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

November 17, 2005

10 CFR 50.59
10 CFR 72.48

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

Gentlemen:

In the Matter of	)	Docket Nos. 50-327
Tennessee Valley Authority	)	50-328

SEQUOYAH NUCLEAR PLANT - UNITS 1 AND 2 - 10 CFR 50.59, AND 10 CFR 72.48 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) and 10 CFR 72.48. The evaluations occurred since the last Unit 2 refueling outage.

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

Sincerely,

for

P. L. Pace
Manager, Site Licensing and
Industry Affairs

Enclosure

IE47
NMSS01

ENCLOSURE

SEQUOYAH NUCLEAR PLANT

10 CFR 50.59 AND 10 CFR 72.48

SUMMARY REPORT

SEQUOYAH NUCLEAR PLANT

CHANGES IN THE FACILITY FOR AMENDMENT - 19

DCN	DESCRIPTION	SAFETY ANALYSIS
D-20922-A	This change involves the construction of a cask storage pad for the Independent Spent Fuel Storage Installation (ISFSI). Major modifications implemented as part of this change include the construction of the cask storage and fabrication pad including lighting, drainage features, grounding, a nuisance fence and an access road. Additional interface work involving grading rework, equipment abandonment and potable water line rerouting was also performed as part of the change.	Construction of the access road from the Auxiliary Building to the ISFSI pad required installation of a thick layer of roller compacted concrete adjacent to existing tornado missile protection slabs at a grade above existing safety-related underground essential raw cooling water (ERCW) piping. The road excavation required temporarily removal of soil to a depth below the bottom of the missile protection slabs. As such, a portion of the soil that provides the barrier against low missile flight paths to the ERCW piping was removed. This condition was temporary for the time from soil excavation to the placement of the roller compacted concrete (which provides missile protection superior to the present soil barrier). During this interim configuration, 80 percent of the possible tornado missile flight paths to the ERCW piping were blocked by the remaining soil and concrete slab. The other 20 percent of the flight paths affected by soil removal were covered with 4.3 feet of soil. Protection against missiles larger than six inches is not diminished by this configuration. For smaller size missiles to damage the ERCW piping in this configuration, the missiles must maintain a non-tumbling flight path, strike within the 20 percent angle range and penetrate 4.3 feet of soil. While this is possible, such a missile will not damage two or more ERCW pipes. All active components of the ERCW supply are redundant and can tolerate a single failure in the short or long term. The functional capabilities of the ERCW system were not affected by the construction configuration between excavation and placement of the roller compacted concrete.
D-21247-A D-21248-A	This change involves the replacement of the original electromechanical controls for the main control room (MCR) and electric board room (EBR) air conditioning (A/C) condensing units with replacement digital controls. The MCR and EBR A/C condensing units are components of the MCR and EBR A/C subsystems. These subsystems are part of the Control Building Environmental Control System (CBECS) which also includes the MCR emergency pressurization system, the MCR emergency air cleanup	The CBECS functions to mitigate the radiological consequences of accidents to control building personnel. The CBECS utilizes the control room emergency pressurization system and the control room emergency air cleanup system to limit radiological exposure. These systems automatically actuate upon receipt of a control room isolation (CRI) signal generated manually, automatically from a safety injection signal, or upon detection of high temperature or high radiation levels in the Control Building fresh air intake duct. The MCR and EBR A/C condensing units are not directly used to mitigate the consequences of any UFSAR Chapter 15 event and their failure

DCN	DESCRIPTION	SAFETY ANALYSIS
	system, the battery room exhaust system, and miscellaneous ventilating systems. The normal function of the MCR and EBR A/C subsystems is to provide cooling to the designated areas of the Control Building to maintain design temperatures in accordance with environmental design requirements. The replacement digital control systems were designed specifically to replace the electromechanical control systems used on the existing A/C packages. The control system changes included replacement of temperature, pressure, and motor current sensor inputs with units compatible with the digital control unit. Where possible, the digital control unit outputs drive existing interface relays, motor starters, and solenoid valves. The digital controller for each train of A/C condensing units is powered from an independent Class 1E power supply. The digital controller interfaces with the same power source used in the previous installation. Bus loading for the new controls remains within previous design parameters with acceptable margin. There was no modification of the power system or wiring to the controllers. Therefore, there was no change in power system redundancy, independence, or separation. The digital controller provides for improved control and operational flexibility of the A/C condensing units. System reliability improvements are anticipated based on the ability to discriminate and compensate between sensor malfunctions (and other minor problems) and actual operational issues requiring the condensing units to be shutdown.	would not create a Chapter 15 initiating event. However, as a subsystem of the CBECS, the MCR and EBR A/C condensing units provide cooling to designated areas of the Control Building to maintain temperatures in accordance with environmental design requirements. The MCR and EBR A/C system condensing units consist of trained, 100 percent redundant units. The replacement digital controllers are robust and provide diagnostic information to personnel for assessing and correcting potential problems. In some cases, a single failure of a digital controller will result in the loss of a single A/C condensing unit. Because the redundant unit in each system is unaffected, there is no corresponding loss of safety function. This is similar to the existing electromechanical control scheme in that a single failure results in loss of a single condensing unit. Unlike the electromechanical controls which lack the ability to evaluate process parameters or restart an A/C condensing unit following a trip, the digital control system is designed to maximize system performance while minimizing operator intervention. It is therefore expected that the digital control system will provide more reliable operation of the A/C condensing units than the existing electromechanical control system. The digital controller represents a potential common cause failure attributable to software. Review of the software code, the software design process, operational history of similar units, and factory acceptance tests on the digital control system for each compressor supports the conclusion that the system control software does not increase the probability of a common cause failure. Replacing the existing electromechanical MCR and EBR A/C condensing unit controls with digital controllers does not increase the likelihood of equipment malfunctions, change equipment failure modes, impact fission product barriers, alter the frequency of previously analyzed events, or create an event of a type not previously analyzed.

SEQUOYAH NUCLEAR PLANT

CHANGES IN THE FACILITY FOR AMENDMENT - 19

DCN	DESCRIPTION	SAFETY ANALYSIS
D-21640-A D-21641-A	This change involves the deletion/abandonment of eight radiation monitors as part of the Radiation Monitoring System Optimization Project. The change also removes the automatic closure signal to the containment isolation valves for the reactor coolant drain tank lines and the reactor building floor and equipment drain sump lines based on signals from the deleted/abandoned radiation monitors. The change was performed in five separate stages. The first four stages removed the high radiation isolation signal from the affected containment isolation valves. Stage 5 removed/abandoned all eight radiation monitoring instrumentation loops. The radiation monitors affected by this change include the Reactor Coolant Drain Tank Monitors (2), the Reactor Building Floor and Equipment Drain Sump Monitors (2) and the Residual Heat Removal (RHR) Accident Line Monitors (4).	None of the deleted monitors are used to mitigate or recover from a UFSAR Chapter 15 accident. The radiation monitoring system at Sequoyah Nuclear Plant was designed early in plant construction. Some radiation monitors were included in the original design in response to industry events or were added in anticipation of regulatory issues. As a part of the Radiation Monitoring System Optimization Project, all radiation monitors ware reviewed to determine if they were a) redundant with other monitors and could be eliminated, b) the function was accomplished by other means such as health physic surveys, or c) the function was not needed. It was concluded that the affected monitors could be eliminated based on the following considerations. The Reactor Coolant Drain Tank Monitors and the Reactor Building Floor and Equipment Drain Sump Monitors automatically isolate the reactor coolant drain tank lines and the reactor building floor and equipment drain sump lines on detection of high radiation. These lines are also automatically isolated on a Phase A containment isolation signal independent of the signal from the radiation monitors. Resetting the Phase A signal does not allow the valves to automatically reopen. Thus the lines are protected from inadvertent opening after an accident. There is no regulatory requirement (NUREG-0737 or NUREG-0696) which requires these lines be isolated using a high radiation signal. Therefore these monitors have been eliminated. The RHR Accident Line Monitors are area type monitors located adjacent to the RHR lines. These monitors were added in anticipation of a regulatory requirement that was not promulgated. Therefore, these monitors have been eliminated. In summary, these radiation monitors were eliminated from the plant design because they provided redundant or unnecessary functions. Their deletion does not affect safe plant operation.

DCN	DESCRIPTION	SAFETY ANALYSIS
D-21854-A	This change involves modification of the emergency diesel generator (DG) air starting system. Each DG engine is provided with an air starting system using two air accumulator tanks. In the original design, one of the tanks is connected to the air start system while the other is isolated from the system in a standby mode. Should the DG fail to start using the air in the primary tank, the standby tank can be aligned to the air start system by operator action (i.e., manually opening locally mounted isolation valves). The tanks are maintained between 250-300 psig. Pressure control valves (PCVs) control pressure downstream of the tanks between 190-200 psig. During starts, the downstream pressure drops and is limited only by the back pressure in the starter set and associated piping. The large differential pressure across the PCVs when the air motors actuate occasionally causes the PCVs to malfunction. This causes the pressure in the aligned accumulator tank to fall below an acceptable level. (It should be noted that when a malfunction occurs, it occurs after a successful DG start. A malfunction has never caused a failure to start the DG.) To address this issue, this modification 1) changes the alignment of the accumulator tanks such that they are both normally aligned to the air start system in series, 2) removes the PCV between the air start motor and the first accumulator tank, 3) relocates the remaining PVC between the first accumulator tank and the second accumulator tank and 4) reduces the operational pressure in the first tank to 200 psig. This modification will lower the differential pressure across the PCV and allows both tanks to be connected to the starting air system. This increases the amount of available starting air and eliminates the operator action to manipulate the manual isolation valves.	The design function of the DG air start system is to maintain an air storage capacity sufficient to crank the engine five times without recharging. This modification enhances the capability of the air start system to meet this functional requirement based on the following considerations. 1. Although the total amount of stored energy in the accumulators will be reduced by reducing the pressure in one accumulator tank from 300 psig to 200 psig, the normal alignment of both air accumulator tanks to the air start motors will provide additional air capacity to meet the five-start requirement without operator action. 2. The operation/reliability of the air system PCV is improved by reducing the differential air pressure across the valve during air motor operation. Improved reliability of the PVC reduces the likelihood of an inadvertent blow down of the air accumulator tanks. 3. Removal of the PCV between the air start motor and the first accumulator tank simplifies the design and eliminates a potential failure device between the accumulator tank and the air start motor. As a result of this modification, a PCV will no longer have to actuate to pass air from an accumulator to a start motor for the first start of the DG engines. 4. The air start system remains single failure proof as each DG set continues to be equipped with a redundant starting air system to ensure that a single failure will not prevent start of a DG set. As in the previous design, low pressure alarms in the MCR will alert operators to a failed closed condition of the PCV. Given the considerations, the modification replaces a required manual operator action with an adequate and reliable automatic action, reduces the likelihood of equipment malfunctions and has no affect on equipment failure modes. The modification does not adversely affect safe operation of the plant.

DCN	DESCRIPTION	SAFETY ANALYSIS
E-21761-A	This engineering-only change documents the acceptability of the as-installed configuration of the moisture separators contained in the air cleanup subsystem of the Emergency Gas Treatment System (EGTS). As designed, the EGTS moisture separator assembly is composed of two functional elements. The first element is composed of louvers (or wave/bent plate demister blades) which are designed to remove large (50-1000μm) liquid droplets. The second element is composed of fiber-glass, non-woven fiber mat pads (i.e., demister pads) which remove the finer, sensible moisture droplets (1-100μm). During an inspection of the EGTS filter housings (where the moisture separators are installed), it was determined that the demister pads were not installed in the moisture separator assembly. Upon review, it was determined that the demister pads were not installed during original plant construction. This change documents the acceptability of the as-installed configuration to meet the functional requirements of the air cleanup system and revises the design basis documentation to eliminate the requirement for moisture separators in the EGTS air cleanup system.	The function of the EGTS moisture separators is to reduce moisture loading on the pre-filters and high efficiency particulate air (HEPA) filters installed in the air cleanup system downstream of the separators. The original design of the air cleanup system assumed that the EGTS filter housings would be located in a high relative humidity environment. Subsequent environmental condition calculations determined that the relative humidity of the air entering the EGTS air cleanup units is low (45%) during accident conditions. (NRC RG 1.52 defines low relative humidity as less than 70% relative humidity.) These calculations were previously used to establish that the EGTS duct heaters included in the original system design are not required to meet system functional requirements (refer to Sequoyah License Amendment Nos. 103 (Unit 1) and 92 (Unit 2) issued in response to Technical Specification Change Request Nos. TS 80 and TS 136). These calculations do not take credit for any moisture removal by the moisture separators. Based on this result, deletion of the moisture separator functional requirement does not aversely affect the operation of the EGTS system. The moisture separators are not needed to protect the downstream filters and charcoal absorbers due to the low relative humidity service conditions. The non-functional moisture separators will not prevent the EGTS air cleanup unit from performing the required safety functions of 1) keeping a negative pressure within the containment building annulus and 2) reducing the radioactive nuclides in the annulus air released to the environment within acceptable levels.

PROCEDURE	SUMMARY OF DESCRIPTION	SAFETY ANALYSIS
ES-1.3, Revision 12	Step 2 of Emergency Procedure No. ES-1.3, "Transfer to Residual Heat Removal Containment Sump" was revised to permit one train of containment spray to be removed from operation if both trains of containment spray are operating at the initiation of emergency core cooling system (ECCS) pump suction swapover from the refueling water storage tank (RWST) to the reactor building recirculation sump. This change reduces the flow rate out of the RWST and allows for increased operator action time to complete the realignment of the ECCS pump suction from the RWST to the reactor building recirculation sump. Following alignment of the containment spray pump suction to the reactor building recirculation sump, both containment spray pumps will be returned to service.	The function of the containment spray system is to reduce containment pressure following ice bed depletion for the long term containment integrity analysis described in the UFSAR. Containment pressure suppression is provided by the ice condenser until the ice mass is depleted for a large break loss of coolant accident (i.e., the design basis containment pressurization transient). Following ice bed depletion, the long term containment integrity analysis credits one train of containment spray flow for containment pressure suppression. The analysis requires that the ECCS and containment spray pump suction be aligned to the reactor building recirculation sump before the ice mass in the ice condenser is depleted to ensure continued pressure suppression. As such, termination of one train of containment spray flow during the swapover from RWST injection to containment sump recirculation is acceptable because 1) the ice condenser is providing containment pressure suppression during this interval and 2) the containment integrity analysis only assumes operation of one train of containment spray during the entire event. Additionally, the assumption of one train of ECCS and containment spray operation maximizes the drain down time of the RWST. It establishes the limiting margin between the completion of sump recirculation swapover and ice bed meltout. The termination of one train of containment spray flow during the injection to recirculation transition time will not affect the transition time completion to ice bed meltout margin established by the analysis. Based on these considerations, the procedure change does not adversely affect safe operation of the plant.

TACF	SUMMARY OF DESCRIPTION	SAFETY ANALYSIS
1-05-002-063	This change involves the temporary installation of a continuous vent line for the residual heat removal (RHR) discharge piping. The alteration involves the installation of a check valve and a valve station (which includes a pressure reducing valve with upstream and downstream isolation valves, a parallel by-pass valve and a parallel throttle valve). The check valve (which supports a piping classification transition) will be installed downstream of an existing RHR Train B emergency core cooling system (ECCS) high point vent valve (normally closed). The check valve will be connected between the existing vent valve and the skid mounted valve station with stainless steel tubing. Discharge from the valve station will be routed via a floor drain to the Reactor Building Floor and Equipment Drain Sump (RBFEDS). Two additional normally closed ECCS vent valves will be opened such that the RHR pump discharge piping will vent continuously through the valves, the temporary check valve and valve station into the RBFEDS. This temporary alteration is required to mitigate the effects of a leaking primary system check valve which requires operator action to periodically vent the RHR discharge piping. The continuous vent will reduce but not eliminate the need to manually depressurize the RHR discharge piping. The continuous vent will remain in place until the leaking check valve can be repaired/replaced during the next refueling outage.	The design of the temporary alteration is consistent with existing design requirements for interfacing with safety-related systems, structures, and components (SSC's). The design does not affect RHR system operation, it satisfies the normal and post accident limitations for drainage to the RBFEDS and meets all containment isolation requirements For RHR system operation, each RHR pump is credited for supplying a nominal flow 3000 gpm. The maximum RHR inventory lost to the RBFEDS through the temporary vent line (approximately 750 gpd) does not have a significant affect on system operation. In the event of a primary system loss-of-coolant accident, the temporary vent line would allow a portion of the RHR flow to be diverted from the active reactor building recirculation sump to the RBFEDS (i.e., outside the containment building polar crane wall). The portion of the ECCS flow diverted to the RBFEDS will not be available to the long-term ECCS recirculation inventory. A conservative calculation of the recirculation inventory lost through this line for 100 days of post-accident operation confirmed that there is sufficient active sump inventory to support ECCS operation (i.e., to meet pump NPSH and vortex safety limits) when the temporary vent line loss is considered. Changing the position of the ECCS high point vent valves from normally closed to normally open to allow the continuous vent was reviewed relative to the containment isolation function established for Containment Penetration No. X-20A. The evaluation concluded that the temporary change does not affect the containment isolation function for Containment Penetration X-20A. The classification and testing of the temporary materials/components which interface with the established containment isolation boundary were equal to or better than the classification and testing of the existing X-20A pressure boundary components. Based on these considerations, the temporary change does not adversely affect safe operation of the plant.

TACF	SUMMARY OF DESCRIPTION	SAFETY ANALYSIS
1-05-013-063	This change involves the temporary installation of a commercial grade, low volume, high pressure pump in the safety injection system to fill the Cold Leg Accumulators (CLAs). The temporary pump is air driven and operates at station air header pressure (70-100 psig). It consumes approximately 100 scfm while providing a maximum outlet pressure of 850 psig (relief valve setpoint). The pump is manually operated under local operator control. The temporary pump is located in the Auxiliary Building. The pump discharge ties into an existing manual valve in the accumulator fill/test line. An in-line check valve is also installed at the tie-in valve to prevent back flow into the temporary equipment. The temporary pump takes suction from an existing pressure gauge tap on the tube side of the Train A containment spray system (CSS) heat exchanger. An existing instrument isolation valve is used for suction supply isolation. The water source for the pump is the refueling water storage tank (RWST) via the Train A CSS heat exchanger. All connections and added valves are consistent with the permanent piping system design. Hydraulic modeling predicts a pump flow of 3.5 to a maximum of 5.4 gpm in this configuration. This temporary alteration is required to mitigate the effects of a leaking primary system check valve which requires operator action to periodically fill the Loop 3 CLA. The CLAs are normally filled by operation of the safety injection pumps. This temporary alteration will be used as an alternate CLA fill method in order to minimize the number of starts on the safety injection pumps. The pump will remain in place until the leaking check valve can be repaired/replaced during the next refueling outage.	The design of the temporary alteration is consistent with existing design requirements for interfacing with safety-related systems, structures, and components (SSC's). A flow path to the centrifugal charging pump (CCP) suction is prevented by existing check valves and normally closed manual valve alignments. Flow from the temporary pump is isolated from the safety injection system flow by containment isolation valves. Backflow from the safety injection system test header into the CSS, the residual heat removal (RHR) pump, safety injection (SI) pump or the temporary pump connection is prevented by an added inline check valve. The fill pressure is greater than CLA pressure but is much less than normal Reactor Coolant System (RCS) pressure. The CLA will not be overpressurized by the temporary pump because of the low capacity which provides sufficient time for operator action prior to actuation of a CLA relief valve. If pressure were to increase to the point of actuating a CLA relief valve, the relief valves have sufficient capacity to relieve any overpressure. The low pump flow rate also provides sufficient time to secure the pump should a leak develop or an equipment malfunction occur within the fill path. When the temporary pump is aligned for fill operation, administrative controls are in place which require operator action to ensure that the post accident sump inventory is not communicated to the Auxiliary Building via the attachment at the CSS heat exchanger. The operator action ensures that there are no changes to the consequences of any analyzed accident. The failure to perform the manual action does not result in an accident or new malfunction, and does not exceed the consequences already evaluated in the UFSAR.

UFSAR REVISION	SUMMARY OF DESCRIPTION	SAFETY ANALYSIS
Section 10.4.7	This change involves a revision to the Steam Generator Blowdown (SGBD) system automatic isolation logic described in Section 10.4.7 of the UFSAR. The current logic description indicates that the SGBD system automatically isolates when the auxiliary feedwater (AFW) pumps receive an automatic start signal. A design review of the as-installed isolation logic established that the SGBD isolation valves will not receive an automatic closure signal concurrent with an AFW automatic pump start when the AFW pumps are in operation (i.e., when the pumps are operating during startup or shutdown operations or during pump surveillance testing performed during full power operation). The SGDB isolation valves will receive an automatic isolation signal when 1) the AFW pumps receive an automatic start signal when they are not in operation and 2) upon receipt of a Phase A containment isolation signal. Review of the as-installed SGBD isolation logic determined the AFW system functional capabilities are not affected by the isolation logic change. The UFSAR (and other design basis documents) have been revised accordingly.	The AFW system safety function involves the supply feedwater to the steam generators to remove reactor coolant system (RCS) decay heat and other stored energy by steaming through the main steam safety valves. The AFW system is required to provide feedwater when the non-safety grade main feedwater (MFW) system is unavailable and the RCS is above the maximum pressure/temperature for initiation of residual heat removal (RHR) system operation. The specific safety functions and flowrate requirements of the AFW system vary with each design basis event (DBE). The DBEs which impose safety-related performance requirements on the AFW system include the loss of MFW transient, main feedline break (MFLB), main steamline break (MSLB), loss of all alternating current power (station blackout) and a small break loss of coolant accident (SB LOCA). The AFW system performs safety functions to both mitigate and recover from these transients. These transients are divided into two broad classes which include 1) primary system cooldown events which limit the maximum amount of AFW flow and 2) primary system heatup events which establish minimum AFW flow requirements. The MSLB is the limiting RCS cooldown event and establishes the maximum AFW flow limit (2250 gpm AFW flow delivered to a faulted loop until isolated). The loss of MFW and MFLB transients are the limiting RCS heatup events and establish the minimum AFW flow requirement (410 gpm to at least two steam generators one minute following the initiation of a low-low steam generator water level reactor trip signal). In addition to the DBE mitigation, the AFW system supports recovery from all of the DBEs listed above by providing sufficient flow to remove decay heat, stored energy and reactor coolant pump heat until such time as the RHR system can be placed in service. The requirements for performing this recovery function are identical to the minimum AFW requirements for RCS heatup transient mitigation. Based on a detailed evaluation of AFW system operation with the existing SGBD isolation logic, it was determined that all system functional requirements continue to be met. This conclusion is based on 1) the ability of the AFW system to provide adequate RCS cooling during normal startup and shutdown operations

UFSAR REVISION	SUMMARY OF DESCRIPTION	SAFETY ANALYSIS
		with minimum pump capacities and a bounding SGBD flow rate, 2) the availability of diverse SGBD flow isolation signals to maintain system operation consistent with safety analysis assumptions, and 3) the availability of additional AFW flow capacity from the turbine driven AFW pump which is not credited by the loss of normal feedwater transient analysis but is required for system operability. Based on these results, the isolation logic change does not adversely affect safe operation of the plant.
Section 15.2.10	This change involves a revision to Section 15.2.10 of the UFSAR to incorporate a revised analysis of the feedwater system malfunction transient. The revised analysis demonstrates that the reactor coolant system temperature-pressure response to the transient is well within the system design limits, confirms that the minimum calculated departure from nucleate boiling (DNB) ratio is above the established safety limit and confirms that the steam generator does not overfill and flood the main steam piping. The results meet all of the previously established transient acceptance criteria. The revised analysis differs from the existing analysis in that it uses an NRC-approved evaluation methodology developed by Framatome Advanced Nuclear Power (FANP). The FANP methodology establishes feedwater flow rates, core power and reactor coolant system temperature and pressure conditions using the RELAP5/MOD2-B&W computer code and the resultant DNB conditions using the LYNXT computer code. The current analysis uses a methodology developed by Westinghouse Electric Company which establishes the core power transient using the LOFTRAN computer code and is based on a bounding, maximum feedwater flow rate.	The revised analysis uses methodology and computer codes approved by NRC and is in compliance with the conditions and restrictions placed on their application. It does not represent a departure from an approved method of evaluation. As a result, the revised analysis was adopted as the analysis of record under the current licensing basis.

UFSAR REVISION	SUMMARY OF DESCRIPTION	SAFETY ANALYSIS
	The reanalysis was performed to make the methodology used for the feedwater malfunction analysis consistent with other transient analyses which use methodologies developed by FANP. FANP is the current supplier of reload fuel and reload analysis services for Sequoyah.	

ISFSI CHANGE	SUMMARY OF DESCRIPTION	SAFETY ANALYSIS
SQN-04-01	Temporarily removing the HI-STROM 100S, Version B, lower inlet vents gamma shields. The temporary removal assists the annual inspection and if needed, reapplication of a sealant between the overpack baseplate and ISFSI storage pad.	The evaluation concluded that temporary removal of the gamma shields results in no increase in off-site consequences. The site-specific calculations for the SQN ISFSI conservatively did not credit the use of inlet vent gamma shields in the model, therefore, the temporary removal does not affect the existing off-site consequence.