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

August 6, 2008

Tennessee Valley Authority
ATTN: W. R. Campbell, Jr.
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 05000327/2008003 AND 05000328/2008003

Dear Mr. Campbell:

On June 30, 2008, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Sequoyah Nuclear Plant, Units 1 and 2. The enclosed inspection report documents the inspection results, which were discussed on July 9, 2008, with Mr. Timothy Cleary 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.

This report documents one NRC-identified finding of very low safety significance (Green), which was determined to involve a violation of NRC requirements. Additionally, two licensee-identified violations which were determined to be of very low safety significance are listed in this report. However, because of their very low safety significance and because they were entered into your corrective action program, the NRC is treating these findings as non-cited violations (NCVs) consistent with Section VI.A.1 of the NRC Enforcement Policy. If you contest any NCV in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, 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 Nuclear Plant.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the

NRC Public Document Room or from the Publically Available Records (PARS) component of NRCs document system (ADAMS). ADAMS is accessible from the NRC Website at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Eugene Guthrie, Chief
Reactor Projects Branch 6
Division of Reactor Projects

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

Enclosure: Inspection Report 05000327/2008003 and 05000328/2008003
w/Attachment: Supplemental Information

cc: w/encl: (See next page)

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Letter to William R. Campbell, Jr. from Eugene Guthrie dated August 6, 2008

SUBJECT: SEQUOYAH NUCLEAR PLANT - NRC INTEGRATED INSPECTION REPORT 05000327/2008 AND 05000328/2008

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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:	05000327/2008003 and 05000328/2008003
Licensee:	Tennessee Valley Authority (TVA)
Facility:	Sequoyah Nuclear Plant
Location:	Sequoyah Access Road Soddy-Daisy, TN 37379
Dates:	April 1, 2008 – June 30, 2008
Inspectors:	C. Young, Senior Resident Inspector S. Freeman, Senior Resident Inspector M. Speck, Resident Inspector W. Loo, Senior Health Physicist (Sections 2OS1 and 4OA1) E. Michel, Senior Reactor Inspector (Sections 1R08 and 4OA7) L. Suggs, Reactor Inspector (Section 4OA5.3)
Approved by:	E. Guthrie, Chief Reactor Projects Branch 6 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000327/2008-003, IR 05000328/2008-003; 04/01/2008 - 06/30/2008; Sequoyah Nuclear Plant, Units 1 and 2; Maintenance Effectiveness.

The report covered a three-month period of inspection by resident inspectors, a senior health physicist, a senior reactor inspector and a reactor inspector. One NRC-identified Green finding which is a non-cited violation was identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (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 4, dated December 2006.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

Green: The inspectors identified a Green, non-cited violation of 10 CFR 50.65(b)(2)(i) for the licensee's failure to include the gland seal steam supply and supply bypass isolation valves in the scope of their maintenance rule program. These valves are used in the emergency operating procedures to mitigate a steam generator tube rupture if a main steam isolation valve were to fail. The licensee entered the issue into their corrective action program.

The finding was more than minor because it was associated with the mitigating systems cornerstone attribute of equipment performance and affected the cornerstone objective of ensuring the availability and reliability of systems that respond to initiating events to prevent undesirable consequences. In accordance with Inspection Manual Chapter IMC 0609.04, Phase 1 - Initial Screening and Characterization of Findings, the finding was determined to be of very low safety significance (Green) because it did not represent an actual loss of a safety function of one or more non-Technical Specification trains of equipment designated as risk-significant per 10 CFR 50.65 (Section 1R12).

B. Licensee-Identified Violations

Two violations of very low safety significance, which were identified by the licensee, were reviewed by the inspectors. Corrective actions taken or planned by the licensee were entered into the licensee's corrective action program. These violations and corrective actions are listed in Section 4OA7.

REPORT DETAILS

Summary of Plant Status:

Unit 1 operated at or near 100% rated thermal power (RTP) the entire reporting period.

Unit 2 operated at or near 100% RTP until April 9, 2008, when a slow power reduction to 80% RTP commenced. 80% RTP was achieved on May 4, 2008, when Unit 2 was shutdown for a refueling outage. Unit 2 achieved criticality on June 4, 2008, and reached 100% RTP on June 8, 2008, where it remained for the remainder of the reporting period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection

a. Inspection Scope

The inspectors reviewed the licensee's readiness of offsite and alternate AC power systems prior to the onset of the high grid loading season. The inspectors reviewed procedures affecting these areas and the communications protocols between the transmission system operator and the licensee to verify that appropriate information is exchanged when issues arise that could impact the offsite power system. The inspectors walked down offsite power supply systems and emergency diesel generators, reviewed corrective action program documents, and interviewed appropriate plant personnel to assess deficiencies and plant readiness for summer high grid loading. Documents reviewed are listed in the Attachment.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

a. Inspection Scope

Partial System Walkdowns: The inspectors performed a partial walkdown of the following two systems to verify the operability of redundant or diverse trains and components when safety equipment was inoperable. The inspectors attempted to identify any discrepancies that could impact the function of the system, and, therefore, potentially increase risk. The inspectors reviewed applicable operating procedures, walked down control system components and verified that selected breakers, valves, and support equipment were in the correct position to support system operation. The inspectors also verified that the licensee had properly identified and resolved equipment alignment problems that could cause initiating events or impact the capability of mitigating systems or barriers and entered them into the corrective action program. Documents reviewed are listed in the Attachment.

- Unit 2 Containment Spray (CS) Train B During Train A Maintenance
- Control Room Ventilation Train B During Maintenance on Train A

b. Findings

No findings of significance were identified.

1R05 Fire Protection

a. Inspection Scope

The inspectors conducted a tour of the six areas listed below to assess the material condition and operational status of fire protection features. The inspectors verified that combustibles and ignition sources were controlled in accordance with the licensee's administrative procedures, fire detection and suppression equipment was available for use; that passive fire barriers were maintained in good material condition; and that compensatory measures for out-of-service, degraded, or inoperable fire protection equipment were implemented in accordance with the licensee's fire plan. Documents reviewed are listed in the Attachment.

- Auxiliary Building Elevation 714 (Corridor)
- Control Building Elevation 685 (Auxiliary Instrument Rooms)
- Control Building Elevation 706 (Cable Spreading Room)
- Control Building Elevation 669 (Mechanical Equipment Room, 250 VDC Battery and Battery Board Rooms)
- Control Building Elevation 749 (Reactor Motor-Operated Valve Board Rooms, Transformer Rooms, Mechanical Equipment Rooms)
- Control Building Elevation 732 (Mechanical Equipment Room and Relay Room)

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures

a. Inspection Scope

The inspectors reviewed the Essential Raw Cooling Water (ERCW) Structure internal flood design to verify that flood mitigation plans were consistent with the design requirements and risk analysis assumptions and that equipment essential for reactor shutdown was properly protected from a flood caused by pipe breaks in the building. Specifically, the inspectors reviewed the licensee's moderate energy line break flooding study and building piping analyses in order to fully understand the licensee's flood mitigation strategy and then verified that the assumptions and results remained valid. The inspectors also walked down the ERCW structure to verify the assumed flooding sources, adequacy of common area drainage, and status of building compartmentalization to ensure that a flooding event would not impact more than one train of ERCW. The inspectors reviewed licensee Procedure (Abnormal Operating Procedure) AOP-M.01, Loss of Essential Raw Cooling Water, Revision 19, to ensure that if a break actually occurred, procedures existed to identify and isolate the leak.

Documents reviewed are listed in the Attachment.

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance

a. Inspection Scope

The inspectors observed performance and reviewed the results of licensee Procedure 1-PI-SFT-070-001.0, Performance Testing of Component Cooling Heat Exchangers 1A1, 1A2, Revision 15, to verify that the acceptance criteria and results appropriately considered differences between testing conditions and design conditions; that test results were appropriately categorized against pre-established acceptance criteria; that the frequency of testing was sufficient to detect degradation prior to loss of heat removal capability below design basis values; and that test results considered test instrument inaccuracies and differences. The inspectors also observed the inspection and cleaning of the 1A1 and 1A2 Component Cooling System Heat Exchangers. Documents reviewed are listed in the Attachment.

b. Findings

No findings of significance were identified.

1R08 In-service Inspection (ISI) Activities

.1 Non-Destructive Examination (NDE) Activities and Welding Activities

a. Inspection Scope

The inspectors reviewed the implementation of the licensee's ISI program for monitoring degradation of the reactor coolant system (RCS) boundary and risk significant piping boundaries during the Unit 2 Spring 2008 refueling outage (2R15). The inspectors' activities consisted of an on-site review of nondestructive examination (NDE) and welding activities to evaluate compliance with the applicable edition of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC), Sections XI (Code of Record for the third 10-year ISI interval was 2001 Edition with 2003 Addenda), and to verify that indications and defects (if present) were appropriately evaluated and dispositioned in accordance with the requirements of the ASME Code, Section XI acceptance standards.

The inspectors' inspection of NDE activities consisted of direct observation of an ultrasonic examination, a review of examination procedures, NDE reports, equipment and consumables certification records, personnel qualification records, calibration reports, and calibration block fabrication drawings (as applicable) for the following:

- Ultrasonic (UT) examination of SIF-162, a 6" valve to pipe austenitic weld (ASME Class 1).
- Liquid Penetrant (PT) examination of 2-SI-2123, a 2" austenitic socket weld (ASME Class 1)
- PT examination of 2-SI-2124, a 2" austenitic socket weld (ASME Class 2)

The inspectors' review of welding activities included a sample of in process welding activities for ASME Class 1 piping to evaluate compliance with procedures and the ASME Code. The inspector reviewed the repair/replacement package, welding procedures, procedure qualification records, certified material test reports for filler material, welder qualification records, and NDE requirements for the following:

- Welds 2-SI-2123, and 2-SI-2124 for replacement of 2" check valve (ASME Class 1 and Class 2 respectively).

b. Findings

No findings of significance were identified

.2 PWR Vessel Upper Head Penetration (VUHP) Inspection Activities

a. Inspection Scope

The licensee was not required to conduct bare metal visual (BMV), or volumetric examinations of vessel upper head in accordance with NRC Order EA-03-009. However, the inspectors reviewed the licensee's calculations for effective degradation years (EDYs), and reviewed the visual inspection performed to identify potential boric acid leaks from pressure-retaining components above the RPV head as referenced in paragraph IV.D of NRC Order EA-03-009.

b. Findings

No findings of significance were identified.

.3 Boric Acid Corrosion Control (BACC) Inspection Activities

a. Inspection Scope

The inspectors reviewed the licensee's BACC program activities to ensure implementation with commitments made in response to NRC Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary," and applicable industry guidance documents. Specifically, the inspectors performed an on site record review of procedures and the results of the licensee's Mode 3 containment walkdown inspections performed during the Unit 1 Spring 2008 outage. The inspectors also conducted an independent walkdown of the containment to evaluate compliance with licensee's BACC program requirements and verify that degraded or non-conforming conditions, such as boric acid leaks identified during the Mode 3 containment walkdown, were properly identified and corrected in accordance with the licensee's BACC and Corrective Action programs.

The inspectors reviewed a sample of engineering evaluations completed for evidence of boric acid found on systems containing borated water to verify that the minimum design code required section thickness had been maintained for the affected components. The inspectors selected the following evaluations for review:

WO 07-773588-000
WO 07-780947-000
WO 07-776857-000

b. Findings

No findings of significance were identified.

.4 Steam Generator (SG) Tube Inspection Activities

a. Inspection Scope

The inspectors reviewed the Unit 2 SGs tube eddy current testing (ECT) examination activities to ensure compliance with Technical Specifications (TS), applicable industry standards, SG Program Procedures, and ASME Code Section XI requirements. The inspectors reviewed examination status reports and discussed them with the site lead Level III analyst to ensure that all tubes with relevant indications were appropriately screened for in-situ pressure testing. Two tubes were identified for in-situ pressure testing. The inspectors verified applicable industry guidelines (EPRI Steam Generator In Situ Pressure Test Guidelines, Revision 3) were followed. The inspectors reviewed the 2R14 Operational Assessment report in conjunction with the tube plugging report to assess the licensee's ability to predict the number and sizes of flaws detected. In addition, the inspectors reviewed the latest Degradation Assessment report to identify the scope of the inspection and verify it addressed potential degradation areas, plant specific degradation history, and applicable operating experience. The inspectors verified that appropriate inspection scope expansion criteria were applied based on inspection results of active and new degradation mechanisms. The inspectors reviewed licensee procedures for tube repair by plugging to verify that repair methods were approved and in accordance with Quality Assurance requirements. In relation to the tube repair methods, the inspectors reviewed the licensee's implementation of the tube repair criteria to ensure it was consistent with plant TS. The inspectors also reviewed licensee actions in response to primary to secondary leakage; however no primary to secondary leakage was identified during the previous operating cycle. Additionally, the inspectors reviewed documentation to ensure that data analysts, ECT probes, and equipment configurations were qualified to detect the expected types of SG tube degradation. The inspectors selected a sample of site-specific Examination Technique Specification Sheets (ETSS) for acquisition and analysis to ensure that their qualification was consistent with industry standards. The inspectors also observed the licensee's contractor during a portion of data acquisition activities to ensure compliance with industry standards for data quality. The inspectors reviewed a sample of data for tubes identified with cracking mechanisms (primary water stress corrosion cracking and outside diameter stress corrosion cracking) and reviewed historical data for these tubes to assess whether they should have been repaired in a previous outage. Tubes SG3, R4C3; SG2, R6C74; SG2, R6C58; SG2, R7C78; and SG2, R8C3 were reviewed.

b. Findings

No findings of significance were identified.

.5 Identification and Resolution of Problems

a. Inspection Scope

The inspectors performed a review of ISI-related problems, including welding, BACC, and SG inspections that were identified by the licensee and entered into the corrective action program as Problem Evaluation Reports (PERs). The inspectors reviewed the PERs to confirm that the licensee had appropriately described the scope of the problem and had initiated corrective actions. The review also included the licensee's consideration and assessment of operating experience events applicable to the plant. The inspectors performed this review to ensure compliance with 10CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requirements. The corrective action documents reviewed by the inspectors are listed in the report attachment.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program

a. Inspection Scope

The inspectors observed as-found simulator training on April 7, 2008. The training involved an unisolable Reactor Coolant System (RCS) leak followed by a steam generator tube rupture. Operators initiated a manual reactor trip and safety injection and plant cooldown and depressurization. Anomalies included a failure of the turbine-driven Auxiliary Feedwater (AFW) pump, a failure of the 1B Emergency Diesel Generator to auto-start, and failure of the steam dump system. The inspectors observed crew performance in terms of 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 shift manager, including the ability to identify and implement appropriate TS action; and group dynamics involved in crew performance. The inspectors also observed the evaluators' critique and reviewed simulator fidelity to verify that it matched actual plant response. Documents reviewed are listed in the Attachment.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness

a. Inspection Scope

The inspectors reviewed the two maintenance activities associated the below listed

PERs to verify the effectiveness of the activities in terms of: 1) appropriate work practices; 2) identifying and addressing common cause failures; 3) scoping in accordance with *10 Code of Federal Regulations* (*CFR*) 50.65 (b); 4) characterizing reliability issues for performance; 5) trending key parameters for condition monitoring; 6) charging unavailability for performance; 7) classification in accordance with *10 CFR 50.65*(a)(1) or (a)(2); 8) appropriateness of performance criteria for structure, system, or components (SSCs) and functions classified as (a)(2); and, 9) appropriateness of goals and corrective actions for SSCs and functions classified as (a)(1). Documents reviewed are listed in the Attachment

- PER 107149, 2A Emergency Diesel Generator (EDG) Inoperable Due to No Operable Exhaust Fan
- PERs 139779 & 140367, Gland Seal Regulator Failure

b. Findings

Introduction: The inspectors identified a Green, non-cited violation (NCV) of 10 CFR 50.65 (maintenance rule) for failure to include the gland seal steam supply and supply bypass valves in the scope of the maintenance rule program. Isolation of this header was specifically called out in the emergency operating procedures to mitigate a steam generator tube rupture.

Description: On March 8, 2008, during steady state operation, the gland seal regulator on the Unit 2 main turbine failed, resulting in a high steam pressure in the header. This caused the downstream relief valve to lift and increased steam flow, resulting in a small increase in reactor power. When operators attempted to isolate the gland seal header using the motor operated shutoff valve, 2-FCV-47-180, the valve failed to close. The operators subsequently isolated the header using the downstream manual shutoff valve. Because the shutoff valve failed to close, the inspectors reviewed the preventive maintenance performed on the valve and its applicability to the maintenance rule. During this review, the inspectors noted that gland seal steam supply valves 1/2-FCV-47-180 and supply bypass valves 1/2-FCV-47-181 were specifically identified in emergency operating procedure (EOP) E-3, Steam Generator Tube Rupture, Revision 17, as part of the isolation of the main steam header if the Main Steam Isolation Valve (MSIV) failed to close following a steam generator tube rupture. The inspectors reviewed the maintenance rule (10 CFR 50.65) and noted that Paragraph (b)(2)(i) called for non-safety related SSCs used in the plant EOPs to be included in the scope of the rule. In addition, Regulatory Guide (RG) 1.160, Monitoring the Effectiveness of Maintenance at Nuclear Power Plants, Revision 2, which provided a means of implementing the maintenance rule, considered SSCs explicitly used in the EOPs as part of the scope of the maintenance rule. The licensee entered this issue into their corrective action program as PERs 139779 and 140367. The licensee evaluated the risk significance designation per 10 CFR 50.65 and determined that the isolation of the gland seal steam from the main steam line was not risk significant.

Analysis: The licensee's failure to include the gland seal steam supply and supply bypass isolation valves within the scope of the maintenance rule program was a performance deficiency. The finding was more than minor because it was associated with the Mitigating Systems Cornerstone attribute of equipment performance and affected the cornerstone objective of ensuring the availability and reliability of systems that respond to initiating events to prevent undesirable consequences. In accordance with IMC 0609.04, Phase 1 - Initial Screening and Characterization of Findings, the finding was determined to be of very low safety significance (Green) because it did not represent an actual loss of a safety function of one or more non-Technical Specification trains of equipment designated as risk-significant per 10 CFR 50.65. No cross-cutting aspect was assigned since the issue was not indicative of licensee current performance, in that, the error occurred during the original scoping of equipment into the maintenance rule program.

Enforcement: 10 CFR 50.65, Paragraph (b)(2)(i), requires that the scope of the maintenance rule program include those non-safety related SSCs relied upon to mitigate accidents or transients or are used in the EOPs. Contrary to this, as of March 8, 2008, the licensee had not included in the scope of their maintenance rule program the gland seal steam supply and supply bypass isolation valves, 1/2-FCV-47-180 and 1/2-FCV-47-181. Because this violation was determined to be of very low safety significance (Green) and has been entered into the licensee’s corrective action program as PERs 139779 and 140367, it is being treated as an NCV consistent with Section VI.A.1 of the NRC Enforcement Policy and is identified as NCV 05000327,328/2008003-01, Gland Seal Steam Header Isolation Valves Not Scoped in Maintenance Rule.

1R13 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope

The inspectors reviewed the following four activities to verify that the appropriate risk assessments were performed prior to removing equipment from service for maintenance. The inspectors verified that risk assessments were performed as required by 10 CFR 50.65 (a)(4), and were accurate and complete. When emergent work was performed, the inspectors verified that the plant risk was promptly reassessed and managed. The inspectors verified the appropriate use of the licensee’s risk assessment tool and risk categories in accordance with Procedure SSP-7.1, On-Line Work Management, Revision 10, and Instruction 0-TI-DSM-000-007.1, Risk Assessment Guidelines, Revision 8. Documents reviewed are listed in the Attachment.

- Unit 2 Elevated Risk Due to Component Cooling System (CCS) Pump B-Train Scheduled Testing
- Unit 1 Elevated Risk Due to Unit 2 Outage Performance of SI-26B
- Review of Elevated Shutdown Risk for Emergent Work to Replace Shear Pin on Unit 2 Fuel Transfer Canal Wafer Valve
- Elevated Trip Risk Due to Emergent Repairs to 2-FCV-3-48, Loop 2 Main Feedwater Regulating Valve

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

For the seven operability evaluations described in the PERs listed below, the inspectors evaluated the technical adequacy of the evaluations to ensure that TS operability was properly justified and the subject component or system remained available, such that no unrecognized increase in risk occurred. The inspectors reviewed the Updated Final Safety Analysis Report (UFSAR) to verify that the system or component remained available to perform its intended function. In addition, the inspectors reviewed compensatory measures implemented to verify that the compensatory measures worked as stated and the measures were adequately controlled. The inspectors also reviewed a sampling of PERs to verify that the licensee was identifying and correcting any deficiencies associated with operability evaluations. Documents reviewed are listed in the Attachment.

- PER 141305, Blown Fuse in U2 AFW Control Circuit
- PERs 140677 and 142648, Blown Fuse in Emergency Gas Treatment System (EGTS) B Heater Circuit
- PER 141747, Spurious High Temperatures Alarms on U1 Reactor Coolant Pump (RCP) #1 Lower Bearing
- PER 142444, Thermal Performance Testing of 1A1/1A2 CCS HX Fouling Factor Close to Operability Limit
- PER 142870, Vital Battery III Did Not Meet Surveillance Acceptance Criteria
- PER 145600, EDG 2A Received Low Lube Oil Alarm That Would not Reset
- PER 146640, 2B Motor-driven Auxiliary Feedwater Pump (MDAFP) Did Not Meet Test Acceptance Criteria

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). Documents reviewed are listed in the Attachment.

- Work Order (WO) 08-774382-007, Replace Undervoltage Coil UVX in 2A-A Shutdown Board Normal Feeder Breaker
- WO 08-774128-000, Replace Slide Resistor in Vital Battery Charger 4
- WO 07-774613-000, Source Range Channel N31 Failed

- WO 07-778879-000, Troubleshoot/Repair Primary Emergency Core Cooling System (ECCS) Check Valve, 2-VLV-063-562
- WO 08-775365-000, Troubleshoot/Repair Cause of Increased Controller Output to Loop 2 Main Feedwater Regulating Valve, 2-FCV-3-48

b. Findings

No findings of significance were identified.

1R20 Refueling and Other Outage Activities

a. Inspection Scope

For the Unit 2 refueling outage that began on May 4, 2008, the inspectors evaluated licensee activities to verify that the licensee considered risk in developing outage schedules, followed risk reduction methods developed to control plant configuration, developed mitigation strategies for the loss of key safety functions, and adhered to operating license and TS requirements that ensure defense-in-depth. The inspectors also walked down portions of Unit not normally accessible during at-power operations to verify that safety-related and risk-significant SSCs were maintained in an operable condition. Specifically, between May 4, 2008 and June 4, 2008, the inspectors performed inspections and reviews of the following outage activities. Documents reviewed are listed in the Attachment.

- Outage Plan. The inspectors reviewed the outage safety plan and contingency plans 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.

- Reactor Shutdown. The inspectors observed the shutdown in the control room from the time the reactor was tripped until operators placed it on the Residual Heat Removal (RHR) system for decay heat removal to verify that TS cooldown restrictions were followed. The inspectors also toured the lower containment as soon as practicable after reactor shutdown to observe the general condition of the RCS and emergency core cooling system components and to look for indications of previously unidentified leakage inside the polar crane wall.

- Licensee Control of Outage Activities. On a daily basis, the inspectors attended the licensee outage turnover meeting, reviewed PERs, and reviewed the defense-in-depth status sheets to verify that status control was commensurate with the outage safety plan and in compliance with the applicable TS when taking equipment out of service. The inspectors further toured the main control room and areas of the plant daily to ensure that the following key safety functions were maintained in accordance with the outage safety plan and TS: electrical power, decay heat removal, spent fuel cooling, inventory control, reactivity control, and containment closure. The inspectors also observed a tagout of the Safety Injection (SI) pumps to verify they were appropriately configured to safely support the work or testing. To ensure that RCS level instrumentation was properly installed and configured to give accurate information, the inspectors reviewed the installation of the Mansell level monitoring system. Specifically, the inspectors discussed the system with engineering, walked it down to verify that it was installed in accordance with procedures and adequately protected from inadvertent damage, verified that Mansell indication properly overlapped with pressurizer level instruments during pressurizer draindown, verified that operators properly set level alarms to procedurally required setpoints, and verified that the system consistently tracked RCS level while lowering to reduced inventory conditions. The inspectors also observed operators compare the Mansell indications with locally-installed ultrasonic level indicators during entry into mid-loop conditions.

- Refueling Activities. The inspectors observed fuel movement at the spent fuel pool and at the refueling cavity in order to verify compliance with TS and that each assembly was properly tracked from core offload to core reload. In order to verify proper licensee control of foreign material, the inspectors verified that personnel were properly checked before entering any foreign material exclusion (FME) areas, reviewed FME procedures, and verified that the licensee followed the procedures. To ensure that fuel assemblies were loaded in the core locations specified by the design, the inspectors independently reviewed the recording of the licensee's final core verification.

- Reduced Inventory and Mid-Loop Conditions. Prior to the outage, the inspectors reviewed the licensee's commitments to Generic Letter 88-17. Before entering reduced inventory conditions the inspectors verified that these commitments were in place, that plant configuration was in accordance with those commitments, and that distractions from unexpected conditions or emergent work did not affect operator ability to maintain the required reactor vessel level. While in mid-loop conditions, the inspectors verified that licensee procedures for closing the containment upon a loss of decay heat removal were in effect, that operators were aware of how to implement the procedures, and that other personnel were available to close containment penetrations, if needed.

- Heatup and Startup Activities. The inspectors toured the containment prior to reactor startup to verify that debris that could affect the performance of the containment sump had not been left in the containment. The inspectors reviewed the licensee's mode change checklists to verify that appropriate prerequisites were met prior to changing TS modes. To verify RCS integrity and containment integrity, the inspectors further reviewed the licensee's RCS leakage calculations and containment isolation valve lineups. In order to verify that core operating limit parameters were consistent with core design, the inspectors also observed portions of the low power physics testing, including reactor criticality.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

For the four surveillance tests identified below, the inspectors verified that the SSCs involved in these tests satisfied the requirements described in the TS surveillance requirements, the UFSAR, applicable licensee procedures, and that the tests demonstrated that the SSCs were capable of performing their intended safety functions. This was accomplished by witnessing testing and/or reviewing the test data. Documents reviewed are listed in the Attachment. Those tests included the following:

Routine Surveillance Tests:

- 2-SI-ICC-092-N35.1, Channel Calibration of 2-XX-92-5003 (N35) Intermediate Range Nuclear Instrumentation System Channel 1, Revision 13

In-Service Tests:

- 2-SI-SXP-062-203.0, Centrifugal Charging Pump 2A-A Comprehensive Pump Test and Check Valve Test, Revision 29

Ice Condenser System Tests:

- 0-SI-MIN-061-106.0, Ice Condenser Flow Passage Inspection, Revision 3

Reactor Coolant System Leak Detection Tests:

- 0-SI-OPS-068-137.0, Reactor Coolant System Water Inventory, Revision 22

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP6 Drill Evaluation

a. Inspection Scope

Resident inspectors evaluated the conduct of a routine licensee emergency drill on June 24, 2008, to identify any weaknesses and deficiencies in classification, notification, and protective action recommendation (PAR) development activities. The inspectors observed emergency response operations in the simulated control room to verify that event classification and notifications were done in accordance with EPIP-1, Emergency Plan Classification Matrix, Revision 40. The inspectors also attended the licensee critique of the drill to compare any inspector-observed weakness with those identified by the licensee in order to verify whether the licensee was properly identifying failures.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety (OS)

2OS1 Access Control To Radiologically Significant Areas

a. Inspection Scope

Access Controls: The inspectors evaluated licensee guidance and its implementation for controlling worker access to radiologically significant areas and monitoring jobs in-progress. The inspectors evaluated the adequacy of procedural guidance; directly observed implementation of administrative and physical radiological controls; evaluated radiation worker (radworker) and Health Physics (HP) technician knowledge of and proficiency in implementing Radiation Protection (RP) requirements; and assessed worker exposures to radiation and radioactive material.

During facility tours, the inspectors directly observed postings and physical controls for radiation areas and high radiation areas (HRAs) within the Unit 2 (U2) containment building, shared Unit 1 and U2 auxiliary building, external buildings, and the independent spent fuel storage installation. The inspectors independently measured radiation dose rates and contamination levels or directly observed conduct of licensee radiation surveys for selected Radiologically Controlled Area locations. Results were compared to current licensee surveys and assessed against established postings and Radiation Work Permit (RWP) controls. Licensee key control and access barrier effectiveness were observed and evaluated for selected Locked HRA locations. Implementation of procedural guidance for Locked HRA and Very High Radiation Area controls were discussed in detail with HP supervisors and management. Physical controls for storage of irradiated material within the spent fuel pool were observed. In addition, licensee controls for areas where dose rates could change significantly as a result of refueling operations or radwaste activities were reviewed and discussed.

The inspectors observed pre-job RWP briefings and reviewed RWP details, including engineering controls for potential airborne radioactivity and surface contamination, to assess communication of radiological control requirements. Radworkers' adherence to RWP guidelines and HP technician proficiency in providing job coverage, including use of contamination controls and airborne surveys, were evaluated through observation of jobs in-progress. Jobs observed included work activities associated with the U2 Mixed Bed Valve Gallery and U2 Seal Table Incore Detector Drive Box "A". Electronic dosimeter (ED) alarm set points were evaluated against area radiation survey results and ED alarm response actions were discussed with radworkers and HP supervisors.

The inspectors evaluated the effectiveness of radiation exposure controls, including air sampling, barrier integrity, engineering controls, and postings through a review of both internal and external exposure results. Licensee evaluations of skin dose resulting from discrete radioactive particle or dispersed skin contamination events during the last refueling outage were reviewed and assessed.

For HRA tasks involving significant dose rate gradients, the inspectors evaluated procedural guidance and implementation for the use and placement of whole body and extremity dosimetry to monitor worker exposure, including the use of multiple badging for refueling outage activities.

RP activities were evaluated against the requirements of Updated Final Safety Analysis Report Chapter 12; Technical Specifications Sections 6.8 and 6.12; 10 Code of Federal

Regulations Part 20; and approved licensee procedures. Records reviewed are listed in Section 2OS1 and 4OA1 of the report Attachment.

Problem Identification and Resolution: The inspectors reviewed and assessed select Problem Evaluation Reports associated with access control to radiologically significant areas. The inspectors evaluated the licensee's ability to identify, characterize, prioritize, and resolve the identified issues in accordance with procedure SPP-3.1, Corrective Action Program, Revision 13. In addition, the inspectors reviewed self-assessments related to the area of access controls. Specific corrective action program documents associated with access control issues, personnel radiation monitoring, and personnel exposure events reviewed and evaluated during inspection of this program area are identified in Section 2OS1 and 4OA5 of the report Attachment.

The inspectors completed 21 of the required line-item samples described in NRC Inspection Procedure (IP) 71121.01.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification

a. Inspection Scope

The inspectors sampled licensee data for the performance indicators (PIs) listed below. To verify the accuracy of the PI data reported during the period reviewed, PI definitions and guidance contained in NEI 99-02, Regulatory Assessment Indicator Guideline, and Rev. 5 were used to verify the basis for each data element.

Occupational Radiation Safety (OS) Cornerstone: The inspectors reviewed the Occupational Exposure Control Effectiveness PI results from October 2007 through March 2008. For the assessment period, the inspectors reviewed electronic dosimeter alarm logs and assessed corrective action program documents to determine whether HRA, VHRA, or unintended radiation exposures had occurred. The inspectors also reviewed licensee procedural guidance for collecting and documenting PI data. In addition, the inspectors reviewed selected personnel contamination event data and internal dose assessment results. Report section 2OS1 contains additional details regarding the inspection of controls for exposure significant areas. Documents reviewed are listed in sections 2OS1 and 4OA1 of the report Attachment.

The inspectors completed one of the required samples for IP 71151. The one sample was for the OS PI.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems

.1 Daily Review

As required by Inspection Procedure 71152, Identification and Resolution of Problems, and in order to help identify repetitive equipment failures or specific human performance issues for followup, the inspectors performed a daily screening of items entered into the licensee's corrective action program. This was accomplished by reviewing the description of each new PER and attending daily management review committee meetings.

.2 Annual Sample Review of Feedwater Speed Control Issues that Resulted in a Reactor Trip

a. Inspection Scope

On March 13, 2007, Unit 2 was manually tripped when partial loss of Main Feed Pump (MFP) 2A resulted in low level on all four steam generators. The loss of MFP 2A was found to be caused by failures in the pump speed controller, a model Foxboro 62-H. While there were no findings of significance with this event itself in Inspection Report (IR) 2007-003, Section 4OA3.2, generic controller issues could result in similar problems on other systems in the future. Therefore, the inspectors reviewed licensee actions to determine the root cause and the actions to correct the cause. The inspectors reviewed PER 121526 dealing with this event, interviewed maintenance and engineering personnel, and reviewed several of the corrective actions

b. Findings and Observations

There were no findings of significance identified during this review. The inspectors determined that the root cause was thorough and that immediate and long term corrective actions appeared to address the causes. The root cause team performed a Kepner-Tregoe Problem analysis and determined that a Local/Remote switch, internal to the MFP speed controller, failed open and caused erratic operation with the controller in automatic mode. The team further determined that a relay, also internal to the controller, failed to actuate when the operators attempted to place the controller in manual. This prevented any manual control of the feedwater regulating valve. The licensee developed several actions to address these causes and implemented them beginning in August 2007. These included initiating and implementing a design change to place a jumper around the Local/Remote switch, initiate and implement preventive maintenance instructions to periodically exercise the internal controller relay, and to develop a site performance indicator to track the age and resolution of requests for new preventive maintenance requests. The inspectors reviewed these actions and concluded that they addressed the cause and verified that they were being implemented.

.3 Semi-Annual Trend Review

a. Inspection Scope

As required by Inspection Procedure 71152, the inspectors performed a review of the licensee's corrective action program and associated documents to identify trends that could indicate the existence of a more significant safety issue. The inspectors' review was focused on repetitive equipment issues, but also included licensee trending efforts and licensee human performance results. The inspectors' review nominally considered the six-month period of January 2008 through June 2008, although some examples expanded beyond those dates when the scope of the trend warranted. Specifically, the inspectors consolidated the results of daily inspector screening discussed in Section 4OA2.1 into a log, reviewed the log, and compared it to licensee trend reports for the period from November 2007 through February 2008 in order to determine the existence of any adverse trends that the licensee may not have previously identified. The inspectors also independently reviewed RCS leakage data for the six-month period of January 2008 through June 2008.

b. Findings and Observations

No findings of significance were identified. In general, the licensee had identified trends and appropriately addressed them in their corrective action program. The inspectors evaluated the licensee trending methodology and observed that the licensee had performed a detailed review. The licensee routinely reviewed cause codes, involved organizations, key words, and system links to identify potential trends in their data. The inspectors compared the licensee process results with the results of the inspectors' daily screening and did not identify any discrepancies or potential trends that the licensee had failed to identify. There were two trends that had potential significance, both of which were tracked in the corrective action program.

During the Unit 1 End-of-Cycle 15 Outage in November 2007, there were several instances where foreign material was found in or entered unexpected places. These included the reactor vessel, the main condenser, the ice condenser, the main generator, and the reactor cavity. This was discussed in the trend review for IR 2007-005. Subsequently, increased licensee management attention to foreign material controls was observed. Additional training on foreign material prevention and administrative controls and requirements for formal FME plans on certain systems was implemented. During the Unit 2 End-of-Cycle 15 Outage in May 2008, there were several instances of foreign material being introduced into systems, however, none had more than minor safety significance. An increase in PERs identifying inappropriate FME practices was observed indicating an increased management emphasis on foreign material prevention.

The inspectors have noted a continuing tendency to place temporary equipment in safety-related areas, particularly the auxiliary building, without supporting the equipment as described in Procedure 0-TI-DXX-000-013.0, Temporary Equipment Control, Revision 3. Since April 2007 the inspectors have identified seven separate instances where temporary equipment was stored in the auxiliary building without being properly secured, per the procedure. While each incident was of minor safety significance and none affected safety-related equipment, collectively these indicate that licensee personnel were not sensitive to the seismic requirements of the auxiliary building and its contents. The licensee has entered this into the corrective action program as PER 139609 and has taken steps to inform workers of the seismic nature of safety-related areas.

4OA5 Other Activities

.1 Quarterly Resident Inspector Observations of Security Personnel and Activities

a. Inspection Scope

During the inspection period, the inspectors conducted observations of security force personnel and activities to ensure that the activities were consistent with licensee security procedures and regulatory requirements relating to nuclear plant security; these observations took place during both normal and off-normal plant working hours:

These quarterly resident inspector observations of security force personnel and activities did not constitute any additional inspection samples. Rather, they were considered an integral part of the inspectors' normal plant status review and inspection activities.

b. Findings

No findings of significance were identified.

.2 (Closed) Unresolved Item (URI) 05000327,328/2007005-02, Improper Information Provided for Mitigating Systems Performance Index (MSPI)

This URI was opened after the inspectors identified the improper use of importance measures in the unreliability portion of the Emergency Alternating Current (AC) Power MSPI Performance Indicator (PI). This performance deficiency resulted in an indicator that was very close to the green/white threshold in two previous quarters. The PI value for Unit 2, for example, for the June 2006 and March 2007 quarters was 1.04E-6. Because they had already reviewed the failure data that supported the unreliability portion of the PI, the inspectors looked more closely at the unavailability data for previous quarters in order to further verify the accuracy of the PI. This review also included the information used to determine the planned unavailability baseline numbers in the calculation.

The inspectors noted from this review that critical hours for the 12 quarters from January 2002 to December 2004 used to determine the planned unavailability baseline for the Emergency AC Power MSPI PI were not correct. The number of hours used was 26304, while the actual critical hours for Unit 1 during the period were 23377, and for Unit 2 were 23605. The licensee evaluated this discrepancy and determined the effect to be negligible because critical hours were used in the denominator of the unavailability ratio and that would actually make the baseline unavailability ratio greater. Since the baseline was subtracted from the actual unavailable hours this would have the effect of reducing the MSPI PI. The licensee entered these items into the corrective action program as PER 142876 and part of PER 135288.

Based on these reviews the inspectors concluded that the Emergency AC Power MSPI PI remained less than 1E-6 (Green) since the beginning of monitoring in April 2006. Because the MSPI PI did not change color, this issue is considered a minor violation. Although this issue should be corrected, it constitutes a violation of minor significance and is not subject to enforcement action in accordance with Section IV of the Enforcement Policy. This URI is closed.

.3 (Open) Temporary Instruction 2515/174, Hydrogen Igniter Backup Power Verification

a. Inspection Scope

The objective of TI 2515/174, "Hydrogen Igniter Backup Power Verification," was to determine if the licensee had adequately implemented commitments related to providing backup power to containment hydrogen igniters.

During this inspection period, the inspector reviewed the licensee's response to GSI-189, "Susceptibility of Ice Condenser and Mark III Containments to Early Failure from Hydrogen Combustion during a Severe Accident." The licensee committed to revise the back-up generator procedure(s) to include supplying one train of containment hydrogen igniters per unit, and train personnel to the procedure revision, by December 31, 2007. The inspector reviewed the licensee's actions as they related to Unit 1.

The inspector's review of the licensee's commitments consisted of the review and verification that appropriate procedures had been established to govern the provision of backup power to the igniters and included appropriate cautions. Additionally, the feasibility and availability of the power supply to be transported and/or connected from its storage location to the access location for providing power to the igniters using readily available equipment (i.e. fitting, cables, tool, etc.) was evaluated. The inspector also reviewed electrical schematics and capacity/loading calculations to determine if the rating of the portable power supply was adequate to continuously power at least one train of igniters and that unnecessary portions of the internal power distribution system could be separated from portions necessary to provide power to the igniters from the portable supply. The licensee's training records and related vendor documents were reviewed to evaluate whether a suitable training program had been established to train staff in the actions necessary to provide back up power and that all necessary equipment was being tested, stored and maintained consistent with vendor recommendations.

b. Findings and Observations

Evaluation of Inspection Requirements

In accordance with the requirements of TI 2515/174, the results of the inspection are provided below.

(1) Did the licensee modify plant equipment and implement training programs and procedures to provide backup power to at least one complete train of hydrogen igniters?

Yes. Sequoyah procured two trailer-mounted diesel generator sets, each with a capacity of 2 megawatts (MW). These generators could be connected to feed the 6.9kV Shutdown Boards through temporary cables by connecting to the 6.9kV Unit Boards in the Turbine building or by back feeding through the 5th Diesel Generator electrical board at the Emergency Diesel Generator Building. Mitigating Actions 0-MA-REM-000-001.0, Extended Station Blackout procedure provided this guidance and would normally be entered as a result of an extended loss of emergency raw cooling water (ERCW) in conjunction with a loss of all AC power.

The generators could be moved as necessary between the two pre-staged connection areas, using a semi-tractor housed in an onsite location to facilitate connection. This generator capability along with properly made connections would enable at least one train of hydrogen igniters on each unit to be restored once the need had been determined.

Operators and select maintenance staff were trained on applicable procedures required to facilitate provision of backup power.

(2) Did the licensee provide the equipment necessary to provide backup power to the hydrogen igniters?

Yes. The two trailer-mounted diesel generator sets could provide backup power to the hydrogen igniters through the C and A Vent Boards. All necessary cables and fittings were pre-staged at the storage location of the diesel generators and other designated locations in the Turbine Building, 5th Diesel Generator Room and Emergency Diesel Generator Building. Additional tools and tow vehicles that would be needed to facilitate connections were staged onsite and itemized in procedures.

(3) Did the licensee establish appropriate procedures to govern the provision of backup power to the igniters?

The licensee modified their Extended Station Blackout procedure to support the additional function and clarified its intent regarding power to the hydrogen igniters. While preparing to discuss restoration of power to the igniters with the inspector, the licensee identified a deficiency with the Extended Station Blackout procedure. Appendix R of this procedure provided guidance on restoring power to the 480V C and A Vent Boards, which energize the igniters; however, it left all the individual load breakers (including the breakers for the hydrogen igniters) in the OPEN position. Guidance for reclosing the hydrogen igniter breaker was omitted when the igniters were addressed in Revision 3 of the procedure. The licensee revised the procedure and generated a problem report to address this issue (PER 144301). This issue is being considered an Unresolved Item pending further inspection and review: URI 05000327, 328/2008003-02, Procedure 0-MA-REM-000-001.0, Extended Station Blackout, Did Not Close Hydrogen Igniter Breakers.

The licensee established appropriate procedures to govern the Movement of the Blackout Diesel Generators to the 5th Diesel Generator Building (0-MA-ESC-317-300.1), the Connection of the Blackout Diesel Generator to 5th Diesel Generator Board (0-MA-ESC-317-300.2), and the Connection of the Blackout Diesel Generator to 6.9kV Unit Board Compartments (0-MA-ESC-317.300.4).

(4) Did the licensee establish a suitable training program to train selected staff in the actions necessary to provide backup power to the igniters?

In response to this commitment, the licensee issued a training letter STL-0739, Revision to 0-MA-REM-000-001.0, Extended Station Blackout, which notified operators of the revision to the mitigating actions procedure regarding the enhancement of the hydrogen igniters and required each operator’s signature. The training letter was issued in December 2007; however the last two signatures were not obtained until a request to review the training records was submitted by the inspector. The licensee generated PER 144366 and entered this issue into their corrective action program.

Maintenance personnel who also served as emergency responders were trained on separate procedures that govern the movement and connection of the Blackout Diesel Generators to either the 5th Diesel or the Unit Boards in October of 2004. The licensee plans to train the remaining maintenance personnel on these procedures and include such training in the continuing training program by July 25, 2008 (PER 144536 and 130581).

(5) Did the licensee establish maintenance and testing schedules consistent with vendor recommendations for permanently installed equipment?

The contract vendor performed inspection and maintenance on the Blackout Diesel Generators as a part of the licensee's preventive maintenance work. The 2MW Diesel Generator was scheduled to be tested by the vendor on a 12 week frequency; however, the companion transformer is not tested and no routine maintenance is performed. The inspector discussed this observation with the licensee.

The licensee's actions were consistent with vendor recommendations for the pre-staged separable connectors used as fittings to facilitate cable connections. The licensee performed routine surveillance on the permanently installed hydrogen mitigation system consistent with Technical Specifications.

4OA6 Meetings, Including Exit

On July 9, 2008, the resident inspectors presented the inspection results to Mr. Timothy Cleary and other members of his staff, who acknowledged the findings. The inspectors asked the licensee whether any of the material examined during the inspection should be considered proprietary. No proprietary information was identified.

4OA7 Licensee-Identified Violations

The following violations of very low significance (Green) were identified by the licensee and are violations of NRC requirements that meet the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned a an NCV.

- 10 CFR 50.65(a)(4) requires the licensee to assess and manage increases in risk resulting from maintenance. Procedure SPP-7.2, Outage Management, Revision 10, implemented risk management reviews during outage periods and required that a risk assessment should be reperformed when emergent conditions exist which could change the conditions of a previously performed risk assessment. Contrary to this, on May 8, 2008, a shear pin failed on the fuel transfer tube wafer valve 2-VLV-78-610, which prevented achieving containment closure had it been required. This emergent condition resulted in an escalation in risk from a Yellow threshold to an Orange threshold; however, a risk assessment was not reperformed. The risk condition change was not recognized prior to the licensee performing maintenance, specifically detensioning the reactor vessel head and entering Mode 6. The licensee replaced the failed shear pin and reestablished containment closure capability. Following repairs, the licensee realized that an elevated risk condition resulted from the shear pin failure, and this had not been reevaluated as required. The issue was entered into the corrective action program as PER 144542. This finding was determined to be of very low safety significance (Green) because of the short duration of the condition and the number and nature of risk management actions already in place.

- 10 CFR 50, Appendix B, Criterion XVI requires, in part, that "Measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected." Contrary to the above, the licensee failed to identify the presence of, and subsequently repair, axial outside diameter stress corrosion cracking (ODSCC) in the free-span region of steam generator 3 tube R4C3 during the previous refueling cycle. The licensee entered this issue into the Corrective Action Program (CAP) as PER 145579. This finding is of very low safety significance because NRC Inspection Manual, Manual Chapter 0609, Appendix J, Table 1 provides guidance stating that one or more tubes that should have been repaired as a result of a previous inspection is preliminarily Green. Additionally, the indication in tube R4C3 was in-situ pressure tested and withstood $3x\Delta P_{NO}$ with no detected leakage.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel

D. Bodine, Chemistry/Environmental Manager
D. Boone, Radiation Protection Manager
M. Button, Maintenance and Modifications Manager
C. Church, Plant Manager
T. Cleary, Site Vice President
L. Cross, Acting Maintenance Manager
D, Jaquith, Licensing Engineer
K. Jones, Engineering Manager
Z. Kitts, Licensing Engineer
A. Little, Acting Site Security Manager
T. Marshall, Outage and Site Scheduling Manager
J. Proffitt, Licensing Engineer
P. Simmons, Operations Manager
J. Smith, Licensing and Industry Affairs Manager
N. Thomas, Licensing Engineer
R. Thompson, Licensing Supervisor
K. Wilkes, Emergency Preparedness Manager

NRC personnel:

R. Bernhard, Region II, Senior Reactor Analyst
B. Moroney, Project Manager, Office of Nuclear Reactor Regulation

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000327, 328/2008003-02	URI	Procedure 0-MA-REM-000-001.0, Extended Station Blackout, Does Not Close Hydrogen Igniter Breakers (Section 4OA5.3)

Opened and Closed

05000327, 328/2008003-01	NCV	Gland Seal Steam Header Isolation Valves Not Scoped In Maintenance Rule (Section 1R12)

Closed

05000327,328/2007005-02	URI	Improper Information Provided For MSPI (Section 4OA5.2)

Discussed

2515/174	TI	Hydrogen Igniter Backup Power Verification (Section 4OA5.3)

LIST OF DOCUMENTS REVIEWED

Section 1R01: Adverse Weather Protection

1,2-45N500, Switchyard Wiring Diagrams Single Line, Revision 26
GOI-6, Apparatus Operations, Revision 127
IGA-6, Transmission/Power Supply Intergroup Agreement, Revision 10
OPDP-9, Emergent Issue Response, Revision 3
SPP-7.1, On Line Work Management, Revision 10
TRO-TO-SOP-10.129, Sequoyah Nuclear Plant (SQN) Grid Operating Guide, Revision 5

Section 1R04: Equipment Alignment

1,2-47W812-1, Flow Diagram Containment Spray System, Revision 43
1,2-47W866-4, Flow Diagram Control Building Heating Ventilation and Air Conditioning, Revision 33
1,2-47W867-4, Control Building Mechanical Heating Ventilation and Air Conditioning Controls Revision 17

Section 1R05: Fire Protection

Sequoyah Nuclear Plant Fire Protection Report, Revision 22

Section 1R06: Flooding

NUREG 0800, Section 3.6.2, Standard Review Plan Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping, Revision 1
Sequoyah Probabilistic Safety Assessment, Individual Plant Examination, Volume 3, Revision 1
SQN-SQS40056, Moderate Energy Line Break Flooding Study, Revision 12
SQN-DC-V-1.1.11, Evaluating the Effects of a Pipe Failure Outside Containment, Revision 6
SQN-DC-V-13.3, Detailed Analysis of Category I and I(L) Piping Systems, Revision 6
SQN-CEB-N2-67-4A, Summary of Piping Analysis N2-67-4A, Revision 14
PER 141423, MELB Flood Levels ERCW Pump Station

Section 1R07: Heat Sink Performance

1-PI-SFT-070-001.0, Performance Testing of Component Cooling Heat Exchangers 1A1, 1A2, Revision 15
WO 07-774851 Inspect/Clean 1A1 CCS HX
0-MI-MRR-070-611.0, Component Cooling System Heat Exchanger Maintenance, Revision 10

Section 1R08: In-service Inspection Activities

Procedures/Calculations
2-MI-MXX-068-005.0, Appendix E, Rev. 0011, In-Situ Pressure Calculation Sheet, prepared for in-situ pressure testing SG2, R8C3 and SG3, R4C3.
Westinghouse Procedure SQN-006, “Standard In Situ Pressure Test Using the Computerized Data Acquisition System” – including data from in situ pressure testing of tubes SG2, R8C3 and SG3 R4C3, Rev 7

ACTS #SQN2-02-08, Steam Generator Eddy Current Inspection Multi-Frequency Eddy Current Parameters, Rev 0
ACTS #SQN2-03-08, Steam Generator Eddy Current Inspection Multi-Frequency Eddy Current Parameters, Rev 0
ACTS #SQN2-04-08, Steam Generator Eddy Current Inspection Multi-Frequency Eddy Current Parameters, Rev 0
ANTS SQN2C15-Bobbin, Rev 0
Westinghouse Procedure TVA-400-001, Multifrequency Eddy Current Examination of Non-Ferromagnetic Steam Generator Tubing, Rev 12
0-VI-MXX-068-022.0, Vendor Steam Generator Maintenance Instructions, Revision 49
2-SI-SXI-0680114.3, Steam Generator Tubing In-service Inspection and Augmented Inspections, Revision 0003
ETSS #20510.1, Eddy Current Examination Technique Specification Sheet, Rev 7
ETSS #21401.1, Eddy Current Examination Technique Specification Sheet, Rev 4
ETSS #20511.1, Eddy Current Examination Technique Specification Sheet, Rev 8
Sequoyah Nuclear Plant – Unit 2 Steam Generator Eddy Current Examination Guideline, Rev 8, September 26, 2006
MDQ-002-068-2006-0169, "Calculation for the Determination of Effective Degradation Years (EDY) for the Sequoyah Unit 2 Cycle 14 Reactor Pressure Vessel Head Susceptibility," Rev 0
GWPS 1.M.1.2-23, Attachment B, Weld Joint Detail Branch Connections Socket Weld Fittings, Rev 7
0-PI-SLT-068-200.0, "Reactor Building Post Shutdown Leakage Examination," Revision 0001
SPP-9.7, "Corrosion Control Program," Rev. 0015
N-UT-64, "Generic Procedure for the Ultrasonic Examination of Austenitic Pipe Welds," Rev 0011
PDQS No: 569, "PDI-UT-2; Revision: C; Addenda: 3," 10/18/2007

Corrective Action Documents
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* PER 145579, Plug SG 3, 5/21/2008
* PER 145652, SQN U2C15 ISI Inspection, 5/22/2008
PER 109759, CCPIT Leakage, 9/5/2006
PER 115534, Unit 2 Decreasing Pressurizer Level, 11/30/2006
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PER 118464, Unit 2 Turbine Trip, 1/25/2007
PER 139838, Applicability of OE Event Number 44037, 3/12/2008

Other
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MIZ70/MIZ30 Equivalency, 5/14/2002
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Sequoyah Unit 2 Technical Specifications Steam Generator (SG) Program
Zetec Certificate of Conformance for various eddy current probes
Generic Letter 95-05, Voltage Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking
Letter From Brenda Mozafari to Mr. William R. Campbell, Jr., CNO TVA, SUBJ: SEQUOYAH NUCLEAR PLANT, UNIT 2 – REVIEW OF STEAM GENERATOR TUBE INSPECTION REPORTS FOR THE 2006 REFUELING OUTAGE (TAC No. MD5142), 3/5/2008
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Mid-South Nuclear, Inc., Certificate of Conformance, 1/8” ER316/316L, Heat# 271748, 9/2/1998
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WO #06-774890-000, Replace Existing Rockwell Check Valve with a Generic Substitute, SQN-2-VLV-063-0543
SEOR 02-04, “Davis-Besse” – PowerPoint Presentation given as management/supervisor boric acid training
WO 07-773588-000 – Boric Acid Evaluation
WO 07-780947-000 – Boric Acid Evaluation
WO 07-776857-000 – Boric Acid Evaluation

* Corrective Action documents created as a result of this inspection.

Section 1R11: Licensed Operator Requalification

AOP R.05, RCS Leak and Leak Source Identification, Revision 11
E-0, Reactor Trip or Safety Injection, Revision 29
E-3, Steam Generator Tube Rupture, Revision 17

Section 1R12: Maintenance Rule Implementation

TI-4, Maintenance Rule Performance Indicator Monitoring, Trending, and Reporting - 10CFR50.65, Revision 20
SPP-6.6, Maintenance Rule Performance Indicator Monitoring, Trending, and Reporting – 10CFR50.65, Revision 9
AOP-S.05, Steam or Feed Leak, Revision 6
E-3, Steam Generator Tube Rupture, Revision 17
Westinghouse Emergency Response Guidelines, Steam Generator Tube Rupture, Revision 2
RG-1.160, Monitoring the Effectiveness of Maintenance at Nuclear Power Plants, Revision 2
NUREG-1526, Lessons Learned from Early Implementation of the Maintenance Rule at Nine Nuclear Power Plants, dated June 1995
NUREG-1648, Lessons Learned from Maintenance Rule Baseline Inspections, dated October 1999

0-MI-EPM-317-102.0, Insulation Resistance Test of Cables and Motors, Revision 26
0-MI-MXX-000-008.0, Pulley Alignment and Belt Tensioning of Belt Driven Components, Revision 12
PER 141307, 2A-A DG Exhaust Fan 1 Step Increase in Motor Vibrations
PER 131965, 2A1 DG Exhaust Fan Thermal Overloads Open
PER 107235, 2A-A DG Exhaust Fan Thermal Overloads Open
WO 07-780681-000, Troubleshoot/Repair 2A1 Emergency Diesel Generator Exhaust Fan Overloads

Section 1R13: Maintenance Risk Assessments and Emergent Work Evaluation

Sentinel Run April 7-27, 2008 dated April 14, 2008
Operations Directive Manual, Appendix D Protected Equipment, Revision 5
Sentinel Runs May 12 to 1 June, 2008 dated May 19, 2008 and May 20, 2008
Sentinel Run June 16-29, 2008 dated June 17, 2008
2-SO-2/3-1, Condensate and Feedwater System, Revision 65

Section 1R15: Operability Evaluations

FSAR 10.4.7.2 Auxiliary Feedwater System
NUREG 1022, Event Reporting Guidelines 10 CFR 50.72 and 50.73, Revision 2
1,2-45N603-2, Wiring Diagrams Main & Aux Feedwater Sys Schematic Diagrams Sheet 2, Revision 4
1,2-45W657-26, Wiring Diagrams Separation & Misc Aux Relays Schematic Diagrams SH 26, Revision 13
Letter from Black to Kingsley on Unit 1 License Amendment 103 and Unit 2 Amendment 92, dated March 6, 1989
OPDP-4, Annunciator Disablement, Revision 2
Disabled Alarm Checklist/Technical Evaluation-SER 2150- dated April 11, 2008
Alarm Response Procedure 1-AR-M5-B, (E-2), TS-62-42, Reactor Coolant Pumps Lower Bearing Temp High, Revision 36
AOP-R.04. Reactor Coolant Pump Malfunctions, Revision 23
0-SI-EBT-250-100.4, Modified Performance Testing of 125Vdc Vital Batteries and 125Vdc Vital Battery Charger Test, Revision 17 – multiple performances
FE 42570 Excessive Fouling in 1A1/1A2 CCS HX
1-PI-SFT-070-001.0, Performance Testing of Component Cooling Heat Exchangers 1A1, 1A2, Revision 15 – multiple performances
1/2-47W816-2, Flow Diagram Diesel Generator Lube Oil System, Revision 0
2-SI-SXP-003-202.B, Motor-Driven Auxiliary Feedwater Pump 2B-B Comprehensive Performance Test, Revision 0
FE 42682 – MDAFW Pump 2B-B Section XI Pump Test Failure

Section 1R19: Post Maintenance Testing

SQN-VTD-P319-0030, Instruction Manual Three Phase Magnetic Amplifier Controlled Battery Charger/Eliminator, Revision 3
1,2-45N765-1, Wiring Diagram 6900V Shutdown Aux Power Schematic Sh-1, Revision 16
1,2-45N765-5, Wiring Diagram 6900V Shutdown Aux Power Schematic Sh-5, Revision 14
WO 08-774382-009, Attachment A, Operability Test of D/G 2A-A

PER 144705, Normal Supply Breaker for Shutdown Board 2A-A Cell-switch U-channel Installed Incorrectly
TVA Central Laboratories Services Technical Report EQ28-0021, Perform Failure Analysis of 1818UVX Relay
PER 145398, U2 Source Range Channel N31 Failed to Respond to Startup Source
2-SI-ICC-092-500.1, Channel Calibration of 2-XM-92-5001 Wide Range Amplifier Nuclear Instrumentation System Channel 1, Revision 15
2-SI-SXV-063-206.0, Residual Heat Removal Cold Leg Primary and Secondary Check Valve Integrity Test, Revision 11, performance dated 12/23/06
2-SI-SXV-063-206.0, Residual Heat Removal Cold Leg Primary and Secondary Check Valve Integrity Test, Revision 13, performance dated 5/30-31/08
2-47W811-1, Flow Diagram Safety Injection System, Revision 60
PER 146874, Increased Controller Output to U2 Loop 2 Main Feed Water Regulating Valve

Section 1R20: Refueling and Outage Activities

Clearance 2-63-0119A-RFO, SIS Pump Cold Leg Injection
Clearance 2-63-0149A-RFO, Safety Injection Pump 2A-A
Clearance 2-63-0150-RFO, Safety Injection Pump 2B-B
Clearance 2-63-0345-RFO, Accumulators to Holdup Tanks
Clearance 2-68-0102-RFO, Mansell Power Supplies
GOI-6, Apparatus Operations, Section AF – Mansell Operation and Setting Changes, Revision 127
0-GO-15, Containment Closure Control, Revision 24
0-GO-13, Reactor Coolant System Drain and Fill Operations, Revision 45
0-PIC-ICC-068-005.0, Calibration of Mansell Level Monitoring System Transducers, Revision 3
0-RT-NUC-000-003.0, Low Power Physics Testing, Revision 21
0-RT-NUC-000-008.0, Low Power Physics Testing Acceptance Criteria, Revision 8

Section1 R22: Surveillance Testing

0-RT-NUC-000-003.0, Low Power Physics Testing, Revision 21
2-PI-ICC-092-N35.1, Channel 1 Gamma-Metrics Full Power Alignment, Revision 11 performance dated February 24, 2008
PER 142372, U2 Nuclear Instrument N35 “As-found” values out-of-tolerance
FSAR 6.3.2.2, Emergency Core Cooling System – Centrifugal Charging Pumps
FSAR Figure 6.3.2.7, CCP Performance Curve
2-SI-SXP-062-201.A, Centrifugal Charging Pump 2A-A Performance Test performance dated Feb 14, 2008, Revision 11
2-47W809-1, Flow Diagram Chemical and Volume Control System, Revision 71

Section 1EP6: Drill Evaluation

E-0, Reactor Trip or Safety Injection, Revision 30
E-1, Loss of Reactor or Secondary Coolant, Revision 23
FR-P.1, Pressurized Thermal Shock, Revision 13
FR-Z.1, High Containment Pressure, Revision 17

Section 2OS1: Access Control To Radiologically Significant Areas

Procedures, Instructions, Guidance Documents, and Operating Manuals

Tennessee Valley Authority (TVA), Sequoyah Nuclear Plant (SNP), Radiological Control Instruction (RCI)-01, Radiation Protection Program, Rev. 62
TVA, SNP, RCI-14, Radiation Work Permit (RWP) Program, Rev. 42
TVA, SNP, RCI-21, Control of Radioactive Materials, Rev. 13
TVA, SNP, RCI-22, Contamination Control, Rev. 17
TVA, SNP, RCI-24, Control of Very High Radiation Areas, Rev. 8
TVA, SNP, RCI-28, Control of Locked High Radiation Areas, Rev. 6
TVA, SNP, RCI-29, Control of Radiation Protection Keys, Rev. 7
TVA, SNP, Radiation Protection Management Directive (RMD) FO-02, Radiation and Contamination Surveys, Rev. 19
TVA, SNP, RMD FO-03, Alpha Contamination Monitoring and Controls, Rev. 0
TVA, SNP, RMD FO-08, Radiological Surveys of Equipment and Materials Leaving the RCA, Rev. 0
TVA, SNP, Technical Instruction 0-TI-NUC-000-002.0, Storing Material in Spent Fuel Pool or New Fuel Vault, Rev. 13
TVA SNP, Surveillance Instruction, 0-SI-OPS-000-011.0, Containment Access Control During Modes 1-4, Rev. 33
TVA Standard Programs and Processes (SPP) - 3.1, Corrective Action Program, Rev. 13

Records and Data Reviewed

Air Sample Survey ID 43008010, U-2 Seal Table, Dated 04/30/08
LHRA Key Control Log Sheets
Radiation Work Permit (RWP) No. 08000056, Rev. 0, 714’ U2 – Rebuild Resin Fill Valve 2-62-918
RWP No. 08000801, Rev. 0, U2 Lower Ctmt – This RWP is Valid for MOVATS, Flange Removal/Replacement, Scaffold Work, Insulation Removal/Replacement & Other Work as Approved by the RP Shift Supervisor
RWP No. 08000812, Rev. 0, U-2 Seal Table – Replace Incore Detector Drive Unit “A”, “C”, “D”, and “E” and Associated Work
RWP No. 08000855, Rev. 0, U-2 Raceway – Inspection of Moisture Barrier to Include Removal of Insulation, Flashing and Associated Work
RWP No. 08001056, Rev. 0, 714’ Cation Demin Tank – Inspect 2-62-919 Reach Rod and Travel Stops, Repair/Adjust As Needed
RWP No. 08001850, Rev. 0, U-2 Lower LHRA – Pre-Outage U2C15 NOT/NOP Leak Inspection Walkdown
Sequoyah Nuclear Station Visual Survey Data System Survey Report Nos. 021307-1, Dated 02/13/07; 040408-1, Dated 04/04/08; 042108-4, Dated 04/21/08; 042108-5, Dated 04/21/08; 042808-3, Dated 04/28/08; 042808-8, Dated 04/28/08; 042908-2, Dated 04/29/08; 042908-11, Dated 04/29/08; 042908-13, Dated 04/30/08; 043008-9, Dated 04/30/08; 052905-7, Dated 05/30/05; 102107-5, Dated 10/21/07; 111107-6, Dated 11/11/07; 120406-8, Dated 12/04/06
Surveillance Task Sheet, Procedure: 0-SI-OPS-000-011.0, Containment Access Control During Modes 1-4, Dated 04/30/08

Corrective Action Program (CAP) Documents

Problem Evaluation Report (PER) No. 132165, The pre-job briefing given by Mechanical Maintenance for the change out of the Unit #1 Letdown heat exchanger was inadequate, Dated 10/18/07

PER No. 132406, Found the U-1 690' pipe chase unable to close or lock, Dated 10/23/07
TVA Nuclear Assurance – Nuclear Power Group (NPG) Wide – Radiological Protection and Control Programs – Audit Report SSA0702, Dated 02/15/08

Section 4OA1 Performance Indicator Verification

Procedures and Guidance Documents
SPP-3.4, Performance Indicator and MOR Submittal Using INPO Consolidated Data Entry, Rev. 6

Records and Data
PI Summary for 4th Quarter, 2007 through 1st Quarter, 2008
Printout of PER summaries for all dose and dose-rate ED alarms from 10/01/07 – 04/30/08

Section 4OA2: Identification and Resolution of Problems

Integrated Trend Report, dated November 1, 2007 to February 29, 2008

Section 4OA5: Other – 2515/TI-174

Technical Manuals and Vendor Information
Elastimold Separable Connectors 200 AMP Deadbreak
Rockbestos-Surprenant Cable Corp Firewall III-J Power Cable
Okonite Company Okoguard® -Ololon® Type MV-105 8kV Shielded Power Cable
Hypalon® Chlorosulfonated Polyethylene From Dupont Performance Elastomers
Stevens Roofing Systems – Stevens Hypalon® Physical Properties
PowerCommand® 3200 Series Generator Sets
SQN-VTD-P319-0050 Instruction Manual Three Phase Regulating Transformer

Drawings
CCD No: 1, 2-15E500-1 Key Diagram Station Aux Power System, Rev 25
CCD No: 1, 2-45N779-39 Wiring Diagrams 480V Shutdown Aux Power Schematic Diagrams, Sh-39
1, 2-45N756-1 Wiring Diagrams 480V Cont-Aux Bldg Vent BD 1A1-A Single Line, Sh 1, Rev 15
1, 2-45N756-2 Wiring Diagrams 480V Cont-Aux Bldg Vent BD 1A1-A Single Line, Sh 2, Rev 22
1, 2-45N756-6 Wiring Diagrams 480V Cont-Aux Bldg Vent BD 1B1-B Single Line, Sh 2, Rev 23

Procedures
0-MA-REM-000-001.0, Extended Station Blackout, Radiological Emergency Mitigating Actions Rev. 3
0-MA-ESC-317-300.1, Movement of Blackout Diesel Generators to the 5th Diesel Generator Building, Mitigating Actions, Rev. 1
0-MA-ESC-317-300.2, Connection of Blackout Diesel Generator to 5th Diesel Generator Board, Mitigating Actions, Rev. 3
0-MA-ESC-317-300.4, Connection of Blackout Diesel Generator to 6.9kV Unit Board Compartments, Mitigating Actions, Rev. 1
0-SO-268-1, Permanent Hydrogen Mitigation System, System Operating Instruction, Rev. 0002

Training Records

Training Letter STL-0739, Revision to 0-MA-REM-000-001.0, Extended Station Blackout and Sign-Off Sheet

Emergency Procedures Training Station Blackout with Loss of ERCW, Emergency Preparedness Training, Rev. 0

Training and Development Attendance Record (emergency Preparedness) Connection of BODG to 6.9kV Unit BD, Compartments and Connect BODG to 5th DG BD-Cable Runs, dated 10/23/2004

EPT- Training and Development Attendance Record (emergency Preparedness), MSS Training to Start/Run BODGs, dated 10/22/2004

Completed Surveillance Procedures and Test Records

PM 063602395 Preventive Maintenance Work Instructions 0-DG-245-0001, Rev. 1, dated 03/04/2008

PM 063602395 Preventive Maintenance Work Instructions 0-DG-245-0001, Rev. 1, dated 12/19/2007

PM 063602395 Preventive Maintenance Work Instructions 0-DG-245-0001, Rev. 1, dated 08/14/2007

Calculations

SQNETAPAC Auxiliary Power System – Extended Blackout Analysis, Rev. 17, Attachment 11.10

ETAP 5.5.6N, SQNDATA Load Feeder Tag Name and Description, Rev. Base, dated 03/10/2008

Licensing Basis Documents

Letter – TVA to NRC, March 6, 2007, Sequoyah Nuclear Plant (SQN) Units 1 and 2, and Watts Bar Nuclear Plant (WBN) Unit 1 – Enhancement of the Capability of the Containment Hydrogen Igniters, LL44 070306 001

S10 071217 8000 NRC Commitment Management, Commitment Completion Form, Dated 12/10/1999, Commitment Due Date 12/31/07

Problem Evaluation Reports (PERs) Generated as a Result of this Inspection

PER-144301 Deficiency in Extended Station Blackout Procedures

PER-144300 Discrepancy in 1 Key Diagram Station Aux Power System Drawing 1,2-15E500-1

PER-144366 Deficiency in Training Letter Process

PER-144536 Discrepancy in Maintenance and Operations Training on Backup Power Supplies