UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W.
ATLANTA, GEORGIA 30303

Report Nos.: 50-327/84-11 and 50-328/84-11

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 Date: April 6 - May 5, 1984

Inspection at Sequoyah site near Chattanooga, Tennessee

Inspectors: P. R. Holmes, Jr. — 6/13/84
E. J. Ford — Date Signed

P. R. Holmes, Jr. — 6/13/84
S. D. Butler — Date Signed

Approved by: — 6/13/84
C. A. Julian, Section Chief — Date Signed
Division of Reactor Projects

SUMMARY

Scope: This routine inspection involved 180 inspector-hours onsite in the areas of Operational Safety Verification, Plant Operations Following Refueling, Unit 1 Post Modification and Surveillance Testing, Maintenance, Independent Inspection Effort, PORV Miswiring Followup, and Thimble Guide Tube Ejection Followup.

Results: Of the seven areas inspected, no violations or deviations were identified in five areas; five apparent violations were found in two areas (Changing modes with Technical Specification instrumentation out of commission, paragraph 6; Failure to control modifications, paragraph 10; Failure to retrieve QA records, paragraph 10; Inadequate testing of modifications, paragraph 10; and Failure to use appropriate drawings during maintenance, paragraph 10.)

REPORT DETAILS

1. Persons Contacted

   C. C. Mason, Plant Superintendent
   L. M. Nobles, Assistant Plant Superintendent
   J. B. Krell, Assistant Plant Superintendent
   D. H. Tullis, Maintenance Supervisor (M)
   B. M. Patterson, Maintenance Supervisor (I)
   D. C. Craven, Maintenance Supervisor (E)
   J. M. Anthony, Operations Supervisor
   R. W. Fortenberry, Engineering Supervisor
   D. E. Crawley, Health Physics Supervisor
   J. T. Crittenden, Public Safety Service Supervisor
   J. L. Hamilton, Quality Assurance Supervisor
   M. R. Harding, Compliance Supervisor
   W. M. Halley, Preoperational Test Supervisor
   J. Robinson, Field Services Group Director

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

2. Exit Interview

   The inspection scope and findings were summarized on May 8 and 17, 1984, with those persons indicated in paragraph 1 above. The notice of violation was discussed and the licensee acknowledged.

   During the reporting period, frequent discussions are held with the Plant Superintendent and his assistants concerning inspection findings.

3. Licensee Action on Previous Inspection Findings

   Not Inspected.

4. Unresolved Items

   Unresolved items were not identified during this inspection.

5. Operational Safety Verification (71707)

   The inspector toured various areas of the plant on a routine basis throughout the reporting period. The following activities were reviewed/verified:

   a. Adherence to limiting conditions for operation which were directly observable from the control room panels;

   b. Control board instrumentation and recorder traces;

c. Proper control room and shift manning;

d. The use of approved operating procedures;

e. Unit operator and shift engineer logs;

f. General shift operating practices;

g. Housekeeping practices;

h. Posting of hold tags, caution tags and temporary alteration tags;

i. Personnel, package, and vehicle access control for the plant protected area;

j. General shift security practices, on post manning, vital area access control and security force response to alarms;

k. Surveillance testing in progress;

l. Maintenance activities in progress; and

m. Health physics practices.

On April 15, the inspector was informed that the licensee had declared an Unusual Event (UE) at the site in accordance with their Radiological Emergency Plan (REP). The UE was declared when Unit 1 surveillance testing identified Reactor Coolant System (RCS) leakage of approximately nine gallons per minute from an unknown source. The inspector verified that the licensee was complying with Technical Specification requirements for unidentified RCS leakage of greater than one gallon per minute and that the NRC had been properly notified in accordance with 10 CFR 50.72. The source of leakage was traced to a leaking seal injection filter housing isolation valve that was closed for maintenance. The seal filter housing was closed to stop the leak and surveillance testing demonstrated that RCS leakage was less than one gallon per minute. The UE was cancelled later in the day and the NRC was properly notified.

On April 17, the Unit 1 reactor tripped from approximately 15% power during startup testing after the refueling outage. The inspector reviewed the trip and discussed it with Operations personnel to ensure that Safety Systems operated as designed, that the plant was recovered in accordance with approved operating procedures and that the NRC was properly notified in accordance with 10 CFR 50.72. No discrepancies were identified.

On May 4, 1984, the inspector observed portions of the licensee's operational activities to effect a Unit 1 plant status change from mode 5 (Cold Shutdown - ≤200°F) to mode 3 (Hot Standby - ≥350°F). Procedures in use by operations personnel included General Operating Instruction, GOI-1 "Plant Startup from Cold Shutdown to Hot Standby". The inspector verified by observations, interviews and log checks that procedural precautions and prerequisites 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.

6. Plant Operations Following Refueling (71711)

During the reporting period the inspector verified that Unit 1 was being properly returned to service following the refueling outage. The inspector verified that approved procedures were being used to establish configuration control of plant systems and ensure that Technical Specification requirements were being met during plant heatup and startup. The inspector toured the Unit 1 lower containment following the establishment of containment integrity, to ensure work was complete, systems were returned to normal, and that the containment was cleared of debris that could clog the sump. Selected safety systems were inspected to ensure that valve alignment was correct for power operations. No discrepancies were noted.

On April 15, 1984, the inspector witnessed initial criticality of Unit 1 after the refueling which started February 21, 1984. The inspector reviewed startup procedures to ensure that they were adequate and complete, and that startup was proceeding in a controlled manner. The reactor went critical with control rod position and boron concentration within expected design values.

During the heatup and startup, the licensee identified on two different occasions that required instrumentation was not operable. The first occurrence was on April 13, 1984, when Unit 1 entered mode 3 with steam generator level channel 1-LT-3-38 inoperable. The operators involved evaluated the inoperable instrument using Technical Specification 3.3.1.1, which does not require the instrument to be operable until Mode 2. Technical Specification 3.3.2.1, however, requires the instrument to be operable in mode 3 for its Auxiliary Feedwater initiation feature. The licensee failed to evaluate the need for the instrument in accordance with Technical Specification 3.3.2.1 and entered mode 3 with it inoperable. When it was determined that the instrument was required in mode 3, the licensee complied with the Technical Specification action statement by placing the associated bistable in the trip position until the instrument could be returned to service. A second occurrence took place on April 15, 1984, when Unit 1 entered mode 2 with pressurizer level transmitter 1-LT-68-320 inoperable. The pressurizer level instruments are required to be operable in modes 1 and 2 in accordance with Technical Specification 3.3.1.1, Reactor Trip Instrumentation and 3.3.3.7, Accident Monitoring Instrumentation. The level transmitter's variable leg tap was relocated in parallel with a pressurizer sample tap during the refueling outage as part of a modification involving the pressurizer safety valve loop seals. The safety evaluation for the modification stated that measures needed to be taken to ensure that the change in the instrument's sensing line did not affect its safety related function. The licensee did not adequately determine the effect of this change. When the chemistry lab analyst started taking continuous pressurizer boron samples, as required by the approved restart procedure, the level transmitter pegged low. The operator did not notice the downscale reading until after the Unit had entered mode 2. Once the inoperable instrument was recognized, the licensee complied with the Technical Specification action statement by placing the associated bistable in the tripped condition until sampling was terminated and the instrument verified operable. Additional modifications are being made to prevent sampling from affecting the instrument. Both of these occurrences are examples of a violation of Technical Specification 3.0.4, which prohibits entry into operational modes while relying on action statement provisions for satisfying Limiting Conditions for Operation: A Notice of Violation will be issued. (327/84-11-01)

No other violations or deviations were identified.

7. Unit 1 Post Modification and Surveillance Testing (37700, 617726)

During the reporting period the inspector witnessed portions of Post Modification Test PMT-53 (WP10920) "Auxiliary Feedwater System Cavitating Venturi Modification", PMT-50 (WP10909) "Pressurizer PORV Test - Unit 1" and PMT-39 (WP10925) "Reactor Head Vent System - Unit 1". The test procedures were reviewed to ensure that they were adequate and portions of each test were observed to ensure test prerequisites were met, calibrated test equipment was in use when necessary, and that the testing was being performed in accordance with the written test procedures by adequately trained personnel. No discrepancies were noted.

During the reporting period the inspector witnessed portions of Surveillance Instruction SI-43 "Rod Drop Timing Test". The test instructions were reviewed to ensure that instructions were adequate, that testing was witnessed to ensure that prerequisites were met, that calibrated test equipment was in use where necessary and that testing was being performed in accordance with written test procedures by adequately trained personnel. The completed data was reviewed for all control rods to verify that the Technical Specification requirements were being met following the refueling outage.

No violations or deviations were identified.

8. Maintenance (62703)

On April 15, 1984, the inspector observed the reassembly of the 1B seal injection filter. The work was being performed in accordance with Maintenance Request MR-A245521 and Maintenance Instruction MI-8.1 "Seal Water Injection Filter Changeout". The inspector reviewed the work package to ensure that it was adequate and that it contained the necessary quality assurance features. Work was observed to ensure that it was being done in accordance with the written instructions by adequately trained personnel. The Radiological Work Permit was reviewed to ensure that it was adequate for the work and that it was being followed by health physics and maintenance personnel. No discrepancies were noted.

No violations or deviations were identified.

9. Independent Inspection Effort

The inspector routinely attended the morning staff meetings during the reporting period. These meetings provide a daily status report on operational and maintenance activities in progress as well as a discussion of significant problems or incidents associated with the plant.

No violations or deviations were identified.

10. Event Followup of PORV Miswiring (93702, 37700)

The inspector was informed by the licensee of an incident involving the miswiring of bistable contacts associated with auxiliary control loops of the pressurizer power operated relief valves (PORV). The incident was discovered on April 2, 1984. The wiring change errors were made on March 31, 1984. The inspector's inquiries disclosed that Instrument Mechanics (IM) were attempting to perform Surveillance Instruction, SI-92 "Remote Shutdown Monitoring Instrumentation - Pressurizer Pressure Channel Calibrations" an 18-month surveillance requirement of Technical Specification (TS) 3.4.3.2 on Unit 1. The IMs noted the alarm indicating lights on the bistable. The PORV pressure switch appeared to be working the reverse of what was normally expected for the bistable. The IMs attempted to verify the correctness of the bistable indication by comparing the bistable alarm lights, external terminal wiring and internal logic wiring (high or low) of the Unit 1 bistables to those of Unit 2. This unit functional check showed no comparable pattern of bistable lights, terminal wiring or logic switch position. The IMs took no action at the hardware and informed their foreman of the discrepancies. The foreman checked electrical schematics, equipment cards, and the SI. After his review, the foreman consulted with the plant instrumentation engineers to determine correct bistable action. The engineers used an "as constructed" drawing, 45N668-1 Rev. 5, to evaluate the wiring. They then changed the bistables' internal switches from low to high. High was the position in normal usage for similar circuits. When "corrective actions" had been implemented, the result was that all four PORVs, (2 PORVs for each unit), had their internal switch incorrectly placed in the high position and one PORV, (on Unit 2), had an actual error in its terminal wiring corrected. The engineers were not aware of another TVA drawing, D8059895B, which showed the internal wiring of the bistable pressure switches and its relationship to the instrument loop. This other drawing and the manufacturer's manual would have shown the correct internal logic position, (low).

On April 2, 1984, the corrective actions to the perceived problem was discussed with management by the engineers for technical review and reportability consideration. A phone report was made to NRC Operation Center that day. An investigation into how the wiring had been changed and how long it had been "incorrect" was conducted. A review of the manufacturer's manual revealed a fail-safe feature which had been implemented for the PORV bistable pressure switches.

A review of documentation in the inspector's files showed a Potential CDR dated February 15, 1979, and followup Non-Conforming Report, NCR-MEB 79-10 dated March 14, 1979; May 10, 1979; and May 31, 1979; addressing control loop bistables failing to an undesirable contact position on loss of control power to the pressure switch module. The NCR stated that the problem was found during preoperational testing of the Unit 1 facility auxiliary control room controls. The NCR stated the safety implications (such as had the deficiency gone uncorrected), that the air-operated pressurizer power relief valve, upon loss of control power to the valve bistable, would fail open. This would result in depressurization of the reactor coolant system. Inadvertent lifting of a pressurizer safety valve is evaluated as a fault of moderate frequency in Section 15.2 of the Sequoyah FSAR. Inadvertent opening of pressurizer power operated relief valves (PORV) was not analyzed.

The NCR also stated the corrective action as "... The wiring diagram for the air-operated pressurizer power relief valve bistable has been corrected such that the bistable will fail open on loss of control power. This ensures that the valves will fail closed. Modifications for each Unit will be performed before its respective fuel load date. Upon examination of all other control loops in the auxiliary control room controls for similar configuration, no discrepancies were found." The modifications were accomplished by ECN 22-78.

Further investigation by the licensee showed that despite the original wiring inconsistencies, the four PORVs were operable for their normal function with the exception that one PORV on Unit 2 did not have the fail-safe feature referred to above. Also, a review of preoperational test W-1.2A, Reactor Coolant System Function Test, which was used to test the PORV circuitry after the modification revealed that the test did not provide for testing the fail-safe feature on Unit 2, which the modification was intended to incorporate. The four operable circuits were then made inoperable by the March 31, 1984, "corrective action", in that had control been transferred from the main control room (MCR) to the auxiliary control room (i.e. in event of MCR abandonment), the PORVs would have opened. This information was presented to the plant management and they ordered the PORV block valves closed. There was no safety significance for Unit 1 in that it was in a post-refueling Mode 5 condition. For Unit 2, which was at 100% power, one PORV (68-340) block valve had been previously closed due to PORV leakage, thus leaving one PORV (68-334) at risk to inadvertent opening if the ACR transfer switch for PORV control was manipulated.

At this point, with the PORV block valves closed, the possibility of inadvertent operation of the PORVs was eliminated. The licensee then proceeded with a technical analysis of all PORV bistable wiring to assure correct operation of the modules. After performing the analysis, the shutdown unit (Unit 1) PORV circuitry was rewired and tested prior to performing the rewiring and testing on Unit 2. After satisfactory testing, the PORV block valves were reopened (as applicable) and normal operation resumed. Testing included fail-safe operation as well as normal PORV functions.

To recap, prior to performing the SI, one PORV on Unit 2 could have opened if: (1) the transfer switches were placed in the auxiliary position, and (2) there was a coincident loss of A.C. power to the bistable module. After performing the "corrective maintenance", the PORVs on both units could have opened if: (1) the transfer switches were placed in the auxiliary position, and (2) RCS pressure was less than normal bistable setpoint for PORV opening. For the second case, circumstances mitigated the probability of occurrence and potential effects, in that Unit 1 was shutdown and one of the two PORVs on Unit 2 was already blocked (due to previous leakage). Had the Unit 2 PORV opened due to transfer of control to the ACR, indications of the downstream pressurizer relief tank (PRT) temperature and pressure, and position indication via hand switch lights, in addition to other RCS parameters, were available to the operator.

A review for compliance with regulatory requirements of the above sequence of events, disclosed the following items of apparent non-compliance:

a. Failure to Control Modifications

PORV control circuitry is classified by the QA Manual as a CSSC item, and as such, requires design controls be implemented to ensure that engineering modifications to original design are transferred to as built drawings and properly installed in plant hardware. A PORV control circuitry modification, ECN 22-78, was developed to provide corrective action for an ENDES non-conforming report (NCR-MEB 79-10, dated February 12, 1979), which stated that bistables associated with auxiliary pressure loops 68-336 or 68-337, upon loss of control power, will cause the associated pressurizer PORV (PCV-68-340 or PCV-68-334) to receive an open signal. Technical evaluations disclosed that one (PS 68-337 which effects PCV 68-334) of the two Unit 2 PORV control circuits did not have the fail-safe portion of this modification installed prior to fuel load as stated in the NCR.

This failure to have modification control is identified as a violation and a Notice of Violation will be issued (328/84-11-01).

b. Failure to Retrieve QA Records

When the above evaluation was complete it was also disclosed that one of the two PORV's (PS 68-336 which affects PCV 68-340) was wired correctly. However, the licensee was unable to produce required QA records documenting the installation of the modification on the correctly modified PORV. The licensee's QA program and 10 CFR 50, Appendix B, requires that QA records be retrievable.

This failure to retrieve modification records is identified as a violation and a Notice of Violation will be issued (328/84-11-03).

c. Inadequate Testing of Modification

When modifications are made to CSSC designated items, the licensee's QA program requires that testing be performed on those modifications. The preoperational test procedure, W-1.2A, used to test pressurizer pressure control circuitry was inadequate, in that it failed to require testing of the fail-safe feature on Unit 2 as specified in the ENDES NCR-MEB 79-10 and the modification ECN 22-78.

This failure to have an adequate testing procedure is identified as a violation and a Notice of Violation will be issued (328/84-11-04).

d. Failure to Use Appropriate Drawings During Maintenance

The maintenance that was performed on the PORV pressure switches on March 31, 1984, utilized only drawing 45N668-1 to determine wiring changes required. If the internal wiring drawing had also been used, the licensee would have realized that the switches were, in fact, correctly wired. This failure to accomplish activities affecting quality in accordance with appropriate drawings is identified as a violation and a Notice of Violation will be issued (327, 328/84-11-02).

11. Flux Thimble Guide Tube Ejection Incident - Unit 1 (93702)

On April 19, 1984, at approximately 10:00 p.m. (Unit 1 operating at 30 percent power), primary coolant began leaking from a break in a movable incore flux detection system line at the seal table. The inspector was notified by the Shift Engineer (SE) and appraised of the situation. The leak rate was approximately 25-30 gallons per minute based on charging rates necessary to recover pressurizer level.

The SE reported that an in-core flux thimble guide tube (FTGT) cleaning operation had been in progress at the seal table when the workers noted water seepage. There were eight individuals involved in the cleaning of the FTGT which was properly authorized by maintenance request MR-A238084. Upon noticing the seepage, the workers abandoned the cleaning rig and exited the room through a personnel air lock. Written statements of circumstances leading up to the event were provided by the workers involved; an exiting individual reported seeing "...the cleaning tool crank and guide fall from the seal table, water spraying to the ceiling, and the cable starting to lay back on the grating at the head of the stairs..." It was determined that the cable referred to was the FTGT and not the cleaning tool cable being ejected. The workers notified the main control room of the apparent break and proceeded to have whole body counting done. There was no personnel contamination or internal deposition as a result of the incident.

The inspector reviewed logs, interviewed personnel and made observations of ongoing event activities for compliance with regulatory and procedural requirements. Additional details concerning this event and other aspects of the recovery operation are available in inspection report 50-327/84-12.

Pending a determination of causal mechanics, the FTGT ejection incident is identified as an inspector followup item (327/84-11-03).

No violations or deviations were noted.