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

Report Nos.: 50-327/84-38 and 50-328/84-38

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: December 6, 1984 - January 5, 1985

Inspectors: A. J. Ignatonis 2/08/85
for E. J. Ford Date Signed

A. J. Ignatonis 2/08/85
for L. J. Watson Date Signed

Approved by: A. J. Ignatonis 2/08/85
for S. Weise, Section Chief Date Signed
Division of Reactor Projects

SUMMARY

Scope: This routine, unannounced inspection entailed 230 inspector-hours onsite in the areas of plant tour, Technical Specification compliance, operations performance, housekeeping, radiation control activities, surveillance activities, maintenance activities, quality assurance practices, site security, modifications, independent inspection and followup of events.

Results: Three violations were identified - failure to document a prompt Non-Conforming Report (NCR), failure to perform adequate post-maintenance testing on P-11 block switches and inadequate corrective action for a deficient safety injection switch.

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REPORT DETAILS

1. Persons Contacted

Licensee Employees Contacted

*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
*M. Skarzynski, Maintenance Supervisor (E)
D. H. Tullis, Maintenance Supervisor (M)
B. M. Patterson, Maintenance Supervisor
R. W. Fortenberry, Engineering Section Supervisor
J. R. Walker, Assistance Operations Group Supervisor
G. G. Wilson, Assistant Operations Group Supervisor
D. E. Crawley, Health Physics Supervisor
J. T. Crittenden, Public Safety Service Supervisor
D. C. Craven, Quality Engineering Supervisor
*R. E. Alsup, Compliance Supervisor
*R. K. Gladne , Instrument Engineer
*R. W. Olson, Modifications Manager

Other licensee employees contacted included field services craftsmen, technicians, operators, shift engineers, security force members, engineers, maintenance personnel, contractor personnel and corporate office personnel.

Other Organizations:

Office of Engineering

*V. A. Bianco, Project Engineer, (Nuclear Engineering Branch)
*J. E. Staub, Supervisor, Electrical Inst. Services.

*Attended exit interview

2. Exit Interview

The inspection scope and findings were summarized with the Plant Manager and members of his staff on January 11 and 25, 1985. Three violations described in paragraphs 10 and 11.b were discussed in detail. The licensee acknowledged the violations and took no exception. During the reporting period, frequent discussions were held with the Plant Manager and his assistants concerning inspection findings. The licensee did not identify as proprietary any of the materials provided to or reviewed by the inspectors during this inspection. At no time during the inspection was written material provided to the licensee by the inspector.

3. Licensee Action on Previous Enforcement Matters

   This subject was not addressed in the inspection.

4. Unresolved Items

   Unresolved items were not identified during this inspection.

5. Plant Tour (71707, 92706, 71710, 71711)

   a. The inspector conducted plant tours periodically during the inspection interval to verify that monitoring equipment was recording as required, equipment was properly tagged, operations personnel were aware of plant conditions, and plant housekeeping efforts were adequate. The inspector determined that appropriate radiation controls were properly established;, excess equipment or material was stored properly, and combustible material was disposed of expeditiously. During tours the inspector looked for proper equipment oil levels and cooling water availability, the existence of unusual fluid leaks, excessive piping vibrations, pipe hanger and seismic restraint abnormal settings, various valve and breaker positions, equipment clearance tags and component status, adequacy of firefighting equipment, and instrument calibration dates. Some tours were conducted on backshifts. The inspector performed major flowpath valve lineup verifications and system status checks on Unit 1 and 2 on the following systems (both trains):

      (1) Containment Spray System
      (2) Residual Heat Removal System
      (3) Safety Injection System
      (4) Turbine Driven Auxiliary Feedwater (Unit 1 only)
      (5) Motor Driven Auxiliary Feedwater
      (6) Condensate Storage Tank (supply and recirculation flow paths)
      (7) Essential Raw Cooling Water (supply to auxiliary feedwater)
      (8) Refueling Water Storage Tank (supply to centrifugal charging pumps)
      (9) Upper Head Injection System
      (10) Auxiliary Control Air System
      (11) Auxiliary Building Gas Treatment System
      (12) 6.9kV Shutdown Boards
      (13) 480 VAC Shutdown, Reactor MOV, and Containment and Auxiliary Ventilation Boards
      (14) 120 VAC Vital Plant Control Power System
      (15) 125 VDC Vital Plant Control Power System

   b. During the inspection period the inspector conducted a detailed walk-down of the Unit 2 Auxiliary Feedwater (AFW) system. The inspector utilized the following documents:

      - Flow Diagram, Auxiliary Feedwater, dwg. 47W 803-2 Rev. 25

      - System Operating Instruction, SOI 3.2 "Auxiliary Feedwater System"

These documents were used to walkdown accessible portions of the AFW system including flow valve alignment and valve locking verification, instrumentation alignment and power availability checks. Both trains of equipment were checked. Additionally major flow and diversion pathways were checked for Unit 1.

No violations or deviations were identified.

6. Technical Specification Compliance (71707, 61726, 92706)

   a. During this reporting interval, the inspector verified compliance with selected limiting conditions for operation (LCO) and reviewed results of selected surveillance tests. These verifications were accomplished by direct observation of monitoring instrumentation, valve positions, switch positions, and review of completed logs and records. The licensee's compliance with selected LCO action statements were reviewed as they happened.

      No violation or deviations were identified.

   b. During the inspection period the inspector observed the surveillance of ΔT/Tavg channel IV, of the Reactor Protection System. Documents reviewed included:

      - Surveillance Instruction, SI 90.7 "Reactor Trip Instrumentation Monthly Functional Test (Rack 13)", Unit 1, Revision 2.

      - Instrument Maintenance Instruction, IMI-99 "Reactor Protection System, FT 11.4A Online Functional Test of ΔT/Tavg Channel IV, Rack 13", Unit 1, Revision 11.

      The following test equipment in use was verified as properly calibrated:

      - Resistance Temperature Detector Simulator RTD-100 (decade box), ID #466296, calibration due 3-12-85 and ID #502491, calibration due 5-14-85.

      - Ramp Generator, ID #385586, calibration due 10-29-85.

      - Fluke 8600A Digital Multimeter, ID #472628, calibration due 2-16-85.

      For the above testing the inspector verified that testing was being performed in compliance with the procedure in use at the work site, that communications were adequate, that the tested components met the acceptance criteria, and that procedural controls existed for lead lifting/landing and jumpers. The inspector determined that performing personnel were sufficiently knowledgeable for the task through technical discussions of the circuitry involved and the nature and purpose of the test.

      No violations or deviations were identified.

7. Plant Operations Review (71707, 71711)

   a. During the inspection interval, the inspector periodically reviewed shift logs and operations records, including data sheets, instrument traces, and records of equipment malfunctions. The review included control room logs, auxiliary logs, operating orders, standing orders, jumper logs and equipment tagout records. The inspector routinely observed operator alertness and demeanor during plant tours. During abnormal events, operator performance and response actions were observed and evaluated. The inspector conducted random-off hours inspections during the reporting interval to assure that operations and security remained at an acceptable level. Shift turnovers were observed to verify that they were conducted in accordance with approved licensee procedures. The inspector had nc further comments.

   b. During this reporting period Unit 1 was at normal power operation and Unit 2 resumed normal operation after completing its cycle 2 refueling/modification outage which began September 28, 1984.

   c. At various times during the inspection period the inspector observed portions of the license's operational activities to recover Unit 2 from a refueling and modification outage and effect a plant status change from mode 5 (cold shutdown $\leq$ 200°F) to mode 1 (power operation). Procedure usage by operations personnel was confirmed by the inspector and included:

      - General Operating Instruction, GOI-1 "Plant Startup from Cold Shutdown to Hot Standby" Rev 49

      - General Operating, Instruction, GOI-2 "Plant Startup from Hot Standby to Minimum Load" Rev 37

      - General Operating Instruction, GOI-5 "Normal Power Operation" Rev 21

   The inspector verified by observations, interviews and log checks that procedural precautions and prequisites were met. The inspector also verified that instructional steps were followed in accordance with good operating practices. Spot checks were made of items on the various checklists to verify their accomplishment.

   No violations or deviations were identified.

8. Physical Protection

   The inspector verified by observation and interview during the reporting interval that measures taken to assure the physical protection of the facility met current requirements. Areas inspected included the organization of the security force, the establishment and maintenance of gates, doors and isolation zones in the proper condition, that access control and badging was proper, that search practices were appropriate, and that escorting and communications procedures were followed.

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

The inspector reviewed the following LER's to verify that the report d. ls met licensee requirements, identified the cause of the event, described appropriate corrective actions, adequately assessed the event, and addressed any generic implication. Corrective action and appropriate licensee review of the below events were verified. When licensee identified violations were noted, they were reviewed in accordance with the enforcement policy. The inspector had no further comments.

LER	EVENT
327/84075	Failure to Comply With One Hour Fire Watch Requirement of Technical Specification 3.7.12.
327/84069	ERCW Valve For Diesel Generator 2A-A Cooling Inoperable

The inspectors reviewed licensee event report (LER) SQRO-50-327/84069 which provided details concerning failure of an essential raw cooling water flow control valve (FCV-67-66) for diesel generator 2A-A. The thermal overload on the valve had not been properly reset when a channel calibration of the valve's overload relay heater was performed on October 12, 1984, (SI-251.2, "Channel Calibration of Class IE Motor Operated Valve Overload Relay Heaters"). The mechanical device which holds the overload open had apparently not cooled sufficiently to reset when the thermal overload reset button was pushed. The licensee discovered the incident on October 23, 1984, during the performance of surveillance instruction SI-9, "Actuation of Automatic Valves via SI Signal for Non-Testable Boric Acid and ECCS Flow Path Valves," which required operation of diesel generator 2A-A. When the diesel generator was started, the unit operator noted that the "A" train valve did not operate automatically upon diesel generator startup and opened the "B" train valve to provide cooling water to the diesel generator.

The inspector review a December 7, 1984 revision to SI-251.2 which contained a requirement to perform a continuity check across the overload relay control contact to insure the contact has been reset.

The inspector reviewed the flowpath of ERCW cooling water to the diesel generators. The "B" train cooling water supply control valve does not receive an automatic diesel generator startup signal. The valve is maintained closed to provide train separation. The valve is opened by operator action to provide cooling in the event the "A" train valve fails to open. The system operating instruction requires the operator to monitor the operation of the "A" train valve upon actuation of the diesel generator.

The inspector reviewed surveillance instruction SI-7, "Electrical Power System: Diesel Generators" (Rev. 30) and determined that a step in the manual actuation instructions which require verification of the opening of "A" train cooling valve 2-FCV-67-66 had been omitted from the instructions for starting diesel generator 2A-A. The step was included in the portions of the procedure addressing diesel generators 1A-A, 1B-B and 2B-B.

System Operating Instruction SOI-82.3, "Diesel Generator 2-A" requires that the opening of valve 2-FCV-67-66 be verified upon diesel operator startup. SI-7 requires the use of SOI-82.3 to monitor the diesel generator operation which the surveillance testing is in progress, therefore, opening of the valve was verified by the system operating instruction. The licensee has agreed to revise SI-7 to include the missing step. This is identified as Inspector Followup Item 328/84-38-04.

LER 327/84-72 concerning post accident radiation monitors was also reviewed, but will remain an open item pending receipt of supplemental information as discussed in paragraph 10 of this report. The supplement to the LER is expected by February 10, 1985.

10. Modifications (37700, 62703, 61726)

On December 7, 1984, the licensee notified the NRC that post accident radiation monitors on the Reactor Coolant Drain Tank (RCDT) sump line and the Reactor Building Floor and Equipment Drain (RBF&ED) sump line were located on the wrong lines, such that detection of a high radiation level in one sump line by the radiation monitors would result in isolation of the non-affected line. The radiation monitors are required by License Condition 2.c.(22). F for the Unit 1. The licensee identified the discrepancy while performing a visual inspection of a similar installation of radiation monitoring equipment on Unit 2. The monitors are not an NRC requirement for Unit 2. The noncompliance was corrected on Unit 1 and Unit 2 by reviewing the high radiation isolation signals to the correct isolation valves on December 7, 1984.

The inspectors conducted interviews with licensee employees and reviewed the design change requests, the installation work plan, post-modification testing, plant drawings and other documents listed below.

Documentation:

ECN-2779 Rev. 0
ECN-L 5199 Rev. 1
Work Plan 8947 Rev. 0 and Rev. 1
PRO 1-84-421
NCR SQNNEB8407
LER SQRO-50-327/84072
TACF 2-84-122-90
TACF 1-84-123-90

Drawings and procedures:

45N824-14	Rev. 18	Conduit and Grounding Floor EL 690.0 Details - Sheet 2
45W1651-17	Rev. 6	Wiring Diagrams Unit Containment Building PNL 1-M-30 Connection Diagram, Sheet 17

45W1651-18	Rev. 2	Wiring Diagrams Unit Containment Building PNL 1-M-30 Connection Diagram, Sheet 18
45W1651-19	Rev. 3	Wiring Diagrams Unit Containment Building PNL 1-M-30 Connection Diagram, Sheet 19
47W800-1	Rev. 1	Flow Diagram General Plant Systems
47W809-7	Rev. 10	Flow Diagram Flood Mode Boration Makeup System
47W812-1	Rev. 10	Flow Diagram Containment Spray System
47W821-28	Rev. 3	Flow Diagram Chemical Cleaning Waste Disposal System
47W830-1	Rev. 15	Mechanical Flow Diagram Waste and Disposal System
47W851-1	Rev. 15	Mechanical Flow Diagram Floor and Equipment Drains
47W852-1	Rev. 5	Mechanical Flow Diagram Floor and Equipment Drains
SI-685	Rev. 3	Channel Calibration for Low Range Accident Radiation Monitors - 18 Months
SI-688	Rev. 2	Functional Test for Accident Radiation Monitoring System, Unit 1-Monthly
EN DES-EP1.26	Rev. 8	Nonconformance - Reporting and Handling by EN DES
EN DES-EP 1.48	Rev. 1	Preparation of Failure Evaluations/ Engineering Reports of Deficient Conditions for Operating Nuclear Plants.

The inspector reviewed the licensee's design error; the licensee's corrective action process; their reporting methodology; the adequacy of post-modification installation verification and testing; and, the safety significance of the discrepancy.

Field Installation Drawing and Installation Review

A review of Work Plan No. 8947 Rev. 1 indicated that an incorrect conduit and cabling design drawing, 45N 824-14, had been issued for field installation. This drawing showed monitors 1-RE-90-275 and 1-RE-90-276 physically located near the RBF&ED sump line. These monitors were shown to provide an isolation signal to RCDT sump line isolation valves 1-FSV-77-9 and 1-FSV-77-10 instead of to required valves 1-FSV-77-127 and 1-FSV-77-128; monitors 1-RE-90-277 and 1-RE-90-278 located near the RCDT sump line, were shown to provide an isolation signal to incorrect isolation valves 1-FSV-77-127 and 1-FSV-77-128 located in the RBF&ED sump line.

Although the licensee event report (LER) SQRO-50-327/84072 identified the drawing discrepancy, it was found to be inadequate in that it did not address the reason for the original design error. The licensee has agreed to provide additional information including a more comprehensive description for cause of the design error, the personnel involved and the planned corrective action. The supplemental LER will be provided by February 10, 1985.

The engineer in change of installation confirmed that the incorrect drawing had been used for installation; however, ECN 5199 and other documents in Work Plan 8947 were used to prepare the post-modification test, DI-685. The post modification test required the radiation monitors to be source checked with verification made in the control room by control board indication of valve closure in lieu of verification of valve closure in the affected line. Therefore, the test indicated that the correct associated valve was closing and did not identify the mislocation of the monitors. Furthermore, a post installation verification was conducted using the incorrect drawing and as a result it did not identify the discrepancy. LER SQRO-50-327/84072 does not address measures to preclude this type of error in future installations. This is one of two examples constituting a violation: Failure to Conduct an Adequate Post Modification Test (327/84-38-01, 328/84-38-01). The second example is discussed in paragraph 11.b.

Reporting Methodology Review

The reporting methodology utilized to identify and report the error to the responsible individuals for corrective action was reviewed. Interviews with licensee employees revealed the following information. The discrepancy was identified on November 20, 1984, by an individual from the TVA Office of Engineering (OE). The employee while gathering data for a study on the radiation monitors not related to the subject installations noted the monitors to be either mislabeled or mislocated. On November 21, the discrepancy was reported to the person's supervisor. The discrepancy was evident by observation of photographs taken of the monitors on November 20 and then developed on November 21, 1984, and the licensee's OE Site Project Electrical Engineer was asked on November 21, to investigate the discrepancy. Additional calls were made by the individual on November 26, and December 4, 1984. The individual then issued a nonconformance report (NCR). The NCR was dated December 6, 1984 and signed by the Branch Chief of NEB-NAL on December 7, 1984.

Procedure EN DES - EP 1.26, "Nonconformances - Reporting and Handling by EN DES," Rev. 8, Section 10, states that one purpose of the procedure is to implement 10 CFR 50 Appendix B, Criterion XVI, "Corrective Action." Further, under Section 3.0, the procedure states that the nonconformance report is the "only method used by EN DES to officially report to EN DES management identified significant conditions adverse to quality." The individuals involved confirmed that they utilized EN DES-EP 1.26 in their review of the discrepancy. Section 3.0 also states that "a condition adverse to quality is to be promptly documented and all required action expeditiously effected." Section 5.2 allows two working days from identification of a potential nonconformance for the responsible organization to initiate an NCR form. If the responsible organization does not initiate the NCR, the individual discovering the discrepancy has an additional working day to initiate an NCR.

Although the discrepancy on the subject radiation monitors was identified on November 20, 1984, an NCR was not initiated until 16 days later on December 6, 1984. (The responsible individual in the Office of Nuclear Power was informed on December 7, 1984.)

Furthermore, interviews with OE employees and managers responsible for NCR determinations indicates that it was common practice to allow five to seven days and in some cases months, for the determination of whether discrepancy was a "potential nonconformance," and these became an "identified potential nonconformance" to which EN DES-EP 1.26 applied.

Two managers indicated that training classes on EN DES-EP 1.26 were provided. One manager had attended the class in the spring of 1984. The second manager had not attended the class. These managers were responsible for instructing those engineers who reported to them on the application of EN DES-EP 1.26.

In summary, OE employees indicated that it was a common practice upon identification of a discrepancy to allow a period of time for evaluation before the NCR process was entered. This period of time ranged from days to, in some cases, months before the discrepancy was documented. The period of time as described by OE employees is not addressed by the governing procedure EN DES-EP 1.26. EN DES-EP 1.26 requires documentation of potential conformances within three days of the identification of potential nonconformance and allows a total of eight days from identification to the determination of significance. In addition, communications between the OE employee identifying the discrepancy and the responsible organization failed to establish the party who had responsibility for writing an NCR. The inspector also noted that in a review of the procedural steps of EN DES EP 1.26 with OE managers interviewed, the managers interpreted steps which allowed three days to write an NRC to allow five days and also were not familiar with the definition of "significant conditions adverse to quality" which is defined by the procedure.

The interviews described above indicated that the management controls governing the identification, prompt evaluation and processing of nonconformance was not effective. The failure to provide an adequate procedure prescribing the process for identifying and documenting nonconformance was identified to the licensee as a violation: Failure to have adequate nonconformance report procedure (327/84-38-02 and 328/84-38-02). It was noted that once the responsible organization of the Office of Nuclear Power was notified, the nonconformance was corrected on the same day.

Safety Significance

The inspectors also evaluated the safety significance of the mislocation of the radiation monitors. The modifications are an additional safety measure used as backup assurance that high level radioactive effluent is not released from containment. Valves 1-FSV-77-9, 1-FSV-77-10, 1-FSV-77-127 and 1-FSV-77-128 receive a Phase "A" isolation signal in the event of a release of radioactive material to the containment atmosphere or upon pressurization of the containment per Technical Specification 3.6.3. The valves can only be reopened by manual action during accident conditions and effluent from the sump lines is processed through the normal liquid radwaste system. Since the function performed by the radiation monitors is redundant to existing safety features, containment isolation under accident conditions was not compromised due to inoperability of the above described system.

11. Event Follow-Up (93702, 71707, 61726, 62703)

a. The inspector followed-up on an event which occurred at 3:36 a.m. CST on December 15, 1984, with Unit 1 at power and Unit 2 in hot standby (mode 3). An oil fire occurred in a current transformer in the 500 kv line. The fire was extinguished by the site fire brigade by approximately 3:35 a.m. CST. At 4:36 a.m., the returning fire brigade members reported that the fire was the result of the Watts Bar #2 line, B phase current transformer explosion. When informed by the brigade that an explosion was involved, control room personnel implemented IP-2 "Unusual Event" in accordance with their procedures and notified NRC Operations Center. The licensee exited IP-2 at 4:39 a.m. CST. The inspector reviewed the events and chronology, interviewed involved personnel, reviewed applicable procedures and compared the data to the requirements of 10 CFR 50.72, Immediate notification requirements for operating nuclear power reactors which states that the licensee shall notify the NRC not later than one hour after the declaration of one of the Emergency Classes (e.g. Unusual Event). This requirement was met.

   No violations or deviations were identified.

b. A follow-up inspection was performed for an inadvertent Unit 2 Safety Injection (SI) which occurred on December 16, 1984 while the unit was in mode 3 and in the process of increasing pressure to change modes. At 8:26 a.m. CST an SI actuation occurred on low pressurizer pressure. Prior to the event, the operators identified a primary safety relief valve weeping based on elevated tailpipe temperature and reduced primary pressure using pressurizer spray to reseat the valve with the low pressurizer pressure SI channels blocked. To assure SI blockage while performing the evolution, prior to reaching 1870 psig (SI setpoint) the operator cycled the SI block spring-return-to-center switch. Upon release of the A train switch, the switch passed through the center position and momentarily actuated the contacts on the reset side thereby unblocking the low pressurizer pressure signal. A partial emergency core cooling system (ECCS) actuation and BIT injection of approximately 570 gallons occurred. The plant was stabilized at 1850 psig and the SI was reset.

The inspector interviewed the operator and STA, reviewed entries and procedures involved. The examined procedures were:

- Emergency Instruction E-0 "Reactor Trip or Safety Injection" Rev. 0.
- Emergency Instruction ES-0.2 "SI Termination" Rev. 0.
- SQN REP Implementing Procedures Document SQN-IP-2 "Notification of Unusual Event" Rev. 7
- SQN REP Implementing Procedures Document SQN-IP-1 "Emergency Plan Classification Logic", Rev. 6

The inspector concluded, after evaluating the above, that the operators had properly utilized and adhered to the appropriate procedures.

At 9:18 a.m. CST, the licensee notified the NRC Operation Center of the ECCS actuation in accordance with 10 CFR 50.72(b) "Non-Emergency Event" which requires reporting of any event that results or should have resulted in ECCS discharge into the reactor coolant system as a result of a valid signal. At 11:35 a.m. CST, the license declared an Unusual Event as directed by procedures. The NRC was again notified as required by 10 CFR 50.72(a) "General Requirements" which requires NRC notification not later than one hour after the declaration of one of the emergency classes. Although the notifications were made, the inspector discussed with plant management the usefulness of declaring an Unusual Event approximately 3 hours after the initiation of the event. The inspector noted that during this time interval the operators were appropriately concerning themselves with plant stabilization and recovery of BIT to comply with Technical Specifications and that the procedures in use did not direct the implementation of the Emergency Plan until recovery from the inadvertent SI was assured. The inspector expressed a concern to management that the procedures involved needed to initiate the Unusual Event in a more timely manner or that the need to declare an Unusual Event for this type of incident be reevaluated. The inspector reviewed additional procedures which addressed more serious incidents and was satisfied that implementation of the Emergency plan would occur in a timely manner. Pending the licensee's decision on the disposition of the inspector's concern this is designated an inspector follow item (IFI 328/84-38-05).

The inspector performed a detailed review of the SI switch operation and its design. The inadvertent SI was attributed to improper operation of the SI block handswitch (Type OT-2). There are eight OT-2 handswitches in use, four on each unit. The handswitches are spring return-to-neutral (center) from either the block position or the unblock (reset) position. These switches tend to spring back past the neutral position to the reset position when released in the block position.

The handswitch problem was first identified during startup testing for Unit 1 in September, 1979. An investigation of several spurious SIs which occurred during the startup testing indicated that the reset contact could be opened by a very slight movement (approximately a 10 millisecond overshoot) towards the reset position. The licensee placed administrative controls in the appropriate procedures which require reactor operators to release the switch slowly to prevent the switch from springing back to the reset position. Additionally, cautionary placards were placed above the applicable switches. The inspector verified that GOI-3, "Plant Shutdown from Minimum Load to Cold Shutdown", Rev. 28, contained these instructions.

The licensee also initiated a design change request, SQ-DCR-775, on June 26, 1980 to replace the existing switches. A request was sent to Westinghouse to replace the switches on December 10, 1981. Westinghouse initiated Field Deficiency Reports TVAM-10202 (Unit 1, January 11, 1982) and TENM-10130 (Unit 2, January 11, 1982) to replace the switches. The disposition of these field deficiency reports, which was signed off in February, 1982, indicates that the switches were to be replaced, but no design change of existing equipment was needed. The Unreviewed Safety Question Determination (USQD) for SQ-DCR-775, approved on November 5, 1982, indicates that the original handswitches were defective and that the new handswitches would stop in the normal position as required. The USQD states that the new handswitches are the same design as the original ones. There is no indication of testing or other investigations to support a determination that the original handswitches were defective versus a determination that the switch design caused the problem.

On January 15, 1985, Westinghouse indicated to TVA that the reported problem was analyzed and that the resulting "bounce" was not due to defective parts but was characteristic of the particular switch. Westinghouse also stated that the application of the OT-2 switches did not present a safety problem.

The inspector discussed the following areas of concern with the licensee:

(1) The deficiency was identified in September 1979; however, the switches were not replaced until February 1983, for Unit 1 and December 1982 for Unit 2. The licensee stated that the delay was attributed to a disagreement with Westinghouse on the application of a warranty on the switches. The licensee acknowledged that the timeframe was excessive.

(2) The inspector noted that although the original need for a design change to the switches was recognized by the plant, changes in this determination were subsequently made without supporting reason, which resulted in reinstallation of switches with the original design problem. Although administrative controls were utilized on interim measure to preclude recurrence, the correction for the design problem was not made.

(3) The failure to adequately determine the cause of the handswitch design problem and correct the problem to preclude repetition of an inadvertent SI was identified to the licensee as a violation: Failure to Determine and Correct OT-2 Handswitch Problem (327/84-38-03, 328/84-38-03).

(4) It is NRC position that inadvertent actuation of safety injection should be avoided. The design problem in this case has resulted in inadvertent safety injection actuations which are unacceptable. On January 11, 1985, the licensee initiated a design change to replace the switches with a different type to prevent a recurrence of the problem. This is identified as Inspector Followup Item (327/84-38-06, 328/84-38-06).

The inspector also reviewed the post-modification testing which was conducted after the handswitches were installed. SI 90.8 (Unit 1) and 90.82 (Unit 2), "Monthly Functional Test of Reactor Trip Instrumentation" were utilized for the post-modification tests. These tests verify that the switch will perform the block function; however, testing to detect the original problem, i.e., the "bounce" back to reset after release, was not performed. The failure to perform testing to demonstrate that the handswitch would perform satisfactorily was identified to the licensee as a second example of a violation (327/84-38-01, 328/84-38-01) discussed in paragraph 10 of this report.