UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
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ATLANTA, GEORGIA 30303-8931

January 28, 2002

Tennessee Valley Authority
ATTN: Mr. J. A. Scalice
Chief Nuclear Officer and
Executive Vice President
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

SUBJECT: SEQUOYAH NUCLEAR PLANT - NRC INTEGRATED INSPECTION REPORT 50-327/01-04 AND 50-328/01-04

Dear Mr. Scalice:

On December 29, 2001, the NRC completed an inspection at your Sequoyah Nuclear Plant, Units 1 and 2. The enclosed report presents the results of that inspection which were discussed on January 4, 2002, with Mr. Richard Purcell and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

Based on the results of this inspection, the inspectors identified two issues of very low safety significance (Green), that also were determined to involve violations of NRC requirements. However, because of their very low safety significance and because they have been entered into your corrective action program, the NRC is treating these issues as non-cited violations, in accordance with Section VI.A.1 of the NRC's Enforcement Policy. If you deny any non-cited violation in the enclosed report, you should provide a response with the basis for your denial, within 30 days of the date of this inspection report, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001; with copies to the Regional Administrator, Region II; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at Sequoyah.

Immediately following the terrorist attacks on the World Trade Center and the Pentagon, the NRC issued an advisory recommending that nuclear power plant licensees go to the highest level of security, and all promptly did so. With continued uncertainty about the possibility of additional terrorist activities, the Nation's nuclear power plants remain at the highest level of security and the NRC continues to monitor the situation. This advisory was followed by additional advisories, and although the specific actions are not releasable to the public, they generally include increased patrols, augmented security forces and capabilities, additional

security posts, heightened coordination with law enforcement and military authorities, and more limited access of personnel and vehicles to the sites. The NRC has conducted various audits of the Tennessee Valley Authority's response to these advisories and Sequoyah's ability to respond to terrorist attacks with the capabilities of the current design basis threat. From these audits, the NRC has concluded that the Sequoyah security program is adequate at this time.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room

from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Paul E. Fredrickson, Chief
Reactor Projects Branch 6
Division of Reactor Projects

Docket Nos. 50-327, 50-328
License Nos. DPR-77, DPR-79

Enclosure: NRC Integrated Inspection Report
w/Attachment - Supplemental Information

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U.S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos:	50-327, 50-328
License Nos:	DPR-77, DPR-79
Report No:	50-327/01-04, 50-328/01-04
Licensee:	Tennessee Valley Authority (TVA)
Facility:	Sequoyah Nuclear Plant, Units 1 & 2
Location:	Sequoyah Access Road Soddy-Daisy, TN 37379
Dates:	September 30, 2001 - December 29, 2001
Inspectors:	R. Gibbs, Senior Resident Inspector R. Telson, Resident Inspector K. Poertner, Resident Inspector - Surry Plant D. Rich, Resident Inspector - Watts Bar Plant P. Habighorst, Resident Inspector - Indian Point 2 Plant R. Carrion, Project Engineer J. Blake, Senior Project Manager J. Kreh, Emergency Preparedness Inspector

Enclosure

M. Scott, Senior Reactor Inspector
S. Walker, Reactor Inspector
E. Testa, Senior Health Physicist
R. Chou, Reactor Inspector
G. Hopper, Senior Operations Engineer
L. Miller, Operations Engineer

Approved by: P. Fredrickson, Chief
Reactor Projects Branch 6
Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000327-01-04, IR 05000328-01-04, Integrated inspection report, on September 30, 2001 - December 29, 2001, Tennessee Valley Authority, Sequoyah Nuclear Plant, Units 1 and 2. Emergent work, event follow-up.

The inspection was conducted by resident inspectors, a senior health physicist, a senior project manager, reactor inspectors, an emergency preparedness inspector, operations engineers, and a project engineer. The inspection identified three Green findings, which were also non-cited violations. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using IMC 0609, "Significance Determination Process" (SDP). The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described at its Reactor Oversight Process website.

A. Inspector Identified Findings

Cornerstone: Mitigating Systems

- Green. The inspectors identified a non-cited violation of Technical Specification 6.8.1.a, Procedures and Programs, for failure to follow the procedure for realigning the Unit 2 B boric acid tank (BAT) to the reactor coolant system after a transfer of boric acid from BAT C to BAT B was completed. This resulted in the function to provide highly concentrated boric acid to refill the RWST being unavailable for nearly 3 days.

  This finding had a credible impact on safety because it rendered the B boric acid tank unavailable for anticipated transient without scram (ATWS) and steam generator tube rupture (SGTR) mitigation. For ATWS mitigation, highly concentrated boric acid had been available from the RWST through the high pressure charging pumps. Therefore, the portion of the finding related to ATWS mitigation was of very low safety significance. With respect to SGTR event mitigation, an SDP Phase 3 assessment determined that this portion of the finding was also of very low safety significance because of the short time that the equipment was unavailable.

Cornerstone: Barrier Integrity

- Green. The inspectors identified a non-cited violation of Technical Specification 6.8.1.a, Procedures and Programs, for failure to follow work instructions to obtain shift manager concurrence when breaching a control room envelope door.

  This finding had a had a credible impact on safety and could have affected the integrity of the control room envelope because the control room staff was unaware of the door being open and therefore would not have known to contact the involved worker had a control room envelope isolation signal occurred. Therefore, the control room envelope would not have pressurized as required. The finding was of very low safety significance because the finding only represented a degradation of the radiological barrier function provided for the control room.

B. Licensee Identified Violations

A violation of very low safety significance which was identified by the licensee has been reviewed by the inspectors. Corrective actions taken or planned by the licensee appear reasonable. This violation is listed in Section 4OA7.

Report Details

Summary of Plant Status: Unit 1 operated at or near full power until October 21 when the unit was shutdown for a scheduled refueling outage. The unit was restarted on November 22 and was returned to full power on November 26. The unit operated at or near full power until the end of the inspection period.

Unit 2 operated at or near 100 percent power for the entire inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems and Barrier Integrity

1R01 Adverse Weather Protection

a. Inspection Scope

The inspectors verified that the licensee had taken actions against freezing weather conditions to limit the risk of freeze-related initiating events and adequately protect mitigating systems from its effects. The inspectors walked down selected components including those associated with the refueling water storage tank level instruments, condensate storage tank level instruments, main feedwater flow sensing lines, main steam valve vaults, and essential raw cooling water (ERCW) instrumentation and piping to evaluate implementation of licensee procedures and corrective actions to items identified in relevant problem evaluation reports (PERs) and to assess the condition of selected freeze-protection components. The following documents were reviewed during the inspection.

- Periodic Instruction (PI) 0-PI-OPS-000.006.0, Freeze Protection
- PI 1-PI-EFT-234-706.0, Freeze Protection Heat Trace Functional Test
- PI 2-PI-EFT-234-706.0, Freeze Protection Heat Trace Functional Test
- PI 0-PI-MIN-000-706.0, Freeze Protection Insulation Inspection
- PERs 00-009264, 00-011653, 01-003765, and 01-006304

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

.1 Partial Walkdowns

a. Inspection Scope

The inspectors conducted equipment alignment partial walkdowns to evaluate the operability of selected redundant trains or backup systems, listed below, with the other train or system inoperable or out-of-service. The walkdowns included a review of applicable operating procedures to determine correct system lineups and an inspection of

critical components (e.g., power supplies, support systems) to identify any discrepancies that could affect operability of the redundant train or backup system.

- Train B auxiliary building gas treatment system while the A train was out of service
- Unit 2 motor driven auxiliary feedwater (AFW) system train A while the turbine driven AFW pump was in test
- Unit 2 motor driven AFW system train B while the turbine driven AFW pump was in test

b. Findings

No findings of significance were identified.

.2 Complete Walkdown

a. Inspection Scope

The inspectors performed a walkdown of selected accessible portions of the Unit 2 turbine-driven AFW system. The inspectors verified proper equipment alignment by comparing actual equipment configuration to plant procedures, drawings, and the Updated Final Safety Analysis Report (UFSAR). The inspectors reviewed outstanding work requests, PERs, and the system health report, which discussed open engineering issues, to determine if any conditions existed that would have prevented the system from fulfilling its intended safety function. The inspectors also performed a review of the corrective action program for substantive equipment alignment issues for all risk-significant systems to ensure that the licensee was identifying and correcting problems appropriately. Selected documents reviewed during the inspection included the following:

- AFW system flow diagrams, CCD No. 1,2-47W803-2 and CCD No. 1,2-47W803-3
- ERCW system flow diagram, CCD No. 1,2-47W845-2
- UFSAR Section 10.4.7.2, Auxiliary Feedwater System
- Work orders 00-008127-000, 01-000895-000, 01-003485-000

b. Findings

No findings of significance were identified.

1R05 Fire Protection

.1 Routine Walkdowns

a. Inspection Scope

The inspectors conducted tours of areas important to reactor safety, listed below, to evaluate conditions related to (1) control of transient combustibles and ignition sources; (2) the material condition, operational status, and operational lineup of fire protection systems, equipment and features; and (3) the fire barriers used to prevent fire damage or fire propagation. The inspectors referenced SPP-10.10, Control of Transient Combustibles, and prefire plans for the areas listed below, as appropriate.

- Unit 1 main turbine lube oil storage area
- Emergency diesel generator (EDG) 1A-A room
- Spent fuel pool heat exchanger and pump area
- EDG 2A-A and 2B-B exhaust fan rooms
- Unit 2 main turbine lube oil storage area
- Unit 1 turbine building near main steam line piping

b. Findings

No findings of significance were identified.

.2 Annual Fire Drill Observation

a. Inspection Scope

The inspectors observed an off-hours fire brigade drill in the cable spreading room located in the control building below the main control room. This room is one of the highest risk fire areas in the plant. The drill included the use of local offsite fire fighting personnel. The inspectors evaluated the licensee's readiness to fight and prevent the spread of fires, including the use of fire fighting equipment, the use of pre-fire plans, the donning and use of protective clothing and self-contained breathing apparatus, command and control by the fire team leader, and radio communications between the fire brigade and plant operations. In addition, the inspectors verified that the licensee's drill objectives were met.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities

.1 Inservice Inspection (ISI)

a. Inspection Scope

The inspectors observed in-process ISI work activities and reviewed selected ISI records. The observations and records were compared to Technical Specifications (TS) and ASME Boiler and Pressure Vessel Code, Sections V and XI, 1989 Edition and 1995 Edition with Addenda through 1996 to verify compliance. In addition, qualification and certification records for examiners and nondestructive examination (NDE) procedures for the below ISI examination activities were reviewed. Portions of the following Unit 1 ISI examinations were observed:

Ultrasonic (UT)

- Calibration for residual heat removal piping weld RHRF-093 and pressurizer welds WP-1 &WP-6
- Reactor coolant piping welds RCS-051, 069, 089, and 101
- Feedwater piping weld FDS-29
- Feedwater piping thickness 103 BT 135

Visual (VT)

- Safety injection supports 1-SIH-064 and SIH-113
- Steam generator mid support bolt connections SGH-1-1 and SGH-4-1

Radiographic (RT)

- Feedwater piping root pass welds for leading edge flowmeter

The inspectors observed the licensee's UT Level III examiner review a suspected indication for FDS-29. The radiograph for feedwater piping root pass welds were observed, specifically film shot preparation, shot taken, and the review of radiograph film before and after repair for acceptability.

b. Findings

No findings of significance were identified.

.2 Unit 1 Steam Generator (SG) Inspection

a. Inspection Scope

The inspectors reviewed selected inspection records for the eddy current (ET) examination of the SGs. The records were compared to the TS, license amendments and applicable industry established performance criteria to verify compliance. Qualification and certification records for examiners, equipment and procedures for the above eddy current examination activities were reviewed. The inspectors also observed in-situ pressure testing of tube R3 C69 in SG 4, which was found to have an outside diameter, axial flaw in the U-bend region.

The inspectors reviewed the results of the licensee's extent-of-condition and apparent cause reviews for PER 01-010352-000, regarding a circumferential crack found in the hot-leg, tube-sheet region of tube R15 C29 in SG 4, during the current cycle 11 ET examinations, which had previously been called a circumferential crack during cycle 10 ET examinations, but had not been plugged at that time.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

a. Inspection Scope

The inspectors reviewed the facility operating history and licensee event reports since the last requalification program inspection for indications of operator weaknesses and their inclusion in the feedback process. In addition, the inspectors reviewed Sequoyah Nuclear Plant Training Department Self Assessment SQN-TRN-01-001. The inspectors also reviewed the three biennial written examinations administered last year and evaluated their effectiveness in providing a basis for assessing operator knowledge of material covered in the requalification training program. Examination quality, licensee effectiveness in integrating industry events, plant and student feedback into the requalification training program, and examination development methodology were evaluated as well. The inspectors observed annual dynamic simulator examinations (six scenarios) for three operator teams to assess the adequacy of the licensee's evaluation of operator knowledge and abilities. During these observations, the inspectors assessed licensee evaluator effectiveness in pinpointing operator performance deficiencies requiring supplemental training. The inspectors also evaluated and observed portions of the walkthrough examination administered during this requalification segment to assess evaluator performance.

The inspectors reviewed and evaluated the licensee's remedial training program for operator deficiencies identified during the previous two years. The review included remedial training packages and associated training review board minutes. In addition, the inspectors observed the remediation and reevaluation process of two crews who had failed the simulator portion of the examination during the week of the inspection. The inspectors also reviewed a sample of licensed operator requalification attendance records, watchstanding records, reactivation records, and ten percent of the licensed operator medical records to ensure compliance with 10 CFR 55.59, Requalification and 10 CFR 55.53, Conditions of License.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation

a. Inspection Scope

The inspectors sampled portions of selected structures, systems or components (SSCs), listed below, as a result of performance problems, to assess the effectiveness of the licensee's maintenance practices. The inspectors evaluated the licensee's Maintenance Rule (MR) implementation against Procedures SPP-6.6, Maintenance Rule Performance Indicator, Monitoring, Trending, and Reporting - 10CFR50.65 and 0-TI-SXX-000-004.0, same title as SPP-6.6. Reviews focused on: (1) MR scoping; (2) characterization of failed

SSCs; (3) safety significance classifications; (4) 10CFR 50.65 (a)(1) or (a)(2) classifications; and (5) the appropriateness of performance criteria for SSCs classified as (a)(2) or goals and corrective actions for SSCs classified as (a)(1).

SSC	Related Documents
EDG 2A-A failed to start	PER 01-010452-000; CDEF 1393
Miscellaneous auxiliary control room instrument failures	CDEFs 1326, 1380, 1379
Control room ventilation damper failed to close during test	CDEF 1302; PER 01-004394-000
Component cooling water valve 0-FCV-70-198 thermal overload trip	PER 01-009247-000; CDEF 1388
Unit 2 6.9 KV board flood berm leakage	PER 01-008546-000

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope

The inspectors evaluated, as appropriate for the selected work activities: (1) the effectiveness of the risk assessments performed before maintenance activities were conducted; (2) the management of risk that, upon identification of an unforseen situation, necessary steps were taken to plan and control the resulting emergent work activities; and (3) that maintenance risk assessments and emergent work problems were adequately identified and resolved. The inspectors referenced Procedures SPP-7.1, Work Control Process, and Instruction 0-TI-DSM-000-007.1, Equipment to Plant Risk Matrix, during these inspection activities.

Work/Activity	Related Documents
EDG 2A-A emergent maintenance due to start failure	PER 01-010354-000; main control room logs; 12 week rolling schedule
Unit 1 component cooling water system surge tank removal from service during refueling outage	daily outage report; main control room logs
EDG 1B-B 24 hour test	main control room logs; 12 week rolling schedule

EDG 2B-B routine surveillance test	PER 01-009001-000
Main control room door maintenance	PER 01-0011492-000; 0-TI-SXX-000-016.0, Breaching the Shield Building, ABSCE, or Control Room Boundaries, Rev. 14
1B-B 480 V shutdown board transfer to alternate and back to normal	12 week rolling schedule; main control room logs

b. Findings

A finding of very low safety significance (Green) was identified by the inspectors for failure to follow work instructions to obtain shift manager concurrence when breaching a control room envelope door. The finding was also a non-cited violation of TS 6.8.1.a, (Procedures and Programs).

On December 7, while conducting routine plant status reviews, the inspectors observed a maintenance activity on one of the main control room doors (vital area access door C49), which required the door to be kept open for several minutes. The worker was replacing the door security locking device. The inspectors confirmed that a security officer was present with the door open as required. The inspectors immediately discussed the work with control room operators to ensure that they were aware the work was taking place. The Unit 1 unit supervisor informed the inspectors that the no one had discussed the work with the control room staff. Controlling documents for the work were WO 01-0011264-000 and Procedure 0-TI-SXX-000-016.0, Breaching the Shield Building, ABSCE, or Control Room Boundaries, Rev. 14. Procedure 0-TI-SXX-000-016.0 required security personnel to obtain concurrence from the operations shift manager before the work began. Because the concurrence was not obtained, the control room operators were unaware of the activity. The licensee initiated PER 01-0011492-000.

The inspectors discussed the issue with engineering personnel to determine the consequence of having the door open without operator awareness. Step F.2 [2].C of 0-TI-SXX-000-016.0, Rev 14, states that operators must contact the doorwatch upon detection of a control room isolation and the door must be closed within 45 seconds. In addition, as discussed in the licensee's evaluation to PER 01-0011492, the main control room pressurization calculation (SQN-31-D053-EPM-DAP-021689) states that only a few square inches of the control room envelope is allowed to be breached to ensure the proper pressurization of the control room area upon a control room isolation. Having the door open far exceeded this assumption.

The issue had a credible impact on safety and could have affected the integrity of the control room envelope because the control room staff was unaware of the door being open and therefore would not have known to contact the involved worker had a control room ventilation isolation signal occurred. Therefore, the control room envelope would not have pressurized as required. The finding was of very low safety significance (Green) because the finding only represented a degradation of the radiological barrier function provided for the control room.

TS 6.8.1.a. requires that written procedures shall be implemented covering the activities referenced in Appendix A of Regulatory Guide 1.33, February 1978 (Paragraph 9). Paragraph 9 recommends, in part, that maintenance that can affect the performance of safety-related equipment be properly performed in accordance with written procedures appropriate to the circumstances. On December 7, security personnel failed to perform Step 3.0 (A)(1) of 0-TI-SXX-000-016.0 which required shift manager concurrence for the control room C49 being kept open while maintenance occurred. However, because the violation was of very low safety significance and was entered into the licensee's corrective action program, the violation is being treated as a Non-Cited Violation consistent with Section VI.A.1 of the NRC Enforcement Policy, and is identified as NCV 50-327,328/01-04-01, Failure to Obtain Shift Manager Concurrence with Control Room Door Open. This violation was entered into the licensee's corrective action program as PER 01-0011492-000.

1R14 Personnel Performance During Non-routine Plant Evolutions

a. Inspection Scope

The inspectors reviewed operator performance associated with a Unit 2 condensate system flowrate perturbation that occurred on November 11. While Unit 2 was operating at 100 percent power, the condensate demineralizer subsystem was inadvertently bypassed which caused a condensate flowrate reduction of about 1,000 gpm from the operating heater drain pumps. The inspectors reviewed plant operating logs, plant computer information, annunciator response procedure (2-AR-M2-C), plant procedures, and conducted discussions with numerous plant personnel. The inspectors also reviewed the associated PER 01-010453-000 and discussed the transient with operations personnel.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed selected technical operability evaluations (TOEs), functional evaluations (FEs), and PERs, listed below, and related documents for issues affecting risk-significant mitigating systems to assess, as appropriate: (1) the technical adequacy of the evaluations; (2) whether continued system operability was warranted; (3) whether other existing degraded conditions were considered as compensating measures; (4) where compensatory measures were involved, whether the compensatory measures were in place, would work as intended, and were appropriately controlled; and (5) where continued operability was considered unjustified, the impact on TS limiting condition for operation (LCO) and the risk-significance in accordance with the SDP. The inspectors referenced Procedure SPP-10.6, Engineering Evaluations for Operability Determination, as needed, during the course of these inspection activities.

Operability Evaluation Inspected	Related Documents Reviewed
Silt found in Unit 1 motor driven AFW pump suction line	PER 01-009704-000
Evaluation of EDG exhaust system degradation	PER 01-007257-000
Emergency gas treatment system damaged dampers	PER 01-009626-000
Moisture found in safety injection pump 2A-A	PER 01-01109-000; TOE 2-01-063-11091
Control room emergency ventilation system charcoal content	PER 01-011525-000 and associated functional evaluation

b. Findings

No findings of significance were identified.

1R16 Operator Work-Arounds

a. Inspection Scope

The inspectors evaluated the cumulative effects of operator workarounds (OWAs) on the reliability, availability, and potential for misoperation of plant systems. Specifically, the cumulative effects were evaluated for the potential to: (1) increase initiating event frequency, (2) affect multiple mitigating systems, or (3) affect the ability of operators to respond in a correct and timely manner to plant transients and accidents. The inspectors also assessed whether OWAs were being identified and entered into the corrective action program at an appropriate threshold. The inspectors reviewed all open OWAs, control room deficiencies, auxiliary unit operator round deficiencies, and selected caution orders.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications

a. Inspection Scope

The inspectors evaluated the eight modifications listed below to verify the implementation of Procedure SPP-9.3, Plant Modifications and Engineering Change Control and the regulatory design requirements. Activities were reviewed to determine that: (1) system energy requirements could be supplied by supporting systems; (2) materials and replacement components were environmentally qualified; (3) replacement components were seismically qualified; (4) code and safety classification of replacement SSCs were consistent with system design bases; (5) work control and performance instructions were

correct and properly utilized; (6) post-modification testing verified system operability; (7) failure modes were bounded by existing analyses; and (8) new procedures or procedure changes had been initiated. The inspectors also reviewed applicable sections of the UFSAR, site drawings, supporting analyses, calculations, TS, and procedures to ensure that revisions were completed or planned as required. In addition the inspectors reviewed completed modifications and observed work and testing activities during the Unit 1 cycle 11 refueling outage.

Completed Modifications Reviewed

- Unit 2 D-20372, TDAFW Pump Upgrade
- Unit 0 D-20236, Safety-Related Chillers TCV Replacements
- Unit 1 D-20348, Install ECCS Breakdown Orifices
- Unit 1 D-20241, Install Refurbished Spare MDAFW Pump
- Unit 1 D-20805, Replace 480 V Unit and S/D PT primary fuses

Ongoing Outage Modifications Reviewed

- Unit 1 E-20521, Upgrade CCP Shaft to Custom 625 Alloy
- Unit 0 D-20071, Replace Safety-Related Inverters - open
- Unit 1 D-20664A, Replace Cold Leg Accumulator Transmitters

The inspectors also reviewed the corrective action program audits and self assessments listed below to verify that the licensee was identifying modification-related issues and initiating actions to resolve concerns.

Audits and Self-Assessments Reviewed

- Nuclear Assurance-TVAN-Wide-Audit Report SS0006, Engineering Functional Area Audit, dated February 16, 2001
- Self Assessment, SQN-MS-00-02, Processing Design Change Notices, dated June 21 through July 24, 2000
- Self Assessment, SON-M7M-00-002, Plant Modification and Design Change Control, November 15 through 26, 1999

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing

a. Inspection Scope

The inspectors reviewed Procedure SPP-6.3, Pre/Post Maintenance Testing (PMT) which governs the licensee's PMT process, and work orders (WO) and/or test activities, as appropriate, for selected risk-significant mitigating systems to assess whether: (1) the effect of testing on the plant had been adequately addressed by control room and/or engineering personnel; (2) testing was adequate for the maintenance performed; (3) acceptance criteria were clear and adequately demonstrated operational readiness consistent with design and licensing basis documents, (4) test instrumentation had current calibrations, range and accuracy consistent with the application, (5) tests were performed as written with applicable prerequisites satisfied; (6) jumpers installed or leads lifted were properly controlled; (7) test equipment was removed following testing; and (8) equipment was returned to the status required to perform its safety function.

Post Maintenance Test Inspected	Related Documents Reviewed
Replace vital inverter I-I	WO 99-003753-010
EDG 1A-A 24 hour run after cylinder power pack replacements	WO 01-008941-002
Unit 1 RHR pump failed to start	WO 01-009683-000; PER 01-009681-000
Seat leakage repair for RHR valve 1-HCV-74-34	WO 98-010966-000
EDG 2A-A failure to start	0-SI-OPS-082-007.A, Electrical Power System Diesel Generator 2A-A
Unit 1 turbine driven AFW pump replacement	1-PI-SFT-003-001.C, TDAFW Pump Full Flow Test; 1-SI-SXP-003-201.S, Turbine Driven Auxiliary Feedwater Pump 1-S Performance Test
Positioner replacement for Unit 1 main steam loop 4 pressure relief control valve, 1-PCV-001-0030	WO 01-010814-000; 0-SI-SXV-000-206.0, Testing of Category A and B Valves After Work Activities, Upon Release From A Hold Order, Or When Transferred From Other Documents

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities

a. Inspection Scope

The inspectors observed numerous activities associated with the Unit 1 cycle 11 refueling outage. These activities are listed below.

Review of Outage Plan

The inspectors reviewed the Unit 1 Cycle 11 Outage Safety Plan and verified that the licensee had appropriately considered risk, industry experience and previous site specific problems. The inspectors also ensured that the licensee had considered contingencies for loss of key safety functions.

Monitoring of Shutdown Activities

The inspectors observed portions of the plant cooldown and reviewed plant procedures to verify that TS restrictions were satisfied. Procedure 0-SI-SXX-068-127.0, RCS and Pressurizer Temperature and Pressure Limits was reviewed.

Outage Configuration Management

On numerous occasions during the outage, the inspectors verified that the licensee maintained defense-in-depth commensurate with the outage risk plan. The inspectors reviewed the daily outage report which described the defense-in-depth status of the unit and verified operators were aware of changing plant configurations.

Reactor Coolant System Instrumentation

The inspectors verified that selected reactor coolant system (RCS) pressure, level, and temperature instrumentation were installed and provided reliable and accurate indication.

Electrical Power

The inspectors verified that the status and configurations of electrical systems met TS requirements and the outage risk control plan and that switchyard activities were controlled commensurate with safety and the outage risk control plan assumptions. The inspectors reviewed Procedure OPDP-2, Switchyard Access, to confirm that switchyard access was properly controlled.

Decay Heat Removal (DHR) System Monitoring

The inspectors observed DHR parameters on numerous occasions to assess proper system function and that the steam generators, when relied upon, were a viable means of backup DHR.

Spent Fuel Pool Cooling System Operation

The inspectors assessed outage work for potential impact on the ability of the operations staff to operate the spent pool cooling system during and after core offload. The inspectors reviewed Procedure AOP-M.06, Loss of Spent Fuel Cooling, which is the licensee's procedure for mitigation of a loss of spent fuel pool cooling and walked down the system after core offload to confirm proper system operation.

Inventory Control

The inspectors reviewed flow paths, configurations, and alternative means for inventory addition for consistency with the outage risk plan. The inspectors also reviewed activities with the potential to cause loss of inventory for adequacy of controls to prevent inventory loss. The inspectors reviewed the licensee's daily outage report, toured the main control room, and discussed with operators availability of systems needed for inventory control. The inspectors also reviewed Procedure 1-PI-OPS-068-673.D, Daily Requirements for Reduced Inventory/Midloop Operation.

Reactivity Control

The inspectors evaluated licensee control of reactivity for compliance with TS. The inspectors also evaluated outage activities for potential to cause unexpected reactivity changes for inclusion and proper control under the outage risk plan.

Containment Closure

The inspectors reviewed control of containment penetrations for compliance with refueling operations TS to ensure that containment closure could be achieved during selected configurations. Procedure 0-GO-15, Containment Closure Control was reviewed.

Reduced Inventory and Mid-Loop Conditions

The inspectors reviewed numerous activities associated with reduced inventory and mid-loop operations with emphasis on the licensee's ability to monitor and control RCS water level. The inspectors also evaluated the effect of distractions on operator ability to maintain required reactor vessel level during mid-loop operations. Procedures 0-GO-13, Reactor Coolant System Drain and Fill Operations, 1-PI-OPS-068-673.D, Daily Requirements for Reduced Inventory/Midloop Operation, and 0-PI-IXX-068.001.0, Daily Requirements for Reduced Inventory/Midloop were reviewed.

Refueling Activities

The inspectors reviewed fuel handling operations for conformance with TS and approved procedures and confirmed that the location of selected fuel assemblies were tracked from core offload through core reload. The inspectors reviewed Procedures TI-45, Physical Verification of Core Load Prior to Vessel Closure, SPP-5.8, Special Nuclear Material Control (fuel assembly transfer forms) and 0-RT-NUC-000-002.0, Core Reconfiguration, and 0-GO-9, Refueling Procedure during this inspection activity.

Monitoring of Heatup and Startup Activities

The inspectors reviewed on a sampling basis that TS and administrative procedure prerequisites for mode changes were met prior to changing modes or plant configurations. The inspectors walked down accessible areas in the containment building prior to plant startup to verify that debris had not been left which could affect performance of the containment sump. The containment sump was also inspected. The inspectors also reviewed Procedure 0-SI-OPS-000-187.0, Containment Inspection.

Identification and Resolution of Problems

The inspectors verified that the licensee had identified problems related to refueling outage activities at an appropriate threshold and had entered these problems into the corrective action program. The inspectors reviewed PERs initiated by the licensee and for selected PERs, the inspectors reviewed corrective actions plans for their appropriateness.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors witnessed surveillance tests and/or reviewed test data of selected risk-significant SSCs conducted using the surveillance instructions (SI), listed below, to assess, as appropriate, whether the SSCs met TS, the UFSAR, and licensee procedure requirements, and to determine if the testing effectively demonstrated that the SSCs were operationally ready and capable of performing their intended safety functions.

Surveillance/Equipment Test Inspected	Related Documents Reviewed
Loss of offsite power with safety injection test	1-SI-OPS-082-026.A, Loss of Offsite Power with Safety Injection-D/G 1A-A Test
Various valve tests during Unit 1 outage	0-SI-SXV-001-266.0, ASME Section XI Valve Testing
Testing of Unit 1 safety injection system hot and cold leg check valves	1-SI-SXV-063-201.0, Safety Injection System Hot Leg and Cold Leg Injection Check Valve Full Stroke Test
EDG 2A-A starting air system test	0-PI-SXV-082-203.A, Diesel Starting Air Valve Test for DG Set 2A-A, Time Frame A
EDG 2A-A starting air system test	0-PI-SXV-082-203.B, Diesel Starting Air Valve Test for DG Set 2A-A, Time Frame B

Unit 1 ice condenser ice weighing results	0-SI-MIN-061-105.0, Ice Condenser - Ice Weighing; PERs 01-009826-000 and 01-009980-000

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed temporary plant modification temporary alteration control forms (TACFs) 2-01-0011-003, MDAFW Pump Start Circuitry related to the motor driven auxiliary feedwater pump start circuitry. The inspectors reviewed the temporary modifications and associated 10 CFR 50.59 screening and evaluation against the system design bases documentation, including the system drawing, UFSAR, and TS to ensure that risk-significant functions of the circuit were not affected. The inspectors reviewed the results of post-installation testing to confirm the results were satisfactory and that the actual impact of the temporary modification on the permanent systems and interfacing systems have been adequately verified by test. Additionally, the inspectors interviewed operators to verify operator awareness of the modification.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP2 Alert and Notification System Testing

a. Inspection Scope

The inspector reviewed the testing program for the alert and notification system (ANS), which comprised 108 sirens. The schedule consisted of biweekly silent tests, annual growl tests (in conjunction with routine maintenance), and monthly full-volume tests. System changes, post-maintenance testing methodology, test records, and a sample of corrective actions were discussed and evaluated against requirements of the Radiological Emergency Plan (REP).

b. Findings

No findings of significance were identified.

1EP3 Emergency Response Organization (ERO) Augmentation

a. Inspection Scope

The inspector reviewed the maintenance and testing of the licensee's capability to staff emergency response facilities within stated timeliness goals. Documentation of six unannounced pager tests between July 2000 and August 2001 was evaluated. Records of off-hour ERO augmentation drills conducted on December 21, 2000 and June 22, 2001 were reviewed. Follow-up activities for a sample of problems identified through augmentation testing were reviewed to determine whether appropriate corrective actions were implemented as required by the REP.

b. Findings

No findings of significance were identified.

1EP4 Emergency Action Level and Emergency Plan Changes

a. Inspection Scope

The inspector reviewed changes to the REP, as contained in Revisions 59, 60, and 61, against the requirements of 10 CFR 50.54(q) to determine whether any of the changes decreased REP effectiveness. All of the listed revisions involved changes to the generic portion of the REP. In addition, Revision 60 contained modifications to Appendix B (site-specific for Sequoyah), including minor changes to the EALs. The inspector verified whether the EAL modifications were reviewed with, and agreed upon by, state and local officials prior to implementation, as required by Section IV.B of Appendix E to 10 CFR Part 50.

b. Findings

No findings of significance were identified.

1EP5 Correction of Emergency Preparedness Weaknesses and Deficiencies

a. Inspection Scope

The inspector evaluated the efficacy of licensee programs that addressed weaknesses and deficiencies in emergency preparedness. Documents reviewed included PERs, Audit Report SSA0005, self-assessment reports, and critique reports for ERO drills conducted on December 5, 2000 and April 6, 2001. In addition, the inspector reviewed in detail PER 00-008706-000, regarding a failure on September 26, 2000 to properly declare a Notification of Unusual Event.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically-Significant Areas

a. Inspection Scope

The inspectors reviewed radiological surveys and access controls for a resin transfer from the spent resin storage tank into a high integrity container (HIC) located on a trailer in the railroad bay and an at-power entry into the Unit 1 lower containment. The resin transfer was conducted under radiation work permit (RWP) 01000030 and the containment entry was conducted under RWP 01000519. The inspectors observed the pre-job briefing and RWP briefing for both jobs to evaluate their depth and thoroughness. Health physics technician job coverage and radiation worker performance were observed during the course of the resin transfer and for a portion of the containment entry.

In addition, the inspectors independently verified dose rates at selected locations in two rooms of the auxiliary building and checked selected locked high radiation area doors for status and material condition, as well as implementation of Radiological Control Instructions (RCI) -24, Control of Very High Radiation Areas; RCI-28, Control of Locked High Radiation Areas; and RCI-29, Control of Radcon Keys.

The inspectors reviewed licensee self-assessments and PERs addressing radiological surveys, access controls, and unplanned exposures for characterization and timeliness of resolution. The following self-assessment reports were reviewed:

- SQN-RP-01-001, Radiation Worker Compliance
- SQN-RP-01-003, Contamination Control
- SQN-RP-01-004, Radioactive Material Control
- SQN-RP-01-005, Radiological Dose Controls

b. Findings

No findings of significance were identified.

2OS2 As Low As Reasonably Achievable (ALARA) Planning and Controls

a. Inspection Scope

The inspectors reviewed the plant's collective exposure history, current exposure dose trends, and the year 2001 annual site dose goal to determine if the licensee was implementing ALARA practices as required by 10 CFR 20.1101(b) and Procedure RCI-10, ALARA Planning Report Criteria. The inspectors also evaluated source term reduction efforts and the incorporation of ALARA into licensee RWPs. The evaluation included: ALARA planning, dose goals and estimates, daily dose results, job dose trends, problem

identification and resolution, and ALARA Committee meeting minutes. The inspectors attended pre-job briefings for the No. 1 and No. 4 steam generator nozzle dam removal (RWP No. 01034020) and the installation of SG primary covers and inserts (RWP # 01034040). For the former activities, the inspectors observed the job via closed circuit television to evaluate implementation of radiological controls.

The inspectors reviewed the licensee's calculated doses for selected contamination events and independently verified the licensee's calculations. The cavity decontamination results and the health physics confirming surveys were reviewed. Radiation surveys, documenting dose reduction by the addition of a six foot extension of lead shielding around the reactor head, were evaluated. The inspectors observed health physics activities for the lift, transport and installation of the reactor internals and reactor vessel head.

The inspectors evaluated airborne radioactivity surveys which were performed using Procedure RMD FO-05, Airborne Radioactive Surveys, Revision 18. The data included particulate evaluation for airborne transuranic isotopes.

The inspectors toured the Dry Active Waste Building, Auxiliary Building and Refueling Floor, and the contaminated trash storage and sorting area to evaluate housekeeping, outage preparations and radioactive material storage.

b. Findings

No findings of significance were identified.

Cornerstone: Public Radiation Safety

2PS1 Radioactive Gaseous and Liquid Effluent Treatment and Monitoring Systems

a. Inspection Scope

The inspectors reviewed the licensee's 2000 Annual Radiological Effluent Release Report to verify that the program was implemented as described in Radiological Effluent Technical Specifications and Offsite Dose Calculation Manual (RETS/ODCM) and to verify that all radiological effluent releases resulted in doses to members of the public which were within the limits defined in 10 CFR 20.1301. The inspectors also reviewed the report for any anomalous results to determine if their resolution was adequate. In addition, the inspectors reviewed the results of an upper and lower Unit 1 containment purge performed during the inspection period which included implementation of SI 0-SI-CEM-030-410.2, the radioactive gaseous effluent release permit, and the licensee's projected doses to members of the public.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification

Licensee records were reviewed to determine whether the submitted PI statistics were calculated in accordance with the guidance contained in NEI 99-02, Regulatory Assessment Performance Indicator Guideline.

Cornerstone: Mitigating Systems

.1 Safety System Unavailability, Residual Heat Removal

a. Inspection Scope

The inspector assessed the accuracy of the PI for RHR unavailability for the period of July1, 2000, through June 30, 2001, for both units through a review of out-of-service logs maintained by the system engineer, operating logs, and the maintenance rule database and verified selected planned, unplanned, and fault exposure unavailable hours.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

.2 Emergency Response Organization Drill/Exercise Performance PI

a. Inspection Scope

The inspector assessed the accuracy of the PI for ERO drill and exercise performance (DEP) through review of a sample of drill records. Documentation was reviewed for an ERO drill conducted on April 6, 2001 and licensed operator requalification drills conducted on October 31, 2000 to verify the licensee's reported data regarding successes in emergency classifications, notifications, and protective action recommendations.

b. Findings

No findings of significance were identified.

.3 ERO Drill Participation PI

a. Inspection Scope

The inspector assessed the accuracy of the PI for ERO drill participation through review of the training records for 6 of the 48 individuals assigned to key positions in the ERO as of the end of the second quarter of 2001.

b. Findings

No findings of significance were identified.

.4 Alert and Notification System (ANS) Reliability PI

a. Inspection Scope

The inspector assessed the accuracy of the PI for the ANS reliability through review of a sample of the licensee's records of siren tests conducted from July 1, 2000 to June 30, 2001.

b. Findings

No findings of significance were identified.

Cornerstone: Public Radiation Safety

.5 RETS/ODCM Radiological Effluent Occurrences

a. Inspection Scope

The inspectors reviewed licensee PERs for liquid or gaseous effluent releases that were reported to the NRC, Licensee Event Reports, and the 2000 Annual Radioactive Effluent Release Report plus raw/draft data for the first three quarters of 2001 to assess whether all radiological effluent release occurrences in excess of limits were counted as PI occurrences.

b. Findings

No findings of significance were identified.

Cornerstone: Occupational Radiation Safety

.6 Occupational Exposure Control Effectiveness

a. Inspection Scope

The inspectors reviewed licensee PERs and self-assessments for the first three quarters of 2001 for high radiation area, very high radiation area, and unplanned exposure occurrences to assess whether nonconformances were properly classified as PI

occurrences. The licensee's database, which contained RCA exit transactions with exposures greater than 100 mrem, was reviewed by the inspectors to determine whether the exposures were within RWP limits and whether any met the criterion for a PI occurrence.

b. Findings

No findings of significance were identified.

4OA3 Event Follow-up

.1 Unplanned Radiological Release

a. Inspection Scope

The inspectors reviewed the licensee's response to an event which occurred on October 30, 2001, involving a potentially radioactive release of water from the turbine building sump to the yard drainage. During this evolution, the station sump discharge radiation monitor reached its alarm setpoint of 600 counts per minute and alarmed. A water sample taken from the turbine building sump confirmed radioactivity in the sump. The release to the environment was terminated by re-aligning the sump discharge to the low volume waste treatment pond, which is isolated from the environment, and the licensee declared a notification of unusual event (NOUE) due to an unplanned radiological release which exceeded procedure-prescribed effluent radiation levels. The declaration of the NOUE required the licensee to notify the NRC in accordance with 10 CFR 50.72. Further analysis determined that the total activity in the turbine building sump was actually less than 35 percent of the limit specified in the Off-site Dose Calculation Manual (ODCM). Liquid samples taken at the discharge of the yard drainage pond to the dilution pond and at the discharge of the dilution pond to the Tennessee River detected no radioactivity being released into the environment. As no release limits were exceeded, the criteria for declaring a NOUE (two times the ODCM limits) had not been met and the licensee rescinded the 10 CFR 50.72 notification. The licensee's subsequent investigations to determine the source of the radioactivity did not conclusively identify the leak path.

b. Findings

No findings of significance were identified.

.2 (Closed) Unresolved Item (URI) 50-328/01-03-05: Failure to Realign Boric Acid Tank Flow Path to the RCS. On August 8, 2001, during a routine automatic RCS makeup to the volume control tank, operators observed that the makeup automatically stopped. Shortly afterwards, a valve lineup check was performed which identified filter bypass valve 2-VLV-62-1055B to be closed when it should have been open. Having the valve closed isolated the boric acid flow from boric acid tank (BAT) B to the RCS (BAT B is the normal flow path for Unit 2). The licensee's investigation determined that filter bypass valve 2-VLV-62-1055B had previously been closed on August 5, 2001, to transfer boric acid from BAT C to BAT B. When the system was realigned to its normal configuration operators had failed to reopen the valve.

The inspectors evaluated the safety significance of the issue and determined that failure to reopen the valve had a credible impact on safety because it rendered BAT B unavailable for anticipated transient without scram (ATWS) and SGTR mitigation. During this time, however, for ATWS mitigation, highly concentrated boric acid was available from the RWST through the high pressure charging pumps. The ATWS mitigation portion of the finding was therefore of very low safety significance. For SGTR events, the function to provide highly concentrated boric acid to refill the RWST was not available. An SDP Phase 3 assessment was performed using the risk achievement worth values for emergency boration and for makeup to the RWST from the licensee's risk model. The SGTR event portion of the finding was determined to have very low safety significance because of the short time (3 days) that the equipment was unavailable.

TS 6.8.1.a, requires that written procedures shall be implemented covering the activities referenced in applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978 (Paragraph 3). Paragraph 3 recommends, in part, that instructions for operation of safety-related systems should be prepared. Step 5.1.2[5][a] of 0-SO-62-10 was not performed as required to realign BAT B flow path to the RCS after boric acid was transferred from BAT C to BAT B. However, because the violation was of very low safety significance and was entered into the licensee's corrective action program, the violation is being treated as a Non-Cited Violation consistent with Section VI.A.1 of the NRC Enforcement Policy, and is identified as NCV 50-328/01-04-02, Failure to Realign Boric Acid Tank Flow Path to the RCS. This violation was entered into the licensee's corrective action program as PER 01-006956-000.

.3 (Closed) URI 50-327, 328/01-03-03: Failure to Implement 2B-B EDG Maintenance Procedure. Between August 27, 2001 and September 1, 2001, the licensee identified degraded engine components in engine number 2 (2B2) for EDG 2B-B. Specifically, there was severe degradation of a piston wrist pin and bearing insert for cylinder number 11. The licensee determined that a failure to properly implement a maintenance procedure completed five months earlier had occurred. Maintenance personnel had incorrectly marked an out-of-specification condition as acceptable and no corrective action was taken. The out-of-specification condition was the as-found End 2 measurement of 0.069" for the number 11 cylinder piston-to-cylinder head clearance. The maximum allowed by the procedure was 0.068".

The inspectors determined that the licensee's failure to perform corrective action (i.e. assess the degraded condition) of the out-of-specification condition had a credible impact on safety. Had the assessment been performed, the inspectors determined that it was reasonable that the licensee would have identified that the previous measurement result performed in 1999 was significantly less (0.042"). The change from 0.042" to 0.069", which is a difference of 0.027", is significant because the vendor recommends that the associated power pack be condemned when a change of 0.030" is observed. The inspectors discovered this vendor information subsequent to the initial inspectors' review of the issue. This significant change indicated a degradation in the wrist pin and insert bearing and potentially impacted EDG reliability. An assessment would have provided an earlier opportunity to identify the degraded EDG condition and subsequently reduced the period of reduced reliability. Because corrective action was not taken, there was no consideration for replacing the power pack. Discussions with engineering personnel indicated that the power pack would have been replaced had the evaluation been

performed. Although the licensee concluded the EDG was operable, it was indeterminate whether the EDG could achieve its mission of a 24-hour run time assumed in the licensee's probabilistic risk assessment. The inspectors reviewed the licensee's basis for past operability and determined that there was no analysis, testing, operating experience or vendor information that indicated the EDG was not operable under the conditions described. The finding was therefore determined to have a very low safety significance. This licensee identified finding involved a violation of TS 6.8.1.a, Procedures and Programs. The enforcement aspects of the violation are discussed in Section 4OA7.

.4 (Closed) Licensee Event Report (LER) 50-327/328/2000-S01-00: On June 28, 2000, at 6:53 p.m., a security officer was dispatched to assist an employee access through a vital area door. The officer determined that the card reader was not working properly. At 6:55 p.m., a security alert was declared based on a computer security malfunction. Compensatory measures were implemented for the loss of vital area door alarms. The security officers determined that the computer malfunction affected more than the vital area doors and initiated compensatory measures for the protected area perimeters. During the time of the event, the licensee determined that four perimeter alarms had not been received in the central/secondary alarm stations while the system was down. This discovery identified a potential vulnerability in the security system that could have allowed unauthorized or undetected access to the protected area. There were no indications that unauthorized or undetected access to the protected area occurred. Compensatory measures for the computer failure was established in accordance with regulatory requirements. The licensee had met the regulatory requirements for posting during computer system failures: however, the one hour report was not withdrawn because 10CFR 73.71, Appendix G, 1 - c, states, "events to be reported within one hour of discovery are, any failure degradation or the discovered vulnerability in a safeguards system that could allow unauthorized or undetected access to a protected area, material access area....". The inspector reviewed the LER and determined that this event did not constitute a violation of NRC requirements. No new issues of significance were revealed by the LER review.

4OA5 Other

.1 Temporary Instruction 2515/145 Review

a. Inspection Scope

The inspectors witnessed the licensee perform an examination of the Unit 1 reactor vessel head control rod drive mechanism penetrations. The inspectors reviewed the licensee's examination procedure, reviewed training qualifications of those who performed the examination, and reviewed the licensee's examination report. The inspectors also independently checked visible portions of the reactor head for indications of boron deposition due to RCS leakage, debris, insulation, and boron from other sources.

b. Findings

The metallurgical engineer who performed the examination was knowledgeable and had performed numerous examinations during previous refueling outages. The licensee's training requirements were identified in Procedures SPP-9.7, Corrosion Control Program, NEDP-7, Engineering Support Personnel Training, and NEDP-7, Qualification Guides (QG). In order to perform the examination, the licensee required the involved engineer to complete a qualification card addressing knowledge of the boric acid program and necessary corrective actions based on examination results. The qualification card also required the engineer to demonstrate proficiency followed by a task evaluation. In this particular case, the task evaluation was completed based on the individual's experience. The inspectors reviewed the training records and no training issues were identified. The licensee used PM 040851000 and WO 00-007281-000 as controlling documents for the examination. The inspectors did not identify any procedural use issues with the recognition that these implementing procedures were mostly used for tracking work completion versus providing any substantive guidance. The examination was mostly accomplished by skill of the craft methods using a "best effort visual examination." During the examination, the inspectors discussed with the engineer the primary water stress corrosion cracking phenomenon described in NRC Bulletin 2001-01 and the engineer was knowledgeable was capable of identifying potential leakage conditions.

The reactor vessel head was relatively clean. There was very little debris or dirt and no evidence of boron accumulation. There were small amounts of fiber insulation and small boron particles of about 1/32 inch in diameter that the licensee determined was from a previous control rod drive mechanism (CRDM) canopy seal weld repair that occurred during the cycle 10 refueling outage. There was some evidence of boron "streaks" or "staining" on several CRDM housing supports that the licensee attributed to leakage from the CRDM canopy seal weld or conoseal leakage. The physical layout of the reactor head and the vessel shroud made the examination difficult for extensive viewing. There was a viewing area of about two inches between the head surface and the vessel shroud plate. Because of this small viewing area, the only penetrations examined were those on the outer periphery and the area on the downward side of the second row of penetrations. In addition, the top side of the outer periphery penetrations were not readily visible. Lead blankets were installed on the vessel shroud for ALARA purposes. The blankets had to be temporarily moved for the examination which made the examination somewhat difficult. The licensee stated that the method observed for the examination was the normal means for the examination.

4OA6 Meetings, Including Exit

The inspectors presented the inspection results to Mr. Richard Purcell, Site Vice President, and other members of licensee management on January 4, 2002. The inspectors asked the licensee whether any of the material examined during the inspection should be considered proprietary. No proprietary information was identified.

40A7 Licensee Identified Violations

The following finding of very low significance was identified by the licensee and is a violation of NRC requirements which met the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600 for being dispositioned as an NCV.

NCV Tracking Number	Requirement Licensee Failed to Meet
NCV 50-328/01-04-03	TS 6.8.1.a, Procedures and Programs, requires that written procedures be implemented covering the activities recommended in Appendix A of Regulatory Guide 1.33, Rev. 2. (Paragraph 9). Paragraph 9 recommends that maintenance that can affect the performance of safety-related equipment be properly performed in accordance with written procedures appropriate to the circumstances. Contrary to the above, on March 14, 2001, when the measured 2B2 engine cylinder number 11 piston-to-head clearance exceeded the acceptance criteria, the licensee took no corrective action as required in step 6.15 (note 2) of Procedure 2-PI-MDG-082-002.B, Two Year Preventive Maintenance of Diesel Engine Set 2B-B, Rev. 4. The issue is in the licensee's corrective action program as PER 01-007771-000 (Green).

SUPPLEMENTAL INFORMATION

PARTIAL LIST OF PERSONS CONTACTED

Licensee

T. Carson, Maintenance Manager
H. Cothron, SG Program Manager
R. Drake, Maintenance and Modifications Manager
R. Ford, Emergency Preparedness Manager
E. Freeman, Operations Manager
J. Gates, Manager, Business and Work Performance
C. Kent, Radcon/Chemistry Manager
D. Koehl, Plant Manager
M. Lorek, Assistant Plant Manager
D. Lundy, Site Engineering Manager
B. Marks, Corporate Emergency Preparedness Manager
R. Purcell, Site Vice President
R. Rogers, Design Manager
P. Salas, Licensing and Industry Affairs Manager
J. Smith, Site Licensing Supervisor
K. Stephens, Security Manager
J. Valente, Engineering & Support Services Manager
J. Vincelli, Radcon Manager
J. Wilkes, Operations Superintendent

NRC

R. Bernhard, Region II Senior Reactor Analyst
R. Moore, Region II Reactor Inspector

ITEMS OPENED AND CLOSED

Opened and Closed

50-327,328/01-04-01	NCV	Failure to Obtain Shift Manager Concurrence with Control Room Door Open (Section 1R13)
50-328/01-04-02	NCV	Failure to Realign Boric Acid Tank Flow Path to the RCS (Section 4OA3.2)
50-328/01-04-03	NCV	Failure to Implement 2B-B EDG Maintenance Procedure (Section 4OA7)

Attachment

Closed

50-328/01-03-05	URI	Failure to Realign Boric Acid Tank Flow Path to the RCS (Section 4OA3.2)
50-327, 328/01-03-03	URI	Failure to Implement 2B-B EDG Maintenance Procedure (Section 4OA3.3)
50-327/328/2000-S01-00	LER	Malfunction of Security Computer System

LIST OF DOCUMENTS REVIEWED

Section 1R08

Procedures

Steam Generator Repair Guidelines, Sequoyah Nuclear Plant, Rev 9, June 2001

Steam Generator Eddy Current Examination Guideline, Rev 3, June 2001

Other Documents

PER 01-010085-000, Axial Crack found in U-bend Region of Row 3 Tube in SG 4

PER 01-010352-000, During Cycle 11 ET exam, a circumferential crack in SG 4 hot side, tube sheet, was found to have been called a circumferential crack during Cycle 10 ET examinations, but was not plugged.

Section 1R17B

Completed Modifications

Unit 2 D-20372, TDAFW Pump Upgrade - closed
Unit 0 D-20236, Safety-Related Chillers TCV Replacements - closed
Unit 1 D-20348, Install ECCS Breakdown Orifices - closed
Unit 1 D-20241, Install Refurbished Spare MDAFW Pump - closed
Unit 1 D-20805, Replace 480 V Unit and S/D PT primary fuses - closed

Ongoing Outage Modifications

Unit 1 E-20521, Upgrade CCP Shaft to Custom 625 Alloy - open
Unit 0 D-20071, Replace Safety-Related Inverters - open
Unit 1 D-20664A, Replace Cold Leg Accumulator Transmitters
Support Documentation

Completed surveillance 1-SI-SFT-063-001.0, Revision 8, Safety Injection System Hot Leg and Cold Leg Injection Flow Test
Chronological Test Log for 1-SI-SFT-063-001.0, Revision 8, Flow and Vibration Testing Orifices, 3/8/00
WO 99-006732-000 PMT on Flow Orifices
Completed surveillance 1-SI-SFT-062-001.0, Revision 10, Charging Pump Injection Flow Test
Calculation CDQ 1063-982002, Revision 4, Reconciliation of Pressure-Reducing Orifice Installations on SI Piping
CDEF-1322, Train B EBR Chiller [Maintenance Rule Report]
Problem Evaluation Report 01-008859-000, SDBR Chiller ERCW 1-TCV-67-158 Hanging in Midposition
WO98-13096-000, Steps for Setting the Throttle Position of 0-67-623A (TYPICAL)
Completed 0-SI-SXI-000-201.0, Revision 6, ASME Section XI Inservice Pressure Test for 0-TCV-67-197 and Associated Welds (TYPICAL)
Completed surveillance 2-PI-SFT-003-727.C, Revision 5, TDAFW Pump Full Flow Test, 11/12/00
Completed surveillance 2-SI-SXP-003-201.S, Revision 6, Turbine Driven Auxiliary Feed Water Pump 2A-S Performance Test, 11/22/00
Problem Evaluation Report 00-010116-000, Temporary Test Drive Device Failed
Completed surveillance 2-MI-MFT-003-002.0, Revision 7, Auxiliary Feedwater Pump 2A-S Overspeed Trip Tests, 11/8/00
Completed calibration 2-SI-ICC-003-148.B, Revision 15, Channel Calibration of Steam Generator 3 Motor Driven Auxiliary Feedwater Train B Level Loop 2-L-3-1, 11/9/00
Calculation SQN-VD-VDC-001, Revision 25, TDAFWP Control Circuit Analysis
Calculation SQN-03 D53 EPM-GLC-031193. Revision 4, Condensate Storage Tank Useable Volume for Aux Feedwater Use
Completed surveillance 2-SI-IRT-099-701.0, Revision 5, Response Time Testing of Turbine Driven Auxiliary Feedwater Pump, 11/12/00
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TVA-SQN-TS-99-14, Units 1 and 2 - Technical Specification Change No. 99-14, Cold Leg Accumulator Volume and Pressure Limit Change, 2/4/00
WO 00-005918-000, SQN-1-LT-063-0060, SIS Accum Tank 4 Level Xmtr
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0-MI-MRR-062-610.0, Revision 8, Centrifugal Charging Pump Inspection, completed 11/5/01
0-MI-MRR-062-001.0, Revision 10, Inspection/Repair of CVCS Centrifugal Charging Pump Seals, completed 11/5/01
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WO 99-001985-000, Revision 0, Centrifugal Charging Pump 1B-B

Reviewed Procedures

SPP-9.3, Revision 5, Plant Modifications
MMDP-1, Revision 4, Maintenance Management System

Reviewed Audits and Self-Assessments

Nuclear Assurance -TVAN-Wide- Audit Report NO. SS0006 - Engineering Functional Area Audit, February 16, 2001
Self Assessment NO. SQN-MS-00-02, Processing Design Change Notices, 6/21/00 to 7/24/00
Self Assessment NO. SON-M7M-00-002, Plant Modification and Design Change Control, 11/15/99 to 11/26/99