UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W.
ATLANTA, GEORGIA 30323

Report Nos.: 50-327/87-76, 50-328/87-76

Licensee: Tennessee Valley Authority
500A Chestnut Street
Chattanooga, TN 37401

Docket Nos.: 50-327 and 50-328 License Nos.: DPR-77 and DPR-79

Facility Name: Sequoyah Units 1 and 2

Inspection Conducted: December 6, 1987 thru February 5, 1988

Inspectors: J B Brady for 4/8/88
K. M. Jenison, Senior Resident Inspector Date Signed

Accompanied by: P. E. Harmon, Resident Inspector
D. P. Loveless, Resident Inspector
W. K. Poertner, Resident Inspector
W. C. Bearden, Resident Inspector
M. W. Branch, Sequoyah Restart Coordinator

Approved by: [signature] 4/8/88
for F. R. McCoy, Chief, Projects Section 1 Date Signed
Division of TVA Projects

SUMMARY

Scope: This routine, announced inspection was conducted on site in the areas of: operational safety verification (including operations performance, system lineups, radiation protection, safeguards and housekeeping inspections); maintenance observations; review of previous inspection findings; followup of events; review of licensee identified items; review of IE Information Notices; and review of Inspector Followup Items.

Results: Two Violations (VIOs) were identified.

327,328/87-76-01; Failure to perform adequate post maintenance testing. (paragraph 9.c)

327,328/87-76-02; Failure to follow procedure, two examples. (paragraph 6 and 9.b)

One Unresolved Item (URI) was identified during this inspection. URIs are matters about which more information is required to determine whether they are acceptable or may involve violations or deviations.

URI 327,328/87-76-03; Fifth vital battery concerns. (paragraph 3)

Two Inspector Followup Items (IFIs) were identified.

IFI 327,328/87-76-04; Anubar flow instruments and verification of heat exchanger differential pressure. (paragraph 3)

IFI 327,328/87-76-05; Cable routing deficiencies on Unit 1 cables 1V1936A and 1V1881A. (paragraph 3)

REPORT DETAILS

1. Persons Contacted

   Licensee Employees

   H. L. Abercrombie, Site Director
   J. M. Anthony, Operations Group Supervisor
   *R. H. Buchholz, Sequoyah Site Representative
   M. A. Cooper, Licensing Supervisor
   H. D. Elkins, Instrument Maintenance Group Manager
   R. W. Fortenberry, Technical Support Superintendent
   J. L. Hamilton, Quality Engineering Manager
   M. R. Harding, Licensing Group Manager
   *G. B. Kirk, Compliance Supervisor
   J. T. La Point, Deputy Site Director
   L. E. Martin, Site Quality Manager
   R. W. Olson, Modifications Branch Manager
   B. M. Patterson, Maintenance Superintendent
   R. V. Pierce, Mechanical Maintenance Supervisor
   R. J. Prince, Radiological Control Superintendent
   H. R. Rogers, Plant Operations Review Staff
   M. A. Skarzinski, Electrical Maintenance Supervisor
   *E. K. Sliger, Manager of Projects
   *S. J. Smith, Plant Manager
   *J. H. Sullivan, Regulatory Engineering Supervisor
   B. M. Willis, Operations and Engineering Superintendent

   NRC Employees

   *Attended exit interview

2. Exit Interview

   The inspection scope and findings were summarized on February 5, 1988, with those persons indicated in paragraph 1. The inspectors described the areas inspected and discussed in detail the inspection findings listed below. Dissenting comments were not received from the licensee. The licensee did not identify as proprietary any of the materials provided to or reviewed by the inspectors during this inspection.

   NOTE: A list of abbreviations used in this report is contained in paragraph 19.

   Summary of Violations, Deviations, and Unresolved Items

   (Open) Violation (VIO) 327,328/87-76-01; Failure to perform adequate post maintenance testing - (paragraph 9.c).

   (Open) VIO 327,328/87-76-02; Failure to follow procedure, 2 examples: (1) failure to drain a portion of the auxiliary feedwater system in accordance with AI-3 in support of maintenance - (paragraph 9b); and

(2) failure to adequately verify hydrogen recombiner orifice clear of obstructions per SI-153.4 - (paragraph 6).

(Open) Unresolved Item 327,328/87-76-03; Fifth vital battery concerns - (paragraph 3).

(Open) Inspector Followup Item 327,328/87-76-04; Any bar flow instruments and verification of heat exchanger differential pressure - (paragraph 3).

(Open) Inspector Followup Item 327,328/87-76-05; Cable routing deficiencies on Unit 1 cables 1V1936A and 1V1881A.

3. Licensee Action on Previous Enforcement Matters (92702)

(Closed) VIO 327,328/87-42-01; Failure to Perform 10 CFR 50.59 Safety Evaluation for Revision 3 to the FSAR on Hydrogen Analyzer Accuracy. Specific corrective action and steps taken to prevent recurrence have been evaluated and deemed acceptable in NRC Inspection Report 327,328/87-42. Additional corrective action by TVA prior to startup was an evaluation of past FSAR revisions to verify that proper unreviewed safety question determinations (USQDs) had been performed or that no unreviewed safety questions existed. As a result of an NRC review of results of this additional TVA evaluation, the NRC requested supplemental information concerning the number of additional safety evaluations that were performed. TVA has provided this supplemental information to the NRC in a letter (R. Gridley/NRC) dated January 25, 1988. The NRC inspector reviewed this supplemental information and discussed this issue with responsible personnel. TVA's corrective actions appear to be adequate.

This item is closed.

(Closed) VIO 327,328/87-52-01, Example A; Drawings and Instructions Did Not Reflect Skid Mounted Valves and Specified High Point Vent Valves. This item identified that the design basis for the Essential Raw Cooling Water (ERCW) system was not correctly translated into drawings and instructions in that skid mounted valves and specified high point vents were not reflected on the drawings or in the applicable instructions. The inspector reviewed the licensee's response to the violation and related corrective actions. The inspector verified that the ERCW skid mounted valves were added to the applicable system operating instructions and surveillance instructions and that the ERCW Control Room Drawing was red-lined to reflect these valves. The inspector verified that instrument high point vents were controlled by SI-604, Essential Instrument Operability Verification, for those instruments identified as Unit 2 restart instruments. The inspector also verified that the commitments identified in the licensee's response were identified in the licensee's corporate commitment tracking system. The inspector considers that this item has been adequately resolved for restart of Unit 2.

This item is closed.

(Closed) VIO 327,328/87-52-01, Example C; Failure to Assure Applicable Regulatory Requirements and the Design Basis are Correctly Translated into Specifications, Drawings Procedures, and Instructions. This item identified that the four ERCW screen wash pumps were not ASME code stamped as required by the design documents. The inspector reviewed the licensee's response to the violation and related corrective actions. The screen wash pumps were designed to ANSI B58.1, were seismically qualified, and the motor satisfies class 1E requirements. As a result of this violation the licensee performed a 50.59 evaluation to determine that the screen wash pumps are suitable for their application. The licensee also committed to revise the FSAR in the next annual FSAR update. The inspector reviewed the 50.59 evaluation and verified that the commitment to revise the FSAR was contained in the licensee's Corporate Commitment Tracking system.

This item is closed.

(Closed) VIO 327,328/87-52-01, Example D; Failure to Transfer the Design Basis Into Specifications, in that the metal flexible hose on the ERCW inlet to the diesel generator lube oil coolers was purchased to a design pressure of 100 psi instead of 150 psi ERCW design pressure. The inspector reviewed the licensee's action to resolve the discrepancy by requesting the vendor supply an evaluation that the hose be requalified to the 150 psi limit. The vendor, Flexonics Inc. provided a calculation that showed the maximum working pressure for the item was 240 psi and applied a conversion factor of 0.97 for the 150°F working temperature. The qualified pressure for the hose at 150°F is 232.8 psi, which is greater than hydro pressure of 150% of 150 psi = 225 psi. The inspector agrees that the hose is qualified for the 150 psi application.

This item is closed.

(Closed) VIO 327,328/87-52-01 Example E; Failure to Take Adequate Design Control Measures, in that the 2A ERCW traveling screen level differential transformer was disconnectd without proper control and review. The licensee determined that this instrument was inadvertently left off the Critical Structures, System and Components (CSSC) list, which resulted in work on, and changes to, the instrumentation without implementing the controls and reviews associated with CSSC equipment. Temporary Alteration Control Form (TACF) 82-258-67 was written to document the present configuration of the screen wash level instruments and the disablement of the automatic functions they were designed to initiate (i.e., automatic backwash of the travelling screens on high differential pressure). The inspector reviewed the TACF and USQD (RIMS #B25 871008 558). This evaluation required compensatory measures to manually backwash the screens at regular intervals and to dispatch an operator to the ERCW pump house on accident conditions to ensure continued ERCW operability. The inspector reviewed the compensatory measures taken and the procedure changes that implement those compensatory measures. A Condition Adverse to Quality Report (CAQR) has been implemented to review the CSSC list against other

plant documents to ensure that other instruments and components have not been left off or removed from the CSSC list.

This item is closed.

(Closed) VIO 327, 328/87-52-01, Example F; Modification that Resulted in Cross Connecting ERCW Trains. This item identified a piping and valve assembly that had been installed such that it connected the two drain plugs located downstream of valves 2-67-674A and 674B. The inspector reviewed the licensee's response to the violation and corrective actions. The licensee reviewed the DBVP drawing deviations to determine if a similar condition existed elsewhere within the walkdown program. No other similar conditions were identified during this review. The licensee removed the cross connect piping under work request B227538. The inspector verified that the cross connect piping had been removed and walked down other safety related systems to verify that unauthorized cross connects were not installed.

This item is closed.

(Closed) VIO 327,328/87-52-02, Example A; Failure to Seal Vertical Sleeves in the ERCW Pump House As Required by Drawings. The sleeves used to route instrument sense lines from one elevation to another were found not to be packed in accordance with design drawing requirements to prevent flooding from one floor to the next. The drawings required packing methods and materials that were different than those found by the inspection team. The licensee determined that conflicting and confusing instructions on the drawings caused the craftsmen to use materials and methods not in accordance with the requirements. The entire ERCW pump house was inspected by the licensee, and all sleeves were re-packed according to specifications. The drawings were changed to resolve the conflicting instructions. The inspector reviewed the completed workplan that implemented the sleeve work (#120[illegible]5) and the DCN that changed the drawings.

This item is closed.

(Closed) VIO 327,328/87-52-02 Example B; Heat Tracing on ERCW Instrument Lines Not Installed as Required by Drawings and Design Criteria. The inspectors found that heat tracing was missing on the R4 ERCW pump discharge pressure instrument line. Other lines could not be checked to have proper heat tracing due to insulation on the lines. The licensee's response was to correct the deficiency and to investigate other lines to ensure this heat tracing was correctly installed. Work Plan B284704 re-installed the heat tracing that had been removed to allow access to a spool piece being worked. A walkdown of the ERCW pump house and other areas exposed to freezing was performed by the licensee in WP B[illegible]28714 to ensure that this was an isolated problem No other instances were discovered. The inspectors reviewed the completed work plans to verify corrective actions were complete. In addition, the controlling procedure, Standard Practice (Maintenance)-2, Sequoyah Nuclear Plant Maintenance

Program, was revised to require CSSC equipment such as the heat tracing circuit that was inadvertently left off when WR B123964 was worked on the spool piece.

This item is closed.

(Closed) VIO 327,328/87-52-02, Example C; ERCW Flow Transmitter to Station Air Compressors Not on CSSC List. The violation is another example of instruments required to be on the CSSC list found missing during the inspection. In this instance, the instrument was added by an ECN (5414) but was not added to the CSSC list due to a lack of awareness by the modifications and engineering staff that the instrument was important to the reliable operation of safety equipment. The corrective action of the licensee was to place the identified item on the CSSC list, and to review the CSSC list for accuracy against SMI-O-317-61, which lists those instruments required to satisfy surveillance requirements of TS. To preclude further deterioration of the CSSC list, AI-19, part IV has been revised to require the CSSC committee to review work plans and ECNs to ensure CSSC equipment is properly identified and maintained. The inspector reviewed the corrective actions and determined that they were adequate and complete.

This item is closed.

(Closed) VIO 327,328/87-52-02, Example D, Failure to Establish and Implement Written Procedures for Activities Affecting Quality. This item identified that the licensee did not have an adequate method for assuring that inline instrument and vent valves between the primary root valves and the instruments were properly aligned and periodically verified. The inspector reviewed the licensee's response to the violation and related corrective actions. SI-604, Essential Instrument Operability Verification, was revised. An Appendix was added for verification of proper alignment of the inline instrument valves within the Unit 2 DBVP, phase I, system boundaries meeting one of the following: reactor protection set inputs, engineered safety feature actuating, Final Safety Analysis Report, post accident monitoring or Technical Specification devices. The inspector reviewed SI-604 for adequacy and verified that the surveillance instruction had been revised. The inspector verified the valve alignment checklist for numerous safety related instruments and verified that the long term commitments identified in the licensee's response were contained in the licensee's corporate commitment tracking system. This item appears to have been adequately resolved.

This item is closed.

(Closed) VIO 327,328/87-52-02, Example E; Use of Belden Braid Barriers for Control Panel Wiring in Lieu of Required Six Inch Separation. This item concerned the design criteria for installation of circuits in panels containing both train A and train B class 1E cables. The design requirement is six inches of separation or use of a material such as Belden Braid. Where Belden Braid is used for redundant class 1E wiring,

it must be restrained such that the braids do not touch. Numerous examples were identified of wiring whose braids were in physical contact in panel O-M-27. The violation required corrective action to separate those identified deficiencies and assurance from TVA that similar conditions do not exist in other panels. The cause of the design deviation was found to be a result of a lack of design detail on the installation drawings. The use of Belden Braid was determined to be limited to control room and backup control room boards. Drawings used for those installations were revised to clearly specify the criteria. Work plan 0002-01 was written and implemented to walk down all panels using Belden Braid and to correct any installation deficiencies found. The field work was completed on January 12, 1988. The inspector walked down several panels to determine that the identified deficiencies had been corrected and to ensure other panels had been checked and corrected by the work plan.

This item is closed.

(Closed) VIO 327,328/87-52-02, Example F; Failure to Adequately Accomplish General Construction Specification G-38 In Routing Safety-Related Cables. NRC inspection report 327,328/87-52 detailed findings regarding misrouting of safety related cables 1PP718B and 1PP712B which was contrary to the requirements of installation specification G-38, Installing Insulated Cables Rated Up to 15,000 Volts. At the time the cables were installed, Modification and Alteration Instruction (M & AI)-4, Installation of Control Power and Signal Cables, contained additional guidance which allowed routing of cables in short tray lengths near tray intersections and that the short tray length routes would not appear on the cable routing cards. This was allowed if the length of the unspecified cable was less than 10 feet because some tray designations change at tray intersections. The TVA response to the violation dated November 10, 1987, discussed cable routing including problems and actions from several programs. These included NRC report 87-18 (Special Test Inspection), NRC report 87-52 (ERCW As Built Verification), Cable Testing, Appendix R, and Ampacity programs. The TVA response dated November 10, 1987, committed to completing the long term cable routing program by January 1990. NRC review of the TVA response for example F of violation 2 indicated that TVA's conclusion of technical adequacy appeared premature, since walkdowns and an evaluation of the extent of the problem had not been completed. Also, additional misroutings of both 1E and non-1E cables identified in NRC inspection report 327,328/87-18 had not been resolved. The review also noted that TVA should address the interaction of free air space bundling of 1E and non-1E cables together with separation and segregation criteria in the long range cable routing program and submit a supplemental response that would include this information. NRC inspectors reviewed TVA report, "Routing Inconsistencies of Appendix R Cables, Units 1 and 2, Revision 0" and discussed the basis, methodology, and results with TVA's Department of Nuclear Engineering personnel. Two-Hundred/Seventy-six cables required for Appendix R were investigated by TVA to determine routing. Discrepancies were found between actual routing and the computer cable routing data base. The discrepancies were

evaluated on an individual basis for the attributes of train separation, voltage segregation, cable ampacity, tray loading, environmental qualification and Appendix R considerations. The report provided justification that the disagreements between actual and design cable routing schedules had no safety impact and were technically adequate with the exception of two Unit 1 cables which would be addressed by the long term cable management program. The two Unit 1 cables require resolution prior to Unit 1 restart and will be tracked as Inspector Followup Item, 50-327,328/87-76-05, Cable Routing Deficiencies on cables 1V1936A and 1V1881A.

Inspectors concluded that the results appeared to be adequate to justify the discrepancies and that that Cable Management Program completion date of January 1990 is acceptable. The interaction of routed cables in free air was inspected and addressed in report 87-65 as Unresolved Item 327,328/87-18-01. The supplemental response for this violation example requested in NRC inspection report 327,328/87-52 for long term resolution of free air space issues will be reviewed when submitted. The issue and commitment will continue to be tracked under TVA's Corporate Commitment Tracking System number NCO 870324035. The short term actions and evaluation are adequate to support Unit 2 heatup and startup.

This item is closed.

(Open) VIO 327,328/87-66-01; Failure to Establish, Implement, and Maintain Procedures for System Alignment. The licensee upgraded OSLA 58 to an administrative instruction (AI-58) and revised it to correct inadequacies that previously jeopardized system alignment. Inspection 88-06 reviewed this procedure and found that revision 1 to AI-58 was adequate to support heatup and restart. Performance by the system alignment teams and control room operators were also observed in 88-06 and found acceptable to ensure that the configuration control program will accurately account for equipment configuration. This violation is no longer considered as a heatup item due to the corrective actions that TVA has taken.

(Open) VIO 327,328/87-66-02; Failure to Establish, Implement, and Maintain System Operating Instruction Procedures for System 63 (Safety Injection). The licensee conducted a review of all system operating instruction checklists comparing them to the actual configuration in the plant and the system drawings (OSLA-107). Corrections were made to the checklists prior to rerunning them to ensure they were adequate. Some discrepancies were found in this review during inspection 88-06. This item is considered tied to violation 88-06-02 since similar findings resulted. A remedial program to ensure that all hardware is in the proper position for heatup was conducted. The results were reviewed by the NRC and found adequate. This item is considered adequate for heatup.

(Open) VIO 327,328/88-06-01; Failure to Establish, Maintain, and Implement Procedures for System Alignment. This violation involved failures in implementatin of certification requirements, and several isolated examples

of failures to maintain configuration control. These items have been corrected and TVA's corrective actions appear adequate for heatup.

(Open) VIO 327,328/88-06-02; Failure to Establish Adequate System Operating Instructions. This violation is considered similar to violation 87-66-02 in that it represents a failure to correct generic conditions similar to those of the previous violation. TVA conducted a remedial program to ensure that all hardware is in the proper position for heatup. Conditions on the checklists that were considered unclear were reverified and no discrepancies were found. This action is considered adequate for heatup for both this violation and 87-66-02.

(Open) VIO 327,328/88-06-03; Failure to Prescribe and Conduct Activities In Accordance With Documented Procedures, Instructions, and Drawings. This violation relates to conductive material (spare fuses) found in the diesel generator auxiliary board panel during the NRC walkdown that could invalidate seismic qualification of these panels. TVA corrective action was to walkdown all other seismic panels and to correct any discrepancies. This action ensured that the hardware was adequate for heatup and restart. TVA also reviewed procedures relating to fuses and found that AI-3 allowed fuses removed for clearances to be stored in the bottom of panels. This procedure was changed and now appears to be adequate. This action is considered adequate for heatup.

(Closed) URI 327,328/86-20-09; Penetration General Design Criteria (GDC) This issue involved a question on whether or not the licensee met GDC requirements 55 and 56 on specific penetrations for the chemical and volume control system. As a result of the discussions that followed the identification of these specific penetrations, written correspondence between the licensee and NRC ensued. This written correspondence is described in section 3.6 of the Sequoyah SER. As a result the NRC determined that the licensee met the GDC in certain cases, in certain cases modifications had to be made to the plant, and finally two exemptions to the GDC were required. The NRC issued two exemptions to the applicable GDC requirements (Zech/White) dated December 3 and 4, 1987, respectively. This issue has been resolved to the satisfaction of the NRC as stated in the Sequoyah SER.

This issue is closed.

(Open) Unresolved Item, 327,328/87-76-03; Fifth Vital Battery and Battery Room Deficiencies. The safety system outage modification inspection (SSOMI) report 86-68 and Inspection Report 87-40 identified several deficiencies associated with the fifth vital battery installation. Pending correction and NRC review of corrective action, TVA has placed a caution order on the battery to prevent its use in satisfying Technical Specification requirements for operation. The caution order does not prevent using the battery for emergencies. This unresolved item is being opened to track all fifth vital battery concerns that require resolution prior to accepting the battery for safety related use as a power source to

fulfill Technical Specification requirements. Items requiring resolution are:

a. Seismic qualification documentation for correction of gaps between the bottom of the battery racks and the cement pad previously identified by NRC report 86-68, deficiency D-2.4-12.

b. Bend radius violations on battery intercell connectors previously identified by NRC report 86-68, deficiency D-2.4-13.

c. HVAC duct installation seismic qualification documentation previously identified by NRC report 86-68, URI U-2.2-1.

d. Certification for battery rack anchor bolts and battery rack bolts previously identified by NRC report 86-68, deficiency D-2.1-7, Sample 45-1, 45-2.

e. Walkdown inspection observation in 5th battery room identified by IFI 87-40-03.

The items listed above are administratively closed and are now to be tracked under one common unresolved item (327,328/87-76-03) in order to ensure comprehensive NRC review. This item is a long term, non-restart item.

(Open) Unresolved Item 327,328/87-18-01, Potential for Secondary Bridging, Potential Use of Flamemastic as an Alternative to Solid Barriers, and the Lack of Criteria for Separation of Safety-Related Conduits. NRC special test inspection 50-327,328/87-18 documented an example of a train A conduit that was routed in close proximity to an uncovered train B cable tray. Inspectors expressed concern that a lack of criteria for separation of safety-related conduits did not meet the intent of Section 8.3 of the FSAR in that protection and independance of redundant circuits may be compromised. This item also included concerns regarding routing of cables in free air and examples of cables which had been mis-routed in unspecified trays. The status of these items was reported in NRC inspection report 327,328/87-65, which had three remaining items requiring further review after additional licensee action. NRC report 87-65 reported correction of 2 non-divisional cables that had been misrouted in a divisional tray. The three remaining items included: (1) correction of a divisional cable that had been misrouted in a non-divisional tray; (2) the potential for secondary bridging due to misrouting of cables in cable trays; and (3) both an example and the generic concern regarding the lack of criteria at Sequoyah for separation between cable trays and conduit runs. Section 8.3 of the Sequoyah FSAR includes requirements for separation of cable trays, however, it does not address a criteria for conduit/conduit or conduit/tray separation. A review of TVA construction specifications, modification procedures, and conduit installation procedures in conjunction with TVA/NRC discussions indicated a lack of guidance in this area. NRC letter S. D. Richardson/NRC to S. A. White/TVA of December 31, 1987, "Criteria For Separation of Safety-Related Conduits

(NRC Inspection Report 327,328/87-18)", was sent to TVA after a technical review raised significant questions as to the adequacy of installations at TVA. The letter requested TVA provide analyses to demonstrate that faults in one raceway (conduit or cable tray) would not cause internally generated fires that affect the functional integrity of redundant safety-related circuits in adjacent raceways. The letter required a response within 30 days and resolution of the concern prior to the restart of the Sequoyah units. CAQR SQN871585 was written to address item #1, the divisional cable that was misrouted in the non-divisional cable tray. NRC inspectors reviewed the CAQR and performed a field inspection to verify the cable had been restored to the correct divisional tray. Item #1 is closed. Item 2 regarding the potential for secondary bridging due to misrouting of cables in cables trays, is addressed in NRC Inspection Report 327,328/87-52. Item #2 is closed. Item #3, the example and generic concern regarding the lack of criteria for conduit/cable tray separation, has been reviewed. The specific example in NRC Inspection Report 327,328/87-18 envolved a 3-inch train A conduit (2M-3240-A) which was routed across the top of a train B cable tray (MS-B). Inspectors reviewed cable and conduit records for conduit 2M-3240-A and cable tray MS-B at the interaction point. Records indicated that the tray contained several Unit 1 cables and 4 low voltage Unit 2 conductors associated with the spent fuel pit pump B-B, spent fuel pit heat exchanger A & B and CCS booster pump 2B-B. The interacting conduit contained 5 low voltage Unit 2 cables for the pass system. Inspectors concluded that as both interacting raceways contained low voltage cables and there was some separation between the cable tray and conduit, resolution of this specific example is not required for heatup. Inspectors discussed the generic issue with TVA and Licensing personnel. The TVA formal response to the NRC letter had not been drafted. This item is open pending NRC review of TVA's formal response. This is a startup item.

(Closed) Unresolved Item 327,328/86-68, U-2.4-3, Unknown effect on bearing pressure on Class 1E cables in unsupported vertical bundles. Unresolved Item U-2.4-3 was opened in NRC Inspection Report 327,328/86-68 due to field inspection in the cable spreading room which resulted in a concern that vertical bundles of flamemastic coated 1E and non-1E cables had excessive bearing pressure where the bundle drops over the sharp edge of a cable tray. The NRC inspectors discussed the issue with TVA's Department of Nuclear Engineering personnel and viewed the Wyle video tapes of tests run to determine adequacy of the static and dynamic loading on the cables due to weight of the cabling and the weight of the flameastic that was added after construction. Inspectors also reviewed test results used to justify ampacity reduction values used by TVA. TVA staff and Wyle personnel concluded that bearing pressure was not adequately addressed for vertical bundles that drop over sharp edges. This was addressed during the exit and the Site Director committed to addressing the issue prior to unit restart. NRC inspectors reviewed a memorandum (J. Hosmer, TVA/M. Harding, TVA, Sequoyah Nuclear Plant - Resolution of Bearing Pressure on Air Drop of 1E Cables - NRC commitment No. NC087-0029001) which contained summary justification for the issue being a non-restart item. The memorandum transmitted DNE calculation SQN-E2-031 (RIM B25

880108 815, Sidewall Bearing Pressure For Class 1E Free Air Drops) and NUREG/CR-4548 (Correlation of Electrical Reactor Cable Failure With Materials Degradation, March 1986). Inspectors reviewed the references and discussed the issue with DNE engineers that performed the calculation. The DNE calculation indicates that several cables within the bundle have bend radius violations and that the cables exceed the allowed bearing pressure calculations by a factor of 5 to 6. Cable loading due to weight was well within calculated limits. Due to the complexity of the issue, the potential long term corrective action, and existance of a long term post-restart Employee Concern (CATD) No. 109.00-NPS-01 R3 concerning bend radius violations, a brief summary is presented below to facilitate future inspection efforts.

The TVA DNE calculation selected a worst case bundle based on a walkdown of the cable spreading room. Although cables in the worst case bundle exceeded bearing pressure limits, NRC inspectors concluded that the issue did not meet the accepted TVA Sequoyah restart criteria based on the following information: (1) conservative assumptions were made that the entire weight of all cables and all flamemastic was bearing on the sharp edge; (2) the calculation assumed cables with the lowest bearing pressure limit were the cables in contact with the sharp edge; (3) no credit was taken for support offered by the rigid flamemastic; (4) the weight of flamemastic was calculated using the length of run, times the cross-sectional area difference between the sum of the area of a circle that would encompass the bundle and the cable cross-sectional area of the cables; (5) no credit was taken for bottom support where cables entered other trays; and (6) no credit was taken for increased surface area (increased support) due to flamemastic on each side of the cables at the point of contact with the sharp edge. Inspectors concluded that quantitative inclusion of all conservatisms may have reduced the bearing pressure to an acceptable value, however, bend radius violations would still exist. Testing information prepared by Sandia National Laboratory and included in NUREG/CR-4548, Correlation of Electrical Reactor Cable Failure With Materials Degration, indicated that "creep short out" due to applied stress on a cable over a radius causes a slow migration of the conductor through the insulation. The phenomena is temperature dependent with a slow relatively linear creep rate at room temperatures and shows little change between 1 day and 40 years. Several mitigating factors would tend to minimize the effect with time. These factors include: (1) as the applied stress tends to flatten the conductor strands, the surface area in the direction of the stress increases which in turn decreases the cross-sectional force reducing the creep rate; (2) as the insulation deforms, it widens and provides a greater area to support the given stress; (3) as the insulation and conductor conform to the radius at the point of contact the applied stress seeks uniform distribution over the entire contact area; and (4) insulation aging and irradiation (where applicable) tend to harden polymer and rubber insulation reducing the creep rate. Cracking as related to bearing pressure is not of significant concern as it is mainly related to tensile stresses rather than the compressive stresses associated with creep short out. Hot cracking of stressed bends in cables does occur but is associated with multiple

thermal cycling of the cables above 175 degrees F. Inspectors concluded that the conservatism in the calculation and the mitigating factors offer reasonable assurance that the installation is satisfactory for heatup, startup and operation. Further evaluation and possible long term corrective action is necessary as testing data with regard to age (i.e, justification for use as a basis for a 40 year qualified life) contains unverified extrapolations from limit test data in an area that is not well understood. This item is closed and will be continued to be reviewed as Sequoyah Employee Concern Corrective Action Tracking Document (CATD) No. 109.00-NPS-01 R3 and Corporate Committment Tracking System item NCO87-0029001 as a post restart item.

(Closed) Violation 327,328/86-68-02, paragraph c, section 2.1.3.1, inspection sample number 1-2; Lack of Material Certification for 1-1/8" Diameter Studs and Nuts for Auxiliary Feedwater Pump 2A-A Discharge Flange to Navco Spool Piece 2AFD-13. During the NRC Safety System Outage Modification Inspection (SSOMI) 327,328/86-68, material certification could not be located for the 1-1/8" diameter studs for Auxiliary Feedwater Pump 2A-A discharge flange. Subsequent investigation by TVA failed to locate the missing certification. Lacking documentation of certification required removing paint from the studs in order to inspect vendor markings and certify the material. Work request B244476 was completed on January 19, 1988, and documented removal of paint and certified vendor markings on the studs in question. Inspectors reviewed the work request against the original specification for the studs and then conducted a field inspection of the studs. The studs were found to be ASTM A 193, Grade B7 as required by the material specification. Licensee action on this item is adequate.

This item is closed.

(Closed) VIO 327,328/84-11-02; Failure to Use Appropriate Drawings During Maintenance. This violation involved maintenance personnel rewiring a PORV controller they interpreted as being incorrectly wired by reviewing a single drawing. The cause of the violation was determined to be personnel error in failing to obtain and review all applicable drawings. Further review by the licensee has determined that the single drawing reviewed could have been adequate, and that the error made was in the review process. Further review of additional drawings might have prevented the error, but there is no conclusive evidence that the same error would not have been made regardless of the number of drawings reviewed. The misinterpretation of how the PORV's bistable pressure switches actually functioned was an avoidable mistake, but might have been made in spite of a more extensive review, because the other drawings would have shown the same switch arrangement. The misinterpretation of the circuitry involved is considered an isolated event. The individuals involved were counselled and training was conducted to remind instrument maintenance personnel to obtain and review all applicable drawings during maintenance and modifications planning. This corrective action is considered adequate.

This item is closed.

(Closed) VIO 327/84-24-03; Inadequate Tool Control. This flux thimble guide tube ejection event item was reopened in Inspection Report 327, 328/87-43 when all the items associated with this event were reopened for further review for adequate corrective actions. After reviewing the root cause determination and the corrective actions, the inspector determined that this item was adequately identified and corrected. Said review was documented in Inspection Report 327,328/87-50.

This item is closed.

(Closed) VIO 327,328/86-68-01; Failure to Assure that Applicable ASME Code Requirements Were Included in the Procurement of Specific Safety-related Code Class Equipment. This violation had three examples as addressed in the notice of violation dated April 24, 1987. Example A was closed in Inspection Report 87-60. Examples B and C were closed in Inspection Report 87-40. Therefore, violation 327,328/86-68-01 has been addressed in its entirety.

This item is closed.

(Closed) VIO 327,328/86-68-03; Failure to Identify and Control Materials, Parts, and Components by Heat Number, Post Number, Serial Number, or Other Appropriate Means. This violation contained three examples including multiple deficiencies as addressed in the notice of violation dated April 24, 1987. All deficiencies were addressed with adequate results in Inspection Report 87-40. Therefore, violation 327,328/86-68-03 has been addressed in its entirety.

This item is closed.

(Closed) URI 327,328/87-52-05; Concern Over Observed Flow Conditions For Diesel Generator Coolers and Upper Containment Coolers. This item concerned the adequacy of the installed flow instruments and configuration control and equipment degradation aspects of the inspectors' observations. The inspector reviewed the closure package associated with this item and determined that the flow to the coolers was being controlled in that the configuration log reflected that flow was established. With respect to the adequacy of the installed gages the licensee determined that the gages are designed for an overranging application and that the gages are used for local indication only. The licensee is reviewing the system design to determine if larger scale gages would be more appropriate in these applications.

This item is closed.

(Open) VIO 327,328/87-60-02; Failure to Meet The Requirements of 10 CFR 50.59. This violation resulted from the licensee modifying the AFW system such that the 2A-A AFW pump could no longer deliver 440 gpm as required by the TS. This violation is presently under consideration for escalated enforcement. The licensee has replaced the pump internals on the 2A-A AFW pump to ensure that the pump will be able to deliver the required flow to

the steam generators. The inspector witnessed initial testing of the 2A-A AFW pump after maintenance was completed and verified that the pump could deliver greater than 440 gpm. The licensee's actions have resolved the inspector's concern with TS compliance and are adequate for the restart of the Units. This item will remain open pending enforcement action.

(Closed) URI 327,328/87-54-01; Implementation of Commitments. This item identified that the Corporate Commitment Tracking system was not effective in ensuring that commitments were met. The inspector conducted interviews with appropriate licensee personnel and verified that selected open commitments were being tracked by the licensee. Since this item was opened there have been numerous instances where the licensee has notified the NRC verbally, on the day that a commitment was due, that the commitment due date would not be met. The concern over missed commitment due dates was discussed with the plant manager at the exit interview. The problem appears to be a notification problem and not a commitment completion problem. No commitments were identified by the NRC inspector which did not appear in the CCTS and were being tracked by the licensee. The licensee is aware of the commitments and when they are due; however, the licensee has not instituted an effective method of ensuring that the due dates are met or that the NRC is notified prior to the commitment due date being missed. Although no regulatory requirements were violated, the inspector will review this area during routine monthly inspections to determine if the licensee has made progress in resolving missed commitment due dates.

This item is closed.

(Closed) URI 327, 328/85-16-03; Seismic Design of Vital Batteries Concerning Use of Spacers At End Rack. This issue involved the adequacy of the seismic installation of the plant vital batteries. The inspector reviewed TVA and vendor installation drawings against the field configuration of the battery racks for the 125V vital DC battery banks and the diesel generator battery banks. The 125V vital DC battery racks were installed in accordance with applicable vendor drawings; however, TVA had received a memorandum dated, March 27, 1985, from the vendor, GNB Batteries, Inc., stating that there may be a gap between the end cell and end stringer of greater than 1/4-inch in some Class 1E battery installations. A gap greater than 1/4-inch was determined by the vendor to be unacceptable. This recommendation was apparently based on the configuration used for seismic qualification testing of the batteries. A review of the installation indicated that gaps of approximately 1/2-inch existed at one end of some of the racks. The vendor drawing did not show a gap limit or a spacer or the original installation. The vendor recommended that a spacer be inserted to bring this gap to less than 3/8-inch. TVA has initiated workplans to install spacers of an approved material in the vital battery banks. The inspector requested tha' e licensee provide further information concerning TVA's actions in r, onse to the vendor notice, including any determination concerning operability during a seismic event. The site Office of Engineering (OE) field inspected the vital batteries on April 19 and 20, 1985, and determined that the vital batteries had end gaps greater than recommended by the

vendor. A Nonconformance Report (NCR) was written by site OE dated April 24, 1985. The NCR states, "The spacing between the end cell and end stringer of the rack on vital batteries I thru IV was measured and found to exceed 1/4-inch required by seismic testing. The fifth vital battery has one cell missing at the present time and no spacer was added." The NCR further states the following:

> "In the absence of spacers, seismic loading could cause failure of the vital battery cells. There is evidence that structural failure would likely occur at the battery terminal posts. Such a failure of one cell causes the loss of the entire battery system. Although it was not possible to analytically predict the seismic behavior of unqualified (without spacers) configurations, a failure of this type must be considered probable."

The condition was defined as a "significant condition adverse to quality." The NCR was signed by the responsible Branch Chief on April 24, 1985. This date was corrected on a later copy to May 1, 1985, due to changes after an additional OE review. The NCR was received by the site Office of Nuclear Power (NUCPR) on May 1, 1985. In memoranda dated May 29 and June 5, 1985, NUCPR rejected the NCR and subsequent Failure Evaluation/Engineering Report (FE/ER). The memoranda stated that, "This item was handled by Sequoyah Site Services under the operating experience review program. In addition, this item was not an OE nonconformance nor a breakdown in an OE QA program requirement; therefore, it should not have been written. This item required expeditious maintenance activities and was acted upon by the plant staff when a copy of the vendor letter was received." The inspector reviewed the PRO (1-85-1C0) written on this event to document the plant staff review. The documents stated the following conclusions:

- An evaluation of the vendor letter by plant Compliance personnel determined that the recommendation had no effect on battery operability; however, personnel did consider the recommendation to be a prudent maintenance practice.

- Compliance personnel determined the absence of between-cell spacers did not adversely effect operability of the batteries.

- Because the recommendation was associated with seismic testing, it was handled on a priority maintenance basis; however, inspection of the physical mounting of the batteries and discussions of the manufacturer's recommendations led to the conclusion that it was not a condition of operability.

These statements are in direct contradiction to the NCR written by the Office of Engineering, yet no coordination of these conclusions is apparent between OE and Plant Compliance.

10 CFR 50, Appendix A, Criterion 2, Design Bases for Protection Against Natural Phenomena, states: "Structures, systems, and components important

to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and serches without loss of capability to perform their safety functions. The design bases for these structures, systems, and components shall reflect: (1) Appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena and (3) the importance of the safety functions to be performed."

Regulatory Guide 1.10, Seismic Qualification of Electric Equipment for Nuclear Power Plants endorses IEEE standard 344-1975, recommended practices for seismic qualification of class 1E equipment for Nuclear Power generating stations, which states that:

> "The seismic qualification of Class 1E equipment should demonstrate an equipment's ability to perform its required function during and after the time it is subjected to the forces resulting from one SSE. In addition, the equipment must withstand the effects of a number of OBEs (see Sections 5,4 and 6.1.4) prior to the application of an SSE. The equipment to be tested shall be mounted on the vibration generator in a manner that simulates the intended service mounting. The mounting method shall be the same as that recommended for actual service. The mounting method shall use the recommended bolt size and configuration, weld pattern and type, etc. The effect of electrical connections, conduit, and sensing lines, etc shall be considered. The orientation of the equipment during the test shall be documented and shall be the only orientation for which the equipment is considered qualified unless adequate justification can be made to extend the qualification to an untested orientation."

TS 3.8.2.3 requires that the following D.C. vital battery channels be energized and OPERABLE:

CHANNEL I — Consisting of 125-volt D.C. board No. I, 125-volt D.C. battery bank No. I and a full capacity charger.

CHANNEL II — Consisting of 125-volt D.C. board No. II, 125-volt D.C. battery bank No. II, and a full capacity charger.

CHANNEL III — Consisting of 125-volt D.C. board No. III, 125-volt D.C. battery bank No. III, and a full capacity charger

CHANNEL IV — Consisting of 125-volt D.C. board No. IV, 125-volt D.C. battery bank NO. IV, and a full capacity charger.

Contrary to the above, on April 2, 1986, TVA engineering received a letter from GNB Batteries, Inc. stating that the installed 125V vital batteries did not meet the tested configuration. Plant personnel received this

letter on May 1, 1985. Adequate justification was not made to extend the qualification to the untested configuration. Installation of battery end spacers to correct this problem was accomplished on May 14, 1987. During this time period, Unit 2 was operated without declaring the vital batteries inoperable, or meeting the required action statement. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Action," 10 CFR Part 2, Discretionary Enforcement, the event described above meets the following criteria:

- The Licensee was forced into an extended shutdown related to poor performance over a long period following their August 1985 shutdown.
- The Licensee has developed and is aggressively implementing their Nuclear Performance Program for problem identification and correction.
- NRC concurrence is needed by the Licensee prior to restart.
- Enforcement action is not necessary to achieve remedial action.
- The violation occurred prior to the August 1985 shutdown.
- The violation was non-willful and would not have been categorized as higher than Severity Level III under the NRC's enforcement policy.

The inspector reviewed the memorandum from GNB batteries, Inc. to TVA dated March 27, 1985. It contains information as discussed above which appears to be reportable under 10 CFR Part 21. This issue is being reviewed by OSP staff and will be discussed with the vendor. The current configuration of the vital batteries includes end spacers as recommended by the vendor. This was verified by a walkdown conducted by the inspector.

This item is closed.

(Closed) URI 327,328/87-17-01; Loss of ERCW Caused Inoperability of Control Room Ventilation, Radiation Monitors and the Operating Train of RHR. This URI addressed an event which occurred March 11, 1987. An ASE shut valve 1-FCV-67-127 in a hold order for work to be performed on the ERCW piping. This rendered the A train equipment inoperable. The operator failed to realize that 0-FCV-67-208 was shut making inoperable a portion of the B train supplying the station air compressors. Approximately three hours later it was discovered that the station air compressors had tripped on lack of cooling water. This resulted in a loss of control air to valves in both trains of control room ventilation and isolated four radiation monitors. This event occurred because the operator failed to verify that all of the B train ERCW was operable before isolating the A train. In addition, the 1A RHR pump room temperature was noted to be increasing following the event. This was caused by continued operation of the 1A RHR pump following room cooler isolation.

The inspector considered the following in review of this event:

- Appropriate use of procedures
- Independent verification of actions affecting quality
- Configuration control throughout the event
- Corrective actions taken following initial discovery of the problem

Appropriate Use of Procedures/Control of Plant Configuration

During the midnight shift on March 11, 1987, an SRO received a clearance request to tag the A train ERCW discharge header in order to replace the 24-inch piping from 2-FCV-67-146 to the first discharge header tee. This work was to be performed per WP12313 under WR B202762. In an interview, the SRO stated that he began preparing the clearance, but had not fully completed the review by the end of his shift. The oncoming SRO took the clearance and with minimum review (which was normal procedure at the time) proceeded to hang the hold order tags and isolate the header. The clearance did not state that this action would make the operating train of RHR inoperable by isolating the associated room cooler. Interviews with the preparer of the clearance revealed that he had assumed that compensatory measures would be taken at the time the tagging was accomplished in order to keep the operating train of RHR operable. The SRO tagging the system stated that he failed to realize the error.

TS 3.4.1.4 states that, "Two residual heat removal (RHR) loops shall be OPERABLE and at least one RHR loop shall be in operation." The action statement requires that, "With less than the above required RHR/reactor coolant loops OPERABLE immediately initiate corrective action to return the required RHR/reactor coolant loops to OPERABLE status as soon as possible."

Contrary to the above, on March 11, 1987, the licensee made inoperable the operating loop of RHR and did not realize or correct the condition for 10 hours. The pump, however, did continue to operate during this time. This is an example of violation 87-30-01 which was discussed with Mr. S. A. White in a letter dated July 20, 1987. This multiple example violation involved procedural inadequacies and/or inadequate procedure implementation which has given cause for concern over the control TVA's staff has over Sequoyah's operational activities, particularly in the areas of system and equipment status, procedural changes and testing. Corrective actions currently in progress for violation 87-30-01 should correct the problems that caused this event. Therefore, TVA need not respond additionally to this example of the violation.

The SRO who tagged out ERCW A train equipment also did not realize that the B train supply to the station air compressors through valve 0-FCV-67-208 was also isolated. 0-FCV-67-208 was not on SOI-67.1, Essential Raw Cooling Water System, that was currently on record.

Therefore the ASE had no indication that the valve was closed. TVA had already identified this error and had corrected SOI-67.1 in revision 27 dated November 14, 1986. The licensee failed to re-run the procedure or verify the correct position of 0-FCV-67-208 following the update.

TS 6.8.1 requires that written procedures be established, implemented and maintained covering the referenced activities. Contrary to this, SOI-67.1 failed to verify the correct position of 0-FCV-67-208 and avoid making required safety-related equipment inoperable. This situation existed over four months following the discovery of an error in the procedure.

This violation is considered a further example of 87-30-01 and corrective actions in progress should prohibit repeat occurrences of this event. Therefore, TVA need not respond to this example of the violation.

Independent Verification of Actions Affecting Quality

The inspector determined that independent verification was lacking in the preparation of the hold order (number 2262). This problem among others with the clearance program at Sequoyah is being corrected through corrective actions being performed under violation 87-30-01. The inspector determined that the laydown and isolation of the hold order boundary was independently verified and did not contribute to this event.

Corrective Actions Taken Following Initial Discovery of the Problem

Following the initial loss of service air the SRO re-reviewed the clearance and began walking down the hold order boundary. This review found and corrected the problem with the service air compressors. It failed, however, to discover the problem with the operating train of RHR. Additionally, the inspector found the unit operators response to the event to be inadequate in that the root cause for the problems was never determined. Corrective actions currently in progress for violation 87-30-01 should correct the problems that caused this event. Therefore, TVA need not respond additionally to this example of the violation.

The inspector had no further questions concerning this event. This item is closed.

(Closed) VIO 327, 328/86-68-05, paragraph c, section 2.5.5, deficiency D-2.5-1; Auxiliary Control Room Indicator Modules Power Separation and Mounting. This issue involved an observation that Module side panel protectors were missing on indicators for safety train A and B. Modules had engaging latches loose or not engaged, compromising seismic mounting. Additionally, one cable had four bare exposed leads. The inspector reinspected panels in the auxiliary control room and determined that all deficiencies previously noted had been corrected.

This item is closed.

(Closed) URI 327, 328/87-02-05; Adherence to Directions from HP Technicians and Compliance with HP Practices. During a tour of the auxiliary building, an NRC inspector noted that a worker had violated a "notify HP prior to entry barrier" and had entered to perform a task contrary to instructions given to the worker to remain outside the area until the smear in the area had been counted. The HP technician had frisked the area in addition to taking a smear. The entry took place while the HP technician had left to count the smear sample. The smear sample results showed no contamination. The NRC inspector interviewed all workers in the area, as well as the HP technician, and cautioned them on compliance with HP procedures including verbal instructions. The NRC inspector discussed the matter with the HP superintendent and the modifications supervisor and noted that increased emphasis would be placed on monitoring HP practices during inspectors' tours. Due to the low safety significance of this particular incident, it was characterized as unresolved pending future observation of routine HP practices. The inspection staff has noted few discrepancies in this area since the January 1987 instance. In addition, all 1987 radiological incident reports to date have been reviewed for recurrence of incidents of this type. No similar instances were noted. It was noted, however, that second offenses for infractions such as wearing the wrong TLD have resulted in three day suspensions and two cases of entry into a high radiation area with improper dosimetry have resulted in employee termination.

This item is closed.

(Closed) URI 327,328/87-50-03; Containment Spray Pumps. This issue involves a lack of retrievable documentation to substantiate discisions made to install an orifice in the Containment Spray (CS) system. The licensee committed to the following:

- Perform an engineering evaluation of the test results for CS pump 2A-A and verify that test data is not indicative of possible pump failure.

- Discuss further the need to scale control room indication using previous test results.

- Review the location of the Anubar Flow instruments and evaluate the possibility of any needed modifications.

- Verify heat exchanger differential pressure (DP) during each Surveillance Instruction (SI) or American Society for Mechanical Engineers (ASME) boiler and pressure vessel code section XI testing of the pump and determine an acceptable fouling factor.

Disposition of the four items is addressed below.

The pump test run, September 8, 1987, (2B-B) and September 11, 1987, (2A-A) indicates that the pumps are within manufacturer's specifications. The ultrasonic flow meter used during Surveillance Test Instruction

(STI)-65 was approximately 1.9% low at the upper flow ranges used to determine flow characteristics. This, in conjunction with other data taken over a period of years would indicate that the 2A-A pump is still servicable. The licensee's investigation of this matter appears to be adequate.

The need to scale control room indication was discussed with TVA and TVA may decide to mark flow indicators. However, there is no criteria to have a calibrated flow indication in control room. Only an indication of flow is required. The licensee's actions appear to be adequate.

The location of anubar flow instruments and verification of heat exchanger differential pressure are long term issues which are not required to be completed prior to Unit 2 heatup or startup. These long term commitment items will be addressed through review of the corporate commitment tracking system and will be followed with IFI 327, 328/87-76-04.

The heatup requirements of this URI have been satisfied for Unit 2 heatup. This item is closed.

(Closed) VIO 327/84-23-01, 328/84-24-01; Inadequate Corrective Action for Unconfirmed Piping Analysis Operational Modes Input Data. This issue involved a violation for inadequate licensee corrective actions for the resolution of a Non Conformance Report. The licensee admitted the violation with clarification. Prompt corrective action was taken to resolve the concerns expressed in NCR SQNCEB8205. However, the corrective action was not properly documented on the NCR, and the NCR had not been closed upon completion of the corrective action. The licensee reviewed the programmatic control over operational modes for SQN in May 1982. In addition, the adequacy of the SQN piping analysis was to be reviewed, based on the Watts Bar Nuclear Plant (WBN) operational modes review. These reviews were completed in February 1984, with no potential safety problems identified at SQN. The licensee has signed all ENDES action complete on NCR CEB8205, R1, as of December 26, 1984, and the Office of Engineering (OE) civil discipline is tracking completion of the documentation of corrective action requirements via the Tracking and Reporting of Open Items (TROI) System. The licensee's actions appear to be adequate.

This item is closed.

(Closed) VIO 86-68-02; Sample (17-1) Lack of Material Traceability Documentation for Diesel Generator Anchor Bolts. During the performance of a recent safety system outage modifications inspection (SSOMI) several anchor bolts were identified that did not have adequate material traceability to material specifications. In response to this SSOMI violation the licensee issued CAQR SQP870937 which required that samples of 28 of the 144 anchor bolts in question, be tested to determine their chemical and physical properties. The results of these tests were evaluated and it was determined that the installed bolts are adequate to perform their intended function. The inspector reviewed the above test

results and evaluation and determined that, as the reported chemical properties are comparable to those of the specified AISI 1141 material, and the tensile strength is slightly greater than specified, the licensee's evaluation is appropriate and the installed anchor bolts are acceptable.

This item is closed

(Closed) VIO 328/85-17-04; Failure to Follow Emergency Diesel Generator Surveillance Test Procedure. This violation is identical in technical content and required corrective action to violation 327/85-17-05, which was previously reviewed, determined to be acceptable, and closed in Inspection Report 327,328/87-17.

This item is closed.

(Closed) URI 327,328/87-23-01; Control of Safety Related Diesel Generator (DG) Spare Battery Components Beyond the Confines of a Designated Warehouse. Vital EDG battery cells were stored in the EDG building. Several of the cells could not be properly identified in the original report. The loss of control of this material appears to be an isolated case stemming from the inability of power stores to maintain the batteries in normal storage areas, and personnel error in that the transfer of the material was not properly tracked. The inspector reviewed the storage maintenance program for the spare bank of EDG batteries and finds that program to be adequate. Based on an adequate storage and maintenance program, proper identification of the EDG spare batteries, and tagging of 8 batteries as defective to prevent use in safety related systems the actions for this URI are adequate.

This item is closed.

4. Operational Safety Verification (71707) Units 1 and 2

   a. Plant Tours

      The inspector observed control room operations, reviewed applicable documentation, conducted discussions with control room operators, observed shift turnovers, and confirmed operability of instrumentation. Approximately twenty separate NRC tours were conducted of the control room. Each tour consisted of approximately two hours each. In addition, approximately twenty shift briefings and shift turnovers were audited. The inspector verified the operability of selected emergency systems, and verified compliance with Technical Specification (TS) Limiting Conditions for Operation (LCO). In specific, TS 4.2 and 4.8 were reviewed in depth to ensure operability. The inspector verified that Maintenance Work Request/Hold Orders had been submitted as required and that followup activities and prioritization of work was accomplished by the licensee. The following sources of information were reviewed to ensure the appropriate documentation of plant conditions:

Shift Engineer Log, Assistant Shift Engineer Log, AI-6, rev. 13.
Reactor Operators Log, AI-6, rev. 13
Auxiliary Unit Operator Daily Shift Log, AI-6, rev. 13
Temporary Alteration (TACF) Log, AI-9, rev. 25
Hold Order Log, AI-3, rev. 38
Operator Aid Log, AI-6, rev. 13
Configuration Control Log, AI-58, rev. 1
PRO Log, AI-12, rev. 3
SOI-67.1, rev. 34, Essential Raw Cooling Water System
SOI-72.1 rev. 29, Containment Spray Systems

Approximately 20 separate tours of TVA plant spaces were conducted. Tours were conducted in the diesel generator, auxiliary, control, and turbine buildings, and containment in order to observe plant equipment conditions (including potential fire hazards, fluid leaks, and excessive vibrations) and plant housekeeping/cleanliness conditions. The inspectors walked down appropriate portions of the following safety-related systems on Unit 1 and Unit 2 to verify operability and proper valve alignment:

RHR System
Control Room Ventilation System
Containment Hatch Covers

The reason containment hatch covers were toured, was that on August 26, 1987, D.C. Cook (a sister plant to Sequoyah run by Indiana and Michigan Electric) determined that the reactor coolant pump hatch covers on Unit 2 could fail in the event of a loss of coolant accident. The hatch covers provide access from the upper containment to each of the reactor coolant pump areas in the lower containment, and serve as a divider barrier between the upper and lower containment in the ice condenser design.

Indiana and Michigan Electric performed ultrasonic examination of the Unit 2 bolts holding the hatch covers in place. Their examinations determined that 25 of a total 74 bolts had been cut and welded (as opposed to 12-inch embedment in concrete) during the original installation. The inspector discussed the issue with Sequoyah personnel and the licensee issued Nuclear Experience Review 870727 to investigate. The licensee visually inspected the bolting on Unit 2 and UT inspected the bolting on Unit 1. No discrepancies were found. The inspector reviewed the documentation associated with these tests. Additionally, the inspector visually inspected approximately 25% of the bolting on Unit 2 hatch covers and found no descrepancies. This issue is closed.

No violations or deviations were identified.

b. Safeguards Inspection

The inspectors included a review of the licensee's physical security program. The performance of various shifts of the security force was observed in the conduct of daily activities including protected and vital area access controls; searching of personnel and packages; badge issuance and retrieval; patrols and compensatory posts; and escorting of visitors. The inspectors also observed protected area lighting, as well as protected and vital area barrie' integrity.

Additionally, The inspectors verified an interface between the security organization and operations or maintenance; witnessed firearms training and qualification; responded to bomb threats, fires, etc.; interviewed individuals with security concerns; participated in offsite support agencies visit to facility; inspected security during outages; reviewed licensee Security Event Reports; witnessed spent fuel shipment; visited central or secondary alarm station; witnessed power supply test; verified protection of Safeguards Information; verified onsite/offsite communication capabilities; and witnessed Corporate annual audit of site security program.

Security Incident PRO-1-87-438, was reviewed and its contents were discussed with NRC Region II Security Specialists.

No violations or deviations were identified.

c. Radiation Protection

The licensee's Health Physics (HP) and Radiological Protection Programs were observed. Included in this observation were radiation protection controls, use of protective clothing, and proper use of radiation protection instrumentation. The inspector verified implementation of radiation protection control. Specific work activities were monitored to ensure the activities were being conducted in accordance with applicable RWPs. Selected radiation protection instruments were verified operable and calibration frequencies were reviewed. The following occurrences were reviewed by the inspectors:

(1) On December 9, 1987, the licensee informed the resident staff that there was possible Co-60 contamination in the west septic tank of the plant sewage treatment system. This sewage treatment system is not designed to process radioactive waste and is not physically connected to any plant operating system. In September the licensee had discovered trace amounts of I-131 in this tank and had traced its origin back to an employee who had received an I-131 medical injection. The Sequoyah units have been in mode 5 for over 2 years and, therefore, because of the short halflife of I-131, plant systems do not presently contain detectable levels of I-131.

Following the detection of I-131, the licensee began sampling selected tanks on a routine basis. In late November, plant chemistry personnel detected indications of approximately 1 x 10-7 μ Curies/cc Co-60 in the west septic tank. Additional sampling indicated a maximum concentration of about 1 x 10-6 μ Curies/cc Co-60.

On December 8, 1987, per sewage plant operations procedure, sewage plant personnel pumped a partial load of sewage from the equalization tank into a contractor's truck and shipped it offsite. This equalization tank is the collection tank for the sewage prior to sending it to either the east or west septic tank. Shortly after the shipment left the site, plant chemistry personnel postulated that the shipment may contain Co-60, based on Co-60 indications in the west septic tank. A sample was taken from the equalization tank and Co-60 was indicated. The licensee immediately took the same sample to a separate licensee-owned offsite counting laboratory. No indication of Co-60 was detected in the same sample at the offsite counting laboratory. On December 10, 1987, the licensee detected Co-60 in higher concentrations on a laboratory count blotter used to transport samples within the onsite laboratory.

Additional samples were taken and no Co-60 was detected in the equalization tank or either septic tank. The licensee is reviewing their laboratory counting technique and the sewage disposal program. This event will be reviewed by an NRC Region II specialists.

(2) On December 12, 1987, while reviewing the RWP for entry into the Unit 2 upper containment, the inspector observed several workers frisking after exiting from the annulus. One worker, while frisking, accidentally dropped the pancake probe to the floor. He retrieved the probe and continued frisking without further checking to ensure the equipment was operating. Investigation by the inspector revealed the frisker to be damaged and not operating. The inspector pointed out the damaged frisker to the HP technician who immediately removed the frisker from service. The technician stated that frisking when exiting from the annulus was not absolutely required and no further followup action was initiated by the licensee. The inspector had no further questions.

5. Engineered Safety Features Walkdown (71710)

The operability and proper lineup of plant systems is addressed in special NRC team Inspection Reports 327,328/87-66 and 88-06.

6. Monthly Surveillance Observations (61726)

The inspector observed/reviewed the below listed TS required surveillance testing, special tests and postmaintenance/modification tests. The inspector verified that testing was performed in accordance with adequate procedures; that test instrumentation was calibrated; that LCOs were met; that test results met acceptance criteria requirements and were reviewed by personnel other than the individual directing the test; that deficiencies were identified, as appropriate, and that any deficiencies identified during the testing were properly reviewed and resolved by management personnel; and that system restoration was adequate. For completed tests, the inspector verified that testing frequencies were met and tests were performed by qualified individuals.

- SI-1, Surveillance Program; The inspector noted no deficiencies with the portions observed.

- SI-40, Centrifugal Charging Pump; The inspector noted no deficiencies with the portions observed.

- SI-265.0, Hydrostatic Testing; The inspector reviewed the completed Hydrostatic test package associated with WRs B240532, B240533, and B251598, which replaced three sections of ERCW piping that had been removed because of MIC-induced leakage. No problems were noted.

- SI-691, Spent Fuel Pit Pumps; The inspector noted no deficiencies with the portions of the test that were observed.

- SI 153.4, Revision 2, Test Requirements for the Electric Hydrogen Recombiner System, Unit 2; The purpose of this SI is to verify operability of the Unit 2 Hydrogen Recombiners. This SI is performed every 18 months and consists of a visual inspection, electrical checks and operation of the recombiners. The inspector accompanied the maintenance personnel during portions of the 2B-B recombiner attachment of this SI. The prerequisites require that SI-153.1, Periodic Calibration of Hydrogen Recombiner System Instrumentation, be completed; SI-151, Six Month Test Requirement on Electric Hydrogen Recombiner System, be performed just prior to this SI; and hold orders obtained prior to performing visual and resistance to ground checks. The hold order was properly placed and the Assistant Shift Engineer (ASE) signed for the completion of SI-151. However, no verfication signature for completion of SI-153.1, as detailed in the prerequisites, is required in the procedure and there is no clear understanding of what "just prior" means for SI-151. These are typical SI administrative deficiencies which are to be remedied in the second phase of the licensee's SI review program.

  The Maintenance electricians (ME) verified the hold order tags in place and checked the load side of the power supply de-energized. The ME then performed the electrical resistance to ground checks. Although the readings obtained met the acceptance criteria the behavior of the test instrument was suspect. The reported behavior was not similar to that observed on the 2A-A recombiner. The test

instrument was in calibration, however, it was a different instrument than the one used for the 2A-A checks. This situation was resolved by selecting a better suited instrument from those specified in Instrument Change Form (ICF) 88-093.

A loose bolt or support rod was identified inside the recombiner housing. The function of this item was unclear after reviewing the vendor manual. A work request was written by the ME. Two procedural discrepancies were identified. The first involved verifying the orifice at the bottom of the housing clear of obstruction per step 5.3.4 of SI-153.4. The ME did not understand what was required for the inspection of the orifice. Prior to conducting the 2B-B portion of the SI the ME had requested clarification of the step from the cognizant engineer and it was determined by the ME that the orifice was an exterior louver. However, subsequent review of the vendor manual by the NRC inspector, revealed an orifice plate, at the bottom of the heater banks. As a result of the NRC inspector's finding, the ME returned to the 2B-B recombiner to inspect the orifice plate clear of obstruction. The inspector did not observe this re-inspection.

The second discrepancy involved a visual inspection of the wiring from the ring terminal in the main junction box to each heater bank. The ring terminals in the main junction box are covered with Ray chem heat shrink and cannot be inspected without removal of the heat shrink. Removal of the Ray chem heat shrink was not performed during this SI. The inspection of the wiring to each heater bank is unclear as it would require disconnection of wires and removal of an additional panel for access. The procedure does not provide for disconnecting terminals and the actual connection to each heater bank was not inspected. The ME, have asked for a clarification.

Previously, on December 14, 1987, the ME had inspected the 2A-A recombiner orifice plate; a loose bolt, similar to the one in 2B-B, was found. A work request was written to resolve the loose bolt in recombiner 2A-A. The inspector reviewed Westinghouse Technical Bulletin (TB) NISD-TB-85-08, EHR Cable Inspection, provided by the licensee on December 14, 1987. The TB detailed the inspection recommendations of the vendor for the main power cable visual checks. The inspector believes visual inspection as specified in SI-153.1 and as observed during the conduct of this SI is sufficient to comply with the recommendations of the TB even though the visual inspection as outlined in the SI was not adequately implemented.

The licensee failed to adequately implement SI-153.4 in that the requirement of step 5.3.4 to visually verify the hydrogen recombiner orifice clear of obstructions was not properly conducted prior to signing the step as completed due to inadequate management supervision provided to the personnel performing the SI. This is identified as second example of VIO 327/328/87-76-02. A second example of this violation is located in paragraph 9.b of this report.

- STI-68, Refueling Water Storage Tank Heater Control Logic Functional Test; The inspector reviewed STI- 68, Unit 2, Revision 0. The scope of STI-68 was to verify the control logic of the Unit 2 Refueling Water Storage Tank (RWST) immersion heaters C and D, verify operation of the cold leg accumulator nitrogen header vent valve from the Auxiliary Control Room, and verify proper operation of the cold leg accumulator nitrogen header pressure control valve 2-PCV-63-58. The test procedural steps were clear and understandable. Acceptance criteria were stated for attributes to be tested. Inspectors concluded the test was adequate for performance as written. The following observations were discussed with the TVA staff:

  - Section 6.5 of the test relating to the nitrogen header vent valve and nitrogen accumulator header pressure control valve is not related to sections 6.1 through 6.4 of the the test. TVA staff indicated that section 6.5 was added for ease of review and approval.

  - In test section 3.5, the absence of a red indicator light is used as indication that RWST heaters are not energized. As the purpose of this special test is to verify proper heater logic, a more positive indication that heaters are not energized should be used. NRC inspectors inspected several panels in the plant including the Chemical and Volume Control System 480 volt boards and found the following red indicator lights had missing bulbs:

    480 Volt CVCS Control Board A

    Breaker 2D - RWST Immersion Heater A, Unit 1
    Breaker 3C - Caustic Batching Tank Mixer
    Breaker 2B - Evap Cond Drain Pump A

    480 Volt Aux Building Common MCC A

    Breaker 2D - Transformer A Exhaust Fan
    Breaker 2E - Transformer Exhaust Fan
    Breaker 5D - Pri Clg Wtr Loop Pmp B

    480 Volt Aux Vent Board 2A1-A

    Breaker 7A - 480 Bd Rm 2A Pzr Fan 2A1-A (bulb was out of socket loose in indicator).

    Inspectors gave the list of indicators with missing bulbs to the Shift Supervisor for correction.

  - Drawing 47W830-6, Mechanical Flow Diagram Waste Disposal System, Revision 8, contains an error in zone A-1. The continuation drawing for the high pressure nitrogen header that is referenced is in error. This was turned over to the test coordinator for transfer to drawing control. A drawing discrepancy form had

been submitted to drawing control previously by TVA personnel. The drawing error did not affect the outcome or adequacy of the STI and was not safety significant.

- SI-15, Emergency Core Cooling System Loop 4 RCS Isolation; This surveillance tested RHR Isolation valve operation and interlocks associated with RCS pressure on Loop 4. Test personnel conducted a walkdown prior to the test and found errors in switch numbering. This was corrected with an instruction change form prior to the test. No deficiencies were noted.

- SI-97.2, Calibration of Auxiliary Feed-Water Flow Rate for Remote Shutdown and Accident Monitoring; No deficiencies were noted during the performance of this surveillance.

- SI-144.2, Control Room Emergency Ventilation Test; The inspector reviewed the SI and observed portions of the performance of SI-144.2 on December 18, 1987, on the A train of the control room ventilation system. This test is performed at an 18 month interval and is intended to verify that the control room emergency ventilation system maintains the control room at a positive pressure of 1/8 inch of water gauge relative to the outside atmosphere at a cleanup flow rate of 4000 $\pm$ 10% cfm. Additionally, the SI is intended to verify that on a high radiation signal from the air intake stream, the system automatically diverts its inlet flow through the HEPA filters and charcoal absorber banks. The inspector reviewed ICF # 87-2386 and the associated USQD for SI-144.2. This change allowed performance of the test without smoke testing doors C53 and C60 until replaced under ECN L6860 as implemented by Work Plans 12602 and 12604. A new temporary boundary has been established in the Unit 2 stairwell to the control room which temporarily relocated the control room pressure boundary.

  The inspector observed the initiation of the train A control room isolation at the control room ventilation radiation monitor, RM-90-125, in the mechanical equipment room. The indicated differential pressure shown on the temporarily installed inclined manometer had stabilized at about .21 inches of water within 30 seconds of the initiation signal. However it was discovered during the test that the Unit 2 control room stairway door, C-25, was breached affecting results. With the door closed a less positive pressure resulted and the system failed to provide the required 1/8 inch of water positive pressure. The licensee is continuing with the performance of Special Test Instruction (STI) - 83, Rev. 0, Control Building Emergency Pressurization which consists of a ventilation flow balancing involving obtaining data which would allow determination of flow path and alignment changes necessary to resolve control room inleakage/pressurization problems. Additionally the licensee is investigating different corrective action plans.

- STI-83, Rev. 0, Control Building Emergency Pressure Ventilation Test; The inspector reviewed the incomplete package for STI-83. This ongoing test consists of a ventilation system flow balancing operation and obtaining data which would allow determination of flow path and alignment necessary to resolve inleakage and pressurization problems identified in CAQR SQP871226 Rev. 1. The licensee is planning on continuing with this test until the system will support successful completion of SI-144.2, Control Room Emergency Ventilation Test.

7. Monthly Maintenance Observations (62703)

   a. Station maintenance activities of safety-related systems and components were observed/reviewed to ascertain that they were conducted in accordance with approved procedures, regulatory guides, industry codes and standards, and in conformance with TS. In addition an individual review was conducted of each restart WR existing on the licensee's daily work list on a weekly basis. The total outstanding restart WR number was approximately three hundred on January 18, 1988. The licensee's maintenance planning activities were also observed on a daily basis.

   b. Attributes considered during a review of work requests (WRs)/work plans (WPs) were that: LCOs were met while components or systems were removed from service; redundant components were operable; approvals were obtained prior to initiating the work; activities were accomplished using approved procedures and were inspected as applicable; procedures used were adequate to control the activity; troubleshooting activities were controlled and the repair record accurately reflected what actually took place; functional testing and/or calibrations were performed prior to returning components or systems to service; quality control records were maintained; activities were accomplished by qualified personnel; parts and materials used were properly certified; radiological controls were implemented; QC hold points were established where required and were observed; fire prevention controls were implemented; outside contractor force activities were controlled in accordance with the approved Quality Assurance (QA) program; and housekeeping was actively pursued. The following WRs were reviewed:

      WR B-242678

      WR B-231942

      WR B-232394; The inspector observed portions of the removal of the A refueling water purification pump foundation and installation of new foundation. The old foundation was removed and repositioned to a lower location to allow alignment of a new pump/motor. The foundation grout was replaced per M&AI-17.

WR B-208086; The inspector followed the progress of the work associated with repairing the valve cover gasket leak on the Unit 2 RHR suction check valve, 2-63-502. The gasket leak had resulted in a buildup of boron precipitate around the cover bolting and the exterior surface of the valve body which could result in deterioration of the valve cover bolting. The work request had originally been planned to disassemble the 12-inch check valve, replace gasket, clean and inspect for any defective components and reassemble the check valve. This work as planned would require taking both trains of RHR out of service and was to be performed as part of a scheduled RHR outage. However, the WR was replanned to allow the cover bolting to be retorqued per MI-6.15, Rev. 14, General Procedure Tightening Bolted Joints. The check valve surfaces were cleaned and the valve cover was checked for further leakage at different RCS pressures up to and including 350 psig. No leakage was noted and the WR was placed on a planning hold pending final evaluation by the licensee.

c. The licensee has had control difficulties with clearance orders in the recent past. As a result, the licensee has recently implemented a clearance/tagging group to relieve the administrative burden on the ASE associated with these activities. The purpose of this group is to review work requests for adequacy and tagging requirements. The group is also responsible for hanging/clearing tags. The group appears to be a positive step. However, it is newly implemented and its effectiveness is yet to be determined. The group is presently active only on the day shift.

No violations or deviations were identified.

8. Licensee Event Report (LER) Followup (92700)

The following LERs were reviewed. The inspector verified that: reporting requirements had been met; causes had been identified; corrective actions appeared appropriate; generic applicability had been considered; the LER forms were complete; the licensee had reviewed the event; no unreviewed safety questions were involved; and no violations of regulations or Technical Specification conditions had been identified.

(Closed) LER 327/87-057; Control Room Isolation Initiated During Performance of a Special Test Instruction. During the performance of STI-30, DC High Voltage Test for Selected 1E Cable With Potential For Cable Pullby Damage For Conduit MC 1607A, the door to local junction box 0-L-450 was opened and a control room isolation was initiated. The junction box is the control room chlorine detector panel. The inspector reviewed the LER and the associated PRO (PRO 1-87-305). The licensee attempted to reproduce the event by opening the junction box. However, a recurrence of the event could not be initiated. The licensee also inspected the junction box for loose or damaged connections.

This LER is closed.

(Closed) LER 327/87-064: Improper Fit of Emergency Raw Cooling Water Flood Mode Spool Pieces Due to Minor Piping Orientation Changes. This involved an issue identified in the NRC Integrated Design Inspection (IDI) for the Emergency Raw Cooling Water (ERCW) system that found that flood mode spool pieces did not properly line up. The root cause appears to be a combination of system modifications and pipe stresses over the operating years. The licensee committed to performing a piping analysis by December 2, 1987, and to have the spool pieces modified to ensure proper fit by January 1, 1988. The licensee has reviewed all spool pieces and identified thirteen spool pieces for Units 2 and 3 spool pieces for Unit 1 that require modification. These have been modified per work procedure (WP) numbers 12735 (Unit 2) and 0082-01 (Unit 1). The piping analysis has been completed per calculation package N2-IDI-SPOOL-MISC. The long term corrective action to prevent future misalignment is being tracked by corporate commitment tracking system (CCTS) items NCO 870350 003 and NCO 87 0350 004. The licensees actions appear to be adequate.

This LER is closed.

(Closed) LER 327/87-069; Failure to Satisfy A Containment Spray Pump Surveillance Requirement. This issue involves SI-247.900, Engineered Safety Feature Response Time Verification, which did not require the complete containment spray pump control circuitry (i.e., pump start contact) to be included in the response time testing. Furthermore, the valve start time was not included in the overall calculation. The Licensee has committed to revising SI-247.900 by completing tests using instrument maintenance instruction IMI-99 response test RT-643 A and B, "Response Time Testing, Engineered Safety Features, Slave Relay K 643", prior to Mode 4 operation of Unit 2. The inspector verified that SI-247.900 had been revised. This was accomplished by instruction change form (ICF) 82-2140. IMI-99 and RT643B have been completed through the November 18, 1987 performance of SI-247.100B (Train B, Unit 2) and the Novmeber 12, 1987 performance of SI-247.100A, (Train A, Unit 1). A preliminary review of the test results indicate that the response time will meet TS. The Licensee's actions appear to be adequate.

This LER is closed.

(Closed) LER 327/87-066; Potential Loss of Component Cooling Water Inventory Due to Non-safety Related Design of the Pump Seal Leakage Return System. This issue involved a determination by the licensee that the component cooling water system (CCS) pump seal leakage could deplete the CCS water inventory in approximately 52 hours following a loss of coolant accident (LOCA), assuming two CCS pumps running in the same train and a loss of off-site power. A review of the appropriate abnormal operating instructions (AOI) indicated that the problem had not been addressed and procedures were not in place to mitigate the condition. The licensee has revised the system operating instructions (SOI)-55-OM27B-XA-55-27B-C (approved January 17, 1988), (SOI)-55-OM27B-XA-55-27B-D (approved January 17, 1988) and AOI-15 (approved January 18, 1988) and has committed to update the Final Safety Analysis Report (FSAR) to allow entry into the

auxiliary building to install the cross-over spool piece to connect the emergency raw cooling water (ERCW) to the CCS. This is being tracked by corporate commitment tracking system (CCTS) item NCO 87 0351001. The licensee's actions appear to be adequate.

This LER is closed.

(Closed) LER 327/87-048; Failure of Silicone Rubber Insulated Cables During High Voltage Testing. This issue involved testing of worst-case low voltage vertical drop cables subjected to a high-potential test of 10,800 volts. 3 out of 16 conductors had inadequate dielectric strength to withstand that voltage level." A review of the licensee's corporate commitment tracking system (CCTS) indicates the following:

(1) Failed silicone rubber cables have been replaced CCTS no. NCO 87 0322 001 (closed).

(2) Requirement to perform additional test and analysis has been committed to CCTS, no. NCO 87 0322 002.

In a telephone conversation with OSP Management January 19, 1988, it was stated that the NRC would accept the testing to date, for restart and operation, until the next refueling outage; at which time they would ask for sample cables to be pulled and verification tests implemented on the samples. The licensee has completed item 1 and is tracking item 2. The licensee's actions appear to be adequate.

This LER is closed.

(Closed) LER 327/87-010; Numerous Relays, Level Switches, Cycle Timers, Load Controllers, and Meters Have Not Been Calibrated Because They Were Not Identified in Procedures. This issue involved a lack of clearly defined departmental responsibilities that allowed components, requiring calibration on a routine basis, to be left out of the calibration cycle when department prepared calibration procedures are within their jurisdiction. The licensee has developed a series (13.1 and 13.2) of maintenance instructions (MIs) to cover all the components identified as a result of a Division of Nuclear Quality Assurance audit. The MIs and Preventive Maintenance (PM) procedures have been approved and are issued. The components required to be calibrated prior to restart have had issued and completed. The licensee's actions appear to be adequate.

This LER is closed.

(Closed) LER 327/87-045; An Inadequate Design Control Process Resulted in the Lack of Fuse Coordination Potentially Rendering the ERCW System Inoperable. This issue involves engineering change notice (ECN) L5637, dated February 17, 1983. This ECN added a fuse to each of the four essential raw cooling water system (ERCW) traveling screen speed switches to isolate the non-class 1-E speed switches from the class 1-E control circuit. An inadequate design control process did not require

calculations to be made to determine the type of fuses required to protect the class 1-E control system. The licensee has issued design change notice (DCN) 00078c, dated October 24, 1987, to document the fuse requirements necessary to ensure proper coordination between the class 1E and non-class 1E portions of the ERCW traveling screen electrical circuits and to install the proper fuses prior to mode 4 operation.

The inspector has reviewed DCN 00078c. The fuses in question have been evaluated by the licensee and replacement fuses have been installed per work package (WP) 12725. DCN 00078c will not be closed until the configuration control drawings (CCDs) have been issued; tentative issue date January 8, 1988. This DCN is in the process of being revised to allow substitution of fuses and to clarify issuance of CCDs. Sequoyah engineering project (SQEP)- 09, Change Review Checklist for Electrical Calculations, has been revised (approval date May 2, 1987) to require evaluation of each design change with electrical involvement. The modification group has completed the modifications to the equipment referenced in drawings 35W736-1, 35W736-3, 35W736-5, 35W736-7, 35W746-1, 35W746-2, 35W746-3, and 35W746-4. The DCN has been delivered to the DNE drawing unit (DDU) which has six months in which to update the drawings per Administrative Instruction (AI)-19, revision 25, paragraph 4.5.2. The licensee's actions appear to be adequate.

This LER is closed.

(Closed) LER 327/84-59; Failure To Comply With Appendix R of 10 CFR 50. This LER report was the result of inspections of various safety-related systems and details specifics of installations not in compliance with 10 CFR 50, Appendix R. The issues involved inadequate lighting for safe shutdown with the control room inaccessible, overflow of the Reactor Coolant Pump "pocket sump" upon a failure of an oil pump, and numerous instances of inadequate interaction separation between opposite train components. NRC inspection report 327,328/87-41 reported that the licensee, had in Enclosure 5 to their December 21, 1984, submittal, analyzed the Appendix R discrepancies identified in the subject LER and identified the appropriate corrective actions to resolve the discrepancies in Interaction Studies 22 through 27, 30, and 34. The licensee also requested and was granted a deviation to Appendix R, Section III.G which was necessary as a result of corrective actions. The NRC issued an SER granting the request on May 29, 1986. NRC inspection report 327,328/87-41 reported this LER open pending completion of 1B2-B and 2A2-A 480V shutdown board room cable tray water spray system installations. NRC inspectors reviewed this LER, NRC inspection report 327,328/87-41 and Work Plan 12285. Work Plan 12285 completed modifications to the fire protection system for 1B2-B and 2A2-A 480v shutdown board rooms.

This LER is closed.

(Closed) LER 327/86-039; Surveillance Requirements not Performed Because of Inadequate Procedures. This LER reported that procedures were not adequate to test all interlock functions of the reactor trip system

interlocks, reactor trip, P-4 permissive (Technical Specification 4.3.1.1.2, table 3.3-1, 22G) or to test the ice condenser inlet door position at the local panel during the functional test (Technical Specification 4.6.5.3.1). NRC Inspection Report 327, 328/87-65 closed the ice condenser inlet door portion of the LER and reported that a draft TS interpretation had been written to define the boundary of the total interlock function. The function in question testing of the main feedwater valve closure on low reactor coolant system average temperature with reactor trip) was being tested by procedure, but not by surveillance instruction. The draft Technical Specification interpretation, log No. 94, Revision 1 was approved by plant management and issued on November 12, 1987. The inspector reviewed the interpretation and concluded that action on this portion of the LER is adequate.

This LER is closed.

(Closed) LER 327/87-054; Technical Specification Required Action not Taken with Both Control Room Isolation Radiation Monitors Inoperable Due to Required Testing and a Design Deficiency. Control room radiation monitor RM-90-126 was taken out of service to perform SI-83, Channel Calibration for the Radiation Monitoring System. The monitor was taken out of service on August 10, 1987. The surveillance requirement was due on August 12, 1987. This date included the allowable extension for an 18 month surveillance. Radiation monitor RM-90-125 was also inoperable at that time due to high voltage special testing that was being conducted. With both trains inoperable, TS Limiting Condition for Operation (LCO) 3.3.3.1, action b, requires initiating and maintaining control room emergency ventilation in the recirculation mode of operation within one hour. This could not be done as both trains of the control room emergency ventilation were inoperable pending resolution of a design deficiency previously reported by LER 327/87-039. The inspector reviewed the LER and supporting documentation and interviewed TVA instrument maintenance and plant operating review staff (PORS) personnel to evaluate root cause and corrective action. The surveillance package for RM-90-126 was issued in March 1987, six months in advance of the surveillance due date. An extensive SI adequacy review during the same time period determined that SI-83, Channel Calibration for Radiation Monitoring System, required revision prior to performance of the surveillance. The SI was reissued and the actual surveillance commenced on July 23, 1987, and was completed on August 10, 1987, two days prior to the overdue date. Licensee action appeared adequate.

This LER is closed.

(Closed) LER 327/87-039, Control Room Emergency Ventilation System Single Failure Criteria Violated Due To A Design Error Which Could Result In Exceeding Allowable Dose To Operators. During performance of SI-144.2, Control Room Emergency Ventilation Test, the licensee identified a potential for a single failure of the main control room normal pressurization system to allow entry of unfiltered air into the control room during control room emergency ventilation after a control room

isolation. The discovery resulted in both trains of control room emergency ventilation being declared inoperable and entry into the associated TS action statement to suspend actions that would result in positive reactivity addition to the core. The LER reported that the root cause of this deficiency was under investigation along with corrective action and that a supplemental report would be submitted by December 4, 1987. Revision 1 to LER 87-039 was submitted on November 25, 1987, and revised the date to read "A supplement to this report will be submitted within 30 days following Unit 2 restart." The revised LER did not address system operability and planned action to restore the system to an operable status prior to Unit 2 restart.

The inspector reviewed the original LER, Revision 1 to the LER, all CAQRs, PROs, the Technical Specifications, and the FSAR. The inspector interviewed the TVA staff regarding present status of the Control Room Emergency Ventilation system, root cause determination, planned temporary corrective action prior to Unit 2 restart and permanent corrective action. The licensee's staff reported that the system was still inoperable. Corrective action had been taken to correct the single failure criteria problem reported in the LER 87-039, revision 0. However, the system had failed the surveillance (SI-144.2) due to an inability to meet minimum pressurization requirements. Attempts were in progress to improve weatherstripping and investigate ductwork improvements and then to retest the system in accordance with the surveillance instruction. Parallel action was in progress to pursue a change to the TS and FSAR based on a re-analysis of the dose to control room operators with an increased pressurization flow of outside air. This was being pursued in the event that improvements to the system did not result in an operable system. Plant Operating Review Staff indicated that a second revision to LER 87-039 was being prepared to report correction of the single failure criteria problem and corrective action planned to allow Unit 2 restart. The inspector discussed LER detail and content with licensing and PORS staff as the original LER reported the system inoperable and the revised LER reported that the root cause and corrective action would not be reported until 30 days after Unit 2 restart. Inspectors discussed the potential for needing prior NRC approval if changes were made which would affect the system design bases, assumptions, or analysis as detailed in the FSAR. This would be particularly true if pressurization was to be increased by increasing the introduction of outside air. The resolution of Control Room Emergency Ventilation system issues is discussed in paragraph 15 of this report.

This LER is closed.

(Closed) LER 327/87-003; Potential For Loss Of Containment Air Return Fan Due To A Design And Construction Deficiency. This report describes the possible failure of air return fans installed between upper and lower containment areas due to containment spray water draining into the fan intake opening following a design basis accident. This condition resulted from an inadequate design of curbing and the failure to install/reinstall curbing on the operating deck. An additional problem was identified in

that the containment air return fans would remove entrained water from the containment atmosphere and divert it outside the crane wall, thus lowering the reactor sump water level and potentially affecting suction head requirements of residual heat removal and containment spray pumps. Additional notification of these conditions and the licensee's evaluations and corrective actions were provided to the NRC by a letter dated July 8, 1987. Corrective actions included installation of curbing on the operating floor and in accumulator rooms 3 and 4, installing drain lines from accumulator rooms 3 and 4 to the inside of the crane wall, revision of the containment spray system design criteria and the FSAR, and revision of SI - 19 to provide for inspection of the removable curbing on the operating floor prior to entry into Mode 4.

The inspector reviewed the licensee's corrective actions to date and considers that the actions taken are adequate to support Unit 2 startup. The physical modifications have been completed on Unit 2 and SI-19 has been revised. Written commitments for design criteria and FSAR changes have been issued and have been verified to be present in the licensee's CCTS.

This LER is closed.

(Closed) LER 327/87-006; Operation of All Equipment Affected By Manual Safety Injection Test Was Not Verified Because Of Deficient Procedures. The licensee's review of SIs identified numerous equipment functions that were not being verified as required during tests involving manually initiated safety injection signals. The inspector reviewed revised SI-26.1A and SI-26.2A, Loss of Offsite Power With Safety Injection-Containment Isolation Test, and determined that the previously omitted verifications were now included. These function verifications have been satisfactorily completed in accordance with the revised SIs in June and October 1987. The licensee's extensive program to review all SIs and improve procedure preparation controls should prevent recurrence of this type of problem. The licensee's corrective actions appear to be adequate.

This LER is closed.

(Closed) LER 327/87-021, Revision 1; An Inoperable Fire Barrier Which Was Not Brought To The Attention Of The SE For One Day, Thus Causing A Violation Of TS On April 17, 1987. Fire door D-10 in diesel cell 1A-A was found closed, latched and inoperable with a broken self-closing mechanism by an industrial safety individual. An Assistant Unit Operator (AUO) wrote Work Request (WR)B223640 to repair the fire door on April 16, 1987, but failed to write a fire barrier breaching permit as required by Physical Security Instruction (PHYSI)-13, Fire. The industrial safety individual researched the issue further and decided a fire barrier breaching permit was required. He informed the SE and a permit was issued and the LCO was entered. The Licensee has counseled the AUO on the requirements of fire barrier breaching permits, PHYSI-13 was revised, and as of Revision 50 it is required that a person must be on an approved list

(Attachment F, Appendix A to PHYSI-13) to hold a breaching permit. The licensee's actions appear to be adequate.

This LER is closed.

(Closed) LER 327/87-026, Revision 1; Inadequate Radiological Control Coverage in the Transfer of Radioactive Waste Caused by Personnel Error Resulting in Technical Specification Violation. This issue involves a lack of sufficient planning to perform the intended operation per written instructions. Radiological Control Instruction (RCI)-13, Access Control of High Radiation Areas Where Radiation Intensity is Greater Than or Equal to 1000 mrem/hour, was in place during the time the incident took place. RCI-13 was issued to cover TS 6.12.2. The licensee has counseled the personnel involved, has instructed management to provide better planning and has revised RCI-13 to better define personnel responsibilities. A review of TS 6.12.2 and RCI-13 indicates that the licensee has acted in a responsible manner and the licensee's actions appear to be adequate.

This LER is closed.

(Closed) LER 327/87-029; Yearly Reporting of Environmental Impact for Plant Design and Operating Changes Not Made Due To A Lack of Proper Procedures. This issue involves a violation of TS 5.3.c (appendix B) which says that the licensee must maintain records of changes in facility design or operation that could affect an environmental impact. Further the licensee is to furnish to NRC an annual report containing any descriptions, analyses, interpretation, and evaluations of such changes, tests and experiments. The licensee is required to maintain records of changes in facility design or operation that could affect an environmental impact. Plant procedures and responsibility requirements were not sufficient to prevent this event. The licensee has revised Plant Administrative Instructions (AI)-9 and -19 to require a documented evaluation of changes, tests, and experiments to the facility which could affect the environment as required by TS 5.3.C (Appendix B). The chemistry group has completed a study of the facility design and operational changes since September 15, 1984, which could affect the surrounding environs. This study will be included in the next annual report. The licensee's actions appear to be acceptable.

This LER is closed.

(Closed) LER 327/87-036; Nonconforming Timing Relays Installed in Containment Air Return Fan Controls Due to Personnel Errors. Containment air return fan 1B-B failed to start, per TS acceptance criteria, during the performance of SI-2B, Containment Air Return Fans. A modified Agastat relay 7012PH was found to be installed instead of a 7012PF relay. A 7012PF (120VAC coil) relay was modified using a 7012PH (125 VDC coil) relay as a result of TACFs that were prepared to replace the 7012PH relays with the modified 7012PF relays. The manufacturer states that AC relays cannot be modified by using DC coils to obtain the required relay. The licensee has installed administrative controls to require approval of

TACFs by DNE prior to implementation. Agastat 7012PF relays were re-installed per Design Change Notice (DCN)-75 by Work Package (WP)-12707. The licensee's actions appear to be adequate.

This LER is closed.

(Closed) LER 327/87-053; Inadvertent Diesel Generator Start Caused By Personnel Inattention To Detail And Lack Of Procedural Clarity During Installation Of A Jumper. Diesel generator 1B-B was inadvertently started when the terminal screws were loosened to allow installation of a spade lug jumper. The procedure, IMI-99-RT699B, Response Time Testing Of Engineered Safety Feature Actuation-Safety Injection Signal With Station Blackout Units 1 And 2, stated that wires were not to be lifted. The licensee has revised IMI-99-RT699A and IMI-99-RT699B to caution the performer to use clip-on jumpers instead of lug type jumpers. Maintenance Personnel were counselled on the need to be attentive to details and to question unfamiliar procedural steps. The licensee's actions appear to be adequate.

This LER is closed.

(Closed) LER 327/86-061, Revision 1; Personnel Errors Resulting In Failure To Comply With An Action Statement For Limiting Condition for Operation on the Auxiliary Building Vent Radiation Monitor. The root cause was the cancellation of (SI)-470.5, Auxiliary Building Iodine Sampler Flow Estimation, by chemistry personnel in June 1986. SI-470.5 was to be reinstated by December 21, 1986. However, it was not approved until December 24, 1986, and not distributed until December 31, 1986, one day after the auxiliary building vent monitor was declared inoperable and a temporary sampler was installed. The licensee has reissued SI-470.5 and this procedure was used from December 31, 1987, until the radiation monitor 0-R-90-101 was made operable. The licensee's actions appear to be acceptable.

This LER is closed.

(Closed) LER 328/87-004; Containment Penetration Did Not Have Redundant Over Current Protection Because Of A Personnel Error During Construction. The cable supplying radiation monitor 2-RI-90-60B (non-safety related) was not terminated on the load side of fuse 2-FU2-90-60B. The radiation monitor supply cable was connected to the output of circuit breaker 12 which was the supply side of the fuse. The licensee has corrected the problem per work request (WR)-B117832 and verified all of the other circuit breaker/fuse combinations for penetrations are correct. The licensee's actions appear to be acceptable.

This LER is closed.

(Closed) LER 328/86-005; CVI Caused by EMI On A Radiation Monitor. Electric Magnetic Interference (EMI) caused the activation of radiation monitor 2-RM-90-106 which resulted in a Containment Ventilation Isolation

(CVI). No actual high radiation levels were found. The source of EMI was not identified; however, some heli-arc welding had taken place and it was surmised that welding was the cause. The licensee initiated Design Change Notice (DCN)-2276 to add capacitors to several radiation monitors to reduce EMI induced actuation. At the present time the licensee has stated that an ECN will be issued about the first of May and implemented by June. This DCN is currently in the licensce's TROI system for completion. The licensee's corrective action appears to be adequate.

This LER is closed.

(Open) LER 328/87-05; Train B Containment Ventilation Isolation Caused By An Unknown Source. This item involved an inadvertent containment ventilation isolation (CVI) which occurred on March 5, 1987. This CVI resulted from spurious signals from the channel containing Radiation Monitor RM-90-106. After Analysis of the event, the licensee was unable to determine a root cause. Therefore, the only corrective actions taken were to clear the CVI and return the radiation monitor and its associated isolation valves to normal service. During review of this LER, the inspector reviewed four additional CVIs which have occurred on (LER 328/87-008 R1) November 27, 1987, (LER 328/87-009) December 5, 1987, and (LER 328/87-010) December 21, 1987. All of these CVIs are associated with the channel containing RM-90-106, and root cause has not yet been determined for any of the listed events. This item remains open pending determination of root cause and completion of corrective action for all of the above listed events.

(Closed) LER 327/87-059; Alpha Contamination Checks Not Performed As Required Due to Inadequate Accountability for A Surveillance Requirement. This issue involved Alpha Contamination Checks not being performed on two sealed sources (283N and 662N) as required by TS Surveillance Requirement 4.7.10.2.b and Surveillance Instruction SI-56, "Byproduct Material Inventory and Sealed Source Leak Test." The licensee has determined root cause to be an inadvertently misfiled inventory card (283N) and less than adequate accountability for the performance of the surveillances. The two sources were checked for contamination and no detectable leakage was identified. An inventory of known byproduct material was completed and a survey of plant supervision was conducted to identify any other additional sealed sources which may have been missed. These actions are documented via TVA memo dated October 26, 1987, (RIMS S53-871026-863). Appropriate plant personnel were reminded of the importance of controlling byproduct material via TVA memo dated, October 22, 1987, RIMs S53-871022-837. In addition, in order to provide adequate accountability, responsibility for SI-56 is being transferred to the Radiological Control section. This transfer will be completed by March 1, 1988, and is being tracked by Corporate Commitment Tracking System control number NCO-87-0291-003. These actions taken by the licensee appear to be acceptable.

This LER is closed.

(Closed) LER 327/86-010; Diesel Generator Start from Inadvertent Removal of Fuse. During a fuse check the fuse was dislodged, de-energizing the EDG emergency start circuit, and starting the diesel generator. The licensee has added a caution statement to the appropriate sections of SOI82.2, Manual Rolling and Starting/Synchronizing Diesel Generator.

This LER is closed.

(Closed) LER 327/87-033; Adequate Design Calculation are Unavailable for the 125 VDC Vital Battery V Causing Potential Degrading of the System. This issue involved potential problems on the Sequoyah 125 volt direct current (VDC) vital battery V. These problems have been identified on a TVA significant condition report (SCR) SQNEEB8746 which is used by division of nuclear engineering (DNE) to identify problem type issues. These potential problems are due to the inability of the license to produce adequate design calculations to support the plant design bases. DNE, electrical engineering branch (EEB) has established a minimum set of essential calculations to support the design bases of Sequoyah nuclear plant. These calculations are specified in procedure method (PM) 86-02. Hold order 1606 on site operation instruction (SOI)-250.5, 125 Volt DC Battery Board No. V, will be in effect until the deficiencies associated with vital battery V have been resolved. The licensee's actions appear to be adequate.

This LER is closed.

(Closed) LER 327/86-025; Two Inadvertent Diesel Generator Starts During Walkdown Activities. During a fuse identification check a control fuse was pulled. When the fuse was replaced the normal feeder breaker to the 2B-B, 6.9 KV shutdown board tripped and the board transferred to the alternate power supply. Personnel attempted to transfer back to the normal supply without verifying the board deenergized, causing the diesel generator to start. Investigation revealed a component failure in the surge protection circuit. The component was replaced.

A second inadvertent start occurred during removal of a panel from battery board I for breaker name plate verification. The panel slipped and tripped a breaker which caused a loss of voltage signal, starting the diesel generator. Additional personnel were used to assist in removal of the panels to support the weight. A design change to add handles to the panels had been cancelled due to the low frequency of panel removal. This design change is being reevaluated by the licensee. The licensee's actions in both diesel generator starts appears to be adequate.

This LER is closed.

(Closed LER 327/86-022; Inadvertent Containment Ventilation Isolation (CVI) Caused by Radiation Monitor (RM) Spikes. On May 15, 1986, two CVIs occurred, on Unit 1 the CVI has been determined to be caused by Electric Magnetic Interference (EMI) generated by low flow/high vacuum alarm switch chatter on 1-RM-90-131 during purging initiation. On Unit 2 the CVI

occurred due to personnel error when an operator deenergized 2-RM-90-106 to prevent initiation due to EMI generated by welding. The licensee has submitted DCR2276 to add EMI suppression circuitry to the RMs (1RM90 106 and 131, 2RM90-106 and 131). The DCR is being tracked by the licensee's commitment tracking system and has a completion due date of May 1, 1988. Additional corrective action consisted of counselling the personnel involved in the Unit 2 CVI and issuing a training letter to all licensed personnel and STAs, dated June 25, 1986.

This LER is closed.

(Closed) LER 327/86-045; Fuse Short During Maintenance on ERCW Valve Caused Diesel Generator to Start. The cause of this event was personnel error. However, underlying the cause is a fuse coordination problem and a quality problem with the Bussmann MIS-5 fuses used to replace the KAZ actuators. Bussmann has indicated a potential design defect with the MIS-5 fuses and provided TVA with a method of testing to verify fuse operability. TVA committed to replacing all defective fuses (MIS-5) prior to restart. LER 87-030 has documented that Bussmann MIS-5 fuses have been replaced with Littlefuse FLAS-5 fuses. Discussions with the TVA personnel indicated all fuses have been replaced per ECN 6854.

The fuse coordination problem is caused by using the same size fuses for both feeder and branch circuits, rather than coordinating the fuse rating to smaller sizes nearest the load. TVA has committed to review all circuits for similar coordination problems prior to Unit 2 restart and to implement certain modifications after Unit 2 restart per the LER. The fuse coordination issue is being tracked as IFI 327,328/86-57-01.

This issue is closed.

(Closed) LER 327/86-049; Diesel Generator Start on Loss of 6.9 KV Shutdown Board Voltage Due to Personnel Error. Following maintenance, during restoration of electric system line up, an operator opened the wrong breaker (normal feeder breaker to the 1B-B 6.9 KV shutdown board) causing the diesel generators to start. The event was discussed with the operator and signs were added to the 6.9 KV unit boards to clearly identify them.

This item is closed.

(Closed) LER 327/87-051; Skid Mounted Valves Were Not Checked Every 31 Days. Technical Specifications (TSs) 4.7.3.a, Component Cooling Water System (CCS) and 4.7.4.a, Essential Raw Cooling Water (ERCW) System, requires valve position verification every 31 days for valves servicing safety related equipment. The licensee's Department of Nuclear Engineering (DNE) has identified all safety-related vendor supplied skid mounted equipment. Surveillance Instructions for CCS (SI-32) and ERCW (SI-33.1) were revised, revision 19 and 10 respectively, to add the skid mounted valves to the 31 day surveillance verification.The adequacy of the control room drawings is being followed under Violation 327, 328/87- -01.

This issue is closed.

(Closed) LER 327/86-042; Two Surveillance Requirements Not Performed Because of Inadequate Procedures. This issue involved: (1) several safety related pumps not being tested in accordance with 1977 ASME Boiler and Pressure Vessel Code, Section XI, Subsection IWP3110; and (2) the calibration of temperature sensors, thermocouples and resistance temperature detectors not being performed as required by TSs. All actions associated with this issue have previously been reviewed and determined acceptable in Inspection Report 327,328/87-36 with the exception of the licensee's Request for Relief from the above referenced ASME Section XI requirements. On October 23, 1987, the NRC Office of Special Projects issued a letter and safety evaluation granting the Relief requested. Therefore, all required actions pertaining to this issue are complete.

This LER is closed.

(Closed) LER 327/87-056; Combined Bypass Leakage Limit Was Potentially Violated Due to An Inadequate TS and Lack of a Specific Design Criteria. During a design review it was determined that 26 penetrations to the containment were not included in table 3.6-1 of the TS. The licensee has concluded, based on review of SI-158.1, Containment Isolation Valve Leak Rate Test, which included all but two of the subject valves, that a high degree of confidence exists that no violation of TS occurred. The 26 valves identified have been added to the SI-158.1 calculation for inclusion in the requirements of TS 3.6.1.2.c. The licensee has submitted a change to TS to include the above valves to Table 3.6.1, Secondary Containment Bypass Leakage Paths of TS. The TS change request (number TVA-SQN-TS87-33, dated September 17, 1987) changes Table 3.6.1 and defines "Bypass Leakage Paths to The Auxiliary Building". SI-158.1 is scheduled for completion prior to Unit 2 restart.

This LER is closed.

(Closed) LER 327/87-060; Inadvertent Start of Diesel Generator Caused By Inadequate Procedure During Shutdown Board Maintenance. While removing a tee wrap from wiring bundles an operator inadvertently shorted electrical studs in the board causing a diesel generator start. In the process of shutting down the diesel generator the 1A-A EDG tripped on overspeed. Special Maintenance Instructions SMI-1-202-3/4 were revised to cover the exposed electrical studs and to inhibit the start signal by placing a test switch, 43TC, in test. Investigation of the overspeed trip revealed RTV in the oil passage of the governor hydraulic actuator. The RTV had been applied to the electrical connectors to the actuator in an attempt to seal the cables against hydraulic oil intrusion. The actuators on all diesel generator governors were replaced and a caution sign placed near the governors that prohibits the use of RTV on the actuator.

This LER is closed.

(Closed) LER 327/86-41; Two Diesel Generator (DG) Starts Caused By A Personnel Error and Procedure Inadequacy During Fuse Replacement. On September 14, 1986, while replacing KAZ actuators, an inadvertent EDG start occurred when power was removed from the start logic relay panel. During this event a high temperature trip of the 1B-B EDG was actuated. The root cause of the inadvertent EDG start was the improper marking of the Main Control Room (MCR) drawing following a modification. The modification was properly documented on the Unit 2 drawings, but not on the Unit 1 drawings. The licensee's corrective action included the training of personnel on proper markup procedures, combining MCR common equipment drawings, and revising the procedure used for MCR temporary drawing changes. The high temperature trip of EDG 1B-B was determined to be caused by a misaligned cooling water valve lineup. The licensee's corrective action included issuing a training letter to all Unit Operators (UOs) and above, detailing procedures for removing EDG cooling water supply from service and verifying with the manufacturer that no damage occurred to the EDG.

The second EDG start occurred on September 17, 1986, while removing KAZ type fuses. An operator inadvertently pulled a power supply fuse for the EDG control emergency start circuit. The licensee's corrective action consisted of replacing the fuse and cautioning the operator. A review of the above corrective actions and discussions with licensee personnel indicate that actions accomplished are adequate to close this LER.

This LER is closed.

(Closed) LER 327/87-062; Unplanned Loss of Manhole 7B Missile Protection for B Train Diesel Generator Cables Due to Programmatic Deficiency in The Design Change Process. During the work process to install a modification it was determined that missile protection for manhole 7B, containing the Units 1 and 2 B train cables, was not maintained. The cause of this event has been determined to be a failure to properly identify manhole 7B as missile protection during the design change process, compounded by a failure of the approval and review process to adequately evaluate the effect of the construction activity on the plant operability status. As a result of this event both trains of diesel generators were inoperable at the same time. The A train EDGs were inoperable for other reasons at the time. It is noted that only missile protection was lost for the B train and in the event the diesel generators were required for power they would have functioned as designed. The inspector reviewed the report and determined that the event had been adequately resolved. The recommendations provided in the report, when properly implemented, will limit the probability for recurrence of this type event.

This LER is closed.

9. Event Followup (93702, 62703)

a. On December 1, 1987, maintenance was being performed on motor operated gag valve 2-MOV-87-23B, the gagging device for Upper Head

Injection flow control valve 2-FCV-87-23A. Part of this maintenance activity required that the gag motor rotation be checked. This required that the gag valve be placed in the mid position and the gag motor hand switch be placed in the open position to verify proper stem rotation. When the unit operator was requested to open the gag valve the operator went to open on the hydraulic operated valve 2-FCV-87-23A. The valve opened and resulted in mechanical damage to the gagging device. FCV-87-23A should not have opened with the gag valve in the mid-position. The hydraulic valve is interlocked with the gag valve such that it should only open when the gag valve is fully raised (open). The gag valve shut indicating light was also burned out such that the gag valve did not indicate that it was in the mid-position. The licensee is still investigating why the hydraulic valve opened when the gag valve was in the mid-position. The root cause of this event included poor communication and equipment malfunction. The licensee responded with prompt corrective action and in addition established a root cause analysis task force. Because of the licensee's corrective action, and because there was no material damage to the operability of the system or failure to implement procedures, a violation will not be issued.

b. On November 25, 1987, during maintenance on the Unit 2 Auxiliary Feedwater system (fit up check of the fire main supply spool piece), three to five gallons of water spilled from a blank flange which was removed. This maintenance was being performed within the boundaries established by a work authorization clearance. The source of the water was from a section of piping within the clearance boundary that was not adequately drained or depressurized during establishment of the clearance. Unrelated to the spool piece maintenance, but occurring simultaneously, Surveillance Instruction SI-247.100B, Response Time Testing of the Engineered Safety Features Actuations, caused the "A" SG level control valve (2-LCV-171) to open releasing the water trapped between the level control valve and the spool piece maintenance boundary valve. Three to five gallons of water flowed by gravity out of the spool piece flange.

TS 6.8.1 requires that written procedures be established, implemented and maintained covering the referenced activities.

Administrative procedure, AI-3, section 4.2.5 states, "The SE shall verify that pressure is zero and equipment drained prior to issuing a mechanical clearance."

Contrary to the above, the licensee failed to adequately drain and depressurize the portion of the system isolated to support the maintenance activity. This is considered a second example of VIO 327,328/87-76-02, which is addressed in paragraph 6 of this report. This item was determined not to meet the requirements for a licensee identified item because it was event identified and it was the direct result of a failure to adequately implement a procedure.

c. On July 1, 1987, the 1A-A EDG failed to meet acceptable speed and voltage parameters (56Hz vs 58.8 required). Swings on load reject reached 7960, while the maximum allowed was 7656. PRO 1-87-217 and SQP 871238 described the failure of SI-26.1A. The CAQR stated that the voltage regulator stability settings should be optimized to minimize overshooting, and that the 1A-A EDG did not meet Regulatory Guide 1.9 requirements. The CAQR determination was that the voltage regulators are overdamped (i.e., response time reduced beyond nominal).

CAQR SQP871238 was evaluated by PORS. It determined the affected EDG to be operable and the failed SI not a reportable event based on an evaluation that the voltage necessary to cause overcurrent relays on safety equipment to trip is approximately 8800 volts. The voltage observed during the test was 7960 volts and that previous runs of SI-26 had been run without incident. The licensee determined that the "intent" of R.G. 1.9 had been met in that the voltage required to cause over current was not reached.

On October 14, SI-26.2A was run on 2A-A EDG. It failed to meet the acceptable voltage of 5520 volts minimum prior to output breaker closure, but the breaker closed at 5200 volts. The breaker is interlocked to prevent closure until voltage is above 5520 volts, and the speed is above 850 rpm. Troubleshooting on October 21, found a voltage regulator printed circuit card diode installed backwards. PRO 2-87-94, written on October 27, identified a diode in the voltage regulator as being installed backward, causing the improper response of the voltage regulator (according to LER 327/87-70). This particular voltage regulator was installed by the licensee on November 8, 1986, in response to WR B208401, in order to troubleshoot the cause of reported voltage fluctuations on the 2A-A EDG. Post-maintenance testing only required that SI-7 be run to verify that the 2A-A EDG was operable. This test is inadequate to verify proper operation of the voltage regulator in that it does not test operation of the voltage regulator under rapid loading/rejection conditions. As a result of the inadequate post maintenance test, the defective voltage regulator was not discovered until the 18 month surveillance (SI-26.2A) was run in October 1987.

LER 87-70 was written on part 21 (as-found date is October 21). PRO 2-87-94, written on October 27, 1987, documents the troubleshooting findings of the maintenance workers who discovered the improperly installed diode. The LER states that the cause of the early closure is still under investigation. The use of an inadequate test is a violation of the TS 6.8.1 requirement to establish, implement, and maintain written procedures covering the plant activities including surveillance and test activities of safety related equipment. This item is identified as VIO 327,328/87-76-01.

10. Part 21 Reports

(Closed) 327,328/P2184-04, Incore and Excore Neutron Monitoring Fittings. This item was addressed in IE Information Notice 84-55. This item was also reviewed as part of the Sequoyah flux thimble guide tube ejection eve- The inspector reviewed the licensee's action with respect to IE No' 4-55 and found it to be acceptable.

This item is closed.

11. Licensee Actions on Previously Identified Inspection Finding (92701)

Inspector Followup Items (IFIs) are matters of concern to the inspector which are documented and tracked in inspection reports to allow further review and evaluation by the inspector. The following IFIs have been reviewed and evaluated by the inspector. The inspector has either resolved the concern identified, determined that the licensee has performed adequately in the area, and/or determined that actions taken by the licensee have resolved the concern.

(Closed) IFI 327,328/86-31-04; Review of An Inadvertent Control Room Isolation Following Completion of Licensee Investigation. This event was reported to the NRC in LER 327/86-022, Inadvertent Containment Ventilation Isolation (CVI) Caused by Radiation Monitor (RM) Spikes. This LER was reviewed and closed elsewhere in this report.

This item is closed.

(Closed) IFI 327/84-11-03; Flux Thimble Guide Tube Incident Review. This item was the original tracking item for the flux thimble guide tube incident. Subsequent to this identified item, IR 327,328/84-24 provided additional review of the incident and three violations were issued. The violations were closed in IR 327,328/85-27 and later reopened in IR 327, 328/87-43 due to concern for the adequacy of the root cause determination and the corrective actions. This IFI was reopened along with all other items associated with the incident in order to ensure that a clear auditable closure existed.

The relevant issues will be tracked by the final resolution of the violations issued in IR 327,328/84-24.

This item is closed.

(Closed) IFI 327/84-20-02; Verify That Detailed Procedures Have Been Established and Are Being Used. This issue involves the July 9, 1984 disassembly and repair of the B-B Auxiliary Air Compressor (AAC), which is part of the safety-related system auxiliary control air. The disassembly and repair were accomplished using a maintenance request (i.e., without detailed procedures). TS 6.8.1. requires that written procedures be established, implemented and maintained covering activities referenced in Appendix A of Regulatory Guide 1.32, Revision 2, February 1978. Paragraph 9 of Appendix A, Procedures for performing maintenance, requires that maintenance which can affect the performance of safety-related equipment

should be properly preplanned and performed in accordance with written procedures.

The inspector has contacted mechanical maintenance and received verbal conformation that the maintenance of safety-related equipment is being scheduled and approved procedures are being used. Maintenance instruction (MI)-10.36, Auxiliary Air Compressor Rebuild, has been reviewed and appears adequate. A review of corporate commitment tracking system (CCTS) #NCO085-0491-019 identified seventeen Critical Systems, Structures and Components (CSSC) components that required MIs. All seventeen MIs had been prepared and approved prior to October 1986. The licensee's action appears to adequate

This item is closed.

(Closed) IFI 327,328/86-60-07, Generic Fittings. This issue was included in the licensee's program for the control of mixed compression fittings in safety-related applications. Due to an omission, this item was not closed when the other outstanding generic fittings issues were closed in IR 327,328/87-65. The closure statement in 327,328/87-65, is also relevant to this item.

This item is closed.

(Closed) IFI 327,328/86-69-01; Diesel Generator Starting Air System Discrepancies. During a walkdown of the diesel generator starting air system the inspector identified numerous apparent discrepancies between drawing number 47W839-1, the System Operating Instruction (SOI)-82.1F, Starting Air System for Diesel Generator 1A-A, and the system as-built configuration. The inspector evaluated the licensee's response to 16 apparent discrepancies and 2 questions. The licensee responded adequately to the inspector's questions and no further issues were raised. The inspector verified the corrective action in each discrepancy through direct inspection. The items have been corrected in the procedures, in the field, and through red-line corrections to the control room primary drawings. Drawing Deviation forms exist to assure permanent drawing updates will be implemented. The licensee's corrective action is adequate.

This item is closed.

(Closed) IFI 327,328/86-28-18; Software Security in the SPDS Computer System. During the review of the Safety Parameter Display System (SPDS), the inspector expressed concern that access to the software programs should be further controlled. Since that time the licensee has implemented the following controls to guard against software security problems:

- The source code (language that is used to build/modify programs) was removed from the Technical Support Center (TSC) SPDS computer. The

code was placed on magnetic tape held under administrative control by Sequoyah Instrumentation Maintenance section.

- Password protection was provided to the TSC/SPDS Computer system.

- A keycard is required to access the computer.

- Restricted access was established to the computer room.

- Modifications of the TSC/SPDS software shall be in accordance with Sequoyah procedure SQE-13, Software Configuration Control, Rev. 4, and SQA-193, Quality Assurance for Computer Software Systems, Rev. 1.

Adequate measures appear to have been taken to protect against software security problems.

This item is closed.

(Closed) IFI 327/87-11-07; Verification of Corrective Actions for Element Reports CO 11103 - SQN Rev. 5 and CO 11203 - SQN Rev. 3. The inspector determined that the specific concerns addressed in this IFI were addressed by the safety evaluation report (SER) to be completed by NRC. This item is redundant to that closure process. Therefore, this IFI is closed and the resolution of these element report issues will be addressed by the issuance of the NRC SER on CO 11301 - SQN Rev. 3.

This IFI is closed.

(Open) Inspector Followup Item 327,328/86-69-05, Backup Power Supply For The Emergency Notification System (ENS). IE bulletin 80-15 required determination and verification of the status of the backup power supply to the ENS system. This IE bulletin, issued June 18, 1980, required a verification that the ENS telephone be connected to a supply of power that would remain operable upon a loss of site power. The bulletin required that facilities without a redundent supply make necessary modifications and provide such a connection. TVA responded to the NRC in letters, L. M. Mills/TVA to J. P. O'Reilly, August 29, 1980, and J. L. Cross/TVA to J. P. O'Reilly, September 5, 1980, and indicated that a Design Change Request had been submitted and that testing would be accomplished when the modification was complete. TVA sent a followup letter, L. M. Mills/TVA to J. P. O'Reily, March 3, 1981, indicating the modification was expected to be complete at Sequoyan by April 21, 1981. Between March 1981 and November 1986 (almost 6 years), Sequoyah records and licensing could locate no correspondence to indicate followup on this item. TVA letter R. Gridley/TVA to J. Nelson Grace, November 5, 1986, referenced an August 27, 1986 telephone call between TVA and R. Priebe/NRC which indicated that the item need not be resolved before startup, provided a schedule was being developed to correct the problem. The TVA letter indicated that the design work would be completed by February 1987, and that testing would be completed within four weeks once startup issues no longer dictate the schedule. TVA letter R. Gridley/TVA to USNRC document

control desk, May 15 1987, noted that TVA has been unable to meet the February commitment due to allocation of resources to startup items. The letter indicated a new schedule for design completion by the end of January 1988. Modification completion (including required testing) was not addressed in the letter.

The inspector discussed this item with communications personnel, Nuclear Engineering Branch personnel and licensing personnel in an attempt to determine if a schedule existed for the modification (start to completion). No schedule was located. The inspector concluded that this item is a long-term outstanding bulletin requirement that has not been implemented in a timely manner. Lack of documentation and rotation of personnel involved, added difficulty in assessing the cause of the delays. This item will remain open pending further investigation and corrective action by TVA. This is not a startup item and this item is being reviewed for shutdown plant consideration pending licensee corrective action.

(Closed) IFI 328/81-12-01; No Planned Testing To Verify The Operation Of Each Of The PORV Motor Operated Block Valves During A Blowdown Of Pressurizer Pressure Using The PORV At Normal Operating Temperature And Pressure. TS 3-4.3.2 requires that the power operated relief valves (PORVs) be stroked every 18 months and the blocking valves are to be cycled full stroke every 92 days with the PORVs closed (the block valves are not cycled against differential pressure). This is done to verify the operability of the overpressure protection system. The PORVs and blocking valves are stroked per SI-166.40 once very 18 months. The blocking valves are cycled open per SI-116.1.1 once every 92 days. The inspector was not able to identify a legal requirement or a licensee commitment intended to cycle the blocking valves against normal RCS operating temperature and pressure (blowdown). The licensee has decided to forego testing of the blocking valves against temperature and pressure (blowdown) for the following reasons:

- The valves were purchased from the vendor with the requirement that they operate during operating temperatures and pressures. The licensee expects these valves to operate under the conditions for which they were designed.

- The licensee does not wish to breach the primary pressure boundary at normal operating temperature and pressure without adequate justification.

The licensee's actions appear to be adequate, this item is closed.

(Closed) Item 327-328/86-17-04; Pre-startup Training Commitments. This issue involved a letter from NRC region II to TVA subject: NRC inspection report 50-327/328/86-17 and accepting the TVA commitments as stated in TVA's response to the inspection report dated May 28, 1986. The commitments are as follows:

- By startup, all operators will be provided with a training letter on major plant modifications completed during the extended outage.

- By Unit 1 startup, the Unit 1 operators will be provided with a training letter regarding precautions to be taken on a unit startup with high boron concentration and potential positive moderator temperature coefficient.

- By Unit 2 startup TVA will provide additional startup simulator training for licensed employees who will be involved in the Unit 2 startup.

A review of the Corporate Commitment Tracking System (CCTS) indicates that the above mentioned items were committed to in the CCTS. The items required for Unit 2 restart are completed or are being tracked by CCTS. Those items required for unit 1 restart are being tracked by CCTS. The licensees actions appear to be adequate.

This item is closed.

(Closed) IFI 327,328/86-46-04; RHR Pump Vent Arrangement. This issue involved the post-LOCA swapover of the RHR pumps to the containment recirculation sump from the refueling water storage tank. The pumps are vulnerable to loss of pump suction due to air or steam binding if the pump swapover is not timed appropriately. This issue was discussed with Office of Special Projects, Projects Division management (A. Marinos, E. McKenna) the week of January 5, 1988. It was determined that the swapover concerns were specific to plants that did not have automatic swapover features on the RHR pumps. Sequoyah has automatic swapover of the RHR pumps based on refueling water storage tank and containment recirculation sump permissive switches and is not subject to this specific concern. Office of Special Project memo (A. Marino/ K. Barr) dated January 20, 1988, confirms that the concern is not specific to Sequoyah. The generic issue is being forwarded to NRR for review.

This item is closed.

12. Design Change Process Waivers (37700)

To correct past problems in the design change process TVA committed in section II.3.0 of the Sequoyah Nuclear Performance Plan to implement a transitional design change program prior to restart, and a final program after restart. Two previous inspections documented in inspection reports 327,328/87-42 and 327,328/86-63, found the transitional design change program to be acceptable.

During the change from the old method of processing engineering change notices (ECNs) to the transitional design change program, a number of waivers were granted to allow completion of in-process ECNs under the old program. The old program was covered by Administrative Instruction SQEP-AI-11, Handling of ECNs. SQEP-13, Procedure for Transitional Design

Control, covered the transitional program. Unitized ECNs and Configuration Control Drawings (CCDs) were required under the transitional program. The Project Engineer was required to approve waivers to the transitional program, which were usually granted in order to keep some work activities ongoing. For example, waivers were granted to allow use of the existing drawings when CCDs did not exist. A total of 90 ECNs were processed using waivers. The usage of waivers was stopped in March 1987. Revision 5 of SQEP-13, issued in July 1987, removed the allowance for the Project Engineer to grant waivers to allow performance of modification design work under SQEP-AI-11.

In order to document that adequate design control was maintained for those ECNs worked as a waiver to SQEP-13, a verification checklist was included in each ECN package. Each checklist is signed by the preparer, a reviewer, the principal engineer, and the lead engineer. The checklist verifies the following six items:

- Appropriate design criteria/design input requirements have been reviewed including the restart design basis document.
- Appropriate configuration control drawings or the latest revision to the as-constructed drawings were reviewed and approved as input for design.
- All previously approved field change requests (FCRs) that are not yet incorporated into design drawing/documents were reviewed and effects of these FCRs were accounted for in design.
- All previously designed but unimplemented ECNs were reviewed, and effects of these ECNs were considered in design.
- Any outstanding L-Design Change Requests (DCRs) were reviewed and the effects of these L-DCRs were included in design.
- All existing Temporary Alteration Control Forms in affected area were reviewed and determined not to affect the design.

The inspector reviewed four ECNs (L6828, L6844, L6886, L6888) that had been processed using waivers. ECNs L6844 and L6888 contained the completed checklist in the ECN package and the other two did not. For ECN L6844 the checklist was added as revision one to the ECN package. For ECN L6888 the checklist was part of the original ECN revision zero. With the completion of the checklist review and incorporating the checklist into the ECN package, the review of ECNs processed with waivers will be adequately documented to indicate that design control was maintained.

Discussions with engineering personnel indicated that the review for the checklist was time consuming but no conditions adverse to quality had been identified. Also, the inspector reviewed two computer printouts listing all the CAQRs concerning ECNs and none relating to the waiver process were found.

The inspector reviewed two audits conducted by the engineering assurance (EA) oversight review group over the Design Baseline Verification Program (DBVP). The first audit (EA-OR-001) was completed April 27, 1987. For the transitional design change control program two action items (Q-09 and Q-11) were identified. A supplemental report (EA-OR-001-S) was completed September 28, 1987. Q-09 contained recommendations for such items as various additional signoff requirements in the ECN review process, and identifying who approves ECNs. EA concluded that the recommendations were implemented in revision 5 to SQEP-13 issued July 7, 1987, and considered this item closed. Q-11 concerned the usage of waivers to bypass implementation of SQEP-13. The waivers were stopped in March 1987, and revision 5 to the procedure removed the waiver clause. The EA team considered the SQEP-13 implementation using waivers closed.

In conclusion, the inspector found that the usage of ECN waivers has stopped. ECNs that were processed using waivers are receiving an additional review using a checklist to verify that adequate design control was maintained. These reviews are being adequately documented as part of the ECN package and no quality concerns have been identified.

This issue is closed.

13. Cold Overpressure Protection Review (TI 2500/19)

Temporary Instruction 2500/19, "Inspection Of Licensee's Actions Taken To Implement Unresolved Safety Issue (USI) A-26: Reactor Vessel Pressure Transient Protection For Pressurized Water Reactors".

In order to verify the acceptability of the licensee resolution of USI A-26, the inspector, using the guidance of TI 2500/19, reviewed licensee generated documents described below. All of the items requiring verification are not addressed in this report. However, several items have been verified and a number of items requiring further staff review and/or resolution are identified below.

Compliance With Technical Specification Requirements

There is currently no TS pertaining to the Reactor Coolant System (RCS) Cold Overpressurization Mitigation System (COMS). Therefore, there are no limiting conditions for operation or surveillance requirements for COMS as a system. However, testing of various components within the system is being performed and is discussed below. Testing may not be all encompassing, however. Subsequent to conversations between licensee personnel and NRC staff, the licensee has agreed to initiate, and submit for NRC review, a TS for COMS. The licensee's position, however, is that this need not be accomplished prior to restart of Unit 2. This position is currently being reviewed by NRC-OSP-Headquarters.

System Redundancy and Independence

COMS consists of two redundant systems; one powered from Train A and the other from Train B. However, the two systems are not totally independent, in that an interlock exists between the two trains. The function of this interlock is to prohibit actuation of COMS when RCS temperature is above 350 degrees. The acceptability of this condition needs to be reviewed by NRC-OSP Headquarters.

Administrative Controls and Procedures

The licensee has initiated several administrative controls which are intended to: (1) minimize the temperature differentials between the steam generators and reactor vessel while in a water-solid condition; (2) restrict the number of safety injection pumps to zero and centrifugal charging pumps to one when the RCS is in a low temperature condition; and (3) alert operators to the automatic operation of COMS. These administrative controls are contained in General Operating Instruction (GOI)-1, Plant Startup From Cold Shutdown To Hot Standby, rev. 61 and GOI-3, Plant Shutdown From Minimum Load To Cold Shutdown, rev. 33.

Surveillance

As stated above, there are no TS surveillance requirements for COMS. However, because the RCS power operated relief valves (PORVs) and block valves are utilized for COMS, as well as utilized to alleviate high RCS pressure at power, these valves are addressed under TS surveillance requirements 4.4.3.2.1 and 4.4.3.2.2. These TSs require that the PORV be stroked every 18 months and the block valves every 92 days. Surveillance Instructions SI-166.40, Remote Valve Position Indication, and SI-166.1.1, rev. 0, Full Stroking of Category "A" and "B" Valves Required in All Modes, implement these requirements. In addition, channel calibration and operability verification of COMS has been accomplished using instrument maintenance instructions (IMI)-99 CC12.1, RCS Cold Overpressure Protection System Verification - Unit 1 and IMI-99 CC12.2, RCS Cold Overpressure Protection System Verification - Unit 2. In anticipation of the previously referenced to-be-issued TS, the licensee recently performed a review of these IMIs to assure their completeness and technical adequacy. This review revealed that these IMIs were to be performed once every 18 months during modes 4, 5, or 6, and testing of the final actuation devices (separation relays) was not required. As a result of these findings, the licensee has revised the IMIs to include provisions for performance during modes 1, 2, and 3, and to include the testing of the final actuation devices. These revisions were accomplished with Instruction Change Forms 87-2388 and 87-2389, and were approved on December 18, 1987. The inspector has reviewed the revised IMIs and determined that they adequately test the COMS. The inspector did, however, identify one area in which these IMIs need further enhancement. The PORVs in COMS are target rock valves, similar in design and operation to those installed in the Reactor Vessel Head Vent (RVHV) system. As noted in a previous inspection report (327,328/87-65), these target rock valves have exhibited the tendency to inadvertently open to 60% of full open when the block valves are opened at RCS system operating pressure, and then close within

about 5 seconds. The inspector noted that the above IMIs do not contain caution statements to alert the operator to the possibility of this occurrence.

Observance of Branch Technical Position RSB 5-2

The inspector's review of the Sequoyah COMS system also revealed the following items as they pertain to referenced paragraphs of Branch Technical Position RSB 5-2:

- Paragraph B.3; COMS is a fully automatic system and is not manually enabled. Therefore, an alarm to alert the operator to enable the system is not necessary. Positive indication is not provided to indicate when the system is enabled. However, annunciation is provided to indicate when the protective action is initiated.

- Paragraph B.4.a; Although the above referenced functional testing procedures now contain provisions to allow performance in modes 1, 2, and 3, there is currently no requirement for such testing to be performed prior to each shutdown.

- Paragraph B.7; COMS does not depend on the availability of offsite power to function. Train A is supplied from battery board I for Unit 1 and battery board III for Unit 2. Train B is supplied from battery board II for Unit 1 and battery board IV for Unit 2.

As stated above, the NRC will review the adequacy of the system design as part of the TS approval process.

14. Review of Recent Management Changes

The inspector reviewed certain recent management changes to determine whether they met appropriate requirements and commitments. TS 6.3.1 states that, "Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions and the supplemental requirements specified in Section A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensee's, except for the Site Radiological Control Superintendent who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975."

ANSI N18.1-1971 states that, at the time of initial core loading or appointment to the active position, the plant manager shall have ten years of responsible power plant experience, of which a minimum of three years shall be nuclear power plant experience. A maximum of four years of the remaining seven years of experience may be fulfilled by academic training on a one-for-one time basis. To be acceptable, this academic training shall be in an engineering or scientific field generally associated with power production. The plant manager shall have acquired the experience and training normally required for examination by the NRC for a Senior Reactor Operator's License whether or not the examination is taken.

In an organization which includes one or more persons who are designated as principal alternates for the plant manager and who meet the nuclear power plant experience and NRC examination requirements established for the plant manager, the requirements of the plant manager may be reduced, such that only one of his ten years of experience need to be nuclear power plant experience and he need not be eligible for NRC examination.

At least one of the persons filling positions delineated in 4.2.1 should have a recognized baccalaureate or higher degree in an engineering or scientific field generally associated with power production.

a. The inspector reviewed the qualifications of the new plant manager and determined that he has met the above above criteria in the following manner:

   (1) He has over 21 years power plant experience.

   (2) All of the above experience was Nuclear Power Plant experience.

   (3) This organization identifies two individuals with signature responsibility for the plant manager's office and could be considered principal alternates with respect to the site organization approved by the NRC in section 6 of the TS. One individual, the operations superintendent, meets the following criteria:

      (a) He has approximately 20 years responsible power plant experience.

      (b) All of the above experience was Nuclear Power Plant experience.

      (c) He received a PWR Licence Certification in 1982, based on the licensee's SRO equivalency training for managers and engineers.

      (d) He holds a recognized baccalaureate degree in engineering.

      (e) His training met the requirements specified in Section A and C of Enclosure 1 of the March 28, 1980, NRC letter to all Licensees.

   Considering the above, the inspector determined that the plant manager's office currently meets TS 6.3.1.

b. The inspector reviewed the qualifications of the new Assistant Manager of Nuclear Power and determined that he has met the following criteria:

   (1) He has had over 15 years responsible power plant experience.

(2) All of the above experience was Nuclear Power Plant experience.

(3) He holds a recognized masters degree in Nuclear Engineering.

This manager does not meet the requirements for examination for a Senior Reactor Operation License and has not completed the licensee's SRO equivalency training for managers and engineers.

This position is not a line function and therefore is not covered under TS 6.3.1. The inspector's review, of this position, was completed to determine the depth of the qualifications of those persons acting as advisors to the site director only.

c. ANSI N18.1-1971 states that at the time of initial core loading or appointment to the active position, the maintenance manager shall have a minimum of seven years of responsible power plant experience or applicable industrial experience, a minimum of one year of which shall be nuclear power plant experience. A maximum of two years of the remaining six years of power plant or industrial experience may be fulfilled by satisfactory completion of academic or related technical training on a one-for-one time basis. He further should have nondestructive testing familiarity, craft knowledge, and an understanding of electrical, pressure vessel, and piping codes.

The inspector reviewed the qualifications of the new Corporate Maintenance Manager and determined that he has met the above criteria in the following manner:

(1) He has over 33 years of responsible power plant experience.

(2) All of the above experience was power plant experience.

(3) He has the following additional qualifications:

(a) Non-destructive testing familiarity.

(b) Craft knowledge.

(c) Understanding of electrical, pressure vessel, and piping codes.

d. ANSI N18.1-1971 states that, "For those nuclear power plants having a manager designated to supervise the on site professional - technical groups, such managers, at the time of initial core loading or appointment to the active position, should have a minimum of eight years in responsible positions, of which one year shall be nuclear power plant experience. A maximum of four years of the remaining seven years of experience should be fulfilled by satisfactory completion of academic training.

The inspector reviewed the qualifications of the new Project Engineer and determined that he has met the above criteria in the following manner:

(1) He has over 22 years in a responsible industry position.

(2) Eighteen years of the above experience was nuclear power plant experience.

(3) He holds a recognized baccalaureate and masters degree in engineering.

Considering the above, the inspector determined that the Project Engineer's qualifications currently meet TS 6.3.1.

The inspector had no further questions. No violations or deviations were noted.

15. Control Room Ventilation System

The licensee through the performance of SI-144.2, Control Room Emergency Ventilation (CREV) Test, demonstrated the operability of the CREV three successive times for each train of CREV. The inspectors observed/reviewed each of these performances. In addition the following were reviewed:

STI-83, Rev. 0, Control Building Emergency Pressure Ventilation Test
CAQR SQP 87 1226
DCN X000 91A, Damper 31A modification
DCN X000 51A, Single failure modification
DCN X000 119, Damper 31A-271 modification
DCN X000 129, Mechanical equipment room related work
Drawing 47W866 Series

Issues still exist on the general control building flow balancing. However, these issues are not heatup or startup related and do not appear to affect the operability of the CREV. This issue is also discussed under LER 327/87-039. This issue is closed.

(Closed) LII 328/80-09-03; Main Control Room Habitability.

Preoperational test (PT 333, R2 and PT 333, R3) indicated excessive in leakage to the Control Room (CR) during isolation conditions. Modifications were accomplished to correct the identified deficiencies and reported to NRC via reports dated April 3, 1980, December 15, 1980, January 30, 1981, and April 8, 1982. Acceptable in-leakage was reported as achieved in the December 15, 1980 report. Additional modification was planned to further reduce air leakage to increase the margin of safety. The modification accomplished to reduce in-leakage installed blanks in the smoke removal fan ducts. Smoke removal will be accomplished by alternate methods (portable equipment).

This item is closed.

16. Followup on Operational Readiness Inspection Findings (IR 327,328/87--73)

Prior to recommending the approval for Sequoyah Unit 2 heatup the items identified as open in IR 327,328/87-73 were resolved to the satisfaction of the NRC by TVA. These resolutions, in some cases completely closed the item and in some cases resolved the technical issues. The followup inspection and item status is provided below:

a. (Closed) URI 327,328/87-73-01; Problems associated with Containment Sump Level Transmitter. The issue identified involved the need to instruct the control room operators that the containment sump level transmitters had been re-evaluated and found to be operable and should be used in accordance with Emergency Operating Procedures. This was a change to the original plan which included changing procedures and training operators to not use these indicators for the first 6 hours following a loss of coolant accident (LOCA). The inspector verified that procedure IP-6 was revised and a letter was sent to each operator and to the training center to indicate that the original compensatory measure was no longer required.

b. (Closed) URI 327,328/87-73-02; The use of administrative controls to restrict cooling water inlet temperature to safety related coolers. This issue involved air flow problems associated with several safety-related room/space coolers with the auxiliary feed water (AFW) / boric acid transfer (BAT) pump area cooler being the most restrictive. Specifically, TVA had determined that in related equipment they would have to administratively control the ERCW inlet temperature to the coolers to 72°F (most limiting) or take TS limiting conditions for operation (LCO) actions for equipment cooled by these coolers. This approach was discussed with OSP-HQ staff who agreed that the room/area coolers were attendant equipment for the controls could be used as long as TS actions were taken and documented on Form (ICF) 88-0057 to SI-3, "Daily, Weekly and Monthly Logs". When the ERCW inlet temperature reaches 72°F (most limiting), equipment served by the degraded coolers must be declared inoperable and the TS LCO action statement must be entered. It should be noted that history has shown that the Tennessee River will reach this temperature between May and October and a plant shutdown may be required if TVA cannot repair the degraded coolers.

c. (Closed) URI 327,328/87-73-03; Validation of Procedures Implementing Compensatory Measures; Operator Training and Use of Emergency Plan Implementing Procedures (IP) To Carry Out Compensatory Actions. A review of compensatory measures (CMs) by TVA was made to identify implementing procedures which, due to the CM, required validation in the form of a plant walk down of the procedure. Results of the validation are included below. As a result of this review, it was determined that the following procedures needed to be validated:

- IP-6; Activation and Operation of the Technical Support Center. It was determined that more detail was needed in the instructions for placing Lower Compartment Coolers in service in the event of the loss of a train of power. As a result of the procedure validation, a change was initiated to the procedure to include the specific valves which must be operated for loss of a given train of power. A diagram was also added showing the location of these valves in the annulus.

- AOI-8; Tornado Watch/Warning. This procedure contained two checklists; one for verifying tornado doors closed and one for blocking doors open. Consequently, a the operator would have to jump back and forth between checklists and waste time in retracing steps. As a result of the validation, a procedure change was initiated to combine the two checklists into one checklist. The doors on the checklist were arranged sequentially in such a manner that the operator could complete the checklist with minimal back tracking. In addition, detailed locations of dampers were determined to be needed in the procedure. These were also included in the procedure revision.

- AOI-15; Loss of Component Cooling Water (CCS). A review of this procedure revealed that instructions for realigning spent fuel pit cooling from A train to B train CCS did not contain sufficient detail. A procedure change was initiated to include the locations of valves requiring manual operation or local verification of position.

The following changes to procedures were reviewed by the inspector and found to be acceptable:

- IP-6, revision 19; Activation and Operation of Technical Support Center. This revision implemented recommendations from the procedure validation walkdown as to operation of lower compartment coolers and the use of containment sump level transmitter and effects that environmental conditions may have on their accuracies.

- AOI-27, revision 9; Control Rooms Inaccessibility. This procedure change addressed the requirement to remove power fuses to the main steam isolation bypass valves in the event of control room abandonment.

- AOI-15, revision 9; Loss of Component Cooling Water. This revision implemented recommendations from the procedure validation walkdown as to the aligning of "B" train CCS to spent fuel pit heat exchangers upon loss of "A" train power.

- AOI-8, revision 16; Tornado Watch/Warning. This revision implemented recommendations from the procedure validation

walkdown as to the operation of doors and dampers needed to protect the building during a tornado at the site.

A followup review of operator training on the use of compensatory measures and the need to follow radiological emergency plan implementing procedures (IP) was conducted on January 19, 1988. After the inspectors expressed their concerns with specific training of operators on compensatory measures the licensee conducted augmented training which consisted of the following:

(1) Instructions were provided through a letter from the Operations Supervisor to all licensed operators dated January 18, 1988, regarding the use of IPs in general and the mandatory use of IP-6 for compensatory measures in specific.

(2) Approximately 30-40 minutes of detailed training during shift turnover was provided to all of the 6 operating crews and commitments to review the need to include this training in operator requalification training was discussed with Plant management.

d. (Closed) URI 328/87-73-04
(Open) URI 327/87-73-04; Need to Have 1 Additional ASE for Two Unit Operation. This issue was discussed in IR 327,328/87-73 and was addressed in the transmittal letter to that report. The purpose of addressing this issue in this report is to document that an assessment for Unit 2 heat up was made by the NRC and the outstanding issue did not impact the hold point release decision.

e. (Open) Violation 327,328/87-73-05; Failure to Perform Adequate 10 CFR 50.59 Safety Evaluations for Modifications to the Facility Which Involved Compensatory Measures.

The TVA's safety evolutions (USQD) were developed and provided to the NRC for their review prior to plant heatup. The inspector's review is provided below. However, this item will remain open pending issuance of the Notice of Violation and receipt of TVA's response. The inspector performed followup reviews of the unreviewed safety question determinations (USQDs) as each pertains to the applicable compensatory measures (CMs) listed below. Each was reviewed to determine the adequacy of the licensee's analysis of defeated automatic safety function.

(1) UNREVIEWED SAFETY QUESTION DETERMINATION, Identifier SQP871738 / SQT870649. This compensatory measure involved re-alignment of fire pumps 2A-A and 2B-B to require manual start actuation by the operator. This action was necessary to prevent overloading of the Diesel Generator during a loss of coolant accident (LOCA) concurrent with a loss of offsite power The assumption is that the containment temperatures will rise sufficiently during the LOCA to activate the fire system and the additional electrical

load necessary to operate the fire pumps will overload diesel generators.

The licensee determined that the CM provided for the defeated safety function will not degrade the auxiliary power system and the emergency core cooling system (ECCS) in the event of a LOCA concurrent with a loss of off-site power. In addition, the CM will not defeat the fire protection when required. Technical Specifications were reviewed, sections pertaining to electrical power systems and fire suppression systems, and the USQD was found to be acceptable.

(2) UNREVIEWED SAFETY QUESTION DETERMINATION, Identifier CAQR SQP871738RI / CAQR SQT870181IDI R0. A lack of adequate ground fault detection was identified on the 480V class 1E power system. These circuits were tested once per shift. The test results have not been documented. A CM of testing the 480V 1E power system has not been initiated to mandate the testing and to document the results. The USQD determined that no safety functions of the plant were being degraded, and no margin of safety as defined in the basis for any TS was reduced as a result of implementing the CM. The inspector agrees with the licensee's assessment.

(3) UNREVIEWED SAFETY QUESTION DETERMINATION, Identifier CAQR SQP871738 RI/ Memo A27830919018. To minimize the ingestion of hot gases or hydrogen into the air handling unit (AHU) ducts and to minimize heat addition in containment due to pyrolysis of foam insulation, a CM has been adopted to remove power from the ice condenser AHU following a LOCA. The licensee has analyzed the issue and determined that removal of power to the ice condenser AHUs during a LOCA will not reduce any technical specification safety margin nor will it violate any requirements assumed in the safety analysis report (SAR). The inspector determined that this USQD was acceptable.

(4) UNREVIEWED SAFETY QUESTION DETERMINATION, Identifier CAQR SQP871738 RI/IDID-2.09. The ERCW supply header to the station air compressors is a class H, non-seismic, line at the point where it enters the turbine building. A CM requiring the operator to isolate the breaker has been incorporated into AOI-9 which is more explicit then the previous requirement contained in SOI-55-OM-27B-XA-27B-D. The licensee has evaluated this issue and determined that the consequences of an accident are not increased by the revised procedure. In addition, the licensee has committed to correct the deficiency and therefore eliminate the USQD by adding automatic isolation of the ERCW flow that would result from a non-seismic line break. This commitment is to be incorporated prior to U-2, mode #2 operation. The licensee currently is tracking this modification

on the restart P-2 schedule. However, the USQD determined that the CM was acceptable for plant operations.

(5) UNREVIEWED SAFETY QUESTION DETERMINATION, Identifier CAQR SQP871738/ECNL073. Weight indicating system for the upper head injection does not function and the annunciators have been disabled. This item is a mode 3 issue and the licensee currently has a mode 3 RTI 1.1 punchlist item to ensure it is completed prior to mode 3. The inspector will review the USQD and CM prior to mode 3.

(6) UNREVIEWED SAFETY QUESTION DETERMINATION, Identifier CAQR SQP871738 RI/CAQR SQP87183 RO. Containment electrical penetration conductors do not have circuit protection as required by design criteria, SQN-DE-V-11.3. Failure of this electrical equipment could potentially violate containment integrity. However, these are not to be used in modes 1, 2, 3, and 4 without adding the circuit protection. The licensee has evaluated the issue and determined that testing using these penetrations can only be performed in modes 5 and 6. In addition, the test connections are to be de-energized prior to entering mode 4. The inspector agreed with the licensee assessment provided by the USQD.

(7) UNREVIEWED SAFETY QUESTION DETERMINATION, Identifier CAQR SQP871738 RI/Memo S53851206915. Limits on radiation levels in the evaporator bottoms going to the boric acid tanks. (BAT) have been established to ensure that the area around the boric acid tanks could be classified as a mild environment. A CM has been initiated to check for Cs-137 to verify that the liquid is within acceptable activity level prior to transferring to the BAT.

The licensee determined that the CM has no effect on the basis for any TS and therefore the margin of safety is not reduced. This USQD was found to be acceptable.

(8) UNREVIEWED SAFETY QUESTION DETERMINATION, Identifier CAQR SQP 871738 RI/EC 230.01. The ability of Ruskin fire dampers to close against normal operating flow have been questioned. Ruskin filed a 10 CFR, Part 21 with the NRC on this condition. The licensee evaluated the condition and ten fire dampers were identified which require compensatory measures in order to perform their safety function. The inspector questioned the licensee on the time needed to perform these actions and after several discussions was satisfied that the USQD adequately addressed the CM.

(9) UNREVIEWED SAFETY QUESTION DETERMINATION, Identifier CAQR SQP871738 / SCRSQNMER8677. The spent fuel pool (SFP) heat exchangers must be realigned to train 'B' component cooling

system (CCS) upon the loss of train 'A' power. This requires operator action to manually operate certain CCS valves through a mechanical action.

The licensee evaluated this condition and determined that no margin of safety as defined in the TS has been reduced by the implementation of the CM. This USQD was found to be acceptable.

(10) UNREVIEWED SAFETY QUESTION DETERMINATION, Identifier CAQR SQP871738RI / CAQR SQP871477 IDI R0. The flex hoses installed to connect the diesel generator heat exchangers to the essential raw cooling water (ERCW) supply have not been qualified to withstand a seismic event. A CM has been implemented requiring operator action to visually inspect the flex hoses prior to Unit 2 restart and after each diesel generator start. This action was recommended by the licensee's consultant who evaluated the capability of these hoses to withstand a SSE event although they were not purchased to seismic specifications.

The licensee has evaluated this condition and determined that no plant safety function or operation will be adversely affected or degraded by the visual inspection and that no possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is created by the implementation of the CM.

(11) UNREVIEWED SAFETY QUESTION DETERMINATION, Identifier CAQR SQP871738 R1/CAQR SQF870022 R0. Manual opening of various auxiliary building interior doors are required upon a tornado alert. The impact of opening these doors when a tornado warning is in effect was evaluated by the licensee. This evaluation considered fire protection, toxic gas control, electrical separation/isolation, physical security, environmental qualification, external missile protection, procedural error, and radiation hazards. Based on the required fire watches associated with opening the doors, the licensee has determined that no reduction of safety margin as defined in the basis for any technical specification has occurred. Additionally, the licensee modified the procedures to start compensatory actions on a tornado watch.

(12) UNREVIEWED SAFETY QUESTION DETERMINATION, Identifier CAQR SQP871738 RI/IDID-22-7. A requirement for operator action to turn off the ERCW ventilation fans when pumping station temperature reaches 65°F or less has been implemented as a CM. This requirement was implemented to prevent freezing of portions of the ERCW system during the winter months.

The licensee has evaluated this CM and determined that the plant margin of safety has not been reduced by this CM and the changes

does not introduce any events not previously analyzed in the FSAR.

(13) UNREVIEWED SAFETY QUESTION DETERMINATION, Identifier CAQR SQP871738 RI/CAQR SQP870217 RO. Valves on the ERCW header that supply cooling water to the condensers on the air conditioning systems to the (1) main control room, (2) electrical board room and, (3) shutdown board room do not function as designed. This condition prevents proper ERCW flow to these condensers. The standby trains will not automatically start when the respective normal train fails. The CM is for the operator to manually start the standby train and to manually open/throttle the ERCW flows to the subject air conditioning condensers.

The licensee has evaluated this condition and determined that no plant safety function will be adversely affected by the implementation of the subject CM.

(14) UNREVIEWED SAFETY QUESTION DETERMINATION, Identifier CAQR SQP871738 RI/NRC 6.22, Physical separation of auxiliary control air system air headers inside containment to prevent adverse interactions have been previously identified. In addition, the control building air handling system logic switching configuration appears to be inadequate (i.e., once the control logic switches from an operating train which has lost auxiliary control air to a standby train, the control logic will not switch back to the original train if it regains auxiliary control air and the second train fails). Required operator actions have been implemented to perform the necessary changes. A line break would be followed by a header isolation signal. However, a temporary loss of air supply would cause a temporary failure and auto transfer to the backup unit.

The licensee has evaluated this condition and determined that the ability of this equipment to perform its intended function has not been affected adversely by the implementation of the CM and no new malfunctions of accidents, are foreseen. Therefore no margin of safety reduction would result.

(15) UNREVIEWED SAFETY QUESTION DETERMINATION, Identifier SQP871738RI /SCRSQNNEB8617. Manual operation of valve HCV-77-920, is required to prevent flooding of the annulus due to a moderate energy line break. This action is to be performed within 15 minutes of a high level alarm in the annulus drain sump in order to drain the annulus to the auxiliary building sump. Operating instructions have been developed to provide surveillance of the annulus drain sump alarm system and provisions have also been developed for inspection of the annulus for flooding if the alarm system is inoperable. A review of the applicable maintenance "Detailed Work Instruction," PM 2135-040, rev. 0,

revealed that a requirement for posting a watch existed if either of the annulus sump level switches are inoperable. An evaluation by the licensee had determined that the implementation of the CM will not result in a lesser margin of safety than that analyzed.

(16) UNREVIEWED SAFETY QUESTION DETERMINATION, Identifier CAQR SQP870031 / SQN SQA 119. A problem associated with the unreliability of ERCW supply valves providing cooling water to the emergency diesel generators has been identified. CM requiring an operator to provide assurance that these valves open on demand has been initiated for instances when the valves have been overtightened or handtightened. However, the operator for these valves are to be replaced prior to restart which will eliminate the need for this CM.

The licensee has evaluated the condition and determined that with the operations limitations and actions specified, no failure of the Diesel Generators, due to a failure of the ERCW supply valves to operate, is foreseen. However, these limitations and actions may be cancelled when the Rotork operators are replaced. The licensee has further stated that when these actions are completed, prior to mode #2, full reliability of the diesels are assured and the plant can be operated without impact on safety.

17. Containment Walkdown

On January 19, 1988, two NRC inspectors accompanied TVA management on a tour of the Unit 2 containment. The purpose of this containment tour was to evaluate plant material condition within the containment prior to plant heatup. The items identified by the inspector were discussed with TVA for correction. The items included the following:

a. Penetration X79B cap bolts did not have proper thread engagement.

b. Crane stop on manipulator crane is missing.

c. Limit switch for air return fan damper on elevation 734 inside hatch does not contact actuator arm.

d. RHR containment sump screen is frayed and needs to be secured.

e. RHR containment sump ground strap is disconnected.

f. Temperature element 2-TE-68-386 or TE-1324 is not properly secured.

g. Lower containment drain inside polar crane wall near PRT is stopped up.

h. Excore basket positioning devices need to have locking pins installed on upper and lower rods.

i. Reactor coolant SG/RCP cross-under piping on loop 1 has loose cable tension and the whip restraining device is cocked and has been rubbing it's stop.

j. Air return fan flow devices have nylon tubing installed. Is this installation permanent and will it be needed during a LOCA?

k. Air lock doors need gasket cleaned, inspected, and lubricated as necessary.

l. All steam generator manway cover lifting tools are not secured. Is this a problem?

m. 2-PI-68-42 RCS loop 3 hot leg sample pressure gauge needs face glass replaced.

The licensee also identified several items that needed correction. The licensee initiated action to evaluate and correct the above items along with items they identified. Items a. - e., h., k.,l., and m. were corrected with work requests (WR). Item f. was determined to be associated with the RVLIS system and a WR was issued to correct the problem. Item i. was corrected with a WR however, the licensee was requested to evaluate thermal movement of loop 1 cross-under piping during thermal expansion testing STI-62. Item g. was reported corrected by the cleanup crew. Item j. was corrected by field change request (FCR) 6698 which removed the tubing and blanked the lines. The instruments which were supplied by these instruments were determined to be non-safety and are not used to operate the equipment during accident conditions. Additionally, the instruments in the control room were marked as not operational. Item m. was evaluated by licensee as not requiring to be restrained and an engineering evaluation was performed to support this decision.

18. IE Bulletins (92071)

IE Bulletins (IEBs) are documents issued by the NRC which require certain specific actions of the addressee. The inspector has reviewed the actions taken by the licensee as a response to the below listed IEBs. The inspector verified that corrective actions appeared appropriate; generic applicability had been considered; the licensee had reviewed the event and that appropriate plant personnel were knowledgeable; no unreviewed safety questions were involved; and that violations of regulations or Technical Specification conditions did not appear to occur.

(Open) IEB 87-02; Fastener Testing To Determine Conformance with Applicable Material Specifications. The inspector participated with licensee materials engineers in selecting the 40 fastener samples required to be tested per action paragraph two (2) of the bulletin and TI 2500/26.

Twenty samples were selected from safety-related and twenty samples from non-safety related. The samples were further broken down into ten from studs, bolts, and/or capscrews and ten from nuts for both the safety related and the non-safety related categories. The sample included fasteners from the categories of the bulletin including some from SAE grade J429 with markings RT, H, and KS. The samples were tagged and bagged for shipment to the test lab. The data sheet for each item denoted the fastener description, location, contract item, material specification, head markings and QA level. The data sheet was stapled to the bag containing the sample. This information will allow TVA to identify any application that the parts being tested are used for. The following is a list of material selected:

SAFETY RELATED BOLTS/STUDS

Material	Size
A193, B7	.5 x 13 UNC x 5
A193, B7	9/16 x 12 UNC x 4
A193, B7	.5 x 12 UNC x 4
316	.875 x 9 UNC x 6
A193, B8M	.5 x 13 UNC x 4
A449	1/4 x 20 UNC x 3
A307, GR B	1.375 x 6 UNC x 6
A307, FR A	.562 x 12 UNC x 4
304	.412 x 18 UNC x 1.5

SAFETY RELATED NUTS

Material	Size
A194, 2H	.5 x 13 UNC
A194, 2H	3/4 x 10 UNC
A563, A	.625 x 11 UNC
A194, 8M	.312 x 18 UNC
SA194, 8M	.562 x 12 UNC
A563, B	.25 x 20 UNC
SAE J995, 8	.875 x 9 UNC
A194, 7	1.75 x 8 UNC
A194, 2H	1 x 8 UNC
316	.875 x 9 UNC

NON-SAFETY BOLTS/STUDS

Material	Size
A193, B7	3/4 x 10 UNC x 5
SAE J429, 5	1/4 x 28 UNF x 2 1/2

SAE J429, 8	7/8 x 9 UNC x 3 1/2
SAE J429, 8	3/8 x 16 UNC x 1.5
SAE J429, 8	1/2 x 13 UNC x 3.5
SAE J429, 8	3/4 x 10 UNC x 2
SAE J429, 8	3/8 x 10 UNC x 3.5
18-8	1/2 x 13 UNC x 1 3/4
18-8	3/8 x 16 UNC x 1.5
A193, B8	1/4 x 20 UNC x 1.25

NON-SAFETY NUTS

Material	Size
CS	.25 x 28 UNF
SS	.25 x 20 UNF
A563, A	1.5 x 6 UNC
Steel	1.75 x 5 UNC
J995, 8	.164 x 32 UNC 5 samples
CS	.875 x 9 UNC
CS	.19 x 24 UNC
SS	.625 x 11 UNC

19. List of Abbreviations Unit 1 and 2

AI - Administrative Instruction
AFW - Auxiliary Feedwater
AUO - Auxiliary Unit Operator
AOI - Abnormal Operating Instruction
ASME - American Society of Mechanical Engineers
BIT - Boron Injection Tank
C&A - Control and Auxiliary Buildings
CAQR - Conditions Adverse to Quality Report
CCP - Centrifugal Charging Pump
CCS - Component Cooling System
CCTS - Corporate Commitment Tracking System
COPS - Cold Overpressure Protection System
CS - Containment Spray
CSSC - Critical Systems and Components
CST - Condensate Storage Tank
DBVP - Design Baseline Verification Program
DC - Direct Current
DCN - Design Change Notice
DNE - Division of Nuclear Engineering
ECCS - Emergency Core Cooling System
ECN - Engineering Change Notice
EDG - Emergency Diesel Generator
EGTS - Emergency Gas Treatment System
EMI - Electric Magnetic Interference
EQ - Environmental Qualification
ERCW - Essential Raw Cooling Water

ESF	-	Engineered Safety Feature
FCR	-	Field Change Request
FSAR	-	Final Safety Analysis Report
HO	-	Hold Order
HP	-	Health Physics
HQ	-	Headquarters
HVAC	-	Heating, Ventilation, and Air Conditioning
IDI	-	Integrated Design Inspection
IE	-	Inspection and Enforcement
IEB	-	Inspection and Enforcement Bulletin
IFI	-	Inspector Followup Item
IMI	-	Instrument Maintenance Instruction
KV	-	Kilovolt
LER	-	Licensee Event Report
LCO	-	Limiting Condition for Operation
LOCA	-	Loss of Coolant Accident
MI	-	Maintenance Instruction
MOVATS	-	Motor Operated Valve Testing
MSIV	-	Main Steam Isolation Valve
NEP	-	Nuclear Engineering Procedures
NRC	-	Nuclear Regulatory Commission
ODCM	-	Offsite Dose Calculation Model
OSP	-	Office of Special Projects
PD	-	Positive Displacement
PI	-	Pressure Instrument
PM	-	Preventive Maintenance
PMT	-	Post Modification Test
PORV	-	Power Operated Relief Valve
PORS	-	Plant Operation Review Staff
PRO	-	Potentially Reportable Occurrence
QA	-	Quality Assurance
QC	-	Quality Control
RARC	-	Radiological Assessment Review Committee
RCS	-	Reactor Coolant System
RCP	-	Reactor Coolant Pump
RHR	-	Residual Heat Removal
RO	-	Reactor Operator
RTD	-	Resistance Thermal Devices
RTI	-	Restart Test Instruction
RWP	-	Radiation Work Permit
RWST	-	Reactor Water Storage Tank
SER	-	Safety Evaluation Report
SFP	-	Spent Fuel Pool
SG	-	Steam Generator
SI	-	Surveillance Instruction
SIS	-	Safety Injection System
SMI	-	Special Maintenance Instruction
SOI	-	System Operating Instructions
SRO	-	Senior Reactor Operator
SSOMI	-	Safety System Outage Modification Inspection

STI	-	Special Test Instruction
TACF	-	Temporary Alteration Control Room
TAVE	-	Average Reactor Coolant Temperature
TDAFP	-	Turbine Driven Auxiliary Feedwater Pump
TS	-	Technical Specifications
TSC	-	Technical Support Center
TVA	-	Tennessee Valley Authority
UHI	-	Upper Head Injection
URI	-	Unresolved Item
USQD	-	Unresolved Safety Question Determination
VCT	-	Volume Control Tank
VIO	-	Violation
WCC	-	Work Control Center
WP	-	Work Plan
WR	-	Work Request