November 15, 2002 10 CFR 50.71(e)

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 (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 November 14, 2000 to May 20, 2002.

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

Sincerely,

Original signed by

Pedro Salas
Licensing and Industry Affairs Manager

Enclosure

ENCLOSURE

SEQUOYAH NUCLEAR PLANT

10 CFR 50.59 SUMMARY REPORT

SEQUOYAH NUCLEAR PLANT

CHANGES IN THE FACILITY FOR AMENDMENT - 17

DCN	DESCRIPTION	SAFETY ANALYSIS
D20035-A	Numerous reach rod remote valve operator problems exist and cannot be resolved via preventive maintenance (PM) or work orders (WOs). Eight reach rods were identified by the plant (Operations and Systems Engineering) as the highest priority that need to be fixed to operate per design. Problem Evaluation Report (PER) (SQN940106PER), initiated in 1994, concluded that position indicators are unreliable. Approximately 43 Stow Reach Rod Position Indicators continue to be unreliable; therefore, the position indication hardware is being removed. To increase compression on the clutch spring, approximately 50 Roto-Hammer remote operators will require the addition of spacers/washers in the clutch pack to eliminate unwarranted slippage. These changes will improve the functionality and reliability of the remote operator so the remote operator handwheel will function in the same manner as the valve handwheel. The position indicators will not be used to verify the valve position; however, it will remain in place while personnel verifies the valve is in motion when the handwheel is operated. Tagging information such as valve manufacturer, valve size, and number of turns to open/close will be added on the remote operator handwheel for valves 2 inches in diameter and less. Additional personnel will be necessary to prevent over torqueing the 2-inch or less diameter size valves. As part of design change notice (DCN) D20035A, 14 remote operators will be deleted; however, the maximum dose rate at the handwheel is less than 100 mrem/hr. Therefore, this change does not vary from the requirements of final safety analysis report (FSAR) Section 12.1.2. Two of the fourteen remote operators (i.e., valves 1-062-0653 and -0671) subject for deletion are shown on FSAR Figure 9.3.4-1; and therefore, will be revised to delete the remote operators for these valves. Additionally, FSAR Figures 9.3.4-2 and 9.3.4-3 will also be revised to document a existing remote operator at valves 1-062-0922, 0-62-0959, 1-062-0971, and 2-062-0971 and to delete the remote operator that is shown in error for valves 1-062-0918, 1-062-0948, and 2-062-0948.	Based on the results of this evaluation, it was concluded the scope of DCN D20035A will not invalidate any assumption in FSAR Chapter 15 accident analyses, will not challenge any technical specification and (TS) will not introduce new failures or design basis accidents. Therefore, implementing these modifications will not result in a unreviewed safety question (USQ).
D20071-A	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,	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

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	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 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.	support the identification and mitigation of accidents as well as system control functions during normal plant operations. The function of the inverters are 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 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 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.
D20225A	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 (MCR). All inputs on recorders being replaced will be included as inputs on the new recorders. The exception is the 1-RR-90-1A and B recorder where the inputs from monitors other than the auxiliary building (AB) are being placed on the plant computer instead.	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 MCR 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 offsite dose calculation manual (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

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	This modification also split the "Instrument Malfunction" and "Hi Rad" alarms for the non AB monitors to a separate alarm. All inputs on removed recorders are on the plant computer or trending information is not required.	involve an safety analysis report (SAR) change. This change, therefore, did not involve an USQ.
D20236-A	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-inch 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.
D20291-A	Valves 1-TCV-024-73 and 1-TCV-24-74 (temperature control valves for Unit 1 turbine generator air side and hydrogen side seal oil heat exchangers) were not properly controlling the flow of raw cooling water (RCW) through the heat exchangers. Valves, when placed in auto, were cycling excessively, due to the valves being oversized. This created oscillations in the seal oil temperature, which could result in seal rubbing, leaking, or other damage. Operations had to manually utilize the bypass lines and/or the manual isolation valves downstream of the TCVs to control/throttle the RCW flow through the heat exchangers. The scope of this design change (DCN D-20291) replaced the existing 2-inch bypass valves 1-VLV-024-0548 and 1-VLV-024-0556 and associated piping with stainless steel valves and piping. Smaller 1-inch valves and piping will be installed to bypass 1-VLV-024-0548 and 1-VLV-024-0556 in order to allow more precise control of flow when manual bypass is used. This modification met the design, material, and construction standards applicable to the plant. No direct or indirect adverse impacts on the operation or function of any quality related, safety related, or safe shutdown systems, structures, or components were identified.	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 1 turbine generator air side and hydrogen side seal oil heat exchanger. 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 1 turbine generator air side and hydrogen side seal oil heat exchanger are located in the turbine building, 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, this modification will not affect any design basis accidents or anticipated operational transients. For these reasons this activity does not constitute

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	This modification impacted FSAR figure 9.2.7-2 by adding a second bypass line around both raw cooling water temperature control bypass valves (1-VLV-024-0548 and -0556); therefore, a FSAR change was required. No other required changes to the FSAR were identified. TS Section 3/4.7.1 "Turbine Cycle" was reviewed. There are no TSs related to RCW temperature control on the turbine seal oil skid.	a USQ.
D20393-A	DCN D20393 replaced the obsolete switching valves and check valves on the two auxiliary control air (ACA) dyers A-A (0-DRYA-32-001A) and B-B (0-DRYA-32-002-B). This DCN also replaced the electrical on/off switch mounted adjacent to each dryer which allows each dryer to be turned off when not needed. A manual isolation valve was also installed on each dryer. Each of the dyers contain two drying towers, and one tower is drying the compressed air while the second tower is being regenerated (dried) with dry purge air. The towers "switchover" from drying to regenerating approximately every 5 minutes. The new isolation valves on the supply line for each dryer will allow bypass of the dryers and after filters without affecting availability of the ACA compressors. Until this modification, there were no means of isolating an ACA dryer from its associated ACA compressor, meaning that an ACA compressor must be taken out of service (an limiting condition for operation [LCO] condition) for any repair or maintenance on the associated dryer or after filter that requires opening the dryer or afterfilter pressure boundary, including the routine aligned to the ACA header dryer or after filter or maintenance. The new switching valves/actuators and check valves are equivalent in fit, form, and function to the original components and therefore, introduce no new failure modes. The addition of an electrical switch to each dryer adds a new potential failure mode in that the switch may fail open or close. The new switch will be procured as a Class 1E component consistent with the valve actuators. However, as the dryers are not Class 1E components and are not relied upon to electrically function post accident, this does not constitute a failure mode which could result in an accident or adversely impact the ability to mitigate an accident.	The ACA system is discussed in FSAR Sections 1.2, 3.5, 7.7, 7.3, and 9.3. Section 9.3 including FSAR Figure 9.3.1-1 required a revision by this DCN to reflect addition of the manual isolation valve in each ACA dryer. The FSAR Chapter 15 accidents were reviewed, and no accidents were determined to potentially be affected by this modification. The ACA dryers are not credited in the FSAR as being functional for shutdown of the plant. They only require a safety-related air source be available. This is accomplished with the safety related ACA compressors and safety-related air accumulators. This modification does not reduce the capability of the ACA system to provide air to the safety-related end devices and will not result in a USQ.

DCN	DESCRIPTION	SAFETY ANALYSIS
D20421-A	DCN D20421 provided the installation of a tail pipe on vent-overflow valve 1-FCV-70-66 to prevent water spray from affecting adjacent components in the event of tank overflow, and the installation of new level transmitters and rework of the sense lines will preclude the lose of level indication in the event of tank overflow. The modification raised the surge tank level switch setpoints for low level annunciation, and start and stop of the auto makeup, provided additional margin above the internal baffle, and component cooling water system (CCS). Design criteria SQN-DC-V-13.9.9 was revised to reflect the new level switch elevations. Thus, this design change will improve the ability of the affected components to perform their design basis functions. The CCS surge tank level instrument loops are non-1E components, do not perform a TS compliance function, and are not safety related. However, the power for the loops are fed from 1E safety-related power boards, 120-V AC vital instrument power board. The breakers for these instrument loops are listed on the "Non-1E Loads Powered from Class 1E Power Systems" drawings, and thus these breakers will have the extra testing and maintenance required for Class 1E isolation breakers. The replacement of the CCS Surge Tank level transmitters will not adversely affect the electrical boards supplying power to the instruments, or any other components within the loops. The new Rosemount transmitters have essentially the same power demands as the GEMAC transmitters being replaced, operating on 52.5-V DC with an output range of 10-50mA. The existing GEMAC power supplies will be reused for supplying dedicated power to the replacement transmitters.	SAR Figure 9.2.1-1 will be revised to reflect replacement of the CCS surge tank lower tap root valves. The SAR Figure currently depicts globe-type valves, and these valves will be replaced with ball-type valves. No other change to the SAR is required. These design changes do not affect any system descriptions as presented in the SAR. The function, operation, and performance of the CCS surge tank, vent/overflow valve, and level loops as described in the SAR are not altered, and this change does not affect any operational or functional descriptions in any text, tables or graphs in the SAR. Thus, the only impact to the SAR is to Figure 9.2.1-1, which will require a revision to reflect a change to the lower tap root valves. Based on the results of the evaluation performed, it is concluded that the replacement Unit 1 CCS surge tank level transmitters, rework of the sense lines, revision of the level switch setpoints, and the installation of the tail pipe on vent valves 1 and 2-FCV-70-66, 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 change. Therefore, this design change does not result in an USQ.
D20596A	DCN D20596A provided modifications to the condenser vacuum exhaust vent radiation monitors 1,2-RE-90-404 (including 404A, 404B, and RM-405) and instrument loops 1,2-RE-90-255and -256. The function of the condenser vacuum exhaust vent monitor provides gross measurement of effluent releases out the condenser vacuum exhaust vent stack. The condenser vacuum exhaust monitor is required by Regulatory Guide (Reg. Guide) 1.97 and is a post accident monitoring (PAM) Category 2 variable. This variable is required for the ODCM which follows the guidance of Reg. Guide 1.21 (only the 99 and 119 monitors are required for the ODCM). This modification replaced the existing 404 loop functions with the existing	FSAR Chapter 11 was revised to delete the text and table information on the 404 loops and to update the information on the 255 and 256 loops. The PAM required range in Table 7.5-2 is still covered by this modification; and therefore, no change to this table was required. The FSAR Chapter 15 accidents associated with the condenser vacuum exhaust features are the steam generator tube rupture and a main steam line break. These accidents do not take credit for isolation by the radiation monitors. These loops are not required to initiate any function to mitigate a design basis accident or transient as regulatory commitment nor are they

DCN	DESCRIPTION	SAFETY ANALYSIS
	255 and 256 loops. The 404 loops will then be abandoned in place. The 255 is a mid range and the 256 is a high-range radiation detector. The 255 and 256 loops are an area type detector in close proximity with the exhaust pipe while the 404 loops used a piped sample. The 255 and 256 loops provide adequate range and overlap (between the low [99/119] and the mid [255] and high range [256]) to satisfy Reg. Guide 1.97 requirements for this variable.	designated as TS functions. The monitors may be subject to damage from a turbine-generated missile (reference FSAR 10.2) due to their location in the turbine building. This is not a change from the condition of the 404 monitors which are also located in the same physical location in the turbine building. DCN D20596A does not affect any SAR evaluations (accident analysis or equipment malfunction failures) previously performed. No new accidents or equipment malfunction failures are created. TSs are not affected. The condenser vacuum exhaust radiation monitor may be disabled by a turbine-generated missile; however, the system it monitors would no longer be in service after this event so there is no reduction in nuclear safety. Therefore, on the basis of the evaluation of effects, it is concluded that the proposed change is acceptable from a nuclear safety standpoint and no USQ exists.
D20732A	The auxiliary feedwater system (AFW) provides an emergency source of feedwater for decay heat removal through the steam generators. DCN D-20732 replaces the turbine driven auxiliary feedwater pump (TDAFWP) miniflow recirculation (recirc) line valve 1-VLV-003-0944 internals (stem /disk assembly and gasket) to convert valve to a stop (globe) valve. This modification was necessary to support the corrective action of SQ980016PER that identified a stop check valve was inadvertently installed in lieu of a stop (globe) valve by DCN M-06185-A. Implementing the proposed activity will eliminate a failure mode "check valve fail to open" with two check valves currently in series. The modified valve will be aligned in the open position to provide a miniflow path for the TDAFW pump, thus, pump reliability is not degraded. The flow coefficient (Cv) value for the stop check valve and the stop valve are equivalent in the full open position (normal operating position). Existing hydraulic calculations are not impacted by this change; therefore, this activity does not alter the system hydraulic characteristics or invalidate any accident mitigating assumptions as identified in the accident analyses for SQN. An AFW self-assessment revised FSAR Figure 10.4.7-5 and flow diagram 1,2-47W803-2 to show valve 1-	As previously stated, the modified valve will be aligned in the open position to provide a miniflow path for the TDAFW pump, thus, pump reliability is not degraded. The minimum AFW flow rates assumed in the accident analysis are not changed by this proposed change and the ability of the AFW system to meet these flow requirements has not decreased. This activity does not alter the system hydraulic characteristics or invalidate any assumptions for the FSAR chapter 15 accident analyses, nor does it change the basis for TS 3. 7.1.2 and associated basis contains the bases for operation of the AFW system or any other TS. Therefore, implementing this change will not result in a USQ.

DCN	DESCRIPTION	SAFETY ANALYSIS
	VLV-003-0944 as a stop check valve. These documents along with other affected drawings/documents have been generated/revised to document the conversion of valve 1-VLV-003-0944 to a stop valve (isolation) per original design.	
D20733A	The AFW provides an emergency source of feedwater for decay heat removal through the steam generators. DCN D-20733 replaces the TDAFWP miniflow recirc line valve 2-VLV-003-0944 internals (stem /disk assembly and gasket) to convert valve to a stop (globe) valve. This modification was necessary to support the corrective action of SQ980016PER that identified a stop check valve was inadvertently installed in lieu of a stop (globe) valve by DCN M-06186-A. Implementing the proposed activity will eliminate a failure mode "check valve fail to open" with two check valves currently in series. The modified valve will be aligned in the open position to provide a miniflow path for the TDAFW pump, thus, pump reliability is not degraded. The flow coefficient (Cv) value for the stop check valve and the stop valve are equivalent in the full open position (normal operating position). Existing hydraulic calculations are not impacted by this change; therefore, this activity does not alter the system hydraulic characteristics or invalidate any accident mitigating assumptions as identified in the accident analyses for Sequoyah (SQN). An AFW self-assessment revised FSAR Figure 10.4.7-5 and flow diagram 1,2-47W803-2 to show valve 2-VLV-003-0944 as a stop check valve. These documents along with other affected drawings/documents have been generated/revised to document the conversion of valve 2-VLV-003-0944 to a stop valve (isolation) per original design.	As previously stated, the modified valve will be aligned in the open position to provide a miniflow path for the TDAFW pump, thus, pump reliability is not degraded. The minimum AFW flow rates assumed in the accident analysis are not changed by this proposed change and the ability of the AFW system to meet these flow requirements has not decreased. This activity does not alter the system hydraulic characteristics or invalidate any assumptions for the FSAR chapter 15 accident analyses, nor does it change the basis for TS 3. 7.1.2 and associated basis contains the bases for operation of the AFW system or any other TS. Therefore, implementing this change will not result in a USQ.
D20938A	The SQN units have experienced spurious control rod stepping during steady-state operation when the rod control system was in automatic mode of operation. This rod stepping is a result of T-hot process temperature variations. The T-hot signals are used to generate an RCS Tavg signal which is an input to the rod control system. If the variation in the input signal is large enough, rod motion will be demanded. The phenomena is not unique to the SQN units; it has been noted in several other Westinghouse plants as well. At SQN a revision to the signal compensation on the Tavg signal used as an input to the rod control system has eliminated the rod stepping during steady-state	The changes described above do not affect any SAR evaluations (accident analysis or equipment malfunction failures) previously performed. The decreased sensitivity of the of the control system will slow the response of the system to transients. An evaluation of affected events performed by Westinghouse and Framatome concluded that all acceptance criteria continue to be met and that the results and conclusions documented in SAR remain valid. No new accidents or equipment malfunction failures are

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	plant operation. The Tavg signal used by the rod control system incorporates a lead/lag/lag signal compensation 1-ZC-85-5082B (TY-412P in the Precautions Limitations and Setpoint Document). This compensation was provided to enhance rod control performance, but increases the potential for rod stepping due to variations in the measured Tavg. The present and proposed compensation values are as follows: Present Value / Proposed Value Lead/Lag: 50/10 / 40/10 Filter: 5 / 9 Per Westinghouse analysis the values recommended are 40/10/10, however the Foxboro vendor manual states the filter value is limited to 9.5 seconds by design therefore 9 seconds is being used to allow for field adjustment. Per the Westinghouse Parameter Response Evaluation section 2.2 Data Analysis the 40/10/10 values are nominal bounds used in the analysis. Therefore lead values from 50 to 40 seconds and filter (lag) values 5 to 10 seconds are covered by the analysis. As an additional precaution, the break points of the non-linear gain unit 1-ZC-85-5081B (JY-412B in the Precautions Limitations and Setpoint Document) of the Power Mismatch channel will be changed from the current value of ± 1% to a revised value of ± 2%. This change in the non-linear gain will minimize the temperature error contribution to the rod speed demand signal due to steady state nuclear (neutron) power variation. Since these functions are assumed in both the plant operability analyses and in the FSAR safety analyses, the effect of any changes in these setpoints have been analyzed by Westinghouse and Framatome the fuel vendor. Included in this design change is the change to the value that represents the turbine 100% power value derived from the turbine impulse pressure and plant calorimetric measurement. The existing 628 psia pressure value is the average of four transmitters, two each on unit 1 and unit 2. Loop 1-P-1-73 has an input into the rod control system, allowing the transmitter span to be	created and no Technical Specifications are affected. The changes affects a non-safety grade rod control system which has no accident mitigation function and improves the span on the impulse pressure transmitter by using individual instead of averaged value for span. Therefore, on the basis of the evaluation the proposed changes are acceptable from a nuclear safety standpoint and NRC approval is not required prior to implementation.

DCN	DESCRIPTION	SAFETY ANALYSIS
	adjusted to the actual 100% power will also help with rod control stepping using a more accurate input value. Loop 1-P-1-72 does not have an input to the rod control system but will be adjusted the same way as 1-P-1-73. These transmitters are also used with the steam dump and steam generator level controls, these functions will not be adversely affected by this change.	
D20951A	The SQN units have experienced spurious control rod stepping during steady-state operation when the rod control system was in automatic mode of operation. This rod stepping is a result of T-hot process temperature variations. The T-hot signals are used to generate an reactor coolant system (RCS) Tavg signal which is an input to the rod control system. If the variation in the input signal is large enough, rod motion will be demanded. The phenomena is not unique to the SQN units; it has been noted in several other Westinghouse plants as well. At SQN a revision to the signal compensation on the Tavg signal used as an input to the rod control system has eliminated the rod stepping during steady-state plant operation. The Tavg signal used by the rod control system incorporates a lead/lag/lag signal compensation 2-ZC-85-5082B (TY-412P in the Precautions Limitations and Setpoint Document). This compensation was provided to enhance rod control performance, but increases the potential for rod stepping due to variations in the measured Tavg. The present and proposed compensation values are as follows: Present Value / Proposed Value Lead/Lag: 50/10 / 40/10 Filter: 5 / 9 Per Westinghouse analysis the values recommended are 40/10/10; however, the Foxboro vendor manual states the filter value is limited to 9.5 seconds by design. Therefore, 9 seconds is being used to allow for field adjustment. Per the Westinghouse Parameter Response Evaluation Section 2.2 “Data Analysis,” the 40/10/10 values are nominal bounds used in the analysis. Therefore, lead values from 50 to 40 seconds and filter (lag) values 5 to 10 seconds are covered by the analysis. As an additional precaution, the break points of the non-linear gain unit 2-ZC-85-5081B (JY-412B in the precautions limitations and setpoint document) of the Power Mismatch channel will be changed from the current value of ± 1%	The changes described above do not affect any SAR evaluations (accident analysis or equipment malfunction failures) previously performed. The decreased sensitivity of the control system will slow the response of the system to transients. An evaluation of affected events performed by Westinghouse and Framatome concluded that all acceptance criteria continue to be met and that the results and conclusions documented in the SAR remain valid. No new accidents or equipment malfunction failures are created and no TSs are affected. The changes affects a non-safety grade rod control system which has no accident mitigation function and improves the span on the impulse pressure transmitter by using individual instead of averaged value for span. Therefore, on the basis of the evaluation the proposed changes are acceptable from a nuclear safety standpoint and NRC approval is not required prior to implementation.

DCN	DESCRIPTION	SAFETY ANALYSIS
	to a revised value of ± 2%. This change in the nonlinear gain will minimize the temperature error contribution to the rod speed demand signal due to steady state nuclear (neutron) power variation. Since these functions are assumed in both the plant operability analyses and in the FSAR safety analyses, the effect of any changes in these setpoints have been analyzed by Westinghouse and Framatome the fuel vendor. Included in this design change is the change to the value that represents the turbine 100% power value derived from the turbine impulse pressure and plant calorimetric measurement. The existing 628 psia pressure value is the average of four transmitters, two each on Unit 1 and Unit 2. Loop 2-P-1-73 has an input into the rod control system, allowing the transmitter span to be adjusted to the actual 100% power will also help with rod control stepping using a more accurate input value. Loop 2-P-1-72 does not have an input to the rod control system but will be adjusted the same way as 2-P-1-73. These transmitters are also used with the steam dump and steam generator level controls, these functions will not be adversely affected by this change.	
E20693A	EDC D20693 incorporated the Tennessee River Flood Reassessment analyses that reduced the probable maximum flood (PMF), design basis flood (DBF) run-up on vertical, external, unprotected walls, and the DBF surge level within flooded structures by 3 feet. Over a multi-year period, several changes have been made to dams upstream of SQN. Embankments were raised at Fort Loudoun, Watts Bar, Blue Ridge, Boone, Cedar Creek, Chatuge, Cherokee, Douglas, Nottely and Watauga. Additional spillways were added at Tellico and Blue Ridge. New UFSAR Table 2.4.1-5 (part of this change package) contains a complete list of dam modifications. These changes were made to prevent failure of those dams during severe flood events (rainfall and seismically induced), and do not affect the postulated dam failures due to seismic forces. The net effect of the changes is to lower predicted flood levels at SQN for all postulated flood and combination seismic/flood events. This provides additional margin in the design of most features and allows relaxation of requirements in a few areas while maintaining the design basis margins. In addition, it increases the ability to predict floods with increased warning times. Both of these positive impacts result from elimination of the previously postulated dam failures. The flood analysis has been revised	Implementing this design change had a significant impact on FSAR Section 2.4, Hydrologic Engineering, FSAR Section 2.4A, Flood Protection Plan, and affected several other sections of the FSAR. This change does not constitute an USQ because the only impact on the plant is to increase the margin available for response to the DBF event both by increasing available warning time and by decreasing the predicted worst case flood elevation. This conclusion is also based on the following: • The reanalysis used the same meteorological inputs as the original analysis. • All potentially controlling events were reanalyzed. • The same analytical techniques were used as in the original analysis. • The same river system model was used. Only the dam outflows and retention capabilities were adjusted to reflect the elimination of dam embankment overtopping and breachment failures.

DCN	DESCRIPTION	SAFETY ANALYSIS
	accordingly.	• The flood mode response plan remains unchanged. • All plant system and component designs and functions remain unchanged. • The design basis for plant structures remains unchanged. No structural calculations were revised to take advantage of the lower water elevation. • The warning plan has been simplified and verified to ensure that the required 27-hour advance warning will be provided. Interfaces with TVA organizations outside TVAN have been reconfirmed.
E20869A	EDC E20869A revised 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 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 250 volt station batteries supply power to nonsafety-related loads such as the preferred inverters, Technical Support Center (TSC) inverters, turbo-generator auxiliaries, controls for 6.9-kV and 480-V nonsafety-related boards, and switchyard control and relaying equipment.	The station batteries are required for station blackout. However, changing the testing frequency from 3 years to 5 years will not affect the ability of the batteries to perform their intended function. Battery parameters such as battery terminal voltage, pilot cell voltage, pilot cell electrolyte level, and pilot cell specific gravity are monitored on a monthly basis to ensure the battery is in good condition. Additional checks are made of the batteries on a quarterly and annual basis to ensure they are not degraded. The 250-V station batteries do not perform a safety function. During a station blackout the batteries are required to provide control power to reconnect off-site power to the safety related shutdown busses when it becomes available in the switchyard. This capability will not be degraded by changing the frequency of the battery discharge test.
M08655B	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	The CVCS is designed 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

DCN	DESCRIPTION	SAFETY ANALYSIS
	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 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.	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 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.
M12729-A	DCN M12729A removed, relocated, and replaced various MCR data recorders, and added a cathode ray tube (CRT) display to MCR panel 2-M-23A. This DCN contains 20 stages, which were assigned to facilitate individual component replacement. Stages 1, 15, and 16 were implemented to complete recorder replacement for this DCN. The subject recorders were replaced because they were nearing the end of their useful life and replacement parts were unavailable. Three recorders were removed from the facility because their respective process system is no longer in service or because their function is being performed by another device. FSAR table 11.4.2-3 was revised by this change to denote the removal of the remote display associated with recorder 2-RR-90-170 (boric acid evaporator distillate radiation monitor) from MCR panel 0-M-12 the Unit 2 CVCS positive.	This change involves both 1E and non 1E components. Those components that are non 1E do not perform any safety function. The devices that are 1E will continue to meet or exceed the same electrical and seismic requirements as the instruments they are replacing. The new configuration has been verified by engineering analysis and design to conform to design basis seismic, electrical, and physical separation and isolation criteria. This activity does not affect any system function or operation, and does not add any new functions. The subject recorders do not perform any automatic actuations and no TS setpoints are involved. There is no increased probability of malfunction or accidents previously analyzed; no new malfunctions or accidents not analyzed are created; and there is no increase to consequences of malfunctions or accidents. DCN M12729A does not constitute a USQ and does not result in a decrease in any margin of safety.
M13987-A	M13987 modifications replace the Unit 2 lower containment air radiation monitor RE-90-106 with a new system and modified the Units 2 upper	The containment upper and lower compartment air monitors sample lines are isolated during an accident on a CVI signal.

DCN	DESCRIPTION	SAFETY ANALYSIS
	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 Reg. 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 and 112 monitors (DCN M07148, Unit 2), the monitors no longer served a safety-related function. Therefore, the monitor were downgraded to nondivisional 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 nuclear instrumentation module (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.	The area-type containment building monitors (RE-90-271 through 274) measure the accident range radiation exposure rate during an event. RE-90-106 and 112 detector channels are provided to satisfy the requirements of Reg. 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 DBE. 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 Reg. Guide 1.45 and will continue to have limited compliance to Reg. Guide 1.45 for the containment noble gas monitors. Therefore, this activity does not result in an USQ.
T13210C	This change functionally abandons ERCW equipment and components associated with the auxiliary essential raw cooling ater (AERCW) structure	FSAR Figures 8.3.1-7, -12, and -13, and FSAR Figure 9.2.2-1 were revised to reflect the functionally

DCN	DESCRIPTION	SAFETY ANALYSIS
	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 components such as pumps and compressors which are functionally abandoned.	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.
T13211-B	DCN T13211B provides for functional abandonment of equipment/components which are no longer used for operation or shutdown of the plant. Equipment/components addressed by this DCN are: • Unit 1 gross failed fuel detector system (GFFDS) • Unit 1 emergency gas treatment system (EGTS) flow elements 1-FE-65-79, EGTS shield building vent flow, and 1-FE-65-84, EGTS containment annulus flow Functional abandonment includes the revision of primary drawings, Mechanical Design Criteria, and the equipment management system (EMS) to reflect that these components are functionally abandoned. This DCN also provides for the closing of boundary valves on fluid systems to establish a definable boundary of functionally abandoned equipment. This DCN deenergizes powered boundary valves and major components (valves, pumps, compressors, etc.) which are functionally abandoned, or isolates the control air supply where necessary, which prevents inadvertent operation. Boundary components are administratively controlled through tagging and/or hold orders to prevent inadvertent operation.	As a result of the Fiscal Year 1997 (FY97) FSAR Verification Project, discussion of this functionally abandoned equipment was removed from Section 9.2.1.2 of the FSAR text as well as from the FSAR Figure depicting the Component Cooling System flow. The abandoned EGTS flow elements were not discussed in the FSAR text but were removed from the FSAR Figure depicting the EGTS flow diagram. Functionally abandoning these components does not adversely alter the operation or the capability of any equipment important to safety to perform its design function. These changes reduce potential malfunctions by isolating and, where possible, deenergizing these unused portions of the systems, thereby reducing the potential for inadvertent operation or interactions with functioning equipment/ systems. This modification does not involve an USQ.
T13212-B	DCN T13212B provides for functional abandonment of equipment/ components which are no longer used for operation or shutdown of the plant. Equipment/components addressed by this DCN are: • Unit 2 GFFDS • Unit 2 EGTS flow elements 2-FE-65-79, EGTS shield building vent flow, and 2-FE-65-84, EGTS containment annulus flow • 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	As a result of the Fiscal Year 1997 (FY97) FSAR Verification Project, discussion of this functionally abandoned equipment was removed from Section 9.2.1.2 of the FSAR text as well as from the FSAR Figure depicting the CCS flow. The abandoned EGTS flow elements were not discussed in the FSAR text but were removed from the FSAR Figure depicting the EGTS flow diagram. Functionally abandoning these components does not adversely alter the operation or the capability of any

DCN	DESCRIPTION	SAFETY ANALYSIS
	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 (Demin 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. Functional abandonment includes the revision of primary drawings, Mechanical Design Criteria, and the EMS to reflect that these components are functionally abandoned. This DCN also provides for the closing of boundary valves on fluid systems to establish a definable boundary of functionally abandoned equipment. This DCN deenergizes powered boundary valves and major components (valves, pumps, compressors, etc.) which are functionally abandoned, or isolates the control air supply where necessary, which prevents inadvertent operation. Boundary components are administratively controlled through tagging and/or Hold Orders to prevent inadvertent operation.	equipment important to safety to perform its design function. These changes reduce potential malfunctions by isolating and, where possible, deenergizing these unused portions of the systems, thereby reducing the potential for inadvertent operation or interactions with functioning equipment/systems. This modification does not involve an USQ.
T13213-A	DCN T-13213-B completed functionally abandoning-in-place heating, ventilating, and air conditioning (HVAC) supply to the condensate demineralizer waste evaporator (CDWE) building, heating and ventilating in the ADGB and MCR and electrical board room steam humidifiers (steam generator) equipment/ components which are no longer used for operation or shutdown of the plant. Only the CDWE supply and return isolation dampers (1-FCO-30-296, -297, -298, and -299) are safety-related components. These dampers must close, and remain closed, upon receipt of an auxiliary building isolation (ABI) signal. The dampers are being placed in their safe (closed) position, and control air is being isolated to prevent a change in position. This will not adversely affect the capability of the ventilation system to isolate upon an ABI to mitigate the consequences of an accident. The remaining ventilation components within the scope of this change are non safety-related components which perform no limiting or mitigating function for the accidents evaluated in Chapter 15 of the FSAR. Abandoning the components will not adversely affect the ability of accident mitigating equipment to perform its design function.	As a result of the Fiscal Year 1997 (FY97) FSAR Verification Project, discussion of this functionally abandoned equipment was removed from FSAR Section 9.4.9 and FSAR Figure 9.2.7-4 depicting the raw cooling system flow to the CDWE. Functionally abandoning these components does not adversely alter the operation or the capability of any equipment important to safety to perform its design function. These changes reduce potential malfunctions by isolating and, where possible, deenergizing these unused portions of the systems, thereby reducing the potential for inadvertent operation or interactions with functioning equipment/systems. This modification does not involve an USQ.

PROCEDURE	SUMMARY OF DESCRIPTION	SAFETY ANALYSIS
1-SO-70-1R25	This 50.59 evaluation evaluates the compensatory measures being taken for the degraded condition of 2-FCV-70-156. This valve is used to control component cooling system (CCS) flow to the residual heat removal system (RHR) heat exchanger (HX) 2A-A and during normal operation is maintained closed or throttled as necessary to maintain CCS pump minimum flow requirements. When the RHR HX 2A-A is placed in service this valve is opened. Due to the degraded condition of 2-FCV-70-156, the capability to throttle or change the position of this valve cannot be guaranteed. The compensatory measure being evaluated is to align 2-FCV-70-156 to the full open position and leave the valve in this position for normal and accident operation. In the full open position, 2-FCV-70-156 is in the alignment required for shutdown cooling or for accident mitigation. The effect of this alignment during normal operation is that total CCS flow increases and the CCS flow rate to some components has decreased. However, flow rate data taken during normal operation with 2-FCV-70-156 in the full open position demonstrates that all components are receiving adequate flow as evaluated and that total CCS pump flow is within analysis limits provided the spent fuel pit cooling system (SFPC) loads are on the Unit 1 CCS. Therefore, aligning 2-FCV-70-156 in the full open position does not adversely effect the SAR described design function of the CCS in providing (adequate) cooling water to its served components. A compensatory measure will be required to align the SFP Hx load to the Unit 1 CCS. In the event SFP cooling is transferred from Unit 1 to the Unit 2 CCS, the Unit 2 A train CCS must be evaluated for operability under TS 3 / 4.7.3. In addition to controlling flow to RHR HX 2A-A, 2-FCV-70-156 may be closed during a CCS line break to isolate the broken pipe. Due to the degraded condition of 2-FCV-70-156, this valve is no longer available (i.e., is not to be relied upon) to isolate a CCS line break. However, alternate CCS valves can be used to isolate a CCS line break and therefore, there is no adverse effect due to no longer being able to close 2-FCV-70-156 in the event of a CCS line break.	Based on the above alignment of 2-FCV-70-156 does not adversely effect a SAR described design function and does not require further evaluation under 10CFR50.59. However, due to a TS LCO being entered as a result of the degraded condition of 2-FCV-70-156, this change is being evaluated as an adverse change (i.e., a 10CFR50.59 SE is performed).

PROCEDURE	SUMMARY OF DESCRIPTION	SAFETY ANALYSIS
FR-Z.1R12 & EA-32-3R0	PER 00-004645-000 describes a previously unanalyzed accident scenario in which consequential damage from a LOCA results in a break in the non-essential, control air piping inside containment, and the associated control air supply containment isolation valve (CIV) fails to close. This results in an air leak into containment in addition to the mass and energy released by the loss of coolant accident (LOCA). In order to prevent the design pressure of containment from being exceeded, the control air supply to containment must be isolated. The method for isolating the control air supply to containment under this scenario is by operator action and is implemented by procedures FR-Z.1 (high containment pressure) and EA-32-3 (isolating nonessential air in containment). If the control air supply CIV fails to close procedure, FR-Z.1 directs operators to procedure EA-32-3 which instructs operators to manually close valves upstream of the failed open CIV or if necessary shutdown the control air compressors. The adverse aspect of these additional operator actions is that they could accidentally or purposefully result in a loss of control air to the non-accident unit, resulting in a SAR Condition II event. However, since these actions are only taken as a result of a Condition III or IV event, they only minimally increase the frequency of occurrence of a SAR Condition II event on the non-accident unit and; therefore, may be implemented under 10CFR50.59.	These procedures are being issued as a compensatory measure for the degraded/non-conforming condition of PER 00-4645. NRC approval will be obtained prior to making the operator actions in FR-Z.1 and EA-32-3 the permanent fix for mitigating this accident scenario.

TACF	SUMMARY OF DESCRIPTION	SAFETY ANALYSIS
0-00-018-024	In order to replace the auxiliary building (AB) Chiller A, 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 AB) 2) E RCW (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 and 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.
1-01-003-062	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 CVCS charging header isolation valve 1-FCV-062-89 used during normal plant operations. Bypass valve 1-VLV-062-0538 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.	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.

WORK ORDER	SUMMARY OF DESCRIPTION	SAFETY ANALYSIS
WRC346046/WO9711867 000	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.