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

Report Nos.: 50-327/87-65, 50-328/87-65

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: October 6, 1987 thru November 5, 1987

Lead Inspector: Robert L Carroll for 12/21/87
K. M. Jenison, Senior Resident Inspector Date Signed

Accompanying Inspectors: 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: 12/22/87
F. R. McCoy, Chief, Projects Section 1 Date Signed
Division of TVA Projects

SUMMARY

Scope: This routine, announced inspection involved inspection onsite by the Resident Inspectors 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: Three violations were identified.

(327, 328/87-65-01), Inadequate Corrective Actions - paragraph 12
(327, 328/87-65-02), Inadequate Response Time Test - paragraph 6
(327, 328/87-65-03), Failure to Adequately Control Changes to Control Room Drawings - paragraph 3

One unresolved item was identified.

(327, 328/87-65-04), Surveillance Discrepancies - paragraph 13

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REPORT DETAILS

1. Licensee Employees Contacted

H. L. Abercrombie, Site Director
J. T. La Point, Deputy Site Director
*L. M. Nobles, Plant Manager
*B. M. Willis, Operations and Engineering Superintendent
*B. M. Patterson, Maintenance Superintendent
R. J. Prince, Radiological Control Superintendent
*M. R. Harding, Licensing Group Manager
L. E. Martin, Site Quality Manager
D. W. Wilson, Project Engineer
R. W. Olson, Modifications Branch Manager
J. M. Anthony, Operations Group Supervisor
*R. V. Pierce, Mechanical Maintenance Supervisor
M. A. Scarzinski, Electrical Maintenance Supervisor
*H. D. Elkins, Instrument Maintenance Group Manager
R. S. Kaplan, Site Security Manager
J. T. Crittenden, Public Safety Service Chief
*R. W. Fortenberry, Technical Support Supervisor
*G. B. Kirk, Compliance Supervisor
D. C. Craven, Quality Assurance Staff Supervisor
J. H. Sullivan, Regulatory Engineering Supervisor
J. L. Hamilton, Quality Engineering Manager
D. L. Cowart, Quality Engineering Supervisor
H. R. Rogers, Plant Operations Review Staff
R. H. Buchholz, Sequoyah Site Representative
M. A. Cooper, Compliance Licensing Engineer

Other licensee employees contacted included technicians, operators, shift engineers, security force members, engineers and maintenance personnel.

*Attended exit interview

2. Exit Interview

The inspection scope and findings were summarized with the plant manager and members of his staff on November 5, 1986. Three violations described in this report's summary paragraph were discussed. No deviations were discussed. The licensee acknowledged the inspection findings. The licensee did not identify as proprietary any of the material reviewed by the inspectors during this inspection. During the reporting period, frequent discussions were held with the Site Director, Plant Manager and other managers concerning inspection findings.

During the exit interview plant management committed to revising AOI-27, Control Room Inaccessibility, per paragraph 3 below, in order to meet the guidance of regulatory guides 1.68.2 and 1.68.

3. Licensee Action on Previous Inspection Findings (92702)

(Closed) Unresolved Item (URI) 327, 328/87-24-02, Control of Temporary Changes to Drawings. This URI has been determined to constitute a violation. Drawings in the control room are marked by the modifications engineers immediately following the completion of physical changes to the plant's as-built condition. These temporary drawing changes consist of red marks for additions to the drawings, and green marks to designate removals. During the inspection described in IR 327, 328/87-24, several instances were identified where errors were introduced to the control room drawings by the modifications engineers when they marked up the drawings. 10 CFR 50, Appendix B, Criterion VI requires that changes to documents (including drawings) be reviewed for adequacy and approved for release by authorized persons. This requirement was not met in that changes to primary control room drawings were routinely made with no verification or review by second parties. This is a violation VIO 327,328/87-65-03.

(Closed) URI 327, 328/87-08-02, Abnormal Operating Instruction (AOI) Personnel Required for Remote Shutdown. The inspector reviewed AOI-27, Control Room Inaccessibility. The procedure describes actions to be taken should the control room become uninhabitable. The procedure requires the dispatching of more personnel than are required as a minimum on-shift in TS 6.2.2.a. Appendix A of 10 CFR 50 requires in GDC 19 that equipment at appropriate locations outside the control room be provided with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown. This criteria is addressed in regulatory guide (RG) 1.68.2, Initial Startup, Nuclear Power Plants. This RG states that startup testing should demonstrate that the number of personnel available to conduct the shutdown operation is sufficient to perform the many actions required by the procedure in a timely, coordinated manner.

Startup test documentation, as described in item SU-1.2A of Table 14.1-3 in the final safety analysis report (FSAR), was reviewed. In addition, the following issues, identified in inspection report 327, 328/87-08, were also reviewed:

- Whether there is sufficient personnel and guidance to perform a safe and orderly shutdown from outside the control room with a minimum shift crew per the TS.

- Whether procedures are adequate to address the limiting case of minimum shift manning.

- Whether the initial startup test was performed utilizing the personnel indicated in the procedure or the TS minimum, and whether the procedure was adequate.

The inspector walked down the procedure utilizing licensed personnel and determined that the plant could be shutdown using minimum shift crew. However, performance with a minimum crew would require some step sequence modifications to AOI-27. A draft update to AOI-27 was prepared and reviewed making it possible to run with minimum shift crew. During the exit interview, plant management committed to revising AOI-27 in order to meet the guidance of regulatory guides 1.68.2 and 1.68.

The inspector did determine that the initial startup test utilized normal shift manning as opposed to the minimum shift crew in TS. This action to use the normal shift manning was according to the TVA commitment set in the FSAR. This item is closed.

(Open) Violation (VIO) 327, 328/86-37-06, Containment Sump Level Test Deficiencies. The inspector reviewed this issue which involved a prolonged history of test deficiencies with respect to the calibration of the containment sump level transmitters. These particular transmitters were identified as being out of TS tolerance six times during the surveillances conducted between 1984 and 1986. The NRC identified that the licensee did not initiate a quality assurance document identified as a corrective action report (CAR). Failure to initiate a CAR was identified as a violation and the cover letter for inspection report 327, 328/86-37 stated that, "violation 327, 328/86-37-06 was also identified and is described in paragraph 13 of the enclosed inspection report. This additional violation is under consideration for escalated enforcement action. Accordingly, a notice of violation addressing this particular violation is not being issued at this time, and therefore no response to this violation is required." The cover letter for the inspection report further stated that "the number and characterization of violations described in paragraph 13 of the enclosed inspection report may change as a result of further NRC review".

The licensee's corrective actions with respect to the containment sump level transmitters included issuance of a condition adverse to quality report (CAQR) CAQR-SQP870043 and the establishment of a plant tracking and trending program under standard practice SQM-58, maintenance history and trending. The general adequacy of the QA CAQR and the maintenance tracking and trending processes are subjects of separate NRC inspections. The specific issues dealing with the identification of the containment sump level transmitters as a condition adverse to quality and tracking and trending of those specific material deficiencies appear to have been adequately addressed by the licensee. This item will remain open, pending escalated enforcement action.

(Closed) VIO 327, 328/86-39-01, Failure to Report Computer Program Errors. This violation involved the use of incorrect axial flux curve and rod bow penalty data in monthly surveillances of incore reactor parameters. In response to the violation, the licensee wrote a licensee event report (LER) 328/86-004 which was closed in NRC inspection report 327, 328/87-08. The LER appears to address adequate corrective action to violation 327, 328/86-39-01. The inspector examined the root cause of this issue in detail and evaluated the corrective action with respect to the security and accuracy of data being placed into safety related plant computer systems in general and the INCORE program in particular. In LER 328/86-004 the licensee committed that before installing vendor-supplied data into software in the future, verification will be provided that the correct data is being used. This is accomplished through an independent verification process. This item is closed.

(Closed) URI 327, 328/87-30-03, Reactor Coolant System Sight Glass Design. The design of the sight glass used to indicate reactor coolant system (RCS) inventory level during partial loop drain down conditions was questioned by the NRC inspection staff as well as the licensee's nuclear manager's review group (NMRG) following the RCS spill event of January 28, 1987. Several design deficiencies were identified and later determined by the NRC not to be safety related. In particular, the monitoring arrangement using a TV camera and monitor was not well designed to allow the control room operator to readily determine the level in the sight glass. Resolution of the above deficiency and several other minor deficiencies is scheduled during the next refueling outage for each unit. The corrective actions are presently being tracked under the licensee's tracking and reporting of open items system (TROI). The actual site glass modifications will be reviewed when the appropriate reviews and corrections have been accomplished during the unit 2 cycle 3 outage. This item is closed.

(Closed) URI 327, 328/87-30-04, Change to TS Basis. Sequoyah issued Technical Specification (TS) basis change 87-14 directly to the NRC without a nuclear safety review board (NSRB) review. The licensee's position is that the TS basis is not an actual part of the TS as defined by 10 CFR 50.36(a). Therefore, TVA's position is that changing the basis does not require an NSRB review. NRC approval of the change to the TS Basis was issued on August 18, 1987. The inspector discussed the licensee's position with OSP Technical Programs management, and determined that this generic issue is currently under NRC review. TVA indicated that they would have the NSRB perform such reviews until such time this generic issue is resolved. This item is closed.

(Open) VIO 327, 328/86-73-04, Failure to Properly Evaluate the Generic Applicability at Other Nuclear TVA Facilities of Conditions Adverse to Quality (CAQ). The violation pertained to the adequacy of engineering evaluations. Deficiencies identified included Bellefonte CAQR BLN4929, dated June 30, 1986. Sequoyah did not properly analyze the cause of the CAQR and did not have documentation for justification of the determination that the CAQR did not apply to Sequoyah. TVA gave as reasons for the violation "the failure on TVA's part to provide adequate and sufficient problem descriptions and detailed information in the assessment of the potential generic condition as recorded in the P&CE memorandum so that other TVA facilities could properly assess the generic implications of the cited conditions." Their corrective action includes a revision 3 to TVA's nuclear quality assurance manual (NQAM) part I, Section 2.16, which requires generic reviews to be completed after root cause analysis and recurrence control actions have been determined. This was implemented July 1, 1987. TVA further revised procedures for more controlled and centralized requirements for the conduct and timing of generic reviews. The inspector reviewed CAQRs at Sequoyah that have been determined by TVA to be generic. This was a random review of CAQRs initiated by Sequoyah and by other plants. The following deficiencies were noted.

- CAQR number WBP870420 was initiated on June 6, 1987, at Watts Bar and concerned the storage of electrical cables in the warehouse and craft storage area at Watts Bar. The CAQR listed 8 items to address and gave as the root cause of the CAQ, "inadequate procedures for cable transactions and storage which lead to personnel being trained incorrect." On August 1, 1987, the generic review sheet was signed with a determination that this CAQ did not exist at Sequoyah. The justification was based on inspection of cable storage areas and addressed the 8 items, but failed to address the root cause, i.e., inadequate procedures and training.

- CAQRs WBP870420 and BFQ870375 attachment 7 CAQ generic review sheets were not properly reviewed. NQAM Part 1, Section 2.16, step 10.3 states," the justification for determining that a CAQ is not generic shall be documented. The individual who prepares the justification shall provide a dated signature, and the individual's supervisor shall approve the justification by dated signature." On August 1, 1987, the same individual signed WBP 870420 as reviewer and as supervisor of reviewer. On June 26, 1987, the same individual signed BFQ 870396 as reviewer and as supervisor of reviewer. This is a failure to follow procedures.

- On July 9, 1986, Bellefonte (BFN) identified a CAQ and initiated SCR BLN4929. The root cause of this CAQ was an operator attempting to close the diesel generator (D/G) output breaker out of phase. Sequoyah misidentified the root cause. The recommendations made at Bellefonte to prevent recurrence are: (1) additional simulator training for operators, (2) review of the incident with operators, and (3) addition of a permissive synchronize and voltage check scheme on the diesel generator output breaker.

  Sequoyah CAQR SQP870943, signed on July 2, 1987, states in the description of proposed disposition, that "items 1 and 2-POTC provides D/G simulator training during RO certification. Operators are already aware that the D/G should be in phase before synchronization. No additional training is proposed. Item 3, we do not recommend adding an interlock to the D/G breaker. This accident appears to have been due to human error and was an isolated incident." The recommended actions at Sequoyah appear to be inconsistent with those at Bellefonte. Further, NQAM Section 1, Part 2.16 step 10.5.2 states "for CAQs determined to be generic, affected organizations should communicate with each other in the development of corrective action plans to ensure that, when appropriate, such plans are consistent."

Further inspections of the generic review issue will be conducted prior to the startup of either unit. This violation remains open.

(Closed) 327, 328/87-06 observation MEB-8, Inconsistent Equipment Qualification Temperature. Calculation NEB 811007235 had been prepared to analyze a deficiency in the pre-operational test results for the turbine driven auxiliary feedwater pump (TDAFW) room ventilation system and to provide suggestions to reduce the temperature rise in the room. The calculation uses a temperature rise to 125 degrees F, which the calculation states is the equipment qualification temperature. This is not consistent with plant data sheet, 47E235. TVA recalculated the room heat loads in B44870609010, dated April 10, 1987, and the required air flow rates in B44870716007, dated July 16, 1987. The conclusion is that the installed exhaust fans are adequate to meet the requirements of normal operation and LOCA, with and without loss of offsite power, with a maximum of 110 degrees F in the room. The corrective action in PIRSQNMEB8773 is to change FSAR section 9.4.2.2.7 to delete the room temperature rise criterion and to replace it with a maximum room temperature of 110 degrees F. This corrective action appears to be an adequate response to NRC's concern. This item is closed.

(Closed) 327, 328/87-06 observation GEN-1, Substantiated Condition for a CAQ. Nuclear engineering procedure (NEP)-9.1, Corrective Actions, revised July 1, 1986, defines the controlled system within DNE to document, evaluate and resolve conditions adverse to quality (CAQs). NEP-9.1 was being revised to agree with the corporate QA procedure, NQAM, Part I, Section 2.6, which includes in the definition of CAQs the statement that "unsubstantiated conditions are not defined as CAQs." The concern was that this statement had the net effect of eliminating a set of CAQs that had previously been identified and resolved. NEP-9.1, Revision 2, Section 2.2, dated June 30, 1987, eliminates the questionable statement from the definition. Section 1.1b of NEP-9.1, Revision 2, provides for a problem identification report (PIR) system to document problems, and potential problems that are not CAQs as defined. The licensee's action resolves the concern. This item is closed.

(Closed) 327, 328/87-14 observation 6.19, 480 Volt Board Room Air Handling Unit Control Logic. Cut-out switches had been installed in each of the heating, ventilation and air conditioning (HVAC) systems for the 480V board room and the 6.9KV shutdown board room to shut off the room's cooling system on a high temperature signal in the room. The NRC concern was that the close location of the temperature sensors was such that a common mode failure could disable all cooling. CAQR871011 for the 480V board room and CAQR871279 for the 6.9KV shutdown board room were issued to disconnect the room high temperature cut-out switches since the switches had been installed for economic reasons to protect equipment in the event of refrigerant loss, and had no safety function. ECN7263 followed up on this item and the switches have been disconnected. This item is closed.

(Closed) VIO 327, 328/87-02-02, Failure to Establish, Maintain and Implement Safety-Related Procedures. This violation resulted from three engineered safety feature (ESF) initiations reported in licensee event reports (LERs) 328/86-08, 328/86-09 and 328/86-10, and from observations of the performance of general operating instruction (GOI)-6H, Freeze

Protection. The violation consisted of an inadequate work plan (WP), instrument maintenance instruction, GOI documents, and personnel errors during procedure performance. The inspector reviewed TVA's corrective actions detailed in the June 12, 1987 written response to this violation and in the LERs. The specific procedural errors and weaknesses have been corrected. Site procedures such as surveillance instructions have been through an extensive review process and are now developed and revised using checklists and a writer's guide. Maintenance procedures are currently undergoing a similar process. Personnel involved in the specific incidents have been reinstructed and more extensive training has been conducted regarding adherence to procedures. The licensee's corrective actions appear to be adequate. This item is closed.

(Closed) VIO 327, 328/86-68-05, paragraph c, Section 2.4.5, deficiency D-2,4-6, Loose and Broken Flexible Conduits. NRC inspectors reviewed the July 16, 1987 TVA response to this portion of the violation and the corrective action taken. All discrepancies except the loose conduit on valve 2-FSV-30-14 were addressed in NRC report 87-57. Electrical maintenance request B234106 completed corrective action on valve 2-FSV-30-14. NRC inspectors performed field inspection of the valve and found corrective action to be adequate. This violation is closed.

(Open) URI 327, 328/86-68, section 2.4.10, U-2.4-2, Flamastic Thickness on Cable Trays. NRC report 327, 328/86-68 noted several areas in safety-related cable trays that had flamastic thicknesses in excess of the 1/4-inch that was assumed for ampacity derating due to thermal loading. During discussions between NRC inspectors and site management, TVA committed to an examination of the as-installed flamastic thicknesses at Sequoyah. The TVA draft commitment was reviewed by the NRC and discussions were held with TVA's department of nuclear engineering personnel. TVA sampled flamastic thicknesses at 68 node locations of the 720 total node locations associated with class 1E control power and power cable trays. The node locations were selected by random number generation. Actual flamastic thickness examination identified 16 cables that had thicknesses in excess of 1/4-inch, but that no thickness exceeded 1/2-inch. Calculation SQN-E2-025 was performed on all class 1E control power and power cables based on a thickness of 1/2-inch. The results indicated that only three cables (2PL3091A, 2PL4091B, and 2PL4958A) would be inadequately sized if the flamastic exceeded the 1/4-inch maximum assumed thickness. A supplemental field walkdown measured flamastic thickness at 22 locations on these three cables. The supplemental walkdown identified that only one cable (2PL4901B) of the three has a flamastic coating exceeding 1/4-inch. That cable was determined to be adequate because it was a normally deenergized alternate feeder; it was routed in a mild environment; the flamastic coating exceeded 1/4-inch for only part of the length of the cable; and the ampacity reduction was only 0.67% below full load current. NRC performed inspections on cable trays that had been previously identified as having thickness greater than 1/4-inch and noted that the trays were control and signal cable trays. A satisfactory re-inspection of class 1E control power and power cable trays was conducted by TVA. The licensee's corrective actions appear to be adequate. However, this item remains open pending review of the final TVA resolution memo B25 87-1029-008 and tracking numbers NCO 87-0029-002 and NCO 0484-005.

(Closed) URI 327, 328/85-45-18, Provide Complete Description of Corrective Actions Taken To Preclude Circumstances Which Resulted In The Installation of Upper Head Injection (UHI) Level Switches With Incorrect QA Level Designations. The licensee identified in PRO-2-85-008 that 3 UHI level switches did not have proper documentation of seismic qualification and QA level designation. The subject switches were replaced and a review commenced on both 1E components that had been replaced and components in stock in the warehouse. NRC report 327, 328/86-48 reported status of the review and the ongoing resolution of 138 components that had inadequate seismic qualification documentation. NRC report 327, 328/87-37 provided further status and reported that not all devices that required replacement had been replaced. TVA has completed the review and replacement of required items that are documented in TVA memorandum, D. W. Wilson, DNE/H. L. Abercrombie/ONP dated March 19, 1987. NRC inspectors reviewed seismic and QA level designation documentation for the devices and found them to be acceptable. This item was reviewed against the NRC enforcement criteria, and was determined to be licensee identified. Therefore, this item is closed.

(Closed) URI 327, 328/87-24-01, Drawing Control. TVA administrative instruction (AI)-25, part I, Drawing Control After Unit Licensing, revision 20, details the methods utilized by Sequoyah for control of drawings. This procedure has been revised in response to NRC concerns regarding drawing control. AI-25 provides definitions of the various drawings; responsibilities for control of drawings; procedures for receipt, inspection, filming, distribution, and filing; guidelines for determination of primary drawings and utilization of drawing criteria. AI-25 specifically addresses guidelines for selection of those drawings considered primary drawings (ie., required in the control room for safe startup, operation and shutdown of the Unit. Appendix 1 to AI-25 lists those drawings considered to be primary. Included in Appendix 1 are drawings designated as critical drawings (required for technical support center and Chattanooga emergency operations center). Primary drawing changes will be controlled by the operations group manager and the plant manager as provided in AI-25. This procedure was approved by the plant operations review committee (PORC) on September 25, 1987. The primary drawings in the control room are required to be inventoried against Appendix 1 of AI-25 annually. Currently, the site document control section is ensuring primary drawings are maintained up to date. This URI is closed.

(Closed) VIO 328/85-28-05, Accumulator Boron Concentration out of TS Limit. The following actions were accomplished by the licensee: caution statements were added to the "drain" and "fill" portions of SOI-63.1, concerning the potential for uncertainties of boron samples taken during these processes; and a training letter was issued to licensed personnel, discussing the occurrence and the SOI revision. The licensee's actions appear to be adequate. This item is closed.

(Open) URI 327, 328/87-18-01, Potential for Secondary Bridging, Potential Use of Flammastic As An Alternative To Solid Barriers, and The Lack of Criteria for Separation of Safety-Related conduits. This unresolved item contains three specific examples related to the three generic concerns stated above. The examples include a divisional cable that was misrouted in a non-divisional tray, two non-divisional cables that were misrouted in a divisional tray, and a division A conduit that was routed nearly in contact with an uncovered division B tray. The concern for potential secondary bridging (a non-divisional cable routed with one division and then routed with the opposite division elsewhere in the plant) results from the indeterminate path of the misrouted cables. A second potential for secondary bridging existed in free air routing which is not documented in cable routing records. Long vertical runs of flammastic cable bundles in the cable spreading room contain both divisional and non-divisional cables. Because routing is not documented in free air, the potential for secondary bridging exists. TVA has instituted a change to the cable routing computer program which adds free air nodes to the cable tray system. A change in the computer program will prohibit routing a non-divisional cable which has been run with a divisional cable, from being routed with the opposite division. This action is adequate to satisfy the concern regarding secondary bridging in free air. The concern for secondary bridging due to misrouting of cables in cable trays is not resolved and requires further review. Documentation reviewed indicated that the non-divisional cables running in the divisional tray had been corrected by condition adverse to quality report (CAQR) SQP870702 which changed the routing cards for the misrouted cables. As a result of the CAQR, a category D FCR (5572) was written and completed to change the routing cards. This example is closed. The potential use of flammastic as a fire barrier which is prohibited by subparagraph 8.3.1.4.2, but was allowed by section 4.2.9 of Sequoyah design criteria SQN-DC-V-12.2, has been corrected. A design input memo (temporary change to the design specification) entitled, Design Input Memo On Separation Of Electrical Equipment And Wiring Design Criteria, SQN-DC-V-12.2, dated June 25, 1987, added information to the design specification that precludes use of flamastic as a fire barrier. This generic issue is closed. The conduit/cable tray separation criteria issue is not resolved and will be the subject of further review. Items that remain open include the divisional cable that is routed in a non-divisional tray, the potential for secondary bridging due to misrouting of cables in cable trays, and both the example and the generic concern regarding the lack of criteria for conduit/cable tray separation. The correction of the divisional cable that is in the non-divisional tray, and the potential for secondary bridging due to misrouting of cables in cable trays will be followed up under violations 87-52-01 and 02. The conduit/cable tray separation issue is a startup item and will be followed under URI 87-18-01. Accordingly, this item remains open.

(Closed) Violation 327, 328/86-68-05, paragraph d, section 2.4.14, deficiency D-2.4-16, Pipe Support Discrepancies. TVA's July 16, 1987, response to this violation and completed corrective actions were reviewed. Field change requests (FCRs) were issued to document the as-built clamp gap and change weld sizes and configuration. Space plates were added to correct the potential rotation problem with the one snubber and bracket to clamp assembly and weld data sheets documented that six undersized welds were built up to drawing requirements. Programmatic changes related to procedure adherence by craft technicians and quality control (QC) inspectors, detailed in the Sequoyah nuclear performance plan and elsewhere, have addressed the root causes of this violation. TVA's corrective actions are adequate and this portion of the violation is closed.

4. Unresolved Items

Unresolved items are matters about which more information is required to determine whether they are acceptable or may involve violations or deviations. One unresolved item was identified during this inspection, and is identified in paragraph 13.

5. Operational Safety Verification (71707)

a. Plant Tours

The inspectors observed control room operations, reviewed applicable logs, conducted discussions with control room operators, observed shift turnovers, and confirmed operability of instrumentation. The inspectors verified the operability of selected emergency systems, and verified compliance with Technical Specification (TS) Limiting Conditions for Operation (LCO). The inspectors verified that maintenance work orders had been submitted as required and that followup activities and prioritization of work was accomplished by the licensee.

Tours of the diesel generator, auxiliary, control, and turbine buildings, and containment were conducted to observe plant equipment conditions, including potential fire hazards, fluid leaks, and excessive vibrations and plant housekeeping/cleanliness conditions.

The inspectors walked down accessible portions of the safety injection system on Unit 2 to verify operability and proper valve alignment.

No violations or deviations were identified.

b. Safeguards Inspection

In the course of the monthly activities, 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.

In addition, the inspectors observed protected area lighting, protected and vital areas barrier integrity. The inspectors verified an interface between the security organization and operations or maintenance. Specifically, the resident inspectors: responded to bomb threats, fires, etc.; interviewed individuals with security concerns; inspected security during outages; reviewed licensee security event report; visited central or secondary alarm stations.

No violations or deviations were identified.

c. Radiation Protection

The inspectors observed health physics (HP) practices and verified implementation of radiation protection control. On a regular basis, radiation work permits (RWPs) were reviewed and 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.

During a tour of the auxiliary building the inspector identified two separate deviations from the licensee's health physics procedures. These involved dress out noncompliances which were minor in nature and did not result in any personnel contamination or overexposure. The licensee currently has a long term health physics corrective action effort in progress which includes improvements in dress-out and frisking issues. The two issues were discussed with operations section management, NRC management, and Sequoyah site management, during the exit conducted for this inspection period. Sequoyah health physics management initiated two radiological incident reports (RIR) at the time the two noncompliances were identified by the inspector. The inspector will review the resolution of RIR 87-24 dated October 16, 1987, and RIR 87-23 dated October 16, 1987, after completion of licensee corrective action. The inspector had no further questions.

No violations or deviations were identified.

6. Monthly Surveillance Observations (61726)

The inspectors observed/reviewed the TS required surveillance testing listed below and 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 complete tests, the inspector verified that testing frequencies were met and tests were performed by qualified individuals.

a. SI-166.36, Diesel Starting Air Valve Test

SI-7, Electrical Power System: Diesel Generators

The above two surveillances were performed in tandem with a test director in charge of SI-166.36 and an auxiliary unit operator (AUO) in charge of SI-7. During the performance of the two surveillances the test director of SI-166.36, assumed that the AUO performing SI-7 would record a piece of data that was necessary for the performance of SI-166.36 (the time for the diesel to reach and attain 900 rpm). After the diesel generator had been started and was running at the required rpm, the test director realized that the AUO was not required by SI-7 to record the time it took the diesel to reach the required speed. The test director retrieved the information from an alternate data source (control room operator) without repeating the diesel generator starting sequence. The inspector had no further questions.

b. SI-94.5, Reactor Trip Instrumentation Refueling Outage Channel Calibration. Portions of this SI were observed and the inspector had no questions.

c. SI-98.1, Channel Calibration for Engineered Safety Feature Instrumentation (Steam Flow & Pressure).

IMI-99 CC 9.10B, Offline Channel Calibration of Loop 3 Steam Generator Steam Pressure Channel II.

On October 7, 1987, the inspector observed testing in progress on CC 9 10B of IMI-99. The inspector noted that a step in the procedure was signed off stating that the precautions and prerequisites of FT/CC-9 were completed. FT/CC-9 was not at the work site and the technician that had signed it off stated that she knew what the steps were. The following were stated:

(1) Assure double person signoff for critical steps
(2) Notify operations prior to starting work
(3) Place orange stickers in the control room alarm windows
(4) Assure other loops are in service as necessary

At a later date the inspector determined that the precautions and prerequisites of FT/CC-9 were somewhat different than what he had been told. The actual procedure stated the following:

3. PREREQUISITES

3.1 Copy of functional test or channel calibration procedure and its associated data sheet(s).

3.2 Provide adequate communication between instrument locations.

3.3 Equipment required - see data sheet of functional test or channel calibration being performed.

3.4 For channel calibrations only, calibration data card for each instrument listed in component list at the beginning of each channel calibration procedure.

4. PRECAUTIONS

4.1 Calibration date on test instruments must be current.

4.2 Test may be performed on only one protection set at a time. When one protection set is being tested, the remaining protection sets must be in normal (untripped) mode.

4.3 Notify shift engineer of maintenance, calibration, or functional test to be performed.

4.4 Instrumentation and test equipment must be energized at least the minimum length of time to achieve stability in accordance with applicable manufacturers instruction manual.

4.5 If during any at-power test, a related reactor protection channel trip is actuated from another protection set, the test must be terminated and all channels returned to normal (untripped) condition.

4.6 Control systems are to be placed in the manual control mode before any change is made in a channel defeat and/or channel transfer switch position. After the change is made, the control system may be returned to automatic control.

4.7 Observe all posted health physics precautions obtaining a special work permit when required or if work to be performed could result in changes in radiation levels in work area. Tools or equipment being moved from contaminated to regulated zones or from regulated to clean zones must be surveyed by health physics unit prior to removal.

4.8 Use IMI-118 for backflushing and filling of possible contaminated instrument lines.

4.9 Loops containing Barton transmitters, Models 763 and 764, used in conjunction with the Foxboro 610A power supply shall be de-energized prior to switching the analog channel test to normal, and then re-energized after placing the test switch in the normal operating position.

Although no violation of the procedure was observed, there was an appearance that the procedure was being performed without appropriate controls. The inspector discussed with licensee management the importance of the technicians either knowing explicitly, through training, their responsibilities or having a copy of an approved, procedure in-hand.

d. SI 247.900, Engineered Safety Features Response Time Verification

During review of SI-247.900, Engineered Safety Features Response Time Verification, the inspector determined that the SI does not meet the requirements of TS 4.3.2.1.3. TS 4.3.2.1.3 requires that containment spray response time be demonstrated to be within the limit at least once per 18 months. The inspector determined that the time period for the containment spray pump start interlock to close was not included as part of the response time for the containment spray isolation valve to open. The pump start interlock must be satisfied before valve movement will begin. TVA procedures bypassed this interlock during response time testing. The failure to have an adequate SI to measure containment spray response time is identified as violation 327,328/87-65-02.

7. Monthly Maintenance Observations (62703)

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.

The following items were considered during this review: 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.

a. Work Request (WR) B298870; Maintenance on Temperature Monitor TM-68-43K

During the performance of the above corrective maintenance, the technicians implemented configuration control through the use of instrument maintenance instruction IMI-134, Configuration Control of Instrument Maintenance Activities. When this maintenance was observed the TM-68-43K circuit card had been placed into its cabinet and troubleshooting was in progress. The configuration indicated on the IMI-134 data sheet did not indicate the placement of the circuit card into the cabinet.

IMI-134 section 3.2.7 states that the technician is to list on the data sheet (work performance sheet) any configuration changes. "This includes: jumpers, wire lifts, inhibits, temporary instrument settings, unbolting flanges, disconnecting tubing and pipe fittings, temporary connection, etc."

During this troubleshooting period, the configuration work performance sheet did not accurately reflect the configuration of the equipment covered by the work activity or the trouble shooting that was conducted due to the fact that the card was in the rack when the configuration log indicated it was out of the rack. This is due to the licensee policy of logging the card one time for trouble shooting and then removing and reinstalling the card at will to accomplish that troubleshooting. The corporate maintenance manager had identified a similar programmatic issue involving adequate functional testing of equipment after maintenance is performed. During a meeting which was attended by the plant maintenance superintendent, corporate maintenance manager, and NRC management, it was agreed that selection of adequate post maintenance testing (PMT) depended on an accurate understanding of the troubleshooting that was conducted and the configuration changes that occurred in the equipment. This issue will not result in a violation because the equipment was out of service during this period and the licensee had previously identified the need for functional testing and was implementing corrective action when this issue was identified. The inspector had no further questions.

b. Preventive Maintenance (PM) 0961-068; Main Control Room Recorder 2-M-5

Portions of the 2-REC-068-VAR scaling were observed by the inspector. The PM referenced the vendor's manual and indicated that the scaling should be accomplished within the limits set by the vendor. The vendor set the scaling limits between zero and fifty milli-amps on the wand type indicator pens. The inspector observed that the technicians had applied sixty milli-amps to one of the indicator pens. The technicians had applied the sixty milli-amps to the pen in order to move it aside and had decided that the additional amperage would not damage the recorder. This particular equipment is not a critical safety system component (CSSC) and therefore the requirements of TS 6.8.1 do not apply and no violation will be issued. This issue was discussed with the plant maintenance superintendent, the corporate maintenance superintendent, NRC management and other plant management. The inspector had no further questions.

c. Work request B28468-d; Calibration of Indicating Instruments for 480-volt Shutdown Board (2B2-B)

This item was performed to validate the voltage, amperage, and watt meters for the 480 volt shutdown boards. All three meters were found within tolerance. No adjustments were necessary.

d. Work Plan (WP) 12360; Fire Protection Connections to the Control Room Computer.

On September 28, 1987, the inspector observed work in progress on WP 12360. This WP implements a portion of engineering change notice (ECN) L5841 on connections between fire detection panel 0-L-633 and the plant computer. An engineer was in charge of testing and was acting as the test director. Procedures were out and being followed by the individuals performing the WP. The engineer and testing personnel appeared to be knowledgeable of the work and their responsibilities. The inspector had no further questions.

No violations or deviations were identified.

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

The following LERs were reviewed and closed. 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 TS conditions had been identified.

a. Closed LERs

LER 327/86-044, Inadequate Verification of ECCS Flow. As previously stated in report 8 -36, the remaining required licensee action in this matter was to revise procedure SI-137.3 to include reactor coolant pump (RCP) seal differential pressure requirements. Revision 4 to this procedure was approved on May 29, 1987. NRC review of the revised procedure reveals that the necessary changes have been adequately incorporated. This item is closed.

LER 327/87-001, Trip Setpoints for Air Circuit Breakers (ACBs) Incorrect. The trip setpoints for ACB's on shutdown boards that feed control and auxiliary building vent boards were incorrect due to a design error. ECN L6883 has been issued and the loads have been analysed to determine proper trip setpoints. WP 12636 has been issued and is being worked. The work required to satisfy this LER has been completed. Licensee's corrective actions appear to be acceptable. This item is closed.

LER 327/87-02 , Failure to Cycle Test Six Fire Protection Valves per TS Surveillance Requirement 4.7.11.1.d Due to an Inadequate

Procedure. The inspector reviewed revision 11 to surveillance instruction SI-172, Fire System Testable Valve Cycling, and verified that the six valves have been included. These valves are not critical safety system components per SQA-134, Critical Structures, Systems and Components (CSSC) list. The inspector also reviewed completed data sheets indicating these valves had been satisfactorily cycle tested per SI-172 on August 5, 1987. This item is closed.

LER 327/87-035; Diesel Generator 2B-B Start During Replacement of Fuses. The licensee was unable to reproduce the event. Subsequent testing verified that all diesel generators started as required when a fuse was removed. Conclusion: event may have been generated by momentary high impedance in relay (ES2BY) circuit causing only that relay to pickup. Licensee's investigation and results appear to be adequate. This issue is closed.

LER 327/87-038; 6.9.KV Circuit Breakers Not Tested. The 6.9 KV circuit breakers for pressurizer control heaters that are used to protect 1E busses from faults on non-1E loads have not been tested because of an engineering oversight. The licensee has generated surveillance instruction (SI) 737-1A, 1B, 2A, 2B to cover TS surveillance requirement 4.8.3.3 for the 6.9 KV pressurizer control heater circuit breakers. These have been completed. Licensee's corrective actions appear to be acceptable.

LER 327/87-042; Inadvertent Starting of The Fire Pumps During a Loss of Coolant Accident. The licensee submitted an information LER to identify a potential problem with the shutdown power capabilities at the Sequoyah plant. The licensee determined, as a result of an electrical calculation program review, that an event could occur which was not analyzed for in the final safety analysis report. This event was a loss of coolant accident concurrent with the inadvertent starting and running of the station fire pumps. The licensee determined that starting the fire pumps concurrent with a LOCA could potentially degrade the auxiliary electric power system voltage and thereby prevent safety related equipment from performing its intended function. The NRC is reviewing this and other issues in the electrical calculations design review process to determine if the potential problems identified by the licensee are adequately resolved. The licensee determined that no immediate corrective action was necessary because of the mode condition of each of the units. Long term licensee corrective action will include administrative compensatory measures on the control of the unit 2 fire pumps. During its operational readiness inspection, the NRC will examine all administrative compensatory measures employed by the licensee prior to the startup of either units. The generic issue of compensating issues for degraded auxiliary electric power systems will be addressed as a topic in the NRC operational readiness inspection 327, 328/87-73. This LER is closed.

LER 327/87-024; 2-FCV-74-3 Exceeded Allowable Stroke Time. The local stroke time of a flow control valve 2-FCV-74-3 exceeded the maximum allowable stroke time due to an inadequate procedure. The inspector reviewed the stroke time test results for 2-FCV-74-3, which showed an acceptable local and remote test after maintenance on June 12, 1987. TVA's review data comparing the most recent local stroke times for all power operated valves which did not identify any additional discrepancies, was also reviewed. A review of the applicable surveillance instructions indicated that they have been revised to require comparison of local stroke times to the maximum allowable. The licensee's corrective actions appear to be adequate. This item is closed.

LER 327/87-023; Inspection of Ice Condenser Assemblies. Inspection of the ice condenser vent assemblies has not been performed in accordance with the TS due to lack of guidance in the surveillance instruction. The licensee determined that the total requirements of TS surveillance requirement 4.6.5.3.2.b was not being met due to ice condenser vent assemblies being incorrectly interpreted as being the intermediate deck doors. This interpretation did not support an inspection of the vent curtains for free movement. The immediate corrective action was to verify free movement of the vent curtains. Long term corrective action revised surveillance instruction (SI)-108, Ice Condenser Doors, to provide proper guidance for ensuring free movement of the vent curtain during conduct of the TS surveillance requirement. The inspector reviewed revision 9 to SI-108 and found that it contained adequate guidance for ensuring free movement of the vent curtain in step 6.2.4. The licensee's corrective action appears adequate. This item is closed.

LER 327/87-046; Possible Dilution of The Emergency Core Cooling System During Large Break LOCA Events. This LER was issued by the licensee for information only. The licensee is currently reviewing the possible safety implications that a given plant configuration would have on a post-LOCA long term cooling condition. The concern involves the possibility that nonborated water from such sources as the essential raw cooling water system, high pressure fire protection system, primary water system and component cooling water system will affect the long term shutdown capability of the unit. This issue is being handled as a startup item by the licensee. The potential generic issue has been transferred to the NRC, Office of Special Projects for disposition. This LER is closed.

b. Reviewed LERs which Remain Open

LER 327/87-030; Blown Fuse in Emergency Start Circuits Result in Spurious Emergency Diesel Generator Starts on Two Occasions. A review of the fuses indicates that the failed fuses came from (FLAS-5) lot no. 3. As of July 13, 1987, 69 FLAS-5 fuses have failed with 67 confirmed from lots 2 and 3 and two indeterminate. A change in manufacturing process was initiated (between lots 3 and 4) by the manufacturer as improvements in the production process. The licensee is in the process of changing out all FLAS-5, lot 2 and 3 fuses. The emergency diesel generator, emergency start circuit fuses have been verified or replaced with fuses from lots manufactured after lot 3. The licensee's corrective actions appear to be acceptable. This issue remains open pending completion of NRC review of the fuse replacement program.

(Open) LER 327/86-039, Surveillance Requirements Not Performed Because of Inadequate Procedures. Licensee event report (LER) 327/96-39 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 inspectors reviewed surveillance instruction SI-108, Ice Condenser Doors. The procedure had been revised to include verification of the inlet door position at the local panel as required by the technical specification. This item is closed. Surveillance instruction SI-268-3, periodic verification of the P-4 interlock, was written and provides required testing for the "turbine trip on reactor trip" function of the reactor trip P-4 permissive. Testing of the "main feedwater valve closure on low reactor coolant system average temperature with reactor trip" function of the reactor trip, P-4 permissive interlock is in the process of being evaluated. A draft technical specification interpretation entitled "technical specification interpretation log No. 94 revision 1-definition of total interlock function" dated October 7, 1987 was written and discussed the total interlock function as including the process input, solid state protection system (SSPS) logic (if any), and output function. The output function would include verifying the output of the logic circuitry including the energizing of the SSPS master relays for ESF permissives and verifying the logic through the output of the SSPS undervoltage card for reactor trip permissives. The requirement to test the slave relay output contacts as a surveillance instruction (SI) requirement of the technical specifications is the issue under TVA review. The P-4 permissive slave relay contact for the main feedwater valve closure is presently tested but not by a technical specification surveillance instruction. This item will remain open pending review of TVA action concerning the slave relays.

(Open) LER 327/86-048; Inadequate Verification of ECCS Flow Due to Procedural Inadequacy. As previously stated in report 87-36, procedure SI-260.2.1 is to be instituted and procedure SI-260.2 is to be revised to ensure that Centrifigal Charging Pumps are tested under conditions as specified in the TS. Completion of these actions is not required for Unit 2 restart, but is required for restart of Unit 1. Conversations with licensee personnel reveal that these corrective actions are not yet complete. This item remains open.

(Open) LER 328/87-041 revision 1, Loss of RHR Flow Resulting From Mispositioning of a Breaker Due to Personnel Error. A student assistant unit operator (AUO) opened a circuit breaker while performing a routine surveillance to verify alignment of breaker 2 on the 120 VAC vital instrument power board 1-IT. Licensee's actions prevented damage to RHR and flow was restored within 4 minutes. The student AUO was counselled on the necessity for good communications, of following procedures (attention to detail) and the consequences of failure to do so. The licensee's specific corrective actions appear to be acceptable. The generic issue of allowing student operators to unilaterally operate plant equipment will be reviewed at a later date. This item remains open.

(Open) LER 327/86-042, Two Surveillance Requirements Not Performed Because of Inadequate Procedures. As previously stated in report 87-36, the licensee has requested relief from American Society of Mechanical Engineers (ASME), section XI, subsection IWP-3100, for several safety related pumps because of possible damage to the pump by throttling the pump miniflow recirculation valves during testing. Current information obtained from the licensee indicates that the granting of such relief is in the final approval process at NRC, and will be issued in the near future. This item remains open until the relief request is granted.

(Open) LER 327/86-020, Failure to Perform a TS Required Quarterly Functional Test. As previously stated in report 87-36, procedures SI-244, Periodic Functional Tests of Radioactive Effluent Monitoring Instruments and SI-244.2, Peridodic Functional Tests of Radioactive Effluent Monitoring Instruments were to be revised to include a functional test for channel F-15-43. These procedures have been revised by the licensee. However, NRC review has identified several questions concerning the adequacy of the tests. These questions are; (1) the procedures provide no guidance as to what position flow control valve FCV-15-43 is to be left upon completion of the test; (2) is flow indicating controller FIC-15-43 required to function in both "auto" and "manual" modes, or in the "manual" mode only. The latter question is due to conflicting statements in licensee generated reports. Attachment 1 to PRO 3-86-031, under "additional comments", stated "F-15-43 is used for auto isolation flow control", while the "analysis of event" section of the LER report states that "this instrument provides only an alarm on the plant process computer" and that "the flow channel does not provide an isolation function"; and (3) presently, SI-244 requires that this channel in unit 1 be tested in the "manual" mode only, while SI-244.2 requires that both "manual" and "auto" modes be tested in Unit 2. As a result of these questions, additional information and/or action is required from the licensee before this item can be considered complete. This item remains open.

(Open) LER 327/87-027; Surveillance Requirement Was Not Fulfilled Because Four Essential Raw Cooling Water (ERCW) Valves Were Not

Verified in the Correct Position. This event occurred because the subject valves were only verified to be in the correct position (throttled) every 90 days in accordance with surveillance instruction (SI-682), ERCW Flow Balance Valve Position Verification. Technical Specification surveillance requirement 4.7.4.a required them to be verified in the correct position every 31 days. These valves had been verified to be in the open position every 31 days in accordance with SI-33, "ERCW Valves Servicing Safety-Related Equipment." TVA concluded that because the valves were tagged as throttled valves and the knowledge of plant operators, these valves had remained in the throttled position. Corrective action was to delete the four subject valves from SI-33 and revise SI-682 to ensure that these valves are verified in the correct position every 31 days. SI-682 was also to be rerun prior to restart.

The NRC inspector confirmed that SI-33 and SI-682 have been revised as indicated. However, review of the pertinent documentation has resulted in the following concerns related to the performance of these SIs and the evaluations of discrepancies identified.

- The LER analysis does not address the question of how operators could sign off SI-33, verifying valves in the open position, if they were in fact in a throttled position.
- A review of SI-682 data packages performed in 1986 and 1987 indicates that a number of valves have been found improperly positioned at each inspection. Mispositioning ranged from one turn or less to the completely opposite position from that specified (for open or closed valves). Numerous potential reportable occurrence (PRO) reports have been generated with no apparent identification or correction of the root cause. PRO investigations do not appear to be accurate or adequate.
- Completed SI-682 data packages had not always listed improperly positioned valves in a deficiency log and there were inadequate dispositions of the discrepancies (June, September and December SI-682 performances).

The plant operations review staff (PORS) is reviewing these concerns and is performing an indepth root cause analysis. Pending NRC review of this reassessment by TVA this item remains open.

9. Sequoyah Requalification Program Corrective Actions

Two license examiners of Region II's Operator Licensing Section conducted an unannounced inspection of Sequoyah's corrective actions for deficiencies noted during the December 1986 requalification examination. Of particular interest were actions taken to upgrade training of Reactor Operators on the bases for steps in the Emergency Operating Procedures (EOPs).

On October 8-9, 1987, the inspectors observed ongoing requalification training for a group of operators involved in the last week of the 1987 requalification cycle. This consisted of simulator and classroom training, a final written examination and a graded simulator evaluation. Lesson plans, examination questions and simulator scenarios utilized in requalification training were also reviewed with the focus being on EOP related areas.

The findings of this inspection are as follows:

a. The requalification lesson plans for EOP training contained learning objectives which specifically addressed questions asked on previous NRC examinations and did not incorporate any further objectives which would be appropriate for this training. The objectives should be more broadly based to cover the pertinent topics within the lesson plan, not just previously asked NRC examination questions.

b. Specific requalification lesson plans did not exist for coverage of individual emergency procedures. It was noted that some questions related to procedural bases were developed, however, there were no requalification learning objectives associated with these questions. Lesson plans utilized in the hot license training program were reviewed and could readily be adapted to the requalification program.

c. The written examination questions utilized for evaluating the requalification training in EOP usage were phrased exactly the same as the learning objectives. No other questions were developed to examine this area, and further effort should be conducted to create a broader sample of questions to ensure a valid examination can be generated for each requalification group.

d. Simulator training covered the learning objectives, however, the instructor-to-student ratio for training sessions was 1:5 and for final evaluations was 2:4. Consideration should be given to improving this ratio in order to provide more attention to individuals within the crew and equalize the burden on the instructors to control the simulator scenario as well as evaluate the operators.

During this inspection, the corrective actions initiated by the facility were identified and their implementation was found to be adequate. As noted above, however, further effort is required to improve requalification training on the EOPs. These findings were presented to facility staff members at an exit meeting on October 9, 1987.

10. Plant Operations Review Committee (PORC) (40700)

The inspector conducted a functional review of the PORC which performs as the onsite safety review committee. This functional review was intended to evaluate if the activities of PORC could support the heatup and eventual startup of unit 2. The site is currently changing to a qualified reviewer concept as a result of TS change 87-34. This review however, has been conducted prior to the implementation of the TS change. The inspectors observed the PORC review proposals which affected nuclear safety including issuance of plant procedures and changes thereof, modifications to systems and equipment, and unreviewed safety question determinations (USQDs). The inspectors attended PORC meetings conducted on October 7, 8, 13, 14, 26, and 28. The inspectors had the following comments:

- One plant modification reviewed by PORC on October 8, 1987, addressing manholes which held safety related cables, was presented to PORC without an approved USQD. PORC rejected the modification.
- One plant special maintenance instruction (SMI) addressing thermal excursion of piping had been rejected by a previous PORC. The observed PORC was not aware of why the SMI had been rejected and had to request information from the individual that was presenting the SMI to explain why the previous PORC meeting had rejected it. This is an example of a loss of continuity with respect to PORC oversite activities. In addition, the individual presenting the SMI stated that the thermal property being tested for would result again because of a lack of a program to control field routed lines and conduits with respect to thermal interference. The SMI was rejected by PORC a second time.

No violations or deviations were identified.

11. IE Bulletins (92701)

IE bulletins (IEB) 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 IE bulletins. 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 TS conditions did not appear to occur.

IEB 83-01, Salem Anticipated Transient Without Scram. The inspector reviewed licensee response dated March 4, 1983, A27 830304 010, and maintenance instruction MI-10.9, Removal, Inspection, Lubrication, and Replacement of Control Rod Drive MG Set, Reactor Trip, and Bypass Circuit Breakers. After a review of the supporting documentation and the implementation of the actions required initially by this bulletin the licensee's actions were determined to be adequate. This item is closed.

12. Inspector Followup Items

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 327/87-11-01, Employee Concern Program (ECP) Element 11301, Design of Plates Rev. 6. The inspector determined that the specific concerns addressed in this item were to be closed out by the safety evaluation report (SER) to be completed by NRC. This item is redundant to that closure process. Therefore, this item is closed and element report 11301 will remain open until the issuance of the SER.

(Closed) IFI 327/87-11-03, Pipe/Fittings As Related To Construction. This is part of the on-going employee concern element report 17105 Revision 2. Parts of the essential raw cooling water system were changed from carbon steel to stainless steel without a quality seismic analysis being performed. This has now been accomplished by the civil section of TVA's division of nuclear engineering. IFI 327/87-11-03 is redundant to element report 17105 Revision 2 which will be addressed by the NRC in an SER. The following references were reviewed: (1) TVA's Engineering Change Notice L-5009; (2) Memorandum from H. L. Abercrombie to D. W. Wilson dated, February 6, 1986, (RIMS B 25 870312040); (3) Memorandum from J. A. Southers to Sequoyah Engineering Project Files dated March 25, 1987, (RIMS B 25 870325048); and (4) Memorandum from D. C. Hatcher to Sequoyah Engineering Project Files dated, April 8, 1987, (RIMS B 25 870408078). This IFI is closed. The element report 17105 will remain open pending issuance of the SER.

(Closed) Observation 327,328/87-06-EEB-1, Battery and Charger Sizing. This observation was related to errors in the calculation for sizing of the class 1E batteries. The NRC design calculation review team reviewed TVA's revised calculations and found them acceptable. This calculation was performed using a maximum inverter load of 17.5 KVA instead of its name-plate rating of 20 KVA. TVA's electrical engineering branch informed the team that they intend to establish design and administrative limits to keep inverter loads less than 17.5 KVA. TVA also committed to reevaluate sizing criteria for the battery and charger whenever the DC system load changes. Design criteria procedure SQN-DC-V-11.6 has been changed to include in step 1.2.1 system description..."The maximum allowable load to each inverter is 15 KVA." Also, drawings 45N703-2, 45N703-3, 45N703-4, and 45N706-1, 45N706-2 45N706-3, 45N706-4, and 45N703-1 have been revised to include the note "although each vital inverter is rated to deliver 20 KVA, the maximum design load on each is 15 KVA." Sequoyah engineering procedure SQEP-09 has been revised to adequately require a calculation for the battery and charger whenever a change in loading of the DC system occurs. Also, assumption 2.7 in calculation procedure SQN-CPC-004 addresses calculating inverter load at 17.5 KVA and a design limit of 15 KVA. Licensee action on this item is adequate. This item is closed.

(Closed) IFI 327, 328/85-26-02, Ice Condenser Door Blocks. This inspector followup item addressed the control of the devices used to block shut the ice condenser intermediate doors when the units were in modes 4 and 5. The devices are controlled through the use of work requests and verified through routine auxiliary unit operator performance. This item is closed.

(Closed) IFI 327, 328/86-69-02, Effluent Releases In Accordance With Surveillance Instruction SI-400.2. On November 19, 1986, a ten minute transfer was conducted from the high crud tank B to the turbine building sump without the appropriate setpoint change adjustment to the in-line radiation monitor, 0-RM-90-225, being performed as required by surveillance instruction SI-400.2, Condensate Demineralizer Waste Effluent To The Turbine Building Sump - Periodic Continuous Releases. The system, by design, provides for a pathway from the tank to cooling tower blowdown which goes immediately to the river. In this case, however, the transfer was made to the turbine building sump which is only released itself through another monitored pathway. The interpretation given the inspector from the OSP technical staff was that a release occurred only if it went directly to the environment. Therefore, the transfer of liquid from the high crud tank to the turbine building sump was not considered to be a release even though it left the radiological controlled area.

The licensee discovered this occurrence 10 minutes into the event and the transfer was immediately terminated. Once this condition was discovered the licensee complied with the action statements of Technical Specification 3.3.3.9 for this event. Additionally, licensee samples reflected that the activity of the transferred water was within the release limits of Technical Specification 3.11.1.

Personnel involved were reinstructed in the requirements of SI-400.2 for setpoint changes on rad monitors at a January 7, 1987 safety meeting. In addition, the individual involved was privately counselled in following procedures. Because this issue was identified by the licensee, the licensee took immediate corrective actions, and no release to the environment occurred, no violation will be issued and IFI 327, 328/86-69-02 is closed.

(Closed) IFI 327, 328/86-28-19, Unanalyzed Installation of Piping Insulation. This item involved the processing of nonconformance report SQN-QAB-8105 and the necessity to determine if pipe insulation affected the structural integrity of safety related piping. The issue was addressed by the Watts Bar nuclear plant employee concerns task group (ECTG) and in a letter (Brown/Abercrombie, T25 870106 855) dated January 6, 1987. No structural issues as a result of insulation appear to exist at Sequoyah. This item is closed.

(Closed) IFI 327, 328/86-71-08, Surry Plant Feedwater Line Rupture. Following the Surry feedwater line rupture this IFI was opened to ensure that the licensee initiated a program to evaluate the occurrence of wall thinning in the feedwater system. The licensee initiated a voluntary program employing various inspection and testing techniques. This voluntary program was implemented prior to the issuance of IEB 87-01, Thinning of Pipe Walls In Nuclear Power Plants, dated July 9, 1987. The licensee responded to the bulletin in a letter (Gridley/NRC, L44 870918 806) dated September 18, 1987. All further NRC questions concerning the adequacy of the licensee's feedwater system pipe thinning will be addressed within the bounds of the IEB. This item is closed.

(Closed) IFI 327, 328/84-24-04, and 327, 328/86-60-07, Generic Fittings. This item was initially closed in inspection 327,328/86-15 and subsequently reopened in inspection 327, 328/87-43 to reevaluate all aspects of the seal table tube ejection event. The inspector reviewed the licensee's evaluation of the generic (mixed) fittings as contained in problem identification report (PIR) SQNEEB87131. This PIR referenced a report from Singleton Materials Engineering Laboratory (SME). This report was conducted to determine the adequacy and safety of compression fitting assemblies which were installed contrary to manufacturers' recommendations and/or using mixed manufacturer fitting components. The SME report concluded that regardless of various improper assembly techniques, the assembly is acceptable if the joint was assembled to leak tightness as displayed during hydrostatic testing or in-service leak checking. Various low-amplitude (service) vibration and high amplitude (seismic) vibration tests were performed, as well as axial tension (pullout) tests to support this conclusion. The licensee has performed walk-down inspections of various plant areas and found no leaking fittings on instruments or drain lines. A procedure (MMT-28) was written to implement formal fitting assembly requirements. Site personnel have been trained in the use of this procedure and are scheduled for periodic retraining. This item is closed.

(Closed) 327, 328/87-06 Observation EEB-3, 120V AC and DC Solenoid Valve Voltage. This observation noted that electrical calculations of cable impedance do not consider cable slack, high ambient temperature, and drops across connections. Additionally, assumptions are not listed in a dedicated section. TVA's technical justification is documented in B43870529908 and is included in TVA's response to the NRC dated July 2, 1987. Nuclear engineering procedure (NEP)-3.1, Calculations, requires assumptions to be listed and documented and includes a sample format. The licensee's response, as described in the July 2, 1987 letter, is adequate. This item is closed.

(Open) 327, 328/86-27 D.3.3-1, Pipe Support Friction Design. This item described TVA's failure to analyze pipe supports for friction forces due to thermal displacements. TVA has responded to the NRC in letters dated July 28, 1986, December 31, 1986 and August 21, 1987, with a program to evaluate 60 randomly selected supports on major systems. Design criteria SQN-DC-V-24.2, Supports For Rigorously Analyzed Category I Piping, defines the requirements used in the evaluation program. The general, restart criteria contained in volume 2 of TVA's nuclear performance plan (NPP) have been made specific to pipe supports in CEB-CI 21.89, Modification Priorities for Pipe Supports on Rigorously Analyzed Category I Piping.

These criteria have been transmitted to the NRC in a letter dated, August 31, 1987, and will be used to determine whether an action is required pre-or/post-startup. This item will remain open until the evaluations are complete.

(Closed) IFI 327, 328/87-02-06, Review Supplemental Report on Glycol Valve Stroke Failure. The inspector reviewed revision 1 to LER 327/084-070. TVA's further investigation into the valve failure concluded that failure was due to a buildup of sediment on the stem, was an isolated incident and that no further action was required. The maintenance history of all valves of this type were reviewed by this inspector. Total failure to operate has not been a problem with these valves. Maintenance attention to these valves has been increased and improved since the incident. This inspector also reviewed the complete history of stroke times for the two valves that failed and the work requests that overhauled them after failure. No degradation trends in stroke times were apparent, either before or after the event. Although the inspector does not concur that the failure to operate in this event can clearly be blamed on "sediment buildup," the failure does appear to be an isolated case. This item is closed.

(Closed) 327, 328/86-55 observation 6.15, Testing of 0.5 Second Time Delay Relays. This item noted that the 0.5 second time delay relays on critical safety systems component (CSSC) systems had not been calibrated. TVA had transmitted to the NRC reportable occurrence report SQRO-50-327/871010 on February 27, 1987, which stated that a QA audit had found that numerous relays, switches and controllers had not been identified on any procedure to require calibration. Revision 1 to Maintenance Instruction MI-13.1.3, Setpoint Verification and Calibration for Time Delay Relays Associated with Load Shedding Logic, was issued on June 9, 1987, and these relays have now been calibrated. TVA has committed, in its response to NRC, to incorporate the subject instrumentation into Sequoyah standard practice SQE-8, Control of Installed Permanent Process Instrumentation, by December 31, 1987. The licensee's actions are satisfactory and this item is closed.

(Closed) VIO 327, 328/86-68-05, paragraph c, section 2.4.4 deficiency D-2.4-5, Loose Debris in Valves 2-FCV-1-17 and 2-FCV-1-18. NRC inspectors reviewed the July 16, 1987 TVA response to this portion of the violation and the corrective action taken. The loose debris in the limit switch area of the valves was corrected under work plan 12305. Licensee action on these items is adequate. This portion of the violation is closed.

(Closed) IFI 327, 328/86-48-05, Reportable Occurences. Ten potential reportable occurrences (PROs), identified by the licensee, with the status of corrective actions were reviewed by the inspector. Five of the items were reported as completed, the remaining five were reported as in progress. The five items still in progress are:

Item 3 - MOV (CSSC) MOVATs testing
Item 7 - Pressurizer instrumentation

Item 8 - RHR switches
Item 9 - Backfill instrument sensing lines procedures
Item 10 - Condenser vacuum exhaust flow monitor

The licensee assigned Sequoyah Activities List (SAL) tracking numbers to the above items as follows:

Item 3 - SAL 0317
Item 7 - SAL 0023
Item 8 - SAL 0699
Item 10 - SAL 0699

Item 9 was assigned a corporate commitment tracking system (CCTS) number NCO 870114003 and NCO 870114005. All items are reported as complete for Unit 2 except item 9 (NCO 870114003) which is being evaluated by TVA licensing for satisfactory completion per the commitment. A third commitment, evaluation of outgassing of sense lines for devices required to operate during and after design basis accident (DBA) was not provided for review. Discussions with TVA personnel indicated that the outgassing evaluation was not part of this PRO concern. Based on the above, and TVA's completion of the procedures required for backfill, this item is closed.

(Closed) IFI 327,328/86-49-05, Apparent Deficiency of Adequate Instrumentation to Monitor Cooling Water to the Emergency Diesel Generators (EDGs). A valid emergency start, coincident with a blackout could result in severe damage to the EDGs under certain operational ERCW conditions (reliance on cross connect system). Sequoyah has taken steps to reduce the risk by implementing attachment 1, page 3 of OSLA-30, dated June 11, 1985. This attachment requires the EDG to be made inoperable if normal emergency raw cooling water (ERCW) supply is removed from service. It also requires the assigning of a dedicated operator for the EDG's, if two (2) trains are made inoperable by the action. This corrective action should eliminate damage to the EDGs, due to administrative action making ERCW unavailable. A review of the FSAR did not locate a requirement for flow measurement of ERCW to the EDGs. The generic issues associated with this issue will be addressed under "compensatory measures for defeated safety functions" during the NRC operational readiness review. This item is closed.

(Closed) IFI 327, 328/84-43-06, Training For Reactor Vessel Head Vent (RVHV) System. The training for the RVHV system was reviewed as part of a recent emergency operation procedures insepction conducted November 2 through 5, 1987. However, during a review of Sequoyah action list (SAL) Item 13, NRC inspectors noted that there were several discrepancies with the operation of the RVHV system. The discrepancies were noted in the performance of work plan (WP) 10597 which was written to perform post modification testing (PMT) of the system. This PMT tested the operability of the RVHV system. During the performance of this PMT many exceptions and deficiencies were noted. The testing to prove the operability of the system was conducted in October of 1983. All deficiencies except for one was addressed by a department of nuclear engineering document entitled, Interim Review and Approval of Post Modification Test Results, PMT-39, dated June 6, 1986. The final discrepancy, a change to the controller to provide more accurate valve position for the head vent throttle valves, is to be completed under ECN 5160 during a future outage. The post modification testing ECN 2777 along with workpackage 19597 was closed out on a partial modification completion form which included an attached USQD dated January 13, 1987. During the inspectors review of the test deficiencies and associated documentation, it was noted that during testing when the block valve (2-FSV-68-395) was opened, both throttle valves (2-FSV-68-396, 2-FSV-68-397) inadvertently opened to 60% of full open and then shut within about 5 seconds. This inadvertent opening was noted during the resolution of deficiency DN-6 in work package 10597, PMT-39, Appendix D, Deficiencies and Exceptions. However, the inadvertent opening did not appear to have been made part of the deficiency reviewed by DNE in either the PMT or USQD evaluation. A review of the functional restoration guides (FR-I.3, revision 0, page 9 and 10, and FR-H.1, page 10) indicated that the procedures do not address the system abnormality. During discussions with the TVA staff, TVA indicated that the problem with this target rock valve was well known. However, when questioned by NRC inspectors, some control room operators were not aware of the problem.

The NRC inspectors requested the licensee address the following concerns:

a. Why does the ACTION/EXPECTED RESPONSE column, step 19 of SQNP FR-I.3, Unit 1 or 2, revision 0, page 9, and step 13 RESPONSE NOT OBTAINED, SQNP FR-H.1, Unit 1 or 2, revision 1, page 10, not provide precautions or describe the expected system response?

b. Provide documentation which indicates that the vendor has evaluated the fact that the Target Rock throttle valves may "pop" open when subjected to inlet pressure transients.

c. Verify that DNE agrees that the valve performance is acceptable and documented by an unreviewed safety question determination (USQD).

d. Provide documentation that the problem has been addressed to the operating crew.

Items (a) and (d) above, regarding the functional restoration guides and training on valve response, was addressed by the recent NRC EOP inspection. Item (b) above was addressed by the licensee by providing the inspector with copies of Target Rock report #2866, Solenoid Valve Response to Inlet Pressure Transient (December 17, 1980), along with ASME publication 81-PVP-39 (April 1981). Item (c) above was addressed by the licensee by providing the inspector a copy of the test deficiency review and approval memorandum (D. W. Wilson to H. L. Abercrombie, RIM B25 860606 012).

The inspectors reviewed the above information and determined that items (a), (b), and (d) were adequately addressed. However, item (c) above was determined to be a violation of corrective action requirements, in that, the department of nuclear engineering review of the test deficiencies documented in deficiency report 2-PT-789 (dated April 12, 1984) did not specify that emergency procedures should be changed and personnel trained to cope with the described condition. Additionally, the evaluation of the test deficiency did not reference the vendor's test report as justification for accepting the test deficiency. This item, which involved questionable performance of equipment is identified as a violation of the 10 CFR part 50, appendix B requirements for effective corrective action, and is identified as violation 327, 328/87-65-01.

(Closed) IFI 328/86-62-04, Review of Final Revision of AI-19, part IV, Plant Modification After Licensing, To Ensure Time Commitments for ECN Closeout and Drawing Update Is Established. During a review of the transitional design change program conducted in November 1986, and June 1987, (inspection reports 327, 328/86-62 and 327, 328/87-42) the inspectors identified a weakness in the new program. Specifically, there was no commitment on TVA's part to ensure timely closure of ECNs. This issue was seen as a significant weakness, because the lack of timely closure was determined by TVA and the NRC to be a contributor to past design control problems. Additionally, the inspectors determined that without timely closure of ECNs the requirements of 10 CFR 50.71, which requires the annual update of the FSAR to be current within 6 months of the modification, could not be assured. The licensee determined that it would take approximately 6 months to complete the closure of the ECNs from the point of field completion and operations acceptance of the modification. This time commitment was established by revision 24 of AI-19, part IV and appears to reflect the actual time needed by the licensee to complete all reviews and closure effort. However, this 6 month closure of the ECNs will not ensure that the requirements of 10 CFR 50.71 are satisfied. Specifically, TVA uses the closure of the ECN and not the field completion of physical work, as the starting point of updating the FSAR. This process could cause as much an 18 month lag between the actual plant configuration and the FSAR.

The above issue was discussed with the licensee and a commitment to start the FSAR update process at field completion vs. ECN closeout was made by the licensee. However, the licensee should have changed the FSAR update process as part of the transitional design control program based on corrective action report (CAR) 86-04-21 and the findings in inspection report 327, 328/86-62. This lack of complete corrective action is identified as a violation of 10 CFR part 50, appendix B, criterion XVI and is a second example of violation 327, 328/87-65-01. IFI 328/86-62-04 is closed.

13. Restart Test

a. SI-26.2B Loss of Offsite Power With Safety Injection D/G 2B-B Containment Isolation Test, Revision 14, and STI-78, D/G 2B-B Load Sequence Test, Revision 0. STI-78 was performed in conjunction with SI-26.2B. NRC inspectors observed the performance of all of STI-78 and test 1 and test 2 of SI-26.2B. Test sequence 1 and test sequence 2 tested loading/shedding and safety injection/phase B actuation. NRC inspectors attended the shift turnover briefing prior to the testing. The off-going and on-coming shift supervisors discussed plant activities for the previous shift, maintenance activities performed during the previous shift and the performance of the load shedding, and the containment isolation test. NRC inspectors attended a second briefing for the on-coming operating crew for the performance of the test. The briefing covered personnel assignments, location of personnel, communications, and the details of how the timing would be conducted. The importance of good communication was stressed by the shift supervisor. Each person responsible for sections of the test was given highlighted copies of applicable sections of the test. The following observations were noted during the testing:

- All pressurizer heaters were energized rather than just group B and C heaters as specified by the test procedure. The assistant test director told the unit operator to energize pressurizer heaters. The unit operator energized all heaters and then questioned the assistant test director as to whether he wanted all heaters on. The lineup was promptly restored to the correct lineup. The unit operator did not, like other test personnel, have a copy of the test procedure to check his particular actions. During subsequent steps it was noted that the unit operator was careful in requesting information on what switches and controls to operate by number. However, he was not given a copy of the test. Having a copy of the procedure or repeating back exact verbage of the step would have prevented the occurrence.

- During the initiation of the safety injection (SI) signal all personnel were on location to start recorders from a countdown and "go" signal from the assistant test director. The assistant test director did not check to see that the unit operator was ready, consequently all recorders were started on the "go" signal, but the safety injection signal was not activated. The test was delayed to change paper in the recorders and rerun the step. The test director was not directly overseeing this portion of the test which may have contributed to the error.

- During the tests, the unit operator secured an SI pump with the SI signal still inserted. This resulted in the SI pump stopping and immediately restarting. The assistant test director noted the mistake and reminded the unit operator to place the pump in the "lockout" position. The pump was secured due to isolation valve leakage that was increasing pressurizer level. Securing the pump was not part of the test procedure, but was allowed by precautions in the procedure to control plant parameters.

- Loads normally powered from vital inverter 1-4 was inadvertently deenergized during the test when the 6.9KV shutdown board was blacked out in step 6.2.8. During the shift turnover briefing, it was discussed that maintenance was being performed on the 1-4 vital inverter. It was not brought out that the inverter was powered from its maintenance supply (the 480V shutdown board) nor that the maintenance supply would be deenergized during the blackout of the 6.9 KV shutdown board. When power was lost, the CO_2 fire doors for the 2B-B board room tripped. This caused considerable confusion in the control room as the cause was not known. The test was stopped and eventually it was determined that the vital inverter had been deenergized. The problem could have been prevented by a careful review of the effects of plant maintenance on testing that was to be conducted.

The following material deficiencies or potential material deficiencies were noted during the testing:

- The 2B-B diesel exhaust fans did not restart when the diesel started and returned power to the 6.9KV shutdown boards. The probable cause was determined to be a failure of relay R1X2. A work request was written to investigate and correct the cause of the failure. The relay problem did not effect continuing the test and was noted as a test deficiency.

- During a review of the recorder data from the recorder connected to the 6.9KV shutdown board, it was determined that the recorder had stopped working during the SI activation test. Investigation revealed that a lead had become loose causing the malfunction. The malfunction resulted in having to rerun a section of STI-78, Load Shedding. Steps 6.5.3 through step 6.5.6 had to be repeated. These steps pertained to the simultaneous starting of a main fire pump and containment spray pump on the 6.9KV shutdown board that was being supplied by the 2B-B diesel generator. The inspector determined that the licensee's actions were in compliance with AI-47.

- One procedure problem was noted with SI-26.2B, step 6.2.2.2 in resetting the phase A isolation signal. The step read to reset the phase A isolation signal using reset button HS-30-63D rather than using HS-30-63D and HS-30-63E. The result was that some of the relays in the following steps were not deenergized as required by the procedure. The test was stopped while a temporary change to the test procedure was obtained. The test was recommenced after the temporary change was processed and approved.

- During the test the 2B-B diesel generator fuel oil transfer pump did not restart as expected upon a low day tank level. The 2B-1 diesel generator fuel oil transfer pump was tagged out for maintenance. Fuel oil to support the diesel during the test was transferred manually. The licensee is investigating why the fuel oil transfer operation did not function as expected. The failure of the fuel oil transfer pump to start did not affect the performance of the test.

The discrepancies identified during the performance of the above testing activities was discussed during the exit meeting with Sequoyah plant management. The NRC will further review these discrepancies under unresolved item (URI) 327, 328/87-65-04.

b. Followup of open Sequoyah activity list (SAL) item 51 L-finger Gap Adjustment (MI-11.2A). This SAL item addressed the resolution of Limitorque valve maintenance regarding adjustment of limit switch L-fingers. The inspector reviewed the Limitorque vendor manuals available in Sequoyah document control facility. No provisions or reference to the process of measuring the gap or bending the L-Fingers could be found in the manuals reviewed. The vendor manuals do suggest that the geared limit switches should be replaced as an assembly. The diagrams pictured show the L-fingers as part of the assembly. Discussions with TVA personnel indicated that the vendor is planning to include the measurement of the L-finger gap in future maintenance instructions. TVA also presented a Commonwealth Edison internal memo that, in part, states: the vendor, Limitorque, has addressed the concern of insufficient contact pressure by inspection of the L-finger gap and bending the L-finger to achieve a 1/32-inch nominal gap. This issue is closed.