UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
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January 29, 2001

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 50-327/00-07 AND 50-328/00-07

Dear Mr. Scalice:

On December 30, 2000, the NRC completed an inspection at your Sequoyah Units 1 & 2 reactor facilities. The enclosed report presents the results of that inspection which were discussed on January 9, 2000, 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.

No findings of significance were identified.

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/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
cc w/encl: (See page 2)

TVA

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U.S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos:	50-327, 50-328
License Nos:	DPR-77, DPR-79
Report Nos:	50-327/00-07, 50-328/00-07
Licensee:	Tennessee Valley Authority (TVA)
Facility:	Sequoyah Nuclear Plant, Units 1 & 2
Location:	Sequoyah Access Road Soddy-Daisy, TN 37379
Dates:	October 1, 2000 - December 30, 2000
Inspectors:	R. Gibbs, Senior Resident Inspector D. Starkey, Resident Inspector R. Telson, Resident Inspector E. Testa, Senior Health Physicist P. VanDoorn, Senior Reactor Inspector R. Chou, Reactor Inspector D. Thompson, Physical Security Inspector J. Blake, Senior Project Manager W. Sartor, Senior Emergency Preparedness Inspector J. Kreh, Emergency Preparedness Inspector
Approved by:	P. Fredrickson, Chief Reactor Projects Branch 6 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000327-00-07, IR 05000328-00-07, on October 1, 2000 - December 30, 2000, Tennessee Valley Authority, Sequoyah, Units 1 & 2. Event follow-up.

The inspection was conducted by resident inspectors, a senior health physicist, a senior reactor inspector, a physical security inspector, a senior project manager, a senior emergency preparedness inspector, a reactor inspector and an emergency preparedness inspector.

A. Inspector Identified Findings

No finding of significance were identified.

B. Licensee Identified Violations

Violations of very low significance which were identified by the licensee have been reviewed by the inspectors. Corrective actions taken or planned by the licensee appear reasonable. These violations are listed in Section 4OA7.

Report Details

Summary of Plant Status: Unit 1 started the inspection period in cold shutdown to repair a reactor coolant pump (RCP) No. 4 high vibration problem. The Unit was restarted on October 5 and had reached about 48 percent power on October 6 when the Unit had to be removed from service to repair continued high vibration of RCP No. 4. The Unit was restarted on November 13 and reached 100 percent power on November 16. Unit 1 operated at or near 100 percent for the remainder of the inspection period.

Unit 2 started the inspection period at 90 percent power in coast-down for a scheduled refueling outage. The Unit was shutdown on October 22. The Unit was restarted on November 14 and automatically tripped on November 17 from about 53 percent power due to a main transformer bushing failure. The Unit was restarted on November 18 and reached 100 percent power on November 20. Unit 2 operated at or near 100 percent power for the remainder of the inspection period.

1. **REACTOR SAFETY**

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

1R01 Adverse Weather Preparations

.1 Freeze Protection Readiness

a. Inspection Scope

The inspectors verified that the licensee had taken actions for freezing weather conditions to limit the risk of freeze related initiating events and adequately protect mitigating systems from the effects of freezing weather. The inspectors walked down selected components including those associated with the refueling water storage tank level instruments, condensate storage tank level instruments, and main feedwater flow sensing lines. The inspectors reviewed the licensee’s freeze protection procedures and selected work orders (WOs) and problem evaluation reports (PERs) to evaluate the condition of selected freeze protection components. The following documents were reviewed during the inspection.

- Periodic Instruction (PI) 0-PI-000.006.0, Freeze Protection, Rev. 26
- PI 1-PI-EFT-234-706.0, Freeze Protection Heat Trace Functional Test, Rev. 16
- PI 2-1-PI-EFT-234-706.0, Freeze Protection Heat Trace Functional Test, Rev. 2
- PI 0-PI-MIN-000-706.0, Freeze Protection Insulation Inspection, Rev. 4
- Self-Assessment SQN-M&M-00-001
- PERs 99-010455 and 99-010468

b. Findings

No findings of significance were identified.

.2 Tornado Watch Response

a. Inspection Scope

The inspectors observed the licensee respond to a tornado watch on November 9. The inspectors also reviewed licensee Procedure AOP-N.02, Tornado Watch / Warning, Rev. 7, for its effectiveness to limit the risk of tornado-related initiating events and to adequately protect mitigating systems from the effects of a tornado.

b. Findings

No findings of significance were identified.

1R02 Evaluation of Changes, Tests, or Experiments

a. Inspection Scope

The inspectors reviewed nine samples of safety evaluations to confirm that the licensee had appropriately reviewed and documented changes in accordance with 10 CFR 50.59 and licensee procedures and had considered the conditions under which changes to the facility or procedures may be made without prior NRC approval. The inspectors also reviewed 15 samples of design changes (DCNs) and procedure changes for which the licensee had determined that evaluations were not required to confirm that the licensee's conclusions to "screen out" these changes were correct and consistent with 10 CFR 50.59. The inspectors reviewed additional information as necessary such as applicable sections of the Updated Final Safety Analysis Report (USAR), supporting analyses, Technical Specifications (TS), and procedures.

In addition, the inspectors reviewed a licensee self-assessment of the 10 CFR 50.59 program and reviewed three PERs to confirm that the licensee was identifying issues, entering these into the corrective action program, and resolving the concerns.

The following documents and procedures were reviewed to evaluate the licensee's 10 CFR 50.59 program for processing evaluations of changes, test and experiments.

Safety Evaluations

- DCN D-20292-A, Replacement of the Carbon Steel Bypass Valves 2-VLV-024-0548 and 0556 and Associated Piping with Stainless Steel, Rev.0
- DCN D-20348, Installation of Pressure Reducing Orifices in the Centrifugal Charging Pump Cold Leg and Safety Injection Hot and Cold Leg Injection Branch Lines, Rev. 0
- DCN D-20472, Installation of Breakdown Orifice at Downstream of Emergency Core Cooling System Throttle Valves, Rev. 0

- DCN —13953-B, Replacement of Containment Spray Heat Exchanger 1B and Flow Restricting Orifices, Rev. 2
- DCN S-13744-B, Revise Switch Setting for Auxiliary Feedwater Trip and Throttle Valve, Rev. 1
- Work Request C412052, Repair of Unit 1 Loop 1 Reactor Cooling System Pressurizer Spray Valve 1-PC-68-340D, Rev. 1
- Technical Requirements Manual, Rev. 10, Addition of Section 3/4.7.13 for Control Room Air Temperature Control System, Rev. 0
- TS Bases 3/4.4.2 and.3, Change to Pressurizer Power Operated Relief Valve Operability Criteria, Rev. 0
- TS Bases 4.0.3, Removal of Reporting Requirements, Rev. 0

Safety Screening Documents

- DCN T-14381, Replacement of Unit 2 Motor Driven Auxiliary Feedwater Pump 2B-B, Rev. 0
- DCN D-20241, Replacement of Unit 1 Motor Driven Auxiliary Feedwater Pump 1B-B, Rev. 0
- DCN D-20242, Addition of a Pressure Reducing Orifice in the Common Bearing Cooling Water Line for Turbine Driven Auxiliary Feedwater Pump 1A-S, Rev. 0
- DCN D-20280-A, Replacement of the Unit 2 Turbine Driven Auxiliary Feedwater Pump Steam Supply Piping High Pressure Steam Trap with Orifices and Valves, Rev. 0
- DCN D-20372, Replacement of the Unit 2 Turbine Driven Auxiliary Feedwater Pump 2A-S, Rev. 0
- DCN D-20120-A, Modification of the F delta I Parameters Associated with the Overpower and Overtemperature Delta T Reactor Trip Setpoints, Rev. 0
- DCN D-20129, Replacement of the Existing Flexible Hoses for the Unit 2 Centrifugal Charging, Safety Injection, and Containment Spray Pump Drains, Rev. 0
- Temporary Alteration 1-TACF-016-068, Add Time Delay to Unit 1 Reactor Coolant Pump No. 4 Oil Level Switch Alarm, Rev. 0
- 0-GO-13, Reactor Coolant System Drain and Fill Operations, Rev. 29
- 0-SO-74-1, Residual Heat Removal System, Rev. 27
- 2-SO-63-5, Emergency Core Cooling System, Rev. 26
- 0-SO-61-1, Ice Condenser Cooling, Rev. 9
- 1-SI-OPS-000-003.D, Daily Shift Log, Rev. 30
- 0-SI-OPS-083-151.A, Six Month Test Requirement on Electric Hydrogen Recombiner System Train A, Rev. 5
- 2-SI-OPS-088-006.0, Containment Building Ventilation Isolation (18 Month/100 Hours/7 Days), Rev. 13

Other Documents

- SPP-9.5, Temporary Alterations, Rev. 3
- SPP-9.4, 10CFR50.59 Evaluations of Changes, Tests, and Experiments, Rev. 2
- SPP-9.4, 10CFR50.59 Evaluations of Changes, Tests, and Experiments, Rev. 3
- Self-Assessment CRP-ENG-00-029, Technical Adequacy of 50.59's and Calculations, dated 4/17/00-8/4/00

- PER 99-012003, Failure to Identify UFSAR Change Requirement for Battery Load Shedding Time
- PER 99-011594, Procedure Non-compliance for Screening Reviews
- PER 00-005484, Inadequate Documentation for Screening Reviews

b. Findings

No findings of significance were identified.

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 operating procedures 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.

- Emergency diesel generator 1A-A (EDG) starting air system
- Unit 2 main generator output breaker-PCB 928
- Centrifugal charging pump 2A-A (CCP)

b. 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, Rev. 0, and prefire plans for the areas listed below, as appropriate.

- Cable spread area
- EDG fuel oil room
- EDG switchgear and ventilation rooms
- Unit 2 temporary radcon control point building
- Auxiliary building 734' elevation refueling floor
- Main turbine lube oil storage areas
- EDG building

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures

a. Inspection Scope

The inspectors reviewed the selected risk important internal and external flood protection barriers listed below. The inspectors walked down selected areas which contain risk important equipment and are located below design flood levels to evaluate the adequacy of those attributes protecting risk important equipment. The inspectors also reviewed licensee instructions for shutdown in the event of severe flooding and evaluated the availability of selected structures, systems, and components (SSCs), listed below, for safe shutdown under design worst case assumed water levels.

- EDGs
- Essential raw cooling water (ERCW)
- 1E power and signal cabling to and from ERCW and EDG systems
- Flood mode makeup pumps
- Auxiliary charging system

The inspectors also reviewed the following documents which specifically addressed the acceptability of issues related to monitoring and cable splicing of 6.9KV cables for the EDG and ERCW systems:

- SQNP, Cable Splices in Underground Ductbanks, dated July 26, 2000
- SQNP, 600 Volt Connectors in Medium Voltage Cable Splices, dated August 25,2000.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation

a. Inspection Scope

The inspectors sampled portions of 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, Rev. 4, NUMARC 93-01, Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants, Rev. 2 and Instruction 0-TI-SXX-000-004.0, title same as SPP-6.6, Rev. 10. Reviews focused on (1) MR scoping; (2) characterization of failed SSCs; (3) safety significance classifications; (4) 10 CFR 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 Performance Problem Inspected	Additional Documents Reviewed
Unit 1 main generator loss of field detection resulting in unit trip	Cause Determination Evaluation Form (CDEF) 1063
Excessive unavailability of 1A-A 480 volt board room chiller	PER 00-008019-000 PER 00-008083-000
Failure of RHR 2B-B mini-flow valve to open automatically	PER 00-010636-000, CDEF 1168
Functional failure of EDG 2A-A	PER 00-010238-000, CDEF 1177
Functional failure of EDG 1B-B	PER 00-010357-000, CDEF 1181
Functional failure of 1B-B 480 volt board room chiller	PER 00-006946-000, CDEF 1135, CDEF 1142

b. 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 such 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, Rev. 1, and Instruction 0-TI-DSM-000-007.1, Equipment to Plant Risk Matrix, Rev. 1, during these inspection activities.

Selected Maintenance Activity	Additional Documents Reviewed
Condensate booster pump 2A outage	None
Centrifigal charging pump 2A-A mechanical seal cooler low flow	PER 00-010124-000; WO 00-009904-000
Vital battery III high equalize voltage	PER 00-008936-000; 0-PI-EBM-000-001.1, Battery Equalize Charge, Rev. 4; 0-SO-250-1, 125 Volt DC Vital Power System, Rev. 24
Main feedwater pump 2A-A thrust bearing wear detector	WO 00-010499-000, 10CFR50.59 Screening Review

1N43 power range nuclear instrument bistable failure	PER 00-011068-000; AOP-I.01, Nuclear Instrument Malfunction, Rev. 4; TS 3/4.3.1, Reactor Trip System Instrumentation
Unit 2 main generator loss-of-cooling trip disablement and repair to sensing line hydraulic snubbers	PER 00-010840-000

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Non-routine Plant Evolutions and Events

a. Inspection Scope

The inspectors reviewed human performance during the following non-routine plant evolutions and events. Specifically, the inspectors reviewed operator logs, plant computer data, procedures, and related training to determine what occurred and how operators responded. The inspectors also evaluated the occurrences and subsequent personnel responses using the Significant Determination Process (SDP).

- Unit 1 automatic reactor trip on September 25, 2000, discussed in NRC Inspection Report (IR) 50-327, 328/00-06
- Unit 1 restart activities on October 5
- Unit 1 manual reactor trip on October 6 discussed in Section 1R20.1 of this report
- Unit 2 manual reactor trip on October 22 for scheduled refueling outage
- Unit 2 automatic reactor trip on November 17 discussed in Section 4OA3 of this report
- Unit 2 restart activities on November 18

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed selected technical operability evaluations (TOEs) 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 LCOs and the risk-significance in accordance with the SDP. The inspectors referenced Procedure SPP-10.6, Engineering Evaluations for Operability Determination, Rev. 2, as needed, during the course of these inspection activities.

Operability Evaluation Inspected	Additional Documents Reviewed
Unit 1 auxiliary feedwater recirculation piping vibration	PER 00-0008963-000
Potential failure to close on demand for DS 206 circuit breakers (shock-out)	Cutler-Hammer Part 21 letter dated August 30, 2000; TOE 0-00-201-7879; PER 00-007879-000; PER 00-002194-000; PER 00-00-3517-000; MI-10.5, Westinghouse Type DS Breaker and Switchgear Maintenance, Rev. 50.
Manufacturer’s pump performance data incorrect for three ERCW pumps and fire/flood mode pump A-A	TOE 0-00-067-8611, PER 00-008611-000
Unit 1 pressurizer TS 3.4.9.2 cooldown and heatup rate limits exceeded.	TOE 1-00-068-8691; PER 00-008691-000; PER 00-009812-000; PER 00-008956-000; TS 3.4.9.2, Reactor Coolant System Pressurizer Limiting Condition for Operation (and basis); 0-GO-7, Unit Shutdown From Hot Standby to Cold Shutdown, Revs. 20, 21, 23; September 26 Unit 1 Control Room Narrative Log.
Numerous Unit 2 ice condenser ice basket as-found weights outside of analytical limits	PER 00-009666-000

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications

a. Inspection Scope

The inspectors evaluated a modification to the ERCW cooling system, a risk-significant system, in which piping connecting ERCW to the abandoned No. 5 EDG was removed and capped. This modification was selected because of its potential for common cause failure of ERCW to the remaining EDGs which are risk-significant. The objectives of the inspection were to verify that the design bases, licensing bases, and performance capability of risk significant SSCs had not been degraded through the modification and to verify that the modifications did not place the plant in an unsafe condition. The inspectors observed work in the field, discussed the modification with the engineering field supervisor, reviewed TS 3.8.1.2, the licensee's risk assessment, PER 00-009815-000, WOs 99-007800-000 and 99-007800-001 and Procedures 0-AR-M26-A (C-4), Diesel Generator 1A-A Jacket Water Temperature High-Low Engine 1 or 2 Annunciator Response Procedure, Rev. 20; AOP-P.01, Loss of Offsite Power Abnormal Operating Procedure (AOP), Rev. 8

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing (PMT)

a. Inspection Scope

The inspectors reviewed Procedure SPP-6.3, Pre/Post Maintenance Testing, Rev 0 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.

Post Maintenance Test Inspected	Additional Documents Reviewed
Unit 2 stator cooling differential pressure switch sensing line repair	WO 00-010825-000; 0-SO-35-2, Stator Cooling System, Rev. 27
Unit 2 loop 2 hot leg injection check valve integrity test following valve replacement	Surveillance Instruction (SI) 2-SI-SXV-063-204.0, Safety Injection/Residual Heat Removal Hot Leg Primary Check Valve Integrity Test, Rev. 4

Unit 2 turbine driven auxiliary feedwater pump (TDAFW) overspeed trip test	Maintenance Instruction 2-MI-MFT-003-002.0, Auxiliary Feedwater Pump 2A-S Overspeed Trip Tests, Appendix F, Rev. 7; PER 00-010116-000
Unit 2 stroke testing of main steam isolation valves (MSIV's) following refueling outage	PI 0-PI-SXV-001-001.0, Stroke Testing of MSIV's at Operating Temperature, Rev. 3; PER 00-009479-000
Unit 1 safety injection relief valve testing prior to and following September 25, 2000, shutdown loss-of-coolant event.	SI 0-SI-SXV-000-264.0, Testing Setpoint of Safety and Relief Valves (ASME Section XI, Category C Valves), Rev. 4 (As-installed and as-found testing of valve 1-63-627 which stuck open in).
Ice condenser lower plenum floor monitoring	PI 0-PI-SXX-061-001.0, Ice Condenser Lower Plenum Floor Monitoring Periodic Instruction, Rev. 5.

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities

.1 Unit 1 Forced Outage

a. Inspection Scope

The inspectors observed numerous activities associated with a Unit 1 forced outage which began on October 6 due to a high vibration condition for RCP No. 4. These activities are listed below:

Clearance Activities - Checked that tags were properly hung and equipment appropriately configured to support the function of the clearance, including a walkdown and review of tagout 1-TO-2000-0001, Section 1-63-0862-F/O

Reactor Coolant System Instrumentation - Checked that RCS pressure, level, and temperature instrumentation were installed and configured to provide accurate indication and that instrument error was accounted for:

- PI 1-PI-IXX-068-005.0, Installation and Removal of the Mansell Level Monitoring System, Rev. 1
- Engineering Assistance Request 2000-NSS-068-1348, Benchmark Elevation for the Mansell Level Monitoring System Channel “A” and Channel “B”; Unit One Forced Outage Hot Midloop for RCP No. 4 Impeller Replacement
- WO 00-009569, Troubleshooting & Correct Excessive Level Deviation between Unit 1 Mansell Level Detectors

- PER 00-010468-000, Recurring Erratic and/or Inaccurate Unit 1 and Unit 2 Mansell Indications RCS Vacuum Refill Conditions
- PER 00-010243-000, Mansell Channel Deviation Excessive During RCS Draindown, Transducer Found Out of Calibration.

Electrical Power - Checked that 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.

- Procedure OPDP-2, Switchyard Access and Switching Order Execution, Appendix C, Switchyard Access While on RHR and Defense-In–Depth Measures are in Force, Rev. 0

- Procedure O&SSDM 4.0 (Outage and Site Scheduling Directive Manual), Operational Defense-in-Depth Assessment, Rev. 6

- Procedure SPP-7.2, Outage Management, Appendix C, Outage Risk Assessment Reviews, Rev. 3

Decay Heat Removal (DHR) System Monitoring - Observed DHR parameters to assess proper system function and that the SGs, when relied upon, were a viable means of backup DHR.

Inventory Control - Reviewed flow paths, configurations, and alternative means for inventory addition for consistency with the outage risk plan. Reviewed activities with the potential to cause loss of inventory for adequacy of controls to prevent inventory loss. Observed drain-down to mid-loop conditions, using Procedure 0-GO-13, Reactor Coolant System Drain and Fill Operations, Rev. 29

Reactivity Control - Evaluated licensee control of reactivity for compliance with TS. Also evaluated activities or SSCs for potential to cause unexpected reactivity changes for inclusion and proper control under the outage risk plan.

Containment Closure - Ensured that containment closure could be achieved during selected configurations.

- Procedure 0-GO-15, Containment Closure Control, Rev. 11
- Forced Outage Checklist for Mode 5 to Mode 4-Unit 1

Reduced Inventory and Mid-Loop Conditions - 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. Also evaluated the effect of distractions on operator ability to maintain required reactor vessel level during mid-loop operations. Observed drain-down to mid-loop conditions on October 25, using Procedure 0-GO-13.

Monitoring of Heatup and Startup Activities - Reviewed on a sampling basis that TS and administrative procedure prerequisites for mode changes were met prior to changing modes or plant configurations.

b. Findings

No findings of significance were identified.

.2 Unit 2 Cycle 10 Refueling Outage

a. Inspection Scope

The inspectors observed numerous activities associated with the Unit 2 Cycle 10 refueling outage that began on October 22. These activities are listed below.

Clearance Activities - Checked that tags were properly hung and equipment appropriately configured to support the function of the clearance.

Reactor Coolant System Instrumentation - Checked that RCS pressure, level, and temperature instrumentation were installed and configured to provide accurate indication and that instrument error was accounted for, and reviewed PER 00-010468-000.

Electrical Power - Checked that 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, using Procedures OPDP-2, O&SSDM 4.0 and SPP-7.2

Decay Heat Removal (DHR) System Monitoring - Observed DHR parameters to assess proper system function and that the SGs, when relied upon, were a viable means of backup DHR.

Spent Fuel Pool Cooling System Operation - 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. Also assessed the licensee's procedure for mitigation of a loss of spent fuel pool cooling, using Abnormal Operating Procedure AOP-M.06, Loss of Spent Fuel Cooling, Rev. 1

Inventory Control - Reviewed flow paths, configurations, and alternative means for inventory addition for consistency with the outage risk plan. Reviewed activities with the potential to cause loss of inventory for adequacy of controls to prevent inventory loss.

Reactivity Control - Evaluated licensee control of reactivity for compliance with TS. Also evaluated activities or SSCs for potential to cause unexpected reactivity changes for inclusion and proper control under the outage risk plan.

Containment Closure - Reviewed control of containment penetrations for compliance with refueling operations TS to ensure that containment closure could be achieved during selected configurations, using Procedure 0-GO-15, and Refueling Outage Checklist for Mode 5 to Mode 4-Unit 2

Reduced Inventory and Mid-Loop Conditions - 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. Also evaluated the effect of distractions on operator ability to maintain required reactor vessel level during mid-loop operations. Observed drain-down to mid-loop conditions on November 9, and reviewed Procedure 0-GO-13.

Refueling Activities - Reviewed fuel handling operations for conformance with TS and approved procedures. Confirmed that the location of selected fuel assemblies were tracked during core reload.

- 0-GO-9, Refueling Procedure, Rev. 16
- 0-RT-NUC-000-002.0, Core Reconfiguration, Rev. 16

Monitoring of Heatup and Startup Activities - Reviewed on a sampling basis that TS and administrative procedure prerequisites for mode changes were met prior to changing modes or plant configurations, including a review of SI 0-SI-SXX-068-127.0, RCS and Pressurizer Temperature and Pressure Limits, Rev. 3

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, listed below, to assess, as appropriate, whether the SSCs met TS, the 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.

Surveillance Test Inspected	Related SIs Reviewed
Phase A Isolation Test	2-SI-OPS-088-001.0, Phase A Isolation Test, Rev. 3
Unit 2 main steam safety valves	0-SI-SXV-001-859.0, Testing and Setting of Main Steam Safety Valves, Rev. 6
Loss of offsite power with safety injection	2-SI-OPS-082-026.A, Loss of Offsite Power with Safety Injection-D/G 2A-A Test, Rev. 22

Actuation of automatic valves via SI signal	0-SI-OPS-000-009.0, Actuation of Automatic Valves Via SI Signal for Non-Testable Boric Acid and ECCS Flow Path Valves, Rev. 16
Centrifugal charging pump 2A-A performance test	2-SI-SXP-062-201.A, Centrifugal Charging Pump 2A-A Performance Test, Rev. 6
RHR discharge piping vent	2-SI-OPS-074-128.0, Unit 2 RHR Discharge Piping Vent, Rev. 4

b. 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 RCP No. 4 oil level annunciator. The inspectors reviewed the modification to ensure that risk-significant functions of the RCP were not affected.

b. Findings

No findings of significance were identified.

1EP1 Exercise Evaluation

a. Inspection Scope

The inspectors reviewed the objectives and scenario for the biennial, full-participation emergency preparedness exercise to determine whether the exercise was designed to suitably test major elements of the licensee’s emergency plan. The criteria against which the exercise scenario, the licensee’s performance, and the licensee’s critique were evaluated are contained in 10 CFR 50 Appendix E.

The inspectors observed and evaluated the licensee’s performance in the exercise, as well as selected activities related to the licensee’s conduct and self-assessment of the exercise. The exercise was conducted on October 4 from 8:00 a.m. to 2:00 p.m. Licensee activities inspected during the exercise included those occurring in the control room simulator, technical support center (TSC), operational support center, and central emergency control center. The inspectors’ evaluation focused on the risk-significant activities of event classification, notification of governmental authorities, onsite protective actions, offsite protective action recommendations (PARs), and accident mitigation. The inspectors also evaluated command and control, the transfer of emergency responsibilities between facilities, communications, adherence to procedures, and the overall implementation of the emergency plan. The inspectors attended the post-exercise critique to evaluate the licensee's self-assessment process, as well as the presentation of critique results to plant management.

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation

a. Inspection Scope

The inspectors observed a licensee emergency response team (Orange) perform the quarterly emergency plan drill on December 5. The inspectors evaluated drill conduct and the adequacy of the licensee’s critique of performance to identify weaknesses and deficiencies. The inspectors reviewed the drill scenario and plan, and observed drill performance in the TSC. The inspectors also attended the TSC post-drill critique.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstones: Occupational Radiation and Public Radiation Safety

2OS1 Access Control to Radiologically Significant Areas

a. Inspection Scope

The inspectors reviewed radiological surveys, access controls and verified their implementation for on-line power and shutdown maintenance work for Unit 1 and Unit 2. The work was conducted in accordance with radiation work permits. The review included administrative and engineering controls for high radiation, locked-high radiation, and very high radiation areas. Pre-job briefings, observations of work-in-progress, and Health Physics (HP) technician job coverage were observed. Personnel dosimetry results, personnel contamination event reports and exposure investigation reports were reviewed and discussed in detail. Radiation worker training and whole-body count data analysis were reviewed. Communication protocol, worker direction and remote radiation dose teledosimetry were observed during SG work. Safety injection valve replacement work was observed. Work planning and dose control for workers were observed during movement of the RCP No. 4 impeller, in a shielded storage box from the Unit 1 containment to the decontamination pit for temporary storage. Licensee activities were reviewed against the UFSAR, TS, and 10 CFR Part 20 requirements.

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 collective exposure history, current exposure dose trends, and the year 2000 annual site dose goal to verify that the licensee was implementing ALARA practices as required by 10 CFR 20.1101(b) and Procedure RCI-10, ALARA Planning Report Criteria, Rev. 27. The inspectors also reviewed dose controls for pregnant females, source term reduction efforts, the incorporation of ALARA into licensee radiation work permits, and ALARA job evaluations for five dose significant outage jobs. The review included: ALARA planning, dose goals and estimates, daily dose results and job dose trends and problem identification. The chemistry crud burst and cleanup results from Procedure O-GO-7, Unit Shutdown From Hot Standby to Cold Shutdown, Rev. 23 were reviewed.

The following ALARA Planning Reports (APR) were reviewed for job scope adequacy, accuracy of dose estimates, and the adequacy of licensee dose tracking:

- 2000-50 Refueling Operations
- 2000-52 S/G Primary Maintenance
- 2000-58 Valve Modifications
- 2000-59 In service Inspections
- 2000-65 Scaffolding
- 2000-67 Temporary Shielding
- 2000-70 RCP Platform Mods
- 2000-74 ECCS Pressure Reducing Orifice

b. Issues and Findings

No findings of significance were identified.

2PS3 Radiological Environmental Monitoring Program

a. Inspection Scope

The inspectors reviewed radiological procedures, calibration data files, interviewed health physics technicians, health physics shift supervisors, and health physics section supervisors and managers to evaluate compliance with the Radioactive Material Control Program, UFSAR, TS, and 10 CFR Part 20 requirements. In addition the inspectors accompanied and observed a technician performing operational checks on tool monitors, personnel contamination monitors and a portal monitor.

Procedures reviewed included the following:

- SPP-5.1,TVA Standard Programs and Processes Radiological Controls, Rev. 3
- RCDP-1,TVA Standard Department Procedure Conduct of Radiological Controls, Rev. 1
- RCI-1, Radiological Control Instruction Radiological Control Program, Rev. 59
- RCI-17, Control of By-product and Source Material, Rev. 13
- RCI-21, Control of Radioactive Material and Storage Areas, Rev.10
- RCI-22, Contamination Control, Rev. 7A
- RMD FO-2, Radcon Management Directive Field Operations Radiation and Contamination Surveys, Rev. 10
- RMD FO-10, Personnel Contamination Reports, Rev. 9
- RCDP-8, Radiological Instrumentation/Equipment Controls, Rev. 1
- RCI-5, Radiological Control Instrumentation Program, Rev. 32
- RCI-5, Attachment 19 Calibration of the Eberline Personnel Monitor, Rev. 32
- RCI-5, Attachment 3 Operation of Laboratory Counters/Scalers, Rev. 32
- RCI-5, Attachment 5 Operation of Tennelect LB5100 Counting Systems, Rev. 32
- RCI-5, Attachment 10 Response Check for Portable Friskers, Rev. 32
- RCI-5, Attachment 15 Calibration and Operational Checks of the Eberline Bag Waste Monitor, Rev. 32
- RCI-5, Attachment 16 Calibration and Operational Checks of the Eberline Tool Contamination Monitor, Rev. 32
- RCI-5, Attachment 17 Calibration of the Eberline Personnel Contamination Monitor, Rev. 32
- RCI-5, Attachment 18 Operational Checks of the PCM-1B Personnel Contamination Monitor, Rev. 32
- RCI-5, Attachment 20 Operational Checks for Portal Monitors, Radcon Management Directive Instrumentation, Rev. 32
- RMD I-01, Radiological Control Instrument Inventory and Response Criteria, Rev. 6

The inspectors reviewed environmental sample collection procedures, air sampler calibration data, accompanied an environmental sample collection laboratory technician during the collection of air samples. Selected thermal translucent dosimetry TLD locations and air sample locations were verified. The inspectors observed the performance of the meteorological tower weekly operational checklist and reviewed files containing calibration and maintenance data for the meteorological system. The Annual Radiological Environmental Operating Report Sequoyah Nuclear Plant 1999 and the Offsite Dose Calculation Manual Rev. 45 were reviewed to evaluate compliance with the Radiological Environmental Monitoring Program, Offsite Dose Calculation Manual, Technical Specification, and 10 CFR Part 50 requirements.

Procedures reviewed included the following:

- SC-01, Collection of Environmental Monitoring Samples, Rev. R12
- SC-02, Preventive Maintenance for Radiological Environmental Monitoring Air Sampling System, Rev. R1
- SC-03, Calibration Procedure for Radiological Environmental Monitoring Air Sampler System Gas Meter, Rev. R2

- Self Assessment Report CRP-RP-00-002 dated 7/28/00, Conduct of Radiological Environmental Monitoring Program and 2000 Environmental Cross Check Analysis data.

b. Issues and Findings

No findings of significance were identified.

3. SAFEGUARDS

Cornerstone: Physical Protection

3PP1 Access Authorization

a. Inspection Scope

The inspectors reviewed licensee procedures, Fitness For Duty (FFD) reports, and licensee audits. Additionally, the inspectors interviewed five representatives of licensee management and five escort personnel concerning their understanding of the behavior observation portion of the personnel screening and FFD program. In interviewing these personnel, the inspectors reviewed the effectiveness of their training and abilities to recognize aberrant behavioral traits.

b. Issues and Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification

Licensee records were reviewed to determine whether the submitted PI statistics were calculated in accordance with the guidance contained in NEI 99-02, Revision 0, Regulatory Assessment Performance Indicator Guideline.

Cornerstone: Emergency Preparedness

.1 Emergency Response Organization (ERO) Drill/Exercise Performance

a. Inspection Scope

The inspectors performed a review of PI data over the past eight quarters through review of a sample of drill records, to verify the accuracy and completeness of the ERO Drill and Exercise Performance PI. Detailed documentation was reviewed for drills conducted in November - December 1999 and May 2000. In addition the reactor coolant system leak that occurred on September 26, 2000, and briefly met the Emergency Action Level for a Notification of Unusual Event was reviewed. These activities were evaluated to verify the licensee’s reported data regarding successes in emergency classifications, notifications, and PARs.

b. Findings

No findings of significance were identified.

.2 ERO Drill Participation

a. Inspection Scope

The inspectors performed a review of PI data over the previous eight quarters by review of the training records for the 52 personnel assigned to key positions in the ERO to verify the accuracy and completeness of the ERO Drill Participation PI. Drill participation was verified by reviewing training attendance records for approximately 10 percent of key ERO personnel against the drill/event participation matrix for specific drill dates.

b. Findings

No findings of significance were identified.

.3 Alert and Notification System Reliability

a. Inspection Scope

The inspectors performed a review of PI data over the previous 12 months to verify the accuracy and completeness of the Alert and Notification System Reliability PI. The inspectors reviewed the licensee's records of the siren tests, which comprised biweekly silent tests, annual growl tests, and monthly full-cycle tests.

b. Issues and Findings

No findings of significance were identified.

Cornerstones: Occupational and Public Radiation Safety

.4 Occupational Exposure Control Effectiveness

a. Inspection Scope

The inspectors reviewed chemistry and the radiation protection issues in the licensee's corrective action program to determine whether any should have been reported as PI occurrences during the period April through December 2000, to verify the accuracy and completeness of the Occupational Exposure Control Effectiveness and the RETS/ODCM Radiological Effluent Occurrences PIs.

b. Findings

No findings of significance were identified.

Cornerstone: Physical Protection

.5 Fitness-for-Duty, Personnel Screening, and Protected Area Security Equipment

a. Inspection Scope

The inspector reviewed licensee programs for gathering and submitting data to verify the accuracy and completeness for the Fitness-for-Duty, Personnel Screening, and Protected Area Security Equipment PI. The review included TVA's tracking and trending reports and security event reports for the PI data submitted from the first quarter 2000 to the fourth quarter of 2000.

4OA3 Event Follow-up

.1 (Closed) Licensee Event Report (LER) 50-327/2000-03-00: Reactor Trip Caused from a Detected Loss of Excitation Field to the Main Generator Because of a Design Error. On March 21, 2000, an automatic Unit 1 turbine trip and reactor trip were caused by an erroneous "main generator loss-of-excitation field" protective signal. Errors in a design change specifications package caused the protective circuitry to be incorrectly wired and tested. The event was complicated by secondary plant challenges. The event is discussed in Section 4OA3.2 of IR 50-327, 328/00-02, in which a finding of very low significance was identified, related to the main generator design change error. A review of the secondary challenges is discussed in Section 40A3.2 of this report. The licensee's root cause determination in PER 00-002540-000 identified that "The barriers of design change checking, design verifications and reviews were not completed to the proper level." The inspectors 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.

.2 (Closed) Unresolved Item URI 50-327/00-02-03: Challenges to Secondary Plant Systems After Unit 1 Reactor Trip. On March 21, 2000, an automatic Unit 1 turbine trip and reactor trip were caused by an erroneous main generator protective signal. During the automatic reactor trip, the main feedwater system experienced excessive pipe movement, water hammer and vibration. Main feedwater pump (MFP) 1B was damaged when: (1) it failed to stop rotating due to high pressure (HP) steam stop valve leakage, (2) it lost suction due to the inadvertent isolation of all intermediate pressure (IP) feedwater heaters, and (3) it's recirculation valve opened to the main condenser depressurizing hot feedwater in the pump allowing it to flash to steam. Although these individual conditions are common to both units, and had resulted in previous plant challenges, this was the first observed challenge of greater than minor safety significance. This URI was opened to further review secondary plant equipment performance issues observed during the event to determine their acceptability and safety significance.

The inspectors reviewed: (1) the licensee's completed post-trip review package, (2) PER 00-002540-000 which addressed trip-related deficiencies and corrective actions, (3) PER 00-002548-000 which addressed the 1B MFP damage and corrective actions, (4) PER 00-010688-000 which addressed the observation of leaking high pressure stop valves on both the 2A and 2B MFP during a November 17 Unit 2 trip, (5) PER 00-010690 which addressed the unintended isolation of the B and C low pressure heater strings during the November 17 Unit 2 trip, and (6) operator work-arounds (OWAs) 99007WA and 99008WA, which were reinstated on March 23, 2000, following the event, to address the undesired automatic isolation of intermediate pressure feedwater heaters following reactor trips. The review revealed that these secondary plant equipment performance issues were the result of ineffective: (1) maintenance and modification efforts to address MFP HP steam stop valve leakage, (2) modifications to address original design problems with MFP recirculation valves, and (3) modifications to address original design problems with feedwater heaters. The main feedwater system challenges had a very low risk significance due to the available remaining mitigating systems and the probable restoration of the 1A MFP. The inspectors determined that the planned corrective actions, if effectively implemented, would address the above challenges.

.3 <u>(Closed) LER 50-327/2000-02-00</u>: Loss of Pressurizer Level as a Result of a Relief Valve Failing to Reseat. On March 13, 2000, with Unit 1 in cold shutdown, operators used a deficient procedure to vent the RHR discharge piping. A damaging pressure pulse (water hammer) occurred causing a safety injection system relief valve to actuate and stick open. A loss of reactor coolant inventory resulted. This event is discussed in section 40A3.4 and in NRC IR 50-327, 328/00-02. A similar event occurred on September 25, 2000, and was discussed in NRC IR 50-327, 328/00-06. URI 50-327, 328/00-06-05, RHR Operating Procedures and ECCS System Gas Accumulation Contributes to Loss of Reactor Coolant and ECCS Inoperability, was opened to further evaluate the extent of condition and related issues addressed in TOE 00-063-8645. No new issues of significance were revealed by the LER review. However, an NCV was identified related to this event-LER and is discussed in Section 40A7.

.4 <u>(Closed) URI 50-327, 328/00-02-02</u>: Failure to Identify and Correct Deficiencies in Procedures to Vent RHR System Discharge Piping Contributing to a Shutdown Loss of Coolant Inventory Event. On March 13, 2000, with Unit 1 in cold shutdown, operators used a deficient procedure, Procedure 1-SI-OPS-074-128.0 Unit 1 RHR Discharge Piping Vent, to vent the RHR discharge piping. A damaging pressure pulse (water hammer) occurred causing safety injection system relief valve 62-626 to actuate and stick open. A loss of reactor coolant inventory resulted. This unresolved item (URI) was opened pending completion of a safety assessment of this corrective action finding.

The inspectors determined that the inadequate procedure contributed to the damaging system pressures which increased: (1) the frequency for challenging the relief valve, (2) the likelihood that the relief valve would fail open, and (3) the risk to the valves needed to isolate the failed relief valve. The inspectors further determined that operators had ample time to terminate the leak before loss of RHR cooling, and reviewed other available methods of mitigating the event had RHR cooling been unrecoverable. The inspectors determined that sufficient time, indication, procedural guidance, reserve coolant inventory, and a robust mitigation capability were available during use of the procedure. It was further determined that the shutdown mitigation capability was not affected by use of the procedure. Thus, the resulting plant impact from the inadequate procedure was determined to be of very low safety significance.

In October 1998, following an unanticipated loss of pressurizer level, similar to that which occurred on March 13, 2000, the licensee initiated PER SQ981446PER. This PER, which questioned the adequacy of RHR system vent and fill procedures, was closed on June 2, 1999. A procedure revision to Procedure 1-SI-OPS-074-128.0, made as part of corrective action to the PER, was inadequate in that it failed to correct procedural deficiencies which subjected RHR and SI system piping to the potentially damaging pressure surges and contributed to the loss of reactor coolant inventory event on March 13. Because the resulting plant impact from the inadequate procedure was determined to be of very low safety significance, the inspectors determined that the corrective action finding was also of very low safety significance. This finding involved a violation of 10 CFR 50, Appendix B, Criterion XVI, Corrective Action. The corrective action enforcement aspects of this finding are discussed in Section 4OA7.

.5 (Closed) URI 50-327, 328/00-02-04: Risk Significance and Regulatory Impact of Issues Related to TI-142 Inspection. During an inspection of Temporary Inspection (TI) 2515/142, Draindown During Shutdown and Common-Mode Failure, the inspectors identified two potential enhancements to the licensee's response to Generic Letter 98-02 for the potential risks associated with reactor coolant draindown events during plant shutdown conditions. These were discussed in IR 50-327, 328/00-02, Section 4OA4. These enhancements were associated with operator training and administrative controls to reduce the impacts of a draindown event during hot, pressurized RCS conditions. As a result of these identified enhancements, the licensee initiated PER 00-001404-000 to address appropriate corrective actions. The inspectors reviewed these corrective actions and determined that the licensee had taken reasonable actions to lower the risks of draindown events during hot and pressurized conditions. In particular, the inspectors noted that a recent revision to AOP-R.02, Shutdown LOCA, Rev. 4, was performed which addressed the inspectors previous concerns. No findings of significance were identified and the issue did not constitute a violation of NRC requirements.

.6 Unit 2 Trip Due to Faulted Main Transformer

On November 17, an automatic Unit 2 reactor trip occurred as a result of a turbine trip which was actuated by all three sudden pressure relays for the "C" phase main transformer. The unit was in a power ascension from a refueling outage and was at about 53 percent power when the reactor trip occurred. The inspectors observed portions of trip response in the main control room, walked down the secondary plant for unusual conditions, and observed operation of the AFW system. The inspectors also discussed the cause of the trip with licensee management. The licensee determined that an actual fault condition occurred in the "C" transformer as a result of the failure of a 24KV bushing which had been replaced during the refueling outage. The Unit was subsequently restarted on November 18 after a spare main transformer was placed in service. The inspectors determined that the trip response was uncomplicated and sufficient mitigating systems were available during the trip recovery. Operator response to the event was evaluated in Section 1R14. No findings of significance were identified related to the event.

4OA5 Other

.1 Review of Institute of Nuclear Power Operations (INPO) Evaluation Report

On November 6, the inspectors reviewed the results of an INPO evaluation of licensee performance conducted during July 2000. The report was dated September 8, 2000. The report did not identify any significant licensee performance issues that had not been previously addressed and/or reviewed by the NRC.

.2 Inspection of Unit 2 SG Eddy Current Examinations

A special regional initiative inspection was conducted in response to NRC questions about industry's response to the Indian Point 2 SG tube rupture event, and recent dialog between TVA and NRR questioning the extent of inspection of low-voltage, dented intersections in the Sequoyah Unit 1 SGs during the last outage. The inspection involved observations of examination activities and discussions with licensee and contractor personnel concerning the following documents:

- SG Integrity Guidelines - Sequoyah Nuclear Plant - SG Group, Rev. 7, October 2000.
- Sequoyah Nuclear Plant - Units 1 & 2 SG Eddy Current Examination Guideline, Rev. 2, October 2000.
- Sequoyah Nuclear Plant Unit 2 Cycle 10 Degradation Assessment, October 2000.
- Sequoyah Nuclear Plant Unit 2 Cycle 10 SG Tubing Examination Scan Plan, Rev. 0, July 2000.
- Industry Lessons Learned Implemented at Sequoyah Unit 2 SG Inspections, November 2000.
- Traveler No. SQN-006, In Situ Pressure Test Using the Computerized Data Acquisition System, Rev. 1, October 17, 2000.

As described in the industry lessons learned document, the licensee had implemented a process for conducting quality assurance checks of all of the eddy current data prior to analysis by primary and secondary analysts. The quality assurance (QA) checks verified that the probe and probe speed were correct, correct calibration standard was used, and also looked for evidence of probe wear and excess noise.

After the initial QA checks of the data, primary and secondary data analysts employed by different contractors, located in different cities analyzed the SG tube data for flaw indications. Subsequent to the primary and secondary data analyses, there were also data QA checks conducted. Resolution analysts at the site also reviewed for questionable data as well as resolving discrepancies between primary and secondary analyst's results. Independent quality data analysts (QDAs) reviewed the final determinations by the resolution analysts as well as conducting sampling reviews of data which did not go to the resolution analysts because the primary and secondary analysts were in agreement. A senior analyst reviewed all U-bend indications, as well as all flaws, permeability, and geometry indications. The senior analyst also conducted historical reviews to document and assess the significance of the indication changes/growth.

The inspector observed activities of the data QA analysts, the resolution analysts, the QDAs and the senior analyst. As a part of interviewing resolution analysts and licensee representatives, the inspector selected several tubes at random and also some with known degradation, and reviewed the data with the resolution and QDA analysts. Reviews of historical data were also conducted to compare the current data with the last time that the tube was analyzed. The inspector also selected a sample of tubes in which the primary and secondary analysts were not in agreement to review the differences. The methodology by which feedback was provided to primary and secondary analysts whose results had been overruled, as well as the process by which the licensee tracked productivity and accuracy of each of the individual analysts, was reviewed.

During the inspection, eddy current indications were found in second and third row U-bends. The inspector reviewed how the licensee had factored the identification problems at another utility's U-bend indications into the training of the Sequoyah analysts, and also into the analyst's guidelines. The licensee also pointed out that an indication that had been considered to be a geometric indication at another utility had leaked during an in-situ pressure test, and the signature for that type of indication had also been factored into the analyst guidelines for Sequoyah as a pluggable indication.

The inspector witnessed the in-situ pressure testing of a U-bend indication that exceeded the licensee's in-situ screening criteria. During witnessing of the test, the inspector noted that the licensee had factored lessons learned from pressure tests described in NRC Regulatory Issues Summary 2000-22 dated November 3, 2000, in that the pressure test restricted the speed of pressure increase, and specified intermediate hold times between the main steam line break and the three times normal operating pressure delta pressure (3xΔP) calculated pressures.

b. Findings

No findings of significance were identified.

4OA6 Management Meetings

The inspectors presented the inspection results to Mr. Richard Purcell, Site Vice President, and other members of licensee management at the conclusion of the inspection on January 9, 2001. 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.

40A7 Licensee Identified Violations

The following findings of very low significance were identified by the licensee and are violations of NRC requirements which meet the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600 for being dispositioned as Non-Cited Violations.

NCV Tracking Number	Requirement Licensee Failed to Meet
NCV 50-327, 328/00-07-01	10 CFR 50, Appendix B, Criterion XVI, Corrective Action, requires that conditions adverse to quality be promptly identified and corrected. An inadequacy of Procedure 1-SI-OPS-074-128.0, Unit 1 RHR Discharge Piping Vent, a condition adverse to quality, was not identified and corrected by a procedure revision made in response to an October 1998 RCS event. This procedural inadequacy was revealed during a similar event on March 13, 2000. This issue was placed in the licensee's corrective action program as PER 00-002224-000.
NCV 50-327, 328/00-07-02	10 CFR 73.21(d) requires unattended safeguards information to be stored in a locked security storage container. TVAN Standard Programs and Processes, SPP -1.4, Appendix E, requires Safeguards Information when not in use to be stored in an approved locked storage container. On October 18, 2000, during a licensee quality assurance audit the auditors concluded that information that should have been classified as safeguards information was left unattended and not stored in an approved safeguards container. This issue was placed in the licensee's corrective action program as PER 00-009322-000.

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, Security Manager
J. Valente, Engineering & Support Services Manager

NRC

R. Bernhard, Region II Senior Reactor Analyst
W. Rogers, Region II Senior Reactor Analyst
M. Pohida, NRR Reliability and Risk Analyst

ITEMS OPENED AND CLOSED

Opened and Closed

50-327, 328/00-07-01	NCV	Failure to Identify and Correct Deficiencies in a Procedure to Vent RHR System Discharge Piping Contributing to a Shutdown Loss of Coolant Inventory Event (Section 4OA7).
50/327, 328/00-07-02	NCV	Failure to Store Unattended Safeguards Information in a Locked Security Storage Container (Section 4OA7).

Closed

50-327/2000-03	LER	Reactor Trip Caused from a Detected Loss of Excitation Field to the Main Generator Because of a Design Error (Section 4OA3.1).
50-327/00-02-03	URI	Challenges to Secondary Plant Systems After Unit 1 Reactor Trip (Section 4OA3.2).
50-327/2000-02-00	LER	Loss of Pressurizer Level as a Result of a Relief Valve Failing to Reseat (Section 4OA3.3).
50-327, 328/00-02-02	URI	Failure to Identify and Correct Deficiencies in Procedures to Vent RHR System Discharge Piping Contributing to a Shutdown Loss of Coolant Inventory Event (Section 4OA3.4).
50-327, 328/00-02-04	URI	Risk Significance and Regulatory Impact of Issues Related to TI-142 Inspection (Section 4OA3.5).