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November 18, 2008

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 previous submittal dated June 12, 2008. There were no 10 CFR 72.48 evaluation performed in this timeframe.

If you should have any questions, please contact me at (423) 843-7170.

Sincerely,

Original signed by

James D. Smith
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Enclosure

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I: License TS/ FSAR 21/ FSAR 21-5059 Summary report

ENCLOSURE

SEQUOYAH NUCLEAR PLANT

10 CFR 50.59 AND 10 CFR 72.48

SUMMARY REPORT

SEQUOYAH NUCLEAR PLANT

CHANGES IN THE FACILITY FOR AMENDMENT - 21

DCN	SUMMARY OF DESCRIPTION	SAFETY ANALYSIS
D-21831-A	This modification involves the removal of an original canister type containment electrical penetration device and replacement with a modular type penetration assembly for Containment Penetration No. X-128E on Sequoyah Unit 2. To support the interface with the new modular penetration, changes to the electrical cable connections and the fusing for the electrical loads passing through the penetration were also implemented.	The original canister type penetration was replaced to address an adverse trend of increased leakage through the penetration. The adverse trend resulted from equipment aging issues associated with the original penetration assembly. Replacement of the assembly eliminated the observed leakage and restored the penetration integrity. Evaluation of the replacement penetration assembly confirmed compliance with the original component design standards (i.e., ASME Section III - 1986 and IEEE 317-1983). Also, the electrical interfacing changes were evaluated against the existing design basis requirements and were found to be acceptable in terms of component and system qualification. Performance of the penetration replacement was limited to the core empty period of a refueling outage when containment integrity requirements support the in-process configuration.
E-22316-A	This modification established a revised setpoint tolerance for relief valves which provide piping system overpressure protection for a portion of 1) the centrifugal charging pump (high head) suction piping and 2) the safety injection pump (intermediate head) suction piping. The original tolerance for the spring loaded relief valves was +/- 3 percent of the nominal relief setpoint for both as-found and as-left test conditions. This change established a revised setpoint tolerance of +3 percent,-5 percent for as-found conditions. The original +/-3 percent setpoint tolerance was not changed for as-left conditions.	This change was implemented to address the tendency of the nominal relief valve setpoint to drift downward during the established component test interval. The nominal setpoint for each of the affected relief valves is 220 psig. Each valve functions to relieve system pressure in the event the pump suction isolation valves experience small amounts of in-leakage from the connected higher pressure piping systems when closed. The design capacity of the relief valves is approximately 20 to 25 gallons per minute (gpm). Evaluation of this change focused on the effects of the lower valve actuation pressure afforded by the -5percent setpoint tolerance. The evaluation concluded that 1) sufficient margins exist between the range of normal operating pressures in the affected piping and the revised relief valve lower setpoint to prevent spurious operation of the relief valves during normal operation and 2) the valve reset (blowdown) pressure following actuation (the valves reseat at approximately 10 percent of the nominal lift pressure) has no effect on system operation of functional capabilities. The change did not affect the overpressure protection function of the relief valves. Since the change has no effect on the system or component functional capabilities and is consistent with the component Code of Record, the change does not adversely affect the safe operation of the plant.

PROCEDURE	SUMMARY OF DESCRIPTION	SAFETY ANALYSIS
0-SI-OPS-068-137.0, R22	This change involves a revision to the primary system water inventory surveillance procedure. The revision involves a change to two of the three equations used to determine total leakage from the reactor coolant system (RCS). Calculation of the total RCS leakage value under this procedure involves the primary system makeup rate, volume control tank (VCT) leakage rate, pressurizer leakage rate, and RCS temperature correction. The changes involve normalization of two of the inputs for total leakage to an established reference temperature. The first change alters the RCS temperature correction factor such that the volumetric leak rate is calculated using the density of water at 70 degrees F and atmospheric pressure (instead of using the average density of water based on RCS normal operating temperature and pressure). The second change alters the pressurizer leakage rate equation to use the actual pressurizer pressure instead of the primary system design pressure (2235 psig). These changes allow for a more direct comparison of the leak rates that make up the total system leakage. The changes affect the unidentified leakage rate values as total RCS leakage is one of the two components used to quantify unidentified leakage. The changes do not affect the identified RCS leakage calculations.	This procedure requires the collection of measured RCS data and contains the data reduction requirements to quantify RCS leakage in accordance with plant Technical Specification Surveillance Requirement No. 4.4.6.2.1. It is used as a surveillance tool to quantify RCS leakage and to backup the leakage detection systems located inside of the containment building polar crane wall. This procedure is not relied upon to detect Reactor Coolant Pressure Boundary (RCPB) leakage of 1 gpm in 1 hour or less, as specified in Regulatory Guide 1.45. That function is provided by the leakage detection systems monitoring RCPB external leakage located inside of the polar crane wall which include containment humidity, charging pump operation and excessive makeup volume detection, sump level detectors, and radiation monitors. The surveillance procedure changes are designed to reduce the standard deviation of the primary inventory leakage calculations. The changes are based on the best industry practices and are consistent with the recommendations of the Pressurized Water Owners Group (PWROG) as documented in Topical Report No. WCAP-16423-NP, Revision 00. The procedure revisions improve the accuracy of the primary system leakage surveillance calculations and do not negatively impact nuclear safety.
0-SO-202-1, R12 (OTOC)	This activity involves a one time only change (OTOC) to the system operating instruction for the plant 6.9kV start buses. The procedure contains instructions for 1) energizing the start buses from available sources, 2) taking a start bus out of service for maintenance, modification or testing, 3) transferring power supplies to the normal start buses and 4) transferring power supplies to the DC	Evaluation of the one time only start bus alignment involved 1) review of the modified loading on CSST B, 2) review of power source independence and 3) review of the postulated failure of one of the active CSSTs. The CSST B load review confirmed that several large unit board motor loads aligned to Start Bus 2A would be removed from service and administratively controlled (locked out) during the modified alignment.

PROCEDURE	SUMMARY OF DESCRIPTION	SAFETY ANALYSIS
	control bus on the start buses. Under normal conditions, Start Bus 1A and Start Bus 2A are fed from common service station transformer (CSST) A and Start Bus 1B and Start Bus 2B are fed from CSST C. The CSST design is such that a loss of a CSST (either A or C) will generate a fast transfer to CSST B. Breakers are electrically interlocked such that 1) only one start bus is aligned to each CSST B winding and 2) only one start bus power train (either the A or B) is aligned to CSST B at a time. These interlock features prevent overloading of the windings on CSST B for normal operating loads. To minimize the risk to Unit 1 operation during the CSST C maintenance outage which was conducted during the Unit 2, Cycle 15 refueling outage (Spring 2008), Start Bus 1B was aligned to CSST B. In support of planned maintenance activities, a transfer of Start Bus 2A from its normal feeder breaker to its alternate feeder breaker was also required. This transfer placed Start Bus 2A on the "Y" winding of CSST B at the same time the 1B Start Bus was aligned to the same winding. This one time only procedure change permitted the interlock in the control circuit of the alternate feeder breaker for Start Bus 2A to be bypassed to allow CSST B to carry both Start Bus 2A and Start Bus 1B on the "Y" winding during the maintenance activity. This configuration also prevented the automatic or manual transfer of Start Bus 1A and Start Bus 2B to the "X" winding of CSST B.	Review of the remaining loads applied to the CSST B in this configuration for both normal and accident operation confirmed that they were within the capacity of the CSST B winding such that an overload condition would not occur. As a result of the one time procedure change, Start Bus 1B and 2A were aligned to the "Y" winding of CSST B while the 2B start bus was aligned to the "X" winding of CSST C and Start Bus 1A was aligned to the "X" winding of CSST A. In this configuration, the 1A and 1B start buses remain powered from separate CSST transformers as well as the 2A and 2B start buses. This configuration maintained independent power sources to the trained start buses under expected operating conditions. If failure of the either CSST A or CSST B were postulated to occur such that off-site power is lost to one of the Unit 1 power trains, off-site power can be restored to the affected power train by manual operator alignment of the 6.9kV shutdown board normal/alternate feeder breakers. Based on these results, the loading on CSST B remains within the functional capabilities of the component. One immediate source of off-site power remains available for the 1A and the 1B start buses as well as one delayed source (based on manual operator action if failure of the credited CSSTs is postulated) in accordance with the requirements of IEEE-308 and General Design Criterion (GDC)-17. As such, this one time only CSST alignment change had no effect on nuclear safety.

UFSAR	SUMMARY OF DESCRIPTION	SAFETY ANALYSIS
Section 2.4.12	This activity involves a revision to the UFSAR which evaluates the dilution and disbursement effects of radiological effluents released to the surface waters in proximity to the plant. The revision is required to address changes in river flow control operations associated with a downstream dam which result in a reduction in the river flow rate assumed in the original evaluation.	Section 2.4.12 of the UFSAR evaluates the ability of the surface waters near the plant to dilute and disburse a number of postulated accidental radioactive liquid releases. The evaluation confirms that the level of activity at the first downstream surface water intake will be below 10CFR20 limits. The evaluation is performed in accordance with the requirements of 10CFR50, Appendix I. In this evaluation, the river flow is specified as the flow which is equaled or exceeded approximately 50 percent of the time. Based on recent changes to the operational practice for flow through a downstream dam, the average river flow consistent with this criterion has been reduced from 29,000 cubic feet per second (cfs) to 27,474 cfs. The reduction in the credited river flow was incorporated into the effluent disbursement and dilution analysis. Revised dilution values were calculated using the existing evaluation methodology and model. The calculation established the revised concentration for a continuous plane source release as 3.4E-11 mCi/gm. There was no change in the instantaneous plane source release concentration. The revised continuous plane source release concentration value remains well below the acceptable limit of 1.0E-9 mCi/gm for a liquid effluent release of Iodine-131.
Section 9.4.2.2.6	This activity involves a revision of the environmental conditions described for the shutdown transformer rooms. The current UFSAR indicates that the ventilation system for these rooms is designed to maintain a minimum room temperature of 60ºF. This change revises the description to indicate that the ventilation system is designed to maintain the room temperature within a range of 15°F to 97°F consistent with ventilation equipment functional capabilities and services equipment requirements.	Section 9.4.2.2.6 of the UFSAR contains a general description of the shutdown transformer room ventilation system. The environmental design criteria for these rooms conservatively establish the minimum room temperature as 15 degrees F. A review plant records since initial operation indicate that the lowest recorded temperature within these rooms is 24 degrees F. The subject rooms contain the 480V shutdown transformers. Review of the transformer design indicates that the limiting low temperature consideration is the transformer oil. The pour point of the transformer oil is established as -50 degrees F and represents the temperature at which the oil will not flow. There is no concern with moisture entering the transformer since there is a positive pressure nitrogen blanket over the transformer. An oil temperature of 15 degrees F has been established to be acceptable based on a pour temperature limit of

UFSAR	SUMMARY OF DESCRIPTION	SAFETY ANALYSIS
		-50 degrees F. Since the minimum 15 degrees F design basis room temperature is conservative with respect to actual conditions and the functional capabilities for the transformers are maintained with the 15 degrees F minimum temperature limit, the proposed change to the UFSAR is consistent with the established design basis. The change to the UFSAR represents a correction to be consistent with the design and has no effect on nuclear safety
Section 15.3.1	The emergency core cooling system (ECCS) pump performance capabilities assumed in the plant accident and transient analyses have been revised to 1) reduce the minimum developed head values for the charging (high head) and safety injection (intermediate head) pumps by 5 percent and 2) increase the maximum charging pump developed head values for pump flows which exceed 250 gpm. These changes have been previously made to the realistic large break loss-of-coolant accident (LOCA) analyses documented in Sequoyah Technical Specification Change Requests TS-07-04 and TS-08-01. The large break LOCA analysis was the limiting design basis transient for minimum ECCS pump performance. Application of the realistic analysis methodology to the large break LOCA transient demonstrated acceptable results with the revised ECCS pump performance characteristics. This activity extends the applicability of the revised performance assumptions to the balance of the plant transient and accident analyses.	The ECCS pumps are designed to provide emergency core cooling and reactivity control in the event of high energy piping breaks or other transients which involve primary system inventory losses or reactivity insertions. Evaluation of the minimum pump performance assumption changes resulted in a complete re-analysis of the small break LOCA transient. For the revised ECCS pump performance characteristics, the calculated peak fuel cladding increased from the previous value of 1162 degrees F to 1403 degrees F. This result remains well below the established analysis acceptance criteria of 2200 degrees F. The re-analysis also confirmed that the revised ECCS performance assumptions are sufficient to demonstrate compliance with the balance of the small break LOCA analysis acceptance criteria. The evaluation of the balance of the plant transients and accidents concluded the current analyses of record are either 1) not affected by the changes, 2) bounded by existing conservatisms built into the analysis or 3) bounded by crediting the full capability of the ECCS which was conservatively minimized in the analysis of record through the use of simplifying assumptions. Based on this result, the existing analyses were established to be conservative and bounding for the revised ECCS performance assumptions.

Work Order	SUMMARY OF DESCRIPTION	SAFETY ANALYSIS
99-007763-000	Under this work order, one of the essential raw cooling water (ERCW) pumping station traveling water screens was dissembled and replaced. The disassembly operation consisted of raising the entire screen assembly and removing one section at a time as required for replacement. Re-assembly was accomplished by reversing the process. The lifts were performed by a crane supported on a barge parked adjacent to the ERCW pumping station. The traveling screen sections were placed on the barge and were moved to and from a staging area. The following administrative controls were applicable to the work order. 1. All tornado missile protection roof panels remained in place during the load lifts except for those permitted to be removed by prior analysis. 2. Load lifts were not permitted during a tornado watch, warning, or any other periods of high winds. 3. The barge was not permitted to be positioned near the ERCW pumping station when work was not in-progress or during any flood conditions. 4. The barge and tug boat were not anchored during the lifts. The stationary position was better maintained by using the vessel motors and provided the ability to move the barge/tug quickly.	Evaluation of this activity included review of the actual lifting operation as well as the maneuvering of surface craft near the ERCW pumping station. The evaluation concluded: 1. The screen lifting operation placed loads in excess of 2100 lbs above safety-related equipment. As such, lifting of the traveling screens was performed in accordance with all the requirements of NUREG-0612 for a heavy load lift. 2. Design and operation of the barge and tug boat were consistent with applicable industry standards. Controls and restrictions placed on the maneuvering of the craft are conservative and effective in preventing impact with the ERCW pumping station. 3. Blocking flow to the ERCW intakes due to the postulated sinking of the barge/tug has been evaluated and is not credible. A trench exists immediately in front of the intakes. The width of the barge/tug is greater than the trench and the slope of the trench walls is steep such that it is not possible for the submerged barge to block the intakes. As such, nuclear safety was not affected by this work activity.