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June 12, 2007

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 November 17, 2005. 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

Glenn W. Morris
Manager, Site Licensing and
Industry Affairs

Enclosure

ENCLOSURE

SEQUOYAH NUCLEAR PLANT

10 CFR 50.59 AND 10 CFR 72.48

SUMMARY REPORT

SEQUOYAH NUCLEAR PLANT

CHANGES IN THE FACILITY FOR AMENDMENT - 20

DCN	SUMMARY OF DESCRIPTION	SAFETY ANALYSIS
D-21837-A	This modification involves replacement of the turbine building sump radiation monitor. This instrument does not provide a nuclear safety function. It is used to monitor the turbine building sump liquid effluent consistent with the requirements of NRC Regulatory Guide 1.21. The original monitor drew an effluent sample directly from the sump discharge. Both radiation and sample line flow were monitored. The replacement monitor is a non-intrusive device attached to the sump discharge piping. Use of a sample line is not required. Sample line flow monitoring has been eliminated.	The sample line associated with the original monitor had a tendency to become clogged with particulate from the turbine building sump effluent. This resulted in reduced or low sample line flow, low flow setpoint drift and frequent cleaning which adversely affected component reliability. Replacement of the monitor with a non-intrusive device eliminated the equipment reliability issues. The turbine building sump effluent is not radioactive unless there is or has been a primary to secondary system leak. The effluent is monitored to indicate the presence of elevated activity levels in the sump. The monitor is set to alarm if the activity in the discharge piping is larger than expected or has the potential to exceed 10CFR20 limits after dilution. The monitor's setpoint is calculated in accordance with the plant Off-Site Dose Calculation Manual (ODCM). As a non-intrusive component, the replacement monitor is less sensitive than the original based on detection through the wall of the discharge piping. While the replacement monitor will have a higher minimum detectable activity threshold, evaluation of the replacement monitor range has established the adequacy of the device to detect activity over the range required to meet the existing ODCM requirements.
E-22012-A	This change involves a modification to the reload fuel assembly top nozzle hold-down springs to reduce the hold-down force acting on the fuel assembly by approximately 200 lbs (or 20 percent). This change was implemented on new fuel assemblies installed beginning with the Unit 1, Cycle 14 refueling outage (Spring 2006).	This modification was implemented to reduce excessive hold-down forces imposed on the original fuel assembly design which contribute to fuel assembly dimensional distortion during service. A statistical methodology approved by NRC in September 2005 (see Topical Report No. BAW-10243P-A) was used to verify the modified hold-down springs maintain acceptable margin to fuel assembly lower core plate liftoff under all service conditions. Application of the methodology to the modified top nozzle confirmed that the revised spring forces are adequate to meet the existing fuel assembly structural and functional requirements under all service conditions.

PROCEDURE	SUMMARY OF DESCRIPTION	SAFETY ANALYSIS
ES-1.1, Revision 09	Step 2 of ES-1.1, “Safety Injection Termination” was revised to require termination of one train of containment spray system (CSS) flow if both trains of the CSS are running following a steam line break. The reduction in the number of operating CSS trains from two to one occurs prior to manual actions to terminate emergency core cooling system (ECCS) injection flow. This action reduces the refueling water storage tank (RWST) drain down rate and increases the amount of time to ECCS swap-over from injection mode to recirculation mode. The delay of the required manual operator swap-over actions allows focus on ECCS termination actions required to prevent water solid conditions in the primary system.	The function of the CSS 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. Following ice bed depletion (which occurs after swap-over from RWST injection to containment sump recirculation) the long term containment integrity analysis credits one train of containment spray flow for containment pressure suppression. As such, termination of one train of containment spray flow prior to swap-over 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 limiting event (i.e., large break loss of coolant accident). The proposed procedure revision only affects actions taken when both trains of the CSS are operating following a secondary system break inside containment. Subsequent failure of the operating containment spray train requires manual operator action to restart the available train. A secondary line break is less limiting than the design basis large break loss of coolant accident. The total mass and energy release from a secondary break (equal to the blowdown of a single steam generator) is not sufficient to cause ice bed melt-out. As long as the ice remains in the ice condenser, a reduction in the containment spray flow rate (or a total loss of containment spray) does not result in an increase in containment pressure. Given that the proposed change only affects containment spray system operation following a secondary system pipe break, the procedure change does not adversely affect safe operation of the plant.

PROCEDURE	SUMMARY OF DESCRIPTION	SAFETY ANALYSIS
0-SO-250-5, Revision 23	The system operating procedure for the plant 250 Volt DC Power System was revised to include a mechanical "Make-before-Break" (MBB) manual transfer of the normal and alternate supply breakers for the turbine building power distribution boards. The standard design for this board is a "Break-before-Make" transfer such that isolation of the batteries is maintained. The mechanical MBB transfer is needed to transfer supply power without interrupting control power to the 6.9 kV unit board protective relays.	During the MBB transfer, independence of the 250 Vdc battery systems is lost as both 250Vdc turbine building distribution boards are supplied by a single 250Vdc battery system. (The equivalent of one battery with twice the capacity is supplying all the required loads). The 250 Vdc boards will regain independence once the transfer is complete (i.e., the batteries will no longer be in parallel). While the 250 Vdc systems are momentarily tied together for the MBB transfer, both chargers and both batteries will remain in service supplying their respective loads (and Technical Specification Limiting Condition for Operation (LCO) No. 3.8.1.1 and Technical Requirements Manual LCO No. 3.8.3.1 be in effect during the short duration of the activity). The voltage of the battery chargers will be adjusted before the 250 Vdc systems are tied together to induce load sharing at the 250 Vdc turbine building distribution boards. Either charger is adequate to carry the load normally seen on both battery boards. On a loss of offsite power, the proximity of the batteries to their associated battery board is adequate to ensure one battery does not assume supply of all the loads on the second battery. As such, a controlled MBB manual transfer of the 250 Vdc turbine building power distribution boards does not reduce nuclear safety.
EA-67-1, Revision 05 and 1-SI-OPS-000-002.0, Revision 82	Changes have been made to the plant shift surveillance procedure and Essential Raw Cooling Water (ERCW) system emergency abnormal operating procedure to allow personnel to enter the auxiliary building during normal plant operations (and following the design basis loss-of-coolant accident) to stroke the ERCW isolation valves to the following Unit 1 engineered safety features (ESF) coolers. 1. Residual Heat Removal Pump Room Cooler 1A 2. Containment Spray Pump Room Cooler 1A 3. Centrifugal Charging Pump Room Cooler 1A 4. Centrifugal Charging Pump Oil Cooler 1A	The compensatory measures established by these procedure revisions do not require any temporary modifications or equipment substitutions. All components are operated as designed and do not alter system design requirements or functional characteristics/capabilities. The compensatory measures were evaluated in accordance with ANSI/ANS-58.8-94, "Time Response Design Criteria for Safety Related Operator Actions" and were found to be acceptable. A mission dose analysis was performed in accordance with 10CFR50, Appendix A (General Design Criteria 19) which confirmed the ability to perform the required operator actions under loss-of-coolant accident conditions.

PROCEDURE	SUMMARY OF DESCRIPTION	SAFETY ANALYSIS
	The ERCW system has experienced flow reductions from debris collecting on valve seats, tube sheets and other places. The procedure revisions represent compensatory measures to support system operation until the condition is resolved. The compensatory measures involve the actuation (i.e., stroking) of ERCW isolation valves in the cooler supply lines to dislodge debris and restore adequate ERCW system flow to the coolers. The need for operator action is based on readings from local differential pressure measurement instrumentation.	Based on these considerations, the procedure change does not adversely affect safe operation of the plant.