UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
SAM NUNN ATLANTA FEDERAL CENTER
61 FORSYTH STREET, SW, SUITE 23T85
ATLANTA, GEORGIA 30303-8931

January 29, 2010

Mr. R. M. Krich
Vice President, Nuclear Licensing
Tennessee Valley Authority
3R Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

SUBJECT: SEQUOYAH NUCLEAR PLANT - NRC INTEGRATED INSPECTION REPORT 05000327/2009005, 05000328/2009005 AND EMERGENCY PREPAREDNESS INSPECTION REPORT 05000327/2009501, 05000328/2009501

Dear Mr. Krich:

On December 31, 2009, 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 discussed on January 7, 2010 with Mr. Chris Church and other members of your staff and the results of the Emergency Preparedness inspection presented to Mr. T. Cleary and other members of his staff discussed on October 23, 2009.

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 and one self-revealing finding of very low safety significance (Green). One of these findings was determined to involve a violation of NRC requirements. However, because of the very low safety significance and because it is entered into your corrective action program, the NRC is treating this finding as a non-cited violation (NCV) consistent with Section VI.A 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 Sequoyah Nuclear Plant.

In addition, if you disagree with the characterization of any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region II, and the NRC Resident Inspector at the Sequoyah Nuclear Plant. The information you provide will be considered in accordance with Inspection Manual Chapter 0305.

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

Sincerely,

/RA/

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

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

cc w/Encl: (See page 3)

Enclosure: Inspection Report 05000327/2009005, 05000328/2009005 And Emergency Preparedness Inspection Report 05000327/2009501, 05000328/2009501 w/Attachment: Supplemental Information

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Letter to R. M. Krich from Eugene Guthrie dated January 29, 2010

SUBJECT: SEQUOYAH NUCLEAR PLANT - NRC INTEGRATED INSPECTION REPORT 05000327/2009005, 05000328/2009005 AND EMERGENCY PREPAREDNESS INSPECTION REPORT 05000327/2009501, 05000328/2009501

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U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos.:	50-327, 50-328
License Nos.:	DPR-77, DPR-79
Report Nos.:	05000327/2009005, 05000328/2009005 05000327/2009501, 05000328/2009501
Licensee:	Tennessee Valley Authority (TVA)
Facility:	Sequoyah Nuclear Plant, Units 1 and 2
Location:	Sequoyah Access Road Soddy-Daisy, TN 37379
Dates:	October 1, 2009 – December 31, 2009
Inspectors:	C. Young, Senior Resident Inspector M. Speck, Resident Inspector D. Simpkins, Senior Technical Training Program Specialist W. Deschaine, Project Engineer H. Gepford, Senior Health Physicist (4OA5) L. Miller, Senior Emergency Preparedness Inspector (1EP2, 1EP3, 1EP4, 1EP5, 4OA1, 4OA5) B. Collins, Reactor Inspector (1R08, 4OA5) E. Michel, Senior Reactor Inspector (1R08)
Approved by:	Eugene F. Guthrie, Chief Reactor Projects Branch 6 Division of Reactor Projects

Enclosure

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SUMMARY OF FINDINGS

IR 05000327/2009005, 05000328/2009005, 05000327/2009501, 05000328/2009501; 10/01/2009 – 12/31/2009; Sequoyah Nuclear Plant, Units 1 and 2; Refueling and Other Outage Activities and Event Followup.

The report covered a three-month period of inspection by resident inspectors and announced inspections by regional inspectors. Two Green findings, one of which was a non-cited violation (NCV), were 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: Initiating Events

- Green. A self-revealing finding was identified for an inadequate maintenance procedure which was used to perform a periodic maintenance activity to clean and inspect the bus duct associated with the 'D' common station service transformer (CSST). This resulted in the bus duct being left in a condition that allowed for water intrusion to occur, which led to a fault that caused a loss of one offsite power supply and an automatic reactor trip of both units with main feedwater unavailability. The licensee entered this issue into the corrective action program (CAP) as PER 166884.

  The finding was determined to be greater than minor because it was associated with the procedure quality attribute of the initiating events cornerstone and affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions. Specifically, the use of an inadequate procedure directly contributed to the loss of one offsite power supply and an automatic reactor trip of both units with main feedwater unavailability. Using Inspection IMC 0609, "Significance Determination Process," Attachment 4, "Phase 1 - Initial Screening and Characterization of Findings," the finding was determined to be applicable to a Phase 2 analysis since the finding contributed to both the likelihood of a reactor trip and the likelihood that mitigating systems will not be available. Using IMC 0609 Appendix A, "Determining the Significance of Reactor Inspection Findings for At-Power Situations," a Phase 2 analysis was performed using the site specific risk-informed inspection notebook. The finding was assumed to affect the initiating event likelihood (IEL) of a Transient With Loss of Power Conversion System (TPCS), since power availability to the unit boards affects reactor coolant pump function as well as main condenser availability. A regional Senior Reactor Analyst performed a Phase 3 Significance Determination Process evaluation. The evaluation concluded the finding was of very low safety significance (Green) based on an assumed unavailability of the CSST 'B' fast transfer function of 0.11/yr. No cross-cutting aspect was identified since the issue was not reflective of current licensee performance, in that the inadequate maintenance procedure was implemented in December 2006 (Section 4OA3.2).

Cornerstone: Occupational Radiation Safety

- Green. The inspectors identified a non-cited violation (NCV) of Units 1 and 2 Technical Specification 6.8, "Procedures & Programs," for the licensee's failure to follow procedures involving the review and approval of revisions to a plant abnormal operating procedure (AOP). The incorporation of manual operator actions regarding closure of the containment equipment hatch in the event of a fuel handling accident into a plant AOP without performing a mission dose evaluation resulted in the likelihood that personnel involved with the activity would receive a dose in excess of regulatory limits for occupational exposure. The licensee entered this issue into their corrective action program as PERs 167420 and 167428.

  The finding was determined to be greater than minor because it was associated with the program and process attribute of the occupational radiation safety cornerstone and affected the cornerstone objective to ensure the adequate protection of the worker health and safety from exposure to radiation from radioactive material during routine civilian nuclear reactor operation. The cornerstone objective was affected since adequate worker protection from exposure to radiation was not ensured through the AOP revision process. Using Inspection IMC 0609, "Significance Determination Process," Attachment 4, "Phase 1 - Initial Screening and Characterization of Findings," and Appendix C, "Occupational Radiation Safety Significance Determination Process," the finding was determined to be of very low safety significance (Green) because it did not affect the licensee's ability to assess dose, did not involve an overexposure or substantial potential for overexposure, and was not related to ALARA planning. No cross-cutting aspect was identified since the issue was not reflective of current licensee performance, in that the performance deficiency occurred in 2004 (Section 1R20.1).

B. Licensee-Identified Violations

None.

REPORT DETAILS

Summary of Plant Status:

Unit 1 operated at or near 100 percent rated thermal power (RTP) for the entire inspection period, with the exception of a power reduction to approximately 82 percent RTP on October 15, 2009, for repairs to #7 heater drain tank level control valve. Unit 1 returned to 100 percent RTP on October 17, 2009, and operated there for the remainder of the inspection period.

Unit 2 operated at or near 100 percent RTP until October 25, 2009, when Unit 2 was shut down for a planned refueling outage. Following the outage, Unit 2 achieved criticality on November 24, 2009. While operating at approximately 30 percent RTP on November 26, 2009, Unit 2 was manually tripped in response to indications of degrading main feedwater pump turbine condenser vacuum. Following evaluation and corrective actions for the cause of the trip, Unit 2 achieved criticality on November 27, 2009, and reached 100 percent RTP on November 29, 2009, where it operated for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection

.1 Readiness for Seasonal Extreme Weather Readiness

a. Inspection Scope

The inspectors reviewed design features and licensee preparations for protecting the essential raw cooling water (ERCW) intake structure and both Unit 1 and 2 refueling water storage tanks (RWSTs) from extreme cold and freezing conditions. The inspectors reviewed the Updated Final Safety Analysis Report (UFSAR) and Technical Specifications (TS), reviewed and observed implementation of licensee freeze protection procedures, and walked down portions of the systems to assess deficiencies and the system readiness for extreme cold weather. Documents reviewed are listed in the Attachment. This inspection satisfied one inspection sample for extreme weather readiness.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

.1 Partial System Walkdowns

a. Inspection Scope

The inspectors performed one partial walkdown of the following system to verify the operability of redundant or diverse trains and components when safety equipment was inoperable. The inspectors focused on identification of 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 determined whether 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 (CAP). Documents reviewed are listed in the Attachment.

- Unit 1 Emergency Core Cooling System (ECCS) Train B During Train A Maintenance

b. Findings

No findings of significance were identified.

1R05 Fire Protection

Fire Protection Tours

a. Inspection Scope

The inspectors conducted a tour of the seven areas important to safety, listed below, to assess the material condition and operational status of fire protection features. The inspectors evaluated whether: combustibles and ignition sources were controlled in accordance with the licensee's administrative procedures; fire detection and suppression equipment was available for use; passive fire barriers were maintained in good material condition; and 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 749 Reactor Motor-Operated Valve (MOV) and Vital Battery Rooms
- Control Building Elevation 669 (Mechanical Equipment Room, 250 Volts Direct Current (VDC) Battery and Battery Board Rooms)
- Auxiliary Building Elevation 714 (Corridor)
- Auxiliary Building Elevation 690 (Corridor)
- Control Building Elevation 706 (Cable Spreading Room)
- Control Building Elevation 732 (Mechanical Equipment Room)
- Control Building Elevation 685 (Auxiliary Instrument Rooms)

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures

.1 Internal Flooding

a. Inspection Scope

The inspectors reviewed one internal flood protection measures sample for the Reactor Building Annulus 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 to fully understand the licensee's flood mitigation strategy and then verified that the assumptions and results remained valid. The inspectors walked down the Unit 2 annulus prior to unit startup from a refueling outage to verify the assumed flooding sources, adequacy of common area drainage, and status of flood detection instrumentation to ensure that a flooding event would not impact reactor shutdown capabilities. The inspectors walked down the control room to ensure that if a break occurred, procedures existed to identify and isolate the leak. Documents reviewed are listed in the Attachment.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities (71111.08P)

From November 2, 2009 to November 13, 2009, the inspectors conducted a review of the implementation of the licensee's Inservice Inspection (ISI) Program for monitoring degradation of the reactor coolant system, steam generator tubes, emergency feedwater systems, risk-significant piping and components and containment systems.

The inspections described in Sections 1R08.1, 1R08.2, 1R08.3, 1R08.4 and 1R08.5 below constituted one inservice inspection sample as defined in Inspection Procedure 71111.08-05.

.1 Piping Systems ISI

a. Inspection Scope

The inspectors evaluated the following non-destructive examinations mandated by the ASME Code Section XI to verify compliance with the ASME Code Section XI and Section V requirements and, if any indications and defects were detected, to evaluate if they were dispositioned in accordance with the ASME Code or an NRC-approved alternative requirement.

- Ultrasonic Testing (UT) examination of weld MSS-16, ASME Class 2, Main Steam System, 32-inch diameter pipe-to-tee weld – Direct Observation.

- UT examination of RCF-23A-OL, ASME Class 1, Reactor Coolant System (Pressurizer Spray Nozzle overlay) – Document Review
- Liquid Penetrant Testing (PT) of CRDM #75 Canopy Seal Weld – Document Review

The inspectors reviewed the following examination records (volumetric or surface) with recordable indications that were analytically evaluated and accepted for continued service against the ASME Code Section XI or an NRC-approved alternative.

- Indication found during UT of SGW-E1, ASME Class 2, Steam Generator shell-to-transition cone weld
- Indication found during UT of BIT-4, ASME Class 2, Boron Injection Tank shell-to-lower head weld

The inspectors reviewed documentation for the following pressure boundary welds completed for risk-significant systems during the outage to evaluate if the licensee applied the preservice non-destructive examinations and acceptance criteria required by ASME Code Section XI. In addition, the inspectors reviewed the welding procedure specification, welder qualifications, welding material certification and supporting weld procedure qualification records, to evaluate if the weld procedures were qualified in accordance with the requirements of Construction Code and the ASME Code Section IX.

- Work Order 04-775484, replacement of valve MS-621
- Work Order 04-778294, repair of valve ERCW-589D

b. Findings

No findings of significance were identified.

.2 Reactor Pressure Vessel Upper Head Penetration Inspection Activities

a. Inspection Scope

For the Unit 2 vessel head, a bare metal visual examination was required this outage pursuant to 10 CFR 50.55a(g)(6)(ii)(D).

The inspectors reviewed records of the visual examination conducted on the Unit 2 reactor vessel head to evaluate if the activities were conducted in accordance with the requirements of ASME Code Case N-729-1 and 10 CFR 50.55a(g)(6)(ii)(D). Specifically, the inspectors reviewed the following documentation and/or observed the following activities:

- Evaluated if the required visual examination scope/coverage was achieved and limitations (if applicable) were recorded in accordance with the licensee procedures.
- Evaluated if the licensee's criteria for visual examination quality and instructions for resolving interference and masking issues were adequate.

The inspectors reviewed records of welded repairs on the upper head penetration CRDM #75 completed during the C16 outage to evaluate if the licensee applied the preservice non-destructive examinations and acceptance criteria required by the ASME Code Section XI. In addition, the inspectors reviewed the welding procedure specification and supporting weld procedure qualification records to evaluate if the weld procedure used was qualified in accordance with the Construction Code and the ASME Code Section IX requirements.

b. Findings

No findings of significance were identified.

.3 Boric Acid Corrosion Control (BACC)

a. Inspection Scope

The inspectors performed an independent walkdown of portions of the containment building which recently received a licensee boric acid walkdown and evaluated if the licensee's BACC visual examinations emphasized locations where boric acid leaks could cause degradation of safety-significant components.

The inspectors reviewed the following licensee evaluations of reactor coolant system components with boric acid deposits to evaluate if degraded components were documented in the corrective action system. The inspectors also evaluated the corrective actions for any degraded reactor coolant system components against ASME Code Section XI.

- PER 124682, 2-VLV-074-0511 Boron Leakage Evaluation
- PER 146295, 2-STBL-094-0100 Boron Leakage Evaluation
- PER 157712, SQN-1-LCV-062-178 Boron Leakage Evaluation

The inspectors reviewed the following corrective actions related to evidence of boric acid leakage to evaluate if the corrective actions completed were consistent with the requirements of the ASME Code Section XI and 10 CFR Part 50, Appendix B, Criterion XVI.

- WO 09-777018-000, Boron Leakage from RHR Pump 2A-A
- WO 08-780864-000, Boron Leakage on Root Valve to PT-68-340
- WO 09-777878-000, Boron Leakage from Centrifugal Charging Pump 2B-B

b. Findings

No findings of significance were identified.

.4 Steam Generator (SG) Tube Inspection Activities

a. Inspection Scope

The NRC inspectors interviewed eddy current testing (ET) personnel including the licensee SG engineer, vendor lead ET Level III, and vendor ET Qualified Data Analysts (QDAs); and reviewed documentation related to the SG ISI program. The following items were evaluated against the requirements of the ASME B&PV Code, Section XI; the Technical Specifications; and the guidance documents referenced in NEI 97-06, "Steam Generator Program Guidelines," Revision 2:

- Reviewed a sample of the licensee's in-situ SG tube pressure testing screening criteria. In particular, assessed whether assumed NDE flaw sizing accuracy was consistent with data from the EPRI examination technique specification sheets (ETSS) or other applicable performance demonstrations.
- Reviewed ET data from SG #4 row 21, column 83 (including historical ET data); and SG #4 row 23, column 54.
- Compared the numbers and sizes of SG tube flaws/degradation identified, against the licensee's previous outage Operational Assessment predictions.
- Reviewed the SG tube ET examination scope and expansion criteria.
- Evaluated the licensee's SG tube ET examination scope for potential areas of tube degradation identified in prior outage SG tube inspections and/or as identified in NRC generic industry operating experience applicable to the licensee's SG tubes.
- Reviewed the licensee's examination scope expansion plans and implementation following the discovery of flaws in cold leg W* sample plan.
- Reviewed the licensee's repair criteria and processes.
- Evaluated ET equipment and techniques used by the licensee to acquire data from the SG tubes for site-validation in accordance with Appendix H, Performance Demonstration for Eddy Current Examination, of EPRI Pressurized Water Reactor Steam Generator Examination Guidelines, Revision 7.
- Reviewed the licensee's secondary side SG Foreign Object Search and Removal (FOSAR) activities.
- Reviewed the licensee's disposition of irretrievable foreign objects left within the secondary side of the steam generators.
- Participated in the conference call between NRR/DCI staff and the licensee which detailed the licensee's SG tube examination activities and results.
- Verified the licensee was complying with appropriate probe wear criteria during implementation of Generic Letter 95-05.

b. Findings

No findings of significance were identified.

.5 Identification and Resolution of Problems

a. Inspection Scope

The inspectors performed a review of ISI/SG related problems entered into the licensee's corrective action program and conducted interviews with licensee staff to determine if:

- the licensee had established an appropriate threshold for identifying ISI/SG related problems;
- the licensee had performed a root cause (if applicable) and taken appropriate corrective actions; and
- the licensee had evaluated operating experience and industry generic issues related to ISI and pressure boundary integrity.

The inspectors performed these reviews to evaluate compliance with Title 10 of the Code of Federal Regulations (CFR) Part 50, Appendix B, Criterion XVI, "Corrective Action," requirements. The corrective action documents reviewed by the inspectors are listed in the Attachment to this report.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program

a. Inspection Scope

The inspectors performed one quarterly licensed operator requalification program review. The inspectors observed a simulator session on October 6, 2009. The training scenario involved a loss of a running main feedwater pump resulting in turbine runback, followed by a steam generator tube leak. While performing required actions for the leak, the leak degraded to a tube rupture requiring a manual reactor trip, initiation of safety injection, plant cooldown and depressurization, and declaration of an alert. Additional anomalies included a failed train of control room isolation, emergency core cooling train-A pumps failed to auto-start. 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 listed below to verify the effectiveness of the activities in terms of: appropriate work practices; identifying and addressing common cause failures; scoping in accordance with 10 CFR 50.65 (b); characterizing reliability issues for performance; trending key parameters for condition monitoring; charging unavailability for performance; classification in accordance with 10 CFR 50.65(a)(1) or (a)(2); appropriateness of performance criteria for structure, system, or components (SSCs) and functions classified as (a)(2); and, appropriateness of goals and corrective actions for SSCs and functions classified as (a)(1). Documents reviewed are listed in the Attachment.

- Shutdown Board Room Air Conditioning System
- Problem Evaluation Report (PER) 210886, Vital Battery V Charged at Wrong Voltage

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope

The inspectors reviewed the following activities to determine whether appropriate risk assessments were performed prior to removing equipment from service for maintenance. The inspectors evaluated whether 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 reviewed whether plant risk was promptly reassessed and managed. The inspectors also assessed whether the licensee's risk assessment tool use and risk categories were in accordance with Standard Programs and Processes Procedure (SPP)-7.1, "On-Line Work Management," Revision 12, and Instruction 0-TI-DSM-000-007.1, "Risk Assessment Guidelines," Revision 8. Documents reviewed are listed in the Attachment. This inspection satisfied two inspection samples for Maintenance Risk Assessment and Emergent Work Control.

- U2 Cycle 16 outage Reactor Coolant System (RCS) Midloop Risk Management Actions-ORAM Orange
- U1 Turbine-driven Auxiliary Feedwater Pump (TDAFP) Scheduled Maintenance

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

For the two 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 compared the operability evaluations to the UFSAR descriptions to determine if the system or component's intended function(s) were adversely impacted. In addition, the inspectors reviewed compensatory measures implemented to determine whether the compensatory measures worked as stated and the measures were adequately controlled. The inspectors also reviewed a sampling of PERs to assess whether the licensee was identifying and correcting any deficiencies associated with operability evaluations. Documents reviewed are listed in the Attachment.

- PER 204589, Missed UT Inspection on ERCW Emergency Supply Line to Unit 2 Turbine-driven AFW Pump
- PER 203991, 2A-A Diesel Generator Battery Cell Electrolyte Levels Found Above Maximum

b. Findings

No findings of significance were identified.

1R18 Plant Modifications

.1 Temporary Modifications

a. Inspection Scope

The inspectors reviewed the two temporary modifications listed below and the associated 10 CFR 50.59 screening, and compared each against the UFSAR and TS to verify that the modification did not affect operability or availability of the affected system.

- Temporary RHR Flow Control Valve Flow Restrictors Installed for Unit 2 Mid-loop Operations
- Unit 2 Mansell Reactor Vessel Level Indicator Installation, Operation, and Removal

Following installation and testing, the inspectors observed control room indications affected by the modification, discussed them with operators, and entered reactor containment to verify that the modification was installed properly and its operation did not adversely affect safety system functions. Inspectors observed the removal of the temporary systems and supporting documentation to ensure its completion. 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 three post-maintenance tests associated with the work orders (WOs) listed below to assess whether procedures and test activities ensured system operability and functional capability. The inspectors reviewed the licensee's test procedure to evaluate whether: the procedure adequately tested the safety function(s) that may have been affected by the maintenance activity; the acceptance criteria in the procedure were consistent with information in the applicable licensing basis and/or design basis documents; and the procedure had been properly reviewed and approved. The inspectors also witnessed the test or reviewed the test data to determine whether test results adequately demonstrated restoration of the affected safety function(s). Documents reviewed are listed in the Attachment.

- WO 09-777378-001, Troubleshoot/Repair 2B-B Emergency Diesel Generator (EDG) Jacket Water Temperature Switch
- WO 06-775191-000, CCP 2A Motor Replacement
- WO 09-781990-000, 2B-B EDG Immersion Heater Repairs

b. Findings

No findings of significance were identified.

1R20 Refueling and Other Outage Activities

.1 Unit 2 Refueling Outage

a. Inspection Scope

For the Unit 2 refueling outage that began on October 25, 2009, 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 2 not normally accessible during at-power operations to verify that safety-related and risk-significant SSCs were maintained in an operable condition. Specifically, between October 25, 2009 and November 24, 2009, the inspectors performed inspections and reviews of the following outage activities. Documents reviewed are listed in the Attachment. This inspection satisfied one inspection sample for Refueling Activities.

- 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 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 reactor coolant system (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 containment spray heat exchanger to verify that the equipment was 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

Introduction: The inspectors identified a Green non-cited violation (NCV) of Units 1 and 2 TS 6.8, "Procedures & Programs," for the licensee's failure to follow procedures involving the review and approval of revisions to a plant AOP. The incorporation of manual operator actions regarding closure of the containment equipment hatch in the event of a fuel handling accident into a plant AOP without performing a mission dose evaluation resulted in the likelihood that personnel involved with the activity would receive a dose in excess of regulatory limits for occupational exposure.

Description: In April 2009, during a refueling outage of Unit 1, the inspectors observed that the licensee conducted irradiated fuel movement in containment with the containment equipment hatch open. The inspectors noted that the plant TS Bases specified that a method to promptly close the containment equipment hatch during movement of irradiated fuel assemblies will be in place. This commitment was introduced into the plant's licensing basis as part of a license amendment issued on October 28, 2003, which was TS change 02-08, "Partial Scope Implementation of the Alternate Source Term and Revision of Requirements for Closure of the Containment Building Equipment Door During Movement of Irradiated Fuel." This TS change revised Limiting Condition for Operation (LCO) 3.9.4 to remove the requirement for the containment equipment hatch to be closed during movement of fuel within the containment, unless the fuel had been irradiated (i.e. part of a critical core) within the previous 100-hour period.

The change included a commitment to establish the capability to close the equipment hatch in the event of a fuel handling accident, which was reflected in the revision to the TS Bases, as noted above. The licensee implemented this commitment through a revision to procedure AOP-M.04, "Refueling Malfunctions," Revision 6, on October 25, 2004. This procedure revision incorporated actions to close the equipment hatch in response to a fuel handling accident.

The inspectors noted that licensee design basis document SQN-DC-V-21.0, "Sequoyah Nuclear Plant – Environmental Design," Revision 20, identified that a fuel handling accident is among those design basis accidents that could result in plant personnel approaching GDC-19 dose limits, and requires that a post accident mission dose analysis shall be performed where plant personnel are required to enter vital areas of the plant via a preplanned procedure to maintain the plant design basis following a fuel handling accident. The inspectors also noted that plant procedure EPM-7-1, "EOI Administrative Controls," Revision 8, required that the mission dose estimate be evaluated, prior to implementing new manual operator actions in EOPs or AOPs, for all activities required to be performed outside the control room in the event of a design basis accident as identified by SQN-DC-V-21.0. The inspectors identified that this evaluation had not been performed in conjunction with Revision 6 to AOP-M.04. Pending additional information from the licensee's evaluation of their ability to close the equipment hatch following a fuel handling accident, the inspectors opened unresolved item (URI) 050000327,328/2009003-01, "Containment Equipment Hatch Closure Capability During Fuel Handling Accident."

This issue was entered into the licensee's corrective action program as PERs 167420 and 167428. The inspectors reviewed the licensee's corrective actions, which included performing an evaluation of the mission dose involved with the closure of the equipment hatch in response to a fuel handling accident. The evaluation concluded that the mission could not be accomplished within regulatory dose limits without the use of respiratory protection, which had not been available or specified for use since the implementation of Revision 6 to AOP-M.04. Based on these results, the licensee required the containment equipment hatch and personnel airlocks to remain closed for irradiated fuel movement during the October-November 2009 Unit 2 refueling outage.

Analysis: The licensee's failure to follow requirements for evaluation of AOP revisions was a performance deficiency. This resulted in the likelihood of overexposure for personnel involved with the performance of required actions in response to a fuel handling accident. The finding was determined to be greater than minor because it was associated with the program and process attribute of the occupational radiation safety cornerstone and affected the cornerstone objective to ensure the adequate protection of the worker health and safety from exposure to radiation from radioactive material during routine civilian nuclear reactor operation. The cornerstone objective was affected since adequate worker protection from exposure to radiation was not ensured through the AOP revision process. Using Inspection IMC 0609, "Significance Determination Process," Attachment 4, "Phase 1 - Initial Screening and Characterization of Findings," and Appendix C, "Occupational Radiation Safety Significance Determination Process," the finding was determined to be of very low safety significance (Green) because it did not affect the licensee's ability to assess dose, did not involve an overexposure or substantial potential for overexposure, and was not related to as low as is reasonably achievable (ALARA) planning. No cross-cutting aspect was identified since the issue was not reflective of current licensee performance, in that the performance deficiency occurred in 2004.

Enforcement: Units 1 and 2 TS 6.8.1.a required, in part, that written procedures be established, implemented, and maintained covering the activities specified in Appendix A, “Typical Procedures for Pressurized Water Reactors and Boiling Water Reactors,” of Regulatory Guide (RG) 1.33, “Quality Assurance Program Requirements (Operations),” Revision 2, dated February 1978. RG 1.33 Appendix A Section 6.x required procedures for combating emergencies and other significant events, including irradiated fuel damage while refueling. RG 1.33 Appendix A Section 1.e required administrative procedures for procedure review and approval. Procedure EPM-7-1, “EOI Administrative Controls,” Revision 8, was a plant procedure that implemented requirements for revision and approval of plant AOPs. Specifically, Section 3.2.3.B required that the mission dose estimate be evaluated, prior to implementing new manual operator actions in EOPs or AOPs, for all activities required to be performed outside the control room in the event of a fuel handling accident. Contrary to the above, on October 25, 2004, the licensee failed to implement written procedures for procedure review and approval. Specifically, prior to effecting a change to AOP-M.04, “Refueling Malfunctions,” Revision 5, to incorporate new manual operator actions outside the control room, the licensee failed to perform the required mission dose evaluation. Because the finding was of very low safety significance and has been entered into the licensee’s CAP as PERs 167420 and 167428, this violation is being treated as an NCV, consistent with Section VI.A of the Enforcement Policy: NCV 05000327,328/2009005-01, “Failure to Evaluate Mission Dose for Manual Operator Actions Required by Plant Procedures.”

.2 Unit 2 Forced Outage

a. Inspection Scope

Following the manual trip of Unit 2 on November 26, 2009, the licensee maintained Unit 2 in Mode 3 until conditions to support restart were established on November 27, 2009. The inspectors reviewed the licensee's mode change checklists to verify that appropriate prerequisites were met prior to changing TS modes. The inspectors observed containment entry controls and reviewed Procedure 0-SI-OPS-000-011.0, “Containment Access Control During Modes 1-4,” for the associated containment entries to ensure that all items that entered containment were removed so nothing would be left that could affect performance of the containment sump. The inspectors observed portions of the plant startup including reactor criticality and power ascension. This inspection satisfied one inspection sample for Outage Activities.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

For the seven surveillance tests identified below, the inspectors assessed whether the SSCs involved in these tests satisfied the requirements described in the TS surveillance requirements, the UFSAR, applicable licensee procedures, and 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.

Routine Surveillance Tests:

- 2-SI-OPS-082-026.A, Loss of Offsite Power with SI – D/G 2A-A Test, Revision 41
- 2-SI-OPS-088-001.0, Phase A Isolation Test, Revision 17
- 2-SI-ICC-094-101.A, Channel Calibration of Incore Thermocouple Monitor System (2-T-94-101), Revision 25
- 0-SI-OPS-000-187.0, Containment Inspection, Revision 37 - (U2C16 outage)

In-Service Test:

- 2-SI-SXP-063-202.0, Safety Injection Pumps 2A-A and 2B-B Comprehensive Performance and Check Valve Test, Revision 002

Ice Condenser Test:

- 0-SI-MIN-061-109.0, Ice Condenser Intermediate and Lower Inlet Doors and Vent Curtains, Revision 5

Containment Isolation Valve Test:

- 0-SI-SLT-061-258.1, Containment Isolation Valve leak Rate Test Ice Condenser (Unit 2), Revision 6

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP2 Alert and Notification System Testing

a. Inspection Scope

The inspector evaluated the adequacy of licensee's methods for testing the alert and notification system in accordance with NRC Inspection Procedure 71114, Attachment 02, "Alert and Notification System Evaluation". The applicable planning standard 10

CFR Part 50.47(b)(5) and its related 10 CFR Part 50, Appendix E, Section IV.D requirements were used as reference criteria. The criteria contained in NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," Revision 1, was also used as a reference.

The inspector reviewed various documents which are listed in the Attachment. This inspection activity satisfied one inspection sample for the alert and notification system on a biennial basis.

b. Findings

No findings of significance were identified.

1EP3 Emergency Response Organization Augmentation

a. Inspection Scope

The inspector reviewed the licensee's Emergency Response Organization (ERO) augmentation staffing requirements and process for notifying the ERO to ensure the readiness of key staff for responding to an event and timely facility activation. The qualification records of key position ERO personnel were reviewed to ensure all ERO qualifications were current.

The inspection was conducted in accordance with NRC Inspection Procedure 71114, Attachment 03, "Emergency Response Organization Staffing and Augmentation System." The applicable planning standard, 10 CFR 50.47(b)(2) and its related 10 CFR 50, Appendix E requirements were used as reference criteria.

The inspector reviewed various documents which are listed in the Attachment to this report. This inspection activity satisfied one inspection sample for the ERO staffing and augmentation system on a biennial basis.

b. Findings

No findings of significance were identified.

1EP4 Emergency Action Level and Emergency Plan Changes

a. Inspection Scope

Since the last NRC inspection of this program area, revision 89 of the Sequoyah Nuclear Plant Emergency Plan were implemented based on the licensee's determination, in accordance with 10 CFR 50.54(q), that the changes resulted in no decrease in the effectiveness of the Plan, and that the revised Plan continued to meet the requirements of 10 CFR 50.47(b) and Appendix E to 10 CFR Part 50. The inspector conducted a sampling review of the Plan changes and implementing procedure changes made between October 1, 2008, and September 30, 2009, to evaluate for potential decreases in effectiveness of the Plan. However, this review was not documented in a Safety Evaluation Report and does not constitute formal NRC approval of the changes. Therefore, these changes remain subject to future NRC inspection in their entirety.

The inspection was conducted in accordance with NRC Inspection Procedure 71114, Attachment 04, “Emergency Action Level and Emergency Plan Changes.” The applicable planning standard (PS), 10 CFR 50.47(b)(4) and its related 10 CFR 50, Appendix E requirements were used as reference criteria.

The inspector reviewed various documents which are listed in the Attachment. This inspection activity satisfied one inspection sample for the emergency action level and emergency plan changes on an annual basis.

b. Findings

No findings of significance were identified.

1EP5 Correction of Emergency Preparedness Weaknesses and Deficiencies

a. Inspection Scope

The inspector reviewed the corrective actions identified through the Emergency Preparedness program to determine the significance of the issues and to determine if repeat problems were occurring. The facility’s self-assessments and audits were reviewed to assess the licensee’s ability to be self-critical, thus avoiding complacency and degradation of their emergency preparedness program. In addition, inspector reviewed licensee’s self-assessments and audits to assess the completeness and effectiveness of all emergency preparedness related corrective actions.

The inspection was conducted in accordance with NRC Inspection Procedure 71114, Attachment 05, “Correction of Emergency Preparedness Weaknesses.” The applicable planning standard, 10 CFR 50.47(b)(14) and its related 10 CFR 50, Appendix E requirements were used as reference criteria.

The inspector reviewed various documents which are listed in the Attachment to this report. This inspection activity satisfied one inspection sample for the correction of emergency preparedness weaknesses on a biennial basis.

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation

a. Inspection Scope

Resident inspectors evaluated the conduct of a routine licensee emergency drill on October 6, 2009, 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 42. 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. This inspection satisfied one inspection sample for Drill Evaluation.

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 six PIs listed below for the period from October 1, 2008 through September 30, 2009 for both Unit 1 and Unit 2. Definitions and guidance contained in Nuclear Energy Institute (NEI) 99-02, Regulatory Assessment Indicator Guideline, Revision 6, were used to determine the reporting basis for each data element in order to verify the accuracy of the PI data reported during that period.

Cornerstone: Mitigating Systems

- Mitigating Systems Performance Index: Emergency AC Power
- Mitigating Systems Performance Index: High Pressure Injection System
- Mitigating Systems Performance Index: Heat Removal System (AFW)
- Mitigating Systems Performance Index: Residual Heat Removal System
- Mitigating Systems Performance Index: Cooling Water System
- Safety System Functional Failures

The inspectors reviewed portions of the operations logs and raw performance indicator (PI) data developed from monthly operating reports and discussed the methods for compiling and reporting the PIs with engineering personnel. The inspectors also independently calculated selected reported values to verify their accuracy and compared graphical representations from the most recent PI report to the raw data to verify that the data was correctly reflected in the report. Specifically, for the Mitigating Systems Performance Index (MSPI), the inspectors reviewed the basis document and derivation reports to verify that the licensee was properly entering the raw data as suggested by NEI 99-02. For Safety System Functional Failures, the inspectors also reviewed LERs issued during the referenced timeframe. Documents reviewed are listed in the Attachment.

During the Emergency Preparedness Inspection, the inspector sampled licensee submittals relative to the PIs listed below for the period January 1, 2008, through June 30, 2009. To verify the accuracy of the PI data reported during that period, PI definitions and guidance contained in NEI 99-02, "Regulatory Assessment Performance Indicator Guideline", Revision 5, was used to confirm the reporting basis for each data element.

Emergency Preparedness Cornerstone

- Emergency Response Organization Drill/Exercise Performance (DEP)
- Emergency Response Organization Readiness (ERO)
- Alert and Notification System Reliability (ANS)

For the specified review period, the inspector examined data reported to the NRC, procedural guidance for reporting PI information, and records used by the licensee to identify potential PI occurrences. The inspector verified the accuracy of the DEP through review of a sample of drill and event records. The inspector reviewed selected training records to verify the accuracy of the ERO PI for personnel assigned to key positions in the ERO. The inspector verified the accuracy of the PI for ANS reliability through review of a sample of the licensee's records of periodic system tests. Licensee procedures, records, and other documents reviewed within this inspection area are listed in the Attachment.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems

.1 Daily Review

a. Inspection Scope

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 follow-up, the inspectors performed a daily screening of items entered into the licensee's CAP. This was accomplished by reviewing the description of each new PER and attending daily management review committee meetings.

b. Findings and Observations

No findings of significance were identified.

.2 Annual Sample Review of Operator Work Arounds

a. Inspection Scope

The inspectors reviewed the operator workaround (OWA) program to verify that OWAs were identified at an appropriate threshold, were entered into the CAP, and that corrective actions were appropriate and timely. Specifically, the inspectors reviewed the licensee's workaround lists and repair schedules, performed CAP word searches, conducted tours and interviewed operators and operations department support staff. Additionally, the inspectors checked for undocumented workarounds by observing operators perform rounds, reviewed operator deficiency lists, reviewed appropriate system health documents, and attended plant health committee meetings. The inspectors evaluated all workarounds for their aggregate impact. Documents reviewed are listed in the Attachment. This inspection satisfied one inspection sample for Annual Review of Operator Work Arounds.

b. Findings and Observations

There were no findings of significance identified during this review. However, the inspectors noted that plant procedure were not consistent in describing deficiencies which impact operators. Specifically OPDP-1, Conduct of Operations, Revision 15, defines such deficiencies as Operator Workarounds, Operator Burdens, and Operator Challenges while SPP-7.1, On Line Work Management, Revision 15, classifies them as Priority 1, 2, or 3 Ops Work Arounds. The inspectors also noted that Work Order 09-774792-000, fit the definition of an operator burden but was not classified as such. This WO pertains to Reactor Coolant Pump 1-4 exhibiting reduced seal leakoff flow and required operators to make smaller but more frequent RCS volume additions (50 gallon vice normal 200-400 gallons per makeup) to reduce thermal cycling of reactor coolant pump (RCP) seals. Based on the inspectors' comments, the WO classification was changed and is planned for correction during the next scheduled plant outage in November 2010. The inspectors' observations were captured in the licensee's CAP as Service Requests 108310 and 108315.

.3 Semi-Annual Trend Review

a. Inspection Scope

As required by Inspection Procedure 71152, the inspectors performed a review of the licensee's CAP 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 July through December 2009, 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 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 July 2009 through December 2009. This inspection satisfied one inspection sample for Semi-annual Trend Review.

b. Findings and Observations

No findings of significance were identified. In general, the licensee had identified trends and appropriately addressed them in their CAP. 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 having potential significance noted by the inspectors. During calendar year (CY) 2009, six unit trips occurred (three automatic trips and three manual trips), with three occurring on each unit. These events have been entered in the licensee's CAP, and corrective actions have been identified. The inspectors observed that all but one of these events contained a common causal element of inadequate procedures that contributed to each of the circumstances. The licensee entered this observation into their CAP as PER 213055.

The inspectors also noted eight examples which occurred during CY2009 of missed or inadequately performed surveillance testing. These included the following: On two occasions, periodic surveillance tests (one quarterly and one weekly) were not performed within their respective required surveillance intervals. On two occasions, SR 4.8.1.1.1.a was not performed within one hour of the inoperability of either one offsite power circuit or one EDG, as required by TS LCO 3.8.1.1.a and 3.8.1.1.b, respectively. Each of the above instances required entry into TS SR 4.0.3 pending the completion of the required testing. On one occasion, inservice testing of a safety related pump was not completed prior to its required due date. On one occasion, a quarterly surveillance test was performed and documented as being satisfactorily completed when stated acceptance criteria were not met. Also, two surveillance procedures were found to contain inappropriate testing methodology for ensuring that the respective systems were capable of meeting the required acceptance criteria. The above referenced events and conditions have been entered into the licensee's CAP as PERs 166892, 167573, 169399, 172329, 174514, 174600, 175211, and 202616, and corrective actions have been identified. The inspectors noted the possible existence of common underlying causal elements associated with these issues. The licensee entered this observation into their CAP as PER 213052 for evaluation.

4OA3 Event Follow-up

.1 (Closed) LER 05000327,328/2009-002-00, Lower Containment Gaseous Radiation Monitor Channel Inoperable, a Resultant of Leakage Detection Capabilities

On November 6, 2008, the gaseous lower containment atmosphere radioactivity monitors on both Units 1 and 2 were determined to be inoperable based on their inability to perform their safety function of detecting a reactor coolant pressure boundary leak of 1 gallon per minute in one hour due to improvements in reactor fuel quality. The licensee initiated PER 156667, declared the equipment inoperable, complied with the applicable actions of TS 3.4.6.1 which allowed up to 30 days of continued operation with compensatory actions in place, and submitted a license amendment request to change the TS. The TS amendment was issued on December 4, 2008, which removed the requirement to maintain the gaseous channel of the containment atmosphere radiation monitors as a method of RCS leakage detection.

Further details regarding this issue were discussed in NRC Inspection Report 05000327,328/2008005, in which the inspectors identified a violation of Unit 1 and 2 TS 3.4.6.1, "Leakage Detection Instrumentation." The NRC exercised enforcement discretion not to issue enforcement action for this violation in accordance with Enforcement Guidance Memorandum (EGM) 09-001, "Dispositioning Violations of NRC Requirements for Operability of Gaseous Monitors for Reactor Coolant System Leakage Detection."

The inspectors reviewed the LER and PER 156667, discussed the issue with operations, engineering, and licensee management personnel, and assessed the adequacy of the licensee's corrective actions. One violation was identified for which enforcement discretion was granted, as documented in NRC Inspection Report 05000327, 328/2008005. No additional findings of significance were identified. This LER is closed.

.2 (Closed) LER 05000327/2009-003-00 and 2009-003-01, Units 1 and 2 Reactor Trip on Reactor Coolant Pump (RCP) Buses Undervoltage

a. Inspection Scope

On March 26, 2009, following an automatic reactor trip of both Units 1 and 2 as a result of a loss of power to two reactor coolant pumps per unit, 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. The events were reported to the NRC as event notifications (EN) 44934 and 44935 and documented in the licensee corrective action program as PER 166884.

The inspectors discussed the event with operations, maintenance, engineering, and licensee management personnel to gain an understanding of the conditions leading up to the event and assess licensee actions taken following the event. Additionally, the inspectors reviewed the root cause report to assess the detail and thoroughness of the evaluation and the adequacy of the proposed corrective actions.

The inspectors reviewed the LER and PER 166884 to verify that the cause of the reactor trips was identified and whether corrective actions were appropriate. On March 26, 2009, an electrical fault developed in a bus duct associated with the 'D' common station service transformer (CSST). Protective relaying responded as designed to de-energize the common power supply that feeds CSST 'D' as well as CSST 'C,' which serves as one of the two normally aligned offsite power supplies to each unit. Because of an alternate alignment of the station's start buses that had been temporarily established, the normally available backup offsite power supply (CSST 'B') was unavailable to prevent the interruption of power to the start buses to which CSST 'C' was supplying power. The consequence was a loss of power to two (of four) unit boards (non-safety related buses) per unit, which caused a loss of power to two (of four) reactor coolant pumps on each unit, and a loss of two (of three) condenser circulating water pumps on each unit. This resulted in an automatic reactor trip of both units, with a loss of condenser vacuum.

The fault inside the CSST 'D' bus duct was the result of water intrusion, which accumulated at the location of a horizontal support plate adjacent to the six energized bus bars (two per phase) that were individually encapsulated with Noryl protective sleeves, along with a breakdown/degradation of one or more of the insulating sleeves at that location. The inspectors concluded that the licensee's corrective actions to this event were appropriate, including inspection and repairs to all similarly configured switchyard bus ducts and Noryl bus bar insulating sleeves, and revision of all corresponding bus duct preventive maintenance procedures. The inspectors also verified that timely notifications were made in accordance with 10 CFR 50.72, that licensee staff properly implemented the appropriate plant procedures, and that available plant equipment performed as required.

The inspectors noted that the licensee failed to categorize this event as an unplanned scram with complications (USwC) as indicated in section VII.E of the LER, as well as in their quarterly PI data submitted pursuant to the guidelines of NEI 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 5. The inspectors determined that the event constituted an USwC based on the loss of condenser vacuum which resulted in the unavailability of the main feedwater system to be placed in service if necessary as required by the licensee's emergency operating procedures. The licensee entered this issue into their CAP as PER 203448. The inspectors determined that the licensee's failure to report complete and accurate performance indicator data constituted a minor violation of 10 CFR 50.9 that is not subject to enforcement action in accordance with the NRC's Enforcement Policy.

On December 21, 2009, the licensee submitted revision 1 to this LER to indicate that the event constituted an unplanned scram with complications.

One finding of significance was identified, as discussed below. These LERs are closed.

b. Findings

Introduction. A Green self-revealing finding was identified for an inadequate maintenance procedure, as specified by site standard MMDP-1, Maintenance Management System, Revision 10, which was used to perform a periodic maintenance activity to clean and inspect the bus duct associated with the 'D' CSST. This resulted in the bus duct being left in a condition that allowed for water intrusion to occur, which led to a fault that caused a loss of one offsite power supply and an automatic reactor trip of both units with main feedwater unavailability.

Description. On March 26, 2009, an automatic reactor trip of both units occurred as described above. The licensee entered this issue into their CAP as PER 166884. The cause of the water intrusion into the affected bus duct was determined to be a lack of instructions for reassembly, sealing, and inspection of the as-left condition of the duct when the last preventive maintenance activity was performed, which resulted in inadequate sealing of the duct. The last cleaning and inspection maintenance activities were implemented via WOs 06-771216-000 and 06-771215-000 in December 2006 on the affected bus duct. The work instructions in these WOs did not include steps for reassembly, sealing, and inspection of the as-left condition of the duct that were contained in the previous WO for this activity in November 2000 as well as in the WOs for the last performance of similar maintenance activities on other transformer bus ducts in the switchyard.

These WOs also did not contain instructions to document what type of degradation, if any, was found during the inspection, and did not address actions to take if degradation was found. The inspectors noted that NRC Information Notice 89-64, "Electrical Bus Bar Failures," was issued to alert licensees to potential problems resulting from the failure of electrical bus bars caused by cracked and degraded Noryl insulation and moisture or debris buildup in bus bar housings. Several events were cited that involved degraded Noryl insulation, including a 1983 event at Sequoyah Unit 1. The licensee's root cause report also identified additional sources of information and operating experience that had been issued regarding the Noryl insulation degradation, such as Electric Power Research Institute (EPRI) documents and switchgear vendor information.

The inspectors noted that licensee Procedure MMDP-1, "Maintenance Management System," Revision 10, Section 3.2.3 contained guidance pertaining to the level of detail of work order content. In particular, "Work orders are to be complete and accurate… Completeness and accuracy for the content of a work order package includes the following considerations: …Level of detail of work instructions is right for the task and for the performers." Also, "The work order package should contain sufficient controls and instructions to perform the activity in a safe, quality manner without unanticipated impact on the plant and without the introduction of latent problems into the equipment." The inspectors concluded that the inadequate maintenance procedures constituted a failure to meet this site standard. Specifically, the work order instructions were incomplete in that pertinent instructions were missing from the work order package covering the tasks of duct reassembly and Noryl degradation assessment. Consequently, the activity did result in unanticipated impact on the plant.

Analysis. The licensee's failure to ensure that an adequate level of detail, as specified by site standard MMDP-1, Revision 10, Section 3.2.3, was contained in the work order instructions for performing transformer bus duct periodic maintenance was a performance deficiency. This resulted in the introduction of a latent problem by leaving the equipment in a condition that led to a partial loss of offsite power and a plant trip. The finding was determined to be greater than minor because it was associated with the procedure quality attribute of the initiating events cornerstone and affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions. Specifically, the use of an inadequate procedure directly contributed to the loss of one offsite power supply and an automatic reactor trip of both units with main feedwater unavailability. Using IMC 0609, "Significance Determination Process," Attachment 4, "Phase 1 - Initial Screening and Characterization of Findings," the finding was determined to be applicable to a Phase 2 analysis since the finding contributed to both the likelihood of a reactor trip and the likelihood that mitigating systems will not be available. Using IMC 0609 Appendix A, "Determining the

Significance of Reactor Inspection Findings for At-Power Situations," a Phase 2 analysis was performed using the site specific risk-informed inspection notebook. The finding was assumed to affect the IEL of a Transient With Loss of Power Conversion System (TPCS), since power availability to the unit boards affects reactor coolant pump function, as well as main condenser availability. A regional Senior Reactor Analyst performed a Phase 3 Significance Determination Process evaluation. The evaluation concluded the finding was of very low safety significance (Green). The major assumptions of the evaluation were that a lack of Common Station Services Transformer B fast transfer was necessary to cause an initiating event, a common cause failure of other transformers from this performance deficiency was not credible and that the rainfall necessary to cause transformer failure will happen once per the exposure period. Based upon historical information the unavailability and unreliability of the fast transfer function was approximately 0.11/yr. The dominant accident sequence which led to core damage in the risk evaluation involved the plant electrical fault caused by the performance deficiency with the fast transfer scheme out of service followed by the failure of the Emergency Diesel Generators and a failure to recover offsite power within four hours.

No cross-cutting aspect was identified since the issue was not reflective of current licensee performance, in that the inadequate maintenance procedure was implemented in December 2006.

Enforcement. Enforcement action does not apply because the performance deficiency did not involve a violation of regulatory requirements. No violation of NRC requirements was identified since the subject transformer bus duct was not a safety-related component. Because this finding has been entered into the licensee's corrective action program as PER 166884, and has very low safety significance, it is identified as FIN 05000327,328/2009005-02, "Reactor Trip due to Inadequate Transformer Bus Duct Maintenance Procedure."

.3 (Closed) Licensee Event Report (LER) 05000328/2009-001-00, Unit 2 Automatic Reactor Trip Following a Power Range Negative Rate Trip

On May 27, 2009, Unit 2 automatically tripped due to a group of 10 control rods dropping into the core as a result of a loss of the power supplies associated with a rod control system power cabinet. The loss of power coincided with a lightning strike on site. The reactor was automatically tripped, as designed, due to a negative power range flux rate trip. The event was reported to the NRC as event notification (EN) 45097 and documented in the licensee corrective action program as PER 172287.

The inspectors reviewed the LER and PER to verify that the cause of the reactor trip was identified and that corrective actions were appropriate. The cause of the event was determined to be the power supplies' susceptibility to lightning-induced overvoltage conditions because of a lack of a filter in the overvoltage protection circuit for the control rod drive system. The inspectors concluded that the licensee's corrective actions were appropriate, including revising maintenance procedures to use correct overvoltage devices in the power supplies when replacement is necessary in the future.

The inspectors discussed the trip with operations, engineering, and licensee management personnel to gain an understanding of the event and assess follow-up actions. The inspectors reviewed operator actions taken to determine whether they were in accordance with licensee procedures and TS, and reviewed unit and system indications to verify whether actions and system responses were as expected and designed. The inspectors verified that timely notifications were made in accordance with 10 CFR 50.72, that licensee staff properly implemented the appropriate plant procedures, and that plant equipment performed as required. No findings of significance were identified. This LER is closed.

.4 Unit 2 Manual Reactor Trip

a. Inspection Scope

On November 26, 2009, the inspectors responded to a manual trip of Unit 2. Operators responded to indications of degrading main feedwater pump turbine condenser vacuum by manually tripping the reactor in accordance with plant procedures. The inspectors discussed the trip with operations, engineering, and licensee management personnel to gain an understanding of the event and assess followup actions. The inspectors reviewed operator actions taken to determine whether they were in accordance with licensee procedures and TS, and reviewed unit and system indications to verify whether actions and system responses were as expected and designed. The inspectors found that operators responded to the situation appropriately and in accordance with plant procedures, and that plant systems responded to the trip as designed. The inspectors also reviewed the initial licensee notifications to verify that they met the requirements specified in NUREG-1022, "Event Reporting Guidelines." The event was reported to the NRC as EN 45520, and documented in the licensee's CAP as PERs 209482 and 209487.

b. Findings

No findings of significance were identified.

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) URI 05000327,328/2009003-01: Containment Equipment Hatch Closure Capability During Fuel Handling Accident

This unresolved item was opened and discussed in NRC Inspection Report 05000327,328/2009003 section 1R20. It involved the licensee's implementation of a commitment concerning closure of the containment equipment hatch in the case of a fuel handling accident in the containment building. The licensee entered this issue into their CAP as PERs 167420 and 167428. The disposition of this issue is discussed in section 1R20 of this report. This URI is closed.

.3 (Closed) NRC Temporary Instruction (TI) 2515/173 Review of the Implementation of the Industry Ground Water Protection Voluntary Initiative

a. Inspection Scope

The inspectors reviewed elements of the licensee's environmental monitoring program to evaluate compliance with the voluntary Groundwater Protection Initiative (GPI) as described in NEI 07-07, Industry Ground Water Protection Initiative – Final Guidance Document, August 2007 (ADAMS Accession Number ML072610036). The inspectors interviewed personnel and reviewed the following items:

- Records of the site characterization of geology and hydrology

- Evaluations of systems, structures, and or components that contain or could contain licensed material and evaluations of work practices that involved licensed material for which there is a credible mechanism for the licensed material to reach the groundwater

- Implementation of an onsite groundwater monitoring program to monitor for potential licensed radioactive leakage into groundwater

- Procedures for the decision making process for potential remediation of leaks and spills, including consideration of the long term decommissioning impacts

- Records of leaks and spills recorded, if any, in the licensee's decommissioning files in accordance with 10 CFR 50.75(g)

- Licensee briefings of local and state officials on the licensee's groundwater protection initiative

- Protocols for notification to the local and state officials, and to the NRC regarding detection of leaks and spills

- Protocols and/or procedures for thirty-day reports if an onsite groundwater sample exceeds the criteria in the radiological environmental monitoring program

- Groundwater monitoring results as reported in the annual effluent and/or environmental monitoring report

- Licensee and industry assessments of implementation of the groundwater protection initiative. (Note: the NEI audit of GPI implementation was in-progress at the time of the inspection but unavailable for NRC review).

b. Findings

No findings contrary to the requirements of NEI 07-07 were identified. This TI is closed.

.4 Reactor Coolant System Dissimilar Metal Butt Welds (TI 2515/172, Revision 1)

a. Inspection Scope

The inspectors conducted a review of the licensee's activities regarding licensee dissimilar metal butt weld (DMBW) mitigation and inspection implemented in accordance with the industry self-imposed mandatory requirements of Materials Reliability Program (MRP-139), "Primary System Piping Butt Weld Inspection and Evaluation Guidelines." Temporary Instruction (TI) 2515/172, "Reactor Coolant System Dissimilar Metal Butt Welds," was issued February 21, 2008, to support the evaluation of the licensees' implementation of MRP-139. This inspection was limited to review of MRP-139 activities performed after September 2008.

TI 2515/172 was performed in September 2008 as documented in Inspection Report 2008004. During that time a complete program review (per TI 2515/172 paragraph 03.05) was performed.

The documents reviewed by the inspector for this inspection are listed in the Attachment to this report.

From November 2 – 6, 2009, the inspectors performed a review in accordance with TI-172 as described in the Observation Section below:

b. Observations

In accordance with requirements of TI 2515/172, Revision 0, the inspectors evaluated and answered the following questions:

(1) Implementation of the MRP-139 Baseline Inspections

1. a. Have the baseline inspection been performed or are they scheduled to be performed in accordance with MRP-139 guidance?
   Yes. The licensee has performed all required baseline inspections at the time of this review.

UT exams of all Unit 2 Pressurizer overlay welds were completed during this refueling outage. For Unit 1, no credited MRP-139 follow-on exams occurred since the baseline inspections had been performed. Based on the categorization of the welds in the program, no follow-on exams were required to have been completed at the time of the inspection.

Therefore, the licensee has met the MRP-139 deadlines for baseline examinations of all welds scoped into the MRP-139 program

2. Is the licensee planning to take any deviations from the MRP-139 baseline inspection requirements of MRP-139? If so, what deviations are planned, what is the general basis for the deviation, and was the NEI-03-08 process for filing a deviation followed?

   No, the licensee has not submitted any requests for deviation from MRP-139 requirements.

(2) Volumetric Examinations

1. Were the examinations performed in accordance with the MRP-139, Section 5.1 guidelines and consistent with NRC staff relief request authorization for weld overlaid welds?

   Yes, all examinations were performed in accordance with applicable requirements.

2. Were examinations performed by qualified personnel? (Briefly describe the personnel training/qualification process used by the licensee for this activity.)

   Yes, all personnel performing the examinations were qualified under the Performance Demonstration Initiative (PDI) program.

3. Were examinations performed such that deficiencies were identified, dispositioned, and resolved?

   Yes, examinations were performed in a manner where deficiencies were identified, dispositioned and resolved.

(3) Weld Overlays

This portion of the TI was not inspected during the period of this report.

(4) Mechanical Stress Improvement (SI)

There were no stress improvement activities performed or planned by this licensee to comply with their MRP-139 commitments.

(5) Application of Weld Cladding and Inlays

This portion of the TI was not inspected during the period of this report.

(6) Inservice Inspection Program

This reporting requirement was addressed previously in inspection report 2008004; no new information was noted during this inspection.

c. Findings

No findings of significance were identified.

.5 (Closed) NRC Temporary Instruction (TI) 2515/174 Hydrogen Igniter Backup Power Verification

a. Inspection Scope

Inspection scope and results for inspection related activities for this TI are documented in Sequoyah Nuclear Plant NRC Integrated Inspection Report 05000327/2009002 and 05000328/2009002, section 4OA5.5.

b. Findings

No additional findings were identified. This TI is closed.

4OA6 Meetings

Exit Meeting Summary

On October 23, 2009, the lead inspector for the Emergency Preparedness inspection presented the inspection results to Mr. T. Cleary and other members of his staff. The licensee did not identify any material provided to the inspectors to be proprietary.

Exit meetings for the ISI and SGISI portions were conducted on November 6, 2009 and November 13, 2009 (respectively) with licensee management. The licensee did not identify any material provided to the inspectors to be proprietary.

On January 7, 2010, the resident inspectors presented the inspection results to Mr. Chris Church 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.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel

J. Barrick, ISI: RPVH & NDE Examinations and Planning
S. Bowman, Licensing Engineer
J. Brown, Site Welding Engineer
E. Camp, Maintenance Program Manager
C. Church, Site Vice President
D. Clift, Site Support Manager
J. Dvorak, Outage and Site Scheduling Manager
D. Folsum, ECT Level III
D. Foster, PI Manager
J. Furr, Nuclear Assurance Manager
A. Keyser, ISI/BACC/Repair & Replacement Program Owner
W. Kimsey, Chemistry
R. Krich, Licensing Vice President
K. Langdon, Plant Manager
T. Marshall, Maintenance and Modifications Manager
F. Mashburn, Corporate Licensing
J. McCamy, Radiation Protection Manager
W. Nurnberger, Chemistry/Environmental Manager
J. Parshall, Corporate Emergency Preparedness Manager
D. Porter, Operations Procedures
R. Proffitt, Licensing
P. Simmons, Operations Manager
R. Thompson, Emergency Preparedness Manager
C. Weber, Steam Generator Engineer
B. Wetzel, Director, Safety and Licensing
K. Wilkes, Operations Superintendent
J. Williams, Site Engineering Director
S. Young, Site Security Manager

NRC personnel:

W. Rogers, Region II, Senior Reactor Analyst
S. Lingam, Project Manager, Office of Nuclear Reactor Regulation

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000327,328/2009005-01	NCV	Failure to Evaluate Mission Dose for Manual Operator Actions Required by Plant Procedures (Section 1R20.1)
05000327,328/2009005-02	FIN	Reactor Trip due to Inadequate Transformer Bus Duct Maintenance Procedure (Section 4AO3.2)

Closed

05000327,328/2009-002-00	LER	Lower Containment Gaseous Radiation Monitor Channel Inoperable, a Resultant of Leakage Detection Capabilities (Section 4OA3.1)
05000327/2009-003-00	LER	Units 1 and 2 Reactor Trip on Reactor Coolant Pump (RCP) Buses Undervoltage (Section 4AO3.2)
05000327/2009-003-01	LER	Units 1 and 2 Reactor Trip on Reactor Coolant Pump (RCP) Buses Undervoltage (Section 4AO3.2)
05000328/2009-001-00	LER	Unit 2 Automatic Reactor Trip Following a Power Range Negative Rate Trip (Section 4OA3.3)
05000327,328/2009003-01	URI	Containment Equipment Hatch Closure Capability During Fuel Handling Accident (Section 4OA5.2)
05000390/2515/173	TI	Review of the Implementation of the Industry Ground Water Protection Voluntary Initiative (Section 4OA5.3)
05000390/2515/174	TI	Hydrogen Igniter Backup Power Verification, (Section 4OA5.5)

LIST OF DOCUMENTS REVIEWED

Section R01: Adverse Weather Protection
SPP-10.14, Freeze Protection, Revision 0
M&AI-27, Freeze Protection, Revision 12
0-PI-OPS-000-006.0, Freeze Protection, Revision 47
1-PI-EFT-234-706.0, Freeze Protection Heat Trace Functional Test, Revision 36
2-PI-EFT-234-706.0, Freeze Protection Heat Trace Functional Test, Revision 21

Section R05: Fire Protection
PFP Aux-0-749-01, Fire Protection Pre-fire Plan Auxiliary Building 749 Unit 1 Side, Revision 6
PFP Aux-0-749-02, Fire Protection Pre-fire Plan Auxiliary Building 749 Unit 2 Side, Revision 7
1,2-47W491-27, Mechanical Fire Protection, Revision 5
0-SI-FPU-026-241.R, Visual Inspection of the Fire Protection Sprinkler Systems in the Auxiliary Building, Revision 4

Section R06: Flood Protection Measures
SR 95702, Debris Found in Unit 2 Annulus Prior to Startup
WO 07-778408-000 and 08-775214-000, Test Annulus Drain Sump Level Switches
1-AR-M15-A, Service Water-CCW-HPFP 1-XA-55-15A, Revision 26
0-SO-77-10, Auxiliary Building Floor and Equipment Drain Sump, Revision 8
0-GO-14-1, Operator Rounds-Aux Bldg 1 Round, Revision 19
SPP-10.7, Housekeeping/Temporary Equipment Control, Revision 3
DCN M-02001-A, Modify Annulus Drain System for MELB Concerns
1,2-47W852-1, Mechanical Flow Diagram Floor and Equipment Drains, Revision 19
1,2-47W476-1, Mechanical Annulus Floor Drains and Embedded Piping, Revision 2
2-45N2632-10, Wiring Diagram Miscellaneous Controls Connection Diagram-SH 10, Revision 1

Section R08: Inservice Inspection Activities
Procedures
0-PI-DXI-000-116.2, ASME Section XI IWE/IWL Containment Inservice Inspection Program (CISI) Unit 1 and Unit 2, Rev. 0000
0-PI-DXX-068-100.R, Monitoring of Reactor Head Canopy Seal Welds for Leakage, Rev. 0002
0-PI-SLT-068-200.0, Reactor Building Post-Shutdown Leakage Examination, Rev. 0002
0-TI-DXX-000-097.1, Boric Acid Corrosion Control Program, Rev. 0005
MI-10.2.3, Removal and Replacement of Reactor Coolant Pump Cartridge and Number 1 Seals, Rev. 19
N-UT-76, Generic Procedure for the Ultrasonic Examination of Ferritic Pipe Welds, Rev. 0007
PDI-UT-1, Generic Procedure for the Ultrasonic Examination of Ferritic Pipe Welds, Rev. D
ANTS # SQN2C16-Bobbin, Rev 0
ANTS # SQN2C16-3-Coil+Pt, Rev 0

Corrective Action Documents
08-780864-000, Boron Leakage on Root Valve to PT-68-340, dated 20-Jan-09
09-777018-000, Boron Leakage from RHR Pump 2A-A, dated 9-Apr-08
09-777878-000, Boron Leakage from Centrifugal Charging Pump 2B-B, dated 18-Jul-09
PER 124682, 2-VLV-074-0511 Boron Leakage Evaluation, dated 05/25/07
PER 146295, 2-STBL-094-0100 Boron Leakage Evaluation, dated 06/04/08
PER 157712, SQN-1-LCV-062-178 Boron Leakage Evaluation, dated 5/26/09

PER 176254, Missed UT Exam on ASME Section XI Category B-B Weld, dated 7/14/2009
PER 204204, Unit 1 Pressurizer Spray Nozzle Shrinkage Analysis, dated 10/10/2009
SR 73834, VT-2 Exam Inadvertently Deleted and Subsequently Missed during U1C16, dated Sep 29, 2009
PER 207505, Steam Generator 4 eddy current tube support indication of 6.55 volt, 11/8/2009
PER 203445, Potential primary to secondary leakage, 10/1/2009

Other
CISI-2000-C-59, Drawing: Sequoyah Nuclear Plant Unit 2, Steel Containment Penetration Details, Rev. 03
DWPS GT11-0-1-N, Detailed Welding Procedure Specification for 2-VLV-067-594D Replacement, Rev. 2
GT11-0-1-N, Detailed Welding Procedure Specification – Manual GTAW, Rev. 2
ISI-0394-C-01, Drawing: Sequoyah Nuclear Plant Unit 1, Pressurizer, Rev. 10
ISI-0396-C-01, Drawing: Sequoyah Nuclear Plant Unit 2, Pressurizer, Rev. 11
PQR 01-01-S-001, Procedure Qualification Record, Rev. 0
PQR 01-01-T-001, Procedure Qualification Record, Rev. 0
PQR 01-01-T-002, Procedure Qualification Record, Rev. 0
PQR 01-01-TS-001, Procedure Qualification Record, Rev. 0
R-0025, Examination Summary and Resolution Data Sheet: UT on BIT-4, dated 4/24/08
R-0087, Examination Summary and Resolution Data Sheet: UT on RCF-23A-OL, dated 11/4/09
R-0088, Examination Summary and Resolution Data Sheet: UT on RCF-24H-OL, dated 11/4/09
R-0089, Examination Summary and Resolution Data Sheet: UT on RCF-36A-OL, dated 11/4/09
R-0090, Examination Summary and Resolution Data Sheet: UT on RCF-42A-OL, dated 11/4/09
R-0091, Examination Summary and Resolution Data Sheet: UT on RCF-45A-OL, dated 11/4/09
R-0114, Examination Summary and Resolution Data Sheet: UT on RC-35A-OL, dated 11/5/09
R-0150, Examination Summary and Resolution Data Sheet: UT on SGW-E1, dated 5/24/08
R-3591, Record of Liquid Penetrant Exam, dated 10/30/2009
WPQ11243, Weld Procedure Qualification – Manual GTAW (Barnes), dated 10/16/2009
WPQ11245, Weld Procedure Qualification – Manual GTAW (Borden), dated 10/16/2009
WPQ11246, Weld Procedure Qualification – Manual GTAW (Holden), dated 10/16/2009
WPQ11247, Weld Procedure Qualification – Manual GTAW (Haddock), dated 10/16/2009
WPS-01-01-TS-200, Welding Procedure Specification – Manual GTAW/SMAW, Rev. 3
Letter from TVA to NRC, Subject: Sequoyah Nuclear Plant (SQN) – Unit 2 – Steam Generator Tube Inspection Information, Response to Request for Additional Information (RAI) (TAC No. MD9595), 4/14/2009
Sequoyah Nuclear Power Plant Unit 2 Use of Appendix H and Appendix I Qualified Techniques U2C16 Refueling Outage, October 2009
Sequoyah Nuclear Plant Unit 2 Cycle 16 Degradation Assessment, Rev 0
Letter from NRC to TVA, Subject: Sequoyah Nuclear Plant, Unit 2 – Issuance of Amendment to Allow Use of the W* Alternative Repair Criteria for Steam Generator Tubes (TS-09-02) (TAC No. ME1343), 10/19/2009
Letter from NRC to TVA, Subject: Sequoyah Nuclear Plant, Unit 2 – Summary of Conference Call Regarding Steam Generator Tube Inspections (TAC No. MD8723), 6/18/2008
Letter from TVA to NRC, Response to Request for Additional Information Concerning – Technical Specifications Change 09-02 – W-Star Alternative Repair Criteria for Steam Generator Tubes Cold Leg (TAC No. ME1343), 9/29/2009
QR-08-48, Westinghouse Quality Release & Certificate of Conformance, Plug, ASME, Mech,

Mod. 44/51, I-690, 9/25/2009
SGMS 2.2.1 GEN-011, Appendix 12.3, Typical Customer Approval of Base Scope Inspection Plans, Sequoyah Unit 2, Outage S2C16, 11/01/2009
Sequoyah Nuclear Plant Unit 2 Cycle 15 Refueling Outage May-2008 Steam Generator Operational Assessment Report, Rev 1
Letter from TVA to NRC, Subject: Sequoyah Nuclear Plant (SQN) – Unit 2 – Unit 2 Cycle 15 (U2C15) 90-Day Steam Generator (S/G) Report for Voltage-Based Alternative Repair Criteria and W* Alternative Repair Criteria, 8/27/2008
Letter from NEI to NRC regarding probe wear criteria while implementing GL 95-05
Video of Unit 2 SG FORSAR activities
ZETEC Drawing 10007478, ASME/AVB Calibration Standard, Frame C, Sheet 1, 1/19/2000
ZETEC Test Certificate, Heat 830094, 1-1/4" Dia. Round, ASTM A-108 (90A), 8/18/1994
ABB Field Quality Operations, Certification of Eddy Current ASME/AVB Calibration Standards for TVA Sequoyah Unit 1 & 2 Contract 99NAN-251785, 2/25/2000
CoreStar Certificate of Conformance, Westinghouse Purchase Order 4500305830 (CoreStar W.O. # 7947), 8/10/2009
TVA Training Attendance Record, Eddy Current Demonstration and Owner Experience, Sequoyah Unit 2 Cycle 16, November 2009
Westinghouse Site-Specific Evaluation TVA Sequoyah Unit 2 MRPC +Point Equivalency, Validation of 300 KHz for Detection of Flaws in 0.050" Wall I-600 Material

Section R11: Licensed Operator Requalification
E-0, Reactor Trip or Safety Injection, Revision 30
E-3, Steam Generator Tube Rupture, Revision 17
ES-1.1, SI Termination, Revision 10
FR-P.1, Pressurized Thermal Shock, Revision 14
EPIP-1, Emergency Plan Classification Matrix, Revision 42

Section R12: Maintenance Rule Implementation
SPP-6.6, Maintenance Rule Performance Indicator Monitoring, Trending and Reporting – 10CFR50.65, Revision 9
TI-4, Maintenance Rule Performance Indicator Monitoring, Trending and Reporting – 10CFR50.65, Revision 22
0-PI-EBM-000-001.2, Battery Bank High Level Equalize Charge Systems 82, 244, 250, Revision 18 performance dated 9/18/09
0-PI-EBM-000-001.2, Battery Bank High Level Equalize Charge Systems 82, 244, 250, Revision 19

Section R13: Maintenance Risk Assessments and Emergent Work Evaluation
SPP-7.1, On Line Work Management, Revision 15
SPP-7.3, Work Activity Risk Management Process, Revision 4
Unit 2 Cycle 16 Refueling Outage Safety Plan
Sentinel Risk Evaluations dated 12/16, 12/17, 12/18/2009

Section R15: Operability Evaluations
ER 151905, ERCW Piping Below T-min
FE 42946, Evaluation of ERCW Piping Less Than T-min

NRC Inspection Manual, Part 9900 Technical Guidance, Operability Determinations and Functionality Assessments for Resolution of Degraded or Nonconforming Conditions Adverse to Quality or Safety
0-SI-EBT-082-238.2, Diesel Generator Quarterly Operability, Revision 16 performances dated October 7, 2009 and September 18, 2009
SI-238.1, Diesel Generator Battery System Weekly Inspection, Revision 31 performances dated September 20 and 28, 2009 and October 5, 2009

Section R18: Plant Modifications
0-GO-13, Reactor Coolant System Drain and Fill Operations, Revision 62
2-PI-IXX-068-005.0, Installation and Removal of the Mansell Level Monitoring System During Refueling Outages, Revision 12
EDC E21005, Install Mansell LMS Hardware, Revision A
Engineering Evaluation for Mansell Power Supply Changes, RIMS #B85070712003
1,2-47W813-1, Flow Diagram Reactor Cooling System, Revision 53
SPP-9.5, Temporary Alterations, Revision 9
WO 07-775481-000, SIS Flow to RCS 2&3 Cold Legs 2-F-063-0091B Periodic Calibration performance dated 7/16/08
Setpoint and Scaling Document, 2-F-63-91B, Revision 2
RIMS B38-900323-802, Engineering Review – RHR Flow Rates During Mid-loop Operation dated March 20, 1990
0-SO-74-1, Residual Heat Removal System, Revision 69
RIMS B88-900907-001, Safety Evaluation-RHR Flow Reduction, dated August 30, 1990

Section R19: Post Maintenance Testing
SPP-6.5, Foreign Material Control, Revision 14
0-PI-ICC-082-02B.N, Periodic Calibration of the Standby Diesel Generator 2B-B (non-outage), Revision 4
Setpoint and Scaling Document 0-T-82-5007/4, Diesel Generator ENG-2B-2 Immersion Heater Control, Revision 4
WO 07-778119-002, CCP 2B-B Discharge Check Valve
2-SI-SXP-062-203.0, Centrifugal Charging Pumps 2A-A and 2B-b Comprehensive Pump Test and Check Valve Test, Rev. 0001
TVA 40897, NPG Pre-Job Briefing Checklist, Rev. 04-2009
SPP-8.1, Conduct of Testing, Rev. 0006
2-SI-SXP-062-201.A, Centrifugal Charging Pump 2A-A Performance Test, Rev. 0012
2-SI-SXP-062-201.B, Centrifugal Charging Pump 2B-B Performance Test. Rev. 0015

Section R20: Refueling and Outage Activities
SPP-10.2, TVA Form 40832[06-2003]: Clearance Coversheet for Tagout #:2-TO-2009-0024; Clearance #: 2-68-1608-RFO, Date: 10/29/2009
0-MI-MXX-061-003.0, Ice Condenser Maintenance Inspections, Revision 14
0-SI-OPS-000-187.0, Containment Inspection, Revision 37
0-RT-NUC-000.003.0, Low Power Physics Testing, Revision 21
0-GO-7, Unit Shutdown From Hot Standby to Cold Shutdown, Revision 59
0-GO-1, Unit Startup From Cold Shutdown to Hot Standby, Revision 54
0-SI-OPS-068-137.0, Reactor Coolant System Water Inventory, Revision 22
Calculation SQS20133, Midloop Design Information Calculation, Revision 6

Section R22: Surveillance Testing
0-SI-SLT-061-258, CIV LLRT Ice Condenser, Rev. 6
SPP-8.1, Conduct of Testing, Rev. 16
WO: 08-778637-000/2-VLV-061-0692
WO: 09-779786-000/2-VLV-061-0745
PER: 206112, U2C16 RFO LLRT Component Failures
2-SI-OPS-088-001.0, Phase A Isolation Test, Rev. 17
0-SI-SLT-061-258.1, Containment Isolation Valve leak Rate Test Ice Condenser, Rev. 6
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