Figure is revised to reflect the additional piping. This portion of the radwaste system is not nuclear safety related and is not relied upon to mitigate any design basis events. Addition of this piping will not impose any detrimental effects upon the ability to safely shut down the plant. The change does not increase the probability or consequences of an accident or malfunction of equipment previously evaluated in the SAR, does not create a possibility for an accident or malfunction of a different type than any evaluated previously in the SAR, and does not reduce the margin of safety as defined in the bases for any TS. Therefore this activity does not result in an USQ.
D20417 & FSAR Change #16-58	The change abandons in place the permanent Fuel Oil supply to the Auxiliary Boiler (AB) from the yard Storage Tanks. The change includes the supply pumps located by the Fuel Oil Storage Tanks and the connecting piping from the pumps to the Turbine building. The pump motors will have their power lifted and the breakers spared. The piping that is being	Two FSAR figures will be revised to reflect the abandoned/removed equipment and piping. This change does not impact any safety-related piping and does not adversely impact any safety related equipment. The Auxiliary Power System is not adversely impacted by the load removal and the safety related portion of the fuel oil system is not impacted by

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	removed from service will be disconnected from the functional piping and either abandoned in place or removed. The existing stub-up connections by the Turbine Building will be abandoned in place and new connections will be added with isolation valves in the pipe trench adjacent to the existing connections. The Auxiliary boiler connections at the supply tanks will be rerouted to connect to the existing yard transfer pump suction piping. The yard transfer pump is used to transfer oil between the two storage tanks, and the Diesel Generators 7 Day Tanks. This will allow the fuel oil for the Diesel Generator 7 Day tanks to be taken from the side rather than the bottom of the yard storage tanks.	this change. Therefore, this activity does not result in an USQ.
D20736 & FSAR Change #16-73	This modification cuts, caps, and removes section of downstream excess heat exchanger sampling tubing between isolation valves and abandons one section of tubing, and removes obsolete valves associated with this sampling function, including removal of two active containment isolation valves. Removal of these two containment isolation valves allows their removal from the 10CFR50 Appendix J leak rate test program and the ASME Section XI ISI program. Inside containment, tubing passing through penetration X-85A will be cut and a cap will be installed on the annulus side of the penetration. All of the affected sample tubing is located in the Unit 2 Reactor Building. The impacted sample line serves no purpose because the tubing downstream of the affected tubing was previously cut and capped in the Auxiliary Building during the abandonment of sample sink number C7.	A FSAR change is required to text section 6.2.4.3, Design Evaluation , Item "r." This item is a list of spare containment penetrations. This modification results in a new spare penetration, number X-85A. This list is also included in the SQN Containment Isolation System Description, N2-88-400. As a result, instead of providing the list of spare mechanical penetrations in the FSAR, this list will now be referenced to and contained in the SQN Containment Isolation System Description. The new configuration does not result in any other FSAR text or figure or table impacts. Containment integrity is not adversely impacted and the containment isolation system function or operation is not impacted. New unanalyzed failure modes are not introduced by the new configuration. The function of the sampling system is not changed. This change does not involve an USQ.
E20286A & FSAR Change #16-22	This change involves the replacement of the existing moisture barrier seal for the interface between the containment liner and the concrete slab for the raceway floor. The existing moisture barrier is comprised of a fiberglass filler with a polysulfide sealant. The change will introduce a polyurethane-based non-porous elastomeric sealant into the crevice that will reduce the potential for water intrusion. Evaluations of the seal indicate	This change impacts design drawing 1,2-48N401, section A-A. This drawing is included in the FSAR as Figure 3.8.2-7. Because the figure is being revised, a 50.59 SE of this change is required. The figure will be changed to remove notes in reference to the filler and sealant materials with the crevice at the interface between the raceway fill slab and the steel containment liner. This change does not impact any system or

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	that high chloride concentrations were present as a result of the fiberglass filler with a polysulfide sealant. Residual water in the crevice wall was also removed which further reduced the potential for a corrosive environment. The new sealant material was evaluated for materials compatibility, fire protection, mechanical design, structural and seismic design, and for hazardous materials considerations. The change does not adversely impact any system or structural design function or requirement. No safety function is impacted in any way.	structure safety feature or safety function. This change does not impact the design basis of the plant as described in the FSAR. Therefore, a change to this figure does not involve an USQ.
E20335A & SAR Change 16-26	This SE considers the incorporation of new top and bottom nozzles that enhance the performance of the current FCF Mk-BW fuel assemblies, while maintaining proper structural characteristics. The new nozzles provide improved debris filtering capabilities and decrease the pressure drop across the fuel assembly. Evaluations have been performed to support the use of the low pressure drop (LPD) nozzles at SQN. While offering smaller individual flow holes, the coarse mesh filter, provides a larger total flow area. The effects of this increase in flow area have been analyzed for SQN and determined to decrease pressure drop due to lower fuel assembly lift, resulting in a holddown margin benefit. LOCA Peak Clad Temperature (PCT) remains below the 2200 °F regulatory limit. In addition to these primary feature changes, the LPD top nozzle implements minor dimensioning improvements, designed to facilitate machining and enhance control of some critical features.	The incorporation of the LPD top and bottom nozzles to the Mark-BW design fuel assembly does not constitute an USQ because: Evaluation of the use of the LPD nozzles on the Mark-BW fuel assemblies determined that the new design will meet the same design criteria and licensing basis criteria as the current Mark-BW design. No new performance requirements are being imposed on any system or component that exceed design criteria. The impact of the LPD nozzle design change for the Mark-BW fuel on the SQN accident analyses has been evaluated. All accidents previously evaluated in the SAR remain valid and bounding. No reduction to relevant acceptance criteria was found. The margin of safety, as defined in the basis for any TS, has not been reduced, which ensures that the consequences of accidents remain within known and acceptable limits.
E20399A & FSAR Change #16-27	This design change is a "documentation only" change issued to resolve discrepancies between the master equipment list (MEL), design drawings, and the as-built plant configuration. All of the discrepancies fell into one of the following categories: 1. Drawing configuration does not match field configuration. 2. Component is identified in MEL but not shown on a drawing.	The changes made by this DCN are documentation only changes to correct MEL entry discrepancies and minor drawing discrepancies with the field and/or with MEL, or to correct or clarify piping class changes for various systems. None of these changes impact any system or component function, or operation, specifically any safety system. Most of the changes involved deleting spurious MEL entries for components which are not in the design and not installed in the plant or are duplicate entries. A SE was written because one of the

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	3. Component is in MEL but information incorrect (e.g., safety related vs. quality related). 4. Class break on drawing missing, incorrect, or unclear. All of the discrepancies were corrected by documentation only changes in E20399A. The DCN includes MEL changes to correct information or to delete components which do not exist in the plant, in the design, or are duplicate entries. DCAs are issued in E20399A to correct inconsistencies or errors in pipe class changes and where the design configuration does not match the field. None of the changes affected the function of any system, specifically any safety system. None of the changes affected the operation of any system.	impacted design drawings is the drawing on which FSAR Figure 11.3.2.1 is based therefore necessitating a revision to that SAR figure. It is recognized that documentation only changes can introduce design configurations which are unanalyzed by modifying a design configuration which has been previously analyzed because it is described in the SAR. All of the documentation only changes made by this DCN were evaluated and none were significant with respect to changes which could affect any safety function or feature, or the means of accomplishing a safety function, as described in the SAR. Therefore, no unanalyzed design configuration is introduced by this change. This change, therefore, does not result in an USQ.
E20642 & FSAR Change #16-64	This modification introduces four ALLIANCE (TM) Lead Test Assemblies (LTAs), beginning with the Unit 2 Cycle 11 core. The ALLIANCE (TM) LTAs are the next generation Mark-BW 17X17 fuel assembly for Westinghouse type plants and were jointly designed by Framatome and Framatome Cogema Fuels (FCF) using common methods and codes for licensed use in Europe and the US. As such, the LTAs are designed to be compliant with approved Mark BW methodologies, have the same licensing basis, and the same exposure limit of 60,000 MWD/MTU. The significant differences between the fuel assemblies are the use of the NRC approved M5 alloy for fuel cladding and structural material, the use of MONOBLOC guide thimbles, and the change of the end and intermediate grids. The Mark BW and ALLIANCE (TM) fuel assemblies are neutronically equivalent, and the two types of fuel assemblies are interchangeable. The LTAs will be placed in nonlimiting core locations. No modifications to the core control strategy or operational procedures are required as a result of introduction of the LTAs.	The use of the ALLIANCE (TM) LTAs remains within the current fuel design bases. The manufacturing, material, and design features of the LTAs do not modify the current fuel assembly functional requirements. Use of the ALLIANCE (TM) LTAs was evaluated and found to not conflict with the continued use of the four blended uranium LTAs loaded in SQN unit 2 in cycle 10. Design basis accidents and safety analysis impacts, including mechanical, nuclear, and thermal hydraulic design, LOCA, and Non-LOCA events, were considered. All accidents previously analyzed in the FSAR remain valid and bounding. Core specific operating parameters were reviewed and confirmed to be within the safe operation envelop established by the plant transient and accident analyses. No TS margin of safety is reduced which ensures that the consequences of accidents remain within known and acceptable limits. Therefore, implementation of this change does not involve an USQ.
E20757 & FSAR Change #16-63	The current SQN steam dump logic will block all of the condenser dump valves when the plant average temperature (T_{AVE}) is reduced below the low-low T_{AVE} interlock (P-12)	FSAR text and tables will be revised to address the use of additional condenser dump valves below a T_{AVE} of 350 °F. Plant cooldown procedure changes will address Mode 4 boron

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	value of 540 °F. Manual interlock bypass switches are provided to permit the use of the designated cooldown valves (3 specific valves out of the 12 total condenser dump valves). Based on plant operating experience, the 3 cooldown valves have insufficient steam flow capacity to maintain the administrative cooldown limit of 75 °F/hr prior to Residual Heat Removal (RHR) system cut-in (approximately 285 °F). This modification allows the P-12 interlock to be disabled during the plant cooldown phase, below a Reactor Coolant System (RCS) average temperature (T_{AVE}) of 350 °F (i.e., MODE 4) thereby enhancing the capability of maintaining the desired administrative cooldown limit. The interlock is restored prior to exiting the applicable procedure for plant cooldown.	concentration prior to disabling the P-12 interlock to endure adequate shutdown margin is maintained and return to criticality is not possible due to positive reactivity resulting from the additional cooldown capability afforded by the additional dump valves. An evaluation of the possibilities for incorrect re-termination of the wire leads when the interlock is restored was included in the modification safety review. Overcooling was also evaluated in the safety review. No unacceptable interactions were identified regarding either concern. Pressurized Thermal Shock (PTS) events were evaluated and were not adversely impacted by this change. Overcooling events as described and analyzed in the FSAR accident analysis were reviewed and the alternate cooldown method established by this modification does not impact the analysis or mitigation of any of these events. No new equipment failure modes or malfunctions are created by the procedurally controlled alternate cooldown method established by this modification. This change does not involve an USQ.
E20798 & FSAR Change #16-68	E20798A allows the site telecommunications group to remove the connection of the NRC Emergency Notification System (ENS) telephones from the connection to the NRC supplied direct lines and tie them to TVA's telephone switching system as requested by the NRC. This is being done as a cost saving measure for the NRC. The only change apparent to the NRC will be new telephone numbers assigned to the SQN ENS. There will be one operational change to the phones. The old system required the operator to lift the telephone and dial the area code and number to reach the NRC, but the revised system will require the operator to dial "9" and "1" prior to the area code. The only physical work required is to change connections in the communications room from the Bell system to the TVA telephone system. The telephone system at SQN is not under configuration control and is maintained by the Telecommunications Group.	The SQN telephone system performance is unaltered by this design change, it will maintain its intended functional capabilities. The components, structures, and systems involved in this modification are not safety related and are not required to support the operation of any safety or quality related components. Therefore, should a failure occur, there will be no impact on the safety of the plant and no unreviewed safety question exists. The SE was necessary to cover changes of the physical descriptions of the telephone system and ENS contained in the SQN SAR. Therefore, this activity does not result in an USQ.

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E20869 & FSAR Change #16-84	EDC E20869A is a documentation only change that revises drawing 1-45N704-1 and section 8.3.2.1 of the SAR to change the specified discharge testing frequency of the 250 volt station batteries from every two to three years to every five years per the guidelines in IEEE-450.	The 250 volt station batteries supply only nonsafety-related loads. The two to three year testing frequency is more frequent than what is recommended in IEEE-450 for nonsafety-related batteries. The existing frequency puts the plant at risk more often than is necessary. The change to the battery maintenance practices will still adhere to industry standards and will ensure that the battery is being monitored and tested at acceptable intervals. Neither the function nor the operation of the non-vital DC system will be changed or affected by this documentation-only change. Therefore, this activity does not constitute an USQ.
G12529A & FSAR Change #16-72	This modification (DCN G12529-A) will replace the obsolete readout modules for the radiation monitoring system. The DCN scope covers the replacement of 56 readout modules, with 53 being located in the main control room, and 3 located on skids/panels in the plant. The scope of Rev A of DCN G12529-A does not include replacement of any safety-related (Class 1E) modules. This DCN will allow the replacement of any of the listed readout modules independent of the other modules. The replacement modules (RM-1000) will perform the same basic functions as the existing modules. The stated accuracy for the RM-1000 module is as good as or better than the obsolete module being replaced. In addition to the replacement of the rate meters, this DCN will also remove the Hi-Radiation Alarm isolation relays, and the V/V isolators from the circuits of ERCW liquid radiation monitor loops. The ERCW liquid radiation monitor loops were downgraded from safety-related class 1E to nonsafety-related class non-1E under previous DCN S08816-A, and the class 1E isolation function provided by the relays & isolators is no longer required.	FSAR Section 5.2.7.4 will be revised to increase the minimum primary-to-secondary detectable leakage rate value for the Condenser Vacuum Pump Monitors and the Steam Generator Liquid Monitors, due to the RM-1000 sensitivity, and to simplify the description of the Component Cooling System Monitors. Also FSAR Table 11.4.2-2 will be revised to raise the minimum detectable concentration for Radiation Monitor loops 0-R-90-118 and 1 & 2-R-90-119, in order to maintain the standard counting interval of one minute, for consistency of calibration and operation. Also FSAR Tables 11.4.2-1 and 11.4.2-2 will be revised to clarify that the actual demonstrated range of each monitor within the scope of this design change encompasses the minimum / maximum detectable concentrations listed in the tables. No other changes are required for the SAR due to the installation of the replacement RM-1000 rate meters. Based on the results of the evaluation performed, it is concluded that the replacement RM-1000 will not invalidate any assumptions for the SAR chapter 15 accident analysis, will not change the basis for any TS, and no new failures or accident initiators are introduced by this generic replacement. Therefore this design change does not result in an USQ.
M08654B	Design Change Notice (DCN) M08654A was initiated in 1992 to replace the Unit 1 Chemical and Volume Control System	The CVCS is expected to function as part of the Emergency Core Cooling System (ECCS) in that the Centrifugal Charging

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	(CVCS) Positive Displacement Pump (PDP) with a new centrifugal charging pump. However, the modification was only partially completed (the PDP was disconnected from its piping) before it was decided that a third pump was not necessary. DCN M08654A was then revised to M08654B to implement design work and field work necessary to complete the PDP removal project. During this reporting period, Stages 6, 7, 8, 10, and 11 of the DCN were completed. This consisted of removing the PDP tripout signal from the motor tripout annunciation separation relay, removing the "PDP Running" signal from the letdown orifice isolation valves, removing the PDP-related handswitches from MCR panel M-5, deenergizing the interlocks associated with the CVCS makeup and recirculation control valve, and removing the Component Cooling System (CCS) flow indicator and temperature indicator along with its associated alarm from the M-27B panel. These modifications were addressed in Revision 5 of the SE.	Pumps A and B serve as the high head injection pumps. The PDP, also referred to as the C Charging Pump, was a safety related component only for pressure boundary integrity of the CVCS and was not relied upon to operate for accident mitigation. Removal of the PDP was addressed in Revisions 1 through 4 of this SE in the 1992 through early 1994, while this Revision 5 adds only "cleanup" items to complete the PDP removal project. The components being removed from the plant are not active safety grade components, and the effects upon active safety grade equipment have been evaluated and found not to be an adverse impact to nuclear safety. Removal of interfaces such as Pump Run inhibits and permissive to flow control valves between the removed PDP and other components reduces the potential for adversely impacting system operation. There are no new failure modes introduced through this modification, and the passive safety function of pressure boundary retention of the remaining piping is not diminished by this modification. This modification required revisions to several FSAR sections and figures to reflect removal of the PDP and related components. This modification does not involve an USQ.
M08655B & FSAR Change #16-71	Design Change Notice (DCN) M08655A was initiated in 1992 to replace the Unit 2 Chemical and Volume Control System (CVCS) Positive Displacement (sometimes referred to as a Reciprocating Charging) Pump (PDP) with a new centrifugal charging pump. However, the modification was only partially completed (the PDP and associated room cooler were removed) before it was decided that a third pump was not necessary. The DCN was then revised to provide for the cutting and capping of the former PDP piping during the SQN Unit 2, Cycle 9 Refueling Outage, and this work was previously addressed in Revision 5 of this SE. DCN M08655A was then revised to M08655B to implement design work and field work necessary to complete the PDP removal project. During this reporting	The CVCS is expected to function as part of the Emergency Core Cooling System (ECCS) in that the Centrifugal Charging Pumps A and B serve as the high head injection pumps. The PDP, also referred to as the C Charging Pump, was a safety-related component only for pressure boundary integrity of the CVCS and was not relied upon to operate for accident mitigation. Removal of the PDP and associated piping was addressed in Revisions 1 through 5 of this SE in the 1992 through early 1999 timeframe, while this Revision 6 adds only "cleanup" items to complete the PDP removal project. The components being removed from the plant are not active safety grade components, and the effects upon active safety grade equipment have been evaluated and found not to be an adverse

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	period, Stages 6 through 12 of the DCN were completed. This consisted of removing the PDP tripout signal from the motor tripout annunciation separation relay, removing the "PDP Running" signal from the letdown orifice isolation valves, removing the PDP-related handswitches from MCR panel M-5, removing the PDP speed control from the pressurizer level signal and charging flow control valve control loop, deenergizing the interlocks associated with the CVCS makeup and recirculation control valve, and removing the CCS flow indicator and temperature indicator along with its associated alarm from the M-27B panel. These modifications were addressed in Revision 6 of the SE.	impact to nuclear safety. Removal of interfaces such as Pump Run inhibits and permissives to flow control valves between the removed PDP and other components reduces the potential for adversely impacting system operation. There are no new failure modes introduced through this modification, and the passive safety function of pressure boundary retention of the remaining piping is not diminished by this modification. This modification required revisions to FSAR sections and figures to reflect removal of the PDP and related components. This modification does not represent an USQ.
M09939	DCN-M09939C added nonsafety-related enclosed manual transfer switches and enclosed receptacles to the 480v power circuits of the Unit 2 Control Rod Drive Mechanism (CRDM) cooler fans inside lower containment. These changes were made to allow the power cables for these fans to be used as temporary power feeds during outages.	This modification has a SE because FSAR Figures 8.3.1-7 and 8.3.1-8 (drawings 1,2-45N749-1 & -2 respectively) will be revised to show the new transfer switches and outage related receptacles as being part of the 480-V power circuits for nonsafety-related Unit 2 CRDM cooler fans. There will not be any change to the system function or operation as described in the FSAR for normal reactor operation. The CRDM air cooling system provides no nuclear safety function, the normal operation of the CRDM fans will not be altered by this modification. Therefore, this activity does not result in an USQ.
M11521A & FSAR Change #16-02	This modification designs and installs piping, piping components, and equipment that will be associated with a new and upgraded secondary sampling system. This modification will complete all tie-ins with mechanical and electrical systems which require a unit outage. A subsequent modification will complete the installation and upgrade. The secondary sampling system covers various process systems associated with the secondary side. The equipment impacted by this change is all nonsafety-related and is located in the nonsafety-related Turbine Building. New sample points with isolation valves will be installed serving the individual hotwell discharge lines, the No. 7 heater drains, and the No. 3 heater drains.	This modification will require a revision to the text and figures of the FSAR which define and describe the configuration of the process sampling system and the interface of other plant systems with process sampling. None of the changes impact any safety-related system, component, or structure. None of the changes impact the operation or performance of any secondary system which could be credited or evaluated in a plant safety analysis or conclusion of any safety analysis regarding transient or accident consequences. None of the changes affect any safety margins associated with nuclear safety or operation of any systems important to the safe shutdown of the unit under normal or accident conditions. No TS or bases are impacted;

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	Modification of the existing sample collection points at the main steam lines, the discharge of the hotwell pumps, the inlet to the condensate booster pumps, and the feedwater header after the no. 1 feedwater heater will be made. Mechanical and electrical connection point to support utility systems which will be needed for new sample analyzer panels will be made by the modification.	no TS margins of safety are impacted in any way. Therefore, this modification does not involve an USQ.
M12158A & FSAR Change #16-04	DCN M12158A addresses replacement of the existing obsolete steam generator blowdown (SGBD) sampling system with a new online sampling and monitoring system, and implements that portion of the Unit 2 SGBD modifications which will not require a unit outage. The upgraded system enables the plant to control feedwater chemistry in a manner commensurate with current ASTM, INPO, and ASME guidelines and protects the steam generator to a greater degree than was previously possible. This modification replaces existing sampling equipment in the Unit 2 Hot Sample Room (HSR); the modification and/or installation of sampling lines; installation of a new local grab sample/corrosion products monitoring (CPM) station; and the modification and/or installation in the Unit 2 HSR of component cooling (CCS), Demineralized Water, and chilled water piping associated with the new equipment.	There are no design basis accidents or anticipated operational transient evaluations in the FSAR which are adversely affected by the proposed modifications. A steam generator tube rupture (SGTR), which results in a release of primary coolant into the secondary side of the affected steam generator, is the only design basis accident which is associated in any way with the proposed modifications. An SGTR would cause increased radioactivity levels in the SGBD sample streams. If the radioactive levels exceed acceptable limits, the SGBD and sample panel drains would have to be manually realigned from the turbine driven auxiliary feedwater (TDAFW) pump room sumps to the floor drain collector tank (FDCT) and the CPM station drains would have to be isolated. The applicable plant procedures will be revised to incorporate administrative controls (similar to those in place for the Unit 1 sample panel) to ensure the proper operator actions are taken in this event. Appendix R impacts are limited to minor changes in combustible loadings associated with the installation of the new sampling equipment and temporary impairment of fire barrier walls and/or floors during implementation. These impacts have been addressed in the DCN package. The proposed modifications are nonsafety-related and interface with, but do not adversely affect, certain nuclear safety-related systems or components. The new secondary sampling equipment is designed to upgrade secondary sampling capabilities during normal operation and do not perform any post-accident sampling or monitoring functions. Therefore, the proposed modifications defined in the scope of DCN M-12158A will not

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		directly or indirectly impact the nuclear safety margins defined for the plant. This modification does not represent an USQ.
M12159A & FSAR Change #16-03	The online instrumentation and sampling system for the condensate and feedwater chemistry system is inadequate for continuous monitoring and does not meet current ASTM, INPO, and ASME guidelines. A new online condensate and feedwater chemistry sampling and monitoring system will enable the plant to control feedwater chemistry in a way that will protect the steam generators to a greater degree than currently possible. This DCN covers the design and implementation of that portion of the Unit 2 Secondary Chemistry Sampling system upgrade modifications which does not require a unit outage. Specifically, this DCN addresses the mechanical, electrical, I&C, and/or civil design and field implementation of the following modifications.· Installation of new sample lines for Condenser 2A, 2B, and 2C hotwell outlet and new sample lines for No. 3 and No. 7 heater drains. Modification of sample collection points for main steam, the hotwell pumps discharge header line, the condensate booster pumps inlet sample line, and the feedwater header sample line downstream of the No. 1 feedwater heater. Installation of new sample booster pumps to increase condensate sample pressure for the individual hot outlet samples, the new SCS sample panel, the additional Ion chromatograph analyzers, and the emergency shower and eye wash to the Secondary Chemistry Lab· Addition or relocation of roughing coolers for secondary chemistry lines· Modification of existing raw cooling water to/from the new roughing coolers to suit installation conditions.	There are no design basis accidents or anticipated operation transients evaluated in the FSAR which are adversely affected by the proposed modifications. A SGTR results in a release of primary coolant into the secondary side of the affected steam generator. However, mitigation of a SGTR does not require nor depend on any equipment which has been installed or modified under DCN M-12159A. The new secondary sampling equipment to be installed under the proposed modification is designed to upgrade secondary sampling capabilities during normal operation; no post-accident secondary sampling components or functions are affected. There are no Appendix R equipment or components nor any nuclear safety-related systems or portions of such systems affected by the proposed modifications. Therefore, the proposed modifications will not directly or indirectly impact nuclear safety. This change does not involve an USQ.
M12563A M13145A & FSAR Change #16-28	Following implementation of the Unit 1 and Unit 2 Secondary Sampling upgrades / installation and successful operation of the Unit 1 and Unit 2 Secondary and Steam Generator Blowdown sample panels, the old Titration Room sample panels and associated sampling equipment will be demolished/removed under DCNs M12563A (U1) and M13145A (U2). The	The impacted sampling equipment located in the Titration Room is not safety related. Abandoned equipment process lines are being cut / capped and are therefore incapable of influencing operational equipment. The sampling functions of the equipment being demolished / abandoned have been completed assumed by other equipment, and all physical /

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	Titration Room will not be refurbished, but will be left suitable for personnel traffic and storage. DCNs M12563A and M13145A will isolate (cut and cap as necessary) sample lines which formerly connected Units 1 & 2 secondary side steam cycle systems in the Turbine Building to the Titration Room in the Auxiliary Building. This modification will also remove existing obsolete condensate sample booster pumps and abandon foundations in place, and abandon or delete electrical loads. This modification will remove obsolete / replaced sample panels from the Titration Room. Prior to the implementation of this DCN, the above described equipment and piping / tubing / conduit will have been deactivated as a result of the implementation of previous DCNs.	electrical ties will be severed. The impact on the FSAR involves descriptions of the sampling system. All of the sampling functions have been absorbed by the new, replacement equipment and panels. Changes to the FSAR related to the description of the sampling system to address the changes made under this modification have no impact on any safety function, safety analyses or TS margins of safety. Therefore, this change does not involve an USQ.
M12745A & M12746A	The previously approved SA/SE's for DCNs M-12745-A and M-12746-A were revised to address the deletion of requirements to protect Channel III and IV NSSS systems with Thermo-Lag Electric Raceway Fire Barrier Systems (ERFBS) after removing the previously installed Kaowool ERFBS.	Kaowool was previously approved and installed at SQN for Appendix R fire protection purposes. The Kaowool was used to protect four channels (I-IV) of NSSS instrumentation for each unit. DCN M12745A removed Kaowool on Unit 1 raceways and DCN M12746A removed Kaowool on Unit 2 raceways and replaces it with Thermo-Lag type ERFBS, due to the documented deficiencies of the Kaowool as a qualified 1-hour fire barrier. Since Kaowool is relatively inexpensive and easy to install, all four channels were wrapped originally, instead of performing an analysis to determine if any of the channels could be left unprotected and still achieve Appendix R compliance. However, Thermo-Lag is a much more expensive and labor-intensive material to install, so a review of Kaowool wrapped raceways was performed to identify the channels required for plant safe-shutdown. It was determined from this analysis that the channel III and IV raceways did not require installation of a Thermo-Lag fire barrier after removal of the Kaowool, due to the redundancy provided by the channel I and II instrumentation. Therefore, no USQ is involved.
M13986 M13987	M13986 (Unit 1) and M13987 (Unit 2) modifications replace the Unit 1 & 2 Lower Containment Air Radiation Monitor RE-	The Containment Upper and Lower Compartment Air Monitors sample lines are isolated during an accident on a Containment

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DCN	DESCRIPTION	SAFETY ANALYSIS
& FSAR Change #16-77	90-106 with a new system and modified the Units 1 & 2 Upper Containment Air Radiation Monitor RE-90-112. The existing radiation monitors consisted of three separate detection elements, particulate, gas, and iodine. The iodine channels which are not required per SQN TSs or Regulatory Guide 1.45 have been deleted. To increase flexibility with TS compliance, the monitor's instrument malfunction alarm has been split into separate gas and particulate instrument malfunction alarms. When the Containment Vent Isolation (CVI) signal was removed from 106 & 112 monitors (DCN M07148, Unit 2 & DCN M07147, Unit 1), the monitors no longer served a safety related function. Therefore, the monitor were downgraded to non-divisional during this modification. This modification also required relocating the Unit 2 Component Cooling System rate meter 2-RM-90-123A. The existing rate meter location on 0-M-12 is needed to allow a new type NIM bin for the new RM-23 digital display device 2-RI-90-106A to be installed for the new system. The relocation of the 2-RM-90-123A allows the Unit 2 rate meter arrangement to match the Unit 1 arrangement on 0-M-12.	Vent Isolation signal. The area-type Containment Building monitors (RE-90-271 through 274) measure the accident range radiation exposure rate during an event. RE-90-106 & 112 detector channels are provided to satisfy the requirements of Regulatory Guide 1.45 for early leak detection of the reactor coolant pressure boundary and for airborne radioactivity monitoring systems. The only system affected by this modification which serves to mitigate an accident is the power distribution system, and the board feeder breakers are going to serve as isolation devices. Therefore, this modification is not associated with any equipment required to mitigate a Design Basis Event. The new system for Lower Containment Air Radiation Monitor 2-RE-90-106 will function essentially the same as the presently installed radiation monitor without the iodine channel. Due to adequate isolation devices and qualification of tubing, the only credible failure modes of the proposed activity would be inaccurate or total loss of radiation monitoring indication. The monitors have a channel check every 12 hours, a channel functional test every quarter, and a channel calibration every refueling outage per TSs. On a failure to meet these channel requirements or total loss of indication the other monitor is available to satisfy the indication requirements for TSs. This SE also revised the descriptive information in FSAR Section 5.2.7.4 provided for reactor coolant leakage detection containment particulate and noble gas radiation monitors. FSAR Section 5.2.7.4, provides specific characteristics for radioactive particulate and gas monitors. The replacement radiation monitors will meet the current SQN FSAR containment upper and lower compartment air monitoring licensing basis. SQN was licensed with limited compliance to RG 1.45 and will continue to have limited compliance to RG 1.45 for the containment noble gas monitors. Therefore, this activity does not result in an USQ.
M14150	The wear slab in the lower ice condenser has moved upward impacting the operability of the lower inlet doors. During	The addition of PG to the foam concrete beneath the wear slab is added to the FSAR description of measures to control floor

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	previous ice outages it was common practice to get water and ice on the wear slab due to wall panel and floor defrosting activities and thermal drilling of the ice baskets. The standing water on the floor migrated into the foam concrete beneath the wear slab and freezing within the foam concrete resulted in the upward slab movement. While measures are now taken to minimize the duration any water is allowed to remain on the wear slab, existing trapped water continues to affect the wear slabs. The change implements dewatering of selected bays using well points as a means to control slab movement. Well points will be permanently installed in additional bays to permit dewatering. In addition, propylene glycol (PG) is to be added to two selected bays to suppress excessive floor movement by limiting the saturated foam concrete from freezing.	movement. The addition of slab well points or injection of PG does not change the function of the ice condenser. There is potential for introduction of small amounts of PG into RHR sump inventory. However, previous analysis shows that the intrusion of large amounts (2000 gallons) of ethylene glycol into the RHR sump during design basis accident conditions is acceptable. PG is very similar in chemical and physical attributes to ethylene glycol, therefore the effects would be comparable. Therefore, this activity does not result in an USQ.
M14153	This modification upgrades the chemical treatment system for the Raw Cooling Water (RCW) and the Raw Service Water (RSW) systems by installing hardware having the capability to reliably inject several raw water treatment chemicals. Portions of the old hypochlorite system will no longer be used and will be abandoned by this modification. The chemical injection portion of the new system will be skid mounted modules which are vendor owned and utilized under contract. Procedures will be in place to allow joint operation of the equipment by the vendor and TVA. The vendor injection equipment will be capable of mixing and delivering a variety of select pre approved treatment chemicals to the RSW/RCW system. Two injection units will be installed, one in the Turbine Building Railroad Bay and one in the Turbine Building near the common RCW suction header. The interconnecting piping and valves will be designed and installed by TVA. In addition, the abandoned auxiliary caustic storage tank will be refurbished for use as a bulk chemical storage tank for this modification. The equipment installed or impacted by this change is all, non seismic, non safety related, non quality related. The electric	This change does not modify or impact any safety related system feature or function. No TS requirement or surveillance is impacted. All of the chemicals utilized for water treatment were included in a NPDES permit change and are approved for use. The SQN fire hazards analysis and MCR habitability analysis were both evaluated and have accounted for the chemicals and their storage. A written SE is required to support a SQN FSAR text revision. The FSAR text was revised to more accurately describe the features of the treatment system. No accident analysis consideration is impacted in any way. This change, therefore, does not result in an USQ.

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	power involved is all non divisional. All of the equipment installed or impacted by this design change is located in the Turbine Building.	
M14220A	DCN M14220 will reconfigure the pressurizer upper instrument taps for the pressurizer level transmitters on Unit 2. The existing Unit 2 configuration which uses an angle root valve and locates the condensate pot at an elevation higher than the instrument tap caused problems on unit 1 and resulted in a conservative lowering of the TS compliance function indication limit from 92% to 80% for both Units 1 and 2. To remedy this situation on unit 2, DCN M14220 will replace angle root valves with Y-Globe valves. The condensate pots will be lowered to the centerline elevation of the new globe valves, and the associated Hot Cal bellows will be lowered to maintain a slope from the relocated condensate pots. This DCN will modify all three unit 2 loops. Minor revisions were completed during reporting time frame not affecting safety analysis conclusion.	Of the accidents in the SAR, the loss of reactor coolant from small ruptured pipes or from cracks in large pipe which actuates emergency core cooling system and the accidental depressurization of the reactor coolant system are identified to be potentially affected by the modification. A loss of coolant from a break in the instrument line would not prevent the normal charging system from making up the level in the pressurizer and would allow the operator to execute an orderly shutdown. With an accidental depressurization of the reactor coolant system, as the accident occurs the average coolant temperature would slowly decrease, but the pressurizer level increases until the reactor trips. The reactor trips during accidental depressurization are due to the reactor protection signals of pressurizer low pressure or overtemperature DT. The credible failure modes associated with this modification include: 1) the level instrumentation losing its water filled reference leg causing it to malfunction, leading to loss of reliable pressurizer level indication, which this modification is designed to correct, and 2) the possibility of a loss of pressure boundary integrity which could result in a SBLOCA. No new failure modes are introduced. Neither of these failure modes constitutes an USQ because the plant is in the same configuration as analyzed and approved in the original plant design. No USQ is introduced by this modification.
M14221 & M14222 & FSAR Change #16-06	These modification allows RE-90-99 to be used as well as RE-90-119 for complying with ODCM requirements for the Unit 1 & Unit 2 Condenser Vacuum Exhaust monitoring. The existing low and mid range monitors experience low flow problems due to insufficient flow rates out the condenser exhaust. This modification will provide flexibility in complying with ODCM requirements. A SE is required for this modification to cover description and table changes to the	The RE-90-99 radiation monitor provides essentially the same information as RE-90-119. These changes do not affect the physical characteristics of any plant equipment relative to the operation of the units. No current plant equipment or operational methodologies are changed. Therefore, these changes have no impact on any FSAR accident analysis. The information provided by these monitors can be provided by other instrumentation. Radioactive effluent monitoring is part

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	FSAR. This modification will replace RE-90-99 radiation monitor scintillator assembly to become the spare/backup for the RE-90-119 radiation monitor. This will make monitor RE-90-99 a low range radiation monitor. The mid range for RE-90-404A covers the existing RE-90-99 range and the category 2 PAM (Reg Guide 1.97) mid range indication will be monitored by the RE-90-404A radiation monitor. The plant computer software has been updated to reflect these changes.	of the Administrative Section of TS and is not addressed in the "Basis". The ODCM monitors and reports radioactive effluents ensuring that no offsite dose limits are exceeded. This change has no impact on TS margin of safety or any offsite dose. No increase in the relationship between operating points, acceptance limits, and actual failure points used as a basis for the margin of safety has occurred as a result of this change. This change, therefore, did not involve an USQ.
M14223A & FSAR Change #16-60	This DCN removes the real time particulate and iodine channel detectors from 0-R-90-101 (Auxiliary Building Ventilation) and 0-R-90-132 (Service Building Ventilation) Process Radiation Monitor loops. The Auxiliary Building Isolation (ABI) signal will no longer be initiated by the particulate and iodine channels of the Auxiliary Building Monitor. Fixed filters for use in chemistry sampling and laboratory analysis will remain in place for the Auxiliary Building Vent Monitor. Finally, three pen recorder 0-RR-90-132 will be removed from 0-M-12 due to the Service Building Vent Monitor iodine and particulate channel inputs being deleted and the noble gas channel information being available on the ICS. Replacement of the globe isolation valves with qualified ball valves for ERCW monitors 0-RE-90-133 & 140, and 0-RE-90-134 and 141 will provide better assurance of monitor availability since there is less potential for the build up of sediment which exists in the ERCW system in these ball valves.	The two FSAR Chapter 15 accidents associated with the auxiliary building features are the Fuel Handling Accident and the Waste Gas Decay Tank Rupture. Detection of the fuel handling accident and isolation of the auxiliary building and start up of the Auxiliary Building Gas Treatment System (ABGTS) is accomplished for the fuel handling accident by the redundant safety grade fuel pool area monitors, 0-RE-90-102 and 0-RE-90-103. The Waste Gas Decay Tank rupture, which does not take credit for isolation, are shown to be within the 10CFR100 limits without filtration; therefore, the consequences of an accident will not be increased. Neither the gas, particulate, or iodine channels of the Auxiliary Building Vent Radiation Monitor are required to initiate any function such as an ABI signal to mitigate a design basis accident or transient as regulatory commitment nor are they designated as TS functions. The Auxiliary Building Vent Monitor will continue to provide a high radiation initiation signal to the ABGTS from the noble gas channel in accordance with defense in depth philosophy. The auxiliary building release is diverted to the shield building vent for design basis accidents. Auxiliary building vent isolation and operation for the ABGTS, an engineered safety feature, provide the required high-range post accident monitoring capability for the auxiliary building post accident release path. There is no unreviewed safety question in deletion of ABI from the particulate and iodine real time channels of the Auxiliary Building Vent Monitor.
M14430B	This SE supports the issuance of DCN M14430B. Numerous	The proposed activity will not impact the control, logic, or

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DCN	DESCRIPTION	SAFETY ANALYSIS
& FSAR Change #16-78	Reach Rod Remote Valve Operator problems exist which cannot be resolved via PMs or WRs. Seven Reach Rods were identified by the plant (Operations and Systems Engineering) as most needing to be fixed to operate per design. Also, a new reach rod for valve 0-VLV-077-1360 will be installed by this DCN. Approximately 43 Stow Reach Rod Position Indicators continue to be unreliable; therefore, the position indication hardware is being removed. Position indicators for Roto-Hammer will remain in place as a Operators aid to provide indication that the valve is in motion when the handwheel is operated. Also, as a result of low dose rates, 14 Stow flexible shaft reach rods are no longer required and will be deleted. Changes will also be made to improve the functionality and reliability of the remote operator so that the remote operator handwheel will function in the same manner as the valve handwheel, therefore the position indicators are no longer required.	function of any equipment or procedure required to mitigate a FSAR Chapter 15 accident. The addition, deletion, and modification of the reach rods will improve the operation of the affected valves and will not increase the 10CFR20 or 10CFR100 post accident dose limits previously established for the facility or restrict access to vital areas or otherwise impede action to mitigate the consequences of reactor accidents. Based on the location of the deleted reach rods for the Residual Heat Removal System, a 1" thick blank plate will be installed at the penetrations for radiological considerations (i.e., radiation shine). Also, in accordance with design criteria for environmental design, none of SQN reach rods are associated with Post-Design Accident Mission Dose (GDC-19) criteria. Therefore, this change will not result in an USQ.
S14227B & FSAR Change #16-15	This safety assessment evaluates restoration of the Units 1 and 2 Boric Acid Filter flow paths which have been bypassed for several years. This safety assessment also evaluates installation of reduced size filters to allow more effective particulate removal at the boric acid and ion exchange filter locations. The normal flow path described in the FSAR will be restored. Documentation changes to allow finer filtration media are evaluated. This design change restores a configuration that is already analyzed. The boric acid filter is described in the FSAR. FSAR Figure 9.3.4-4 shows the boric acid filter bypassed. This DCN differs from that figure, so the figure was changed. All other changes in this DCN are consistent with the FSAR or are not addressed in the FSAR.	Return to service of the boric acid filter is consistent with FSAR descriptions. Use of a filter in the boric acid injection flow path does not inhibit response to any events requiring RCS boration and does not increase the probability of operational transients or accidents discussed in Chapter 15. The restored configuration will not adversely affect safety equipment. Finer filtration at the boric acid filter will increase the cleanliness of the water used for RCS makeup and RCP seal injection. This could result in benefits to RCS seal performance and CCP shaft longevity. The head loss due to finer filtration is acceptable. No new accident scenarios or malfunctions are created by this change. TS margins are not affected. Therefore, there is no USQ associated with this DCN.
S14364B	DCN S14364B will expand the cold leg accumulator contained volume safety limits. The revised safety limits have been analyzed using the SQN plant-specific emergency core cooing system (ECCS) evaluation models (both Framatome and	SQN has both Westinghouse and Framatome fuel in the reactor at the present time, therefore, both types are evaluated for impact on the licensing basis for ECCS operation at SQN. The Framatome analysis for LBLOCA assumes a nominal

DCN	DESCRIPTION	SAFETY ANALYSIS
	Westinghouse) to confirm that all 10CFR50.46 acceptance criteria are met. These values as currently stated represent both the operational and safety limit. The information provided in TS Bases 3/ 4.5.1 is being changed to state that the values for cold leg accumulator level in limiting condition for Operation 3 /4.5.1 are the operating limits. The safety limits are being changed to cover these operational limits plus the instrument uncertainty. The change to the bases will further clarify this point. The new accumulator level safety limits are different than those assumed in the ECCS analysis for large or small break LOCA. The cold leg accumulators were included in the design of the plant specifically to aid in core cooling for a large break LOCA (LBLOCA). However, in a small break LOCA the predicted Peak Clad Temperature (PCT) is not sensitive to the accumulator flow and the core does not typically completely uncover. Thus, these changes will have a negligible impact on the SBLOCA results.	accumulator level (1050 ft^3) and used the minimum TS pressure of 600 psi. A set of sensitivity analyses has been performed to look at the new safety limits. The sensitivity analysis confirmed the ability of the revised accumulator level limits to keep the fuel assembly PCT from exceeding 2200°F. The sensitivity evaluation for the Westinghouse V5H fuel type concluded that the revised accumulator level limits are adequate to keep the fuel assembly PCT from exceeding 2200°F. Based on the ability of the revised limits to meet the 10CFR50.46 acceptance criteria, the expansion of the accumulator level safety limits will not have an adverse impact on mitigating small and large brake LOCA analyses described in FSAR Chapter 15.3.1 and 15.4.3 respectively; and therefore, does not involve an USQ.
T12508B	The DCN revision is to remove monitors that were functionally abandoned under Rev. A of the DCN. The Boric Acid Evaporator Condensate Radiation Monitors are added to the scope of this DCN. They were functionally abandoned under DCN T13217A. The equipment to be removed includes local monitor skids and recorder/instruments in the MCR (0-M-12).	The radiation monitors affected by this DCN do not have an impact on the control, logic or function of any equipment required to mitigate a design basis event. They are not addressed in the TS or in the basis for any TS. Compliance with the TS and OSDM has been maintained without the use of the subject monitors. The Auxiliary Power System is not adversely impact by the removal of these loads.
T13210 & FSAR Change #16-76	This change functionally abandons Essential Raw Cooling Water (ERCW) equipment and components associated with the Auxiliary Essential Raw Cooling Water (AERCW) structure and the Additional Diesel Generator Building (ADGB). The equipment/components are not in service and are not actively used for operation or shutdown of the plant. The change includes closing boundary valves on fluid systems and control air lines to establish a definable boundary of functionally abandoned equipment. To prevent inadvertent operation, the change also deenergizes powered boundary valves and major	A FSAR Figure is revised to reflect the functionally abandoned piping and equipment and closed boundary valves. The change applies only to equipment and components which are functionally abandoned and are not used for plant operation or shutdown, and does not alter the design, function, or method of performing a function described in the FSAR. The design change is associated with a revision to the FSAR which removes discussion of the abandoned equipment. Therefore this activity does not result in an USQ.

DCN	DESCRIPTION	SAFETY ANALYSIS
	components such as pumps and compressors which are functionally abandoned.	
T13212B & FSAR Change #16-80	DCN T13212B is for SQN Unit 2. The DCN revision is to permanently abandon the Unit 2 Gross Failed Fuel Detector (GFFD) by cutting and capping the connecting lines. The Hot Reactor Coolant Sample System module will also be abandoned in place. The GFFD instruments in the control room 2-M-11 will be removed. The GFFD was functionally abandoned under Rev. A of this DCN. This SE also supports a FSAR change to two figures. Figure 7.1.4-1 (MCR layout) will be revised to change the function of MCR panel M-11 (from GFFD to spare). Figure 9.2.3-2 (Demineralized Water flow diagram) will be revised to show the removal the supply to the hot reactor coolant sample system module. The text changes for the GFFD were included in Amendment 13.	The renaming of the main control panel (back row vertical) to spare and the deletion of the hot reactor coolant sample feed from demineralized water (both nonsafety-related systems) does not impact any accident or transient analysis. Neither system is relied upon for accident mitigation. This modification revision does not introduce an USQ.
T13371B	This revision to the SA/SE supports Rev B of the DCN which permanently abandons the Laundry Room Radiation Monitor, the Personnel Air Lock Radiation Monitor and the Condenser Vacuum Exhaust Particulate Monitor. The Radwaste Gas Decay Tank Particulate Radiation Monitor will also be removed. These monitors were functionally abandoned under Rev A (summarized in the Amendment 15 50.59 Report). Two SAR figures (10.4.2-1 Condensate Flow Diagram and 11.3.2-1 Waste Disposal Flow Diagram) will be updated to remove the abandoned monitors.	The radiation monitors affected by this DCN do not have an impact on the control, logic or function of any equipment required to mitigate a design basis event. They are not addressed in the TS or in the basis for any TS. Compliance with the TS and OSDM has been maintained without the use of the subject monitors. The Auxiliary Power System is not adversely impact by the removal of these loads. Therefore, this activity does not result in an USQ.
T14048	DCN-T14048A changes the method by which the diesel generator and electrical panel vent fan is energized. The purpose of this modification is to prevent condensation that occurs when the moist air is blown into the DG rooms when the components in these rooms are relatively cool causing corrosion on ferrous components, degrading cable insulation resistances, and causing corrosion of the generator slip rings and brushes. Following this modification, the vent fans will only operate when the diesels are running and producing heat in these areas	A SE was generated by this modification because the diesel generator and the electrical cabinet fan is shown on SAR figures 8.3.1-3 and discussed in SAR Section 9.4-19. Since this modification has no significant affect on the performance of the diesel generators and does not differ with the operation described in the SAR, no USQ is created by this modification.

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	thus reducing the probability of corrosion due to condensation.	
T14144A & FSAR Change #16-23	DCN-T14144A removes the trip feature from circuit breaker 107 on each of the primary vital battery boards and from circuit breaker 102 on the Fifth Vital Battery Board. The circuit breaker will function as a switch following the modification. This affects only system 250. The protective function of the circuit breaker is provided by an existing fuse.	A SE was required for this change because it affected FSAR figure 8.3.2-1. This modification does not introduce any new failure modes or mechanisms. The protection of the battery board busses is provided by existing fuses. The potential for breaker failure is reduced since the breaker trip function is being removed. Therefore, this activity does not result in an USQ.
T14261	This modification installs connection points on the safety related Emergency Raw Cooling Water (ERCW) system supplying the lower compartment coolers. During periods of high river water temperatures, the lower containment cooling capacity has needed improvement to facilitate maintaining the working environment during outages (i.e., lower temperatures). The connection points added by this modification will allow the installation and use of temporary skid mounted chillers to cool the ERCW supply water which serve lower compartment coolers. The permanent modification to the ERCW system which comprises the connection points will be normally isolated (closed isolation valves installed at each point) and the free end flanged. The connection points will be accessed under plant procedures and normally used during mode 5 and/or 6 and/or defueled.	A previous FSAR change was made to support the companion unit 1 modification implemented under design change T14137A. That change now applies to unit 2 based on implementation of this modification. Specifically, the FSAR text describing the purpose of the chilled water system and the lower compartment air cooling system was revised to include their use for temporary cooling of the ERCW supply to the lower containment ventilation coolers. The ERCW safety function and design is not impacted or challenged by this change. The chilled water system is not safety related and will be used in support of ERCW during periods requiring temporary cooling. Power requirements for the temporary chiller were evaluated and do not impact safety related power supplies nor create unacceptable interactions. This change does not involve an USQ.

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EAR	DESCRIPTION	SAFETY ANALYSIS
EAR 99-NSS-062-1209	This safety assessment evaluates temporary disablement of the standpipe alarm for Unit 1 Reactor Coolant Pump (RCP 1-4). The alarm was received on 6/29/99. Attempts to clear the alarm in accordance with Annunciator Response procedure were unsuccessful. Trends of RCDT level, Reactor Bldg sump inleakage, RCS unidentified leakage, lower compartment air temperature, upper compartment dew point temperature, RCP 1-4 seal supply flow, seal return flow, seal water temperature, lower bearing temperature, motor upper and lower radial and thrust bearing temperatures all appear normal. A previously worked 1998 work order found the lower level switch not working properly. Based on these facts, the probable cause of the alarm is a failed lower level switch. Limited maintenance troubleshooting on the level switch circuit is possible at power. The switch will be replaced next outage and the annunciator will be enabled. FSAR section 5.5.1.3.11 describes RCP shaft seal leakage and standpipe function, including alarms and warnings. Disablement of the RCP 1-4 standpipe alarm is inconsistent with FSAR text. For this reason, a SE has been prepared.	This SE concerns the disablement of the RCP 1-4 standpipe alarm by removing it from the annunciator system scan. The alarm window has been "sealed-in" since 6/29/99 from what appears to be a failed open lower level switch. Alarm functions (high/low level) are presently masked because of this condition. The standpipe level instrumentation provides no system interactions or protective features. Alarm disablement will not defeat any additional protective functions or warnings. The lack of a standpipe alarm does not enhance the probability of a LOCA nor does it introduce any new credible failure modes. The equipment involved with the change is not expected to perform an active safety function. There is no increased likelihood of offsite dose resulting from an accident or equipment malfunction associated with the proposed change. RCP shaft seal leakage is described in Section 5.5.1.3.11 of the FSAR. The proposed standpipe alarm disablement does not increase the consequences of any evaluated malfunctions. Nor does it introduce a new accident or malfunction. No margin of safety information relates directly or indirectly for evaluation against the proposed change. Therefore, this change does not introduce an USQ.

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CHANGES TO THE FIRE PROTECTION REPORT (FPR)
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FPR CHANGE N0.	DESCRIPTION	SAFETY ANALYSIS
Fire Protection Report R5	Revision 5 to the Fire Protection Report (FPR) incorporates the following changes: (1) Added Part II, Section 14.7 to incorporate new Fire Operating Requirement (FOR) 3.7.14 and Surveillance Requirement (SR) 4.7.14 for Emergency Battery Lighting (EBL) units, including compensatory actions and testing frequencies. Also revised Part V "Emergency Lighting and Reactor Coolant Pump Oil Collection" to address new FOR and SR; (2) Revised Part II, Section 14.5 (FOR/SR 3/4.7.11.4) for Fire Hose Stations to allow use of portable hose packs and removal of fire hoses from the hose stations inside the Reactor Buildings; (3) Clarified the definitions for continuous and roving fire watches in Part II, Section 13.0; (4) Clarified compliance with NFPA-72D regarding disabling normal operation of the audible annunciation system in the Main Control Room (MCR), allowable only in the presence of a dedicated operator stationed at the alarm console (Panel 0-M-29); (5) Revised Section 3.31 of Part VII to replace summary of superseded calculation MDQ0026-980017, "Fire Barrier Rating Evaluation for Hollow Block and Partially Filled 8" Concrete Block Walls" with calculation SCG1S591, "Fire Ratings of Hollow Core Masonry Walls"; (6) Corrected minor documentation discrepancy in Part II, Table 3.3-11, in which the number of ionization fire detectors for Zone 230 was listed as 9, instead of the correct number of 10 detectors for the zone; (7) Revised Part II, Section 14.0 to discuss Calculation SQN-SQS2-203, which provides information regarding measures taken to address inoperability of Appendix R equipment that is not currently bounded by existing Tech Specs. The calculation determines that between existing Surveillance Requirements and high priority work orders to support plant operability, sufficient assurance exists such that inoperability of all equipment required for FSSD is recognized and addressed in a timely manner.	The revision did not constitute an USQ. The changes to the Fire Protection Report have been evaluated according to established procedures, and it has been determined that no adverse impact or decrease in nuclear safety has occurred. The changes are administrative, conservative, and provide enhancements to the engineering design bases for the fire protection program.
Fire Protection Report R6	Revision 6 to the FPR eliminated fire detector testing in	The change does not adversely affect the ability to achieve and

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FPR CHANGE N0.	DESCRIPTION	SAFETY ANALYSIS
	inaccessible areas during forced outages requiring cold shutdowns which exceed 24 hours when the testing has not been performed in the previous 6 months. Detectors in inaccessible areas will only be tested during refueling outages. Inaccessible areas are defined in the FPR as 1) areas having a radiation expose rate greater than 100 mrem/hr and 2) areas designated as inaccessible by the Fire Protection Manager due to conditions that pose immediate danger to life and health from environmental or operational conditions.	maintain safe shutdown in the event of a fire due to 1) the very low overall fire detector failure rate of 0.07% per year, and 2) the provision of two or more detectors in almost all areas which contain equipment required for FSSD. Therefore, this activity does not result in an USQ.

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AMENDMENT 16 CHANGES TO THE UFSAR
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FSAR CHANGE	DESCRIPTION	SAFETY ANALYSIS
FSAR Change #16-07	This change is a revision to FSAR Section 13.5 to correct and clarify the description of system operating instructions, emergency instructions, and abnormal operating instructions. The following clarifications/revision are made: 1. The terms "immediate" and "subsequent" operator actions are being deleted from the discussion of abnormal operating procedures. Immediate actions are defined as those actions which are expected to be performed immediately from memory prior to consulting the associated procedure. There are no immediate actions identified in the SQN AOPs; therefore, a distinction of "immediate" and "subsequent" is not needed. 2. The description of the contents of AOPs and EOPs is revised to eliminate automatic actions which may occur during an event. 3. The description of the emergency procedures is revised to eliminate the discussion of postulated events and to eliminate the unnecessary description of the responsibility of the operator who discovers an emergency, which does not apply in many conditions.	This FSAR change did not impact the actions taken in response to an accident or abnormal plant condition. This change does not involve revision to any procedures nor changes in operator actions in response to any emergency or abnormal condition. The absence of immediate actions in AOPs and the absence of automatic actions and event discussion in EOPs and AOPs does not impact or affect the ability of operators to mitigate any accident or malfunction of plant equipment. This change affects the FSAR description of all SQN AOPs and EOPs and is not limited in impact to any specific accidents. The following operator considerations were evaluated with respect to the FSAR change: 1. Failure to perform required operator actions within required time. 2. Inappropriate actions taken OR appropriate actions NOT taken due to misdiagnosis or misunderstanding of the event. 3. Failure to recognize and take appropriate action as a result of failed automatic actions. None of these considerations resulted in new or adversely impacted failure modes or concerns. Therefore, this change does not involve an USQ.
FSAR Change #16-38	The proposed FSAR change is necessary to delete excessive detailed and incorrect information from FSAR Section 12.1.2, Design Description. The Fuel Transfer Shielding part of this section states that the dose rates three feet above the Spent Fuel Pool is less that 1.0 mrem/hr when fuel is being removed from the reactor vessel. RADCON surveys indicates dose rates at the Spent Fuel Pool Manipulator Crane floor are as high as 2mr/hr during nonrefueling periods and as high as 10mr/hr during refueling activities. It is also noted that the manipulator crane floor is 9.79 feet above the Spent Fuel Pool normal water level. Therefore, the FSAR less than 1mr/hr dose rates at three feet above the water level are exceeded. The FSAR information is	This FSAR change is acceptable and will not delete relevant information from the FSAR. Implementing the proposed activity will not impact the control, logic, or functions of any equipment required for normal plant operations or equipment required to mitigate a FSAR Chapter 15 design basis accident. Furthermore, the proposed FSAR change will not impact any plant procedures including ALARA practices and subsequently adherence to 10CFR20 and 10CFR100 criteria will not be jeopardized. Therefore, the proposed FSAR change is safe from a nuclear safety stand point and is not an USQ.

FSAR CHANGE	DESCRIPTION	SAFETY ANALYSIS
	also contrary to the administrative dose controls utilized for the refueling operators while working on the refueling manipulator crane. The Spent Fuel Pool dose rates will typically increase over the life of the plant due to increasing number of spent fuel rods being stored in the pool. In addition, dose rates to the refueling operator while moving fuel is dependent on the number and size of fuel rod leaks within each fuel bundle and the radioactive contaminants suspended in the pool. Therefore, no attempt to update this information should be implemented but rather replace with text which reflects ALARA practices.	
FSAR Change #16-39 & SQ981671PER	This SA/SE supports an FSAR Change Request. Specifically, the purpose of the change is to revise FSAR Section 8.3.1.1, Diesel Generator Description, to accurately depict the equipment configuration. This is part of corrective action for PER No. SQ981671PER. FSAR Section 8.3.1.1 is being revised to delete the paragraph that details the two devices that produce alarm signals should the two engines of a diesel-generator set receive different amounts of fuel. One of these devices is a synchro device that gives an alarm signal should the difference in the actuator shaft positions for the two engines exceed a certain tolerance. The other device is an injector rack limit switch that will initiate an alarm should one engine be on full rack when the other is not. The four SQN D/Gs are not equipped with the device that produces an alarm for one engine at full rack while the other is not. The D/Gs have the synchro device described in the paragraph but they will be abandoned the next time they malfunction. The electronics (Singer Card) associated with the synchro device is no longer available. The alarm function is redundant and the exhaust temperature differential alarm can detect an imbalance of fuel to the engines. This change will make the information presented in the FSAR agree with that presented in the design criteria and actual plant configuration.	This change involves alarm functions listed in FSAR Section 8.3.1.1. The design function of the deleted alarms is fulfilled by the existing exhaust temperature alarm. System design and functional requirements of the DGs are not affected. Therefore, there is no increase in the probability of an accident, equipment malfunction or increase in the consequences of an accident, or equipment malfunction previously evaluated in the SAR. This activity cannot create a possibility for an accident or malfunction of a different type than any evaluated previously in the SAR and will not reduce the margin of safety as defined in TSs. This change is in compliance with safety requirements as specified in design basis documents. Therefore, on the basis of the evaluation of effects, it is concluded that the proposed change is acceptable from a nuclear safety standpoint and does not involve an USQ.
FSAR Change #16-45	The proposed FSAR Change addresses documentation	The SAR change is a documentation only change to properly

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FSAR CHANGE	DESCRIPTION	SAFETY ANALYSIS
	discrepancies identified during a review of Chapter 4.5. The revisions update incorrect information appearing in the SAR and do not reflect any changes in design bases analyses, documents, hardware changes, or procedure revisions. The specific changes are: 1. Sections 4.5.4.1.3 (Core Flow Design Basis), 4.5.4.3.1.1 (Flow Paths Considered in Core Pressure Drop and Thermal Design), and Table 4.5.4.2-1 (note g) (Reactor Design Comparison Table - Four Loops in Operation) have been revised to reflect the proper bypass flow used in design bases analysis. The bypass flow value in these sections was corrected from 7.0% to 7.5%. A correction was also made to the thermal design flowrate given in Table 4.5.4.2-1 (note g). The flowrate was incorrectly given as 360,000 gpm. This value was corrected to be 360,100 gpm. 2. Figure 4.5.2-15 (Mark-BW Burnable Poison Rod) was revised to delete incorrect information and reflect the proper burnable poison rod in use at SQN.	reflect the current analyses and design bases of the plant. No changes are made to design criteria, licensing basis criteria, or any design basis analyses. These document changes will have no impact on safety, LOCA, and non-LOCA analyses. No new performance requirements are being imposed on any system or component and no Tech Spec margin of safety is adversely impacted. The document change is acceptable from a nuclear safety perspective. Therefore this activity does not result in an USQ.
FSAR Change #16-49	The purpose of these FSAR changes is to provide a more detailed description of Reactor Coolant System (RCS) chemistry used at SQN and the documentation to allow the use of hydrazine in the Residual Heat Removal (RHR) during scheduled shutdowns. Hydrazine is routinely utilized in the RCS during startup and addressed in Section 5.1 Plant Startup, however, the Plant Shutdown section does not contain the use of hydrazine in the RHR. Since hydrazine is utilized in the RCS, there should be no material compatibility issues relative to the use of hydrazine.	The proposed FSAR changes are necessary to clarify information in FSAR Sections 1.2.2.10, 5.2.3.4, 9.3.4.2.2, 9.3.4.2.5 to allow the addition of hydrazine to the RHR. Implementing the proposed activity will not impact the control, logic, or functions of any equipment required for normal plant operations or equipment required to mitigate a FSAR Chapter 15 design basis accident. Furthermore, the proposed FSAR changes will not impact any plant procedures involving ALARA practices but should provide a reduction in outage RHR dose rates. Adherence to 10CFR20 and 10CFR100 criteria will not be jeopardized. These changes to the FSAR are only an update based on a review by Chemistry and recommendations provided by EPRI. Therefore, the proposed FSAR changes are safe from a nuclear safety standpoint and not an USQ.

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FSAR CHANGE	DESCRIPTION	SAFETY ANALYSIS
FSAR Change #16-52	The proposed activity is necessary to change the Tennessee River dilution factor from 20% to 60% in FSAR Section 11.2.8. This change is necessary to be consistent with the current ODCM practices and applicable dose codes used in offsite dose calculations. In 1980 when Sequoyah began operation and making liquid releases to the river 20% of the river flow was assumed and used as part of our offsite liquid dose calculations. This 0.2 factor was based on a model study of the diffusers in 1979. Later in 1982 a field study was performed which documented that the actual available river flow was at least 64%. In 1988 the available dilution flow was questioned and verified to be higher than 0.2. Therefore, in 1989 Sequoyah's ODCM river dilution factor was updated to 0.6. However, the FSAR was not changed and remains at 0.2.	As previously stated, the proposed FSAR change is necessary to correct information in the FSAR Section 11.2.8. Implementing the proposed activity will not impact the control, logic, or functions of any equipment required for normal plant operations or equipment required to mitigate a FSAR Chapter 15 design basis accident. Furthermore, the proposed FSAR change will not impact any plant procedures including ALARA practices and subsequently adherence to 10CFR20 and 10CFR100 criteria will not be jeopardized. This change to the FSAR is only an update based on an engineering study performed following Unit-1 startup. The changes will improve our effluent monitoring program by making it as accurate as possible based on all available data. Therefore, the proposed FSAR change is safe from a nuclear safety stand point and is not an USQ.

SEQUOYAH NUCLEAR PLANT
FUEL RELOAD CHANGES
REQUIRING 50.59 EVALUATIONS

FUEL RELOAD	DESCRIPTION	SAFETY ANALYSIS
Unit 1 Cycle 11 Reload	This evaluation considers reactor core reload and operation for cycle 11 operation in all modes to a maximum cycle core average burnup of 21,196 MWd/MTU, including a power coastdown. Changes to be made for cycle 11 include: · Revised core configuration - SQN unit 1 will be refueled by replacing 84 spent fuel assemblies with 84 fresh FCF Mk-BW fuel assemblies. The remaining irradiated fuel assemblies will be shuffled. All Westinghouse fuel assemblies will be discharged. Fuel inserts including secondary sources, rod cluster control assemblies (RCCAs), and plugging devices will also be shuffled. No discrete burnable poison cluster will be used in cycle 11.· Revised Core Operating Limit Report (COLR) - The following changes will be made to the COLR: 1. Axial flux difference limits will be revised. 2. Heat Flux Hot Channel Factor - revision to the NSLOPEAFD, PSLOPEAFD, NSLOPEf2(DI), and PSLOPEf2(DI) factors; revision to the increase in FQM factor for compliance with SR 4.2.2.2.e. 3. Nuclear Enthalpy Rise Hot Channel Factor - Revision to the TRH values. 4. Deletion of limits and references to Westinghouse fuel. · Fresh fuel assembly design changes - The fresh fuel will utilize top and bottom nozzles of a new low pressure drop design. The design features of the new top and bottom nozzles will enhance the performance of the current FCF Mk-BW fuel assemblies, while maintaining proper structural characteristics. The new nozzles provide improved debris filtering capabilities and decrease the pressure drop across the fuel assembly.	Evaluations have been made of the effect of the core configuration specified for cycle 11 upon the safety analyses described in the SAR. These evaluations included consideration of the mechanical design of the new fuel assemblies, nuclear and thermal-hydraulic design of the cycle 11 core, and effects of the cycle 11 core upon the LOCA and non-LOCA accidents discussed in the SAR. All conclusions presented in SAR were found to remain valid and no new credible failure modes have been created for the cycle 11 reload. Based upon the preceding information and the following: 1) an end-of-cycle 10 burnup between 18,427 and 19,587 MWd/MTU, 2) termination of cycle 11 burnup at or before 21,196 MWd/MTU, including a power coastdown, and 3) adherence to plant protective and operating limitations given in the TSs, the COLR, and the operating guidelines. There are no unreviewed safety questions or TSs changes identified as a result of the SQN Unit 1, cycle 11 core design. Therefore, the cycle 11 reload design is licensable under 10 CFR 50.59, and requires no prior USNRC approval.
Unit 2 Cycle 11 Reload (All Modes)	This evaluation considers reactor core reload and operation for cycle 11 operation in all modes to a maximum cycle core average burnup of 21,314 MWd/MTU, including a power coastdown. Changes to be made for cycle 11 include: · Revised core configuration - SQN unit 2 will be refueled by replacing 80 burned fuel assemblies with 76 fresh Framatome Cogema Fuels (FCF) Mark-BW assemblies and 4 FCF ALLIANCE Lead Test Assemblies (LTAs) and shuffling the	Evaluations have been made of the effect of the core configuration specified for cycle 11 upon the safety analyses described in the SAR. These evaluations included consideration of the mechanical design of the new fuel assemblies, nuclear and thermal-hydraulic design of the cycle 11 core, and effects of the cycle 11 core upon the LOCA and non-LOCA accidents discussed in the SAR. All conclusions presented in SAR were found to remain valid and no new

FUEL RELOAD	DESCRIPTION	SAFETY ANALYSIS
	remaining burned fuel assemblies for cycle 11. No Westinghouse fuel assemblies will be used in cycle 11. Fuel inserts including secondary sources, rod cluster control assemblies (RCCAs), and plugging devices will also be shuffled. Discrete burnable poison rods will be utilized in the LTAs, while integral burnable poison in the form of gadolina will be utilized in the fresh Mark-BW fuel assemblies.· Revised Core Operating Limit Report (COLR) - The following changes will be made to the COLR: 1. Axial Flux Difference limits are revised. 2. Heat Flux Hot Channel Factor - revision to the NSLOPEAFD, PSLOPEAFD, NSLOPEf2(DI), and PSLOPE f2(DI) factors; revision to the increase in FQM factor for compliance with SR 4.2.2.2.e. 3. Nuclear Enthalpy Rise Hot Channel Factor – revision to the Maximum Allowable Peaking (MAP) limits. 4. Deletion of limits and references to Westinghouse fuel. Changes to the Fresh Fuel Assembly Design - The following changes will be made to the fuel assembly design of the fuel to be loaded in cycle 11: 1. ALLIANCE LTAs - The ALLIANCE LTAs are the next generation fuel product from FCF. The design differs from the standard Mark-BW fuel assembly in materials and design of the guide thimbles and grids. The advanced material, M5, is utilized for the fuel rod cladding and guide thimbles. A different fuel rod end plug is also used. The MONOBLOC guide thimble design is introduced which provides for a constant outer diameter with a thicker wall in the dashpot region. 2. All fresh fuel will utilize top and bottom nozzles of a new low pressure design. Nozzles of this design were previously introduced in the fresh fuel for SQN unit 1 cycle 11. The design features of the new top and bottom nozzles will enhance the performance of the fresh fuel assemblies, while maintaining proper structural characteristics. The new nozzles provide improved debris filtering capabilities and decrease the pressure drop across the fuel assembly.	credible failure modes have been created for the cycle 11 reload. Based upon the preceding information and the following: 1) an end-of-cycle 10 burnup between 19,411 and 20,573 MWd/MTU, 2) termination of cycle 11 burnup at or before 21,314 MWd/MTU, including a power coastdown, and 3) adherence to plant protective and operating limitations given in the TSs and the COLR, there are no unreviewed safety questions or TSs changes identified as a result of the SQN Unit 2, cycle 11 core design. Therefore, the cycle 11 reload design is licensable under 10 CFR 50.59, and requires no prior USNRC approval.
Unit 2 Cycle 11	This evaluation considers Unit 2 reactor core reload and	For operation in restricted mode 6 and mode 5, prevention of

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FUEL RELOAD CHANGES
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FUEL RELOAD	DESCRIPTION	SAFETY ANALYSIS
Reload(Mode 5 & 6 w/Restrictions)	operation while in restricted Mode 6 and Mode 5 during the cycle 10/11 refueling outage. Operation in other modes will be considered in a future safety assessment/SE. The planned cycle 11 loading pattern was revised due to the identification of suspect fuel rod(s) in assembly Y04 which was scheduled for reuse in cycle 11. Y04 and it's symmetric partners (Y01, Y02, and Y03) will be discharged. These assemblies will be replaced with 4 assemblies (X15, X16, X22, and X27) that operated in cycle 10, but were scheduled to be discharged. Selected irradiated fuel shuffles were required to reoptimize peaking factors. Note: This screening review/SE only justifies reactor core reload and operation in restricted mode 6 and mode 5 with the revised cycle 11 core loading pattern. Operation in other modes is not permitted until a PORC approved screening review/SE is completed which justifies operation with the revised loading pattern in these modes.	inadvertent criticality is the primary concern related to safety as discussed in the SAR. By ensuring that the RCS and refueling cavity boron concentration is maintained above the refueling boron concentration, or in Mode 5 above the concentration which will provide adequate operator response time to an inadvertent dilution, sufficient subcritical margin has been demonstrated. In addition, administrative and TS controls have been instituted to ensure that all likely sources of dilution will be valved closed and secured while in Mode 6. The control rod drive system will remain inoperable which will prevent an inadvertent rod withdrawal accident. The highly conservative refueling boron concentration ensures that there is greater than 5% delta k/k margin to criticality (calculated with the most reactive control rod fully withdrawn from the core). This conservative limit ensures that no criticality concern exists during the control rod binding tests that are performed after rod latching. These tests manually move each control rod (one at a time) to ensure that no binding exists and that the rod will move freely prior to vessel head reinstallation. Continuous monitoring of the reactor neutron flux will be performed to ensure that any unexpected increase in core reactivity will be detected with sufficient time for operator intervention. There are no unreviewed safety questions identified as a result of loading the revised reload core and operation in restricted mode 6 or mode 5 for cycle 11.

SEQUOYAH NUCLEAR PLANT
CHANGES TO THE OFFSITE DOSE CALCULATION MANUAL (ODCM) REQUIRING 50.59 EVALUATIONS

ODCM	DESCRIPTION	SAFETY ANALYSIS
ODCM R44	DCNs M14221 (Unit 2) and M14222 (Unit 1) modified the condenser vacuum exhaust radiation monitors. The 1,2-RE-90-99 radiation monitor scintillator assembly was replaced with a similar assembly to the 1,2-RE-90-119 radiation monitor. This makes monitor 1,2-RE-90-99 a low-range radiation monitor, and now can serve as a spare/backup for the 1,2-RE-90-119 radiation monitor. This ODCM modification allows 1,2-RE-90-99 to be used as well as 1,2-RE-90-119 for complying with ODCM requirements for condenser vacuum exhaust monitoring. As part of this revision, flow indicators were deleted from the ODCM since they are not being used or needed for condenser vacuum exhaust effluent monitoring. Two flow elements are being used during routine operation.	The 1,2-RE-90-99 radiation monitor provides essentially the same information as 1,2-RE-90-119. These changes do not affect the physical characteristics of any plant equipment relative to the operation of the units. No current plant equipment or operational methodologies are changed. Therefore, these changes have no impact on any FSAR accident analysis. The implementation of this change does not alter the event classification of previously analyzed accident, transients, or equipment failure. The information provided by these monitors can be provided by other instrumentation. Therefore, there is no increase in the consequences of an accident or equipment malfunction previously evaluated in the SAR. This modification cannot create a possibility for an accident or malfunction of a different type than any evaluated previously in the SAR. Radioactive effluent monitoring is part of the Administrative Section of TS and is not addressed in the TS Basis. The ODCM monitors and reports radioactive effluents ensuring that no offsite dose limits are exceeded. This change has no impact on TS margin of safety or any offsite dose. No increase in the relationship between operating points, acceptance limits, and actual failure points used as a basis for the margin of safety has occurred as a result of this change. This change, therefore, did not involve an USQ.
ODCM R45	The changes to the ODCM will result in minor corrections to the setpoint for the Shield Building radiation monitors (1,2-RM-90-400), and Auxiliary Building (0-RM-90-101), radiation monitor. The setpoint calculation for the Service Building (0-RM-90-132) is also impacted by this change but is conservatively set at a small percent of this value; therefore, no changes are required. This change is due to an earlier modification in the Service Building ventilation system which bypassed part of the flow around the radiation monitor. The safety factor for the liquid radwaste monitors is being changed to a variable value, which will be controlled by plant procedures. At this time SQN has higher than normal tritium	SQN TSs were reviewed and there are no identified impacts. The FSAR was reviewed and resulted in no FSAR impacts. The scope of DCN G-12529-A does not include any safety-related radiation monitoring loops. The setpoint changes are only minor and well below the site gaseous effluent release limit of 500 mRem/yr and the liquid 10CFR20 limit of 10 times Effluent Concentration Limits. All other changes can be considered editorial since they do not impact any ODCM calculations or effluent monitoring methodology of 10CFR20.

ODCM	DESCRIPTION	SAFETY ANALYSIS
	concentration in effluents which results in a depressed setpoint value by the ODCM and Effluent Management System software. This change will allow for minor fluctuations in the monitor background and radiation monitor noise. DCN G12529-A replaced the obsolete readout modules (RP-1 and RP-30 rate meters) for the radiation monitoring system. The DCN scope covers the replacement of 56 readout modules, with 53 being located in the main control room, and 3 located on skids/panels in the plant. The scope of Rev. A of DCN G12529-A does not include replacement of any Safety-Related (Class 1E) modules. The replacement of the obsolete rate meters with upgraded rate meters by DCN G-12529-A does not add or change any of the radioactive release pathways currently existing in the plant however the ODCM was revised to reflect these changes. All other changes can be considered editorial since they do not impact any ODCM calculations or effluent monitoring methodology.	

SEQUOYAH NUCLEAR PLANT
OTHER CHANGES
REQUIRING 50.59 EVALUATIONS

OTHER	DESCRIPTION	SAFETY ANALYSIS
PER 99-000151-000 & FSAR Change #16-55	During normal operation, Component Cooling System(CCS) temperature tracks ERCW temperature on Train B and runs approximately 4 degrees F higher than ERCW temperature on 1A and 2A trains. Historically, ERCW temperature has been recorded to have approached 36 degrees F and has maintained temperatures below 40 degrees for short periods. When ERCW temperature drops below 40 degrees F, the CCS B train temperature can also drop below 40 degrees F during normal operation due to minimal heat loads. CCS A Train heat loads are sufficient to maintain CCS temperature above 40 degrees F during normal operations. If ERCW temperature drops below 40 degrees F, CCS B train temperature can drop below the temperature assumed in the CCS operating mode calculation. CCS Design Criteria lists the CCS design temperature as approximately 40 degrees F. The corrective actions will revise the documents above to reflect a minimum CCS temperature of 35 degrees F.	FSAR Section 9.2.1.2 is revised by this change to reflect a minimum CCS temperature of 35 degrees. The CEB piping analysis calculations have been revised to reflect the reduced minimum temperature with no adverse impact on the CCS piping or components. The revised piping analysis demonstrates that this activity will not increase challenges to the CCS, ECCS, or other safety systems credited in the SAR for accident analysis mitigation such that safety system performance is degraded below the design basis. Therefore, as a result of the revised piping analysis, no safety margin is unacceptably reduced and no assumptions of any safety analyses in the FSAR for accident or transient mitigation are impacted in any way, nor is the basis for any TS impacted. This activity does not result in an USQ.
PER 00-008137-000 & FSAR Change #16-62	FSAR Section 8.3.1.1, Standby Diesel Generator Operation, was revised to clarify the operation of the loss-of-voltage/degraded voltage relays. The relays will operate regardless of power feed. However, if the board is being supplied from the Emergency Diesel Generator, an existing interlock circuit prevents load shedding and resequencing of loads if the Diesel Generator output voltage is greater than 70% of nominal. This interlock prevents unnecessary Shutdown Board load shedding during expected voltage transients resulting from diesel load sequencing. The FSAR was revised to reflect this.	This change only involves a clarification of how the loss-of-voltage/degraded voltage relays and the interlock circuitry operate. There are no impacts on plant safety due to this change. Therefore, this activity does not constitute an USQ.
Storage of Low Level Radioactive Wastes & FSAR Change #16-10	This SE is prepared to allow the storage of resins and/or solid wastes at the Low Level Radioactive Waste, On-Site Storage Facility (LLRW OSF) at Sequoyah. This SE justifies the current physical configuration of the LLRW OSF. The total approved accumulated activity (taking credit for decay if	The generation of and storage of LLRW in the OSF module represents normal activities associated with nuclear plant operation. Under the conditions of dose term limits, the accident potential related to storing LLRW is no greater than that for existing operations associated with waste handling for

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OTHER CHANGES
REQUIRING 50.59 EVALUATIONS

OTHER	DESCRIPTION	SAFETY ANALYSIS
	desired) that can be stored within the OSF is 88,500 Curies. The total yearly generated activity stored in the facility shall not exceed 17,744 Curies. This limitation is based on limiting the yearly cumulative exposure received by LLRW handling personnel. The curie limit will be included in implementing procedures. This change requires a revision to the FSAR. Specifically, FSAR Section 11.5.6.3 is being revised to properly describe the use of a site crane by deleting the mention of a "mobile" gantry crane that was part of the original LLRW OSF design. Reference to a security "gatehouse" is also being deleted from the FSAR text. Both of these original design features are not required to safely store LLRW on-site, consistent with the guidance provided by Generic Letter 81-38.	off-site disposal. FSAR Chapter 15 and Section 11.5.6.3 do not discuss any accident scenarios relative to the use of the LLRW OSF. Design Criteria DC-V-9.2 describes radiological events relative to operation of the LLRW OSF; however, the consequences of these events are bounded by Chapter 15 accident analyses. The two design basis waste handling events were previously analyzed during the original licensing effort for the facility, that of dropping one cell cap into one loaded, open resin storage module cell and dropping a resin liner from a height of 25 feet to a point immediately adjacent to and outside a storage module. These evaluations determined that the OSF facility has been designed to withstand normal, severe, and extreme loading conditions. This change does not involve an USQ.

SEQUOYAH NUCLEAR PLANT
CHANGES TO PROCEDURES
REQUIRING 50.59 EVALUATIONS

PROCEDURE	DESCRIPTION	SAFETY ANALYSIS
0-SI-SFT-030-132.A & B	Procedures 0-SI-SFT-030-132.A and 0-SI-SFT-030-132.B test the TS operability of A & B train of the Auxiliary Building Gas Treatment System (ABGTS). There have been problems in the past meeting the total flow requirements with the normal, non safety related, Auxiliary Building supply and exhaust fans running at the same time the test is being performed. This proposed change to the test procedures will allow alternate fan alignments from the normal alignment while maintaining a negative pressure in the Auxiliary Building.	The proposed activity will temporarily place the normal operation fan alignment different than as described in the SAR during normal operation. However, the intended design features for both normal and accident conditions remain effectively unchanged. The Auxiliary Building will still be maintained at a slightly negative pressure with respect to the outside during normal operation, and the ABGTS will still perform its safety function upon receipt of an ABI signal. LOCA analysis provided in FSAR Section 15.5.3 is not challenged since the system is capable of maintaining a slightly negative pressure. Likewise, the FHA analysis provided in FSAR Section 15.5.6 will not be challenged since the system is capable of functioning as designed. No increase in the relationship between operating points, acceptance limits, and actual failure points used as a basis for the margin of safety in the TSs will occur as a result of this activity. Therefore, this activity does not involve an USQ.
0-SI-SFT-030-149.A & 0-SI-SFT-030-149.B	This change is a revision to 0-SI-SFT-030-149.A & 0-SI-SFT-030-149.B to remove the 2000 CFM vacuum relief flow requirement listed in the acceptance criteria of the surveillance as an acceptance criteria for operability of the ABGTS system. This criteria will remain in the SI as a requirement for passing the surveillance. This SA/SE assesses the operability of the ABGTS system with respect to the 2000 CFM vacuum relief flow listed in the acceptance criteria of 0-SI-SFT-030-149.A & 0-SI-SFT-030-149.B. The ABGTS draw down test is performed to determine if the ABGTS can maintain a 0.25 inch w.g. or greater vacuum on the ABSCE. The 2000 CFM vacuum relief flow was established to simulate additional in leakage to account for possible degradation in the ABSCE boundary over the time between testing. This is defined in the basis to TS 3.7.8 which states: "The minimum vacuum relief flow requirement in TS Surveillance Requirement 4.7.8.d.3 is for test purposes only. It is intended to demonstrate an acceptable	Maintaining a 2000 CFM minimum vacuum relief flow on the ABGTS system is not required for this system to perform it's design basis accident mitigation functions. The ABGTS will still be able to maintain 0.25 inch w.g. or greater vacuum and establish that vacuum within the accident analysis time requirements The 2000 CFM flow will still be met to assure operating margin over the extended time period between tests; however, it will be removed as a requirement for operability of the ABGTS system. This is consistent with the Basis of TS 3/4.7.8. Therefore, this change does not constitute an USQ.

PROCEDURE	DESCRIPTION	SAFETY ANALYSIS
	level of ABGTS performance margin by simulating an ABSCE boundary breach. The inability to meet the specified minimum test condition under other circumstances does not challenge the operability of the ABGTS." The ability to establish a 2000 CFM vacuum relief flow while maintaining vacuum is a good indication of system performance but is not required for the ABGTS to perform it's accident mitigation functions.	
0-SO-74-1	Revised procedure to allow isolation of the Residual Heat Removal (RHR) heat exchanger bypass line on a temporary basis in Mode 5 to support maintenance/testing activities. This change requires throttling total RHR flow and Component Cooling System (CCS) flow to the RHR heat exchangers (HX) as necessary to control Reactor Coolant System (RCS) temperature. This method of controlling RCS temperature deviates from the description of RHR system operation during plant cooldown in FSAR Section 5.5.7.	This procedure revision did not alter the shutdown cooling function of the RHR system and did not significantly change the method of accomplishing that function. Isolation of the RHR HX bypass line and controlling temperature by throttling CCS flow to the in-service RHR HX does not conflict with or violate any MODE 5 Tech Spec requirements and will not prevent the RHR system from performing its Tech Spec function in this MODE. Administration procedural controls will prohibit performance operation with this configuration in MODE 4 because the minimum flow rate of 2000 GPM required by The SQN TSs could result in a violation of LCO 3.9.8.1. The current FSAR description of operation specifies that during plant cooldown the RHR HX by outlet valves will be regulated to control RHR return temperature and that the bypass valve will be regulated to give the total required RHR flow. This procedural change will allow the HX bypass line to be isolated and temperature to be controlled by adjusting CCS flow to the RHR HX. This is an acceptable alternate configuration for the applicable MODE of operation and meets the intent of the present operating requirements as described by the FSAR. The impact on system pressure, shutdown cooling, shutdown margin, and ECCS operation were all evaluated and were not adversely impacted by this change. This revision will not increase the probability or consequences of any accident/equipment malfunction and will not place the plant outside the bounds of existing safety analyses. Also, this revision will not impact or degrade the Emergency Core Cooling function of the RHR system in Modes 1-4. Therefore,

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PROCEDURE	DESCRIPTION	SAFETY ANALYSIS
		this revision does not involve an USQ.
0-TI-CEM-000-001.3	To improve RCS chemistry control on Unit 2 only, a constant Tave = 303.4°C pH of 7.15 is being implemented. Based on comparisons with the approved RCS Modified pH = 6.9 to 7.4 control program, it is concluded that implementing a RCS Constant pH = 7.1 or 7.15 control program with a maximum lithium concentration of 3.5 ppm should (1) reduce core and loop ^{58}Co and ^{60}Co inventories and; hence, lower ex-core dose rates, (2) minimize potential for Axial Offset Anomalies problems, (3) reduce likelihood for large crud releases, especially at end of cycle operation and shutdown, (4) prevent increase and possibly reduce thickness of crud on the core, (5) not appreciably increase rate of zircaloy fuel cladding corrosion, (6) have a negligible effect on Alloy 600 Primary Water Stress Corrosion Cracking (PWSCC) crack growth rate, and (7) not significantly increase susceptibility of Alloy 600 steam generator tubing PWSCC.	The changes do not affect the physical characteristics of any plant equipment relative to the operation of the units. No current plant equipment or their operational methodologies are changed; therefore, these changes will have no impact on any FSAR accident analysis. This is a routine change to improve primary chemistry during unit operation and outage activities. It should be considered an overall improvement in the way SQN operates. This change will not impact any plant equipment relative to the safe operation of either unit and will not involve an USQ.
6gpm Circulating Pump OOS	In standby state, the Emergency Diesel Generator (EDG) engines are equipped with two Lube Oil Circulation Systems to provide prelubrication of internal components, and a heated Jacket Water System to maintain the combustion air canals at an elevated temperature. Both the heated oil and jacket water systems aid in internal combustion and acceleration during rapid start conditions. The 6gpm Lube Oil Circulation Pump will be removed from service for maintenance activities while the EDG remains operable from a TS perspective. The EDG must be declared inoperable if the fluid system must be breached for the maintenance activity.	The engine manufacturer has stated in correspondence that occasional EDG starts with the engine at operating temperature are inconsequential to engine integrity. The timeframe the engine manufacturer is discussing will be when the engine has little (if any) prelubrication oil circulation because of low lubrication oil viscosity. The engine manufacturer information to prolong engine life places emphasis on maintaining prelubrication oil flow to the turbocharger bearings to avoid increased wear during rapid starts with hot engine conditions and less emphasis on maintaining prelubrication of the main crankshaft bearings. Based on 1) the information provided by the engine manufacturer stating inconsequential affects on engine integrity with little (if any) prelubrication during hot engine standby conditions, 2) periodic testing that provides information confirming engine capability to meet acceleration requirements in the off normal condition with the 6gpm Lube Oil Circulation Pump out of service, 3) absence of problems

SEQUOYAH NUCLEAR PLANT
CHANGES TO PROCEDURES
REQUIRING 50.59 EVALUATIONS

PROCEDURE	DESCRIPTION	SAFETY ANALYSIS
		identified during maintenance inspections, and 4) no problems identified in periodic lube oil analyses, there is no increase in the probability that a SQN EDG would not perform the intended safety function. Therefore, this activity would not result in an USQ.

SEQUOYAH NUCLEAR PLANT
CHANGES IN THE FACILITY - TEMPORARY MODIFICATIONS
REQUIRING 50.59 EVALUATIONS

TACF	DESCRIPTION	SAFETY ANALYSIS
TACF 0-00-018-024 R1	In order to replace the Auxiliary Building (AB) Chiller A, Raw Cooling Water (RCW) Inlet Isolation Valve, the RCW system will have to be isolated. The lack of individual RCW branch isolation valves and the RCW loads downstream of the RCW Booster Pumps (i.e., Control Rod Drive Equipment Room Coolers, Unit 2 Steam Generator Blowdown (SGBD) Sample Coolers, and Glycol Chillers, et al) make the RCW isolation difficult. Due to the difficulty in obtaining an isolation boundary, both the AB Chiller A and B, RCW Inlet Isolation Valves will be replaced during this Temporary Alteration Control Form (TACF). This activity involves three basic steps: 1) RCW Deadleg Flush; (Flush a 30-foot RCW Deadleg prior to ERCW-RCW Intertie to drain inside Auxiliary Building) 2) Essential Raw Cooling Water (ERCW) -RCW Intertie; (Intertie ERCW to RCW using flexible connection at flood mode spool piece connections in order to provide suction to RCW Booster Pumps & loads) and 3) RCW Valve Replacement (Replace AB Chillers RCW Inlet Isolation Valves with new similar valves.) The existing valves are 8-inch, 150#, Henry Pratt butterfly valves and the new valves are 8-inch, 150#, Allis Chalmers butterfly valves.	A SE was required to support the ERCW-RCW Intertie since the ERCW does not normally supply water to the RCW system - only during flood mode protection. The increased demand on the ERCW system is anticipated to be small and have no adverse impact on the overall ERCW system to supply its required loads. LCO 3.7.4 will conservatively be entered whenever ERCW is used to supply the RCW system. This activity does not result in an USQ.
TACF 0-98-026-241	Due to the recent failures of Qualitrol Sudden Pressure Relays and past failures of Gas Operated Relays on the main transformers causing erroneous unit trips, this activity assesses the temporary removal of Qualitrol Sudden Pressure Relay and Gas Operated Relay protective trips from the Plant's Offsite Power Transformers until a more reliable relay scheme can be implemented. The transformers affected are the actual Offsite Power Transformers (CSST A, B, and C) and the Cooling	The sudden pressure or gas-operated relay cannot cause or prevent a fault in a transformer. Their function is to minimize the energy released from the main transformer if an internal fault were to occur and to minimize the duration of the fault to the system and the main generator. Additionally, the relay provides an initiation of the transformer's fire deluge system. Protection for the transformers is also is provided by the differential, over current, and the gas-operated relays.

TACF	DESCRIPTION	SAFETY ANALYSIS
	Tower Transformers (CTT A and B) which are in parallel with the associated CSST. The sudden pressure relay initiates a trip signal on a rate of rise in the transformer tank pressure and is considered similar in speed to the differential or over current relays. Each of the identified transformers has this relay in its protective relaying trip circuit. The gas-operated relay initiates an alarm on gas accumulation or low oil level, and a trip on oil surging to the conservator or low-low oil level. Only the trip contact is being disabled. The gas-operated relay trip on oil surge is slower than the differential or over current relays. Only the CSSTs have this relay in its protective relaying trip circuit. Revised to address returning each transformers' sudden pressure protection to service successively as they are completed.	Although the gas-operated relay will not provide a trip it does annunciate to alert personnel so that the transformer can be removed from service in the event of excessive gas accumulation, which is indicative of low-level fault currents in the transformer. Fire protection system initiation while less diverse still has redundant thermal sensors located around the transformer. These sensors will initiate the deluge system in the event of an oil fire to minimize propagation of the fire effects to adjacent equipment. The worst case event that a transformer fault could have on nuclear safety would occur if the fault was of sufficient duration to cause the loss of offsite power. For this to occur, multiple failures of power and protective equipment would be required even with the sudden pressure relay disabled. This change is bounded by the loss of offsite power analysis and is not an USQ. Returning the transformer sudden pressure relay protection to a transformer individually or as a group does not change the basis or the conclusions of the Safety Analysis.
TACF 0-99-019-018	Underground Auxiliary Boiler Fuel oil supply and return lines were determined to be leaking. As required by Sequoyah Spill Pollution Prevention plan and 40CFR1112 which prohibits intentional release of fuel oil into the environment, the fuel supply pumps and the fuel oil lines have been isolated from service. This condition has made the auxiliary boilers inoperable. To ensure auxiliary boiler availability, a temporary fuel supply arrangement consisting of an oil tanker of approximately 9000 gallon capacity, a fuel transfer pump and associated piping and petroleum hoses will be required.	The temporary fuel supply arrangement for Auxiliary Boilers will add approximately 9000 gallons of fuel storage within 200 feet of the Control Building. The temporary oil tanker is a closed container like the diesel fuel oil storage tank currently in use on site. The diesel fuel inside the tanker is a Class II combustible, with a flash point between 100 and 140 degrees F. In its normal state, diesel fuel is not an explosive, and therefore does not present any unusual situations beyond the capabilities of the site fire brigade. In case of a fire to the temporary oil tanker, the smoke is not expected to intrude into air intake of the control room. If any smoke should enter the control building air intake, CRI will be ensured by the operator action per AOP-N.01 "Plant Fires." Therefore, control room habitability is not affected. The auxiliary boilers and associated piping are non-safety related and are not addressed in the TSs. The FSAR was reviewed and determined that there is no adverse impact on the FSAR. Based on the results of the

SEQUOYAH NUCLEAR PLANT
CHANGES IN THE FACILITY - TEMPORARY MODIFICATIONS
REQUIRING 50.59 EVALUATIONS

TACF	DESCRIPTION	SAFETY ANALYSIS
		evaluation performed, it is concluded that the temporary diesel fuel oil tanker and associated equipment, will not compromise control building habitability, will not invalidate any assumptions for the SAR chapter 15 accident analyses, will not change the basis for any TS and no new failures or accident initiators are introduced. Therefore, this TACF will not result in an USQ.
TACF 1-00-004-001	Valve 1-VLV-1-485A is the manual isolation for flow indicator 1-FI-1-53. A furmanite repair was performed at this valve in accordance with TACF 1-00-004-001 and WO 00-002885-000 using approved procedures. The repair is necessary due to an external steam leak at a welded connection directly upstream of the valve. The steam leak is sealed by closing the valve and injecting furmanite into the valve body on the upstream side. The valve will stay in the closed position following the furmanite repair and will remain closed until a permanent repair is made at a later time. The purpose of this valve is to provide manual isolation for 1-FI-1-53 so that on line maintenance may be performed. This furmanite repair prevents the use of flow indicator 1-FI-1-53. However, flow indicator 1-FI-1-53 was already inoperable due to the existing steam leak. 1-FI-1-53 indicates high pressure steam flow to the 1A1 Moisture Separator Reheater (MSR).	1-FI-1-53 indicates high pressure steam flow to the 1A1 Moisture Separator Reheater (MSR). This is a local indication only and is located in the 1A1 MSR HP doghouse. This indicator is not safety related and is not required for process control or safe shutdown of the unit. The valve 1-VLV-1-485A is TVA class H and is nonsafety related. This activity does not affect any plant safety system or any plant operation important to safety. Therefore, the possibility of an accident of a different type or the possibility for a malfunction of a different type than any previously evaluated in the SAR is not created. Therefore, this activity did not involve an USQ.
TACF 2-00-015-006	This SE addressed installation of a temporary Furmanite repair to stop a steam leak through the body of 2-HCV-006-2843. Valve -2843 is located in the startup vent line of Moisture Separator Reheater 2B1. The repair involved removal of the handwheel and installation of a "Furmanite box" around the valve. The objectives of the repair were to (1) stop external leakage of steam, (2) eliminate a potential hazard to personnel, (3) prevent further degradation of the leak, and (4) eliminate a potential source of air inleakage to the condenser. Because valve -2843 is shown on SAR diagram 10.4.9-1 and would be left disabled in the open position by the proposed activity, a SE was performed.	The SE determined that the Furmanite repair of valve -2843 did not alter the operation of the MSRs as described in the SAR or invalidate assumptions of any analyses. An additional valve in the flow path (2-VLV-006-1246) was used to provide isolation of the startup vent flow path. The repair did not change the probability of any evaluated accident nor change the classification of any event. Valve -2843 and the MSR HP startup vent function do not initiate any accidents or events evaluated in the SAR. The MSRs and startup vents do not contribute to mitigation of any evaluated event. The repair did not increase challenges to safety systems nor change any offsite dose analysis assumptions. The repair did not change any

SEQUOYAH NUCLEAR PLANT
CHANGES IN THE FACILITY - TEMPORARY MODIFICATIONS
REQUIRING 50.59 EVALUATIONS

TACF	DESCRIPTION	SAFETY ANALYSIS
		accident mitigation assumptions. The repair did not create the possibility of a new accident nor create any new accident initiators. No new credible failure modes affecting event mitigation were introduced. No change in the relationship between operating points, acceptance limits, and actual failure points occurred as a result of this activity. For these reasons, no analyses were impacted and the activity did not constitute an USQ.
1-01-003-062 & FSAR Change #16-82	This TACF documents the out-of-normal configuration status of the disconnected reach rod for 1-VLV-062-0538. This is a manual valve located in the Unit 1 669' pipechase that acts as the bypass valve for 1-FCV-062-89. The valve can be manually operated locally by its handwheel, located in the Unit 1 EL 669 pipe chase. Temporary operation via the local handwheel poses no challenge to operation of the valve. The reach rod will be tagged to document the temporary alteration and Operations procedures will be revised to reflect the interim operating condition of the valve.	This valve is not an EOI controlled valve but it and the reach rod do appear on SAR Figure 9.3.4-1. Therefore, since it appears in the SAR and its configuration is out-of-normal, a SAR change is required. The temporary disconnection of this reach rod will not adversely affect the ability of accident mitigation equipment to perform its design function and will not change or invalidate any assumptions in the offsite dose analyses for SQN. The projected maximum dose rate at the local handwheel is less that 100 mrem/hr. This was documented in DCN D20035A for its 50.50 analysis. Therefore, this change does not vary from the requirement of FSAR 12.1.2. Also, in accordance with the design criteria for environmental design, this reach rod is not associated with Post-Design Accident Mission Dose (GDC-19) criteria. 10CFR100 and 10CFR20 limits as provided in the FSAR Chapter 15.5 are unaffected by this activity. This activity does not involve an USQ.

SEQUOYAH NUCLEAR PLANT
CHANGES IN THE FACILITY - TEMPORARY BUILDINGS
REQUIRING 50.59 EVALUATIONS

TBCF	DESCRIPTION	SAFETY ANALYSIS
TBCF 2000-01	The evaluation for Temporary Building Change Form (TBCF) 2000-01 R0 evaluates the temporary erection of the RADCON building on elevation 690' on the Unit 2 side of the Auxiliary Building between column lines U and S, and A13 and A14. The SR/SA/SE also evaluates the addition of temporary data cables, power cables, video equipment, and/or communication cables which service the elevation 690' building (temporary structure), the elevation 734 RADCON building, the elevation 669' building and RADCON monitoring of personnel activities inside upper and lower containment during the Unit 1 Cycle 10 refueling outage. The temporary buildings are used by RADCON as locations from which to direct refueling outage activities. The fire protection spray/sprinkler system over the 690' temporary building will be functional, but considered inoperable due to the horizontal obstruction to the normal sprinkler system spray area. The 690' temporary building will not be equipped with a fire detection system. Therefore, FORs 3.7.11.2 and 3.3.3.8 will be entered prior to the installation of the first roof panel. The Fire Brigade and manual hose stations are adequate to serve as the backup fire suppression equipment in compliance with the Fire Protection Report requirement. Prior to Unit 1 mode 4, all temporary data cables, power cables, video cameras, MG Electronic Dosimeter equipment, and radio system equipment shall be removed from inside primary containment in accordance with 0-SI-OPS-000-187.0. All temporary power supplies, communication cables, etc., routed outside containment may be used in all modes of plant operation and shall be removed from the elevation 669', 690' and 734' buildings prior to two weeks following the completion of the U1C10 refueling outage. Prior to two weeks following completion of the U1C10 refueling outage, the 690' temporary building shall be disassembled and removed. The FORs can be exited when the elevation 690' temporary building is removed.	The installation of the 690' temporary building and the power supply, data cables, video equipment, etc., has been determined to be acceptable from a safety standpoint and does not involve an unreviewed safety question. The sprinkler system located above the building shall be declared inoperable prior to the installation of the first roof panel. Adequate compensatory measures exist to ensure that a fire will not result in a malfunction of any safety-related piping, cable trays, and/or equipment as previously evaluated in the SAR. Other design considerations for temporary erection of this building such as the effect on deadweight, seismic missiles, electrical loads, electrical separation, etc,. are all within the TSs and Safety Analysis report requirements. In conclusion, the installation of the 690' temporary building and the power supply, data cables, video equipment, etc., has been determined to be acceptable from a safety standpoint and does not involve an USQ.
TBCF 2000-02	The evaluation for Temporary Building Change Form (TBCF) 2000-02 R0 evaluates the temporary erection of the RADCON	The installation of the 690' temporary building and the power supply, data cables, video equipment, etc., has been determined

TBCF	DESCRIPTION	SAFETY ANALYSIS
	building on elevation 690' on the Unit 2 side of the Auxiliary Building between column lines U and S, and A13 and A14. The SR/SE also evaluates the addition of temporary data cables, power cables, video equipment, and/or communication cables which service the elevation 690' building (temporary structure), the elevation 734 RADCON building, the elevation 669' building and RADCON monitoring of personnel activities inside upper and lower containment during the Unit 2 Cycle 10 refueling outage. The temporary buildings are used by RADCON as locations from which to direct refueling outage activities. The fire protection spray/sprinkler system over the 690' temporary building will be functional, but considered inoperable due to the horizontal obstruction to the normal sprinkler system spray area. The 690' temporary building will not be equipped with a fire detection system. Therefore, FORs 3.7.11.2 and 3.3.3.8 will be entered prior to the installation of the first roof panel. The Fire Brigade and manual hose stations are adequate to serve as the backup fire suppression equipment in compliance with the Fire Protection Report requirement. Prior to Unit 2 mode 4, all temporary data cables, power cables, video cameras, MG Electronic Dosimeter equipment, and radio system equipment shall be removed from inside primary containment in accordance with 0-SI-OPS-000-187.0. All temporary power supplies, communication cables, etc., routed outside containment may be used in all modes of plant operation and shall be removed from the elevation 669', 690' and 734' buildings prior to two weeks following the completion of the U2C10 refueling outage. Prior to two weeks following completion of the U2C10 refueling outage, the 690' temporary building shall be disassembled and removed. The FORs can be exited when the elevation 690' temporary building is removed.	to be acceptable from a safety standpoint and does not involve an unreviewed safety question. The sprinkler system located above the building shall be declared inoperable prior to the installation of the first roof panel. Adequate compensatory measures exist to ensure that a fire will not result in a malfunction of any safety-related piping, cable trays, and/or equipment as previously evaluated in the SAR. Other design considerations for temporary erection of this building such as the effect on deadweight, seismic missiles, electrical loads, electrical separation, etc., are all within the TSs and Safety Analysis report requirements. In conclusion, the installation of the 690' temporary building and the power supply, data cables, video equipment, etc., has been determined to be acceptable from a safety standpoint and does not involve an USQ.

SEQUOYAH NUCLEAR PLANT
TECHNICAL SPECIFICATION BASES CHANGES
REQUIRING 50.59 EVALUATIONS

TS BASES	DESCRIPTION	SAFETY ANALYSIS
Tech Spec Bases 3/4.4.2 & .3 Rev	This evaluation supports a revision to the TS Bases and provides clarification to remove automatic mode operation for PORV operability. This clarification does not affect TS LCO Action or Surveillance requirements. The current language provided in the TS Bases contains an error and deviates from information provided in the SAR. The TS Bases contains language that describe one of the PORV functions as: "automatic control of PORVs to control RCS pressure." This is contrary to the FSAR. The FSAR discussions do not credit the automatic actuation of PORVs for controlling RCS pressure. The proposed clarification will provide agreement between information presented in the FSAR and the TS bases.	The proposed revision clarifies the TS Bases with regard to the design basis function of the PORVs for reactor coolant system pressure. Current operating procedures and accident analysis evaluated in the SAR are not affected by this clarification within the TS bases. The proposed revision to TS Bases does not affect any plant equipment physically or operationally. The proposed revision does not affect accidents evaluated previously in the FSAR but updates the TS Bases to be in agreement with these analyses. Plant equipment and operating practices will not be impacted by the proposed revision. The clarification for having manual capability for PORVs in Modes 1, 2, or 3 remains consistent with plant operating procedures and FSAR Chapter 15 accident analysis. The proposed Bases change does not affect setpoints or operational parameters. In addition, PORVs are not credited in the design basis analysis for reactor coolant system overpressure transients. Therefore, the proposed change does not represent an USQ.
Tech Spec Bases 4.0.3 Rev	The proposed TS Bases revision will remove the description of reporting requirements from Section 4.0.3. This portion of the Bases indicates the necessity to report to NRC the missing of a surveillance as a condition prohibited by TSs. This Bases discussion is repetitive and nontypical for information that is normally maintained in the Bases. The requirements of SPP-3.5, 10 CFR 50.72 and 73, and the guidance of NUREG-1022 provide adequate awareness of reporting requirements such that this elimination will not affect appropriate reporting of missed surveillances as necessary. This revision will not affect plant equipment or operating practices at SQN or the process intended to maintain the CFR requirements for reporting required conditions to NRC.	The proposed revision to remove a redundant requirement to report missed surveillances to NRC has been verified to be adequately covered by site processes and regulatory requirements that do not require specific identification in the TS Bases. No plant equipment, operating practices, structures, setpoints, or functions are affected by this change including reporting requirements because of the redundant nature of the discussion proposed for deletion. Based on the plant functions not being impacted and the administrative requirements being maintained without change, the proposed revision to the TS Bases will not impact nuclear safety or result in an USQ if implemented.

SEQUOYAH NUCLEAR PLANT
CHANGES TO THE TECHNICAL REQUIREMENTS MANUAL (TRM) REQUIRING 50.59 EVALUATIONS

TRM	DESCRIPTION	SAFETY ANALYSIS
TRM R10 Change 00-03	This revision to the TRM adds conservative requirements that are not currently addressed by the TS 3/4.7.7. The MCR chillers and air handling units if not treated as "attendant equipment," an LCO does not exist for the equipment and the plant outside of its design basis (two trains of safety-related equipment available for single failure considerations) when one train is out of service. This proposed activity is to replace immediate administrative controls (standing order) and institute temporary TRM administrative controls for inadequate TS requirements until TS Change 99-18 is approved by NRC. This change adds new LCOs TR 3.7.13 which states that: a) With one CRATCS inoperable, restore the inoperable system to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. b) With both CRATCS inoperable, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.	This proposed TRM change is to implement conservative administrative actions, in accordance with NRC Administrative Letter 98-10, while a TS change is being processed. There are no design basis accidents or credible failure modes that are adversely affected as a result of this change. The revision to the TRM is more conservative, with respect to nuclear safety, as it provides additional LCO actions and expands applicability of existing LCO actions as compared to the current TSs. It does not conflict with current TS requirements and does not adversely impact the potential for accidents or malfunctions or the consequences of an accident as well as the margin of safety. Therefore, this change to the TRM does not involve an USQ.
TRM R7 Change 99-05	This activity involves the revision of the TRM to resolve inaccuracies and to add revision numbers to affected pages for historical tracking of changes. The first change removes a provision associated with the ISI and IST requirements for ASME Section XI testing in Section 4.0.5 that would allow the TRM requirements to supersede the provisions of the code. The second change adds a clarification to Section 6.1.2 to allow an NRC approved SER to satisfy the 10 CFR 50.59 requirements for TRM revisions. The revision number additions are an administrative change and only provide the ability to track the revision history of the TRM pages and do not affect the intent of the TRM requirements.	The proposed revision involves the correction of inconsistencies in the TRM from an administrative standpoint. The deletion of the ASME Section XI exception ensures that code compliance is maintained and will not be superseded by a TRM requirement. This change helps maintain the safety functions of plant systems by supporting the testing necessary to ensure operability. The allowance to use an NRC approved SER is an equal to or better provision because an NRC review activity can not be an USQ by definition since NRC has performed a review. The SER eliminates any potential for an USQ to exist and completely satisfies the intent of a 10 CFR 50.59 evaluation. The addition of revision numbers to the TRM pages is administrative change only and does not affect any plant function but will allow for better tracking of revisions to the TRM. Therefore, the proposed revision to the TRM will not create an USQ.

SEQUOYAH NUCLEAR PLANT
CHANGES TO THE TECHNICAL REQUIREMENTS MANUAL (TRM)
REQUIRING 50.59 EVALUATIONS

TRM	DESCRIPTION	SAFETY ANALYSIS
TRM R9 Change 00-02	The proposed activity is necessary to establish an interim I-131 dose equivalent concentration for the Units 1 & 2 Reactor coolant system specific activities of 0.25 mCi/gm rather than operating at the 0.35 mCi/gm concentrations specified in TS 3/ 4.4.8. Based on Westinghouse Nuclear Safety Advisory Letter NSAL-00-004, nonconservative assumptions were generically used in the calculation of the accident initiated iodine spiking rates in the RCS. The advisory letter provides a correction factor to the current TS iodine dose equivalent concentration limits to account for the nonconservative assumptions in the interim. SQN Units 1 & 2 TS I-131 dose equivalent concentrations are 0.35 mCi/gm. Applying the correction factor will yield an interim administrative limit to the I-131 dose equivalent concentration of 0.25 mCi/gm until a SQN specific analysis is completed. Based on I-131 dose equivalent historical data for both units from cycle 5 operation, this new administrative limit has not been challenged.	As previously stated, the proposed activity will limit the I-131 dose equivalent concentration of the RCS to 0.25 mCi/gm on an interim basis until an SQN plant specific analysis for iodine spiking relative to the Main Steam Line Break and Steam Generator Tube Rupture analysis are completed. This interim concentration is conservative relative to the current TS 3/ 4.4.8 value of 0.35 mCi/gm. Therefore, the proposed activity is safe from a nuclear safety stand point and will not result in an USQ.

SEQUOYAH NUCLEAR PLANT
WORK REQUESTS (WR), WORK ORDERS (WO), & CLEARENCES (HO)
REQUIRING 50.59 EVALUATIONS

WR/WO/HO	DESCRIPTION	SAFETY ANALYSIS
1-HO-98-0011 R1	This evaluation addresses long term off-normal alignment of the Unit 1 Ice Condenser system. Since January 5, 1998, floor glycol temperature control valve 1-TCV-61-71 has been isolated and bypassed due to icing of the TCV which renders it incapable of controlling automatically. Manual valves (VLV-61-750 and -752) in line with the TCV are tagged closed and manual bypass valve -751 is throttled partially open. These valves are all located outside of containment. Operating instruction 0-SO-61-1 authorizes this bypassing of the TCV when temperature control is erratic. Isolation of the TCV is implemented and controlled under clearance 1-HO-98-0011 and SPP-12.4, System Equipment and Status Control. This clearance was reviewed due to greater than one year existence, it was determined that the temporary configuration is inconsistent with FSAR paragraph 6.5.6.2, which describes bypassing of the TCV "during brief maintenance periods." This evaluation supports revising FSAR paragraph 6.5.6.2 as follows, "...A manual throttling valve bypassing the temperature control valve can perform the latters' function..." removing the phrase "during brief maintenance periods."	None of the accidents evaluated in the SAR can be initiated by the Ice Condenser System or any of its components. Containment isolation and integrity is not affected by the proposed activity. All equipment important to safety that is potentially affected by the activity remains fully capable of performing intended design functions. TS basis 3/4.6.5.1 identifies that the minimum required ice weight includes a 15% allowance for sublimation of ice during an operating cycle. Although floor glycol temperature will be manually controlled to the normal operating temperature under the proposed activity, SAR paragraph 6.5.6.1 states that an annual sublimation rate of just 4% to 5% would result if there were no cooling of the floor. The proposed activity therefore cannot reduce the margin of safety defined in the applicable TS basis The evaluation concluded that long-term operation of the floor glycol system with 1-TCV-61-71 isolated and manually bypassed does not create an USQ.
1-HO-98-635 /WO9800370100	The Battery Room II Exhaust Fan 1B1-A was tagged out of service per hold order 1-HO-98-635. This was done to allow repair of the fan per WO 98-003701-000. The purpose of the battery room exhaust fans is to ensure that hydrogen concentration in the battery rooms is maintained at less than 2% by volume. Each battery room has two 100% capacity exhaust fans located on the roof to provide ventilation. The Battery Room II Exhaust Fans 1B1-A and 1B2-B provide this ventilation for battery room 1-II. The battery room exhaust fans also assist (but are not required) in maintaining the battery room temperatures within normal operating parameters. The problem with the Battery Room Exhaust Fan 1B1-A was determined to be an electrical ground on the motor. The work was delayed due to a material restraint with obtaining a new	Hydrogen gas can be explosive at concentrations greater than 2%. The hydrogen concentration can reach these levels in a short time period when the batteries are on high level equalize charge and there is no ventilation. However, if the batteries are performing their design function as a backup power supply, they cannot be on high level equalize charge. Therefore, the battery room exhaust fans would not be required to be in continuous operation while the batteries are performing their design function. During high level equalizing of the vital batteries, at least one battery room exhaust fan must be in service per plant procedure. Operation of the battery room exhaust fans also can affect the room temperatures. Plant procedures performed each week verifies that a battery room exhaust fan is in operation and that the room temperature is

WR/WO/HO	DESCRIPTION	SAFETY ANALYSIS
	fan motor.	between 60°F and 82°F. The battery room exhaust fans do not perform any direct accident mitigation function. Each battery room has two 100% capacity exhaust fans. Adequate administrative controls exist to ensure at least one battery room exhaust fan is in operation when required. Therefore, the activity is safe from a nuclear safety stand point and does not result in an USQ.
WO00004465000	This evaluation addressed a maintenance activity to repair internal air leaks within No. 3 Heater Drain Tank (HDT) level controller 1-LIC-6-106. The repair was performed while Unit 1 operated at 100% power. Automatic control of No. 3 HDT level was interrupted during repair of 1-LIC-6-106. Valve 1-LCV-6-106A was mechanically restrained in the full open position. A regulated air supply was connected to quick-disconnect fittings at the instrument air input to 1-LCV-6-106B. Tank level was then continuously, manually controlled by plant personnel who monitor tank level and adjusted the temporary air supply input to valve -106B as needed to maintain level in the desired range. SAR Section 10.4.9.3, (Safety Analysis, Heater Drains and Vents) states that 1-LCV-6-106B trips closed on loss of a No. 3 HDT Pump if unit load is greater than 85% power to prevent damage to the remaining pumps due to insufficient net positive suction head. Because operator control of the valve interfered with this pump protection function, a SE of this maintenance activity was performed.	The SE determined that this activity had no affect on the overall probability of occurrence of any evaluated events and did not change the classification of any event. Neither the No. 3 HDT Pumps nor any other feature of the Heater Drain and Vents system provides a mitigative function for any event evaluated in the SAR. Additionally, the activity (1) did not increase challenges to safety systems or change, (2) did not invalidate assumptions in offsite dose analyses, (3) did not invalidate accident mitigation assumptions, (4) did not introduce new credible failure modes affecting event mitigation and (5) did not change operating points, acceptance limits, or failure points. For these reasons, no analyses were impacted and the activity did not constitute an USQ.
WO00009119000 & FSAR Change #16-70	WO 00-009119-000 replaced Unit 1, loop 4 RCP motor and rotating element assembly (includes main flange, shaft, impeller, bearing, thermal barrier assembly) to resolve unacceptable vibration problems indicated on this pump. The replacement components were evaluated and confirmed to meet or exceed the original components specifications regarding fit, form, function, mechanical specification, and material specification. Because the work activity was a replacement of	As part of the equivalency evaluation for the rotating element assembly installed under WO 00-009119-000, the original design parameters were evaluated and confirmed that the replacement pump rotating assembly will meet or exceed the original deign requirements in Table 5.5.1-1. During the analysis it was determined that the actual pump performance exceeds the original design developed head requirement. To avoid confusion between the original design minimum head

SEQUOYAH NUCLEAR PLANT
WORK REQUESTS (WR), WORK ORDERS (WO), & CLEARENCES (HO)
REQUIRING 50.59 EVALUATIONS

WR/WO/HO	DESCRIPTION	SAFETY ANALYSIS
	an equivalent component, a 10CFR50.59 evaluation was not required to support the WO work scope. However, a FSAR change was made for clarification of the RCP design parameters shown on FSAR Table 5.5.1-1. The RCP design parameters on FSAR Table 5.5.1-1 establish the original functional requirements to which the reactor coolant pumps were built.	requirement and the actual pump developed head, Table 5.5.1-1 is being revised to address both parameters. A revision to Table 5.5.1-1 to update the representative RCP design parameters to clarify the performance of the SQN RCPs does not impact any safety analyses, system operating characteristics or system performance, or component performance described or assumed in the FSAR. These values are not utilized as inputs into any safety analysis or system analysis for the RCS or its components. Therefore, the change to FSAR Table 5.5.1-1 does not involve an USQ.
WO95004417000	An evaluation was performed for the disablement of the annunciator associated with the Auxiliary Building Passive Sump (ABPS) level switches as a result of the failure of these switches to function. The ABPS switches to the flood detector local panel located in the 6.9KV shutdown board room alarm in the Main Control Room (MCR) if a high water level occurred in the ABPS.	A DCN is being processed to provide a replacement for the disabled level switches. An alternate method of ABPS level detection is evaluated by opening the crosstie valve between the ABPS and the Auxiliary Building Floor and Equipment Drain Sump (ABFEDS). By doing this, the ABFEDS high level alarm will provide a similar alarm function as the ABPS high level alarm, but at the MCR Board 1-M-5 instead of 1-M-15. Operations' annunciator response procedure 1-AR-M5-A will be revised as required to reflect use of the alternate ABFEDS switch for passive sump level indication. Operations procedure 0-SO-77-10 will be revised as required to allow the crosstie valve to remain open pending DCN completion.
WRC4120852/WO99024 14000	Work order 99-02414-000 was to troubleshoot and repair the Loop 1 Pressurizer spray line pressure control which would not fully close when its controller had no output demand to open the valve. This valve is an air-to-open valve. The not fully closed valve position was determined by 1) the main control room valve position indication and 2) spray flow to the pressurizer causing the pressurizer backup heater group 1C to be energized during normal plant operation. This evaluation was to support plant operation and valve troubleshooting by placing the controller in manual and allowing the air to be isolated from the valve until maintenance on the valve could be completed during the next outage. Isolation of air to the Loop	Air isolation to the Loop 1 Pressurizer spray line pressure control valve will not create or affect any design basis accidents evaluated in the SAR. Credible failure modes associated with this activity, air isolation to the PCV, considered thermal shock to the pressurizer spray nozzle and the load demand step decrease of ten percent. No new failure modes were created by this activity. No SAR evaluated accidents could be initiated by this activity. The previously analyzed RCS transient response to the 10% step load decrease from 100% power analytically demonstrated that the primary system pressure will not increase to the pressurizer power operated relief valve actuation setpoint of 2350 psia. No FSAR chapter 15 accident scenarios require

WR/WO/HO	DESCRIPTION	SAFETY ANALYSIS
	1 Pressurizer Spray line pressure control valve was performed by closing its essential air containment isolation valve. Air isolation only affected the Loop 1 pressurizer spray line pressure control valve pneumatic components.	the pressurizer sprays to mitigate events. Therefore, this activity will not increase challenges to safety systems assumed to function in the accident analysis such that safety system performance is degraded below the design basis. The redundant pressurizer spray line pressure control valve will supply required spray as necessary to address transients described in FSAR Section 5.2 such as spray line thermal shock and load reduction of ten percent. Therefore, implementation of this change does not introduce an USQ.
WRC346046/WO971186 7000	The containment upper compartment cooler temperature indicating controllers (TIC) are obsolete and spare parts cannot be obtained. Since the cooling requirements for upper containment are seasonally constant, the controllers can be operated in manual even with the obsolete equipment. This evaluation addresses the SAR change to the description of this cooling system to allow it to be operated in either automatic or manual.	The average upper containment temperature is monitored daily and maintained within the TS Limits. The temperature alarms nor the TSs are affected by this change. The Upper compartment cooler or the TICs perform no safety related or accident mitigation functions. No new failure modes, malfunctions, or accident initiators are introduced by this change. Therefore, this change does not involve an USQ.