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

May 2, 2003

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 POWER PLANT - NRC INTEGRATED INSPECTION REPORT 50-327/03-03 AND 50-328/03-03

Dear Mr. Scalice:

On April 5, 2003, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at the Sequoyah Nuclear Power Plant, Units 1 and 2. The enclosed report presents the results of the integrated inspection which were discussed on April 9, 2003, with Mr. Rick Purcell and other members of your staff.

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

Based on the results of this inspection, the inspectors identified one issue of very low safety significance (Green). This issue was determined to involve a violation of NRC requirements. However, because of its very low safety significance and because it has been entered into your corrective action program, the NRC is treating this issue as a non-cited violation, in accordance with Section VI.A.1 of the NRC's Enforcement Policy. If you contest this 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) components of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/ADAMS.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Stephen J. Cahill, 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 50-327/03-03, 50-328/03-03
w/Attachment: Supplemental Information

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U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos:	50-327, 50-328
License Nos:	DPR-77, DPR-79
Report No:	50-327/03-03, 50-328/03-03
Licensee:	Tennessee Valley Authority (TVA)
Facility:	Sequoyah Nuclear Plant, Units 1 & 2
Location:	Sequoyah Access Road Soddy-Daisy, TN 37379
Dates:	January 5, 2003 - April 5, 2003
Inspectors:	S. Freeman, Senior Resident Inspector R. Telson, Resident Inspector R. Carrion, Senior Project Engineer (Section 1R06) S. Vias, Senior Reactor Inspector (Section 40A5)
Approved by:	S. Cahill, Chief Reactor Projects Branch 6 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000327-03-03, IR 05000328-03-03, Tennessee Valley Authority, 1/5/2003 - 4/5/2003
Sequoyah Nuclear Power Plant, Units 1 & 2, Operability Evaluations.

The report covered a three-month period of inspection by resident inspectors and an announced inspection by two regional based inspectors. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using IMC 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG 1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. Inspector-Identified and Self-Revealing Findings

Cornerstone: Barrier Integrity

- Green. Inadequate technical guidance was identified because the associated procedure did not contain the necessary steps to ensure that multiple breaches of the shield building would be adequately controlled.

  This inspector-identified finding was determined to be a non-cited violation of (NCV) Technical Specification 6.8.1.a. It was more than minor, because if left uncorrected it could result in the actual shield building breached area exceeding the margin of operability for the emergency gas treatment system. The finding also affected the configuration control attribute of the containment barrier. The finding is of very low safety significance because the actual margin was not exceeded. It was also considered to constitute a deficiency in the cross-cutting element of Problem Identification and Resolution (Section 1R15 and 40A2).

B. Licensee Identified Violations

None

REPORT DETAILS

Summary of Plant Status

Unit 1 operated at or near 100 percent rated thermal power until March 17, 2003 when it was shut down for a scheduled refueling and steam generator replacement outage.

Unit 2 began the inspection period shutdown for repair of the Number 3 reactor coolant pump motor. The motor was repaired and the unit returned to 100 percent power on January 8, 2003. The unit operated at or near 100 percent rated thermal power until March 10, 2003, when it tripped automatically due to problems with a heater drain tank level control valve and a hotwell pump. The problems were repaired and the Unit returned to 100 percent power on March 18, 2003. Unit 2 was manually shutdown on March 24, 2003, due to a hydrogen leak on the main generator. The leak was repaired on March 30, 2003 and the unit was returned to 100 percent power.

1. REACTOR SAFETY

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

1R01 Adverse Weather Protection

a. Inspection Scope

The inspectors observed the licensee responsed to a tornado watch on March 19, 2003. The inspectors reviewed licensee Procedure AOP-N.02, Tornado Watch/Warning, Revision 11, for its effectiveness to limit the risk of tornado-related initiating events and to adequately protect mitigating systems from the effects of a tornado. In addition, the inspectors verified the securing of large outside cranes in accordance with guidance in Topical Report 24370-TR-C-002, Rigging and Heavy Load Handling.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

a. Inspection Scope

The inspectors conducted partial walkdowns of the following three systems to verify the availability of redundant or diverse systems and components and that defense-in-depth was maintained during periods when safety equipment was inoperable. The inspectors reviewed applicable operating procedures, walked down critical system components, and reviewed identified problems to ensure they were entered into the corrective action program. Documents reviewed are listed in the attachment.

- Alternate Emergency Diesel Generator (EDG) during unavailability of EDG 1A-A

- B-train of auxiliary feed water during unavailability of Motor Driven Auxiliary Feed Water Pump 1A-A

- A-Train electric board room chiller during unavailability of B-train electric board room chiller

b. Findings

No findings of significance were identified.

1R05 Fire Protection

a. Inspection Scope

The inspectors conducted a tour of eight areas to assess the material condition, operational status, and lineup of fire protection systems, equipment, and features. The inspectors assessed control of transient combustibles and ignition sources, and verified fire protection equipment was available for use. Documents reviewed are listed in the attachment. The areas toured are listed below.

- Essential Raw Cooling Water Building
- Auxiliary building Elev. 669 (2A-S turbine driven auxiliary feed water pump room)
- Emergency Diesel Generator Building
- Auxiliary building Elev. 653 (1A-A residual heat removal pump room)
- Auxiliary building Elev. 653 (1B-A residual heat removal pump room)
- Turbine Building Elev. 662
- Unit 1 Annulus
- Auxiliary building Elev. 714 (temporary containment access pathway)

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures

a. Inspection Scope

The inspectors reviewed selected risk-important plant design features and licensee procedures intended to protect the plant and its safety-related equipment from internal and external flooding events. The inspectors reviewed flood analysis and design documents including UFSAR Sections 2.3 and 2.4, including Appendix 2.4A, Flood Protection Plan, and Design Criteria Document SQN-DC-V-12.1, Sequoyah Nuclear Plant - Flood Protection Provisions, for licensee commitments. The inspectors also reviewed licensee instructions for cross-tying systems in the event of severe flooding and evaluated the availability of a selected Unit 1 spool piece identified in the instructions and on Drawing 1,2-47W845-2, Flow Diagram - Essential Raw Cooling Water System. The inspectors reviewed selected risk-important external flood protection barriers to evaluate the inadequacy at protecting risk-important equipment. The inspectors performed a walkdown of risk-significant areas, susceptible systems, and equipment to verify that the respective floor drain system, including room sump pumps, was operable, including:

- Essential Raw Cooling Water (ERCW) pump house elevations 704' and 720'
- Emergency Diesel Generators
- 161-kV cable tunnel

The inspectors reviewed the following plant procedures for coping with flooding events to verify that the actions were consistent with the plant's design basis assumptions:

- AOP-N.03, Revision 16, Flooding
- AOP-N.04, Revision 5, Break of Downstream Dam

The inspectors also reviewed the licensee's corrective action documents with respect to flood-related items identified in Problem Evaluation Reports (PERs) written in 2002 to verify the adequacy of the corrective actions:

- PER 02-003277-000, AOP-N.04, Break of Downstream Dam, was revised to add steps to makeup to the forebay using ERCW to maintain the required level.

- PER 02-005674-000, AOP-N.03, Flooding, was revised to re-align the sluice gates not previously included in the procedure to satisfy the requirements of Design Criteria SQN-DC-V-12.1, Flood Protection Provisions.

The inspectors also reviewed completed preventive maintenance procedures for monthly checks for standing water in manholes/handholes for September and December 2002 and related PER 03-000418-000.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

a. Inspection Scope

The inspectors observed simulator training on February 19, 2003. The scenario involved a leak on the Residual Heat Removal (RHR) system during mid-loop operations. The leak was within the capability of the charging system. This placed the simulated unit in the abnormal operating procedure for RHR malfunctions.

The inspectors observed crew performance to ensure the following criteria were satisfied: appropriate communications; ability to take timely and proper actions; prioritizing, interpreting, and verifying alarms; correct use and implementation of procedures, including the alarm response procedures; timely control board operation and manipulation, including high-risk operator actions; oversight and direction provided by the shift manager, including the ability to identify and implement appropriate Technical Specification (TS) actions; and group dynamics involved in crew performance.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation

a. Inspection Scope

The inspectors reviewed six activities to verify that the appropriate risk assessments were performed prior to removing equipment for work. When emergent work was performed, the inspectors verified that the risk for the work was assessed and required equipment was protected. The inspectors referenced Procedure SPP-7.1, Work Control Process, Revision 4, and Instruction, 0-TI-DSM-000-007.1, Equipment to Plant Risk Matrix, Revision 7, during these inspection activities.

- Removal of the 1A-A Emergency Diesel Generator from service for maintenance
- Removal of the Unit 2 Turbine Driven Auxiliary Feed Water Pump from service for check valve testing
- Loss of Unit 2 Refueling Water Storage Tank (RWST) Instrumentation
- Unavailability of Auxiliary Feed Water level control valve 1-LCV-3-156 following in-service testing
- Operational Defense-in-Depth Assessment for Week of March 21, 2003
- Removal of ERCW header 1A for pipe replacement

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Non-Routine Plant Evolutions

a. Inspection Scope

The inspectors reviewed operating crew performance and plant indications associated with the automatic reactor trip and loss of secondary heat sink that occurred on March 10, 2003, while Unit 2 was operating at 100 percent power. The review evaluated what occurred and how operators responded to the event. The inspectors reviewed plant operating logs, plant computer information, associated PERs, and conducted discussions with operations and engineering personnel. The inspectors also reviewed plant procedures, to determine whether the operator's response was in accordance with those procedures.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed seven selected functional evaluations and related documents to verify that the licensee had adequately assessed TS operability. In addition, the inspectors reviewed applicable documents to verify that the system or component was capable of performing required functions and any required compensatory actions were properly implemented. The inspectors reviewed the functional evaluations against the requirements of licensee Procedure SPP-10.6, Engineering Evaluations for Operability Determination. Additional documents reviewed are listed in the attachment.

- PER 03-000564-000, Rag found under EDG 1A2 engine lube oil strainer
- PER 03-000290-000, Inability to reach Mode 5 in 30-hour limit required by TS
- PER 03-000830-000, Damage to EDG 1B-B battery due to momentary short during testing
- PER 03-000835-000, EDG 1B-B Stator Winding Polarization Index below IEEE Standard 43
- Breach Permit VBP-2002-0071, Core Drill 4 inch diameter scaffold holes in Unit 1 Shield Building
- PER 03-001354-000, EDG 2A1 Pillow Block Bearing Alignment
- PER 03-003216-000, High temperature conductor feeding electric board room chiller B-B

b. Findings

Introduction: In reviewing the operability evaluation and associated aspects of a four-inch diameter hole drilled in the dome of the Unit 1 Shield Building for the steam generator replacement outage, the inspectors identified a green NCV for failure to provide complete instructions governing shield building breaches.

Description: The licensee opened the four-inch diameter shield building breach to allow workers to pass scaffold material through the dome rather than carry it up the ladder in the annulus. The licensee further indicated that the breach area had been analyzed and was less than the area that would affect the operability of the Emergency Gas Treatment System (EGTS) and annulus vacuum systems. The inspectors reviewed the breach permit for the hole; V.P.-2002-0071; the associated operability evaluation; and Calculation 65NQL041296, Determine Maximum Breach Area Between Auxiliary Building Secondary Containment Envelope, Outside, and The Shield Building. The inspectors noted that the area of the shield building breach was 12.57 square inches and that the calculation allowed 14 square inches before affecting the operability of the EGTS System, a margin of less than two square inches.

On February 15, 2003, the licensee initiated Problem Evaluation Report (PER) 03-001577-000, which identified that two other shield building breaches had not been documented on Vent Boundary Tracking Sheets. Because this indicated the possibility of three simultaneous breaches of the shield building, the inspectors reviewed the tracking sheets, Procedure, 0-TI-SXX-000-016.0, Breaching The Shield Building, Auxiliary Building Secondary Containment Envelope (ABSCE), or Control Room Boundaries, Revision 15, and interviewed personnel associated with installation of the breaches. The inspectors determined that the work documents which installed the breaches adequately controlled the work such that the amount of shield building area open at any one time was less than 14 square inches. However, the inspectors subsequently identified that this control was only successful in this case because all three breaches were controlled by the same organization, (i.e. the Steam Generator Replacement Project).

Procedure 0-TI-SXX-000-016.0 controlled breaches via appendices, one for each area, with instructions for anyone using the procedure to go directly to the applicable appendix. The inspectors determined that, even though a general precaution specified a vent boundary log, there were no specific instructions in the appendix for shield building breaches (Appendix D) that required breaches to be logged or required that the breached area be added to the log when a permit was approved. Without these instructions the inspectors concluded that specific breaches could be missed and not logged as happened with the two breaches in PER 03-001577-000.

Analysis: The inspectors determined that this finding affected only the radiological barrier function of the containment. If left uncorrected it could result in the actual breached area exceeding the margin of operablility for the EGTS system, a more significant safety concern. The finding also affected the configuration control attribute of the containment barrier. This makes it more than minor. However, because the actual margin was not exceeded, there was no degradation to the radiological barrier function of the containment. Therefore, the inspectors considered the finding to be of very low safety significance (Green).

The inspectors also determined that there was a Problem Identification and Resolution (PI&R) aspect to this finding. PER 03-001577-000 identified that two shield building breaches had not been tracked. However, the PER did not identify the lack of specific instructions nor consider that the undocumented breaches had the potential to allow enough of the shield building to be breached to exceed the margin of operability for the EGTS system. The inspectors therefore considered this finding to indicate a potential problem identification deficiency and have noted it in Section 4OA2.

Enforcement: TS 6.8.1.a, requires that activities affecting quality be prescribed and accomplished using instructions, procedures, or drawings In Accordance With (IAW) Regulatory Guide 1.33, Revision 2, February 1978. RG 1.33 requires procedures for performing maintenance that can affect the performance of safety-related equipment. Contrary to this, Procedure 0-TI-SXX-000-016.0, which was designated quality related and controlled work on the safety-related containment barrier, did not contain the instruction steps necessary to ensure that multiple breaches of the shield building would be adequately controlled. This is a violation of TS 6.8.1.a. Because it is of very low safety significance and has been entered into the licensee's corrective action program as PER 03-003612-000, this violation is being treated as an NCV, consistent with Section VI.A of the NRC enforcement policy: NCV 50-327/03-03-01, Inadequate Instructions for Controlling Shield Building Breaches.

1R16 Operator Workarounds

a. Inspection Scope

The inspectors reviewed operator actions to manually start and load the security diesel generator while its auto start feature was out of service to determine whether the functional capability of the diesel was affected. The inspectors specifically considered whether the workaround affected the operators' ability to implement abnormal or emergency operating procedures. Documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing

a. Inspection Scope

The inspectors reviewed the five post-maintenance tests listed below to verify that procedures and test activities ensured system operability and functional capability. The inspectors reviewed the licensee's test procedure to verify that the procedure adequately tested the safety function(s) that may have been affected by the maintenance activity, that the acceptance criteria in the procedure were consistent with information in the applicable licensing-basis and/or design-basis documents, and that the procedure had been properly reviewed and approved. The inspectors also witnessed the test or reviewed the test data, to verify that test results adequately demonstrated restoration of the affected safety function(s). Additional documents reviewed are listed in the attachment.

- WO 03-000284-000, Transfer switch 0-XSW-201-DA for the Security BU DG failed to properly transfer during testing
- 1-PI-EFT-082-002.B, Diesel Generator 1B-B Two (2) Year Electrical Inspection, Revision 4

- 0-SI-EBT-082-238.2, Diesel Generator Battery Quarterly Operability, Revision 8

- WO 03-001019-000, Electric Board Room Chiller Package B-B, Repair leaks, Recharge, and Retest

- WO 03-002725-000 Repair 1-PSV-068-0340A0, Unit 1 pressurizer power operated relief valve

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities

a. Inspection Scope

The inspectors reviewed the outage safety plan and contingency plans for the Unit 1 refueling and steam generator replacement outage to confirm that the licensee had appropriately considered risk, industry experience, and previous site-specific problems in developing and implementing a plan that assured maintenance of defense-in-depth. Between March 17, 2003 and April 5, 2003, the inspectors observed portions of the shutdown and cooldown processes to verify compliance with TS cooldown restrictions and monitored licensee controls over the outage activities listed below. Documents reviewed during the inspection are listed in the attachment.

- Licensee configuration management, including daily outage reports, to evaluate defense-in-depth commensurate with the outage safety plan and compliance with the applicable TS when taking equipment out of service.

- Licensee implementation of clearance activities to ensure equipment was appropriately configured to safely support the work or testing.

- Installation and configuration of reactor coolant instruments to provide accurate indication and an accounting for instrument error.

- Controls over the status and configuration of electrical systems and switchyard to ensure that TS and outage safety plan requirements were met.

- Decay heat removal processes to verify proper operation and that steam generators, when relied upon, were a viable means of backup cooling.

- Controls to ensure that outage work was not impacting the ability to operate the spent fuel pool cooling system during and after core offload.

- Reactor water inventory controls including flow paths, configurations, and alternative means for inventory addition, and controls to prevent inventory loss.

- Reactivity controls to verify compliance with TS and that activities which could affect reactivity were reviewed for proper control within the outage risk plan.

- Containment closure for control of containment penetrations in accordance with refueling TS, to ensure that containment closure could be achieved during selected configurations, and to verify maintenance of secondary containment in accordance with TS.

- Defueling activities for compliance with TS and to verify proper tracking of fuel assemblies from the core to the spent fuel pool.

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 six risk-significant structures, systems and components (SSC) conducted using the surveillance instructions, listed below, to assess, as appropriate, whether the SSCs met TS operability requirements, the Updated Final Safety Analysis Report (UFSAR) and licensee procedure requirements, and to determine if the testing effectively demonstrated that the SSCs were operationally ready and capable of performing their intended safety functions.

- 1-SI-SXP-062-201.B, Centrifugal Charging Pump 1B-B Performance Test, Revision 7*

- SI-90.8, Reactor Trip Instrumentation Monthly Functional Test (SSPS), Revision 25 (Unit 1 Train B)

- 2-SI-SXP-003-201.S, Turbine Driven Auxiliary Feed Water Pump 2A-S Performance Test, Revision 13*

- 0-SI-OPS-082-007.W, AC Electrical Power Source Operability Verification, Revision 6

- 1-SI-OPS-000-002.0, Shift Log, Revision 65

- 1-SI-OPS-082-024.A, 1A-A DG 24 Hour Run and Load Rejection Test, Revision 9

*This procedure included inservice testing requirements.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed the temporary modification described in Temporary Alteration Control Form (TACF) 2-02-0018-063, Unit 2 refueling water storage tank heaters, to verify that the design was adequate, the modification was properly installed, the modification did not affect system operability, drawings and procedures were appropriately updated, and post-modification testing was satisfactorily performed. Documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification

a. Inspection Scope

The inspectors sampled licensee submittals for the PIs listed below for the period January 1, 2002, through December 31, 2002. To determine the accuracy of the PI data reported for that period, guidance contained in NEI 99-02, Regulatory Assessment Performance Indicator Guideline, Revision 2, was used to verify the basis for reporting each indicator.

Cornerstone: Initiating Events

- Unplanned Scrams per 7000 Critical Hours
- Scrams With Loss of Normal Heat Removal
- Unplanned Power Changes per 7000 Critical Hours

The inspectors reviewed selected LERs and portions of the operator logs to verify that the licensee had accurately identified the number of scrams and unplanned power changes greater than 20 percent that occurred during the previous four quarters for both units. The inspectors also reviewed the accuracy of the number of critical hours reported and the licensee's basis for crediting normal heat removal capability for each of the reported scrams.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems

Section 1R15 describes a finding for a procedure that did not contain the instructions necessary to ensure that multiple breaches of the shield building would be adequately controlled. The licensee had written PER 03-001577-000 and identified that two shield building breaches had not been tracked. However, the PER did not identify the lack of specific instructions nor consider that the undocumented breaches had the potential to allow enough of the Shield building to be breached to exceed the margin of operability for the EGTS system. The inspectors therefore considered this finding to indicate a potential problem identification deficiency.

4OA3 Event Follow-up

.1 Frozen RWST Level Instrumentation (NOED 03-6-001)

On January 24, 2003, Unit 2 RWST wide range level transmitters 2-LT-63-50 and 2-LT-63-52 failed when the impulse lines froze due to a failure of the strip heaters within the transmitter enclosures. This resulted in the licensee entering TS 3.0.3. The licensee requested and was granted discretion from enforcement of TS 3.3.2.9.a, which required three of four channels to be operable. The NRC granted a one-time 48-hour reduction in the required minimum number of operable channels from three to two to permit the licensee to repair the transmitters without shutting down the unit.

The licensee agreed to the following compensatory measures during the period the NOED was in effect: (1) establishment of temporary heat for all RWST wide level transmitters for both units, (2) the establishment of a fire watch to monitor the additional temporary heat, (3) briefing of licensed operators on the situation and review of procedural operation for manual Emergency Core Cooling System (ECCS) switch over from the RWST to the containment sump, (4) suspension of all work involving the RWST channels for both units, and (5) increased Unit 1 RWST level monitoring frequency from every 12 hours to every 2 hours.

The inspectors reviewed the cause and compensatory measures to ensure they matched the licensee's oral assertions and were consistent with NRC policy and guidance. Pending evaluation of the root cause of the problem leading to the request for enforcement discretion, and any associated enforcement, this issue is identified as URI 50-328/03-03-02 Frozen RWST Instrumentation (NOED 03-6-001). Documents reviewed are listed in the attachment.

.2 Containment Purge Valve Leakage (NOED 03-2-004)

On February 27, 2003, Unit 2 containment penetration X-6 purge valves 2-FCV-30-50 and 2FCV-30-51 failed a local leak rate test due to a broken key on the stem of the inboard valve. The as-found leakage of 29.6 scfm exceeded the TS 3.6.1.9 acceptance limit of 11.25 scfm ($0.05L_a$). The licensee requested and was granted discretion from enforcement of TS 3.6.1.9 Action b, which required the inoperable valves to be restored within 24 hours. The NRC granted an additional 144 hours for the licensee to identify the source of leakage, repair or replace the valve(s), and to perform verification testing without shutting down the unit.

The licensee agreed to the following compensatory measures during the period the NOED was in effect: (1) one valve in Penetration X-6 would be closed and deactivated at all times with leak rate monitoring during the maintenance activity, (2) if overall containment leakage increased to $0.06L_a$ the shutdown actions of TS 3.6.1.9.b would be implemented, (3) the scheduled testing of Diesel Generator 2A-A would be postponed to outside the NOED repair activity and no component would be removed from service that would cause the ORAM-Sentinel risk significance to go above the "Green" level, (4) containment purge operations would not be allowed for the duration of the valve maintenance, and (5) the activity of the reactor coolant would be monitored to provide early detection of an adverse trend.

The inspectors reviewed the cause and compensatory measures to ensure they matched the licensee's oral assertions and were consistent with NRC policy and guidance. Pending evaluation of the root cause of the problem leading to the request for enforcement discretion and any associated enforcement, this issue is identified as URI 50-328/03-03-03 Containment Purge Valve Leakage. Documents reviewed are listed in the attachment.

.3 Unit 2 Manual Reactor Trip With Loss of Normal Heat Sink

On March 10, 2003, following a Unit 2 manual reactor trip due to the loss of condensate pressure and one main feed water pump, the inspectors evaluated plant status, mitigating actions, and the licensee's classification of the event, to enable the NRC to determine an appropriate NRC response. Balance-of-plant problems following the trip resulted in the operators breaking vacuum and closing main steam isolation valves, which resulted in a loss of the normal heat sink. The event was reported to the NRC as event notification (EN) 39652 and documented in the licensee corrective action program as PER 03-002313-000. Evaluation of personnel performance is addressed in Section 1R14.

.4 (Closed) Licensee Event Report (LER) 50-328/2002-003-00, Automatic Reactor Trip Resulting from a Generator Stator Cooling Water High Temperature Caused by a Raw Cooling Water Valve Failure.

The LER was reviewed by the inspectors and no findings of significance were identified. The licensee documented the event and failed equipment in PERs 02-006086-000, 02-006114-000, and 01-005036-000. This event did not constitute a violation of NRC requirements. This LER is closed.

The licensee included additional information in this LER regarding the delayed insertion of Rod Control Cluster Assembly (RCCA) L-11 during the trip. This item has been previously reviewed by the NRC and is the subject of unresolved item (URI) 50-327, 328/02-02-05, Corrective Actions Related to the Apparent Failure of RCCA L-11 to Properly Insert.

.5 (Closed) LER 50-327/2002-002-00, Automatic Reactor Trip Resulting From a Failure of a Breaker Causing an Undervoltage Condition on Two Reactor Coolant Pumps and Failure to Perform a Technical Specification Required Action

The LER was reviewed by the inspectors and no findings of significance were identified. The licensee documented the event and failed equipment in PER 02-008460-000.

In addition to the event described in this report, the licensee identified that offsite power to both units was affected by the loss of Start Bus 2B. With one source of offsite power unavailable to Unit 1, TS 3.8.1.1 required that the remaining source be demonstrated operable within one hour. The licensee determined that this was not done but later demonstrated that the proper offsite sources were operable. This licensee identified violation was previously discussed in Inspection Report 50–327,328/02-04 (Section 4OA7). The inspectors review did not identify any new findings. The licensee documented the problem in PER 02-008493-000. This LER is closed.

.6 (Closed) LER 50-328/2002-004-00, Reactor Trip Resulting From The Loss of a Reactor Coolant Pump

The LER was reviewed by the inspectors and no findings of significance were identified. The licensee documented the event failed equipment in PERs 02-015494 and-000 03-000190-000. This event did not constitute a violation of NRC requirements. This LER is closed.

4OA5 Other Activities

.1 NRC Temporary Instruction (TI) 2515/150, Reactor Pressure Vessel Head and Vessel Head Penetration Nozzles (NRC Bulletin 2002-02)

a. Inspection Scope

The inspectors reviewed the Unit 1 bare metal visual examination and susceptibility calculations performed by the licensee in response to the NRC Order EA-03-009 on interim inspection requirements for reactor pressure vessel heads dated February 11, 2003. The inspection guidelines were provided in TI 2515/150. Additional documents are listed in the attachment.

b. Findings and Observations

No findings of significance were identified. Per the documentation requirements of TI 2515/150, the following attributes were observed:

Verification that visual examination was performed by qualified and knowledgeable personnel

Two teams of three individuals performed the examination of the Unit 1 reactor head. One team worked the day shift and one team worked the night shift. One individual on each shift was a licensee Level III Non-Destructive Examination (NDE) qualified to perform VT-2 inspections. The inspectors reviewed the qualification records and verified that these individuals were certified as Level III VT-2 inspectors.

The other members of each team were vendor employees that operated the remote video camera equipment. These individuals had performed the same examination on Unit 2 in the fall of 2002. The inspectors interviewed all of the individuals and noted they were knowledgeable of the criteria to determine leakage.

Verification that visual examination was performed in accordance with demonstrated procedures

The inspectors reviewed Procedure N-VT-17, Visual Examination for Leakage of Pressurized Water Reactor (PWR) Head Penetrations, Revision 2. The inspectors observed that the examination was done using this procedure. The inspectors verified by direct observation and in discussions with examination personnel that the approved acceptance criteria for head leakage were applied in accordance with the procedures.

Verification that the licensee was able to identify, disposition, and resolve deficiencies

The licensee's examination plan included a VT-2 examination using a remote crawler with attached video cameras in the front and rear. In addition, the examination used the resolution level of a VT-1. The licensee recorded all examinations of the nozzles. Any suspected leakage observed by the visual examination was noted and reviewed by materials engineers. The inspectors verified that the examination results for each nozzle were individually documented.

Verification that the licensee was capable of identifying the Primary Water Stress Corrosion Cracking (PWSCC) phenomenon described in the bulletin

The inspectors visually observed the Unit 1 reactor head during the licensee's examination; observed the licensee conduct the examination; discussed the examination with the licensee examiners prior to, during, and following the examination; and verified the qualifications of the licensee examination personnel. The inspectors concluded that the licensee's visual examination was adequate to identify potential leakage resulting from PWSCC cracking of reactor head penetrations.

Evaluate condition of the reactor vessel head (debris, insulation, dirt, boron from other sources, physical layout, viewing obstructions)

The inspectors viewed the condition of the Unit 1 reactor vessel head via remote video and performed a direct observation of the peripheral portions of the head. There was some debris around several nozzles (metal shavings, nails, other construction type debris). Most of the debris was removed with compressed air. Those nozzles where the debris could not be removed were reviewed by engineering. At four locations the licensee used a wedge to support the insulation shroud to allow room for the crawler to reach the upper portions of the head. The inspectors concluded that this was only a minimal obstruction and that the annulus area of nozzles near the wedge could be viewed from the entire circumference. The inspectors observed no other significant items that prevented a thorough visual examination.

<u>Evaluate ability for small boron deposits, as described in the bulletin, to be identified and characterized</u>

The inspectors observed that the reactor head was generally free of any deposits that would have hindered the visual examination. The licensee observed evidence of boric acid leakage around Nozzle 3, performed follow-up NDE on this nozzle, and found no evidence of a leak path. In addition, the licensee characterized six nozzles on the periphery of the head (Nozzles 53, 64, 65, 72, 73, and 78) as indeterminate due to either heavy debris or boron deposits. They performed NDE on five of these nozzles with no indication of a leak path. The licensee did not perform any extra testing on Nozzle 78, but determined from isotopic analysis that the boron deposits on this nozzle as well as those on Nozzle 3 came from external leakage on a previous cycle.

<u>Determine extent of material deficiencies (associated with the concerns identified in the bulletin) which were identified that required repair</u>

The licensee found boron deposits around Nozzle 3 and boron deposits on six nozzles around the periphery of the head. The licensee performed NDE on these nozzles to clarify whether or not the deposits were from inside the head. All NDE results were negative. Additional inspection per Technical Instruction 2515/150 will be performed for Unit 1 in the next inspection period. This will include a more in-depth review of the identified indications in the nozzles and if Reactor Coolant System (RCS) pressure boundary leakage existed. Technical Instruction 2515/150 will remain open pending completion of the inspection objectives.

<u>Determine any significant items that could impede effective examinations</u>

Other than those minor examples mentioned above, the inspectors observed no examples of significant items that could impede the visual examination process.

<u>Determine the basis for the temperatures used in the susceptibility ranking calculation</u>

The licensee used 547°F as the head temperature in the calculation. This was based on the reactor vessel inlet temperature, T-cold, described in the UFSAR, and test data. In January, 1981, the licensee placed five thermocouples on the Unit 1 head to test bypass flow modifications and to confirm the existence of enhanced flow in the head. The test data showed the head temperature to match T-cold. The inspectors reviewed the UFSAR and the test data. The test data showed that the lowest T-cold measured was generally higher than the highest head temperature. In cases where the head temperature exceeded T-cold, the difference was 3°F or less. Based on this data and the UFSAR T-cold of 544.8°F, the inspectors determined the calculation was conservative. The inspectors also checked Unit 2, which was operating at full power and confirmed that T-cold temperature was less than 547°F.

.2 Steam Generator Replacement (SGR) Inspection Overview

This inspection report documents completion of inspections required by Inspection Procedure (IP) 50001, “Steam Generator Replacement Inspection,” some of which were completed in accordance with baseline inspection procedures. The table below identifies and correlates specific IP 50001 inspection requirements examined during this inspection period with the corresponding sections of this report.

IP 50001 Section	Inspection Scope	Section of This Report
02.02.d.2.	Controls and plans to minimize any adverse impact on the operating unit and common systems	1R4, 1R20, 4OA5.4
02.03.e.1.	Establishment of operating conditions including defueling, RCS draindown, system isolation and safety tagging	1R20
02.03.e.2.	Implementation of radiation protection controls	4OA5.3
02.03.e.4.	Installation, use, and removal of temporary services	1R20, 4OA5.3

.3 SGR Operating Conditions, Radiation Protection Controls, and Temporary Services

a. Inspection Scope

As required by IP 50001 Section 02.03.e, throughout this inspection period, the inspectors routinely inspected the following activities as they occurred:

- Establishment of operating conditions including defueling, RCS draindown, and system isolation and safety tagging/blocking.

- Implementation of radiation protection controls.

- Installation, use, and removal of temporary services directly related to steam generator replacement activities.

b. Findings

No findings of significance were identified.

.4 SGR Controls to Minimize Adverse Impact on Operating Unit

a. Inspection Scope

As required by IP 50001 Section 02.02.d.2, the inspectors reviewed plans and periodically monitored licensee controls to minimize any adverse impact on the operating unit and common systems. Specific areas reviewed included:

- Modifications to the ABSCE
- AOP-M.07, RSG Heavy Load Drop, Rev. 0

b. Findings

No findings of significance were identified.

4OA6 Meetings, including Exit

.1 Exit Meeting Summary

On April 9, 2003, the resident inspectors presented the inspection results to Mr. Rick Purcell and other members of his staff, who acknowledged the findings. The inspectors confirmed that proprietary information was not provided nor examined during the inspection.

.2 Annual Assessment Meeting Summary

Subsequent to the end of this inspection period, on April 10, 2003, the NRC's Chief of Reactor Project's Branch 6 and the Senior Resident Inspector assigned to the Sequoyah Nuclear Plant met with the Tennessee Valley Authority (TVA) to discuss the NRC's Reactor Oversight Process (ROP) and the Sequoyah annual assessment of safety performance for the period of January 1, 2002 - December 31, 2002. The major topics addressed were: the NRC's assessment program, the results of the Sequoyah assessment, and NRC security activities. Attendees included Sequoyah site management, members of site staff, and corporate management.

This meeting was open to the public. The presentation material used for the discussion is available from the NRC's document system (ADAMS) as accession number ML031130023. ADAMS is accessible from the NRC Web site at http://www/nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel:

J. Bajraszewski, Licensing Engineer
T. Carson, Maintenance Manager
H. Cothran, Steam Generator Manager
D. Clift, Acting Maintenance and Modifications Manager
E. Freeman, Operations Manager
J. Gates, Business & Work Performance Manager
C. Kent, Radcon/Chemistry Manager
D. Koehl, Plant Manager
M. Lorek, Assistant Plant Manager
D. Lundy, Site Engineering Manager
R. Purcell, Site Vice President
R. Rogers, Design Manager
P. Salas, Licensing and Industry Affairs Manager
J. Smith, Site Licensing Supervisor
K. Stephens, Security Manager

NRC personnel:

S. Cahill, Chief, Reactor Projects Branch 6
R. Bernard, Region II, Senior Reactor Analyst

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-328/2003-003-02	URI	Frozen RWST Instrumentation (Section 4OA3.1).
50-328/2003-003-03	URI	Containment Purge Valve Leakage (Section 4OA3.2).

Opened and Closed

50-327/2003-003-01	NCV	Inadequate Instructions for Controlling Shield Building Breaches (Section 1R15 and 4OA2).

Closed

50-328/2002-003-00	LER	Automatic Reactor Trip Resulting from a Generator Stator Cooling Water High Temperature Caused by a Raw Cooling Water Valve Failure (Section 4OA3.4).
50-327/2002-002-00	LER	Automatic Reactor Trip Resulting From a Failure of a Breaker Causing an Undervoltage Condition on Two Reactor Coolant Pumps and Failure to Perform a Technical Specification Required Action (Section 4OA3.5).
50-328/2002-004-00	LER	Reactor Trip Resulting From the Loss of a Reactor Coolant Pump (Section 4OA3.6).

Discussed

50-327, 328/02-02-05	URI	Corrective Actions Related to the Apparent Failure of RCCA L-11 to Properly Insert (Section 4OA3.4).

LIST OF DOCUMENTS REVIEWED

1R04 Equipment Alignment

1,2-47W803-2, Flow Diagram, Auxiliary Feedwater, Revision 55

1R05 Fire Protection

0-PI-FPU-410-701.Q, Inspection of Fire Doors, Revision 1
0-SI-FPU-410-703.0, Inspection of FPR Required Fire Doors, Revision 2
0-SI-FPU-013-600.0, Fire Detection Panel 0-L-600 Test, Revision 0
0-SI-FPU-013-601.0, Fire Detection Panel 0-L-601 Test, Revision 0

0-SI-EFT-039-237.0, Diesel Generator Building CO2 Fire Protection System (System 39), Revision 16

Transient Combustible Evaluation 2003-0001, Reactor Building Annulus

Fire Protection Impairment Permit FOR 2003A0074, Sprinklers and detection coverage out of service for U2 Heating and Ventilation Room - Unprotected Security Guard House

FOR 2003A0072, Sprinkler suppression and Detection Out of Service in U2 Heating and Ventilation Room - Covered Walkway

1R15 Operability Evaluations

0-SI-EBT-082-238.2, Diesel Generator Battery Quarterly Operability, Revision 8

1R16 Operator Work-Arounds

ODM-3.7, Operations Directive Manual - Operator Work-Around Program

WO 03-000284-000, Transfer switch 0-XSW-201-DA for the Security BU Diesel Generator failed to properly transfer back to normal during 0-PI-OPS-000-677.0 - Investigate and Repair

1R19 Post-Maintenance Testing

0-PI-OPS-000-677.0, Operability Performance of Security Backup Diesel Generator, Revision 14

WO 03-000030-000, Electric Board Room Chiller Pkg. B-B Maintenance

1R20 Refueling and Other Outage Activities

Outage and Site Scheduling Directive Manual (O&SSDM) 4.0 - Operational Defense-in-depth Assessment

March 21, 2003, Operational Defense-in-depth Assessment
Tagout 1-TO-2003-0001 per 0-GO-7 section 5.2[8] and 1-SI-OPS-068-001.0

1-SI-OPS-088-006.0, Containment Building Ventilation Isolation (18 Month/100 Hours/7 Days), Revision 12

1-PI-OPS-068-673.D, Daily Requirements for Reduced Inventory/Midloop Operation

0-GO-13, Reactor Coolant System Drain and Fill Operations

0-PI-IXX-068.001.0, Daily Requirements for Reduced Inventory/Midloop

SPP-5.8, Special Nuclear Material Control (fuel assembly transfer forms)

AOP-M.04, Refueling Malfunctions

0-SI-OPS-000-187.0, Containment Inspection

1R23 Temporary Plant Modifications

Drawings 1,2-45N746-2, Revision 5, 1,2-45N799-6, Revision 6, 45N776, Revision 3

UFSAR Section 6.3, Emergency Core Cooling System,

Unit 2 Technical Specification (TS) 3/4.5.5, SQN-2-SI-OPS-002.0, SQN-63-D053-EPM-MDE-041593

4OA3 Event Follow-up

Frozen RWST Level Instrumentation

January 28, 2003, TVA NOED Request
January 30, 2003, NRC NOED (03-6-001)

Local Leak Rate Test failure on Unit 2 Containment Purge Exhaust Valves

March 4, 2003, TVA NOED Request
March 6, 2003, NRC NOED (03-2-004)

4OA5 Other Activities

Temporary Instruction 2515/150

Startup Test SU-8.5.1 - Units 1 & 2, Data Sheet 9, dated January 12, 13, 14, 17, 23, 30, and 31, 1981