UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
245 PEACHTREE CENTER AVENUE NE, SUITE 1200
ATLANTA, GEORGIA 30303-1257

July 29, 2011

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

SUBJECT: SEQUOYAH NUCLEAR PLANT - NRC INTEGRATED INSPECTION REPORT 05000327/2011003, 05000328/2011003

Dear Mr. Krich:

On June 30, 2011, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Sequoyah Nuclear Plant, Units 1 and 2. The enclosed inspection report documents the inspection results discussed on July 7, 2011 with Mr. K. Langdon and other members of your staff.

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

This report documents one NRC-identified finding of very low safety significance (Green). This finding was determined to involve a violation of NRC requirements. Additionally, a licensee-identified violation which was determined to be of very low safety significance is listed in this report. However, because of the very low safety significance and because it has been entered into your corrective action program, the NRC is treating these issues as non-cited violations (NCVs) consistent with the NRC Enforcement Policy. If you contest any NCV in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN.: Document Control Desk, Washington DC 20555-0001; with copies to the Regional Administrator, Region II; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at Sequoyah Nuclear Plant.

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

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Website at http://www.nrc.gov/reading-rm/adams.html .

Sincerely,

/CRK RA for/

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 05000327/2011003, 05000328/2011003
w/Attachment: Supplemental Information

cc w/encl: (See page 3)

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Letter to R. M. Krich from Eugene Guthrie dated July 29, 2011

SUBJECT: SEQUOYAH NUCLEAR PLANT - NRC INTEGRATED INSPECTION REPORT 05000327/2011003, 05000328/2011003

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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.:	05000327/2011003, 05000328/2011003
Licensee:	Tennessee Valley Authority (TVA)
Facility:	Sequoyah Nuclear Plant, Units 1 and 2
Location:	Sequoyah Access Road Soddy-Daisy, TN 37379
Dates:	April 1 – June 30, 2011
Inspectors:	C. Young, Senior Resident Inspector W. Deschaine, Resident Inspector M. Coursey, Reactor Inspector (1R08) B. Collins, Reactor Inspector (1R08) A. Nielsen, Senior Health Physicist (2RS1, 4OA1, 4OA7)
Approved by:	Eugene F. Guthrie, Chief Reactor Projects Branch 6 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000327/2011003, 05000328/2011003; 4/1/2011 – 6/30/2011; Sequoyah Nuclear Plant, Units 1 and 2; Surveillance Testing

The report covered a three-month period of inspection by resident inspectors and announced inspections by regional inspectors. One Green finding was identified which involved a non-cited violation (NCV) of NRC requirements. The significance of most findings is identified by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP); the cross-cutting aspect was determined using IMC 0310, "Components Within the Cross-Cutting Areas". Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process" Revision 4, dated December 2006.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

- Green. The inspectors identified a non-cited violation of Units 1 and 2 TS Surveillance Requirement (SR) 4.0.2 for the licensee's failure to perform SRs specified in Units 1 and 2 TS 3/4.3.1, "Reactor Trip System Instrumentation," and 3/4.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," within the required surveillance frequencies. The inspectors identified eight examples over the last three years (five examples on Unit 1 and three examples on Unit 2) where the interval between tests of the automatic actuation logic and reactor trip breaker functions required by SRs 4.3.1.1.1 and 4.3.2.1.1 exceeded the maximum surveillance interval allowed by TS. The licensee entered this issue into their corrective action program as PER 369938. Corrective actions included ensuring that work control processes correctly implement the required surveillance intervals.

  The finding was determined to be greater than minor because it was associated with the equipment performance attribute of the mitigating systems cornerstone and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, extending beyond the required maximum interval between TS surveillance tests affects the ability to confirm continued availability of TS equipment, and the ability to detect potential latent operability concerns in a timely manner. Using Inspection IMC 0609, "Significance Determination Process," Attachment 4, "Phase 1 - Initial Screening and Characterization of Findings," the finding was determined to be of very low safety significance (Green) since it did not represent an actual loss of safety function of a single train for greater than the associated TS allowed outage time. The inspectors did not identify that the cause of this finding was related to any of the cross-cutting aspects defined in IMC 0310, and therefore no cross-cutting aspect was assigned to this finding. (Section 1R22)

B. Licensee-Identified Violations

A violation of very low safety significance which was identified by the licensee was reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program (CAP). This violation and corrective action tracking number are listed in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status:

Unit 1 operated at or near 100 percent rated thermal power (RTP) until June 26, 2011, when Unit 1 experienced an automatic reactor trip due to a turbine trip from 100 percent RTP. Following repairs, Unit 1 achieved criticality on June 27, 2011, and reached 100 percent RTP on June 28, 2011, where it operated for the remainder of the inspection period.

Unit 2 operated at or near 100 percent RTP until April 19, 2011, when end-of-cycle power coastdown began. Unit 2 reached 75 percent RTP on May 23, 2011, when Unit 2 was shut down for a planned refueling outage. Following the outage, Unit 2 achieved criticality on June 22, 2011, and reached 100 percent RTP on June 26, 2011, 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 reviewed the licensee's readiness prior to the onset of adverse weather that poses a risk of flooding. Specifically, the inspectors reviewed flood design documents and abnormal operating procedure (AOP)-N.03, Flooding. The inspectors walked down flood protection barriers in the auxiliary building and verified required temporary spool pieces and required tools used in station procedures were complete and in their specified locations. The inspectors also verified that infrequently operated flood mode pumps were in good working order, that maintenance and testing was current, and that minor deficiencies were identified in the licensee corrective action program with scheduled completion dates. This review constituted one inspection sample. Documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

1R04 Equipment Alignment

.1 Partial System Walkdown

a. Inspection Scope

The inspectors performed partial walkdowns 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. The inspectors completed three samples.

- Unit 2 A-train Motor Driven Auxiliary Feedwater System while B-train was inoperable for maintenance
- 2B-B Diesel Generator while 2A-A was inoperable for 2-SI-OPS-082-026.A, Loss of Offsite Power with Safety Injection – D/G 2A-A Test
- Spent Fuel Pit Coolant System during Unit 2 core offload

b. Findings

No findings were identified.

.2 Complete System Walkdown

a. Inspection Scope

The inspectors performed a complete system walkdown of the Main Control Room Chillers and support systems to verify proper equipment alignment, to identify any discrepancies that could impact the function of the system and increase risk, and to verify that the licensee properly identified and resolved equipment alignment problems that could cause events or impact the functional capability of the system.

The inspectors reviewed the UFSAR, system procedures, system drawings, and system design documents to determine the correct lineup and then examined system components and their configuration to identify any discrepancies between the existing system equipment lineup and the correct lineup. During the walkdown, the inspectors reviewed the following:

- Valves were correctly positioned and did not exhibit leakage that would impact the functions of any given valve.
- Electrical power was available as required.
- Major system components were correctly labeled, lubricated, cooled, ventilated, etc.
- Hangers and supports were correctly installed and functional.
- Essential support systems were operational.
- Ancillary equipment or debris did not interfere with system performance.
- Tagging clearances were appropriate.
- Valves were locked as required by the locked valve program.

In addition, the inspectors reviewed outstanding maintenance work requests and design issues on the system to determine whether any condition described in those work requests could adversely impact current system operability. Documents reviewed are listed in the Attachment. The inspectors completed one sample.

b. Findings

No findings were identified.

1R05 Fire Protection

.1 Fire Protection Tours

a. Inspection Scope

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

- Control Building Elevation 706 (Cable Spreading Room)
- Control Building Elevation 669 (Mechanical Equipment Room, 250 VDC Battery and Battery Board Rooms)
- Control Building Elevation 685 (Auxiliary Instrument Rooms)
- Auxiliary Building Elevation 714 (Corridor)
- Auxiliary Building Elevation 690 (Corridor)
- Control Building Elevation 732 (Mechanical Equipment Room and Relay Room)

b. Findings

No findings were identified.

1R08 Inservice Inspection Activities

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

a. Inspection Scope

From May 30 to June 10, 2011, 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 2. The inspectors' activities consisted of an on-site review of NDE 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 inspections described in Sections 1R08.1, 1R08.2, 1R08.3, 1R08.4 and 1R08.5 below constituted one In-service Inspection sample as defined in Inspection Procedure 71111.08 05.

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

- Ultrasonic Testing (UT) of SIS-244 in 10-inch Safety Injection piping weld.
- UT examinations of RHRF-103 8-inch pipe to pipe weld.

During non-destructive surface and volumetric examinations performed since the previous refueling outage, the licensee did not identify any recordable indications. Therefore, no NRC review was completed for this inspection procedure attribute.

The inspectors reviewed the following pressure boundary welds completed for risk-significant systems during the outage to evaluate if the licensee applied the pre-service non-destructive examinations and acceptance criteria required by the construction Code, NRC-approved Code Case, NRC-approved Code relief request or the ASME Code Section XI. In addition, the inspectors reviewed the welding procedure specification and supporting weld procedure qualification records to evaluate if the weld procedures were qualified in accordance with the requirements of Construction Code and the ASME Code Section IX.

- WO 110993041 SQN-2-VLV-062-525, CENT CHRG PMP DISC CK
- WO 110993043 SQN-2-VLV-062-532, CENT CHRG PMP CK

b. Findings

No findings were identified.

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

a. Inspection Scope

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

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

b. Findings

No findings 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 current spring 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 the following engineering evaluations for evidence of boric acid found on systems containing borated water to verify compliance with generally accepted industry guidance.

- Borated Water Leak Assessment for WO 111593024 1-PMP-074-0020, dated 11/30/10.
- Boric Acid Leakage Evaluation for WO 112087504 2FCV-068-333-A, dated 5/28/11.
- Boric Acid Leakage Evaluation for WO 111466271 SQN-1-Driv-063-0342C, dated 10/15/10.

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

- PER 290082 RHR Pump 1B-B, dated 11/30/10
- PER 286694 Wet Boric Acid Leak on 0-VLV-0322A, dated 11/18/1010
- PER 232296 Active borated water leak from 2-VLV-062-0522 valve packing, dated 6/2/2010

b. Findings

No findings were identified.

.4 Steam Generator (SG) Tube Inspection Activities

a. Inspection Scope

The NRC inspectors observed the following activities and/or reviewed the following documentation and evaluated them against the licensee's technical specifications, commitments made to the NRC, ASME Section XI, and Nuclear Energy Institute (NEI) 97-06 (Steam Generator Program Guidelines):

- Reviewed the licensee's in-situ SG tube pressure testing screening criteria. In particular, assessed whether assumed NDE flaw sizing accuracy was consistent with data from the EPRI examination technique specification sheets (ETSS) or other applicable performance demonstrations.
- Interviewed Eddy Current Testing (ET) data analysts and reviewed 5 samples of ET data.
- Compared the numbers and sizes of SG tube flaws/degradation identified against the licensee's previous outage Operational Assessment.
- Reviewed the SG tube ET examination scope and expansion criteria.
- Evaluated if the licensee's SG tube ET examination scope included potential areas of tube degradation identified in prior outage SG tube inspections and/or as identified in NRC generic industry operating experience applicable to the licensee's SG tubes.
- Reviewed the licensee's implementation of their extent of condition inspection scope and repairs for new SG tube degradation mechanism(s). No new degradation mechanisms were identified during the ET examinations.
- Reviewed the licensee's repair criteria and processes.
- Primary-to-secondary leakage (e.g., SG tube leakage) was below three gallons per day, or the detection threshold, during the previous operating cycle.
- Evaluated if the ET equipment and techniques used by the licensee to acquire data from the SG tubes were qualified or validated to detect the known/expected types of SG tube degradation in accordance with Appendix H, Performance Demonstration for Eddy Current Examination, of EPRI Pressurized Water Reactor Steam Generator Examination Guidelines, Revision 7.
- Reviewed the licensee's secondary side SG Foreign Object Search and Removal (FOSAR) activities.
- Reviewed the licensee's evaluations and repairs for SG tubes damaged by foreign material or tubes surrounding inaccessible foreign objects left within the secondary side of the steam generators.
- Reviewed ET personnel qualifications.

b. Findings

No findings 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) and Service Requests (SRs). The inspectors reviewed the PERs and SRs to confirm 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 10 CFR 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 were identified.

1R11 Licensed Operator Requalification Program

.1 Quarterly Review

a. Inspection Scope

The inspectors performed one licensed operator requalification program review. The inspectors observed a simulator session on May 11, 2011. The training scenario involved Just-In-Time Training for Pre-Refueling Outage risk significant activities such as placing the RHR system in service. 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. This activity constituted one inspection sample.

b. Findings

No findings were identified.

1R12 Maintenance Effectiveness

a. Inspection Scope

The inspectors reviewed the maintenance activities, issues, and/or systems listed below to verify the effectiveness of the licensee's activities in terms of: appropriate work practices; identifying and addressing common cause failures; scoping in accordance with 10 CFR 50.65(b); characterizing reliability issues for performance; trending key parameters for condition monitoring; charging unavailability for performance; classification in accordance with 10 CFR 50.65(a)(1) or (a)(2); appropriateness of performance criteria for structure, system, or components (SSCs) and functions classified as (a)(2); and appropriateness of goals and corrective actions for SSCs and functions classified as (a)(1). Documents reviewed are listed in the Attachment. The inspection performed on the 1A EDG load swing problem utilized OpESS FY 2010-01, "Recent Inspection Experience for Components Installed Beyond Vendor Recommended Service Life." The inspectors completed two samples.

- PER 314771, Unit 1 Loop 3 Feedwater Regulating Valve Failure
- OpESS 2010-01: PER 324530, 1A EDG load swings due to speed/load control motor operated potentiometer failure

b. Findings

No findings were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope

The inspectors reviewed the following activities to determine whether appropriate risk assessments were performed prior to removing equipment from service for maintenance. The inspectors evaluated whether risk assessments were performed as required by 10 CFR 50.65(a)(4), and were accurate and complete. When emergent work was performed, the inspectors reviewed whether plant risk was promptly reassessed and managed. The inspectors also assessed whether the licensee's risk assessment tool use and risk categories were in accordance with Standard Programs and Processes Procedure NPG-SPP-07.1, "On-Line Work Management," Revision 3, and Instruction 0-TI-DSM-000-007.1, "Risk Assessment Guidelines," Revision 9. Documents reviewed are listed in the Attachment. The inspectors completed five samples.

- U2 cycle 17 outage RCS drain to mid-loop risk management actions - ORAM Orange
- Yellow PSA Risk – Units 1 – Component Cooling Water 1B Pump aligned to 'B' Train for SI-26
- Yellow PSA Risk – Unit 1 – Turbine Driven Auxiliary Feedwater Valve Stroke per 1-SI-SXV-000-201.0, App 'L'
- Unit 2 Refueling Outage risk review
- LCO 4.0.3 Risk Assessment/RMAs for SSPS testing

b. Findings

No findings were identified.

1R15 Operability Evaluations

a. Inspection Scope

For the five 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 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. The inspectors completed five samples.

- PER 385549, 2A Safety Injection Pump performance curve below design minimum values at one point
- PER 340794, ERCW instrument line supports
- PER 332977, EDG past operability due to fire/flood mode pump testing
- PER 349161, Feedwater regulating valve 3-90 past operability/reportability
- PER 340456, Probable Maximum Flood flow path through ERCW instrumentation well vent piping

b. Findings

No findings were identified.

1R18 Plant Modifications

.1 Temporary Modifications

a. Inspection Scope

The inspectors reviewed the temporary modification listed below and the associated 10 CFR 50.59 screening, and compared it against the UFSAR and TS to verify whether the modification affected operability or availability of the affected system.

- TACF 1-11-005-063, Unit 1 RHR discharge header continuous vent

Following installation and testing, the inspectors observed indications affected by the modification, discussed them with operators, and verified that the modification was installed properly and its operation did not adversely affect safety system functions. Documents reviewed are listed in the Attachment. The inspectors completed one sample.

b. Findings

No findings were identified.

.2 Permanent Modifications

a. Inspection Scope

The inspectors reviewed DCN D22582A, Unit 2 start bus replacement and breaker manual transfer scheme modification, Unit 2 unit station service transformer disconnect, and temporary power feed to Unit 2 unit boards. The inspectors walked down installed modifications and interviewed engineering and maintenance personnel regarding the modification and associated post-modification testing to verify that (1) the design bases, licensing bases, and performance capability had not been degraded through this modification, and (2) the modification was not performed during increased risk-significant configurations that placed the plant in an unsafe condition. The inspectors also reviewed applicable sections of the UFSAR, plant modification procedures, system drawings, supporting analyses, and related PERs. Documents reviewed are listed in the Attachment. The inspectors completed one sample.

b. Findings

No findings were identified.

1R19 Post-Maintenance Testing

a. Inspection Scope

The inspectors reviewed the post-maintenance tests associated with the seven 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. The inspectors completed seven samples.

- WO 112257547, Fuse blew during performance of Reactor Trip Instrumentation Functional Test (SSPS) Train B
- WO 110851792, Disassemble and Inspect Inboard CIV (check valve VLV-31C-697) for Incore Instrument Room Cooler B Supply
- WO 110774613, Disassemble and Inspect Inboard CIV (check valve VLV-31C-715) for Incore Instrument Room Cooler B Return
- WO 07780697001, Rebuild Spent Fuel Pit Pump A

- WO 09-777459-000, 2A and 2B Start Bus Replacement
- WO 111643152, Calibrate high steam flow isolation to Terry Turbine on Unit 1
- WOs 112272300 and 112274232, Replace Diesel Engine 2A1 Fuel Oil Transfer Pump and Motor

b. Findings

No findings were identified.

1R20 Refueling and Outage Activities

.1 Unit 2 Refueling Outage

a. Inspection Scope

For the Unit 2 refueling outage that began on May 23, 2011, the inspectors evaluated licensee activities to verify that the licensee considered risk in developing outage schedules, followed risk reduction methods developed to control plant configuration, developed mitigation strategies for the loss of key safety functions, and adhered to operating license and TS requirements that ensure defense-in-depth. The inspectors also walked down portions of Unit 2 not normally accessible during at-power operations to verify that safety-related and risk-significant SSCs were maintained in an operable condition. Specifically, between May 23 and June 26, 2011, the inspectors performed inspections and reviews of the following outage activities. Documents reviewed are listed in the Attachment. This inspection satisfied one inspection sample for Refueling Activities.

Outage Plan.

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

Reactor Shutdown

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

Licensee Control of Outage Activities

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

Refueling Activities

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

Reduced Inventory and Mid-Loop Conditions

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

Heatup and Startup Activities

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

b. Findings

No findings were identified.

.2 Unit 1 Forced Outage

a. Inspection Scope

Following the automatic trip of Unit 1 on June 26, 2011, the licensee maintained Unit 1 in Mode 3 until conditions to support restart were established on June 27, 2011. 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-PI-OPS-000-011.0, "Containment Access Control During Modes 1-4," Rev. 1, for the associated containment entries to ensure that all items that entered containment were removed so nothing would be left that could affect performance of the containment sump. The inspectors observed portions of the plant startup including reactor criticality and power ascension. This inspection satisfied one inspection sample for Outage Activities.

b. Findings

No findings were identified.

1R22 Surveillance Testing

a. Inspection Scope

For the nine 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 whether 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. The inspectors completed nine samples.

In-Service Tests:

- 1-SI-SXP-074-201.A, RHR Pump 1A Section XI Test, Revision 16
- 2-SI-SXP-003-202.B, 2B-B MDAFW Comprehensive Performance Test, Revision 5

Routine Surveillance Tests:

- 2-SI-OPS-082-026.A, Loss of Offsite Power with Safety Injection – Diesel Generator 2A-A Test, Revision 42
- 1-SI-IFT-099-90.8A, Reactor Trip Instrumentation Monthly Functional Test (SSPS) Train A, Revision 19
- 1-SI-IFT-099-90.8B, Reactor Trip Instrumentation Monthly Functional Test (SSPS) Train B, Revision 16
- 2-SI-IFT-068-456.0, Functional Test of RCS Cold Over pressurization Protection System PORV PCV-68-334, Revision 16

Ice Condenser Surveillance Test:

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

Containment Isolation Valve (CIV) Surveillance Tests:

- 0-SI-SLT-067-258.2, Containment Isolation Valve Local Leak Rate Test Lower Compartment Essential Raw Cooling Water – Penetration X-58, Revision 11
- 0-SI-SLT-067-258.2, Containment Isolation Valve Local Leak Rate Test Lower Compartment Essential Raw Cooling Water – Penetration X-59, Revision 11

b. Findings

Introduction. The inspectors identified a Green non-cited violation of Units 1 and 2 TS Surveillance Requirement (SR) 4.0.2 for the licensee's failure to perform SRs specified in Units 1 and 2 TS 3/4.3.1, "Reactor Trip System Instrumentation," and 3/4.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," within the required surveillance frequencies. The inspectors identified eight examples over the last three years (five examples on Unit 1 and three examples on Unit 2) where the interval between tests of the automatic actuation logic and reactor trip breaker functions required by SRs 4.3.1.1.1 and 4.3.2.1.1 exceeded the maximum surveillance interval allowed by TS.

Description. On May 11, 2011, the inspectors identified that the monthly surveillance testing required by Units 1 and 2 TS SRs 4.3.1.1.1 and 4.3.2.1.1 for the automatic actuation logic and reactor trip beaker functions had last been performed 54 days prior on Unit 1 and 40 days prior on Unit 2. The inspectors reviewed Tables 4.3-1 and 4.3-2 of both Units 1 and 2 TS, which indicated a testing frequency notation of "M" for channel functional tests of automatic actuation logic and reactor trip breaker functions. Table 1.2 of both Units 1 and 2 TS defined the frequency notation of "M" to be at least once per 31 days. A table footnote which appeared in Tables 4.3-1 and 4.3-2 attached to the "M" frequency notation further explained that each train or logic channel shall be tested at least every 62 days on a staggered test basis. This requirement stipulated that the monthly surveillance test need only be performed on one train of the system, and that successive monthly tests alternate between the two trains of the system. The allowable

62-day interval between consecutive tests of a given train of the system is the result of performing a monthly (i.e. at least once per 31-day) system test on a staggered test basis (i.e. a given train of the system is tested on every other performance).

The inspectors discussed this observation with the licensee, and this issue was entered into the CAP as PER 369938. The inspectors learned that the licensee's existing program for scheduling and implementing TS surveillance testing had been based on not exceeding the above noted 62-day interval between consecutive tests of a given train of the system on each Unit. However, the licensee's program failed to ensure that the required 31-day interval between successive surveillance tests on the system (i.e. one train or the other) was met. The inspectors identified a total of eight examples over the past three years where the maximum surveillance interval allowed by TS (31 days plus 25 percent extension allowed by TS SR 4.0.2) had been exceeded. The licensee subsequently performed the applicable surveillances, all of which were satisfactory.

Analysis. The licensee's failure to perform TS SRs within the required frequency was a performance deficiency. The finding was determined to be greater than minor because it was associated with the equipment performance attribute of the mitigating systems cornerstone and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, extending beyond the required maximum interval between TS surveillance tests affects the ability to confirm continued availability of TS equipment, and the ability to detect potential latent operability concerns in a timely manner (i.e. to reduce potential fault exposure). Using Inspection IMC 0609, "Significance Determination Process," Attachment 4, "Phase 1 - Initial Screening and Characterization of Findings," the finding was determined to be of very low safety significance (Green) since it did not represent an actual loss of safety function of a single train for greater than the associated TS allowed outage time. The licensee subsequently performed the applicable surveillances, all of which were satisfactory. The inspectors did not identify that the cause of this finding was related to any of the cross-cutting aspects defined in IMC 0310, and therefore no cross-cutting aspect was assigned to this finding.

Enforcement. Units 1 and 2 TS SR 4.0.2 required that each surveillance requirement (SR) shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25 percent of the specified surveillance interval. Units 1 and 2 TS SRs 4.3.1.1.1 and 4.3.2.1.1 required that each reactor trip system and ESFAS instrumentation channel shall be demonstrated operable by the performance of the channel functional test operations for the Modes and at the frequencies shown in Tables 4.3-1 and 4.3-2. Tables 4.3-1 and 4.3-2 required that the applicable reactor trip breaker, automatic trip logic, and automatic actuation logic functions be tested with a frequency notation of "M" while in Mode 1. Table 1.2 defined the frequency notation of "M" to be at least once per 31 days. Contrary to the above, on May 10, 2011, April 26, 2011, and on six previous occasions over the last three years, the licensee failed to perform each SR within the specified surveillance interval with a maximum allowable extension not to exceed 25 percent of the specified surveillance interval. Specifically, while operating in Mode 1 over the course of the surveillance interval, the licensee failed to perform the reactor trip breaker and automatic actuation logic function surveillance testing required by TS SRs 4.3.1.1.1 and 4.3.2.1.1 at least once per 31 days plus the 25 percent extension allowed by LCO 4.0.2. The licensee subsequently performed the applicable surveillances, all of which were satisfactory. Because the finding was of very low safety significance and has been entered into the licensee's CAP as PER 369938, this violation is being treated as an NCV, consistent with the NRC Enforcement Policy: NCV 05000327,328/2011003-01, "Failure to Perform Instrumentation Surveillance Testing Within Required Frequency."

2. RADIATION SAFETY

2RS1 Radiological Hazard Assessment and Exposure Controls

a. Inspection Scope

Hazard Assessment and Instructions to workers During facility tours, the inspectors directly observed labeling of radioactive material and postings for radiation areas, high radiation areas (HRAs), and contaminated areas established within the radiologically controlled area (RCA) of the Unit 2 (U2) containment, Unit 1 (U1) and U2 auxiliary buildings, and radioactive waste (radwaste) processing and storage locations. The inspectors independently measured radiation dose rates or directly observed conduct of licensee radiation surveys for selected RCA areas. The inspectors reviewed survey records for several plant areas including surveys for alpha emitters, hot particles, airborne radioactivity, gamma surveys with a range of dose rate gradients, and pre-job surveys for upcoming tasks. The inspectors also discussed changes to plant operations that could contribute to changing radiological conditions since the last inspection. For selected outage jobs, the inspectors attended pre-job briefings and reviewed radiation work permit (RWP) details to assess communication of radiological control requirements and current radiological conditions to workers.

Hazard Control and Work Practices The inspectors evaluated access barrier effectiveness for selected Locked High Radiation Area (LHRA) and Very High Radiation Area (VHRA) locations. Changes to procedural guidance for LHRA and VHRA controls were discussed with 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 (including airborne controls) were evaluated for selected U2 Refueling Outage 17 (2R17) tasks including steam generator (S/G) eddy current testing, thimble tube cutting and removal, and reactor upper internals movement. In addition, licensee controls for areas where dose rates could change significantly as a result of refueling operations were reviewed and discussed.

Occupational workers' adherence to selected RWPs and HP technician (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 S/G eddy current testing. ED alarm logs were reviewed and worker response to dose and dose rate alarms was evaluated. For HRA tasks involving significant dose rate gradients, e.g. S/G maintenance activities, the inspectors evaluated the use and placement of whole body and extremity dosimetry to monitor worker exposure.

Control of Radioactive Material The inspectors observed surveys of material and personnel being released from the RCA using small article monitor, personnel contamination monitor, and portal monitor instruments. The inspectors reviewed the last two calibration records for selected release point survey instruments and discussed equipment sensitivity, alarm setpoints, and release program guidance with licensee staff. The inspectors also reviewed records of leak tests on selected sealed sources and discussed nationally tracked source transactions with licensee staff.

Problem Identification and Resolution Problem Evaluation Reports (PER)s associated with radiological hazard assessment and control were reviewed and assessed. The inspectors evaluated the licensee's ability to identify and resolve the issues in accordance with procedure NPG-SPP-03.1, "Corrective Action Program", Rev. 1. The inspectors also evaluated the scope of the licensee's internal audit program and reviewed recent assessment results.

Radiation protection activities were evaluated against the requirements of Updated Final Safety Analysis Report (UFSAR) Section 12; Technical Specifications (TS) Section 6.12; 10 CFR Parts 19 and 20; and approved licensee procedures. Licensee programs for monitoring materials and personnel released from the RCA were evaluated against 10 CFR Part 20 and IE Circular 81-07, Control of Radioactively Contaminated Material. Documents reviewed are listed in Section 2RS1 of the Attachment.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification

a. Inspection Scope

The inspectors sampled licensee submittals for the 4 PIs listed below for the period from April 1, 2010 through March 31, 2011 for both Unit 1 and Unit 2. Definitions and guidance contained in Nuclear Energy Institute (NEI) 99-02, Regulatory Assessment Indicator Guideline, Revision 6, were used to determine the reporting basis for each data element in order to verify the accuracy of the PI data reported during that period.

Cornerstone: Initiating Events

- Unplanned Scrams per 7000 Critical Hours
- Unplanned Scrams with Complications
- Unplanned Power Changes per 7000 Critical Hours

The inspectors reviewed selected Licensee Event Reports (LERs) and portions of operator logs to verify whether the licensee had accurately identified the number of scrams and unplanned power changes that occurred during the previous four quarters for both units. The inspectors also reviewed the accuracy of the number of critical hours reported and the licensee's basis for addressing the criteria for complications for each of the reported scrams. Documents reviewed are listed in the Attachment.

Cornerstone: Occupational Radiation Safety

The inspectors reviewed the Occupational Exposure Control Effectiveness PI results for the Occupational Radiation Safety Cornerstone from October 2010 through March 2011. The inspectors reviewed ED alarm logs and PERs related to controls for exposure significant areas. The inspectors also evaluated licensee procedural guidance for identifying and reporting PI occurrences. Documents reviewed are listed in sections 2RS1 and 4OA1 of the Attachment.

b. Findings

No findings were identified.

4OA2 Identification and Resolution of Problems

.1 Daily Review

a. Inspection Scope

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

b. Findings and Observations

No findings were identified.

.2 Semi-Annual Trend Review

a. Inspection Scope

As required by Inspection Procedure 71152, the inspectors performed a review of the licensee's CAP and other associated programs and 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 2011, 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 through June 2010. This inspection satisfied one inspection sample for Semi-annual Trend Review.

b. Findings and Observations

No findings 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.

The inspectors identified a trend which included four issues that involved deficiencies in the identification and evaluation of conditions having potentially reportable aspects under 10 CFR 50.73. The inspectors observed that these were examples of a continued trend which includes several previously (2009) identified issues involving deficiencies in regulatory reporting.

On November 16, 2010, PER 284451 was entered into the licensee's CAP for the discovery of the Unit 1 Loop 3 FRV in an inoperable condition. On December 20, 2010, PER 284451 was closed without having been evaluated for cause or reportability in accordance with station procedures for CAP and regulatory screening. On January 28, 2011, PER 314771 was entered into the licensee's CAP in response to the inspectors' questions regarding cause and implications of the inoperable FRV. On February 21, 2011, the licensee completed an apparent cause evaluation under PER 314771 which determined that the FRV had been in an inoperable condition for 3 days prior to discovery. This PER was also not evaluated for reportability. On April 4, 2011, PER 349161 was entered into the licensee's CAP in response to the inspectors' questions regarding potential reportability. This reportability evaluation resulted in the licensee submitting the licensee event report (LER) which is further discussed in section 4OA3.3 of this inspection report.

PER 246090 was entered into the licensee's CAP on August 25, 2010, and documented the need to evaluate past operability of the EDGs in light of time periods when equipment with untested load shedding functions was being powered from safety-related shutdown boards. The inspectors identified that this PER had been closed on March 9, 2011, without having performed the necessary past operability/reportability evaluation. PER 332977 was entered into the licensee's CAP to capture this observation and ensure that the required past operability/reportability evaluation was completed.

The LER discussed in section 4OA3.2 of this inspection report was submitted as required within 60 days of the reportable event which occurred on February 15, 2011. However, the inspectors noted that PER 323782, which was entered into the licensee's CAP to document the event, was not evaluated for reportability in accordance with the licensee's CAP screening procedures listed above. The inspectors noted that the site licensing organization recognized the reportable condition and submitted the required report independently from the CAP processes designed to review conditions for reportability.

The inspectors also observed that the requirement contained in licensee procedures OPDP-8, Limiting Conditions for operation Tracking, revision 5, and NPG-SPP-03.1.3, Regulatory Screening, revision 1, for operations personnel to address reportability of conditions entered into the CAP in conjunction with documenting immediate operability determinations was not being routinely implemented as required. The licensee entered this issue into the CAP as PER 358818.

The inspectors had previously identified several issues in 2009 which involved deficiencies in the reporting of conditions pursuant to the requirements of 10 CFR 50.73. These included: (1) An LER which reported a dual unit trip failed to correctly indicate the trip as being an unplanned scram with complications; (2) The failure to perform an action required by LCO 3.8.1.1.a following a dual unit trip was not reported as required as being a separate reportable condition; (3) A failure to report both trains of CRACS being inoperable as a safety system functional failure under 50.73(a)(2)(v); and (4) A failure to submit an LER as required under 50.73(a)(2)(i)(b) within 60 days of discovery of an inadequate surveillance testing issue which affected the ABGTS system.

The licensee initiated PER 397678 on July 6, 2011 to document the observed trend in issues involving the recognition and evaluation of potentially reportable conditions.

4OA3 Event Follow-up

.1 (Closed) Licensee Event Report (LER) 05000327,328/2009-009-00 and -01, Unanalyzed Condition Affecting Probable Maximum Flood (PMF) Level

a. Inspection Scope

On December 30, 2009, the issuance of an updated calculation titled “PMF Determination for Tennessee River Watershed” increased the Sequoyah Nuclear Plant (SQN) design basis PMF level from Elevation 719.6 feet to Elevation 722.0 feet. This revision to the calculated PMF elevation resulted from several calculational changes. A previous change had decreased the SQN PMF elevation from 722.6 feet to 719.6 feet. However, SQN remained designed for a flood elevation of 722.6 feet with the exceptions of the emergency diesel generator sets and spent fuel pool cooling pumps. Contingency measures were put into place to protect the diesel generator sets and spent fuel pool cooling pumps at the new calculated PMF elevation of 722.0 feet. Although no actual flooding occurred, because of the unanalyzed condition the potential existed for SQN to exceed its PMF design basis and adversely impact plant safety. The licensee documented the issue in PER 162711, which included an apparent cause evaluation.

The inspectors discussed the event with operations, maintenance, engineering, and licensee management personnel to gain an understanding of the conditions leading up to the event and assess licensee actions taken following the event. The inspectors independently verified the adequacy of the compensatory measures to ensure the capability of the EDGs and spent fuel pit coolant system to function under PMF conditions. Additionally, the inspectors reviewed the apparent cause evaluation report to assess the detail and thoroughness of the evaluation and the adequacy of the proposed corrective actions.

The inspectors reviewed the LER and PER 162711 to verify that the cause of the unanalyzed condition was identified and whether corrective actions were appropriate. The root causes of this event were determined to be the failure to establish a PMF procedure or process that could be used to train personnel to perform, revise, and maintain accurate PMF calculations, as well as inadequate communication between organizations, specifically TVA's River Operations (RO) group, who performed the calculation which decreased the PMF level from 722.6 feet to 719.6 feet, and TVA's Nuclear Power Group (NPG). This root cause analysis stemmed from three Notices of Violation (NOV) that the NRC issued in inspection report numbers 05200014/2008-001 and 05200015/2008-001 associated with TVA's Bellefonte (BLN) combined operating license application (COLA) document submittal. These violations identified the root problems which resulted in the calculated PMF level changes. The inspectors concluded that the licensee's corrective actions to this event were appropriate, including developing a PMF evaluation process that will include interface reviews between impacted TVA organizations, define ownership, roles and responsibilities of impacted TVA organizations, include a periodic review of critical PMF inputs, and provide training on management of the PMF process. Also, a design change has been initiated to permanently protect the diesel generator sets and spent fuel pool cooling pumps based on the increased PMF elevation. This LER is closed.

b. Findings

No findings were identified.

.2 (Closed) Licensee Event Report (LER) 05000327,328/2011-001-00, Both Trains of Control Room Air Conditioning Systems Inoperable

a. Inspection Scope

On February 15, 2011, at 0705 Eastern Standard Time (EST), the licensee entered Technical Specification Limiting Condition for Operation (LCO) 3.0.3, for Unit 1 and Unit 2, due to both trains of control room air conditioning systems inoperable. LCO 3.0.3 was entered because "B" train Main Control Room (MCR) chiller failed and "A" train MCR chiller was tagged out of service for scheduled maintenance. At 1005 EST on February 15, 2011, the "A" train MCR chiller was returned to operable status and LCO 3.0.3 was exited on both units. The licensee documented the issue in PER 323782, which included an apparent cause evaluation.

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

The inspectors reviewed the LER and PER 323782 to verify that the cause of the main control room "B" chiller's failure to start was identified and whether corrective actions were appropriate. The licensee's apparent cause evaluation identified that preventive maintenance procedures associated with the main control room "B" chiller did not include requirements for periodic inspection of the internal tubing of the temperature transmitter. The inspectors concluded that the licensee's corrective actions to this event were appropriate, including revision to the applicable preventive maintenance procedures to include inspections of the internal tubing of the temperature transmitter for the main control room "B" chiller. This LER is closed.

b. Findings

No findings were identified.

.3 (Closed) Licensee Event Report (LER) 05000327/2011-002-00, Feedwater Regulator Valve Inoperable

a. Inspection Scope

On November 15, 2010, while operating in Mode 2, the Unit 1 Loop 3 feedwater regulating valve (FRV) was discovered to be in an inoperable condition due to its inability to perform its required feedwater isolation function, and was subsequently isolated on November 16, 2011, as required by TS LCO 3.7.1.6.

The inspectors discussed the event with operations, maintenance, engineering, and licensee management personnel to gain an understanding of the conditions leading up to the event and assess licensee actions taken in response to the event. Additionally, the inspectors reviewed the licensee's cause evaluation report to assess the detail and thoroughness of the evaluation and the adequacy of the proposed corrective actions. This event was documented in the licensee corrective action program as PERs 284451, 314771, and 349161.

The inspectors reviewed the LER and associated PERs to verify that the cause of the condition was identified and whether corrective actions were appropriate. The licensee's cause evaluation determined that cause of the condition was inadvertent valve stem length adjustment due to stem rotation during reassembly from the actuator maintenance performed during outage. The torque applied to the travel stop cap screw during reassembly was sufficient to cause rotation in the actuator/valve stem threaded coupling assembly and a corresponding inadvertent stem length adjustment. A Green NCV was issued in inspection report 05000327/2011002 for inadequate maintenance procedures which led to this condition. The inspectors concluded that the licensee's corrective actions to this event were appropriate, including actions to revise the applicable work procedure to ensure no inadvertent valve stem rotation during reassembly, as well as to evaluate revising post maintenance testing associated with this activity to incorporate a method to verify positive valve seating when stroked to the closed position.

This LER was submitted under the criterion of 10 CFR 50.73(a)(2)(i)(B) (condition prohibited by TS) on the basis that, since the FRV was inoperable as required by LCO 3.7.1.6 in Modes 1, 2, and 3 from the time Mode 3 was entered on November 12, 2011, until the time the FRV was isolated on November 16, 2011, (total of just over 75 hours), the applicable LCO allowed outage time was exceeded. The inspectors noted that LCO

3.7.1.6 actually allowed the FRV to be in an inoperable condition for up to 78 hours before Mode 3 entry was required, and for an additional 6 hours until Mode 4 entry (non-applicable Mode for the LCO) was required. Therefore, the inspectors concluded that a condition prohibited by TS did not actually exist.

The inspectors observed that this LER was not submitted within 60 days of discovery of the condition as required by 10 CFR 50.73(a)(1). The inspectors reviewed NUREG-1022, “Event Reporting Guidelines 10 CFR 50.72 and 50.73,” revision 2, section 2.5, “Time Limits for Reporting,” which stated that “discovery date is generally the date when the event was discovered rather than the date when an evaluation of the event is completed.” However, since a condition or operation prohibited by TS did not actually exist, the inspectors determined that no violation of 10 CFR 50.73 occurred.

The inspectors also observed that Blocks 9 and 10 (Operating Mode and Power Level) of the LER indicated the incorrect information for the Unit at the time of the event, and that Block 2 of the LER indicated the incorrect date of the event. The failure to provide accurate information in these fields of the LER is a violation of 10 CFR 50.9, “Completeness and Accuracy of Information.” This failure to comply with 10 CFR 50.9 constitutes a violation of minor significance that is not subject to enforcement action in accordance with the NRC’s Enforcement Policy. The licensee entered this issue into their CAP as SR 404306.

This LER is closed.

b. Findings

No findings were identified.

.4 Unit 1 Automatic Reactor Trip

On June 26, 2011, the inspectors responded to an automatic reactor trip of Unit 1 due to a turbine trip from 100 percent power. The inspectors evaluated plant status, mitigating actions, and the licensee’s classification of the event, to enable the NRC to determine an appropriate NRC response. The inspectors discussed the trip with operations, engineering, and licensee management personnel to gain an understanding of the event and assess follow-up actions. The inspectors reviewed operator actions taken to determine whether they were in accordance with licensee procedures and TS, and reviewed unit and system indications to verify whether actions and system responses were as expected and designed. The inspectors 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) 46991, and documented in the licensee’s CAP as PER 393838.

4OA5 Other Activities

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

a. Inspection Scope

The inspectors reviewed the final report for the INPO plant assessment of Sequoyah conducted in June 2010. The inspectors reviewed the report to ensure that issues identified were consistent with the NRC perspectives of licensee performance, and determine if any significant safety issues were identified that required further NRC follow-up.

b. Findings

No findings were identified.

.2 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 were identified.

.3 (Closed) NRC Temporary Instruction 2515/183, "Followup to the Fukushima Daiichi Nuclear Station Fuel Damage Event"

a. Inspection Scope

The inspectors assessed the activities and actions taken by the licensee to assess its readiness to respond to an event similar to the Fukushima Daiichi nuclear plant fuel damage event. This included (1) an assessment of the licensee's capability to mitigate conditions that may result from beyond design basis events, with a particular emphasis on strategies related to the spent fuel pool, as required by NRC Security Order Section B.5.b issued February 25, 2002, as committed to in severe accident management guidelines, and as required by 10 CFR 50.54(hh); (2) an assessment of the licensee's capability to mitigate station blackout (SBO) conditions, as required by 10 CFR 50.63 and station design bases; (3) an assessment of the licensee's capability to mitigate internal and external flooding events, as required by station design bases; and (4) an assessment of the thoroughness of the walkdowns and inspections of important equipment needed to mitigate fire and flood events, which were performed by the licensee to identify any potential loss of function of this equipment during seismic events possible for the site.

b. Findings

Inspection Report 05000327/2011010 and 05000328/2011010 (ML111330368) documented detailed results of this inspection activity. Following issuance of the report, the inspectors conducted detailed follow-up on selected issues. No findings were identified during this follow-up inspection.

.4 (Closed) NRC Temporary Instruction 2515/184, "Availability and Readiness Inspection of Severe Accident Management Guidelines (SAMGs)"

On May 27, 2011, the inspectors completed a review of the licensee's severe accident management guidelines (SAMGs), implemented as a voluntary industry initiative in the 1990's, to determine (1) whether the SAMGs were available and updated, (2) whether the licensee had procedures and processes in place to control and update its SAMGs, (3) the nature and extent of the licensee's training of personnel on the use of SAMGs, and (4) licensee personnel's familiarity with SAMG implementation.

The results of this review were provided to the NRC task force chartered by the Executive Director for Operations to conduct a near-term evaluation of the need for agency actions following the Fukushima Daiichi fuel damage event in Japan. Plant-specific results for Sequoyah Nuclear Plant, Units 1 and 2, were provided as an Enclosure to a memorandum to the Chief, Reactor Inspection Branch, Division of Inspection and Regional Support, dated June 02, 2011 (ML111530328).

4OA6 Meetings

.1 Exit Meeting Summary

On June 3, 2011, the inspectors discussed the results of the radiation safety inspection with Mr. Michael Skaggs, Site Vice President, and other responsible staff.

On June 3, 2011, an exit meeting for the ISI portion of the inspection was conducted with licensee management and other licensee staff. An exit meeting for the SGISI portion was conducted on June 10, 2011, with the licensee management and other licensee staff. All proprietary material reviewed during the inspection was returned to the licensee.

On July 7, 2011, the resident inspectors presented the inspection results to Mr. K. Langdon and other members of his staff, who acknowledged the findings. The inspectors asked the licensee whether any of the material examined during the inspection should be considered proprietary. No proprietary information was identified.

4OA7 Licensee-identified Violations

The following violations of very low safety significance (Green) or Severity Level IV were identified by the licensee and are violations of NRC requirements which meet the criteria of the NRC Enforcement Policy, for being dispositioned as a Non-Cited Violation.

TS 6.12.1 requires that entryways into HRAs with dose rates not exceeding 1 rem/hour at 30cm be barricaded. Contrary to this, on May 31, 2011, and again on June 16, 2011, HRA entryways into U2 Containment were not barricaded. In both examples, the accessible areas of U2 Containment contained dose rates >100mrem/hr at 30cm, but less than 1 rem/hr at 30cm. On May 31, without HP present, workers in the area re-positioned the HRA swing gates to facilitate the installation of rail track into U2 Upper Containment. In the second example, on June 16, the HRA swing gate to U2 Lower Containment was propped open with scaffold leveling legs during demobilization. In both cases, the boundary re-positioning was such that the swing gates no longer provided adequate barriers to check the advance of an oncoming worker. These violations were discovered by HPTs performing their normal radiological control duties. Immediate corrective actions were taken upon discovery and documented in PERs 379547 and 390159. Although these events involved the failure to maintain proper controls for HRAs, this finding is of very low safety significance because there was no evidence of unauthorized worker entry into the affected areas, nor any unexpected radiation exposures to licensee personnel.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel
J. Armstrong, RP Support Supervisor
A. Bergeron, Operations Training
S. Bowman, Licensing Engineer
I. Collins, Engineering Programs
S. Connors, Operations Manager
A. Day, Chemistry Manager
R. Detwiler, Director, Safety and Licensing
C. Dieckmann, Manager, Maintenance
J. Dvorak, Outage and Site Scheduling Manager
D. Folsom, ISO Eddy Current Level III
D. Foster, Performance Improvement Manager
J. Furr, Quality Assurance Manager
Z. Kitts, Licensing
R. Krich, Licensing Vice President
K. Langdon, Plant Manager
F. Leonard, Level III Reactor Vessel Head Inspection
J. Mayo, Steam Generator Engineer
S. McCamy, Radiation Protection Manager
D. Porter, Operations Procedures
R. Proffitt, Licensing Engineer
J. Reidy, Operations Superintendant
P. Simmons, Work Control Manager
M. Skaggs, Site Vice President
D. Sutton, Licensing Engineer
N. Thomas, Licensing Engineer
R. Thompson, Emergency Preparedness Manager
G. Cook, Director, Safety and Licensing
C. Webber, TVA Corporate SG Program Manager
C. Ware, Training Director
K. Wilkes, Operations Support Superintendent
J. Williams, Site Engineering Director
S. Young, Site Security Manager

NRC personnel

S. Lingam, Project Manager, Office of Nuclear Reactor Regulation

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000327,328/2011003-01	NCV	Failure to Perform Instrumentation Surveillance Testing Within Required Frequency (Section 1R22)
05000327,328/2515/183	TI	Follow-up to the Fukushima Daiichi Nuclear Station Fuel Damage Event (Section 4OA5.3)
05000327,328/2515/184	TI	Availability and Readiness Inspection of Severe Accident Management Guidelines (SAMGs) (Section 4OA5.4)

Closed

05000327,328/2009-009-00, -01	LER	Unanalyzed Condition Affecting Probable Maximum Flood (PMF) Level (Section 4OA3.1)
05000327,328/2011-001-00	LER	Both Trains of Control Room Air Conditioning Systems Inoperable (Section 4OA3.2)
05000327/2011-002-00	LER	Feedwater Regulator Valve Inoperable (Section 4OA3.3)

LIST OF DOCUMENTS REVIEWED

Section R01: Adverse Weather Protection

Procedures
AOP-N.03, Flooding, Revision 32
0-PI-OPS-510-001.0, Flood Preparation Equipment Inventory, Revision 8

Section R04: Equipment Alignment

Partial System Walkdowns

Procedures
0-SI-OPS-082-007.0, Diesel Generator Operability Verification, Revision 10
0-GO-14-7, Outside AUO Operator Rounds, Revision 35
0-SO-82-8, Diesel Generator 2B-B Support Systems, Revision 15
0-SO-82-4, Diesel Generator 2B-B, Revision 35
2-SO-3-2, Auxiliary Feedwater System, Revision 38
0-SO-78-1, Spent Fuel Pit Coolant System, Rev. 49
0-SO-78-1 Attachment 2, Spent Fuel Pit Coolant System Power Checklist
0-SO-78-1 Attachment 5, Spent Fuel Pit Coolant System Valve Checklist

Complete System Walkdown

Procedures
0-SO-30-1, Control Building Heating, Air Conditioning and Ventilation, Revision 37
0-SO-30-2, Control Room Isolation, Revision 14

Work Orders
08-777122 – “B” Train MCR chiller has tripped and will not restart. Investigate and repair
08-777133 – Inspect relay connectors and logic cards to ensure they are not coming loose on a bi-weekly basis.
08-778582 – Evaluate air regulator and diaphragm for replacement
08-779987 – MCR Air Handling Unit Motor 1B-B appears to have seized up
09-770258-1 – Chiller tripped on low discharge pressure. Investigate cause/repair
09-777093 – “A” Main Control room chiller not loading up (not cooling).
111940742 – Reconfigure Temperature Switch wiring
111940804 – Calibration of SQN-0-TC-311-0039

PERs
149057 - Request FE for MCR Chiller Relays
316871 - Failed motor must be shipped to Power Service Shop for failure analysis
153304 - Unplanned LCO Entry TS 3.0.5 and TS 3.7.15
161252 - Found the A MCR chiller tripped on low discharge pressure and entered unplanned LCO 3.7.15 action a
173196 - A Main Control room chiller not loading.
213177 - Both units entered unplanned LCO 3.0.3 due to 'A' MCR chiller failing to start when placing in-service with 'B' MCR tagged due to failed PMT for WO 09-772143-000.

323782 - "B" Main control room chiller (Upper Tier)
323785 - "B" Main Control Room Chiller tripped
325906 - Latent Vulnerabilities in the MCR Chiller Control logic.
147723 - MCR B Chiller relay PM issue
149022 - Unplanned LCO Entry
149695 - Extended LCO Times
150986 - A-A MCR Chiller Relay Board Replacement
159359 - Unplanned TRM LCO entered on both units
173196 - A MCR chiller INOP
176798 - MCR 'A' Chiller exceeded leak rate
177663 - B Main control chiller head leak.
276746 - MCR and EBR Chiller placed in Maintenance A1 Status (Lower tier)
323780 - "A" Main Control Room Chiller
334231 - Small Oil leak under MCR chiller A

Other documents
Mechanical Drawings: 47W866-4, 47W867-2, 47W867-4, 47W931-1
Logic Drawings: 47W611-31-1
Flow Drawings: 47-W865-3
FSAR Sections 9.4.1 & 6.4

Section R05: Fire Protection

Procedures
FPDP-1, Conduct of Fire Protection, Revision 2
0-PI-FPU-317-299.W, Att. 8, Shift Check List, Revision 32
NPG-SPP-18.4.7, Control of Transient Combustibles, Rev. 0
EITP-100, Environmental Compliance, Rev. 6
0-SI-FPU-410-703.0, Inspection of FPR Required Fire Doors, Rev. 5
SQN-FPR-Part-II, SQN Fire Protection Report Part II – Fire Protection Plan, Revision 28
0-PI-FPU-026-538.R, Fire Extinguisher Installation and Removal, Revision 3

PERs
390623 - Evaluate procedure enhancement to 0-GO-1 and 0-PI-FPU-026-538.R
390613 - Revise 0-PI-FPU-026-538.R to update references
381101 – App-R BPEL Light units found without power

FORs
110082
110083
080183
100434
080183
110101
110006
100439
100448

Other documents
AUX-0-690-00, Fire Protection Pre-Fire Plans Auxiliary Building - El. 690, Revision 2
AUX-0-690-01, Fire Protection Pre-Fire Plans Auxiliary Building - El. 690 (Unit 1 Side), Revision 7
AUX-0-690-02, Fire Protection Pre-Fire Plans Auxiliary Building - El. 690 (Unit 2 Side), Revision 7
AUX-0-714-00, Fire Protection Pre-Fire Plans Auxiliary Building - El. 714, Revision 3
AUX-0-714-01, Fire Protection Pre-Fire Plans Auxiliary Building - El. 714 (Col. A1-15, Q-U), Revision 6
AUX-0-714-02, Fire Protection Pre-Fire Plans Auxiliary Building - El. 714 (Col. U-X and U1 & U2 Additional Equip. Builds), Revision 7

Section R08: Inservice Inspection Activities

Procedures
N-UT-66, Rev. 06, Generic Procedure for the Ultrasonic Examination of Weld Overlaid Austenitic Pipe Welds
N-UT-76, Rev. 07, Generic Procedure for the Ultrasonic Examination of Ferritic Pipe Welds SPP-9.1, Rev. 09, ASME Section XI
N-VT-17 Revision 7, Visual Examination for Leakage of PWR Reactor Head Penetrations
SI-DXI-000-114.3, Rev. 15, ASME Section XI/NDE Program Unit 1 and Unit 2
TI-DXX-000-097.1, Rev. 6, Boric Acid Control Program
SPP-9.7, Rev. 18, Corrosion Control Program
PI-DXX-068-100.R, Rev. 2, Monitoring of Reactor Head Canopy Seal Welds for Leakage
PI-SLT-068-200.0, Rev. 2, Reactor Building Post Shutdown Leakage Examination
PI-DXX-000-105, Rev. 0, Boric Acid Leak Monitoring Program

Corrective Action Documents
SR180482, Only 1 Qualified SG Engineer in TVA, dated 05/17/2010
SR180502, No Designated SG Maintenance Manager, dated 05/17/2010
SR382333, FOSAR in Steam Generator #4 Identified 3 Loose Objects, dated 06/06/2011
SR381443, Possible Indication during Eddy Current Testing, dated 06/05/2011
SR383230, Loss of FME Control on SG #3, dated 06/07/2011
SR380104, Dry, White Boric Acid Leak on Unit 2 Seal Table, dated 6/2/2011
PER 290082 RHR Pump 1B-B, dated 11/30/10
PER 286694 Wet Boric Acid Leak on 0-VLV-0322A, dated 11/18/1010
PER 232296 Active borated water leak from 2-VLV-062-0522 valve packing, dated 6/2/2010
PER 258510 CVCS snubber found out of tolerance during Section XI exam, dated 9/28/2010
PER 327722 ASME Code program enhancements, dated 2/23/2011
PER 226669 ANII Open items for ISI programs, dated 4/23/2010

Other
AREVA Personnel Certifications – Submittals #1 – #6, dated March 18, 2011
WO 110993041 SQN-2-VLV-062-525, CENT CHRG PMP DISC CK
WO 110993043 SQN-2-VLV-062-532, CENT CHRG PMP CK
TVA Site Welder Qualifications and Certifications
Ultrasonic Testing (UT) of SIS-244 in 10 inch Safety Injection piping records
Visual Acuity records of NDE personnel
Qualification/Certification Records of NDE personnel
Certificate of Calibration of Krautkramer Unit E37689

Krautkramer Transducer Certification Record for 00FCY6
Certificate of Calibration of Krautkramer Unit E37690
Krautkramer Transducer Certification Record for 00F8T6
Borated Water Leak Assessment for WO 111593024 1-PMP-074-0020, dated 11/30/10
Boric Acid Leakage Evaluation for WO 112087504 2FCV-068-333-A, dated 5/28/11
Boric Acid Leakage Evaluation for WO 111466271 SQN-1-Driv-063-0342C, dated 10/15/10

Section R11: Licensed Operator Requalification

Procedures
0-GO-6, Power Reduction from 30% Reactor Power to hot Standby, Revision 47
0-GO-7, Unit Shutdown Hot Standby to Cold Shutdown, Revision 65

Section R12: Maintenance Effectiveness

Procedures
TI-4, Maintenance Rule Performance Indicator Monitoring, Trending, and Reporting – 10CFR50.65, Revision 23

Work Orders

PERs

Other documents

Section R13: Maintenance Risk Assessments and Emergent Work Evaluation

Procedures
0-TI-DSM-000-007.1, Risk Assessment Guidelines, Revision 9
NPG-SPP-07.3, Work Activity Risk Management Process, Revision 3
NPG-SPP-07.2.4, Forced Outage or Short Duration Planned Outage Management, Revision 0
NPG-SPP-07.2, Outage Management, Revision 0
GOI-6, Apparatus Operations, Revision 142

Other documents
SQN-0-11-045, LCO 4.0.3 Risk Assessment/RMAs for SSPS testing

Section R15: Operability Evaluations

Procedures
NEDP-22, Functional Evaluations, Rev. 9
OPDP-8, Limiting Conditions for Operation Tracking, Rev. 5
NPG-SPP-03.5, Regulatory Reporting Requirements, Revision 2

Work Orders

PERs
385549 - 2A SI Pump performance curve below Design Min values at one point
340794 - ERCW instrument line supports
332977 - EDG past operability due to fire/flood mode pump testing
340456 - Probable Maximum Flood flow path through ERCW instrumentation well vent piping

Other documents
SR 347208, FRV 3-90 past operability/reportability

Section R18: Plant Modifications

Procedures
NPG-SPP-09.3, Plant Modifications and Engineering Change Control, Revision 4
NPG-SPP-09.4, 10 CFR 50.59 Evaluations of Changes, Tests, and Experiments, Revision 1
NPG-SPP-09.5, Temporary Alterations, Revision 0
0-SI-OPS-082-007.W, AC Electrical Power Source Operability Verification, Rev. 19
PMTI-22582-003, Unit 2 Start Bus Manual Transfer Scheme, Revision 0
1-SO-63-5, Emergency Core Cooling System, Rev. 57
1-AR-M6-DAuxiliary Systems 1-XA-55-6D, Rev. 36

Work Orders

PERs
SR 380465, 2B Rod Drive MG run during temporary outage power lineup
332950, RHR discharge header pressurization

Other documents
DCN D22582A, Unit 2 start bus modifications
TACF 1-11-005-063, Installation of a continuous vent on RHR discharge pipe

Drawings
11291-196 DCN 22582 Sketch No. 1 Page No. 1, Unit 2 Start Bus Alignment for Replacement of Unit 2 Start Bus & CSST B Buswork
1,2-15E500-3, Transformer Taps & Voltage Limits – Aux Power System, Rev. 23
1-47W435-1, Mechanical Safety Injection System Piping, Rev. 8
MDQ00006320100240, Calculation of effects of gas accumulation in ECCS piping, Rev. 0

Section R19: Post Maintenance Testing

Procedures
MMDP-1, Maintenance Management System, Revision 20
MMDP-3, Guidelines for Planning and Execution of Troubleshooting Activities, Revision 6
NPG-SPP-6.5, Foreign Material Control, Revision 0
NPG-SPP-6.1, Work Order Process Initiation, Revision 0
NPG-SPP-06.3, Pre-/Post-Maintenance Testing, Revision 0
NPG-SPP-06.9, Testing Programs, Revision 0

NPG-SPP-06.9.1, Conduct of Testing, Revision 1
NPG-SPP-06.9.3, Post-Modification Testing, Revision 0
OPDP-7, Fuse Control, Revision 4
PMTI-22582-003, Unit 2 Start Bus Manual Transfer Scheme, Revision 1
0-SI-SLT-31C-258.1, Containment Isolation Valve Local Leak Rate Test Chilled Water System, Revision 8
0-TI-SXI-000-200.CV, Check Valve Condition Monitoring Program, Revision 5

<u>Work Orders</u>
112257547 - Fuse blew during performance of Reactor Trip Instrumentation Functional Test (SSPS) Train B
09-777459-000 - DCN 22582 - Phase 2 - Replace Start Bus 2A and 2B, Modify controls to MBB Scheme
09-777459-003 - DCN 22582, stage 3 - Revise the control circuits for Unit 2 start bus breakers to MBB Scheme
09-777459-002 - DCN 22582, stage 2 - Replace Start Bus 2A and 2B
09-777459-009 - DCN 22582, Stage 4 - Remove and Replace CSST B Buses B1 and B2
110851792 – Disassemble and Inspect Inboard CIV (check valve VLV-31C-697) for Incore Instrument Room Cooler B Supply
110774613 - Disassemble and Inspect Inboard CIV (check valve VLV-31C-715) for Incore Instrument Room Cooler B Return
111643152 – Calibrate high steam flow isolation to Terry Turbine on Unit 1
112272300 – Replace Diesel Engine 2A1 Fuel Oil Transfer Pump Motor
07780697001 - Rebuild Spent Fuel Pit Pump A

<u>PERs</u>
368185 - Fuse blew while changing bulb during performance of 1-SI-IFT-099-90.8B
370549 - SE Walkdown – Eng 2A1 Fuel Oil Transfer Pump Motor

<u>Section R20: Refueling and Outage Activities</u>

<u>Procedures</u>
FHI-3, Movement of Fuel, Revision 65
0-GO-15, Containment Closure Control, Revision 34
0-GO-13, Reactor Coolant System Drain and Fill Operations, Revision 71
0-TI-OXX-068-001.0, Reactor Coolant System Hot Leg Vents and Generic Letter 88-17 Issues, Revision 17
0-GO-5, Normal Power Operation, Revision 73
0-GO-6, Power Reduction from 30% Reactor Power to hot Standby, Revision 47
0-GO-7, Unit Shutdown Hot Standby to Cold Shutdown, Revision 65
NPG-SPP-08.1, Nuclear Fuel Management, Revision 00
0-PI-OPS-000-187.0, Containment Inspection, Revision 1
0-GO-2, Unit Startup from Hot Standby to Reactor Critical, Revision 35
0-GO-3, Power Ascension from Reactor Critical to less than 5 percent Reactor Power, Revision 26
0-PI-OPS-000-011.0, “Containment Access Control During Modes 1-4, Revision 1
NPG-SPP-07.2.3, Plant Startup Review/Checklists, Rev 1

0-RT-NUC-000-003.0, Low Power Physics Testing, Rev. 23
NPG-SPP-03.21, Fatigue Management and Work Hour Limits, Rev. 2

<u>Work Orders</u>
112394681 – Unit 2 Elevation 690 Containment Access Control
112397001 - Unit 1 Elevation 690 Containment Access Control

<u>PERs</u>
385495 - Clarification needed for NPG-SPP-08.1 Appendix E Section 3.0 D
225364 - SPP-10.8 Qualification of FHS/SROs.

<u>Other documents</u>
Unit 2 Cycle 17 Outage Safety Plan Rev. C Schedule, Revision 0
Tagout: 2-TO-2011-0030, Clearance: 2-3-0035-RFO, TDAFW Pump
Mode Change Checklist for Mode 3 To Modes 1 and 2

<u>Section R22: Surveillance Testing</u>

<u>Procedures</u>
NPG-SPP-06.9.1, Conduct of Testing, Revision 1
1-SI-SXP-074-201.A, RHR Pump 1A Section XI Test, Revision 16
2-SI-OPS-082-026.A, Loss of Offsite Power with Safety Injection – Diesel Generator 2A-A Test, Revision 42
1-SI-IFT-099-90.8A, Reactor Trip Instrumentation Monthly Functional Test (SSPS) Train A, Revision 19
1-SI-IFT-099-90.8B, Reactor Trip Instrumentation Monthly Functional Test (SSPS) Train B, Revision 16
0-SI-SLT-067-258.2, Containment Isolation Valve Local Leak Rate Test Lower Compartment Essential Raw Cooling Water, Revision 11
0-SI-MIN-061-109.0, Ice Condenser Intermediate and Lower Inlet Doors and Vent Curtains, Revision 5
0-SI-MIN-061-105.0, Ice Condenser - Ice Weighing, Revision 7
2-SI-SXP-003-202.B, Motor Driven Auxiliary Feedwater Pump 2B-B Comprehensive Performance Test, Rev. 5
2-SI-IFT-068-456.0, Functional Test of RCS Cold Overpressurization Protection System PORV PCV-68-334, Rev. 16

<u>Work Orders</u>
111595122 - 1-SI-IFT-099-90.8A U1 Rx Trip Inst FT (SSPS) Train A
111721455 - 1-SI-IFT-099-90.8B U1 Rx Trip Inst FT (SSPS) Train B
112257547 - Fuse blew while changing bulb during performance of 1-SI-IFT-099-90.8B
111857728 - 0-SI-SLT-067-258.2 U2 LCC ERCW LLRT
111857384 - 0-SI-MIN-061-105.0 U2 Ice Condenser - Ice Weighing (as left)

<u>PERs</u>
368185 - Fuse blew while changing bulb during performance of 1-SI-IFT-099-90.8B
368204 - Fuse blew during performance of 1-SI-IFT-099-90.8B

377163 – Problems encountered during 2B MDAFWP testing
377953 – Problems encountered during 2B MDAFWP testing

Section 2RS1: Radiological Hazard Assessment and Exposure Controls

Procedures, Guidance Documents, and Manuals
NPG-SPP-05.1, “Radiological Controls”, Rev. 2
RCDP-10, “Personnel Contamination Reporting”, Rev. 4
0-TI-NUC-000-002.0, “Storing Material in Spent Fuel Pool or New Fuel Vault”, Rev. 17
RCI-15, “Radiological Postings”, Rev. 19
RCI-24, “Control of Very High Radiation Areas”, Rev. 11
RCI-28, “Control of Locked High Radiation Areas”, Rev. 11
RCI-29, “Control of Radiation Protection Keys”, Rev. 11
RCI-32, “Alpha Contamination Monitoring and Controls”, Rev. 1
RCI-412, “Radiation Protection Surveys during Initial Spent Fuel Assembly Movement”, Rev. 0
RCI-201, “Radiation and Contamination Surveys”, Rev. 7
RCI-202, “Airborne Radioactivity Surveys”, Rev. 4
RCI-204, “Radiological Surveys of Equipment and Materials Leaving the RCA”, Rev. 2
RCI-208, “Hot Particle Controls”, Rev. 1
NPG-SPP-03.1, “Corrective Action Program”, Rev. 1

Records and Data
RWP 10044110, U-1 Upper CTMT Cleanup/decon, Rev. 1
RWP 11027100, U2 Lower CTMT SGR Support Work, Rev. 1
RWP 11027010, U2 Lower Containment Scaffolding/Insulation, Paint, Etc.
RWP 11001023, Sort High Rad Trash, Rev. 0
RWP 11037004, U2 S/G Nozzle Dam Installation, Rev. 0
RWP 11037002, U2 S/G Manway Insert/Removal, Rev. 0
Radiological Survey 052511-17, Rx Head Survey
Radiological Survey 053111-49, U2 S/G 1&4 platform & S/G Insert Smears
Radiological Survey 103010-24, R152 U1 Upper CTMT TriNuc filter
Radiological Survey 60311001, U2 Upper GA air sample
Radiological Survey 60211020, U2 Upper GA air sample
Radiological Survey 053111-3, U2 Accumulator Room #4
Radiological Survey 051611-14, Waste Package Area Compactor Room
Radiological Survey 041911-4, Waste Package Area Compactor Room
Radiological Survey 53111011, U2 S/G 3 Primary Platform Air Sample
Radiological Survey 53111012, U2 S/G 2 Primary Platform Air Sample
Radiological Survey 052511-12, RHR Pump Room 2A-A
Radiological Survey 22511001, Waste Gas Valve Gallery Gas Grab Sample
Sequoyah Nuclear Plant – Special Monitoring Program, Annual Radionuclide Trending and Assessment Report for 2009 Calendar Year
ALARA Plan 2011-011, S/G Primary and Secondary Maintenance/Inspection
2011 National Source Tracking System Annual Inventory Reconciliation, ID 6017
Work Order 111184189, Byproduct Material Inventory and Sealed Source Leak Test
Small Article Monitor #860495 Calibration Forms, 7/26/10 and 1/21/11
ARGOS-5AB #860588 Calibration Data Sheets, 9/14/11 and 11/23/11

CAP Documents
QA-SQ-11-004, Assessment of Radioactive Contamination Control, Control of Radioactive Materials, and Radiation Protection Measurement
SQN-RP-S-11-39, Snapshot Self-Assessment, Radiological Hazard Assessment and Exposure Controls
PER 379547
PER 390159
PER 345980
PER 378011
PER 375752
PER 330983
PER 285850
PER 276799
PER 315450
PER 276878
PER 378160

Section 4OA1: Performance Indicator Verification

Procedures
NPG-SPP-02.2, Performance Indicator Program, Revision 2
NEI 99-02, Regulatory Assessment Performance Indicator Guideline, Revision 6
NPG-SPP-02.2, “Performance Indicator Program”, Rev. 2

PERs
PER 378844
PER 318603
PER 378839

Other Documents
ED Alarm Logs, 6/1/10 – 6/1/11

Section 4OA2: Identification and Resolution of Problems

Procedures
NPG-SPP-03.1, Corrective Action Program, Revision 1
NPG-SPP-03.1.4, Corrective Action Program Screening and Oversight, Rev. 3
NPG-SPP-03.1.7, PER Actions, Rev. 1
NPG-SPP-01.15, Service Request Initial Review, Rev. 2
OPDP-8, Limiting Conditions for Operation Tracking, Rev. 5
NPG-SPP-03.1.3, Regulatory Screening, Rev. 1

Work Orders

PERs
314771 – feedwater regulating valve apparent cause evaluation
284451 – FRV excessive leakage

Other documents

Section 4OA3: Event Follow-up

Procedures
0-TI-QXX-000-001.0, Event Critique, Post Trip Report, and Equipment Root Cause, Rev. 11

Work Orders
112396996, main steam dump valve controller
09-771564-000, Diesel Generator Building flood compensatory actions
09-777032-000, replace FRV diaphragm
09-771161-000, replace FRV positioned
111644616, verify FRV on seat

PERs
323782 - Main Control Room “B” chiller – Failure to Start
162711 - Errors in Hydrology Analyses
201568 – Flood studies with a TVA model indicates PMF higher elevations at TVA dams
227488 – Hydrology Probable Maximum Flood (PMF) Calculations
162711 – River System Operation Impact on PMF
SR 393260 – Unit 1 Reactor Trip
314771 – feedwater regulating valve apparent cause evaluation
284451 – FRV excessive leakage

Other documents
Apparent Cause Evaluation Report – Main Control Room “B” chiller – Failure to Start, dated 4/6/11
Apparent Cause Evaluation Report – Errors in Hydrology Analyses dated 9/10/10
Licensee Event Report (LER) 05000327, 328/2011-001-00, Both Trains of Control Room Air Conditioning Systems Inoperable dated 4/15/11
Licensee Event Report (LER) 05000327, 328/2009-009-00 and -01, Unanalyzed Condition Affecting Probable Maximum Flood (PMF) Level dated April 14, 2010
Functional Evaluation 43152, River System Operation Impact on PMF, Revision 3
SR 393260 Unit 1 Reactor Trip Report
Standing Order SO-09-006, Probable Maximum Flood Issue, Rev. 1
SQN-DC-V-12.1, Sequoyah Nuclear Plant – Flood Protection Provisions, Rev. 10
CDQ000020080054, PMF Determination for Tennessee River Watershed, Rev. 0

Section 4OA5: Other Activities

TRN-34, Severe Accident Management Training, Revision 5
NPG-SPP-18.3.1, Severe Accident Management Guideline (SAMG) Program Administration, Revision 0
EPT500.032, Severe Accident Management Guideline Training, Revision 1
Training records for SAMG Annual Training 2008, 2009, 2010, and 2010
SPM-SAMG-0, Deviation Document for Severe Accident Management Guidelines, Revision 0
NPG-SPP-09.3, Plant Modifications and Engineering Change Control, Rev. 4

LIST OF ACRONYMS

2R17	Unit 2 Refueling Outage 17
CAP	Corrective Action Program
CFR	Code of Federal Regulations
ED	Electronic Dosimeter
HPT	Health Physics Technician
HP	Health Physics
HRA	high radiation area
IP	Inspection Procedure
LHRA	locked high radiation area
NEI	Nuclear Energy Institute
No.	Number
NSTS	National Source Tracking System
PERs	Problem Evaluation Report
PI	Performance Indicator
PM	portal monitor
QA	Quality Assurance
Radwaste	Radioactive Waste
RCA	radiologically controlled area
Rev.	Revision
RS	Radiation Safety
RWP	radiation work permit
S/G	Steam Generator
SFP	Spent Fuel Pool
TI	Temporary Instruction
TLDs	thermoluminescent dosimeters
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
U1	Unit 1
U2	Unit 2
VHRA	very high radiation area