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

Report Nos.: 50-327/85-17 and 50-328/85-17

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 1 and 2

Inspection Conducted: May 6 - June 5, 1985

Inspectors: A. J. Ignatonis 7/29/85
for K. M. Jenison, Senior Resident Inspector Date Signed

for A. J. Ignatonis 7/29/85
L. J. Watson, Resident Inspector Date Signed

Approved by: S. P. Weise 7/30/85
S. P. Weise, Section Chief Date Signed
Division of Reactor Projects

SUMMARY

Scope: This routine, announced inspection involved 416 resident inspector-hours onsite in the areas of operational safety verification including operations performance, system lineups, radiation protection, security and housekeeping inspections; ESF walkdown; surveillance and maintenance observations; review of previous inspection findings; followup of events; review of licensee identified items; and in-office review by the Regional staff.

Results: In the areas inspected, four violations were identified.

1) Failure to establish adequate procedures for: a) tests of diesel generator relays; b) limit switch adjustments for motor operated valves; and c) fill and vent of the Reactor Vessel Level Indication System.

2) Failure to follow a radiation protection procedure.

3) Failure to follow procedure for surveillance testing of the emergency diesel generator.

4) Failure to follow procedure for installation of intercell spacers for vital battery V.

REPORT DETAILS

1. Licensee Employees Contacted

   *H. L. Abercrombie, Site Director
   *P. R. Wallace, Plant Manager
   L. M. Nobles, Operations and Engineering Superintendent
   *J. B. Krell, Maintenance Superintendent
   M. R. Harding, Engineering Group Supervisor
   J. M. Anthony, Operations Group Supervisor
   D. C. Craven, Quality Assurance Supervisor
   B. M. Patterson, Maintenance Supervisor (I)
   *D. E. Crawley, Health Physics Supervisor
   J. L. Hamilton, Quality Engineering Supervisor
   *G. B. Kirk, Compliance Supervisor
   *M. E. Frye, Compliance 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 June 7, 1985. Violations described in paragraphs 3, 5, 7, 8, and 10 were discussed. The licensee acknowledged the inspection findings. The licensee did not identify as proprietary any of the materials provided to or reviewed by the inspectors during this inspection. During the reporting period, frequent discussions were held with the Site Director, Plant Manager and his assistants concerning inspection findings. At no time during the inspection was written material provided to the licensee by the inspector.

3. Licensee Action on Previous Enforcement Matters

   (Open) Unresolved Item 327/85-16-03, 328/85-16-03: The inspector reviewed TVA and vendor installation drawings against the field configuration of the battery racks for the 125V vital DC battery banks. The inspector found that gaps existed between the end cells and end stringers of vital batteries I, II, III, and IV. Continued review resulted in the following findings:

   a. The 125V vital DC battery racks for vital batteries I, II, III, and IV were determined to be installed in accordance with approved drawings; however, on April 2, 1985, the TVA Office of Engineering (OE) received an information notice from the battery vendor dated March 27, 1985, stating that there may be a gap between the end cell and stringer of greater than one quarter inch in some Class 1E battery installations.

The original vendor drawings did not show this gap or a spacer installation. The vendor recommended that a spacer be inserted to bring this gap to less than three-eighths of an inch to conform to the configuration used during seismic testing of the battery. The vendor letter indicated applicability to both the Watts Bar and Sequoyah plants. Discussions between Region II and the vendor indicated that the vendor intended for the licensee to install spacers if they wanted to be able to use the vendor seismic test results.

TVA personnel stated that the engineer who received the letter did not associate the letter with Sequoyah for several weeks; therefore, the letter was not received by the Sequoyah site OE personnel until April 18, 1985. The site OE field inspected the vital batteries on April 19 and 20, 1985, and determined that the vital batteries had end gaps greater than recommended by the vendor. A Nonconformance Report (NCR) was written by site OE dated April 24, 1985. The NCR states, "The spacing between the end cell and end stringer of the rack on vital batteries I thru IV was measured and found to exceed one quarter inch required by seismic testing. The fifth vital battery has one cell missing at the present time and no spacer was added." The condition was defined as a "significant condition adverse to quality." The NCR was signed by the responsible Branch Chief on April 24, 1985. This date was corrected on a later copy to May 1, 1985, due to changes after an additional OE review. The NCR was received by the site Office of Nuclear Power (NUCPR) on May 1, 1985.

In memoranda dated May 29 and June 5, 1985, NUCPR rejected the NCR and subsequent Failure Evaluation/Engineering Report (FE/ER), discussed below, stating that the vendor letter was a recommendation and was not an NCR. NUCPR stated that the item would be handled under the Nuclear Experience Review Program.

The TVA review of the installation (on April 19 and 20 by OE and on May 2 by NUCPR) indicated that gaps of up to 2 inches existed at the end of some of the racks of vital batteries I, II, III, and IV and up to 5 inches at the end of the racks of vital battery V. TVA stated that the cognizant NUCPR engineers determined that the lack of end spacers did not have an effect on the operability of the batteries. This evaluation was not documented. On May 2, 1985, TVA initiated immediate action (I4L) maintenance requests (MRs) to install end spacers of an approved material in vital battery banks I, II, III, and IV. On May 2, 1985, NUCPR requested that OE issue a Design Change Request (DCR) or Engineering Change Notice to revise the appropriate drawing to indicate the end spacers. OE provided an informal memorandum on May 7, 1985, which stated that no DCR or ECN was needed for the installation. The inspector reviewed Field Change Requests Nos. 3530 and 3536 which requested updates of the drawings.

The MRs were completed by May 13, 1985. A second set of MRs were issued on May 13, 1985, for vital batteries I, II, III, and IV to install additional spacers in gaps where the rack end stringers were not square and therefore, the one quarter inch requirement was not met. The MRs were completed on May 14, 1985.

The inspector has received additional information indicating that an NCR was written on end spacers on vital batteries for TVA's Watts Bar Nuclear Plant due to a letter received from the vendor, Gould, in May 1984. Sequoyah has the same type of vital batteries as Watts Bar. This Unresolved Item will remain open pending receipt of additional information on TVA's procedures for assuring timely review of nonconforming conditions at its nuclear sites for applicability to other TVA sites. An order dated June 14, 1985, was issued to TVA requiring a complete evaluation of nonconformance handling procedures and an appropriate corrective action plan. In addition, the NRC is also reviewing TVA's corrective action for a violation in this area cited in IE Inspection Report 327, 328/84-38.

b. The inspector reviewed Workplan 11188 which covered installation of vital battery V. The workplan required installation and inspection of the racks in accordance with vendor drawing 410336C, which showed intercell spacers between the batteries, and installation and inspection of the battery cells in accordance with vendor drawing 400197C which did not show the intercell spacers. Technical Specification 6.8.1 requires that written procedures be established, implemented and maintained covering activities referenced in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978. Section 9.e of Appendix A requires procedures for the control of modifications. Workplan 11188 was established for the control of the installation of vital battery V. Workplan 11188 was not implemented in that the installation of the battery racks did not conform to the vendor drawing 410336C, as required by the workplan, since the intercell spacers were not installed when the inspection of the racks took place. This resulted in the failure to install the spacers. Failure to follow the procedure for installation of vital battery V constitutes a violation (327, 328/85-17-01).

A Failure Evaluation/Engineering Report (FE/ER) was issued by OE on May 20, 1985. The FE/ER addresses the seismic requirements for spacings between batteries and battery racks for vital batteries I, II, III, IV, and V. The FE/ER states, "In the absence of spacers, seismic loading could cause failure of the vital battery cells. There is evidence that structural failure would likely occur at the battery terminal posts. Such a failure of one cell causes the loss of the entire battery system. Although it was not possible to analytically predict the seismic behavior of unqualified (without spacers) configurations, a failure of this type must be considered probable." As stated above, NUCPR rejected this FE/ER stating that the conclusion was technically inaccurate. TVA stated in Potential Reportable Occurrences Report (PRO) 1-85-160 that, after inspection of the physical mounting of the batteries and discussions of the manufacturer's recommendations, the lack of spacers did not affect operability and the event was not reportable. TVA also stated that conversations with the author of a Sandia study on aged Gould batteries indicated that seismic test results appeared unaffected by a battery configuration without spacers.

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 new unresolved item identified during this inspection is discussed in paragraph 7.

5. Operational Safety Verification (71707)

a. 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, reviewed tagout records, verified compliance with Technical Specification (TS) Limiting Conditions for Operations (LCO) and verified return to service of affected components. Tours of the diesel generator, auxiliary, turbine buildings and reactor containment were conducted to observe plant equipment conditions, including potential fire hazards, fluid leaks, and excessive vibrations and plant housekeeping/cleanliness conditions. The inspectors verified that maintenance work orders had been submitted as required and that followup and prioritization of work was on-going. During the course of the inspection, observations relative to protected and vital area security were made, including access controls, boundary integrity, search, escort, and badging.

b. The inspectors walked down accessible portions of the following safety-related systems on Unit 1 and Unit 2 to verify operability and proper valve alignment:

Containment Spray System (Units 1 and 2)
Residual Heat Removal System (Units 1 and 2)
Safety Injection System (Unit 2)
Turbine Driven Auxiliary Feedwater System (Unit 2)
Motor Driven Auxiliary Feedwater System (Unit 2)
Condensate Storage Tank (supply and recirculation flow paths)
Upper Head Injection System (Unit 2)
Auxiliary Control Air System
125 VDC Vital Plant Control Power System

In addition, normally inaccessible portions of the following safety-related systems on Unit 1 were walked down to verify operability and proper valve alignment:

Ice Condensers
Residual Heat Removal System
Reactor Coolant System (Pressurizer Safety and Power Operated Relief Valves)
Reactor Vessel Level Indication System
Reactor Vessel Head Vent System
Upper Head Injection System
Lower Containment Air Coolers

While touring containment on Unit 1, the inspector noted that a limit switch on one cold leg accumulator valve appeared to be broken. A maintenance request was written by the licensee and the switch was determined to be an unused annunciation limit switch.

No violations or deviations were identified in these areas.

c. Radiation Protection Control

The inspectors observed Health Physics 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 assure the activities were being conducted in accordance with applicable RWPs. Selected radiation protection instruments were verified operable and calibration frequencies were reviewed for completeness. The inspectors had the following findings:

(1) On May 15, 1985, while conducting a walkdown of normally inaccessible safety-related systems, a Quality Assurance (QA) auditor in the company of two technicians, was observed performing an audit in the Unit 1, number 4 accumulator room. The individual had entered containment under Radiation Work Permit (RWP) 1-85-105 to observe the installation of certain limit switch gaskets. The RWP required that, in addition to other protective clothing, a canvas hood be worn. The QA auditor was observed by the inspectors without a canvas hood. When questioned, the individual stated that his canvas hood had fallen off and that he intended at some later time to retrieve it. Licensee procedure RCI-1, which implements TS 6.11, requires that each employee adhere to radiological work procedures and protective measures, and to report to the appropriate supervisor any differing circumstances. Failure to comply with the protective dress requirements of RWP 1-85-105 is a violation (327/85-17-02).

(2) On May 17, 1985, a Health Physics (HP) technician was observed passing a meter outside a regulated area without frisking or smearing the meter immediately prior to exiting the regulated area boundary. The technician stated that the meter had been frisked and smeared in the decontamination room, and then hand carried through the Auxiliary Building hatch (elevation 690). The meter was placed outside the regulated area, the technician exited

through the portal monitor, retrieved the meter from the unregulated area and entered the HP office.

Licensee procedure RCI-1, Radiological Hygiene Program, and HPSIL-2, Contamination Survey, require only that an HP technician survey the instrument without specifying what actions are or are not acceptable to prevent the spread of contamination outside the regulated area. The general HP practices observed were not in violation of 10 CFR 20, since the meter remained under the technician's control. The licensee has committed both at the monthly exit meeting and in a telephone conversation with the NRC Region II Health Physics Section (TVA-Crawley, NRC-Weddington) to review the above two procedures, to clarify the wording to prevent procedural errors, and ensure that all material is appropriately surveyed and smeared prior to exit from the regulated area. This issue is an Inspector Followup Item (327/85-17-03 and 328/85-17-02).

(3) On June 4, 1985, during a plant tour, the inspector discovered an unattended contaminated tool in an unsealed yellow plastic bag on EL 690 of the Auxiliary Building. Health Physics was called and the bag was surveyed (15,000 dpm reading). RCI-1, "Radiological Hygiene Control," establishes controls on the movement and storage of equipment within regulated areas. These controls were not implemented. This issue is included as a further example of a violation described in IE Inspection Report 327, 328/85-20.

6. Engineered Safety Features Walkdown (71710)

The inspector verified operability of the residual heat removal system (RHR) on Units 1 and 2 by performing a complete walkdown of the accessible portions of the systems. The following specifics were reviewed and/or observed as appropriate:

a. that the licensee's system lineup procedures matched plant drawings and the as-built configuration;

b. that equipment conditions were satisfactory and items that might degrade performance were identified and evaluated (e.g., hangers and supports were operable, housekeeping, etc., was adequate);

c. with assistance from licensee personnel, the interior of the breakers and electrical or instrumentation cabinets were inspected for debris, loose material, jumpers, evidence of rodents, etc.;

d. that instrumentation was properly valved in and functioning and calibration dates were appropriate;

e. that valves were in proper position, breaker alignment was correct, power was available, and valves were locked as required; and

f. local and remote instrumentation was compared and remote instrumentation was functional.

No violations or deviations were identified.

7. Monthly Surveillance Observation (61726)

a. The inspectors observed TS required surveillance testing and verified that testing was performed in accordance with adequate procedures, that test instrumentation was calibrated, that limiting conditions for operation were met, that test results met acceptance criteria requirements and were reviewed by personnel other that the individual directing the test, that deficiencies identified during the testing were properly reviewed and resolved by management personnel, and that system restoration was adequate. The inspector verified that testing frequencies were met and tests were performed by qualified individuals.

The inspector witnessed/reviewed portions of the following test activities:

Calibration of NIS Power Range (NI 42) - Instrument Maintenance Instruction IMI-92-PRM-CAL, Rev. 24

Surveillance Instruction, SI-484, "Periodic Calibration of Reactor Vessel Level Instrumentation (RVLIS) and RCS Wide Range Pressure Channels," Rev. 0

b. Semi-annual Emergency Diesel Generator (EDG) surveillance was conducted by the licensee and observed by the inspectors on May 21, 1985. The following documents were reviewed in order to ensure compliance with the applicable Technical Specifications (TS) in connection with the subject surveillance:

DPSO-SMI-1-DG, Relay Functional Tests for Diesel Generator Protective Relays

SI-102, Inspection of Diesel Generators
MI-10.1, Diesel Generator Inspection
Division of Power System Operation (DPSO) Field Test Manual
Power System Operations (PSO) Quality Assurance Manual

Several examples of procedural inadequacies were identified by the inspectors:

(1) MI-10.1 states in Paragraph 5.0 that:

"The following steps were written to provide a complete check of the emergency diesel generator and associated systems. Since the completion of one step is not a prerequisite for continuing to the next step, it is not mandatory that the steps in this section be completed in numerical sequence.

The intent is not that steps be left out or short-cuts be taken but rather that this instruction be adaptable to unique operating conditions and limitations at the time of performance and that all steps be performed in a timely and professional manner."

Licensee management interpreted this paragraph to mean that each numbered step in the entire paragraph could be completed in any order desired, depending on the individual performance requirements at the time the test was performed. There were approximately 100 separate numbered steps in this procedure covering 21 pages in which certain steps were preceded by notes and cautions requiring action to be completed prior to executing a procedural step. Two examples of the notes written into the steps were:

"QC Holdpoint: Insure MO-2 is used if lubrication is added to generator bearing."

"Note: The following checks shall be made with the diesels not running, the maintenance-auto selector switch in the maintenance position, and the local-remote selector switch in the local position."

The above statements required that their attendant steps have a mandatory orientation with respect to preceding or following steps. In addition, there were several cautions within the surveillance document that also required a mandatory orientation with respect to certain steps. Thus, the guidance on step completion in paragraph 5.0 appeared to be incorrect and could lead to improper procedural performance. The licensee's control of the sequence of critical activities by appropriate orientation of procedural steps is under review by the NRC. This is identified as Unresolved Item (327/85-17-09 and 328/85-17-08).

(2) DPSO-SMI-1-DG paragraph 9.A(3), states the following:

"Trip device 86GA by "A" phase differential relay 87 and verify correct target operation."

This test is performed locally in the EDG building. The test requires energizing relay 87 which causes trip device 86GA (reverse power protection feature) to actuate. In order to make up relay 87, an electrical current must be passed into the circuit using a test source. The above procedure did not specify either how the relay is to be made up or the test source to be used, although the licensee manufactured a special calibrated relay testing kit (TVA - Relay Test Set - Model C - TVA 266151) to perform the function. During the testing witnessed, the special relay testing kit was not used and instead the test engineer chose to use another uncalibrated, uncontrolled kit that generated an uncontrolled signal source.

Step 4.4 of Administrative Instruction (AI) 4, "Plant Instruction Document Control," Revision 49, requires that the prerequisites section of the procedure identify special equipment requirements. The special calibrated relay testing kit described above, intended for circuit testing was not specified in surveillance procedure DPS-SMI-1-DG and reflected an inadequacy in the procedure.

Secondly, DPSO-SMI-1-DG Paragraph 9.A(4) 4.1, states the following:

"Attempt to start the diesel generator from local start without resetting 86GA. Verify that the diesel does not start."

This step did not require the local-remote selector switch to be in the local position, although this position is crucial to the test desired. With the selector switch in the remote position (which was how the surveillance was initially conducted), the test trip device 86GA was not tested because the start signal was blocked by the selector switch position. When brought to the licensee's attention, the test was properly run with satisfactory results.

Procedure DPSO-SMI-1-DG failed to incorporate the required switch position needed to conduct the test. As a result for in the instance above, this in adequacy contributed to test invalidation.

The two procedural discrepancies described above are examples of a failure to establish adequate procedures and constitute a violation (327/85-17-04 and 328/85-17-03). Additional inadequate procedure examples for this violation are discussed in paragraphs 8 and 10.

c. Procedure MI-10.1, paragraph 5.3.1.2.2.4 states that the DPSO technicians are to set up their test equipment prior to the engine start required in paragraph 5.3.1.2.2.5. As observed, the equipment was set up during the completion of paragraph 5.3.1.2.4, after engine start. Paragraph 5.3.1.2.4 states that the operator is to verify that the engine running annunciators (0-M-26 and 0-L-4) are energized when the engine speed reaches 850 rpm. The actual speed at which the annunciation was energized, was approximately 875 rpm; however, the technician erroneously recorded 850 rpm on Inspection Sheet 5. These examples of failure to follow procedure constitute a violation (327/85-17-05 and 328/85-17-04). While the safety significance of these specific examples is minimal, they are indicative of a lack of personnel compliance with procedural requirements.

8. Monthly Maintenance Observations (62703)

a. Station maintenance activities of safety-related systems and components were observed/reviewed to ascertain that they were conducted in accordance with approved procedures, regulatory guides, industry codes and standards, and in conformance with TS. 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.

b. During the Unit 2, cycle 2, refueling outage, the motor operators on the Main Feedwater System (MFW) isolation valves for steam generators 1 through 4 (2-FCV-33, 47, 87, and 100 respectively) were replaced. All work was completed by November 27, 1984. On May 4, 1985, with Unit 2 in mode 2 returning to power, MFW valves 2-FCV-3-33 and 100 would not allow water flow to the steam generators. The restart was discontinued and the unit was placed in the hot shutdown mode. During the examination to determine the extent of the valve failures, it was discovered that each of the two valve stems had sheared from its disc. The disc had remained in the closed position within the valve seat. The stem had suffered brittle fracture failure through approximately three quarters of the diameter of the shaft, in addition to stress failure of the remaining quarter. One of the remaining operable valves was examined employing the use of Motor-Operated Valve Analysis and Test Service (MOVATS) equipment. The Limitorque operator had been installed to use a limit switch to control valve motion in the open direction. The limit switch was set at approximately 97% of full valve travel in the open direction. These MFW system valves are large, fast acting (154 inches per minute) valves. Because of the speed of these valves, the relation of the mass of the disc to the stem diameter and the close proximity of the limit switch setpoint with the full stroke travel of the valve, the disc was impacting with the backseat (causing fracture). This apparently resulted in a stress failure of the remaining portion of the stem on the opening stroke of the valve.

The licensee examined the remaining two MFW system isolation valves and found no indication of stem failure. In addition, the licensee evaluated valves in the Containment Spray, Residual Heat Removal Spray, Reactor Coolant, and Safety Injection systems. The licensee evaluated ten valves out of a total population of approximately 40 valves that met the criteria for susceptibility to failure. In general, the largest, fastest acting valves were chosen for evaluation. Based on the weighted sample, the licensee determined that no multi-system valve backseating issue existed.

As a result of this issue, the resident inspectors reviewed the following documents:

MI-11.2	Motor Operated Valve Adjustment Guidelines
AI-18	Appendix B Trip Report
SQM-24	Torque and Limit Switch Settings for Motor-Operated Valves
Work Plan 11099	
Limitorque Technical Manual	

Step 5.1.6 of MI-11.2 states that for valves with high speed stroke times, the limit switch should be initially adjusted to approximately 90% of travel. The valve is then operated electrically and the amount of stem travel is measured. The valve limit switch is readjusted to allow the valve to open or close between 99 and 100%, but the limit switch actuation should not be set to allow exceeding 98% of valve travel. Work Plan 11099 utilized procedure SQM-24 during replacement of the Limitorque operators on the Main Feedwater System isolation valves. While SQM-24 required that motor-operated valves not be backseated by motor operation, SQM-24 made no reference to the MI-11.2 process and stated only that the opening limit switch be set to operate between 97 to 98% of full travel of the stem from the closed position.

Based on the above discussion, procedure SQM-24, used for Limitorque operator replacement per Work Plan 11099, did not incorporate the necessary controls on the valve limit switch adjustment activity, which were established by MI-11.2. Failure to incorporate appropriate and necessary controls in procedure SQM-24 for maintenance on safety-related equipment resulted in incorrect limit switch settings and subsequent Main Feedwater System valve failures. This constitutes a violation (327/85-17-04 and 328/85-17-03). This is a second example of the same violation discussed in paragraph 7 of this report.

The valve stem from Unit 1 valve 1-FCV-47 was used to replace one of the failed Unit 2 stems and a spare stem from power stores was used to replace the second failed valve. A replacement stem was provided by a vendor for the Unit 1 valve, and it was installed by the licensee. The vendor acceptance documents had licensee identified QA exceptions, however, the stem was installed. A review of the licensee exceptions is an Inspector Followup Item (327/85-17-06 and 328/85-17-05). The setting of the limit and torque switches was observed by the inspectors.

c. An electrical maintenance activity was observed on the 6.9 KV Shutdown Board, panel 1B. The maintenance involved the installation of a bypass function for the 2B-B centrifugal charging pump auxiliary lubrication oil pump, pressure interlock. Work plan 11529 and field change request (FCR) 3528 were reviewed. No violations or deviations were identified.

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

a. 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 completed; no unreviewed safety questions were involved; and violations of regulations or Technical Specification conditions had been identified.

LERs Unit 1

327/83014	Hydrogen Recombiner Inoperable Because of a Bad Kilowatt Meter.
327/83075	Hydrogen Recombiner Inoperable Due to a Bad Kilowatt Meter.
327/83098	Glycol Containment Isolation Valve Discovered Failed Closed and Subsequent Rise of Ice Bed Temperature Above 27°F.
327/83111	Oil in the Glycol (coolant) Expansion Tank of Diesel Generator 1A2.
327/83112	Simultaneous Removal of Both Trains of Automatic Actuation Logic for Reactor Trip Function From Service.

LERs Unit 2

328/83028	Hydrogen Recombiner Being Inoperable Due to a Bad Kilowatt Meter.
328/83085	Reactor Coolant System Subcooling Margin Monitor Inoperable Because of Loss of the Plant Process Computer.
328/84013	Unit Shutdown Due to a Rupture of the Pressurizer Relief Tank Relief Disc.
328/84014	Automatic Reactor Trip on Lo-Lo Steam Generator Level.
328/84015	Reactor Trip Due to Failure of the Turbine Generator Electrohydraulic Control System.

328/84016 Reactor and Generator Trip Due to the Actuation of the Generator Neutral Overvoltage Alarm.

b. The following licensee identified items were reviewed in order to evaluate management initiatives and overall corrective actions. The NRC encourages licensee initiatives for self-identification and correction of problems. In support of these goals, the NRC will not generally issue a Notice of Violation (if applicable to the situation) for a situation which was identified by the licensee; fits the Severity Level IV or V classification; is reported if necessary; was or will be corrected including measures to prevent recurrence within a reasonable time; and was not a violation that could reasonably be expected to have been prevented by the licensee's corrective action for a previous violation. Not all of the below listed issues were potential violations; however, the issues were identified and administered as if they were.

Issue	Responsibility	Date Reported
RWP checkout procedures	Health Physics	May 8
Acoustic monitor storage	Instrument control	May 14
Dropped screw into unit board	Electrical - 0588	May 15
Employee lost key card	Security	May 18
Keys left in a vehicle	Security	May 8
RWP signout missed	Electrical	May 16
Inadequate procedure	Operations	May 21

10. Event Followup (93702, 92706, 62703, 61726)

a. Unit 1 Residual Heat Removal Isolation

On May 14, 1985, while in Mode 4 at 140°F and 10 psig, both trains of the Unit 1 Residual Heat Removal System (RHR) were isolated by a false high pressure signal from Reactor Coolant System (RCS) pressure transmitter PT-68-66. Unit 1 had been in cold shutdown for approximately one month prior to the event. The RCS temperature increased from 140°F to 149°F during the event.

The Train B RCS transmitter was on a common sense line with the Train B Reactor Vessel Level Indication System (RVLIS), which was undergoing a high pressure test to assure adequate fill of the RVLIS sensing lines. The transmitter sensed the high pressure in the RVLIS and isolated FCV-74-2, the RHR suction line isolation valve, at 500 psig, as designed. Operators promptly responded to of indication FCV-74-2 closing and secured the operating RHR pump. The RHR system was isolated for 16 minutes while operators diagnosed the problem and depressurized the RVLIS. The RCS pressure transmitter setpoint reset, thus allowing operators to reopen FCV-74-2 and restore RHR core cooling. During the event, the centrifugal charging pump (CCP) was being utilized in the first stages of RCS pressurization. With suction from the RWST, the CCP supplied approximately 500 gallons of 2000 ppm borated water to the RCS while the RHR was isolated.

The event was caused by an inadequacy in Surveillance Instruction SI-484, "Periodic Calibration of Reactor Vessel Level Instrumentation (RVLIS) and RCS Wide Range Pressure Channels (P-403, P-406) (Refueling Outage)," which prescribed the configuration of the RVLIS for the test. The test was performed in accordance with Special Maintenance Instruction, SMI-0-68-26, "Partial Fill of RVLIS System - Upper Plenum Sense Lines (Trains A and B)." Steps to preclude this event, i.e., isolation of the RCS transmitter from RVLIS or disabling the pressure signal to the RHR suction isolation valve, were not included in procedure SI-484 or SMI-0-68-26. Licensee personnel indicated that the Westinghouse instructions for RVLIS calibration were utilized to review the procedures for completeness without using the proper TVA drawings and procedures. Failure to provide an adequate procedure for testing the RVLIS is a further example of violation (327/85-17-04 and 325/85-17-03) discussed in paragraph 7.

The licensee stopped work on the RVLIS test after the system was depressurized. The procedures were reviewed in detail by the licensee, revised as needed, and were reviewed and approved by the Plant Operations Review Committee. In addition, the licensee conducted a review of other procedures being utilized to perform outage work to assure that no other conflicts existed. No further problems were identified.

The inspector reviewed the licensee's evaluation of the event and determined that the licensee reported the event to the NRC in accordance with NRC regulations.

b. Unit 2 Trip on Erroneous Over Power Delta Temperature Signal

On May 22, 1985, with the reactor at 100% power, a reactor trip occurred on Sequoyah Unit 2 as a result of a reactor protection logic signal for excess Over-Power Delta Temperature (OPDT). The actual OPDT limit was not exceeded by the unit. An erroneous signal was introduced during the execution of plant test, TI-2, "Calorimetric Calculation," by an Instrument Maintenance technician. The erroneous signal was produced by insertion of an electrical ground into the Reactor Coolant System (RCS) hot leg and cold leg temperature instrument test points (ITP-411C, ITP-411D, ITP-421C, and ITP-421D) due to improper use of digital voltmeter. The reactor protection system properly sensed the grounded test points as two average temperatures (Tave) below the value required for the existing power level of the unit. The two out of four low Tave signals resulted in the OPDT trip.

A review of control room operator action was conducted by the inspectors, in addition to a verification of TS required staffing requirements. The following documents were reviewed:

Unit 2 Reactor Operator Log
Unit 2 Assistant Shift Engineer (ASE) Notebook
Unit 2 Reactor Trip Report
AI-2 Authorities and Responsibilities for Safe Operation and Shutdown
E-0 Reactor Trip
ES-0.1 Reactor Trip Response
TI-2 Calorimetric Calculation

Interviews were held with key individuals on shift in order to evaluate the root cause of the above trip. The following work practices were found to have contributed to the reactor trip:

1. The Shift Engineer (SE) and the ASE were not notified that a test was being conducted.

2. The Lead Reactor Operator (RO) was notified of the test, but did not notify the Balance of Plant (BOP) RO that testing was being conducted. The BOP RO was at the controls while the Lead RO left the horseshoe area and went behind the panels.

3. The Instrument Technician taking data in the reactor protection system racks did not correctly use a digital volt meter. TI-2 did not contain expected values, although the technician should have recognized the obviously inappropriate readings.

4. The Reactor Engineer supervising the test in the Unit 2 auxiliary instrument room did not recognize that faulty data was being taken and recorded in TI-2 Appendix G during four successive erroneous readings.

No procedural violations were identified, although personnel error was the root cause.

Following the reactor trip a startup was conducted in which there was difficulty maintaining number four steam generator (SG) level. Reactor power had reached between one and four percent and feed water was being supplied by the two motor driven auxiliary feed pumps following the removal from service of the "A" main feed pump (MFP). The "A" MFP was removed from service to repair a speed controller problem which would not allow the pump to rotate at desired speed. Loop 4 SG level began to decrease as a result of a failure of level control valve 2-LCV-3-171 to allow adequate flow. MFP "A" was restarted and the Unit Lead Reactor Operator drove control rods in, in an attempt to lower power below that which could be handled by the two motor driven AFW pumps. The combination of the flow of cold water to the SG and the action of driving control rods in caused Tave to drop to 521 degrees F, with the reactor still critical. The "A" MFP was tripped and control rods were driven in to decrease reactor power. The reactor was stabilized in

Mode 2 with primary pressure at 1990 psig and Tave at 521. The operator actions were reviewed with key personnel and appeared to be adequate and conservative. The inspectors had no further questions.

c. Control Room Isolation

On May 23, 1985, a control room isolation occurred on the "A" train of Control Room Ventilation. The reported cause of the isolation was an electrical spike on the 24V DC bus. There also have been several instances of auxiliary building isolations at Sequoyah Nuclear Facility during calendar year 1985. A review of the root cause of these isolations and the licensee's corrective actions is an Inspector Followup Item (327/85-17-07 and 328/85-17-06).

d. Unit 1 RCS Unidentified Leakage

On May 30, 1985, Unit 1 entered the action statement of TS 3.4.6.2 which limits unidentified leakage in the RCS to 1 gpm. The unit was in Mode 4. Unidentified leakage had been determined to be 1.94 gpm per Surveillance Instruction SI 137.2, "Reactor Coolant System Water Inventory - Units 1 and 2," Rev. 16. In accordance with SQN-IP-1, "Emergency Plan Classification Logic," Rev. 6, the Shift Engineer initiated a Notification of Unusual Event (NOUE). A report was made to the NRC within one hour of the event. The licensee continued actions to identify the leakage until June 1, 1985 (approximately 30 hours after determination of the excessive leakage rate). The unit was then placed in Mode 5, and the NOUE was terminated. The licensee subsequently attributed the majority of the excessive leakage to a seal failure on reactor coolant pump #4.

The inspectors reviewed the event and found no violations or deviations. The inspectors reviewed SI 137.2 and determined that the procedure was ambiguous in the description of the TS term UNIDENTIFIED LEAKAGE. The use of the term in the procedure implied that the completion of necessary actions to determine unidentified leakage had been earlier than the indicated entry into the LCO. In addition, the procedure did not provide precautions to assure timeliness in the determination of leakage other than the requirement to meet the TS surveillance time limit of 72 hours. These items were discussed with the licensee and the licensee committed to revise the procedure to eliminate the ambiguous wording and provide guidance on timeliness in the determination of the leakage. This item is identified as an Inspector Followup Item (327/85-17-08 and 328/85-17-07).

11. In Office Review (92700, 92701)

The following items (18 months and older) were reviewed by the Regional staff for safety significance. Based on this review and the results of the latest Resident and Region based inspection activities in the affected functional areas, the following items were determined to require no additional specific followup due to lack of safety significance and are closed.

a. Docket 50-327

78-05-01	81-10-06
79-16-05	81-20-15
79-30-03	81-21-03
79-PA-06	82-24-01
79-BU-15	82-24-03
80-SB-01	
80-34-03	

b. Docket 50-328

CDR81-13	79-35-06
CDR81-19	80-SB-01
CDR81-22	81-CD-01
CDR81-28	81-CD-13
CDR81-29	81-49-06
CDR81-35	82-25-01
CDR81-39	