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

July 23, 2009

Mr. Preston D. Swafford
Chief Nuclear Officer and
Executive Vice President
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

SUBJECT: SEQUOYAH NUCLEAR PLANT - NRC INTEGRATED INSPECTION REPORT 05000327/2009003 AND 05000328/2009003

Dear Mr. Swafford:

On June 30, 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, which were discussed on July 1, 2009 with Mr. Timothy Cleary and other members of your staff.

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

This report documents one self-revealing finding of very low safety significance (Green). This finding 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.1 of the NRC Enforcement Policy. If you contest any NCV in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN.: Document Control Desk, Washington DC 20555-0001; with copies to the Regional Administrator, Region II; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at Sequoyah Nuclear Plant.

Additionally, 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

Enclosure: Inspection Report 05000327/2009003 and 05000328/2009003
w/Attachment: Supplemental Information

cc: w/encl: (See page 3)

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Letter to Preston D. Swafford from Eugene F. Guthrie dated July 23, 2009

SUBJECT: SEQUOYAH NUCLEAR PLANT - NRC INTEGRATED INSPECTION REPORT 05000327/2009003 AND 05000328/2009003

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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.:	05000327/2009003 and 05000328/2009003
Licensee:	Tennessee Valley Authority (TVA)
Facility:	Sequoyah Nuclear Plant, Units 1 and 2
Location:	Sequoyah Access Road Soddy-Daisy, TN 37379
Dates:	April 1, 2009 – June 30, 2009
Inspectors:	C. Young, Senior Resident Inspector M. Speck, Resident Inspector P. Higgins, Project Engineer E. Michel, Sr. Reactor Inspector (1R08, 40A5) R. Hamilton, Sr. Health Physicist (2OS2, 4OA1, 4OA7) D. Forbes, Sr. Health Physicist (2OS1, 4OA1) A. Nielsen, Health Physicist (2PS2)
Approved by:	Eugene F. Guthrie, Chief Reactor Projects Branch 6 Division of Reactor Projects

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SUMMARY OF FINDINGS

IR 05000327/2009-003, 05000328/2009-003; 04/01/2009 – 06/30/2009; Sequoyah Nuclear Plant, Units 1 and 2; Event Followup.

The report covered a three-month period of inspection by resident inspectors and announced inspections by regional inspectors. One Green finding, which was a non-cited violation (NCV), was identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Initiating Events

Green. A self-revealing NCV of Unit 1 Technical Specification 6.8, "Procedures & Programs," was identified for the licensee's failure to maintain adequate procedures for plant startup and power operation to support an as designed plant response following a turbine trip. This resulted in an automatic isolation of intermediate pressure feedwater heaters, which caused a loss of condensate flow and a reactor trip. This issue was entered into the licensee's corrective action program as Problem Evaluation Report (PER) 169976. The licensee revised the applicable procedures to address the inadequacies.

The finding was 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. 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 pre-solved table derived from the site specific risk-informed inspection notebook. Based on the results of the Phase 2 analysis, the finding was determined to have very low safety significance (Green). No cross-cutting aspect was identified since the issue was not reflective of current licensee performance, in that the failure to maintain plant operating procedures to appropriately address heater drain operations occurred following the identification of the issue in 1998, and the procedure inadequacies were promptly identified and corrected by the licensee following the April 2009 event (Section 4OA3.1).

B. Licensee-Identified Violations

None.

REPORT DETAILS

Summary of Plant Status:

Unit 1 began the inspection period in a planned refueling outage for operating cycle 16. Following the outage, Unit 1 achieved criticality on April 27, 2009, and entered Mode 1 on April 28, 2009. While operating at approximately 18 percent rated thermal power (RTP) on April 28, 2009, Unit 1 was manually tripped due to a loss of condensate flow which occurred following a main generator turbine trip. Following repairs, Unit 1 achieved criticality on May 1, 2009, and reached 100 percent RTP on May 5, 2009. On May 6, 2009, Unit 1 was manually tripped due to a loss of feedwater to Loop 1 steam generator caused by a failure of Loop 1 feedwater regulating valve. Following repairs to the valve, Unit 1 achieved criticality on May 8, 2009, and reached 100 percent RTP on May 10, 2009, where it operated for the remainder of the inspection period.

Unit 2 operated at or near 100 percent RTP until May 27, 2009, when Unit 2 was automatically tripped in response to a loss of power that occurred on one rod control power cabinet, which resulted in 10 control rods dropping into the core. Following evaluation of the cause of the trip, Unit 2 achieved criticality on May 30, 2009, and reached 100 percent RTP on June 3, 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

a. Inspection Scope

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

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

.1 Partial System Walkdowns

a. Inspection Scope

The inspectors performed a partial walkdown of the following three systems 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 to this report.

- Unit 2 emergency core cooling systems (ECCS) train A during unplanned train B unavailability
- Spent fuel pool cooling train A and B following unit 1 core off-load
- Unit 2 motor-driven auxiliary feedwater trains A and B during turbine-driven auxiliary feedwater pump unavailability

b. Findings

No findings of significance were identified.

1R05 Fire Protection

Quarterly Fire Protection Inspection

a. Inspection Scope

The inspectors conducted a tour of the six areas listed below to assess the material condition and operational status of fire protection features. The inspectors 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 to this report.

- Control building elevation 706 (cable spreading room)
- Control building elevation 685 (auxiliary instrument rooms)

- Auxiliary building elevation 734 6.9kV and 480V shutdown board rooms
- Control building elevation 669 (mechanical equipment room, 250 VDC battery and battery board rooms)
- Auxiliary building elevation 714 (corridor)
- Essential raw cooling water (ERCW) building

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection (ISI) Activities (71111.08P)

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

a. Inspection Scope

From April 6 to April 10, 2009, the inspectors reviewed the implementation of the licensee's In-service Inspection (ISI) program for monitoring degradation of the reactor coolant system (RCS) boundary and risk significant piping boundaries of Unit 1. The inspectors' activities consisted of an on-site review of NDE and welding activities to evaluate compliance with the applicable edition of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC), Section XI (Code of record: 2001 Edition with 2003 Addenda), and to verify that indications and defects (if present) were appropriately evaluated and dispositioned in accordance with the requirements of the ASME Code, Section XI acceptance standards.

The inspectors observed and/or reviewed portions of the NDE activities listed below. The review included examination procedures, NDE reports, video of the inspection, equipment and consumables certification records, personnel qualification records, and calibration reports (as applicable).

- Ultrasonic Testing (UT) of ASME Class 2 Steam Generator "B" tubesheet-to-shell weld, RSGW-B2
- Visual Exams (VE – as per ASME Code Case N-722) of ASME Class 1 Reactor Pressure Vessel Bottom Mounted Instrumentation

The inspectors observed portions of the welding activity listed below in order to evaluate compliance with Sections II and IX of the ASME Code, and the licensee's Repair and Replacement Program and welding procedures. The inspectors also reviewed the work order, repair and replacement plan, weld data sheets, detailed welding procedure, welder qualification records, and conducted interviews with plant personnel.

- Work Order 09-773437-000 – Replace section of piping downstream of Check Valve 1-63-587, ASME Class 1

The inspectors reviewed aspects of containment sump construction for compliance with ASME B&PV Code section IWE-2000 inspection requirements.

The inspectors reviewed portions of the Flow Accelerated Corrosion Program for compliance with appropriate industry guidance.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

No volumetric/surface or bare metal visual (BMV) inspections were planned this outage. The inspectors reviewed the licensee's scheduled activities for compliance with the requirements of 10 CFR 50.55a and ASME Code Case N-729-1, and reviewed available documentation documenting completion of a VT-2 exam completed per the requirements of N-729-1 footnote (4).

b. Findings

No findings of significance were identified.

.3 Boric Acid Corrosion Control (BACC) Inspection Activities

a. Inspection Scope

The inspectors reviewed the licensee's BACC program activities to ensure implementation with commitments made in response to NRC Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary," and applicable industry guidance documents. Specifically, the inspectors performed an on-site record review of procedures and the results of the licensee's containment walk-down inspections performed during the Unit 1 Spring 2009 outage. The inspectors also interviewed the BACC program owner, conducted an independent walk-down of containment to evaluate compliance with licensee's BACC program requirements, and verified that degraded or non-conforming conditions, such as boric acid leaks, were properly identified and corrected in accordance with the licensee's BACC and corrective action programs.

The inspectors reviewed an engineering evaluation completed for evidence of boric acid found on systems containing borated water to verify compliance with generally accepted industry guidance.

- PER 167216

b. Findings

No findings of significance were identified.

.4 Steam Generator (SG) Tube Inspection Activities

a. Inspection Scope

No SG eddy current testing (ECT) or secondary side visual exams were scheduled for this outage. The inspectors reviewed the licensee's "Degradation Assessment and Technical Review and Justification for Not Performing Primary or Secondary Inspections of the Steam Generators SQN Unit 1 Cycle 16 Outage," Revision 0 for compliance with the EPRI Pressurized Water Reactor Steam Generator Examination Guidelines, Rev.7.

b. Findings

No findings of significance were identified.

.5 Identification and Resolution of Problems

a. Inspection Scope

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

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program

a. Inspection Scope

The inspectors performed one licensed operator requalification program review. The inspectors observed a simulator session on June 1, 2009. The training scenario involved a condensate system failure resulting in turbine runback, followed by a steam generator tube leak. While performing a rapid shutdown, the leak degraded to a tube rupture requiring a manual reactor trip, initiation of safety injection, plant cooldown and depressurization. Additional anomalies included a failed emergency diesel generator auto-start, turbine-driven auxiliary feedwater (AFW) pump failure, a faulty reactor trip switch and subsequent steam dump failure requiring operators to close main steam isolation valves (MSIV). 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 Technical Specification (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 to this report.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness

a. Inspection Scope

The inspectors reviewed the maintenance activity below to verify the effectiveness of the activity 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 structures, systems, and 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 to this report.

- Unit 1&2 62H line Foxboro controllers maintenance rule (a)(1) corrective action plan

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope

The inspectors reviewed the following five 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 to this report.

- Unit 2 – Safety injection pump B unplanned unavailability
- Unit 1 – Component cooling system pump (CCP), C-S, check valve test

- Unit 2 – Planned auxiliary feedwater maintenance and testing week of May 4-5, 2009
- Emergency diesel generator maintenance extending beyond 50 percent of allowed outage time
- Unit 1 turbine-driven auxiliary feedwater pump unavailability for valve testing

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

For the six 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 updated final safety analysis report (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 to this report.

- PER 165645, Gas accumulation in 1A surveillance instruction (SI) pump discharge piping
- PER 168390, Unit 2 train B safety injection pump recirculation valve
- PER 168601, CCP 1A-A failed to meet acceptance criteria during comprehensive pump test
- PER 162711, Hydrology model review
- PER 169526, ERCW missile shield bolts
- PER 172329, ABSCE boundary surveillance not properly performed

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing

a. Inspection Scope

The inspectors reviewed the eight 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 to this report.

- WO 09-773119-000, Recalibrate emergency diesel generator (EDG) 1B-B EDG Relay 46
- WO 09-773426-000, U2 emergency raw cooling water (ERCW) piping leak repair
- WO 09-774966-000 Replace eagle partial trip card in rack 2-R-5
- WO 09-776531-000, Special performance of EQ maintenance and inspection 2-MVOP-003-179A
- WO 09-776537-002, Replace control air diaphragm unit 1 loop 1 main feedwater regulating valve 1-FCV-3-35
- WO 09-776754-000, Troubleshoot/repair high vibrations on 1-CLR-030-186, elevation 669' penetration room cooler 1A-A
- WO 08-773684-003, Rebuild Unit 1 loop 4 main steam isolation valve
- WO 09-773202-001, Unit 2 train B safety injection pump control circuit repairs

b. Findings

No findings of significance were identified.

1R20 Refueling and Other Outage Activities

.1 Unit 1 Refueling Outage

a. Inspection Scope

For the Unit 1 refueling outage that began on March 26, 2009, the licensee placed Unit 1 in Mode 5 and commenced a scheduled refueling outage.

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 1 not normally accessible during at-power operations to verify that safety-related and risk-significant SSCs were maintained in an operable condition. Specifically, between April 1 and April 28, 2009, the inspectors performed inspections and reviews of the following outage activities. Documents reviewed are listed in the Attachment.

- Outage Plan. The inspectors reviewed the outage safety plan and contingency plans to confirm that the licensee had appropriately considered risk, industry experience, and previous site-specific problems in developing and implementing a plan that assured maintenance of defense-in-depth.

- Reactor Shutdown. Following the automatic reactor trip from full power, the inspectors observed the operators' actions in the control room to maintain the unit in stable conditions in Mode 3. The inspectors also observed the cooldown of the reactor into Modes 4 and 5, including placing it on the residual heat removal (RHR) system for decay heat removal, to verify that TS cooldown restrictions were followed and stable conditions were established. 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 ECCS 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. To ensure that RCS level instrumentation was properly installed and configured to give accurate information, the inspectors reviewed the installation of the outage 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 outage level monitoring system 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 reactor vessel head removal 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.

- 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 an unresolved item (URI) involving 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. This issue is unresolved pending further NRC inspection and review of additional information to be provided by the licensee.

Description. On April 1, 2009, during a refueling outage of Unit 1, the inspectors performed a routine inspection of the licensee's ability to close the containment equipment hatch should residual heat removal (RHR) cooling be lost while the reactor coolant system (RCS) is open to the containment atmosphere. The inspectors noted that the licensee had performed an analysis of containment environmental conditions, following the loss of RHR, to determine how much time was available to close the hatch prior to conditions within the containment becoming so harsh as to potentially prohibit hatch closure. The licensee conducted a drill to ensure that personnel could be mobilized and the hatch could be closed within analyzed time limits.

Upon review of the analysis, the inspectors noted that it covered only the condition of loss of RHR. Further inspection revealed that the licensee also intended to leave the equipment hatch open during fuel movement in the containment building, and that the plant Technical Specification (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 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. This commitment was reflected in the revision to the TS Bases, as noted above, and was implemented through a revision to the licensee's procedure AOP-M.04, "Refueling Malfunctions," revision 6, on October 25, 2004.

The inspectors requested that the licensee provide a copy of the analysis which determined that the environmental conditions which would be present within the containment following the design basis fuel handling accident would not prohibit plant personnel from closing the hatch in accordance with the commitment reflected in the TS Bases. The licensee was unable to provide such an analysis. 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 requested the mission dose calculation for hatch closure following a design basis fuel handling accident. The licensee was unable to provide such a calculation. It was identified that this evaluation had not been performed in conjunction with revision 6 to AOP-M.04.

These issues were entered into the licensee's corrective action program as PERs 167420 and 167428. Pending additional information from the licensee's evaluation of their ability to close the equipment hatch following a fuel handling accident, this item is identified as URI 050000327,328/2009003-01, "Containment Equipment Hatch Closure Capability During Fuel Handling Accident."

.2 Unit 1 Reactor Trip

a. Inspection Scope

Following the manual trip of Unit 1 on April 28, 2009, the licensee maintained Unit 1 in Mode 3 until conditions to support restart were established on May 1, 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.

b. Findings

No findings of significance were identified.

.3 Unit 1 Reactor Trip

a. Inspection Scope

Following the manual trip of Unit 1 on May 6, 2009, the licensee maintained Unit 1 in Mode 3 until conditions to support restart were established on May 8, 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.

b. Findings

No findings of significance were identified.

.4 Unit 2 Reactor Trip

a. Inspection Scope

Following the automatic trip of Unit 2 on May 27, 2009, the licensee maintained Unit 2 in Mode 3 until conditions to support restart were established on May 30, 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.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

For the eight 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 to the report.

Reactor Coolant System Leak Detection Tests:

- 0-SI-OPS-068-137.0, Reactor coolant system water inventory, Revision 22 – Unit 1

In-Service Tests:

- 1A/B CCP Comprehensive flow test - 1-SI-SXP-062-203.0
- 1-SI-SXV-063-204.0, Safety injection/residual heat removal hot leg primary check valve integrity test, Revision 7

Routine Surveillance Tests:

- 1-SI-ICC-077-410.0, Revision 8 and 1-SI-ICC-077-411.0, Revision 7, Channel calibration of reactor building auxiliary floor and equipment drain sump level (1-L-77-410/-411)
- 1-SI-OPS-088-014.0, Verification of containment integrity, Revisions 22 and 23
- 1-SI-OPS-000-009.0, Actuation of ECCS and boron injection flowpath valves via SI signal, Revision 2
- 0-SI-SFT-030-149.B, Auxiliary building gas treatment system vacuum test train B, Revision 16/ 0-SI-SFT-030-149.A, Auxiliary building gas treatment system vacuum test train A, Revision 16

Ice Condenser System Tests:

- 1-SI-MIN-061-105.0, Ice condenser-ice weighing, Revision 5

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 May 12, 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 and technical support center to verify that event classification and notifications were done in accordance with EPIP-1, Emergency Plan Classification Matrix, Revision 41. The inspectors also attended the licensee critique of the drill to compare any inspector-observed weakness with those identified by the licensee in order to verify whether the licensee was properly identifying failures.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstones: Occupational (OS) and Public Radiation Safety (PS)

2OS1

1. Access Control to Radiologically Significant Areas

a. Inspection Scope

Access Controls The inspectors evaluated licensee performance in controlling worker access to radiologically significant areas and monitoring jobs in-progress associated with the Unit 1 Cycle 16 Refueling Outage (U1C16 RFO). The inspectors directly observed implementation of administrative and physical radiological controls; evaluated radiation worker (radworker) and health physics technician (HPT) knowledge of and proficiency in implementing radiation protection requirements; and assessed worker exposures to radiation and radioactive material.

During facility tours, the inspectors directly observed postings and physical controls for radiation areas, high radiation areas (HRAs), and potential airborne radioactivity areas established within the radiologically controlled area (RCA) of the Unit 1 (U1) Reactor Containment Building, U1 and Unit 2 (U2) auxiliary buildings, U1 and U2 turbine buildings radioactive waste (radwaste) processing and storage locations, and the Independent Spent Fuel Storage Installation (ISFSI) dry cask storage location facility. The inspectors independently measured radiation dose rates or directly observed conduct of licensee radiation surveys for selected RCA areas. Results were compared to current licensee surveys and assessed against established postings and Radiation Work Permit (RWP) controls. Licensee key control and access barrier effectiveness were evaluated for selected U1 and U2 Locked High Radiation Area (LHRA) and Very High Radiation Area (VHRA) locations. Changes to procedural guidance for LHRA and VHRA controls were discussed with the Radiation Protection Manager (RPM) and health physics (HP) supervisors. Controls and their implementation for storage of irradiated material within the spent fuel pool (SFP) were reviewed and discussed in detail. Established radiological controls were evaluated for selected U1C16 RFO tasks including fuel movement, reactor coolant pump replacement, reactor head lift, removal and installation of insulation and lead, and entries into the keyway for purposes of inspection. In addition, licensee controls for areas where dose rates could change significantly as a result of plant shutdown and refueling operations were reviewed and discussed.

For selected tasks, the inspectors attended pre-job briefings and reviewed RWP details to assess communication of radiological control requirements to workers. Occupational workers' adherence to selected RWPs and HPT proficiency in providing job coverage were evaluated through direct observations and interviews with licensee staff. Electronic dosimeter (ED) alarm set points and worker stay times were evaluated against area radiation survey results for U1 containment and refueling floor activities observed.

The inspectors evaluated the effectiveness of radiation exposure controls, including air sampling, barrier integrity, engineering controls, and postings through a review of both internal and external exposure results. Worker exposure as measured by ED and by licensee evaluations of skin doses resulting from discrete radioactive particle or dispersed skin contamination events since the last inspection of this area were reviewed. For HRA tasks involving significant dose rate gradients, e.g. refueling activities, the inspectors evaluated the use and placement of whole body and extremity dosimetry to monitor worker exposure. The inspectors also reviewed and discussed selected whole-body count analyses conducted during U1C16 RFO.

Radiation protection activities were evaluated against the requirements of Updated Final Safety Analysis Report (UFSAR) Section 12; Technical Specifications (TS) Sections 5.4 and 5.7; 10 Code of Federal Regulations (CFR) Parts 19 and 20; and approved licensee procedures. Records reviewed are listed in Section 2OS1 of the report Attachment.

Independent Spent Fuel Storage Installation (ISFSI) Radiological Controls

The inspectors observed and evaluated implementation of radiological controls, including RWPs and postings, and discussed the controls with a HPT and Health Physics supervisory staff. Radiological controls for loading Hi-Storm ISFSI casks were also reviewed and discussed. Radiological control activities for ISFSI areas were evaluated against 10 CFR Parts 20 and 50, NRC Certificate of Compliance (CoC) No. 1014 and applicable licensee procedures. Documents reviewed are listed in section 2OS1 of the report Attachment.

Problem Identification and Resolution Licensee Corrective Action Program (CAP) documents associated with access control to radiologically significant areas were reviewed and assessed. This included review of selected Condition Report Quality Assurance records related to radworker and HPT performance. The inspectors evaluated the licensee's ability to identify, characterize, prioritize, and resolve the identified issues. The inspectors also evaluated the scope of the licensee's internal audit program and reviewed recent assessment results. Licensee CAP documents reviewed are listed in Section 2OS1 of the report Attachment.

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

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning and Controls

a. Inspection Scope

As Low As Reasonably Achievable (ALARA) The inspectors evaluated ALARA program guidance and implementation for ongoing tasks associated with the U1C16 (RFO). The inspectors reviewed and discussed with licensee staff various ALARA work plan documents, including dose estimates and prescribed ALARA controls for selected outage work activities expected to incur significant collective doses. The inspectors reviewed the implementation of dose-reduction initiatives for high person-rem expenditure tasks.

The inspectors evaluated these elements of the ALARA program for consistency with the methods and practices delineated in applicable licensee procedures.

The inspectors evaluated the implementation and effectiveness of ALARA planning and program initiatives during work in progress. The inspectors made direct field observations of work activities involving a filter change out. The inspectors interviewed radiation workers and HPT staff to assess their understanding of dose reduction initiatives and their current and expected final accumulated occupational doses at completion of the task.

Projected RWP dose expenditure estimates from refueling outage efforts were compared to actual dose expenditures, and noted differences were discussed with cognizant ALARA staff. Changes to dose budgets relative to changes in job scope were identified and discussed. The inspectors attended pre-job briefings and evaluated the communication of ALARA goals, RWP requirements, and industry lessons-learned to job crew personnel.
The inspectors evaluated the implementation and effectiveness of selected program initiatives with respect to source-term reduction. The effectiveness of selected shielding packages installed for the current outage was assessed through completion of independent radiation surveys and comparison to applicable licensee survey records and expected planning data.

The plant collective exposure histories for calendar years (CY) 2005, 2006 and 2007, taken from data reported to the NRC pursuant to 10 CFR 20.2206(c), were reviewed and discussed with licensee staff, as were established goals for reducing collective exposure. The inspectors reviewed the applicable guidance and examined dose records of declared pregnant workers during CY 2008 and 2009 to evaluate current gestation doses for declared pregnant workers.

ALARA activities were evaluated against the requirements specified in 10 CFR 19.12; 10 CFR Part 20, Subparts B, C, F, G, H, and J; and approved licensee procedures. In addition, licensee performance was evaluated against Regulatory Guide (RG) 8.8, Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations will be As Low as Reasonably Achievable, and RG 8.13, Instruction Concerning Prenatal Radiation Exposure. Procedures and records reviewed within this inspection area are listed in Sections 2OS1 and 2OS2 of the report Attachment.

Problem Identification and Resolution The inspectors reviewed PER documents listed in Section 2OS2 of the report Attachment that were related to the ALARA program. The inspectors assessed the licensee’s ability to identify, characterize, prioritize, and resolve the identified issues in accordance with procedure PIDP-4, Corrective Action Program Screening and Oversight, Revision (Rev. 1).

The inspectors completed 21 of the required line item samples described in IP 71121.02.

b. Findings

No findings of significance were identified.

2PS2 Radioactive Material Processing and Transportation

a. Inspection Scope

Waste Processing and Characterization During inspector walk-downs, accessible sections of the liquid and solid radioactive waste (radwaste) processing systems were assessed for material condition and conformance with system design diagrams. Inspected equipment included floor drain tanks; resin transfer piping; resin and filter packaging components; and abandoned evaporator equipment. The inspectors discussed component function, processing system changes, and radwaste program implementation with licensee staff.

The 2007 Effluent Report and radionuclide characterizations from 2006 - 2008 for each major waste stream were reviewed and discussed with radwaste staff. For primary resin and Dry Active Waste (DAW) the inspectors evaluated analyses for hard-to-detect nuclides, reviewed the use of scaling factors, and examined comparison results between licensee waste stream characterizations and outside laboratory data. Waste stream mixing and concentration averaging methodology for resin and filter wastes was evaluated and discussed with radwaste operators. The inspectors also reviewed the licensee's procedural guidance for monitoring changes in waste stream isotopic mixtures.

Radwaste processing activities and equipment configuration were reviewed for compliance with the licensee's Process Control Program (PCP) and UFSAR, Chapter 11. Waste stream characterization analyses were reviewed against regulations detailed in 10 CFR Part 20, 10 CFR Part 61, and guidance provided in the Branch Technical Position on Waste Classification and Waste Form. Reviewed documents are listed in Section 2PS2 of the report Attachment.

Transportation The inspectors directly observed preparation activities for a shipment of contaminated outage equipment. The inspectors noted package markings and placarding, performed independent dose rate measurements, and interviewed shipping technicians regarding Department of Transportation (DOT) regulations. The inspectors also observed instructions provided to the driver including emergency response information.

Five shipping records were reviewed for consistency with licensee procedures and compliance with NRC and DOT regulations. The inspectors reviewed emergency response information, DOT shipping package classification, radiation survey results, and evaluated whether receiving licensees were authorized to accept the packages. Licensee procedures for opening and closing Type A boxes and Type B shipping casks were compared to recommended vendor protocols and CoC requirements. In addition, training records and training curricula for selected individuals currently qualified to ship radioactive material were reviewed.

Transportation program implementation was reviewed against regulations detailed in 10 CFR Part 20, 10 CFR Part 71, 49 CFR Parts 172-178; as well as the guidance provided in NUREG-1608. Training activities were assessed against 49 CFR Part 172 Subpart H. Documents reviewed during the inspection are listed in Section 2PS2 of the report Attachment.

Problem Identification and Resolution The inspectors reviewed selected PERs in the area of radwaste/shipping. The inspectors evaluated the licensee's ability to identify, characterize, prioritize, and resolve the identified issues in accordance with procedure PIDP-4, Corrective Action Program Screening and Oversight, Rev. 1. The inspectors also evaluated the scope of the licensee's internal audit program and reviewed recent assessment results. Licensee CAP documents reviewed are listed in Section 2PS2 of the report Attachment.

The inspectors completed 6 of 6 samples as required by inspection procedure 71122.02.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification

a. Inspection Scope

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

Occupational Radiation Safety Cornerstone The inspectors reviewed the Occupational Exposure Control Effectiveness PI results from January 2007 through February 2009. For the assessment period, the inspectors reviewed electronic dosimeter alarm logs and assessed corrective action program documents to determine whether HRA, VHRA, or unintended radiation exposures had occurred. The inspectors also reviewed licensee procedural guidance for collecting and documenting PI data. In addition, the inspectors reviewed selected personnel contamination event data and internal dose assessment results. Report section 2OS1 contains additional details regarding the inspection of controls for exposure significant areas. Documents reviewed are listed in sections 2OS1 and 4OA1 of the report Attachment.

Public Radiation Safety (PS) Cornerstone To evaluate the Radiological Effluent Technical Specification/Offsite Dose Calculation Manual Radiological Effluent Occurrences PI the inspectors reviewed data from January 2007 through February 2009.

The inspectors reviewed documents listed in Section 4OA1 of the report Attachment The inspectors completed two of the required samples for IP 71151, one sample for the OS PI and one sample for the PS PI.

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 followup, 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: Inspection of Motor-Operated Valve (MOV) Not Performed Appropriately (PER 154634)

a. Inspection Scope

On October 14, 2008, the licensee identified that a maintenance activity to inspect two containment isolation valves had been conducted on September 4, 2008, and that post-maintenance test (PMT) valve strokes had not been conducted until regularly scheduled quarterly valve strokes occurred on October 13, 2008. Since the maintenance activity, as planned, involved the manipulation of the MOV such that the limit switches would be affected, the valves would have been made inoperable for over a month as a result of the maintenance with no PMT stroke. The inspectors reviewed licensee actions to determine and correct the cause of the incident, as well as actions to evaluate similar plant equipment and activities for operability implications. The inspectors reviewed PER 154634 dealing with this event, interviewed maintenance and operations personnel, and reviewed several of the corrective actions.

b. Findings and Observations

No findings of significance were identified. The inspectors determined that the licensee's apparent cause evaluation was thorough and that the corrective actions appeared to be adequate to address the identified cause. The licensee's investigation found that the inspection activity was performed as a visual inspection only, contrary to the maintenance instructions. Therefore, the operability of the MOV was never in question since the MOV itself was not manipulated as planned and a PMT stroke was not necessary. The inspectors verified that the licensee also evaluated other prior performances of this maintenance activity to determine whether they had been adequately performed and if the required PMT strokes had been conducted. It was determined that other performances of this activity were followed by an immediate PMT stroke as required, such that no operability concerns existed with the affected valves. The inspectors also verified that the licensee's procedure for conducting MOV maintenance of this type requires that a PMT stroke be performed.

The licensee's corrective actions included conducting training with maintenance personnel to emphasize the requirements for procedure adherence when conducting maintenance, along with a revision to the maintenance procedure to clarify the requirement for a PMT stroke of MOVs whose operation is impacted by maintenance activities.

.3 Annual Sample: Inadvertent Start of All Four Emergency Diesel Generators (EDGs) (PER 168117)

a. Inspection Scope

On April 9, 2009, all four of the EDGs at the station automatically started. The start was coincident with maintenance that was being conducted to replace a relay inside of a solid state protection system (SSPS) cabinet. The inspectors reviewed licensee actions to determine and correct the cause of the incident. The inspectors reviewed PER 168117 dealing with this event, interviewed maintenance and operations personnel, and reviewed several of the corrective actions.

b. Findings and Observations

No findings of significance were identified. The inspectors determined that the licensee's apparent cause evaluation was thorough and that the corrective actions appeared to be adequate to address the identified cause. The licensee's investigation found that the EDGs started as designed in response to an automatic start signal that was generated by a relay that was inadvertently actuated by a maintenance technician who was working on a nearby relay inside the SSPS cabinet. The inspectors verified that the licensee complied with the applicable reporting requirements of 10 CFR 50.73 (a)(2) (iv) (A) and 50.73 (a)(1), which required that this invalid safety system actuation be reported to the NRC within 60 days. The event was reported to the NRC as event notification (EN) 45085.

The licensee's corrective actions included enhancement of the work order instructions to better alert workers to the situation and prevent future similar occurrences, as well as the conduct of training for maintenance personnel training to emphasize work practice behaviors to prevent future similar occurrences.

.4 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 January through June 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 January 2009 through June 2009.

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 was one trend that had potential significance, which was identified by the inspectors and entered into the licensee's CAP.

The inspectors identified four examples of performance indicator (PI) data which met the reporting criteria prescribed by NEI 99-02, "Regulatory Assessment Performance Indicator Guideline," revision 5, that the licensee had failed to report. These included an unplanned scram, an unplanned power change, an unplanned power change that was prevented by a notice of enforcement discretion, and a safety system functional failure. The licensee entered these issues into their CAP as PERs 163637 and 167436.

The licensee acknowledged the omissions in their PI submittals, corrected the submitted data, and conducted training to review the applicable reporting guidelines and criteria in order that their internal review and reporting processes would not fail to recognize reportable conditions and events in the future.

4OA3 Event Followup

.1 Unit 1 Manual Reactor Trip

a. Inspection Scope

On April 28, 2009, the inspectors responded to a manual trip of Unit 1 due to a loss of condensate flow that occurred following a manual trip of the main turbine generator. 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 event notification (EN) 45029, and documented in the licensee's CAP as PER 169863.

b. Findings

Introduction. A Green self-revealing NCV of Unit 1 TS 6.8, "Procedures & Programs," was identified for the licensee's failure to maintain adequate procedures for plant startup and power operation to support the plant responding as designed following a turbine trip. This resulted in an automatic isolation of intermediate pressure feedwater heaters, which caused a loss of condensate flow and a reactor trip.

Description. On April 28, 2009, with Unit 1 operating at approximately 27 percent RTP during startup from a refueling outage, a moisture separator reheater (MSR) shell side relief valve lifted. Operators responded by reducing power to approximately 18 percent RTP in accordance with plant procedures. With the affected relief valve still open, operators tripped the turbine in accordance with plant procedures. Approximately 10 minutes after the turbine trip occurred, two of the three parallel "strings" of intermediate pressure feedwater heaters had automatically isolated due to high level on the shell side of the #2 heaters in each string, with the third string isolation imminent for the same reason. Operators responded in accordance with plant procedures by manually tripping the reactor due to imminent loss of condensate supply to the main feedwater pumps, and, thus, main feedwater supply to the steam generators.

The inspectors reviewed the UFSAR and noted that following a turbine trip from an initial power level below 50 percent, the reactor will not be tripped, but instead the reactor plant is designed to be maintained in a stable and controlled manner by plant systems.

This event was entered into the licensee's corrective action program as PERs 169863 and 169976. The licensee evaluation determined that the heater string isolations occurred due to an elevation difference between the #2 heaters and the #3 heater drain tank (HDT), combined with the lack of residual extraction steam pressure (to overcome the elevation difference) following a turbine trip from low power. This configuration resulted in the inventory in the #3 HDT gravity draining back to fill the #2 heaters, which caused the heater string isolations to occur when heater shell side levels reached their respective high level setpoints. This susceptibility was identified by the licensee in 1998 following a similar event.

A nominal operating level in the #3 HDT must be established prior to placing the #3 HDT pump(s) in service, which is required for power operation above approximately 80 percent RTP, as noted in the UFSAR section 10.4.9.3: "With all drains from the No. 3 heater drain tank being bypassed to the condenser (and being passed through the hotwell, demineralized condensate, and condensate booster pumps) the Condensate-Feedwater System can deliver approximately 82 percent (Unit 2) and 81.6 percent (Unit 1) guaranteed flow to the steam generators."

Licensee procedure 0-GO-5, "Normal Power Operation," Revision 60, which was in effect at the time of the event, directed operators to establish level in the #3 HDT when increasing power from 30 percent power. Approximately two weeks later, the inspectors noted that licensee Procedure 0-GO-4, "Power Ascension From Less Than 5% Reactor Power to 30% Reactor Power," Revision 59, which was also in effect at the time of the event, contained similar requirements regarding the operation of #3 HDT.

Three days after the event took place, as an interim corrective action, the licensee revised Procedure 0-GO-5 to require that the #3 HDT remain drained and bypassed to the condenser until power exceeds ~45-50 percent power. The licensee had identified this, as well as the similar deficiency in Procedure 0-GO-4, and revised Procedure 0-GO-4 on May 14, 2009, to also require that the #3 HDT remain drained and bypassed to the condenser until power exceeds ~45-50 percent power.

Since plant systems are designed to prevent a reactor trip following a turbine trip from less than 50 percent power, the inspectors concluded that the operating procedures in effect at the time of the event were inadequate. This was reasonably within the licensee's ability to foresee and correct, and should have been prevented, since the issue was identified following a similar event in 1998. However, corrective actions to eliminate this susceptibility by controlling, via operating procedures, the power level at which the #3 HDT would be placed in service were not taken at that time.

Analysis. The licensee's failure to maintain adequate procedures for plant startup, which resulted in a loss of condensate flow and a reactor trip, was a performance deficiency. 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. 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 pre-solved table derived from the site specific risk-informed inspection notebook. The appropriate target for the finding in the pre-solved table was assumed to be the "Loss of Main Feedwater," since the finding (together with an independently occurring turbine trip from low power) resulted in the isolation of all condensate flow. An exposure time of <3 days was assumed, since main feedwater is designed (and expected) to be isolated on a feedwater isolation (FWI) signal following a reactor trip, and the licensee's procedures for restoration of main feedwater flow, if necessary, would have provided for the restoration of condensate flow following the heater string isolations. Based on the results of the Phase 2 analysis, the finding was determined to have very low safety significance (Green).

No cross-cutting aspect was identified since the issue was not reflective of current licensee performance, in that the failure to maintain plant operating procedures to appropriately address heater drain operations occurred following the identification of the issue in 1998, and the procedure inadequacies were promptly identified and corrected following the April 2009 event.

Enforcement. Unit 1 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 2, "General Plant Operating Procedures," required procedures for plant start-up and power operation. Procedures 0-GO-5, "Normal Power Operation," Revision 60, and Procedure 0-GO-4, "Power Ascension From Less Than 5% Reactor Power to 30% Reactor Power," Revision 59, were plant procedures that implemented these requirements. Contrary to the above, on April 28, 2009, the licensee failed to maintain adequate written procedures for plant start-up and power operation. Specifically, since plant systems are designed to prevent a reactor trip following a turbine trip from less than 50 percent power, and #3 HDT operation is not required by plant design at less than 50 percent power, plant Procedures 0-GO-5, "Normal Power Operation," Revision 60, and Procedure 0-GO-4, "Power Ascension From Less Than 5% Reactor Power to 30% Reactor Power," Revision 59, were inadequate in that the operation of #3 HDT in accordance with these procedures resulted in a loss of condensate flow and reactor trip following a turbine trip from less than 50 percent power. Because the finding was of very low safety significance and has been entered into the licensee's CAP as PER 169976, this violation is being treated as an NCV, consistent with Section VI.A of the Enforcement Policy: NCV 05000327/2009003-02, "Reactor Trip due to Inadequate Plant Operating Procedures."

.2 Unit 1 Manual Reactor Trip

a. Inspection Scope

On May 6, 2009, the inspectors responded to a manual trip of Unit 1 due to a loss of feedwater flow to the loop 1 steam generator. A failure of the loop 1 feedwater regulating valve control air diaphragm caused the valve to shut, which resulted in a loss of feedwater flow to the loop 1 steam generator. 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 event notification (EN) 45045, and documented in the licensee’s CAP as PER 170598.

b. Findings

No findings of significance were identified.

.3 Unit 2 Automatic Reactor Trip

a. Inspection Scope

On May 27, 2009, the inspectors responded to an automatic trip of Unit 2 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 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 event notification (EN) 45097, and documented in the licensee’s CAP as PER 172287.

b. Findings

No findings of significance were identified.

.4 (Closed) LER 05000327/2008-002-01, Loss of a Main Control Room Air Handling Unit In Conjunction With an Emergency Power Source Out-of-Service

On September 25, 2008, while EDG 1A was inoperable for scheduled maintenance, the train A main control room (MCR) chiller was in service. The emergency power supply to the train A MCR chiller is EDG 1A. During an attempted start, the train B MCR chiller air handling unit (AHU) motor failed, resulting in the train B of the control room air conditioning system (CRACS) being inoperable. With the train A of the CRACS also being inoperable due to its emergency power supply being inoperable, operators entered TS LCO 3.0.5 which required the licensee to shut down both units within 6 hours. The licensee requested a Notice of Enforcement Discretion (NOED) to allow continued operation beyond the 6 hours allowed by TS LCO 3.0.5, which was approved. The licensee entered this issue into their corrective action program (CAP) as PER 153304. The inspectors performed a followup inspection and verified that the licensee fully complied with the specified conditions of the notice of enforcement discretion (NOED) request. As part of this followup inspection, the inspectors discussed the event with licensee management, operations personnel, work scheduling staff, and maintenance personnel to gain an understanding of the conditions leading up to the event and actions taken immediately following to assess licensee actions. Additionally, the inspectors reviewed the root cause report to assess the detail and thoroughness of the evaluation and proposed corrective actions. Additional details of this event were discussed in NRC Inspection Reports 05000327,328/2008004 and 05000327, 328/2008005.

The licensee submitted LER 05000327/2008-002-00 on November 25, 2008, to report this event pursuant to the requirements of 10 CFR 50.73(a)(2)(i)(B) since both units remained in Mode 1 with both trains of CRACS inoperable beyond the six hours allowed by TS LCO 3.0.5. However, the inspectors noted that the licensee failed to report this event as a safety system functional failure (SSFF) in their quarterly performance indicator (PI) data, as well as via an LER. Following discussion with the licensee, entered this issue in their CAP as PER 167436, and submitted LER 05000327/2008-002-01 on June 16, 2009, to report this event as an SSFF.

The inspectors determined that the licensee's failure to report this event as an SSFF within 60 days constituted a violation of 10 CFR 50.73(a)(2)(v)(D), and that the failure to report complete and accurate performance indicator data constituted a violation of 10 CFR 50.9. This failure to comply with above reporting requirements constitutes a violation of minor significance that is not subject to enforcement action in accordance with the NRC's Enforcement Policy. This LER is closed.

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 NRC Temporary Instruction (TI) 2515/172, Reactor Coolant System Dissimilar Metal Butt Welds (DMBWs)

a. Inspection Scope

From April 6-10, 2009, the inspectors reviewed the licensee's activities related to the inspection and mitigation of DMBWs in the Reactor Coolant System (RCS) to ensure that the licensee activities were consistent with the industry requirements established in the Materials Reliability Program (MRP) document MRP-139, "Primary System Piping Butt Weld Inspection and Evaluation Guidelines", July 2005. This inspection was limited to the review of licensee activities with regard to the scoping, classification, inspection, and mitigation of dissimilar metal butt welds in accordance with the industry requirements of MRP-139.

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

b. Findings and Observations

No findings of significance were identified.

MRP-139 Baseline Inspections

1) Have the baseline inspections been performed or are they scheduled to be performed in accordance with MRP-139 guidance.

All baseline exams required by paragraph 2.1 of MRP-139 have been completed as follows: all 12 pressurizer nozzles have had full structural weld overlays applied and received post-overlay baseline exams. Exams were not required of the remaining typical DM butt weld locations as follows: Unit 1 SGs were replaced and both hot and cold leg nozzles were welded with Alloy 690 materials. Unit 2 SG welds were not within the scope of MRP-139 as the hot and cold leg nozzle welds were performed with stainless steel filler metal. The reactor coolant pump bowl material was cast stainless steel, therefore there was no DM butt weld associated with the reactor coolant pumps. The reactor vessel inlet and outlet nozzle-to-safe end DM welds were constructed with stainless steel and had no Alloy 82/182 filler metal.

Therefore all required baseline exams are fully completed.

2) Is the licensee planning to take any deviations from MRP-139 requirements?

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

Volumetric Examinations

Sample not available.

Weld Overlays

Sample not available.

Mechanical Stress Improvement (Not Applicable)

Sample not available.

In-service Inspection Program

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

.3 Groundwater Monitoring

a. Inspection Scope

The inspectors discussed current and future programs for onsite groundwater monitoring with licensee corporate staff, including number and placement of monitoring wells and identification of plant systems with the most potential for contaminated leakage. The inspectors also reviewed procedural guidance for identifying and assessing onsite spills and leaks of contaminated fluids.

In addition, the inspectors reviewed records of historical contaminated spills retained for decommissioning purposes as required by 10 CFR Part 50.75(g).

In 2007, hydrological studies were performed and several groundwater monitoring wells were installed. Analyses have been performed quarterly for tritium and, for selected samples, hard-to-detect radionuclides. To date, tritium has been the only radionuclide identified in the well samples. One of the wells shows elevated levels of tritium due to historical spills. The sample results have not exceeded the EPA drinking water limit either in the on-site or off-site monitoring wells. Based on samples not approaching the EPA limits, absence of other nuclides in samples and a downward trend in measured tritium the licensee is considering reducing the well sampling frequency.

Findings

No findings of significance were identified.

4OA6 Meetings

Exit Meeting Summary

The engineering inspectors conducted an exit meeting on April 10, 2009 with licensee management.

On April 17, 2009, the health physics inspectors discussed results of the onsite radiation protection inspection with Mr. T. Cleary, Site Vice-President, and other responsible staff.

The inspectors noted that proprietary information was reviewed during the course of the inspection but would not be included in the documented report.

On July 1, 2009, the resident inspectors presented the inspection results to Mr. Timothy Cleary and other members of his staff, who acknowledged the findings. The inspectors asked the licensee whether any of the material examined during the inspection should be considered proprietary. No proprietary information was identified.

ATTACHMENTS: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel

D. Bodine, Chemistry/Environmental Manager
D. Boone, Radiation Protection Manager
C. Church, Plant Manager
T. Cleary, Site Vice President
D. Clift, Site Support Manager
L. Cross, Maintenance Manager
J. Dvorak, Outage and Site Scheduling Manager
N. Eggemeyer, Site Security Manager
M. Halter, Support Manager, RP
S. Holdeffer, Health Physics Supervisor
K. Jones, Engineering Manager
M. Kerwin, Nuclear Assurance Manager
T. Marshall, Maintenance and Modifications Manager
S. McCamy, Health Physics Supervisor
G. Morris, Manager, Site Licensing
P. Simmons, Operations Manager
R. Thompson, Emergency Preparedness Manager
B. Wetzel, Licensing and Industry Affairs Manager
K. Wilkes, Operations Support Superintendent

NRC personnel:

R. Bernhard, Region II, Senior Reactor Analyst
S. Lingam, Project Manager, Office of Nuclear Reactor Regulation

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

050000327,328/2009003-01	URI	Containment Equipment Hatch Closure Capability during Fuel Handling Accident (Section 1R20.1)

Opened and Closed

05000327/2009003-02	NCV	Reactor Trip Due to Inadequate Plant Operating Procedures (Section 4OA3.1)

Closed

05000327/2008-002-01	LER	Loss of a Main Control Room Air Handling Unit In Conjunction With an Emergency Power Source Out-of-Service (Section 4OA3.1)

LIST OF DOCUMENTS REVIEWED

<u>Section R01: Adverse Weather Protection</u>

0-AR-ECB6-A, Electric Control Board 0-XA-55-ECB6-A, Revision 39
0-AR-ECB6-B, Electric Control Board 0-XA-55-ECB6-B, Revision 21
0-GO-14-9, Unit Operator Rounds – Transformer Yard Tour, Revision 8
GOI-6, Apparatus Operations, Revision 130
SPP-7.1, On Line Work Management, Revision 13
TRO-TO-SOP-10.129, Sequoyah Nuclear Plant (SQN) Grid Operating Guide dated 12/15/2008
IGA-6, Transmission/Power Supply Intergroup Agreement, Revision 10
OPDP-9, Emergent Issue Response, Revision 5

<u>Section R04: Equipment Alignment</u>

1,2-47W855-1, Mechanical Flow Diagram Fuel Pool Cooling and Cleaning System, Revision 45
0-SO-78-1, Spent Fuel Pit Cooling System, Revision 41

<u>Section R05: Fire Protection</u>

TVAN Fire Protection Report, Revision 25

<u>Section R08: Inservice Inspection (ISI) Activities</u>

<u>Procedures</u>
N-VT-17, Visual Examination for Leakage of PWR Reactor Head Penetrations, Rev 0007
N-UT-19, Ultrasonic Examination of Welds in Vessels Greater than 2-Inches in Wall Thickness Other Than Reactor Vessels, Rev 0017
N-GP-18, Ultrasonic Testing Supplements, Rev 0017
N-UT-26, Ultrasonic Examination for the Detection of ID Pitting, Erosion, and Corrosion, Rev 026

<u>Corrective Action Documents</u>
PER 163598, Failure to submit ASME relief requests as required, 02/13/2009
PER 167713, Through wall pipe leak on ERCW piping, 4/4/2009
PER 167216, SIS L2 INJ CHECK (including boric acid evaluation)
PER 168318, White staining due to ice melt, 4/10/2009

<u>Other</u>
0-SI-DXI-114.3, ASME Section XI ISI/NDE Program Unit 1 and Unit 2, Rev 10
1-SI-SXI-068-201.0, Leakage Test of the Reactor Coolant Pressure Boundary, Rev 0005
Sequoyah Unit 1, Cycle 16, Reactor Pressure Vessel Lower Head Remote Visual (VT-2) BMI Penetration Exam, Final Report, April, 2009
SPP-9.7, Corrosion Control Program, Rev 0017
SPP-9.1, ASME Section XI, Rev 0008
0-TI-DXX-000-001.0, Flow Accelerated Corrosion Program, Rev 6
Examination Summary and Resolution Sheet, Report Number R-0106, RSGW-B2
Examination Summary and Resolution Sheet, Report Number R-8317, SI-1610
Examination Summary and Resolution Sheet, Report Number R-7965, SI-1610
Examination Summary and Resolution Sheet, Report Number R-7500, SI-1610

USN-50 Ultrasonic Calibration Data Sheet, Calibration Number R-6857, SI-1620
Drawing SQ-108, Sequoyah Nuclear Plant, Unit 0, Replacement Steam Generator Calibration Block as Builts, Rev 00
Drawing ISI-0504-C-14, Sequoyah Nuclear Plant, Unit 1, Reactor Vessel Bottom Head Penetrations, Rev 02
Drawing 1,2-48N919, Reactor Building Units 1 & 2, Miscellaneous Steel Containment RHR Sump Liner, R2
Drawing 1,2-48N401, Reactor Building Units 1 & 2, Structural Steel Containment Vessel Anchor Bolt Plan & Base Dets – SH1, R1
Drawing 48N402, Reactor Building Units 1 & 2, Structural Steel Containment Vessel Anchor Bolt Plan & Base Dets – SH 2, R4
Drawing 41N707-1, Reactor Building Units 1 & 2, Concrete Structural Slab EL 677.28 - Outline, R3
Drawing 41N707-2, Reactor Building Units 1 & 2, Concrete Structural Slab EL 677.28 – Outline, R3
Degradation Assessment and Technical Review and Justification for Not Performing Primary or Secondary Inspections of the Steam Generators, SQN Unit 1, Cycle 16 Outage, Rev 0
Letter from TVA to NRC, Sequoyah Nuclear Plant (SQN) – American Society of Mechanical Engineers (ASME) Section XI Inservice Inspection (ISI) Program – Relief Requests, 2/29/2009
Report of Visual Examination, SQN, Unit 1, Cycle 16, Reactor Vessel Closure Head, 4/1/2009
Work Order 09-773437-000, Replace section of piping downstream of Check Valve 1-63-587 due to indications found during NDE exam

<u>Section R11: Licensed Operator Requalification</u>

AOP-C.03, Rapid Shutdown or Load Reduction, Revision 19
AOP-R.01, Steam Generator Tube Leak, Revision 25
AOP-S.04, Condensate or Heater Drains Malfunction, Revision 13
E-0, Reactor Trip or Safety Injection, Revision 30
E-3, Steam Generator Tube Rupture, Revision 17
OPDP-1, Conduct of Operations, Revision 12
Simulator Evaluation Guide Assessment Scenario A-4, Cycle 2, 2009

<u>Section R12: Maintenance Rule Implementation</u>

SPP-6.6, Maintenance Rule Performance Indicator Monitoring, Trending and Reporting – 10CFR50.65, Revision 9
PERs 121526, 156301 – Foxboro 62H Controllers

<u>Section R13: Maintenance Risk Assessments and Emergent Work Evaluation</u>

Sentinel Maintenance Risk Evaluation 27-APR-09 to 17-May-09 dated 4/28/09
Sentinel Maintenance Risk Evaluation 27-APR-09 to 17-May-09 dated 5/1/09
Sentinel Maintenance Risk Evaluation 4-MAY-09 to 17-May-09 dated 5/5/09
SPP-7.1, On Line Work Management, Revision 13
SPP-7.3, Work Activity Risk Management Process, Revision 0

Sentinel Maintenance Risk Evaluations May 10 to June 5, 2009
Risk Management Plans for Emergency Diesel Generator Outages

Section R15: Operability Evaluations
1-SI-SXP-062-203.0, Centrifugal Charging Pumps 1A-A and 1B-B Comprehensive Pump Test and Check Valve Test, Revision 2 performances dated April 2009
1-SI-SXP-062-203.0, Centrifugal Charging Pumps 1A-A and 1B-B Comprehensive Pump Test and Check Valve Test, Revision 1 performance dated October 2007
0-TI-SXI-000-200.P, ASME OM Code Pump Testing, Revision 1
ASME OM Code-2001 Subsection ISTB
1-47W809-1, Flow Diagram Chemical and Volume Control System, Revision 74
PER 169526, ERCW missile shield bolts
38N344, Structural Steel ERCW Pumping Station Missile Protection, Revision 2
PHYSI-33, Special Requirements Associated With Safety-Related Missile Protection Panels, Revision 9
45KS002, ERCW Pumping Station Structural and Miscellaneous Steel, Revision 8
PER 168390, Unplanned LCO Unit 2 B Train ECCS
PER 138749, Insufficient identification of design input data
PER 139270, Hydrology Computer Code SOCH90
PER 146852, Insufficient identification of design input data
PER 152316, SQN site review and operability evaluation of corporate PER 151414
PER 152317, SQN site review and operability evaluation of corporate PER 151412
PER 153297, Error in WBN SQN FSARs regarding seismic induced failure of Tellico Dam
PER 154499, SQN site review of corporate PER 153297
PER 158788, Anomalies/Issues for probable maximum flood
PER 162711, Hydrology Model Review

Section R19: Post Maintenance Testing
PER 167056, 1B-B DG Relay 46 Failure
PER 166924, Unplanned LCO Entry for 1B DG tripping on phase imbalance
Work Order 09-773119-000 calibrate EDG 1B-B Relay 46
TVA Periodic Instruction 1-PI-TFT-082-102.B Functional Test of the Diesel Generator 1B-B Protective Relays, Revision 3
TVA Maintenance Instruction 1-MI-TDC-082-301.B Diesel/Generator 1B-B Relay Calibration, Revision 8
ASME Boiler and Pressure Vessel Code, Code Case N-416-3, Alternate Pressure Test Requirements for Welded Repairs
0-SI-SXI-000-201.0, ASME Section XI Inservice Pressure Test, Revision 21
2-47W845-4, Mechanical Flow Diagram Essential Raw Cooling Water , Revision 16
2-SI-IFT-003-038.2, Functional Test of Steam Generator 1 Level Channel II Rack 5 Loop L-3-38 (L-519), Revision 11
2-SI-IFT-003-106.2, Functional Test of Steam Generator 4 Level Channel II Rack 5 Loop L-3-106 (L-549), Revision 12
2-SI-IFT-003-519.2, Functional Test of Environmental Allowance Modifier (EAM)/ Trip Time Delay (TTD) Protection Set II, Revision 21
NRC Regulatory Issue Summary 2007-21, Rev. 1, Adherence to Licensed Power Limits
OPDP-1, Conduct of Operations, Revision 12
PER 170233, Thermal Power Variations During Performance of WO 09-774966-000

SPP-10.4, Reactivity Management Program, Revision 6
2-PI-OPS-000-022.1, Operator at the Controls Duty Station Checklist Modes 1-4, Revision 34
0-MI-EMV-317-146.0, Inspection, Preventive and Corrective Maintenance of Limitorque Motor Operators Maintenance Instruction, Revision 21
0-MI-MVV-003-001.0, Fisher Main Feedwater Regulating Valve Maintenance, Revision 11
0-MI-IXX-000-000.D, AOV Actuator Pressure Drop Test, Revision 2
0-MI-IXX-000-000.A, AOV AirCEt Testing, Revision 3
0-MI-MXX-030-001.0, Pulley Alignment and Belt Tensioning of ESF Coolers, Revision 6

Section R20: Refueling and Outage Activities

AOP-R.02, Shutdown LOCA, Revision 11
1-SI-OPS-068-001.0, Low Temperature Overpressure Protection, Revision 4
1-SI-OPS-000-002.0, Shift Log, Revision 93
Tagout 1-TO-2009-0023; Clearance 1-62-0127-RFO, Clearance 1-63-0149-RFO
1-47W811-1, Flow Diagram Safety Injection System, Revision 72
TI-45, Physical Verification of Core Load Prior to Vessel Closure, Revision 25
0-RT-NUC-000-003.0, Low Power Physics Testing, Revision 21
0-RT-NUC-000-008.0, Low Power Physics Testing Acceptance Criteria, Revision 8
0-GO-15, Containment Closure Control, Revision 25
0-GO-9, Refueling Procedure, Revision 36
0-GO-13, Reactor Coolant System Drain and Fill Operations, Revision 59
0-GO-1, Unit Startup From Cold Shutdown to Hot Standby, Revision 54
0-GO-2, Unit Startup From Hot Standby to Reactor Critical, Revision 30
0-GO-3, Power Ascension From Reactor Critical to Less Than 5 Percent Reactor Power, Revision 21
0-GO-4, Power Ascension From Less Than 5 Percent Reactor Power to 30% Reactor Power, Revision 57
0-SI-OPS-000-011.0, Containment Access Control During Modes 1-4, Revision 34
0-MI-MXX-061-001.0, Ice Condenser Ice Servicing, Revision 26
0-TI-OPS-000-270.0, Refueling Outage System Review Program, Revision 10
PER 167428, No mission dose calc for Mode 5 and 6 accidents
AOP-M.04, "Refueling Malfunctions," revision 6,
SQN-DC-V-21.0, "Sequoyah Nuclear Plant – Environmental Design," revision 20
EPM-7-1, "EOI Administrative Controls," revision 8

Section R22: Surveillance Testing

SPP-8.1, Conduct of Testing, Revision 5
1-47W809-1, Flow Diagram-Chemical Volume Control System, Revision 74
1,2-47W810-1, Flow Diagram-Residual Heat Removal System, Revision 53
1-47W811-1, Flow Diagram-Safety Injection System, Revision 72
1-AR-M5-A (C-5), Annunciator Response – Reactor Coolant – LS-77-410A Reactor Bldg Aux FL and EQ Drain Sump Hi, Revision 31
SSD 1-L-77-410, Setpoint and Scaling Document, Revision 6
NRC Inspection Manual – Part 9900 Guidance – Ice Condenser System Inspections
PER 168371, Evaluate Ice Condenser Baskets Outside of Min/Max Weight Criteria
0-MI-MXX-061-003.0, Ice Condenser Maintenance Inspections, Revision 14

1,2-47W866-2, Flow Diagram Heating and Ventilation Air Flow, Revision 13
1,2-47W866-8, Flow Diagram Heating and Ventilation Air Flow, Revision 38
1,2-47W866-10, Flow Diagram Heating and Ventilation Air Flow, Revision 19
1,2-47W866-11, Flow Diagram Heating and Ventilation Air Flow, Revision 10
TI-50, Air Flow Measurement and Balancing Methods, Revision 18

2OS1 Access Control To Radiologically Significant Areas

Procedures, Guidance Documents, and Manuals
RCDP-10, Personnel Contamination Reporting, Revision 0004, 03/03/08
RCI-01, Radiation Protection Program, Revision 0067, 02/23/09
RCI-03, Personnel Monitoring, Revision 48, 02/24/05
RCI-14, Radiation Work Permit Program, Revision 0045, 10/10/08
RCI-15, Radiological Postings, Revision 0016, 08/01/08
RCI-22, Contamination Control, Revision 0019, 08/01/08
RCI-28, Control Of Locked High Radiation Areas, Revision 0009, 07/16/08
RCI-29, Control Of Radiation Protection Keys, Revision 7, 09/12/07
RCI-201, Radiation And Contamination Surveys, Revision 0000, 02/17/09
RCI-208, Hot Particle Controls, Revision 0000, 02/17/09

Records and Data
Occupational Cornerstone Performance Indicator Data for 2008 and 2009
Unit One and Unit 2 Locked High Radiation Area, Very High Radiation Area Key Control Inventory Data, 04/02/09
List Of Internal Dose Calculations For 2008-2009
2008 4th Quarter Dry Cask Storage Area TLD Results
2008-2009 Contamination Event Reports Data
High Radiation Area, Locked High Radiation Area, and Very High Radiation Area Key Control Inventory Data, 04/02/09
Spent Fuel Pool Inventory Data, 04/04/09
RWP Number 09044120, Install/Remove Temporary Shielding
RWP Number 09014550, Remove/Replace/Inspect Insulation
RWP Number 09044134, Remove/Replace CRDM Ductwork
RWP Number 09044040, Scaffolds/Build/Inspect/Remove
RWP Number 09044200, Replace Cables From Reactor Head
RWP Number 09044031, Unit 1 Pressurizer Remove/Replace Safety and PORV Valves
RWP Number 09044141, Reactor Head Lift And Reactor Head Set
RWP Number 09024163, Reactor Coolant Pump Alignment
Survey Number 033109-1, Inside Polar Crane Wall, 03/31/09
Survey Number 033109-6, Unit 1 RCP Seal Platform, 03/31/09
Survey Number 033109-13, Unit 1 Upper Containment – All Areas, 03,31,09
Survey Number 033109-12, Unit 1 Equipment Pit, 03/31/09
Survey Number 033109-18, Unit 1 Reactor Head Work Platform, 03/31/09
Survey Number 040109, Unit 1 Top Of Pressurizer, 04/01/09
Survey Number 010209-6, ISFSI Dry Cask Storage Pad, 01/02/09
Survey Number 020609-5, ISFSI Dry Cask Storage Pad, 02/06/09

CAP Documents
PER 145403, Violation of Radcon Postings
PER 144628, Worker received Dosimeter Alarm
PER 154577, Wrong RWP used, Dose Rate Alarm
PER 158172, Dose Rate Alarm
PER 144857, Dose Rate Alarm
PER 146655, Dose Rate Alarm
PER 149994, Failure to Report Dose Rate Alarm
PER 136968, Wrong RWP Used to Enter Locked High Rad Area
PER 15403, Violation of Radcon Postings
PERs 158027, 158045, 158028, 159055, 159059, 159060, 159065, 159070, 159078, High Radiation Area Concerns and Training Seminar for High Radiation Area Enhancement
PER 144848, Worker Exited RCA with Electronic Dosimeter in Pause Mode

ISFSI-Procedures and Guidance Documents
0-SI-DCS-079-005.0, Hi-Trac Surface Dose Rates NRC COC #1014, Amendment 2, Rev.1

ISFSI-Records and Data
Monthly survey results from October 2007 to February 2009

2OS2 ALARA Planning and Controls

Records and Data
SQN – ALARA Committee Meeting Minutes – Meeting No. 2008-08, 8/20/2008
SQN – ALARA Committee Meeting Minutes – Meeting No. 2008-09, 8/21/2008
SQN – ALARA Committee Meeting Minutes – Meeting No. 2008-10, 9/ 23/2008
SQN – ALARA Committee Meeting Minutes – Meeting No. 2008-11, 12/5/2008
SQN – ALARA Committee Meeting Minutes – Meeting No. 2009-01, 2/19/2009
APR (ALARA Planning Report) 2008-10, U2C15 Refueling Operations
APR 2008-12, U2C15 Steam Generator Primary Side Inspection and Maintenance
APR 2008-13, U2C15 Steam Generator Secondary Side Sludge Lance, Inspection, and Maintenance
APR 2008-14, U2C15 RCP Electrical Work
APR 2008-29, U2C15 Radiation Protection Surveillance
APR 2009-10, U1C16 Refueling Operations
APR 2009-33, U1C16 RPI Cable Replacement
Spreadsheet: U2C15 ALARA Summary
Spreadsheet: U2C15 Scope Growth and Rework
Spreadsheet: U1C16 Baseline Outage Estimate
Spreadsheet FY08 Goals.XLS
Spreadsheet: FY09 Unit Goals.xls
White Paper: U1C16 Radiological Protection Plan
White Paper: Sequoyah Nuclear Plant Source Term Reduction
Sequoyah Nuclear Plant, U1C15 ALARA Outage Report
Sequoyah Nuclear Plant, U2C15 ALARA Outage Report

Procedures, Instructions, Guidance Documents, and Operating Manuals
RCDP-3, Administration of Radiation Work Permits (RWPs), Rev. 2
RCI-10, ALARA Program, (with attachments 1-5), Rev. 31
RCI-14, Radiation Work Permit (RWP) Program (with attachments 3 and 4), Rev. 45
RCI-22, Contamination Control, Rev.19
SPP-7.1, On Line Work Management, Rev. 12
SPP-7.2, Outage Management, Rev. 12

2PS2 Radioactive Material Processing and Transportation

Procedures, Manuals, and Guides
Radioactive Material Shipping Manual, Rev. 38
RWTP-100, Radioactive Material/Waste Shipments, Rev. 5
RWTP-101, 10 CFR 61 Waste Characterization, Rev. 1
RWTP-102, Use of Casks, Rev. 1
RHSI-1.1, Packaging Filters and Items of High Levels of Radiation, Rev. 5
RHSI-6, Bead Resin/Activated Carbon Dewatering Procedure for Energy Solutions 14-215 or Smaller Liners, Rev. 8
HPT 007.001, HP Technician Continuing Training - Transportation and Shipping of Radioactive Material
Process Control Program, Rev. 3
PIDP-4, Corrective Action Program Screening and Oversight, Rev. 1

Shipping Records and Radwaste Data
Effluent and Waste Disposal Annual Report, 2007
08-0305, Radwaste Resin/Charcoal – Low Specific Activity
08-0404, Primary Resin – Type B
08-1104, DAW – Low Specific Activity
09-0102, Outage Equipment – Type A
09-0210, Outage Equipment – Surface Contaminated Object
Report of Calibration, Certificate No. 11006, Torque Wrench ID E06712
DCN D22261, 50.59 Evaluation, Modifications to Allow Installation of New Radwaste Processing Equipment
Waste Stream Reports, CVCS Resin 9/15/08, DAW 12/31/06 and 6/15/08, Filter Clipping 5/1/08, RadDI Charcoal 12/13/08, RadDI Resin 6/8/08
U1 and U2 RCS Scaling Factor Trends, 11/1/04 – 3/1/09
Training Records for Shipping Staff, 2007-2008
CoC No. 9204, Model CNS 10-160B Shipping Package

CAP Documents
SSA0702, Radiological Protection and Control Programs - Audit Report, 2/15/08
160878, Survey instrument with low-level fixed contamination sent to WARL as non-rad shipment
162257, Dose rates on a radwaste resin HIC were overestimated prior to shipment
163412, Letter that designates personnel who are qualified to sign shipping papers needs to be updated
164583, Procedural limits for contamination on HICs need to be changed

4OA1 Performance Indicator Verification

Procedures and Guidance Documents
SPP-3.4, Performance Indicator and MOR Submittal Using INPO Consolidated Data Entry, Rev. 7

CAP Documents
PER 144628, Worker received Dosimeter Alarm
PER 154577, Wrong RWP used, Dose Rate Alarm
PER 158172, Dose Rate Alarm
PER 144857, Dose Rate Alarm
PER 146655, Dose Rate Alarm
PER 149994, Failure to Report Dose Rate Alarm
PER 136968, Wrong RWP Used to Enter Locked High Rad Area

4OA5 Other—Groundwater Monitoring

Procedures, Manuals, and Guides
0-PI-CEM-000-010.3, Ground Water Monitoring, Rev. 0004
0-TI-CEM-000-005.0, Well Sampling, Rev. 0003
RCDP-11, Protocol for Remediation of Inadvertent Spills or Leaks of Contaminated Liquids, Rev. 0
SPP-5.14, Guide for Communicating Inadvertent Radiological Spills/Leaks to Outside Agencies, Rev. 2

Records and Data Reviewed
Vendor presentation: Investigation of Tritium Releases to Groundwater, Sequoyah Nuclear Plant June 2006
EPRI Groundwater Assessment for TVA's Sequoyah Nuclear Plant 3/30/2007
NEI 07-07 Compliance Self Assessment Sequoyah

Section 4OA2: Identification and Resolution of Problems
PER 163637, NRC PI Error
PER 167436, Question Regarding Safety System Functional Failure
PER 169526, ERCW missile shield bolts missing
PER 169935, Pressurizer pressure control system operating instruction
PER 171420, Security procedures
PER 169399, Weekly diesel gen battery inspection
PER 174600, Missed SI on 125vdc Vital Battery I
PER 174979, Ineffective Interim Action
PER 161252, Unplanned LCO
PER 173196, A MCR Chiller INOP
0-MI-EMV-317-146.0, Inspection, Preventive and Corrective Maintenance of Limitorque Motor Operators Maintenance Instruction, Revision 21
PER 154634, Containment Isolation Valves PMT Not Performed
WO 09-773275-000, Replace K604 Relay in SSPS Cabinet
PER 168117, All Four Emergency Diesels Start for Cause Unknown

Section 4OA3: Event Followup
1-47W805-1, Flow Diagram High Pressure Heater Drains and Vents, Revision 55
0-GO-4, Power Ascension From Less Than 5% Reactor Power To 30% Reactor Power, Revision 61
0-GO-5, Normal Power Operation, Revision 61
0-TI-QXX-000-001.0, Event Critique, Post Trip Report and Equipment Root Cause
G-47, Installation, Modification and Maintenance of Electrical Grounding Systems and Lightning Protection Systems, Revision 8

LIST OF ACRONYMS

ALARA	as low as reasonably achievable
CFR	Code of Federal Regulations
CoC	Certificate of Compliance
CVCS	Chemical Volume and Control Systems
DOT	Department of Transportation
ED	electronic dosimeter
HPT	Health Physics Technician
HRA	High Radiation Area
IP	Inspection Procedure
ISFSI	Independent Spent Fuel Storage Installation
LHRA	locked high radiation area
No.	Number
OA	Other Activities
OS	Occupational Radiation Safety
PER	Problem Evaluation Report
PI	Performance Indicator
PS	Public Radiation Safety
radwaste	radioactive waste
radworker	radiation worker
RCA	Radiologically Controlled Area
Rev.	Revision
RP	Radiation Protection
RWP	Radiation Work Permit
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
U1	Unit 1
U1 C16 RFO	Unit 1 Cycle 16 Refueling Outage
U2	Unit 2