UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
SAM NUNN ATLANTA FEDERAL CENTER
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ATLANTA, GEORGIA 30303-8931

October 30, 2000

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 50-327/00-06 AND 50-328/00-06

Dear Mr. Scalice:

On September 30, 2000, the NRC completed an inspection at your Sequoyah 1 & 2 reactor facilities. The enclosed report presents the results of that inspection which were discussed on October 10, 2000, with Mr. Dennis Koehl 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 an apparent violation and three issues of very low safety significance (Green). The apparent violation involved activities in the area of physical protection and has not yet been formally dispositioned by the NRC. 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 these non-cited violation, 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 the Sequoyah facility.

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 or 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/NRC/ADAMS/index.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 Inspection Report w/Attachment

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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:	50-327/00-06, 50-328/00-06
Licensee:	Tennessee Valley Authority (TVA)
Facility:	Sequoyah Nuclear Plant, Units 1 & 2
Location:	Sequoyah Access Road Soddy-Daisy, TN 37379
Dates:	July 2, 2000 - September 30, 2000
Inspectors:	R. Gibbs, Senior Resident Inspector D. Starkey, Resident Inspector R. Telson, Resident Inspector R. Carrion, Project Engineer E. Testa, Senior Health Physicist D. Thompson, Physical Security Specialist
Approved by:	P. Fredrickson, Chief Reactor Projects Branch 6 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000327-00-06, IR 05000328-00-06, on July 2, 2000 - September 30, 2000, Tennessee Valley Authority, Sequoyah, Units 1 & 2. Maintenance risk assessments and emergent work evaluation, response to contingency events, and other activities.

The report covers a thirteen-week period of resident inspection. In addition, it includes the results of inspections by a regional senior health physicist and a physical security specialist.

The significance of issues is indicated by their color (green, white, yellow, red) and was determined by the NRC's Significance Determination Process (SDP), as found in NRC Inspection Manual Chapter 0609 and as discussed in the attached summary of the NRC's Reactor Oversight Process. Findings for which the SDP does not apply are indicated by "No Color" or by the severity level of the applicable violation.

Cornerstone: Mitigating Systems

- Green. Three examples of a non-cited violation of TS 4.3.2.1.3 were identified for failure to conduct response time testing, following maintenance, of three flow control valves. These violations are being treated as additional examples of NCV 50-327/00-02-05.

  The additional examples had a very low safety significance because there was not an actual loss of safety function of any of the systems involved (Section 4OA5.1).

- Green. An NCV of TS 6.8.1.a was identified for failure to schedule a post-maintenance test on a Unit 2 electrical board room air handling unit (AHU), when the test on the AHU could not be performed at the time of field work completion.

  The finding had very low safety significance because a redundant AHU was operable and in service so there was not an actual loss of board room cooling system safety function (Section 1R13).

Cornerstone: Physical Protection

- To Be Determined. An Apparent Violation of the Physical Security Plan and Contingency Plan was identified for the licensee allowing a TVA employee to enter the site without removing his shoes after the first alarm was received on the access portal metal detector (Section 3PP2).

- Green. An NCV of the Training and Qualification Plan was identified for failure to ensure that posted security force personnel have no weaknesses or abnormalities that would adversely affect their performance of assigned duties.

  The finding had very low safety significance due to the non-predictable basis of the single equipment failures and because there was no evidence that the vulnerability had been exploited (Section 3PP3).

Report Details

Summary of Plant Status: Unit 1 operated at or near 100 percent power until September 25 when an uncomplicated automatic reactor trip occurred. The unit remained shutdown for the remainder of the inspection period. Unit 2 operated at or near 100 percent power until September 6 when the unit began coasting down for a scheduled refueling outage. Unit 2 was operating at 90 percent power at the end of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, and Emergency Preparedness

1R04 Equipment Alignment

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 documents to determine correct system lineups and an inspection of critical components (e.g., power supplies, support systems) to identify any discrepancies which could affect operability of the redundant train or backup system.

- The diesel-driven fire pump while electric fire pump out-of-service for breaker replacement
- The A auxiliary air compressor during routine maintenance of the B auxiliary air compressor

b. Issues and Findings

No findings of significance were identified.

1R05 Fire Protection

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 Procedure SPP-10.10, Control of Transient Combustibles, Revision 0, and prefire plans for the areas listed below, as appropriate.

- Essential raw cooling water building
- Unit 1 480V transformer rooms and 480V board room
- 690' elevation of auxiliary building by Unit 2 component cooling system pumps
- Emergency gas treatment room
- Emergency diesel generators

- 250V battery board rooms, No. 1 250V DC battery room, and associated corridor referencing SI-234.1, Functional Test of Operationally Required Fire Detectors in Panel 0-L, 603, 604, 605, 606, 607, and 627, Revision 29

b. Issues and Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

a. Inspection Scope

The inspectors observed operators in a simulator response to a steam generator tube rupture (SGTR) event. SGTR events are risk-significant as evaluated in the Individual Plant Examination. The inspectors observed the crew's (1) clarity and formality of communication, (2) ability to take timely action in the safe direction, (3) prioritization, interpretation, and verification of alarms, (4) correct use and implementation of procedures, including the alarm response procedures, (5) timely control board operation and manipulation, including high-risk operator actions, (6) oversight and direction provided by the shift manager, including ability to identify and implement appropriate TS actions such as reporting and emergency plan actions and notifications, and (7) the group dynamics involved in crew performance. The inspectors used Simulator Exercise Guide 2730023, "SGTR - TDAFW LCV Stuck Open on Ruptured SG," Revision 0, as a reference for this inspection.

b. Issues and 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 Procedure SPP-6.6, "Maintenance Rule Performance Indicator, Monitoring, Trending, and Reporting," Revision 4, NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision 2 and Instruction 0-TI-SXX-000-004.0, title same as SPP-6.6, Revision 9. Reviews focused on (1) MR scoping; (2) characterization of failed SSCs; (3) safety significance classifications; (4) 10CFR50.65 (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
Centrifugal charging pump 1B-B rotating element and room cooler	Cause Determination Evaluation Forms (CDEFs) 807 and 934
Component cooling system surge tank level instrumentation	Problem Evaluation Report (PER) 00-004723-000
Transformer room 1A temperature transmitter failure	PER 00-3461-000
2B-B 480V board room air handling unit	PER 00-006150-000
125V DC vital power	CDEF 1071, Predictive Monitoring Evaluation Report 00-005
480V DS breakers	PER 99-002075-000 and associated 10-Point Plan, CDEF 302, Licensee Event Report (LER) 50-327/98001; SQ980593PER; Work Order (WO) 98-06365-00, WR-C378991, SQ970086PER, PER99-001846, CDEF 857, CDEF 944

b. Issues and Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation

a. Inspection Scope

The inspectors evaluated, as appropriate for 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 Procedure SPP-7.1, Work Control Process, Revision 1, and Instruction 0-TI-DSM-000-007.1, Equipment to Plant Risk Matrix, Revision 1, during these inspection activities.

Work Activity	Related Document
Unit 1 auxiliary feedwater flow demand indication	WO 00-005986-000
250V DC ground at cooling tower lift pump station	PER 00-004660-000

Unit 1 No. 3 heater drain tank level indicating controller 1-LIC-6-106 air leak repair	WO 00-004465-000
2B-B 480V board room air handling unit (AHU) temperature controller repair	WO 00-004041-000
Unit 2 161 KV PCB 924 generator-to-bus 1 output breaker	N/A
Unit 1 LV-2 relay replacement	WO 00-006610-000
2B-B AFW pump maintenance	WR 00-003787-000

b. Issues and Findings

.1 2B-B 480V Board Room Air Handling Unit (AHU) Temperature Controller Failure

A non-cited violation (NCV) of TS 6.8.1.a was identified for failure to schedule a post-maintenance test (PMT) on the 2B-B 480V board room AHU, as required by procedure SPP-6.3, Pre-/Post-Maintenance Testing, Revision 0, when the PMT on the AHU could not be performed at the time of field work completion of a maintenance activity. The 480V board room cooling system is safety-related and risk-significant.

On May 4, 2000, the temperature controller for the 2B-B 480V board room AHU failed which resulted in the AHU being declared inoperable. The temperature controller was successfully calibrated on May 18, using WO 00-004041-000. On May 20, 2000, the temperature controller instrument air tubing, which had been previously identified as kinked and cracked, was replaced. The PMT for these maintenance activities was deferred, but the AHU was considered available for use. On June 10, when the system engineer questioned why the PMT had not been performed, the PMT was attempted but failed due to a ruptured diaphragm in the temperature controller. The temperature controller was replaced and the PMT was successfully completed on June 13, 2000.

The inspectors determined that during the time that the 2B-B 480V board room AHU was inoperable, May 20 to June 13, 2000, the function of the B 480V board room cooling system was maintained because the redundant AHU, 1B-B 480V board room AHU, was operable and in service. Either of the two trains, 1B-B and 2B-B, which comprise the B board room cooling system, was capable of meeting the cooling requirements for the B train 480V board rooms. The inoperability of the 2B-B 480V board room AHU had a credible impact on safety because it reduced defense-in-depth for 480V board room cooling. However, since there was no actual loss of safety function of the B train 480V board room cooling system, the finding was considered to have very low safety significance (Green).

Procedure SPP-6.3 requires, in part, that if a PMT cannot be performed at the time of field work completion then notification to the scheduling group to schedule the PMT is required. Contrary to the requirements of Procedure SPP-6.3, when the PMT on the

2B-B 480V board room AHU could not be performed at the time of field work completion, the licensee failed to schedule the PMT. This failure to adhere to Procedure SPP-6.3 was a violation of TS 6.8.1.a., which requires written procedures to be implemented covering the applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2, dated February 1978. Appendix A recommends procedures for performing maintenance, including general procedures for the control of maintenance. This 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 50-328/00-06-01: Failure to Schedule Post Maintenance Test on 2B-B 480V Board Room Air Handling Unit. The violation is in the licensee's corrective action program as PERs 00-006150-000 and 00-003461-000.

1R14 Personnel Performance During Non-routine Plant Evolutions

a. Inspection Scope

The inspectors reviewed human performance during the following non-routine plant events:

- Unit 1 loss of reactor coolant event and resulting inoperability of emergency core cooling system (ECCS) which occurred on September 24 and 25. Refer to Section 4OA3.2 for the event details.

- Unit 1 operators unable to perform boron dilution with primary water pump control switches in pull-to-lock discussed in PER 00-006128 dated July 24, 2000

The inspectors reviewed operator logs, plant computer data, procedures, and related training to determine what occurred and how operators responded for both events. In addition, the inspectors observed actual operator performance during the loss of reactor coolant event. The inspectors also evaluated the occurrences and subsequent personnel responses using the Significance Determination Process (SDP).

b. Issues and Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed selected technical operability evaluations (TOEs), 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; (5) where continued operability was considered unjustified, the impact on TS LCOs and the risk-significance in accordance with the SDP. The inspectors referenced procedure SPP-10.6, Engineering Evaluations for

Operability Determination, Revision 2, as needed during the course of these inspection activities.

- TOE 1-00-333-6011, Unit 1 Auxiliary Feedwater Flow Demand Indication
- TOE 0-00-032-4645, Containment Over-pressurization Following HELB which Causes Rupture of Control Air Piping
- TOE/PER 00-005707-000, ABSCE Breach - U2 RWST Vent Loop Seal Found Dry, and related document 0-TI-SXX-000-016.0, Breaching the Shield Building, ABSCE, or Control Room Boundaries, Revision 11

b. Issues and Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing

a. Inspection Scope

The inspectors reviewed Procedure SPP-6.3 which governs the licensee's PMT process, and also WOs 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.

Work Activity	Related Document(s)
Electric fire pump breaker replacement	WO 00-002606-000
1A-A shutdown board load shed relay LV-2 replacement	1-SI-EDC-082-307.A, Undervoltage/Degraded Voltage, DG Start and Load Shedding Time Response Relay Test, Revision 5
2-FSV-43-58A, loop 2 steam generator blow down sample valve accident indication light dual indication	WO 00-007078-000
2A centrifugal charging pump (CCP) outage	work schedule line items 9910167000 and 8005060000, PER 00-004238-000

2B-B motor-driven auxiliary feedwater pump	WR 00-003787-000

b. Issues and 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, listed below, to assess, as appropriate, whether the SSCs met TS, the updated final safety analysis report (UFSAR), and licensee procedure requirements, and to verify that the testing effectively demonstrated that the SSCs were operationally ready and capable of performing their intended safety functions.

- 2-SI-OPS-000-002.0, Shift Log, Revision 46 (ultimate heat sink temperature calculation)
- 0-SI-EBT-250-100.4, Vital Battery I Discharge Test, Revision 6
- 1-SI-SXP-070-201.B, Component Cooling Pump 1B-B Performance Test, Revision 3
- 2-SI-SXP-062-201A, 2A CCP Section XI Pump Test, Revision 6
- 0-SI-FPU-026-192.M, Motor-Driven Fire Pump A Mini-Flow Test, Revision 1
- 2-SI-SXP-003-201.B, Motor-Driven Auxiliary Feed Water Pump 2B-B Performance Test, Revision 6

b. Issues and Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed a temporary plant modification, Temporary Alteration Control Form 1-00-016-068, related to the installation of a time delay relay on the Unit 1 reactor coolant pump (RCP) No. 4 oil level annunciator. The inspectors reviewed the modification to ensure that risk-significant functions of the RCP were not affected.

b. Issues and 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, access controls and verified their implementation for at-power maintenance work for Unit 1 and Unit 2. The work was conducted in accordance with Radiation Work Permit (RWP) 516, Unit 1 Inside Polar Crane Wall, and RWP 823, Unit 2 Lower Containment. The following PERs were reviewed for assignment of responsibility, evaluation, and timely closure:

PER 00-003970-000	PER 00-003971-000
PER 00-003972-000	PER 00-003973-000
PER 00-004396-000	PER 00-004526-000
PER 00-006121-000	PER 00-006284-000
PER 00-006632-000	PER 00-007055-000
PER 99-011795-000	

Topics reviewed included administrative and engineering controls for high radiation, locked-high radiation, and very high radiation areas. Pre-job briefings, work-in-progress, and health physics technician job coverage were observed. Personnel dosimetry results and exposure investigation reports were reviewed and discussed in detail. Licensee activities were reviewed against UFSAR, Chapter 9, Radiation Protection, TS and 10 CFR Part 20 requirements.

Self-Assessment RP-00-003, Radiological Control Survey Instrumentation & Equipment, conducted May 1-12, 2000; Self Assessment SQN-RP-00-002, High Radiation Area Controls, dated February 18, 2000; and Self Assessment SQN-RP-00-001, Radiation Dose Control, dated December 6-17, 1999, were reviewed and the findings evaluated for significance and timely correction.

b. Issues and Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors evaluated the plant collective exposure history and current exposure dose trends against the annual site exposure goals for calendar year 2000, as outlined in the licensee's RWP Total Person-REM Person-Hours and Dose Rate Report, dated September 19, 2000. The inspectors also reviewed the following:

- Unit 1 Cycle10 ALARA Outage Report
- Unit 2 Cycle 10 ALARA Planning Report 2000-50, Refueling Operations
- Unit 2 Cycle 10 ALARA Planning Report 2000-52, Steam Generator Primary Side Eddy Current and Tube Plug Operations
- Unit 2 Cycle 10 Refueling Outage Schedule for Key ALARA Activities, Revision C
- Unit 2 Cycle 10 ALARA Planning Report 2000-58, Mechanical Valve Modifications
- Unit 2 Cycle 10 ALARA Planning Report 2000-65, Scaffolding Installation and Removal
- Unit 2 Cycle 10 ALARA Planning Report 2000-73, Refuel Outage Radiological Control Support

The inspectors attended an ALARA meeting concerning Unit 1 Cycle 10 ALARA Planning Report 2000-49, Add Oil to Unit 1 RCP No. 4 lower reservoir. In addition, the inspectors reviewed:

- ALARA committee meeting minutes of January 6, February 1, and March 17, 2000
- Chemistry General Operating Instruction 0-GO-7, Unit Shutdown From Hot Standby to Cold Shutdown, dated February 22, 2000
- Mechanical Design Standard DS-M18.7.1, Incorporating Radiation Protection Into Nuclear Plant Design, dated June 16, 1999
- Sequoyah Nuclear Plant Source Term Reduction Plan
- ALARA activities against TS and 10 CFR Part 20 requirements

b. <u>Issues and Findings</u>

No findings of significance were identified.

3. SAFEGUARDS

Cornerstone: Physical Protection

3PP2 <u>Access Control</u>

a. <u>Inspection Scope</u>

The inspector observed access control activities on September 5, 6 , and 7, 2000, and observed the licensee conduct equipment testing on September 7, 2000, to verify that the licensee was meeting the requirements of the Physical Security Plan and Contingency Plan (PSP/CP). In observing the access control activities the inspector assessed whether officers could detect contraband before it was introduced into the protected area. In addition, the inspector assessed whether the officers were conducting access control equipment testing according to regulatory requirements.

b. <u>Observations and Findings</u>

An apparent violation of the PSP/CP was identified with respect to access control. While reviewing the licensee event logs concerning access issues the inspector noted that in April 2000 the licensee had failed to follow the PSP/CP when granting access to an individual. Specifically, on April 19, 2000, the licensee violated a physical security instruction that supplements the PSP/CP by allowing a TVA employee to enter the site without removing his shoes after the first alarm was received on the access portal metal detector. The individual was allowed to entry after a physical search.

License Amendment No. 259, Paragraph E, dated August 4, 2000, states that the licensee shall fully implement and maintain in effect all provisions of the Commissioned approved physical security, guard training and qualification, and safeguards contingency plans.

Paragraph 3.1.3, of the PSP/CP, dated October 16, 1999, Revision 8, states that "the SQN Site Vice President is responsible for ensuring that security procedures, systems, and programs are developed, implemented, and enforced to meet the NRC approved PSP/CP, and Training and Qualification Plan."

Paragraph 5.3, of the PSP/CP requires that "Individuals entering the Protected Area shall be searched for firearms, explosives, and incendiary devices by use of equipment capable of detecting such objects or they shall undergo a hands-on pat down search." The search requirements as defined in the PSP/CP are supplemented by Physical Security Instruction - 32, dated March 31, 2000, which requires that, "Individuals processing through the metal detector shall stop inside the detector search area. If an alarm (first alarm) is received on the metal detector, the individual who caused the alarm shall be asked to ensure that all metal is removed (including shoes) and to process through the metal detector again. Should the individual alarm the detector again (second alarm), the MSF shall physically search the individual." On April 19, 2000, the licensee failed to require an individual who set off the metal detector during the first entry to remove his shoes and to process through the metal detector a second time. Pending further NRC review and formal disposition of this issue, this item is identified as Apparent Violation 50-327, 328/00-06-02: Failure to Search an Individual Prior to Granting Access to the Protected Area in Accordance with Access Control Procedures.

3PP3 Response to Contingency Events

a. Inspection Scope

The inspector randomly selected and screened security loggable events which related to response events during the period of January to August 2000 to determine if the licensee was identifying problems related to response to contingency events and were these problems being entered into the corrective action program.

b. Observations and Findings

An NCV of the Training and Qualification Plan (T&Q) was identified related to security force member physical weaknesses, affecting performance of assigned duties. While reviewing the licensee's documentation, the inspector noted that on April 29, 2000, an individual was assigned duties as an armed responder, even though the individual had physical weaknesses which could have adversely affected the performance of his assigned duties. The licensee had conducted a review of the event and had concluded that the individual should not have been assigned to this post due to physical weaknesses; however, the licensee believed that the individual in this stationary position could perform armed response duties. The inspector concluded during his review of the licensee's contingencies response requirements that the individual was posted at a location that during certain conditions would have required a response to other positions within seconds and that the individual posted in the armed responder position with the physical weaknesses would not have been capable of meeting their response requirements. This problem, if left uncorrected, would become a more significant concern and become a predictable weakness in the security of the site. However, given the non-predictable basis of this individual being posted with physical weaknesses and because that there was no evidence that the vulnerability had been exploited, this finding is considered to be of very low safety significance (Green).

Paragraph 1.2.1 of the T&Q, dated March 30, 1998, requires "Security Force personnel whose duty assignment position is directly associated with the effective implementation of the Physical Security/Contingency Plan (PSP/CP) shall have no weaknesses or abnormalities that would adversely affect their performance of assigned job duties." Failure to ensure that posted security force personnel have no weaknesses or abnormalities that would adversely affect their performance of assigned duties was a violation of the T&Q Plan. This violation is being treated as a non-cited violation (NCV), consistent with Section VI.A1 of the NRC Enforcement Policy, and is identified as NCV 50-327, 328/00-06-03: Failure to Ensure That Posted Armed Response Security Force Personnel Have No Weaknesses or Abnormalities That Would Adversely Affect Their Performance of Assigned Duties. This violation is in the licensee's corrective action program as PER 00-003775-000.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verifications

Licensee records were reviewed to confirm the accuracy and completeness of PI data in accordance with the guidance contained in NEI 99-02, Revision 0, "Regulatory Assessment Performance Indicator Guideline."

.1 Initiating Events Cornerstone

a. Inspection Scope

The inspectors reviewed monthly operating reports for the period July 1999 through June 2000 to verify the accuracy and completeness of the PI data for the indicators listed below. In addition, the inspectors reviewed the licensee's corrective action program to determine if any problems with the collection of PI data had occurred and if resolution was satisfactory.

- Unplanned Scrams per 7,000 Critical Hours
- Scrams with Loss of Normal Heat Removal
- Unplanned Power Changes per 7,000 Critical Hours

b. Findings

No findings of significance were identified.

.2 Mitigating Systems/Barrier Integrity Cornerstones

a. Inspection Scope

The inspectors verified data reporting elements for the following performance indicators for the last quarter of 1999 and the first and second quarters of 2000.

- Safety System Functional Failures
- Reactor Coolant System (RCS) Specific Activity

For the Safety System Functional Failures PI, the inspectors reviewed all LERs issued during the referenced time frame to determine if the licensee correctly identified the number of functional failures, including:

- LER 50-327/1999-003, Control Room Emergency Ventilation System Start as a Result of the Smell of Smoke in the Control Room

- LER 50-327/2000-001, Failure to Perform Response Time Testing on a Refueling Water Storage Tank Level Transmitter

- LER 50-327/2000-002, Loss of Pressurizer Level as a Result of a Relief Valve Failing to Reseat

- LER 50-327/2000-003, Reactor Trip Caused From a Detected Loss of Excitation Field to the Main Generator Because of a Design Error

- LER 50-328/2000-001, Reactor Trip Caused from a Low-Low Steam Generator Level Resulting From a Static Switch Control Board Circuit Failure

- LER 50-328/2000-002, Inoperability of Both Safety Injection Pumps as a Result of Personnel Error During Performance a Maintenance Activity

For the RCS Specific Activity PI, the inspectors also reviewed daily RCS grab sample results for Dose Equivalent Iodine (DEI)-131. These results were compared to the numbers reported for the PIs for the referenced time periods. The inspectors also observed a chemistry technician perform a routine RCS grab sample and analysis. The purpose of the observation was to verify that the sample and analysis were performed according to the guidance in Instruction 0-TI-CEM-000-016.3, Sampling Methods-Primary Systems, Revision 44. The analyzed value of DEI from such RCS samples provided the data for the PI involving RCS specific activity.

b. Findings

No findings of significance were identified.

4OA3 Event Follow-up

.1 Main Feedwater Flow Transient and Reactor Trip

On September 25, at 10:33 a.m., while operating at 100 percent reactor power, the Unit 1 reactor tripped following the automatic trip of the 1A main feedwater pump (MFP). Prior to the reactor trip, at 9:12 a.m., a low reservoir oil level alarm was received for the 1A MFP. At 10:31 a.m., a low oil pressure alarm was received and the 1A MFP automatically tripped on low oil pressure. As designed, main turbine load was automatically reduced to about 70 percent. Control rods automatically inserted to reduce reactor power, the auxiliary feedwater water system started and injected, and a signal was sent to the 1B MFP speed control system to accelerate to its 6000 rpm high speed stop. The 1B MFP, however, failed to automatically reach its maximum flow rate. Operators attempted, but were unable to manually increase the pump's speed above about 5300 rpm. As a consequence, the No. 2 SG reached its low-low water level trip set point due to inadequate feedwater flow causing an automatic reactor trip. The resident inspectors observed portions of the post-trip response, reviewed the licensee's post-trip report, and attended the plant operations review committee (PORC) post-trip review meeting.

The licensee initiated PER 00-008586-00 to determine the cause(s) and address appropriate corrective actions for the event. During the licensee's initial investigation, it was determined that maintenance procedures used to adjust the MFP speed control system were in error for both units. As a result, an improper adjustment of the 1B MFP speed control prevented it from reaching maximum flow. By design, the loss of a MFP from 100 percent power should not normally result in a reactor trip. Several issues involving this event require additional information for the NRC to determine whether the issues are acceptable. These issues are the circumstances related to the failure of the 1B MFP to reach rated flow, the automatic trip of the 1A MFP, and operator response to the event. Pending review of these issues, they are collectively identified as Unresolved Item (URI) 50-327, 328/00-06-04: Unit 1 Reactor Trip Due to Inadequate Main Feedwater Flow.

.2 Loss of Reactor Coolant Event While in Mode 4

On September 25 at 11:53 p.m., with Unit 1 in Mode 4 (hot shutdown) and 69 percent pressurizer water level, operators opened RHR suction valves 1-FCV-74-1 and -2, which aligned the RHR system to the RCS. During this evolution, operators observed a 2.5 percent pressurizer level step decrease. Nine minutes later at 12:02 a.m., with RCS pressure at approximately 365 psig, operators started the 1A RHR pump to cooldown the RCS to Mode 5 (cold shutdown). Upon starting the RHR pump, operators observed pressurizer level decrease rapidly and pressurizer relief tank (PRT) level similarly increase. Operators, suspecting that a relief valve had lifted, stopped the RHR pump at 12:04 a.m. During the two minutes the pump operated, pressurizer level decreased approximately 5.5 percent and PRT level increased approximately 1.4 percent to a level of 80 percent. At 12:13 a.m. operators, upon recognizing that stopping the RHR pump had not fully stopped the transfer of coolant from the RCS into the PRT (now at 81.5 percent), closed the RHR suction valves. This action stopped the loss of reactor coolant. The RHR system and its associated ECCS subsystem were declared inoperable by the licensee.

Steam dump valves in combination with the auxiliary feedwater system were used to remove decay heat while the licensee identified, isolated, and replaced the malfunctioning relief valve. The licensee completed immediate corrective actions, both trains of RHR were returned to service, and the unit was placed in Mode 5. The licensee also initiated PER 00-008645-000 to determine the cause(s) and address appropriate corrective actions for the event. During the licensee's initial investigation, it was determined that gas had accumulated in the RHR discharge piping resulting in a pressure pulse of sufficient magnitude, when combined with RCS operating and RHR startup pressures, to cause the relief valve to lift. Several issues involving this event require additional information for the NRC to determine whether the issues are acceptable. These issues are the issue extent of condition and a related issue addressed in TOE 00-063-8645. Pending review of these issues, they are collectively identified as URI 50-327, 328/00-06-05: RHR System Operating Procedures and ECCS System Gas Accumulation Contribute to Loss of Reactor Coolant and ECCS Inoperability.

.3 <u>(Closed) URI 50-328/00-03-01</u>: Risk-Significance of Inverter 2-IV Failure. On January 18, 2000, the licensee was in the process of installing a new inverter for the vital 120V AC 2-IV electrical bus. In this process while attempting to disconnect the old inverter, which was to be designated as a spare, the 2-IV vital bus momentarily lost power. As a result, Unit 2 tripped on low-low water level in steam generator 4. This event was documented in NRC Special Report 50-328/00-03. The inspectors reviewed the licensee's root cause evaluation and discussed the event with engineering personnel. The inspectors also reviewed the inverter modification work package and associated electrical drawings. The licensee's evaluation determined that the failure was caused by a voltage spike induced on the system when a vendor supplied neutral wire was removed while determinating a temporary synchronization wire. This voltage spike caused the static transfer switch to swap to the maintenance or alternate power source which was de-energized at that time. The licensee further determined that this condition represented a needed design enhancement, specific to Sequoyah, of the static switch transfer circuit supplied by the vendor. The inspectors discussed with the licensee whether this enhancement was a 10CFR Part 21 issue. The licensee stated that the vendor had evaluated the issue from a Part 21 perspective and had determined that Part 21 was not applicable. No findings of significance were identified.

4OA5 Other

.1 (Closed) Licensee Event Report (LER) Revision 50-327/2000-001-01: Failure to Perform Response Time Testing (RTT) on a Refueling Water Storage Tank (RWST) Level Transmitter. Revision 0 of this LER was closed in Inspection Report 50-327, 328/00-02. As part of this review, NCV 50-327/00-02-05: Failure to Perform Response Time Test for RWST Level Transmitter 1-LT-63-53, was identified. On March 10, 2000, during the licensee's corrective action program extent-of-condition review for Revision 0 of this LER, three additional components: Unit 1 emergency gas treatment system (EGTS), Train B suction valve 1-FCV-065-0030, and Units 1 and 2 turbine electro-hydraulic (EHC) auto-stop interface valves 1-FCV-47-27 and 2-FCV-47-27, were identified as not having been RTT following maintenance in accordance with TS 4.3.2.1.3. The licensee initiated Revision 1 to the LER on April 10, 2000. Following identification of the condition, the inspectors verified that the EGTS valve was RTT, using WO 00-002172-000, and was found to be acceptable. In addition, the inspectors verified, by reviewing licensee records, that acceptable tests had been performed on the two EHC interface valves, approximately 18 months after the missed RTTs, using Procedures 1-SI-IRT-099-601.A, Revision 6, and 2-SI-IRT-099-601.A, Revision 5, Response Time Test of ESFAS Slave Relays K601, K610 and K621 Actuated Devices (Feedwater Isolation and Turbine Trip) Train A.

This finding had a credible impact on safety, however, since each of the missed RTTs subsequently passed an RTT and there was no actual loss of safety function of any of the systems involved, the finding is considered of very low safety significance (Green).

Failure to perform the response time tests on valves 1-FCV-065-0030, 1-FCV-047-0027 and 2-FCV-047-0027 were violations of TS 4.3.2.1.3. These violations are being treated as additional examples of NCV 50-327/00-02-05, consistent with Section VI.A1 of the NRC Enforcement Policy. These violation examples are in the licensee's corrective action program as PER 00-000430-000. Revision 1 to the LER is closed.

.2 (Closed) Violation 50-327, 328/00-01-02: Failure to Include the Storm Drain System Within the Scope of the Maintenance Rule. The inspector verified the corrective actions described in the licensee's response letter, dated February 22, 2000, to be reasonable and complete. No similar problems were identified.

.3 (Closed) LER 50-328/2000-001-01: Reactor Trip Caused From a Low-Low Steam Generator Level Resulting From a Static Switch Control Board Circuit Failure. The inspectors reviewed the revised LER noting that due to additional root cause evaluations the licensee determined that the original suspected cause of the inverter failure and subsequent reactor trip had changed. This issue is discussed in Section 4OA3.3 of this report.

4OA6 Management Meetings

Exit Meeting Summary

The inspectors presented the inspection results to Mr. Dennis Koehl, Plant Manager, and other members of licensee management at the conclusion of the inspection on October 10, 2000. The licensee acknowledged the findings presented.

The inspectors asked the licensee whether any of the material examined during the inspection should be considered proprietary. No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

R. Purcell, Site Vice President
H. Butterworth, Operations Manager
T. Carson, Maintenance Manager
E. Freeman, Maintenance and Modifications Manager
J. Gates, Site Support Manager
C. Kent, Radcon/Chemistry Manager
D. Koehl, Plant Manager
M. Lorek, Assistant Plant Manager
D. Lundy, Site Engineering Manager
P. Salas, Manager of Licensing and Industry Affairs
K. Stephens, Acting Manager of Security
J. Valente, Engineering & Support Services Manager

NRC

R. Bernhard, Region II Senior Reactor Analyst
W. Rogers, Region II Senior Reactor Analyst

ITEMS OPENED AND CLOSED

Opened

50-327, 328/00-06-04	URI	Unit 1 Reactor Trip Due to Inadequate Main Feedwater Flow (Section 4OA3.1).
50-327, 328/00-06-05	URI	RHR Operating Procedures and ECCS System Gas Accumulation Contribute to Loss of Reactor Coolant and ECCS Inoperability (Section 4OA3.2).
50-327, 328/00-06-02	AV	Failure to Search an Individual Prior to Granting Access to the Protected Area in Accordance With Access Control Procedures (Section 3PP2).

Opened and Closed

50-328/00-06-01	NCV	Failure to Schedule Post-Maintenance Test on 2B-B 480V Board Room Air Handling Unit (Section 1R13.1).
50-327/00-02-05	NCV	Three Additional Examples of NCV 50-327/ 00-02-05 for Failure to Perform Response Time Testing (Section 4OA5.1).
50-327, 328/00-06-03	NCV	Failure to Ensure That Posted Armed Response Security Force Personnel Have No Weaknesses or Abnormalities That Would Adversely Affect Their Performance of Assigned Duties (Section 3PP3).

Closed

50-328/00-001-01	LER	Reactor Trip Caused From a Low-Low Steam Generator Level Resulting From a Static Switch Control Board Circuit Failure (Section 4OA5.3).
50-327/00-001-01	LER	Failure to Perform Response Time Testing (RTT) on a Refueling Water Storage Tank (RWST) Level Transmitter (Section 40A5.1).
50-327, 328/00-01-02	VIO	Failure to Include the Storm Drain System Within the Scope of the Maintenance Rule (Section 40A5.2).

50-328/00-03-01	URI	Risk Significance of Inverter 2-IV Failure (Section 4OA3.3).

NRC's REVISED REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) recently revamped its inspection, assessment, and enforcement programs for commercial nuclear power plants. The new process takes into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting and assessing safety performance at NRC licensed plants.

The new process monitors licensee performance in three broad areas (called strategic performance areas): reactor safety (avoiding accidents and reducing the consequences of accidents if they occur), radiation safety (protecting plant employees and the public during routine operations), and safeguards (protecting the plant against sabotage or other security threats). The process focuses on licensee performance within each of seven cornerstones of safety in the three areas:

Reactor Safety

- Initiating Events
- Mitigating Systems
- Barrier Integrity
- Emergency Preparedness

Radiation Safety

- Occupational
- Public

Safeguards

- Physical Protection

To monitor these seven cornerstones of safety, the NRC uses two processes that generate information about the safety significance of plant operations: inspections and performance indicators. Inspection findings will be evaluated according to their potential significance for safety, using the Significance Determination Process, and assigned colors of GREEN, WHITE, YELLOW or RED. GREEN findings are indicative of issues that, while they may not be desirable, represent very low safety significance. WHITE findings indicate issues that are of low to moderate safety significance. YELLOW findings are issues that are of substantial safety significance. RED findings represent issues that are of high safety significance with a significant reduction in safety margin.

Performance indicator data will be compared to established criteria for measuring licensee performance in terms of potential safety. Based on prescribed thresholds, the indicators will be classified by color representing varying levels of performance and incremental degradation in safety: GREEN, WHITE, YELLOW, and RED. GREEN indicators represent performance at a level requiring no additional NRC oversight beyond the baseline inspections. WHITE corresponds to performance that may result in increased NRC oversight. YELLOW represents performance that minimally reduces safety margin and requires even more NRC oversight. And RED indicates performance that represents a significant reduction in safety margin but still provides adequate protection to public health and safety.

The assessment process integrates performance indicators and inspection so the agency can reach objective conclusions regarding overall plant performance. The agency will use an Action Matrix to determine in a systematic, predictable manner which regulatory actions should be taken based on a licensee's performance. The NRC's actions in response to the significance (as represented by the color) of issues will be the same for performance indicators as for inspection findings. As a licensee's safety performance degrades, the NRC will take more and increasingly significant action, which can include shutting down a plant, as described in the Action Matrix.

More information can be found at: http://www.nrc.gov/NRR/OVERSIGHT/index.html.