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

January 30, 2012

Mr. Joseph W. Shea
Manager, Corporate Nuclear Licensing
Tennessee Valley Authority
3R Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

SUBJECT: SEQUOYAH NUCLEAR PLANT - NRC INTEGRATED INSPECTION REPORT 05000327/2011005, 05000328/2011005, 05000327/2011501, and 05000328/2011501

Dear Mr. Shea:

On December 31, 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 January 5, 2011 with Mr. P. Simmons and other members of the Sequoyah 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.

One NRC-identified and two self-revealing findings of very low safety significance (Green) were identified during this inspection.

All of these findings were determined to involve violations of NRC requirements. Further, a licensee-identified violation which was determined to be of very low safety significance (Green) is listed in this report. The NRC is treating these violations as non-cited violations (NCVs) consistent with Section 2.3.2 of the Enforcement Policy.

If you contest these non-cited violations, 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.

If you disagree with a cross-cutting aspect assignment 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.

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 Agencywide Document Access and Management System (ADAMS). ADAMS is accessible from the NRC Website at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Scott M. Shaeffer, Chief
Reactor Projects Branch 6
Division of Reactor Projects

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

Enclosure: Inspection Report 05000327/2011005, 05000328/2011005, 05000327/2011501, 05000328/2011501
w/Attachment: Supplemental Information

cc w/encl: (See page 3)

disagreement, to the Regional Administrator, Region II, and the NRC Resident Inspector at the Sequoyah Nuclear Plant.

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

Sincerely,

/RA/

Scott M. Shaeffer, Chief
Reactor Projects Branch 6
Division of Reactor Projects

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

Enclosure: Inspection Report 05000327/2011005, 05000328/2011005, 05000327/2011501, 05000328/2011501
w/Attachment: Supplemental Information

cc w/encl: (See page 3)

X PUBLICLY AVAILABLE ☐ NON-PUBLICLY AVAILABLE ☐ SENSITIVE X NON-SENSITIVE

ADAMS: X Yes ACCESSION NUMBER:__ML120300579___________ X SUNSI REVIEW COMPLETE X FORM 665 ATTACHED

OFFICE	RII:DRP	RII:DRP	RII:DRP	RII:DRP	RII: DRS	RII:DRS	
SIGNATURE	Via email	Via email	/RA/	SMS	Via email	Via email	
NAME	CYoung	WDeschaine	JHamman	SShaeffer	RBaldwin	MSpeck	
DATE	1/26/2012	1/26/2012	1/26/2012	01/30/2012	1/26/2012	1/26/2012	
E-MAIL COPY?	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO

OFFICIAL RECORD COPY DOCUMENT NAME: G:\DRPII\RPB6\SEQUOYAH\REPORTS\2011\005\SEQUOYAH 2011005 REV 1.DOCX

cc w/encl:
J. T. Carlin
Site Vice President
Sequoyah Nuclear Plant
Tennessee Valley Authority
Electronic Mail Distribution

P. R. Simmons
Plant Manager
Sequoyah Nuclear Plant
Tennessee Valley Authority
Electronic Mail Distribution

G. M. Cook
Manager
Licensing
Sequoyah Nuclear Plant
Electronic Mail Distribution

C. D. Mackaman
Tennessee Valley Authority
Electronic Mail Distribution

E. J. Vigluicci
Assistant General Counsel
Tennessee Valley Authority
Electronic Mail Distribution

County Mayor
Hamilton County
Hamilton County Courthouse
208 Courthouse
625 Georgia Avenue
Chattanooga, TN 37402-2801

Tennessee Department of Environment & Conservation
Division of Radiological Health
401 Church Street
Nashville, TN 37243

Senior Resident Inspector
U.S. Nuclear Regulatory Commission
Sequoyah Nuclear Plant
2600 Igou Ferry Road
Soddy Daisy, TN 37379-3624

Ann Harris
341 Swing Loop
Rockwood, TN 37854

Letter to J. W. Shea from Scott Shaeffer dated January 30, 2012

SUBJECT: SEQUOYAH NUCLEAR PLANT - NRC INTEGRATED INSPECTION REPORT 05000327/2011005, 05000328/2011005, 05000327/2011501, 05000328/2011501

Distribution w/encl:
C. Evans, RII EICS (Part 72 Only)
L. Douglas, RII EICS (Linda Douglas)
OE Mail (email address if applicable)
RIDSNRRDIRS
PUBLIC
R. Pascarelli, NRR ((Regulatory Conferences Only))
RidsNrrPMSequoyah Resource

U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos.:	50-327, 50-328
License Nos.:	DPR-77, DPR-79
Report Nos.:	05000327/2011005, 05000328/2011005, 05000327/2011501, 05000328/2011501
Licensee:	Tennessee Valley Authority (TVA)
Facility:	Sequoyah Nuclear Plant, Units 1 and 2
Location:	Sequoyah Access Road Soddy-Daisy, TN 37379
Dates:	October 1, 2011 through December 31, 2011
Inspectors:	C. Young, Senior Resident Inspector W. Deschaine, Resident Inspector J. Hamman, Project Engineer (4OA1) R. Baldwin, Senior Operations Engineer (1R11.2) M. Speck, Senior Emergency Preparedness Inspector (1EP2, 1EP3, 1EP4, 1EP5, 4OA1, 4OA5)
Approved by:	Scott M. Shaeffer, Chief Reactor Projects Branch 6 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000327/2011005, 05000328/2011005, 05000327/2011501, 05000328/2011501; 10/1/2011 – 12/31/2011; Sequoyah Nuclear Plant, Units 1 and 2; Surveillance Testing, Identification and Resolution of Problems, and Event Followup.

The report covered a three-month period of inspection by resident inspectors and announced inspections by regional inspectors. Three Green findings were identified which involved non-cited violations (NCVs) 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. A self-revealing non-cited violation of Unit 2 Technical Specification (TS) 6.8.1.a was identified for the licensee’s failure to follow station procedures during the performance of a surveillance testing activity. While performing degraded voltage/load shed relay testing associated with the 2B 6.9kV shutdown board, the use of improper test equipment and the incorrect connection of test equipment resulted in a control power circuit fuse being blown, which caused inoperability of an emergency diesel generator and a motor driven auxiliary feedwater train. This issue was entered into the licensee’s corrective action program as Problem Evaluation Report (PER) 415324.

  The finding was determined to be greater than minor because it was associated with the human 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, the failure to follow procedure steps resulted in inoperability of the 2B emergency diesel generator and the 2B motor driven train of auxiliary feedwater. Using Inspection Manual Chapter (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 cause of this finding was determined to have a cross-cutting aspect in the area of Human Performance associated with the Work Practices component. The licensee failed to adequately implement human error prevention techniques such as self and peer checking (e.g. concurrent verification) while connecting test equipment. Additionally, maintenance personnel failed to question the use of test equipment which was different than what was specified in the procedure (i.e. proceeding in the face of uncertainty). [H.4(a)]. (Section 1R22)

- Green. A self-revealing non-cited violation of Unit 2 TS 3.0.3 was identified for the licensee's failure to place the unit in Mode 3 within seven hours when a Limiting Condition for Operation (LCO) was not met in Modes 1 and 2. The requirements of LCO 3.3.1, "Reactor Trip System Instrumentation," associated with the power range neutron flux function in Modes 1 and 2 were not met for a period of approximately 24 hours. This was the result of an error made during the performance of a channel calibration activity, which caused one channel to be left in an inoperable condition. This issue was entered into the licensee's corrective action program as PER 397142.

  The finding was determined to be greater than minor because it was associated with the human 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, the inoperability of the N44 power range Nuclear Instrucment System (NIS) channel low range neutron flux trip function resulted in the failure to meet TS operability requirements associated with reactor trip system instrumentation. Using Inspection IMC 0609, "Significance Determination Process," (SDP) Attachment 4, "Phase 1 - Initial Screening and Characterization of Findings," the finding was determined to be applicable to Phase 2 SDP screening since it represented the loss of a channel of a TS required function for greater than its TS allowed outage time. A Phase 2 analysis using Saphire 8 software with the Sequoyah SPAR model in the SDP mode was performed by a regional SRA. Using an exposure period of 1 day with a truncation value of 1E-13, a result of ΔCDF << 1E-6, or very low safety significance (Green), was obtained. The cause of this finding was determined to have a cross-cutting aspect in the area of Human Performance associated with the Work Practices component. The licensee failed to adequately implement human error prevention techniques, such as self and peer checking, to ensure that the work activity was being performed on the correct component. [H.4(a)]. (Section 4OA3.1)

Cornerstone: Barrier Integrity

- Green. The inspectors identified a non-cited violation of Unit 1 Operating License DPR-77 Condition 2. (C).1 "Maximum Power Level" for the licensee's failure to take prudent action to ensure that the licensed power limit was not exceeded during a pre-planned evolution which involved manual reactivity manipulations. Prompt action was not taken by operators to reduce power when reactor thermal power exceeded the licensed power limit during a control rod full out position reset activity. Additionally, prudent action to sufficiently reduce power prior to the activity to accommodate the power transient was not taken. This issue was entered into the licensee's corrective action program as PER 437068.

  The finding was determined to be greater than minor because it was sufficiently similar to example 8.a. of IMC 0612 Appendix E, in that: 1) prudent action based on prior performance was not taken to reduce power prior to performing the evolution, and 2) operators did not promptly lower thermal power once the licensed limit was exceeded. Additionally, if left uncorrected the finding would have the potential to lead to a more significant safety concern, since operation above the licensed power limit has the potential to reduce analyzed margins to fuel cladding damage. Using 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) under the barrier integrity cornerstone since only the fuel cladding barrier criterion was applicable. The cause of this finding was determined to have a cross-cutting aspect in the area of Human Performance associated with the Decision-Making component. Elements of non-conservative decision making which contributed to this performance deficiency included: (1) Prior to February 2011, station procedures required operators to monitor and maintain the 10-minute average of thermal power below the licensed limit. These requirements were revised and replaced with requirements to maintain the 1-hour average of thermal power below the licensed limit. This was a non-conservative decision made without due consideration of potential consequences or the need for supplemental guidance to maintain an appropriate and conservative approach to controlling thermal power under non-steady state conditions. (2) The decision was made to proceed with the rod withdrawal activity under the non-conservative assumption that there would be negligible reactivity effects, and without considering available data from previous performances. This did not reflect a philosophy of demonstrating that a proposed activity is safe in order to proceed rather than demonstrating that it is unsafe in order to disapprove the action. [H.1(b)] (Section 4OA2.4)

B. <u>Licensee-Identified Violations</u>

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). That 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 RTP for the entire inspection period.

Unit 2 operated at or near 100 percent RTP for the entire inspection period.

1. REACTOR SAFETY

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

1R01 Adverse Weather Protection

.1 Readiness for Seasonal Extreme Weather Conditions

a. Inspection Scope

The inspectors reviewed design features and licensee preparations for protecting the essential raw cooling water (ERCW) intake structure and both Unit 1 and 2 refueling water storage tanks (RWSTs) from extreme cold and freezing conditions. The inspectors reviewed the Updated Final Safety Analysis Report (UFSAR) and Technical Specifications (TS), reviewed implementation of licensee freeze protection procedures, walked down portions of the systems to assess deficiencies and system readiness for extreme cold weather and discussed prioritization and status of correcting deficiencies with licensee personnel. The inspectors completed one sample.

b. Findings

No findings were identified.

1R04 Equipment Alignment

.1 Partial System Walkdown

a. Inspection Scope

The inspectors performed a partial walkdown of the following two systems to verify the operability of redundant or diverse trains and components when safety equipment was inoperable. The inspectors 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 hem into the corrective action program (CAP). Documents reviewed are listed in the Attachment. The inspectors completed two samples.

- 1A-A Safety Injection Pump while the 1B-B Diesel Generator and associated support systems where inoperable for maintenance
- Unit 2 B-train Containment Spray System during A-train planned maintenance

.2 Complete System Walkdown

a. Inspection Scope

The inspectors performed a complete walkdown of the Units 1 and 2 Auxiliary Feedwater System 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 to this report. 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 four 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 four samples.

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

b. Findings

No findings were identified.

.2 Annual Drill Observations

a. Inspection Scope

On December 13, 2011, the inspectors observed an unannounced fire drill in the turbine building railroad bay, 1A 6.9kV Unit Board. The inspectors assessed fire alarm effectiveness; response time for notifying and assembling the fire brigade; the selection, placement, and use of firefighting equipment; use of personnel fire protective clothing and equipment (e.g., turnout gear, self-contained breathing apparatus); communications; incident command and control; teamwork; and fire fighting strategies. The inspectors also attended the post-drill critique to assess the licensee's ability to review fire brigade performance and identify areas for improvement. Following the critique, the inspectors compared their findings with the licensee's observations and to the requirements specified in the licensee's Fire Protection report. This activity constituted one inspection sample.

b. Findings

No findings were identified.

1R06 Flood Protection Measures

.1 Internal Flooding

a. Inspection Scope

The inspectors reviewed one internal flood protection measures sample for the Unit 1 and 2 Turbine Driven Auxiliary Feedwater (TDAFW) Pump Rooms in the Auxiliary Building. The inspectors reviewed the internal flood design for the TDAFW Pump Rooms in the Auxiliary Building to verify that flood mitigation plans were consistent with the design requirements and risk analysis assumptions and that equipment essential for reactor shutdown was properly protected from a flood caused by pipe breaks in these rooms. Specifically, the inspectors reviewed the licensee's moderate energy line break flooding study to fully understand the licensee's flood mitigation strategy, reviewed licensee drawings and then verified that the assumptions and results remained valid. The inspectors walked down the TDAFW Pump Rooms in the Auxiliary Building to verify the assumed flooding sources, adequacy of common area drainage, and flood detection instrumentation to ensure that a flooding event would not impact reactor shutdown capabilities. This activity constituted one inspection sample.

b. Findings

No findings were identified.

1R11 Licensed Operator Requalification Program

.1 Quarterly Review

a. Inspection Scope

The inspectors performed a licensed operator requalification program review. The inspectors observed a simulator session on October 11. The training scenario involved a Power Operated Relief Valve (PORV 68-340) and associated Block valve (68-332) failing open. 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.

.2 Annual Review of Licensee Requalification Examination Results

a. Inspection Scope

Annual Review of Licensee Requalification Examination Results. On December 15, 2011, the licensee completed the annual requalification operating test required to be administered to all licensed operators in accordance with 10 CFR 55.59(a)(2). The inspectors performed an in-office review of the overall pass/fail results of individual operating tests and the crew simulator operating tests. These results were compared to the thresholds established in Manual Chapter 609 Appendix I, "Operator Requalification Human Performance Significance Determination Process."

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 inspectors completed one sample.

- Maintenance Rule 9th Periodic Assessment Report

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 six samples.

- PRA Yellow risk condition not communicated to the station on October 12, 2011
- 2B Emergency Diesel Generator emergent maintenance
- Unit 2 Yellow PSA Risk – Calibration of Time Delay Relays for Degraded Voltage on the 2A 6.9KV Shutdown boards
- Unit 1 Emergency Gas Treatment System maintenance along with 1B-B 6.9KV Degraded Voltage Relay Testing
- Unit 2 Eagle 21 system card failure
- Unit 1 Turbine Driven Auxiliary Feedwater Pump Inoperable

b. Findings

No findings were identified.

1R15 Operability Evaluations

a. Inspection Scope

For the three 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 three samples.

- PER 406695, commercial grade dedication concerns, including 2B EDG ventilation low flow switch
- PER 433761, Issues identified in the Manhole/Handhole 4 week PM 052053000
- SR 417889, inadvertent containment spray actuation analysis

b. Findings

No findings were identified.

1R22 Surveillance Testing

a. Inspection Scope

For the two 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 two samples.

In-Service Tests:

- 2-SI-SXP-074-201.A, Residual Heat Removal Pump 2A-A Performance Test, Revision 16

Routine Surveillance Tests:

- 2-SI-EDC-082-307.B, Undervoltage/degraded Voltage, DG Start and Load Shedding Time Response Relay Test, Revision 19

b. Findings

Introduction. A Green self-revealing non-cited violation of Unit 2 TS 6.8.1.a was identified for the licensee's failure to follow station procedures during the performance of a surveillance testing activity. While performing degraded voltage/load shed relay testing associated with the 2B 6.9kV shutdown board, the use of improper test equipment and the incorrect connection of test equipment resulted in a control power circuit fuse being blown, which caused inoperability of an emergency diesel generator and a motor driven auxiliary feedwater train.

Description. On August 11, 2011, during the performance of surveillance testing pursuant to the requirements of Unit 2 TS SR 4.3.3.11.1 for loss of power diesel generator start instrumentation, the licensee was implementing procedure 2-SI-EDC-082-307.B, Undervoltage/Degraded Voltage, DG Start and Load Shedding Time Response Relay Test, revision 19. The procedure specified the connection of a single pole single throw (SPST) test switch to certain locations in the 125VDC control power circuit associated with the 2B 6.9kV shutdown board degraded voltage/load shedding relays. Maintenance personnel instead used a double pole single throw (DPST) test switch, and mistakenly connected one of the leads to an incorrect terminal location in the circuit. The procedure step which specified the installation configuration required a concurrent verification to be performed in order to verify proper test switch installation. This process failed to identify the incorrect installation. When the switch was closed, a fuse blew due to the short circuit created in the control power circuit associated with the relays being tested. This caused extended, unplanned inoperability of the associated 2B emergency diesel generator, as well as the 2B motor driven train of auxiliary feedwater.

Repairs to the circuit were completed, and the required surveillance testing was satisfactorily completed within 24 hours.

Analysis. The licensee's failure to properly perform the degraded voltage, EDG start, and load shedding relay surveillance testing activity in accordance with procedure 2-SI-EDC-082-307.B, revision 19, was a performance deficiency. The finding was determined to be greater than minor because it was associated with the human 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, the failure to follow procedure steps resulted in inoperability of the 2B emergency diesel generator and the 2B motor driven train of auxiliary feedwater. 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 cause of this finding was determined to have a cross-cutting aspect in the area of Human Performance associated with the Work Practices component. The licensee failed to adequately implement human error prevention techniques such as self and peer checking (e.g. concurrent verification) during the SPST switch connection. Additionally, maintenance personnel failed to question the use of a DPST switch in lieu of a SPST switch as specified in the procedure (i.e. proceeding in the face of uncertainty). [H.4(a)].

Enforcement. Unit 2 TS 6.8.1.a required, in part, that written procedures be established, implemented, and maintained covering the activities specified in Appendix A, "Typical Procedures for Pressurized Water Reactors and Boiling Water Reactors," of Regulatory Guide (RG) 1.33, "Quality Assurance Program Requirements (Operations)," Revision 2, dated February 1978. RG 1.33 Appendix A Section 8.b.(1)(q) required implementing procedures for each surveillance test listed in the TS, including tests associated with emergency power. Unit 2 TS SR 4.3.3.11.1 required channel calibration testing of loss of power diesel generator start instrumentation functions. Contrary to the above, on August 11, 2011, the licensee failed to adequately implement written procedures for surveillance testing required by TS. Specifically, procedure 2-SI-EDC-082-307.B, Undervoltage/Degraded Voltage, DG Start and Load Shedding Time Response Relay Test, revision 19, was not followed as written during TS-required surveillance testing. This resulted in inoperability of the 2B emergency diesel generator and the 2B motor driven train of auxiliary feedwater. Because the finding was of very low safety significance and has been entered into the licensee's CAP as PER 415324, this violation is being treated as an NCV, consistent with the NRC Enforcement Policy: NCV 05000328/2011005-01, "Failure to Follow Procedure for Loss of Power Diesel Generator Start Instrumentation Surveillance Testing."

1EP2 Alert and Notification System Testing

a. Inspection Scope

The inspector evaluated the adequacy of the licensee's methods for testing the Alert and Notification System (ANS) in accordance with Nuclear Regulatory Commission (NRC) Inspection Procedure 71114, Attachment 02, "Alert and Notification System Evaluation" The applicable planning standard, 10 CFR Part 50.47(b)(5), and its related requirements, 10 CFR Part 50, Appendix E, Section IV.D, were used as reference criteria. The criteria contained in NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," Revision 1, was also used as a reference.

The inspector reviewed various documents that are listed in the Attachment to this report. This inspection activity satisfied one inspection sample for the ANS on a biennial basis.

b. Findings

No findings were identified.

1EP3 Emergency Preparedness Organization Staffing and Augmentation System

a. Inspection Scope

The inspector reviewed the licensee's Emergency Response Organization (ERO) augmentation staffing requirements and process for notifying the ERO to ensure the readiness of key staff for responding to an event and timely facility activation. The qualification records of key position ERO personnel were reviewed to ensure all ERO qualifications were current. A sample of problems identified from augmentation drills and system tests performed since the last inspection were reviewed to assess the effectiveness of corrective actions.

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

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

b. Findings

No findings were identified.

1EP4 Emergency Action Level and Emergency Plan Changes

a. Inspection Scope

Since the last NRC inspection of this program area, revisions 93, 94, and 95 of the Emergency Plan were implemented. The licensee determined that in accordance with 10 CFR 50.54(q), the changes resulted in no decrease in the effectiveness of the Plan, and that the revised Plan continued to meet the requirements of 10 CFR 50.47(b) and Appendix E to 10 CFR Part 50. The inspector conducted a review of the Emergency Plan and Emergency Action Level changes and a sampling of the implementing procedure changes made between November 1, 2010, and September 30, 2011, to evaluate for potential decreases in effectiveness of the Plan. However, this review was not documented in a Safety Evaluation Report and does not constitute formal NRC approval of the changes. Therefore, these changes remain subject to future NRC inspection in their entirety.

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

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

b. Findings

No findings were identified.

1EP5 Correction of Emergency Preparedness Weaknesses and Deficiencies

a. Inspection Scope

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

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

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

b. Findings

No findings were identified.

1EP6 Drill Evaluation

a. Inspection Scope

Resident inspectors evaluated the conduct of routine licensee emergency drill on October 4, 2011, to identify any weaknesses and deficiencies in classification, notification, and protective action recommendation (PAR) development activities. The inspectors observed emergency response operations in the simulated control room to verify that event classification and notifications were done in accordance with EPIP-1, Emergency Plan Classification Matrix, Revision 46. The inspectors also attended the licensee critique of the drill to compare any inspector observed weakness with those identified by the licensee in order to verify whether the licensee was properly identifying deficiencies. This activity constituted one inspection sample.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification

a. Inspection Scope

Cornerstone: Mitigating Systems

The inspectors sampled licensee submittals for the six PIs listed below for the period from October 1, 2010 through September 30, 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.

- Mitigating Systems PI: Emergency AC Power (both units)
- Mitigating Systems PI: High Pressure Injection System (both units)
- Mitigating Systems PI: Heat Removal System (AFW) (both units)
- Mitigating Systems PI: Residual Heat Removal System (both units)
- Mitigating Systems PI: Cooling Water System (both units)

- Safety System Functional Failures (both units)

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

Cornerstone: Emergency Preparedness

The inspector sampled licensee submittals relative to the Performance Indicators (PIs) listed below for the period October 1, 2010, through June 30, 2011. To verify the accuracy of the PI data reported during that period, PI definitions and guidance contained in NEI 99-02, “Regulatory Assessment Performance Indicator Guideline,” Revision 6, was used to confirm the reporting basis for each data element.

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

The inspection was conducted in accordance with NRC IP 71151, “Performance Indicator Verification.” For the specified review period, the inspector examined data reported to the NRC, procedural guidance for reporting PI information, and records used by the licensee to identify potential PI occurrences. The inspector verified the accuracy of the PI for ERO drill and exercise performance through review of a sample of drill and event records. The inspector reviewed selected training records to verify the accuracy of the PI for ERO drill participation for personnel assigned to key positions in the ERO. The inspector verified the accuracy of the PI for alert and notification system reliability through review of a sample of the licensee’s records of periodic system tests. The inspector also interviewed the licensee personnel who were responsible for collecting and evaluating the PI data. Licensee procedures, records, and other documents reviewed within this inspection area are listed in the Attachment to this report.

This inspection activity satisfied one inspection sample each for the Drill/Exercise Performance, ERO Drill Participation, and Alert and Notification System as defined in IP 71151-05.

b. Findings

No findings were identified.

4OA2 Identification and Resolution of Problems

.1 Routine 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 Operator Workarounds

a. Inspection Scope

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

b. Findings and Observations

No findings were identified.

.3 Semi-Annual Trend Review

a. Inspection Scope

As required by Inspection Procedure 71152, the inspectors performed a review of the licensee's CAP and 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 July 2011 through December 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 July through December 2011. 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 of issues involving functionality of doors in the plant that serve as fire barriers. For the time period of July through December 2011, the inspectors noted 25 PERs entered into the licensee's CAP documenting conditions relating to fire doors that challenged their ability to perform their required fire barrier functions. Two of these issues were identified by the inspectors. The licensee initiated PER 447967 during this period to identify and address this trend of issues

.4 Selected Issue Follow-up: Adherence To Licensed Thermal Power Limit

a. Inspection Scope

On September 20, 2011, Unit 1 reactor thermal power exceeded the licensed power limit as a result of control rod withdrawal during the performance of a planned activity to reset the fully withdrawn control rod bank position.

The inspectors reviewed the licensee's actions in response to this issue, including actions to determine and correct the cause of the failure to properly maintain reactor thermal power below the licensed limit. The inspectors reviewed PER 437068 dealing with this event, interviewed engineering and operations personnel, and reviewed the licensee's corrective actions.

b. Findings and Observations

Introduction. The inspectors identified a Green non-cited violation of Unit 1 Operating License DPR-77 Condition 2.(C).1 "Maximum Power Level" for the licensee's failure to take prudent action to ensure that the licensed power limit was not exceeded during a pre-planned evolution which involved manual reactivity manipulations. Prompt action was not taken by operators to reduce power when reactor thermal power exceeded the licensed power limit during a control rod full out position reset activity. Additionally, prudent action to sufficiently reduce power prior to the activity to accommodate the power transient was not taken.

Description. On September 20, 2011, the licensee performed procedure 0-PI-NXX-085-001.0, “Resetting Control Rod Fully Withdrawn Position,” revision 23, which involved withdrawing control rod banks to establish a new “full out” position. Prior to commencing the activity, the operations staff consulted with reactor engineering personnel regarding the expected effect on reactivity for the planned control rod withdrawal from 228 steps to 230 steps. Engineering personnel assessed that there would be a negligible effect on reactivity due to the planned rod withdrawal, despite no reference document containing rod worth information above 228 steps being available. Operators questioned this conclusion, but decided to proceed with the activity with the assumption that the magnitude of the resulting transient would be able to be accommodated by the current/existing 2-3 megawatts thermal (MWt) margin to the licensed power limit of 3455 MWt. Data from previous performances of this activity which showed that a transient power increase of several MWt would be expected was available but not referenced. At 11:25am, control rod banks were withdrawn, which induced a power transient that resulted in reactor thermal power exceeding the licensed limit. At 12:32pm, operators inserted a control rod bank in response to 1-hour average of reactor thermal power exceeding the limit. An operations log entry was made to indicate that control rods were inserted in order to reduce power below the limit.

On September 21, 2011, the inspectors noted this log entry and researched further details surrounding the evolution. The inspectors found that instantaneous reactor thermal power had peaked at 3463 MWt (100.2 percent RTP) approximately 12 minutes after rod withdrawal occurred. The inspectors also noted that the 10-minute average of reactor thermal power exceeded the limit of 3455 MWt for a duration of 16 minutes beginning at 11:36am (peak value of 3457.44 MWt, or 100.07 percent RTP), and for another duration of 35 minutes beginning at 12:02pm. The inspectors also noted that the 1-hour average of reactor thermal power exceeded the licensed limit for a period of 18 minutes beginning at 12:21pm. Operator action to reduce power was taken at 12:32pm. The inspectors expressed their concern to the licensee regarding adherence to licensed power limits during pre-planned activities, and the licensee entered this issue into their CAP as PER 437068 on September 22, 2011.

The inspectors reviewed Regulatory Issue Summary 2007-21, Rev. 1, “Adherence To Licensed Power Limits,” which endorsed an NEI Position Statement “Guidance To Licensees on Complying with the Licensed Power Limit.” This included guidance that “no actions are allowed that would intentionally raise core thermal power above the licensed thermal power limit for any period of time,” and “licensees are expected to take prompt action to reduce thermal power whenever it is found above the licensed limit.” This also included guidance that for pre-planned evolutions which could cause a transient increase in reactor power, “prudent action based on prior performance or evaluations should be taken to reduce power prior to performing the evolution.” The inspectors also reviewed the following licensee procedures.

OPDP-1, “Conduct of Operations,” revision 19, step 3.6.A.1 stated that “no actions are allowed that would intentionally raise core thermal power above the licensed thermal power limit for any period of time." Although in this case actions were not taken with the intent of raising core thermal power, it was allowed to remain above the limit for some period of time as a result of the transient caused by planned actions.

OPDP-1 Step 3.6.A.3 stated "if the core thermal power average for a one-hour period is found to exceed the licensed thermal power limit, take prompt (typically no more than 10 minutes from point of discovery) action to ensure that thermal power is less than or equal to licensed thermal power limit." Procedure 0-GO-5, "Normal Power Operation," revision 73, Step 3.2.L stated, "Do not allow average thermal power to exceed 3455 MWt for two consecutive hours. Every effort should be made to maintain core thermal power one (1) hour average less than 3455 MWt." Procedure 1-PI-OPS-000-020.1, "Operator At The Controls Duty Station Checklists Modes 1-4," revision 54, stated "Every effort should be made to maintain core thermal power 1 hour average less than 3455 MWt," and "If core thermal power 1 hour average exceeds 3455 MWt or a rising power trend which will exceed 3455 MWt is observed, than ensure prompt action is taken to lower reactor power as necessary," and "Do not allow average thermal power to exceed 3455 MWt as read on U2125 [1-hour average]."

The inspectors noted that the above requirements are consistent with NEI Position Statement guidance that maintaining the 2-hour average of reactor thermal power below the licensed limit is sufficient to demonstrate compliance with the applicable license condition while accommodating normal short duration fluctuations during steady state power operation. However, if action is delayed until the 1-hour average of power has reached or approaches the limit in the case of a power excursion above the limit due to a non-steady state power transient, the inspectors observed that the above procedural requirements may be insufficient to ensure that "prompt action to reduce thermal power whenever it is found above the licensed limit" is taken in a timely manner. In the case of the power transient on September 20, 2011, the 10-minute average of power exceeded the licensed limit for 46 of the 60 minutes prior to operator action being taken based on the 1-hour average.

OPDP-1 Step 3.6.A.6 stated that for pre-planned evolutions which could cause a transient increase in reactor power, "prudent action based on prior performance or evaluations should be taken to reduce power prior to performing the evolution." The inspectors reviewed data from prior performances of this evolution and concluded that a transient increase in reactor power of several MWt should have been expected. 1-PI-OPS-000-020.1 stated, "If any preplanned activity will be performed which is expected to cause a transient rise in reactor thermal power, then reduce turbine loading and/or insert negative reactivity (control rod use / boration) as necessary prior to starting the activity to ensure reactor thermal power will not exceed 3455 MWt." The inspectors determined that this requirement was not met. 0-GO-5 contained a similar requirement: "If any preplanned activity will be performed which is expected to cause a transient rise in thermal power, then reduce turbine load and/or insert negative reactivity (using control rods or boration) prior to starting activity as necessary to ensure one (1) hour average power will not exceed 3455 MWt." The inspectors noted that the inconsistency between this procedure's step and the former one is the reference to maintaining the 1-hour average of power below the limit, which, as discussed above, has the potential to allow power to exceed the licensed limit as long as the 1-hour average of power does not. 0-GO-5 Step 5.2 [2] and [3] stated, “Trend computer point U2118 (instantaneous reactor thermal power) on a trend recorder in the unit horseshoe and monitor for rising power trends above 3455 MWt. If rising power trend is observed, then ensure prompt action is taken to reduce reactor power as necessary.” The inspectors determined that this requirement, while not explicit regarding what parameter is required to be maintained or what would make action “necessary” in the case of a rising trend, was not adequately followed.

0-GO-5 Step 5.2 Caution (2) stated, “Power shall NOT exceed one hour average of 3455.00 MWt.” The inspectors determined that this requirement was not met, since 1-hour average power (1U2125) indicated in an alarm condition (above the alarm setpoint of 3455 MWt) from 12:21pm until 12:39pm.

Analysis. The licensee’s failure to take appropriate action to maintain reactor thermal power within the licensed limit during a planned control rod withdrawal activity was a performance deficiency. The finding was determined to be greater than minor because it was sufficiently similar to example 8.a. of IMC 0612 Appendix E, in that: 1) prudent action based on prior performance was not taken to reduce power prior to performing the evolution, and 2) operators did not promptly lower thermal power once the licensed limit was exceeded. Additionally, if left uncorrected the finding would have the potential to lead to a more significant safety concern, since operation above the licensed power limit has the potential to reduce analyzed margins to fuel cladding damage. Using 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) under the barrier integrity cornerstone since only the fuel cladding barrier criterion was applicable.

The cause of this finding was determined to have a cross-cutting aspect in the area of Human Performance associated with the Decision-Making component. Elements of non-conservative decision making which contributed to this performance deficiency included:

- Prior to February 2011, station procedures required operators to monitor and maintain the 10-minute average of thermal power below the licensed limit. These requirements were revised and replaced with requirements to maintain the 1-hour average of thermal power below the licensed limit. This was a non-conservative decision made without due consideration of potential consequences or the need for supplemental guidance to maintain an appropriate and conservative approach to controlling thermal power under non-steady state conditions.
- The decision was made to proceed with the rod withdrawal activity under the non-conservative assumption that there would be negligible reactivity effects, and without considering available data from previous performances. This did not reflect a philosophy of demonstrating that a proposed activity is safe in order to proceed rather than demonstrating that it is unsafe in order to disapprove the action. [H.1(b)]

Enforcement. Facility operating license DPR-77 condition 2.(C).1 stated that TVA is authorized to operate the [Unit 1] facility at reactor core power levels not in excess of 3455 MWt. Contrary to the above, on September 20, 2011, the licensee operated Unit 1 at reactor core power levels in excess of 3455 MWt. Specifically, reactor thermal power reached 3457.4 MWt on a 10-minute average as a result of a planned control rod withdrawal activity, and remained above the licensed limit for durations of 16 minutes and 35 minutes. Additionally, 1-hour average thermal power exceeded the licensed limit for a period of 18 minutes. Because this violation was determined to be of very low safety significance and has been entered into the licensee's corrective action program as PER 437068, it is being treated as an NCV consistent with the NRC Enforcement Policy: NCV 05000327/2011005-02, "Failure to Maintain Thermal Power Less Than Licensed Limit."

.5 Selected Issue Follow-up: LCO 3.5.3 One Hour Allowance Time for RHR Check Valve Testing

a. Inspection Scope

In July 2011, following Unit 2 startup from a refueling outage, an issue regarding the performance of RHR check valve surveillance testing during the outage was identified. This testing was performed in Mode 4. LCO 3.5.3, which required one train of ECCS to be operable in Mode 4, contained a provision which allowed the RHR subsystem of the operable ECCS train to be inoperable for up to 1 hour for surveillance testing of valves. The valve surveillance testing activities took longer than 1 hour to complete, and as a result, the 1 hour allowance of LCO 3.5.3 was invoked multiple times over a 24-hour period in order to accomplish the testing.

The inspectors reviewed the licensee's actions in response to this issue, including actions to determine and correct the cause of this issue regarding TS compliance. The inspectors reviewed PER 402304 dealing with this event, interviewed engineering and operations personnel, and reviewed the licensee's corrective actions.

b. Findings and Observations

No findings were identified. The inspectors determined that the licensee's apparent cause evaluation associated with PER 402304 was thorough and that the corrective actions appeared to be adequate to address the issue. The licensee's investigation determined that corrective actions were needed in order to prevent multiple consecutive applications of a 1-hour TS allowance in order to complete valve surveillance activities in the future. The licensee's corrective actions included revising applicable testing procedures to allow the testing to be performed in Mode 3.

4OA3 Event Followup

.1 (Closed) Licensee Event Report (LER) 05000328/2011-001-00 and -01, Nuclear Instrumentation System Power Range Neutron Flux Trip Low Range Bistable Incorrectly Calibrated

On June 23, 2011, during Unit 2 startup from a refueling outage, nuclear instrumentation system (NIS) power range channel N44 was discovered to be in an inoperable condition when, during power escalation, only three of the four NIS channel "power range hi flux lo setpoint" bistable status indicating lights in the main control room came on at the expected setpoint of 25 percent RTP. Upon discovery of the condition, the licensee took action to place the affected channel in the tripped condition. The affected channel had been left in an inoperable condition as a result of an error made during the performance of an outage maintenance activity approximately one month earlier. The licensee documented the issue in PER 397142, which included a root 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 root 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 397142 to verify that the cause of the channel N44 inoperability was identified and whether corrective actions were appropriate. The licensee's root cause evaluation identified that maintenance personnel failed to correctly perform the channel calibration procedure during the outage, which resulted in the improper setpoint adjustment of the incorrect component. The inspectors concluded that the licensee's corrective actions to this event were appropriate, including revision of applicable maintenance procedures to add verification steps and incorporate operational barriers to help prevent incorrect components from being adjusted. This LER is closed.

b. Findings

Introduction. A Green self-revealing non-cited violation of Unit 2 TS 3.0.3 was identified for the licensee's failure to place the unit in Mode 3 within seven hours when a Limiting Condition for Operation (LCO) was not met in Modes 1 and 2. The requirements of LCO 3.3.1, "Reactor Trip System Instrumentation," associated with the power range neutron flux function in Modes 1 and 2 were not met for a period of approximately 24 hours. This was the result of an error made during the performance of a channel calibration activity, which caused one channel to be left in an inoperable condition.

Description. On May 26, 2011, Unit 2 was in a refueling outage in Mode 6, and procedure 2-SI-ICC-092-N44.4, Channel Calibration of Power Range Nuclear Instrumentation System (NIS) Channel N44, revision 24, was being performed. While performing step 6.5.9 to adjust the overpower trip high range bistable setpoint to its required value of 60 percent RTP to support low power physics testing scheduled to occur during startup from the outage, maintenance personnel mistakenly adjusted the low range bistable setpoint to 60 percent RTP. Step 6.3.5 of the procedure included a functional test of the low range bistable setpoint to verify its required setpoint of 25 percent RTP. This step had already been completed satisfactorily at the time the low range bistable was mistakenly adjusted to 60 percent RTP while performing step 6.5.9.

Unit 2 TS Limiting Safety System Setting (LSSS) 2.2.1, "Reactor Trip Instrumentation Setpoints," required the low range trip setpoint to be set at a nominal value of 25 percent RTP (maximum allowable value of 27.4 percent RTP). Unit 2 TS SR 4.10.3.2 required that each power range channel shall be subjected to a channel functional test prior to initiating physics tests in order to ensure that the required trip functions are available to operate at the required setpoint. The above channel calibration procedure implemented this requirement. Unit 2 TS LCO 3.3.1, "Reactor Trip System Instrumentation," required that a minimum of 3 (out of 4) channels of the power range neutron flux trip function be operable in Modes 1 and 2, and that a single inoperable channel shall be placed in the tripped condition within 6 hours. With all four NIS channels operable, the neutron flux trip function operates in a 2-out-of-4 logic configuration. With one channel inoperable and placed in the tripped condition as required, the trip function operates in a 1-out-of-3 logic configuration.

On June 21, 2011, while in Mode 3, power range NIS channel N43 was removed from service and placed in the tripped condition in preparation for low power physics testing. As a result, when Mode 2 was entered on June 22, 2011, the requirements of LCO 3.3.1 were not met due to both channels N43 and N44 being in inoperable conditions. In this condition, channels N41 and N42 remained available to perform the required trip function in a 1-out-of-2 logic configuration. This condition existed from 0411 on June 22, 2011, until 1453 on the same day, when channel N43 was restored to an operable condition. Because channel N44 remained in an inoperable condition, LCO 3.3.1 and associated action requirements continued to not be met since the inoperable channel had not been placed in the tripped condition. In this condition, which existed from 1453 on June 22, 2011, until 0355 on June 23, 2011, during which time Unit 2 had entered Mode 1, the remaining three channels were available to perform the required trip function in a 2-out-of-3 logic configuration.

The inoperable condition of channel N44 was discovered at 0222 on June 23, 2011, when, during power escalation, only three of the four NIS channel "power range hi flux lo setpoint" bistable status indicating lights in the main control room came on at the expected setpoint of 25 percent RTP. Upon discovery of the condition, the licensee took action to place the affected channel in the tripped condition. This action was accomplished at 0355 on June 23, 2011.

The Mode change into Mode 2 on June 22, 2011, constituted a violation of LCO 3.0.4, which prohibits, with some stated exceptions, entry into an applicable Mode of an LCO which is not met. The failure to meet the requirements of LCO 3.3.1 during the period of inoperability of channel N44 as described above constituted a violation of LCO 3.0.3, as described below.

Analysis. The licensee's failure to properly perform the N44 power range nuclear instrumentation channel calibration procedure 2-SI-ICC-092-N44.4, revision 24, was a performance deficiency. The finding was determined to be greater than minor because it was associated with the human 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, the inoperability of the N44 power range NIS channel low range neutron flux trip function resulted in the failure to meet TS operability requirements associated with reactor trip system instrumentation. Using Inspection IMC 0609, "Significance Determination Process," Attachment 4, "Phase 1 - Initial Screening and Characterization of Findings," the finding was determined to be applicable to Phase 2 SDP screening since it represented the loss of a channel of a TS required function for greater than its TS allowed outage time. The inspectors determined that neither the pre-solved Phase 2 spreadsheets nor the plant-specific Phase 2 notebook adequately modeled the condition of concern. The inspectors, in consultation with the Senior Reactor Analysts, performed a Phase 2 analysis using Saphire 8 software with the Sequoyah SPAR model in the SDP mode. The inspectors adjusted the basic event RPS-XHE-SIGNL, "Operator to Respond with Reactor Protection Signal Present" to a failed condition as a surrogate for the NIS channel not being able to perform its function. This was a bounding analysis (because other NIS channels remained operable during this time period) and represented the increase in risk due to the reactor not shutting down when called upon (ATWS). Using an exposure period of 1 day with a truncation value of 1E-13, a result of ΔCDF << 1E-6, or very low safety significance (Green), was obtained.

The cause of this finding was determined to have a cross-cutting aspect in the area of Human Performance associated with the Work Practices component. The licensee failed to adequately implement human error prevention techniques, such as self and peer checking, to ensure that the work activity was being performed on the correct component. [H.4(a)].

Enforcement. Unit 2 TS LCO 3.0.3 required that when an LCO is not met, except as provided in the associated action requirements, within one hour action shall be taken to place the unit in a Mode in which the specification does not apply by placing it, as applicable, in at least Hot Standby (Mode 3) within the next 6 hours. In Modes 1 and 2, Unit 2 TS LCO 3.3.1, Reactor Trip System Instrumentation, required that a minimum of 3 (out of 4) channels associated with the power range neutron flux trip function shall be operable, and that a single inoperable channel shall be placed in the tripped condition within 6 hours. Contrary to the above, on June 22, 2011, while operating in Mode 2 with LCO 3.3.1 not met, the licensee failed to take action within one hour to place the unit in a Mode in which LCO 3.3.1 does not apply by placing it in at least Mode 3 within the next 6 hours. Specifically, operation continued in Modes 1 and 2 with the requirements of LCO 3.3.1 and associated action requirements not met from 0411 on June 22, 2011, until 0355 on June 23, 2011 due to the inoperability of the low range neutron flux trip function of power range nuclear instrumentation channel N44 during this time. Because the finding was of very low safety significance and has been entered into the licensee's CAP as PER 397142, this violation is being treated as an NCV, consistent with the NRC Enforcement Policy: NCV 05000328/2011005-03, "Nuclear Instrumentation System Channel Calibration Error."

4OA5 Other Activities

.1 Quarterly Resident Inspector Observations of Security Personnel and Activities

a. Inspection Scope

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

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

b. Findings

No findings were identified.

.3 Review of the Operation of an Independent Spent Fuel Storage Installation (ISFSI) (60855.1)

a. Inspection Scope

The inspectors reviewed the seventh dry-cask-loading campaign of the ISFSI to verify that operations were conducted in a safe manner in accordance with approved procedures and without undue risk to the health and safety of the public. The inspectors observed fuel loading operations and other processes on several multi-purpose canisters (MPCs) to verify that the specified fuel assemblies were placed in the correct locations and that other MPC processes were implemented in accordance with approved procedures. The inspectors reviewed problem reports discovered during the campaign to ensure that issues were placed in the corrective action program. The inspectors also reviewed ISFSI document control practices to verify that changes to the required ISFSI procedures and equipment were performed in accordance with guidelines established in local procedures and 10CFR72.48. Documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

4OA6 Meetings

Exit Meeting Summary

On October 14, 2011, the inspectors presented emergency preparedness inspection results to Mr. R. Detwiler and other members of licensee management. The inspector confirmed that proprietary information was not provided or examined during the inspection.

On January 5, 2012, the resident inspectors presented the inspection results to Mr. P. Simmons 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 violation of very low safety significance (Green) was identified by the licensee and is a violation of NRC requirements which meets the criteria of the NRC Enforcement Policy for being dispositioned as a Non-Cited Violation.

Title 10 CFR 50, Part 50.65(a)(4), states, in part, that before performing maintenance activities, licensee personnel shall assess and manage the increase in risk that may result from the proposed maintenance activities. Contrary to the above, on November 16, 2011, before performing maintenance activities, licensee personnel did not adequately assess the increase in risk from the proposed maintenance activities. Specifically, the licensee failed to assess the increase in risk associated with Essential Raw Cooling Water (ERCW) pumps Q-A and R-A being unavailable for maintenance. This problem was entered into the licensee's corrective action program as PER 465734. The finding was screened using Inspection Manual Chapter 0609, Appendix K – Maintenance Risk Assessment and Risk Management Significance Determination Process, and was determined to be of very low safety significance (Green).

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel
T. Adkins, Corporate Emergency Preparedness Specialist
J. Carlin, Site Vice President
S. Connors, Operations Manager
G. Cook, Site Licensing Manager
J. Cross, Chemistry Manager
A. Day, Radiation Protection Manager
R. Detwiler, Director, Safety and Licensing
C. Dieckmann, Manager, Maintenance
Z. Kitts, Licensing Engineer
W. Lee, Corporate Emergency Preparedness Manager
A. Little, Site Security Manager
S. McCamy, Quality Assurance Manager
P. Noe, Site Engineering Director
J. Parshall, Corporate Emergency Preparedness
W. Peggram, Emergency Preparedness Specialist
P. Pratt, Work Control Manager
J. Reidy, Operations Superintendant
P. Simmons, Plant Manager
D. Sutton, Licensing Engineer
N. Thomas, Licensing Engineer
C. Ware, Training Director
K. Wilkes, Operations Support Superintendent

NRC personnel
S. Lingam, Project Manager, Office of Nuclear Reactor Regulation
J. Hanna, Senior Reactor Analyst, Region II

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000328/2011005-01	NCV	Failure to Follow Procedure for Loss of Power Diesel Generator Start Instrumentation Surveillance Testing (Section 1R22)
05000327/2011005-02	NCV	Failure to Maintain Thermal Power Less Than Licensed Limit (Section 4OA2.4)
05000328/2011005-03	NCV	Nuclear Instrumentation System Channel Calibration Error (Section 4OA3.1)

Closed

05000328/2011-001-00	LER	Nuclear Instrumentation System Power Range Neutron Flux Trip Low Range Bistable Incorrectly Calibrated (Section 4OA3.1)
05000328/2011-001-01	LER	Nuclear Instrumentation System Power Range Neutron Flux Trip Low Range Bistable Incorrectly Calibrated (Section 4OA3.1)

LIST OF DOCUMENTS REVIEWED

The following is a list of documents reviewed during the inspection. Inclusion on this list does not imply that the NRC inspectors reviewed the documents in their entirety but rather that selected sections of portions of the documents were evaluated as part of the overall inspection effort. Inclusion of a document on this list does not imply NRC acceptance of the document or any part of it, unless this is stated in the body of the inspection report.

Section R01: Adverse Weather Protection

Procedures
0-PI-OPS-000-006.0, Freeze Protection, Revision 49
NPG-SPP-10.14, Freeze Protection, Revision 00

Work Orders
112794097 - 1B1 MSR doghouse
112788006 - Air Intake damper fail open
112794101 - 1C1 and 2A2 MSR's
112794103 - 2B2 MSR doghouse
112794107 - 1E Turbine Building Basement Exhaust Fan Freeze Protection
112805822 - CVCS Heat Trace Circuit 407 not operating
112805823 - Security cage plastic installation per 0-PI-OPS-000-006.0
112805825 - CST Freeze Protection installation per 0-PI-OPS-000-006.0
111480558 - Replace obsolete 8" Deluge Valves in valve vault "C"

PERs
442789 - Air Intake damper fail open

Section R04: Equipment Alignment

Partial System Walkdowns

Procedures
1-SO-63-5, Emergency Core Cooling System, Revision 57
1-SO-63-5.01, Emergency Core Cooling System Power Checklist, Revision 57
1-SO-63-5.02, Emergency Core Cooling System Valve Checklist, Revision 57
0-SO-72-1, Containment Spray Systems, Revision 38
0-SO-72-1.10, Containment Spray Systems Valve Checklist 2-72-1.05, Effective Date: 08-10-2011
0-SO-72-1.04, Containment Spray Systems Power Checklist 2-72-1.02, Effective Date: 08-10-2011

Complete System Walkdown

Procedures
EA-3-2, Local Control of Turbine Driven AFW LCVs, Revision 3
EA-3-4, Local Alignment of TD AFW LCV Backup Air Supply, Revision 4

EA-3-7, Local Operation of TD AFW Pump, Revision 6
EA-3-9, Establishing Turbine Driven AFW Flow, Revision 6
EA-3-10, Establishing Motor Driven AFW Flow, Revision 2
EA-3-11, Local Isolation Of MD and TD AFW, Revision 1
1-SO-3-2, Auxiliary Feedwater System, Revision 44

PERs
356178
203854
211596

Other documents
Drawing 1, 2-47W803-2, Flow Diagram Auxiliary Feedwater, Revision 65

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

Section R06: Flood Protection Measures

Work Orders
111984178 - Check standing water level in manholes/handholes

PERs
433761 – Issues identified in the Manhole/Handhole 4 week PM 052053000
432510 - Special performance needed for PM's per WO# 111984178

Other documents
TVA letter to NRC dated May 4, 2007. TVA response to GL 2007-01
SQS40056, Moderate Energy Line Break Flooding Study, Revision 14

Section R11: Licensed Operator Requalification

Other documents
S-63, Simulator Exam Guide (PORV and Block Valve Fail Open), Rev 0

Section R12: Maintenance Effectiveness

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

Other documents
RIMS B85101005006, Maintenance Rule Ninth Periodic Assessment Report Units 1, 2, & Common, Revision 0

Section R13: Maintenance Risk Assessments and Emergent Work Evaluation

Procedures
NPG-SPP-07.1, On Line Work Management, Revision 4
0-TI-DSM-000-007.1, Risk Assessment Guidelines, Revision 9
NPG-SPP-07.3, Work Activity Risk Management Process, Revision 4
0-GO-16 - System Operability Checklists, Revision 12
0-PI-ICC-046-057.0, Calibration of Auxiliary Feedwater Pump A-S Speed Control Loop F-46-57/S-46-57, revision 26

Work Orders
112973794, removal of EGM cover and inspect/clean electronics as required

PERs
447407 - PRA Yellow risk condition not communicated to the station

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

PERs
433761 – Issues identified in the Manhole/Handhole 4 week PM 052053000
SR 417889 – discrepancy between TS and SAR analysis for inadvertent CS actuation
406695 – commercial grade dedication process

Other documents
15W810-18, Conduit & Grounding Details – Sheet 17
15N810-5, Conduit & Grounding Details – Sheet 4
15W810-17, Conduit & Grounding Details – Sheet 16

Section R22: Surveillance Testing

Procedures
2-SI-EDC-082-307.B, Undervoltage/degraded Voltage, DG Start and Load Shedding Time Response Relay Test, Revision 19
2-SI-SXP-074-201.A, Residual Heat Removal Pump 2A-A Performance Test, Revision 16

Work Orders
112557819 – 2-SI-EDC-082-307.B, Undervoltage/degraded Voltage, DG Start and Load Shedding Time Response Relay Test, Revision 19

Drawings
1,2-45N765-18, Wiring Diagram 6900V SD Aux PW-Train A&B Schmatic Diagram, revision 17

Section 1EP2: Alert and Notification System Testing

Procedures and Manual
EPFS-9, Inspection, Service, and Maintenance of the Prompt Notification System (PNS) at Browns Ferry, Sequoyah, and Watts Bar Nuclear Plants, Rev. 4
EPDP-14, Evaluation of Changes to Alert and Notification Systems (ANS), Rev 0

Records and Data
Bi-weekly and monthly Siren testing data 2009-2011
Maintenance Trouble log entries for 2010/2011

Section 1EP3: Emergency Response Organization (ERO) Augmentation

Procedures
TRN-30, Radiological Emergency Preparedness Training, Rev. 17
EPDP-10, Facilitation of the Alert and Notification System and Pager Tests, Rev. 2
EPIP-3, Alert, Rev 32
EPIP-4, Site Area Emergency, Rev. 32

Records and Data
Exercise Report – Hostile Action Drill – 8/19/2009
Radiological Emergency Response Plan Drill Report 2/4/2009
Radiological Emergency Response Plan Drill Report 5/12/2009
Radiological Emergency Response Plan Drill Report 9/22/2009
Radiological Emergency Response Plan Drill Report 3/9,16, 23, and 30/2010
Radiological Emergency Response Plan Drill Report 6/10/2010
Radiological Emergency Response Plan Drill Report 8/10/2010
Radiological Emergency Response Plan Drill Report 9/1/2010
Radiological Emergency Response Plan Drill Report 3/15/2011
Radiological Emergency Response Plan Drill Report 7/19/2011
2010/2011 ERO Pager tests
Review of ERO position qualifications for selected individuals
Lesson Plan EPT001.010, CECC Director

Section 1EP4: Emergency Action Level (EAL) and Emergency Plan Changes

Procedures
EPDP-1, Procedures, Maps, and Drawings, Rev. 2
NPG-SPP-09.3, Plant Modifications and Engineering Change Control, Rev. 4
NPG-SPP-09.4, 10 CFR 50.59 Evaluations of Changes, Tests, and Experiments, Rev. 4

Change Packages for Plans and Procedures
Radiological Emergency Plan, Rev. 93, 94, and 95
EPIP-1, Emergency Plan Classification Matrix, Rev. 46

EPIP-2, Notification of Unusual Event, Rev. 30
EPIP-3, Alert, Rev. 32
EPIP-4, Site Area Emergency, Rev. 32
EPIP-5, General Emergency, Rev. 40
EPIP-6, Activation and Operation of the Technical Support Center, Rev. 48
EPIP-10, Medical Emergency Response, Rev. 27
EPIP-13, Dose Assessment, Rev. 16
EPIP-14, Radiation Protection Response, Rev. 22

Section 1EP5: Correction of Emergency Preparedness Weaknesses and Deficiencies

Procedures
NPG-SPP-02.1, NPG Self-Assessment and Benchmarking Program, Rev. 1
NPG-SPP-03.1, Corrective Action Program, Rev. 2

Records and Data
Sequoyah Second Tier Emergency Preparedness PI data 2010/2011
2010 and 2011 Augmentation Drill packages

Audits, Oversight Reports, and Self-Assessments
Quality Assurance Oversight Report – January 1, 2010 through March 31, 2010
Quality Assurance Oversight Report – April 1, 2010 through June 30, 2010
Quality Assurance Oversight Report – July 1, 2010 through September 30, 2010
Quality Assurance Oversight Report – October 1, 2010 through December 31, 2010
Quality Assurance Oversight Report – January 1, 2011 through March 31, 2011
Quality Assurance Oversight Report – April 1, 2011 through June 30, 2011
QA-SQ-11-006, Quality Assurance Assessment of Emergency Preparedness Equipment and Facility Readiness, Drills, and Exercises dated March 25, 2011
QA-SQ-11-008, Quality Assurance Fleet Assessment of Emergency Preparedness -Equipment and Facilities Readiness and Duty Teams dated May 9, 2011
QA-SQ-11-014, Quality Assurance Assessment of Emergency Preparedness and Operations Main Control Room Communications Systems and EP Second Tier Performance Indicators dated September 30, 2011
TVA Quality Assurance Emergency Preparedness Audit Report SSA1003 dated May 20, 2010
2010 Biennial Audit interface adequacy results letters to state and local officials

Problem Evaluation Reports (PER)
345406, 345407, 345409: QA Observations
262497: NSRB Concern
239427: EPIP-13 has human performance error trap
225586: REP Scenario development team lacks members
214670: EDMG-1 procedure change
372442, 366996: EAL1.1.5L error

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
Sequoyah Mitigating System Performance Index Basis Document, Rev. 5
EPDP-11 Emergency Preparedness Performance Indicators, Rev 3

Records and Data
Unit 1 and Unit 2 MSPI Derivation Reports, Emergency AC Power, September 2011
Unit 1 and Unit 2 MSPI Derivation Reports, High Pressure Injection, September 2011
Unit 1 and Unit 2 MSPI Derivation Reports, Heat Removal Systems, September 2011
Unit 1 and Unit 2 MSPI Derivation Reports, Residual Heat Removal, September 2011
Unit 1 and Unit 2 MSPI Derivation Reports, Cooling Water Systems, September 2011

Siren System Availability Test Records, 2010-2011
Siren test data from 4th Quarter 2010 through 2nd Quarter 2011
ERO Personnel Participation, September 1, 2010, through June 30, 2011
DEP opportunities documentation from 4th Quarter 2010, through 2nd Quarter 2011
Drill and exercise participation records of ERO personnel from 3rd Quarter 2010, through 2nd Quarter 2011
Various ERO Personnel Qualification and Participation records
Service Request 297335

Corrective Actions Resulting From Inspection
SR445841, Revise EPFS-9
SR 446678, Procedure revision reviews
SR 446690, Format/content of notification test summary
SR 446699, Late drill report
SR 446719, DEP opportunity scheduling

Section 4OA2: Identification and Resolution of Problems

Procedures
NPG-SPP-03.1, Corrective Action Program, Revision 1
0-GO-14-1, Auxiliary Building 1 AUO Operator Rounds, Revision 24
0-GO-14-2, Auxiliary Building 2 AUO Operator Rounds, Revision 28
NPG-SPP-07.1, On Line Work Management, Revision 4
0-PI-NXX-085-001.0, “Resetting Control Rod Fully Withdrawn Position,” revision 23
OPDP-1, “Conduct of Operations,” revision 19
0-GO-5, “Normal Power Operation,” revision 73
1-PI-OPS-000-020.1, “Operator At The Controls Duty Station Checklists Modes 1-4,” revision 54

PERs
402304 – TS 3.5.3 allowance for RHR check valve testing
437068 – Unit 1 core thermal power limit

Other documents
Operations Directive Manual Appendix O, Operations Guide to Daily Work Control, Schedule Review, and Intolerance Item Standards, Revision 5
Regulatory Issue Summary 2007-21, Rev. 1, "Adherence To Licensed Power Limits"

Section 4OA3: Event Followup

Procedures
2-SI-IFT-092-N44.4, Functional Test of Power Range Nuclear Instrumentation System Channel 44, revision 28
0-RT-NUC-000-003.0, Low Power Physics Testing, revision 23
NPG-SPP-06.9.2, Surveillance Testing Program, revision 0
NPG-SPP-06.9.1, Conduct of Testing, revision 1
2-SI-ICC-092-N44.4, Channel Calibration of Power Range Nuclear Instrumentation System Channel 44, revision 24
0-GO-4, Power Ascension From Less Than 5% Reactor Power to 30% Reactor Power, revision 75

Work Orders
111857542 – N44 channel calibration

Section 4OA5: Other Activities
0-GO-17, Spent Fuel/Dry Cask Operations, Revision 5
NFTP-100, Fuel Selection for Dry MPC Storage, Revision 6 completed for campaign #7
SQN-DCS-300.11, Supplemental Cooling System Operation, Revision 6
CTP-DCS-100.0, Dry Cask Storage Campaign Guidelines, Revision 12
SQN-DCS-200.0, Dry Cask Campaign Review Program, Revision 1
SQN-DCS-200.2, SQN-MPC-Loading and Transport Operations, Revision 25
Fuel Assembly Transfer Forms for SQN-1-96 through SQN-1-101

LIST OF ACRONYMS

ANS	Alert and Notification System
CAP	Corrective Action Program
DEP	Drill/Exercise Performance
EAL	Emergency Action Level
ERCW	Essential Raw Cooling Water
ERO	Emergency Response Organization Readiness
LSSS	Limiting Safety System Setting
NI	Nuclear Instrument
NIS	Nuclear Instrument System
NRC	Nuclear Regulatory Commission
PER	Problem Evaluation Report
PI	Performance Indicator
RWST	Refueling Water Storage Tank
SDP	Significance Determination Process

SSC	Structures Systems and Components
TDAFW	Turbine Driven Auxiliary Feed Water
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report