UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA ST., N.W., SUITE 3100
ATLANTA, GEORGIA 30303

Report No. 50-327/80-25

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

Facility Name: Sequoyah Unit 1

Docket No. 50-327

License No. DPR-77

Inspection at Sequoyah site near Chattanooga, TN

Inspectors: C. Julian for W. T. Cottle — 8/19/80 Date Signed

C. Julian for S. D. Butler — 8/19/80 Date Signed

Approved by: H D Martin for H. C. Dance, Section Chief, RONS Branch — 8/21/80 Date Signed

SUMMARY

Inspection on June 1-July 5, 1980

Areas Inspected

This routine, announced inspection involved 241 inspector-hours on site in the areas of Operational Safety Verification, Preoperational test witnessing, Startup test witnessing, Verification of license conditions, Independent inspection effort and followup on plant incidents.

Results

Of the six areas inspected, no items of noncompliance or deviations were identified in five areas; 2 items of noncompliance were found in one area (Infraction - failure to positively control personnel access into vital areas and deficiency - failure to properly perform a critical system temporary alteration.

8010170041

DETAILS

1. Persons Contacted

   Licensee Employees

   J. M. Ballentine, Plant Superintendent
   C. E. Cantrell, Assistant Plant Superintendent
   W. F. Popp, Assistant Plant Superintendent
   J. W. Doty, Maintenance Supervisor (M)
   J. M. McGriff, Maintenance Supervisor (I)
   W. A. Watson, Maintenance Supervisor (E)
   D. J. Record, Operations Supervisor
   W. H. Kinsey, Results Supervisor
   R. J. Kitts, Health Physics Supervisor
   C. R. Brimer, Outage Director
   R. S. Kaplan, Supervisor, Public Safety Services
   W. M. Halley, Preoperational Test Supervisor
   D. O. McCloud, Quality Assurance Supervisor
   J. R. Bynum, Assistant to Plant Superintendent

   Other licensee employees contacted included six construction craftsmen, five technicians, nine operators, ten shift engineers, seven security force members, eight engineers, six maintenance personnel, four contractor personnel, and ten corporate office personnel.

   Other Organizations

   During the inspection period, the following NRC personnel visited the facility:

   Three inspectors from the Office of Inspection and Enforcement
   One representative from the Office of Nuclear Reactor Regulation
   V. Gilinsky, Commissioner, USNRC
   J. Austin, Technical Assistant, USNRC
   V. Stello, Director, Office of Inspection and Enforcement
   D. Okrent, Advisory Committee on Reactor Safeguards

2. Exit Interviews

   The inspection scope and findings were summarized with the Plant Superintendent and members of his staff on June 13, and June 27, 1980. The two apparent items of noncompliance were discussed with the Plant Superintendent.

3. Licensee Action on Previous Inspection Findings

   Not inspected.

4. Unresolved Items

Unresolved items were not identified during this inspection.

5. Operational Safety Verification

The inspector toured various areas of Unit 1 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 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. Fire protection measures for hot work.

i. Posting of hold tags, caution tags and temporary alteration tags.

j. Measures to exclude foreign materials from entry into clean systems.

k. Personnel, package, and vehicle access control for the Unit 1 protected area.

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

m. Surveillance testing and preoperational testing in progress.

n. Maintenance activities in progress.

On June 20, 1980, a protective grating was removed from a vital area barrier opening, to permit access to the area for maintenance activities, without adequate compensatory measures being taken to provide access control to the area. The maintenance activity had been in progress for the previous several days with the grating being removed and reinstalled at the end of each work period and a Public Safety Officer stationed to positively control access during all periods in which the grating was removed. The incident on June 20, 1980 resulted from a failure of the outage personnel performing the maintenance to notify the Public Safety Shift Lieutenant that the grating was being removed. This incident was identified by a Public Safety Officer during a routine patrol. Prompt measures were taken to re-establish the integrity of the vital area boundary.

On June 23, 1980, a vital area door was removed for maintenance without adequate compensatory measures being taken to provide access control to the area. This was identified by the inspector during a plant tour. The inspector notified the Chief, Public Safety Services and prompt measures were taken to re-establish the integrity of the vital area barrier. The door had been removed for approximately 10 minutes when the inspector discovered it and was reinstalled within 20 minutes of discovery.

These two incidents constitute examples of an apparent item of noncompliance in that, in both instances, vital area barriers were breached without adequate compensatory measures being implemented. (327/80-25-01).

On June 25, 1980, the ice condenser inlet doors were blocked in the closed position with temporary mechanical stops. A pressure imbalance in the reactor building had resulted in the doors opening and the blocking action was required in order to allow the reestablishment of a cold air head to hold the doors closed. The unit was in Mode 3 when the inlet doors were blocked. The inspector discussed this action with the Plant Superintendent and the doors were unblocked in an expeditious manner. This was the second incident in which the ice condenser inlet doors were blocked shut with the unit in Modes 3 or 4, (Reference IE Report 50-327/80-20 Section 5). The apparent oversight in the Sequoyah Technical Specifications which allows blocking the inlet doors shut while in Modes 1, 2, 3, and 4 was referred to Region II Management on May 9, 1980, for resolution.

In following up this event, the inspector noted that the use of the temporary mechanical stops constitute a temporary alteration to the ice condenser system. Administrative Instruction AI-9 requires that all temporary modifications to critical systems be performed per an approved Temporary Alteration Control Form to insure proper review, installation and removal of the temporary alteration. There was no Temporary Alteration Control Form initiated for the blocking of the inlet doors. Failure to apply the administrative controls specified in AI-9 is an apparent item of noncompliance. (327/80-25-02)

The inspectors witnessed maintenance activities associated with the stem replacement of the suction isolation valve for the 1A-A containment spray pump. The task required that a freeze seal be established in the 12 inch suction line from the refueling Water Storage Tank since the suction valve is not isolable from the tank. The inspectors observed the work practices, reviewed the Maintenance Instruction, and discussed the work activities with the journeymen and supervisory personnel involved. The freeze seal was established, monitored, and maintained by an outside vendor. The practices and procedures utilized for the activities involving the freeze seal were discussed with the vendor representative.

On June 20, 1980, the inspectors received notification of an apparent problem involving failure of two containment spray valves to meet the system design pressure and temperature ratings. The problem was initially identified at the Watts Bar plant and was thought to be of potential significance to the Sequoyah units. The inspectors reviewed the valve name plate ratings for the valves installed at Sequoyah Unit 1 and found that while the valve pressure ratings were below the system design rating, that the ASME code allowed a sufficient upgrading of the valve pressure rating, based upon the system temperature, to compensate for the name plate pressure rating. This was verified in subsequent discussions with engineering and design personnel from the licensee's corporate office.

On June 30, 1980 while reviewing unit operator logs, it was noted that on the previous day the unit operator found valve 1-FCV-1-18 shut. This valve is one of three steam supply isolation valves for the turbine driven auxiliary feedwater (TDAFW) pump and makes the pump inoperable when it is shut. The valve was immediately opened. Further investigation by the licensee revealed that on June 28, 1980 the valve lineup for the TDAFW pump was performed as required by technical specifications but the required position for 1-FCV-1-18 specified in the surveillance instruction was wrong. The surveillance instruction has subsequently been changed to indicate the correct required position for the valve. In addition, the Operations Supervisor discussed this matter with the operations personnel and instructed them to be aware of this type of problem to prevent its recurrence.

The event will remain as a licensee identified item of noncompliance since the inoperability of the TDAFW pump did not completely disable the auxiliary feedwater system and the licensee's identification and investigation of the problem was prompt and resulted in effective corrective action being taken.

On June 30, 1980, while touring the control room, the inspector questioned the Assistant Shift Engineer about the lineup of the Auxiliary Essential Raw Cooling Water (AERCW) system. The AERCW system is designed to provide a backup to the Essential Raw Cooling Water (ERCW) system in the event of a plant site flood (break of upstream dam) or loss of heat sink (break of downstream dam) to ensure a cooling water supply to safety related components. The inspector's questions concerned a cooling tower makeup pump which was locked out and the positions of four ERCW supply valves to the four AERCW mechanical draft cooling towers. The current revision to the system operating instruction, SOI-67.1 required the makeup pump be in auto-start and the valves 1-FCV-360, 361, 362, and 363 be open. Further investigation revealed that the cooling tower makeup pump was locked out for maintenance on the discharge check valve and the licensee was relying on makeup to the cooling tower basins from the high pressure fire main cross connect, a mode of operation which is described in the FSAR. The reason the valve positions did not correspond to the SOI is that the licensee was in the process of changing the operating instruction lineup from a "standby mode" to an "immediate availability" mode to make the system more readily available to the operator in the control room. The system lineup had been done prior to issuance of the new revision to the SOI. The Assistant Shift Engineer agreed to promptly check the system lineup against the approved system lineup and correct any discrepancies.

The inspector determined that the improper system lineup did not affect the operability of the system since the motor operated valves can be operated from the control room and the system is designed to be initiated by the operator without automatic starting features. The matter was discussed with the Operations Supervisor. There were no further questions on this matter.

On July 3, 1980 the inspector performed valve lineup verifications on selected engineered safety feature systems including Residual Heat Removal, Containment Spray and the Boron Injection Tank. Actual valve lineups were compared to drawings and system lineups prepared by the licensee for the stand-by mode. No discrepancies were noted.

On July 4, 1980 the inspector made a containment tour. Both upper and lower volumes of the containment were inspected with primary emphasis placed on identifying material which could become debris and clog the refueling canal drains or the containment sump. All discrepancies were noted and identified to the Operations Supervisor and Assistant Plant Superintendent and were corrected prior to final containment closeout for initial critically.

During a routine tour of the control room it was noted that the subcooling margin was not continuously displayed on the computer trend recorder. Discussions with unit operators revealed that it was their understanding that it didn't have to be continuously displayed since it was immediately available if needed and in fact it would be called up automatically in an accident situation. The inspector contacted the licensing project manager in the Office of Nuclear Reactor Regulations (NRR) for clarification on this matter. It was determined by further discussion that the intent of the requirement for a subcooling meter was to have it continuously displayed during plant operation. The licensee has subsequently established administrative controls to ensure that the subcooling margin program is continuously displayed on the computer trend recorder for use by operators.

No other items of noncompliance or deviations were identified.

6. Preoperational Test Witnessing

On June 26 and June 27, 1980 the inspector witnessed portions of W-9.3 Control Rod Drop Timing test (hot-full flow segment). The inspector reviewed the official procedure for completeness including sign offs for prerequisites and precautions. Test data for the cold no flow and full flow rod drops was reviewed. Deficiencies were properly noted and the test director indicated that the deficiencies were being discussed with a Westinghouse representative for resolution. It was noted that the identification data for the recorder being used was not entered in the test procedure. This was immediately corrected by the test director. A test procedure was not available in the control room but the unit operator was questioned and found to be familiar with the test precautions related to rod withdrawal.

The inspector witnessed six control rods being tripped and their drop time being measured. The recorder traces were examined after the licensee had analyzed them and recorded drop time for the associated rods. The inspector was satisfied with the interpretation of the recorder traces. All rods witnessed had drop times which were within the tolerances allowed by the test and Technical Specifications. There were no further questions on W-9.3.

No items of noncompliance or deviation were identified.

7. Startup Test Witnessing

On July 5, 1980 the inspector witnessed initial criticality of Sequoyah Unit 1. It was verified that the shift manning requirements were met, the proper revision of the startup procedure was being used and the prerequisites and initial conditions were satisfied and properly signed off in the procedure. Temporary changes to the test procedure were reviewed to ensure they received proper review and approval. Operators were questioned to determine if they were familiar with the procedure and its precautions and limitations. A sampling of Technical Specification requirements were inspected to ensure they were satisfied. During the dilution process, inverse multiplication plots required by the startup procedure were reviewed.

The reactor was critical at 11:40 EDT and critical boron concentration was within the tolerances described by the procedure.

No items of noncompliance or deviations were identified.

8. Verification of License Conditions

On July 2, 1980 the inspector witnessed SQSTEAR-11, the Auxiliary Feedwater Water Hammer test as required by Technical Specification 7.2.3. The procedure was reviewed for completeness including signoffs of prerequisites and precautions and proper incorporation of changes. One deficiency was noted by the test director. The data logger associated with the thermocouples used to measure feedwater nozzle temperatures was past due for its periodic calibration. The deficiency was considered acceptable by the test director since the thermocouples appeared to be indicating normally and the data collected from them was not required for final determination of test acceptability. The inspector determined that the licensee had adequate communication with personnel stationed to monitor the feedwater system during the conduct of the test.

NRC personnel were stationed in the control room and in the east mainsteam valve room. Auxiliary feedwater was injected at a maximum rate of 440 gallons per minute into Steam Generator No. 2 after allowing the feed ring to drain for two hours. No indication of water hammer was observed. The completed test procedure was given final review after system restoration and determined to be satisfactory. This closes item 7.2.3 of Section 7.2 of Technical Specifications.

Subsequent to revising Surveillance Instruction SI-114 to include an augmented inservice inspection program for the reactor vessel closure head flaw, the licensee completed implementing their license requirement 7.2.5 by revising Division Procedure Manual DPM No. N73015 to include the requirement to report any augmented inservice inspection results required by Technical Specifications, surveillance instructions or division procedures to the NRC. This closes item 7.2.5 of Section 7.2. of Technical Specifications.

On July 4, 1980 the inspector reviewed work plans WP8386 and WP 8386 Rev. 1 which documented the relocation of the Auxiliary Building Gas Treatment System (ABGTS) filter D/P gages. The work plans indicated that the work had been completed with exception of marking and accepting a few gage line mounting brackets. The inspector verified by personnel observation that the work was in fact complete and that the gages were relocated to the satisfaction of Region II. This closes inspector Open Item 327/79-48-01 and item 79-48-01 of Table 7.1-2 of Section 7.0 of Technical Specification.

No items of noncompliance or deviations were identified.

9. Independent Inspection Effort

The inspector routinely attended the morning scheduling and staff meetings during the reporting period. These meetings provided a daily status report on the construction and testing activities in progress as well as a discussion of significant problems or incidents associated with the construction, startup testing, and operations efforts.

The inspector reviewed completed Surveillance Instruction SI-656 Waste Gas System Gross Leakage Test. The test was performed in response to Short Term Lessons Learned from the Three Mile Island Accident and was identified as inspector open item 327/80-08-02. The test involved pressurizing the waste gas system including the gas decay tanks, the waste gas compressors and interconnecting piping. The system was pressurized to approximately 100 psig and checked for external leakage to the auxiliary building with soap solution. The system was then checked for internal leakage for information purposes. Two significant valve bonnet leaks were identified and maintenance requests were issued to repair the leaks. The valves will be leak checked independently when repairs have been made. This closes Open Item 327/80-08-02.

The inspector reviewed the Unreviewed Safety Question Determination (USQD) OWIL 0-70-8644 on repair of the "C" Compenent Cooling Water (CCW) heat exchanger. The tube staking of the CCW heat exchangers was described in IE Report 327/80-02. The USQD was in agreement with the inspector's previous determination that taking the "C" heat exchanger out of service for the work in modes 1, 2, 3, or 4 would constitute an unreviewed safety question and require NRC approval. In addition, techincal specification relief would be required for the work in these modes due to loss of independence of the essential raw cooling water (ERCW) headers. The licensee has completed work on the "A" CCW heat exchanger and is presently working on the "B" heat exchanger. Both were previously analyzed and determined not to constitute an unreviewed safety question. The work on the CCW heat exchangers will continue to be followed as open item 327/80-20-01.

During the reporting period the inspector prepared and presented testimony at the Advisory Committee on Reactor Safeguards subcommittee hearings related to approval of Sequoyah for full power licensing.

The inspector prepared comments on Sequoyah's full power license and forwarded them to Region II for submittal to the Office of Nuclear Reactor Regulation (NRR). The inspector also compiled comments from Region II personnel on Sequoyah's Standard Technical Specifications and forwarded them to Region II for submittal to NRR.

During the reporting period the inspector assisted in arranging visits to Sequoyah by NRC Commission V. Gilinsky, Director of Inspection and Enforcement, V. Stello, and Dr. D.Okrent, member of the Advisory Committee on Reactor Safeguards.

Prior to initial criticality, the inspector reviewed work plans and maintenance requests involving repair of pipe hanger discrepancies found during the licensee's inspection required by I&E Bulletin 79-14. The review was concentrated on documentation covering repairs that were committed to be complete prior to initial criticality. This included Category 1 discrepancies (those analyzed to cause pipe failure during a seismic event) and Category 2 discrepancies (those which did not meet design criteria but were not analyzed to cause pipe failure during a seismic event) which would be inaccessible during power operations. Documents reviewed included the following:

Work Plans	Maintenance Requests		
8641-R1, R2, R3, R4, R5, R6	28111	60013	60553
8645	28112	60009	26699
8655	28113	60010	26696
8709	28114	60067	60364
8692	60952	28467	60230
8654	28117	60002	60856
8729	28118	60834	60271
8735	28470	60005	60860
8649	60358	60672	60156
8732	60318	60849	60270
8695	60260	60850	60014
8705	28115	26688	60522
8656	60265	26692	60835
8702	60976	60679	60700
8690	60978	60844	60841
8698	60994	60688	60683
		26692	26688
		60838	60835
		60685	60837

The inspector had no further questions on this matter.

On June 16, 1980 the inspector witnessed the performance of the licensee's Radiological Emergency Plan (REP) drill performed in conjunction with county and state agencies. NRC personnel were present in the Sequoyah control room as well as the licensee's corporate emergency control center, the state emergency control center and the licensee's Radiological Hygiene Center in Muscle Shoals, Alabama. The inspector attended the REP drill critique on June 17, 1980. The inspectors comments and persons contacted were compiled and forwarded to Region II for inclusion in IE Report 327/80-23.

No items of noncompliance or deviations were identified.

10. Plant Incidents

During the reporting period, the inspector conducted followup activities on the following incidents at the facility.

On June 2, 1980 Unit 1 experienced an inadvertent safety injection actuation while in Mode 5. The inadeventent actuation came during the performance of a surveillance which required blocking the low pressure safety injection signal after decreasing pressure indication below the P-11 interlock set point. A pair of spring return switches with a block, reset and mid-position are used for this purpose and the switch is designed such that if it is moved to the block position and allowed to spring return to the mid-position, it can travel past the mid-position and open the reset contacts. This in fact happened and when pressure was reduced below the low pressure set point, the safety injection signal was initiated. The licensee is aware of this problem and had previously placed signs on the control board warning the operator to slowly return the switches to the mid-position. Presently they are considering modifying the switches or replacing them with differently designed switches to prevent recurrence of this problem.

On June 16, 1980 a motor operated disconnect (994) in the 161 KV switch yard overheated and damaged itself and a portion of the related buswork. The loss of a portion of this bus disabled the "A" Common station service (CSS) transformer which is normally one source of offsite power to the plant. No offsite power was lost however because power was also being fed to the plant through the Unit 1 Unit station service (USS) transformers. Subsequent to the occurrence, the licensee performed an infra-red scan of the 161 and 500 KV switch yards to determine if there were any other hot spots. One additional problem was identified and corrected.

Prior to the repair of the damaged disconnect and buswork, the licensee was scheduled to begin a heatup to Mode 3. A representative of the Office of Nuclear Reactor Regulations (NRR) was contacted to ensure that the method that the licensee was using to supply offsite power to the plant met the intent of their technical specification requirement. NRR concurred with this position.

On June 23, 1980 Unit 1 experienced an inadvertent safety injection while in Mode 3. An immediate report was made to NRC headquarters as required by 10 CFR 50.72. The licensee had just completed a unit heatup and had shut the main steam isolation valves (MSIV) to repair a steam trap leak in the turbine hall. Primary temperature had drifted down below the lo-lo Tave setpoint and when the licensce reopened the MSIV's they received a hi steam flow coincident with lo-lo Tave safety injection. All related equipment operated as designed and recovery was uneventful. During the followup the operator indicated that he had properly opened the MSIV bypass valves and no differental pressure was indicated across the valves.

On July 1, 1980 a construction electrician fell from a cable tray in the cable spreading room and sustained an apparent compression fracture of the lumbar vertibrae. He was taken to a local hospital by ambulance. The NRC was properly notified as required by 10 CFR 50.72.

The inspector reviewed the circumstances involved in each incident and, where appropriate, the actions taken by licensee management in response to the incident. Licensee's Management response appeared to be both timely, and adequate in each case.

No items of noncompliance or deviations were identified.