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

May 12, 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 05000327/2011002, 05000328/2011002

Dear Mr. Krich:

On March 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 April 7, 2011 with Mr. M. Skaggs 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 and three self-revealing findings of very low safety significance (Green), three of which involved violations of NRC requirements. However, because of the very low safety significance and because they are entered into your corrective action program, the NRC is treating these findings 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 (the Public Electronic Reading Room).

Sincerely,

/RA/

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

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

Enclosure: Inspection Report 05000327/2011002, 05000328/2011002
w/Attachment: Supplemental Information

cc w/encl: (See page 3)

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Letter to R. M. Krich from Eugene Guthrie dated May 12, 2011

SUBJECT: SEQUOYAH NUCLEAR PLANT - NRC INTEGRATED INSPECTION REPORT 05000327/2011002, 05000328/2011002

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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.:	05000327/2011002, 05000328/2011002
Licensee:	Tennessee Valley Authority (TVA)
Facility:	Sequoyah Nuclear Plant, Units 1 and 2
Location:	Sequoyah Access Road Soddy-Daisy, TN 37379
Dates:	January 1, 2011 – March 31, 2011
Inspectors:	C. Young, Senior Resident Inspector M. Speck, Resident Inspector W. Deschaine, Resident Inspector R. Rodriguez, Senior Reactor Inspector J. Montgomery, Reactor Inspector P. Higgins, Senior Reactor Inspector
Approved by:	Eugene F. Guthrie, Chief Reactor Projects Branch 6 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000327/2011002, 05000328/2011002; 1/1/2011 – 3/31/2011; Sequoyah Nuclear Plant, Units 1 and 2; Operability Evaluations, Identification and Resolution of Problems, and Event Follow-up

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

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Initiating Events

- Green. A self-revealing finding was identified for the licensee’s failure to perform adequate post-maintenance testing, as specified by procedures SPP-8.3, Post-Modification Testing, revision 10, and NPG-SPP-06.3, Pre-/Post-Maintenance Testing, revision 0, in conjunction with a work order which implemented a plant modification on Unit 1 and included the relocation of the steam dump load reject controller. This resulted in a manual trip of Unit 1 following a turbine trip from 26 percent rated thermal power due to the steam dump load reject controller power supply not being properly connected. The licensee entered this issue into their corrective action program as PERs 285349. The licensee implemented corrective actions to include a revision to post-modification testing procedures to require an additional post maintenance testing (PMT) review for large/complex modifications, as well as revision to applicable maintenance procedures to require verification for plug-in type connections.

  The finding was determined to be greater than minor because it was associated with the equipment performance attribute of the initiating events cornerstone and affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, this finding resulted in a reactor trip. 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) because although it did contribute to the likelihood of a reactor trip, it did not contribute to the likelihood that mitigating systems will not be available.

  The cause of this finding was determined to have a cross-cutting aspect of Work Planning, in the area of Human Performance associated with the Work Control component. The work planning processes failed to identify the need to include steps to verify the operational status of the controller following completion of the activity, considering the physical conditions and requirements associated with relocating the device. [H.3(a)]. (Section 4OA3.4)

Cornerstone: Mitigating Systems

- Green. The inspectors identified a non-cited violation of 10 CFR 50, Appendix B, Criterion III, Design Control, for the licensee's failure to assure that appropriate quality standards were specified and included in design documents and that deviations from such standards were controlled. Specifically, the licensee failed to ensure that the molded case circuit breakers utilized in the station 120VAC vital instrument power boards were properly seismically qualified for their application. The licensee entered this issue into their corrective action program as PERs 264271, 266599, 286156, and 319161. Corrective actions included revision of applicable procedures to perform re-alignment of breakers in the vital instrument power boards.

  The finding was determined to be greater than minor because it was associated with the design control 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 ensure that the 120VAC vital instrumentation board components had proper seismic qualification had the potential to affect the ability of safety-related equipment to perform its required function under design basis conditions. Using Inspection IMC 0609, "Significance Determination Process," Attachment 4, "Phase 1 - Initial Screening and Characterization of Findings," the finding was determined to have very low safety significance (Green) because it did not represent an actual loss of safety function. No cross-cutting aspect was identified, since the issue was determined to not reflect current licensee performance. (Section 1R15.1)

- Green. A self-revealing non-cited violation of Unit 1 TS 6.8, "Procedures & Programs," was identified for the licensee's failure to provide adequate procedures for maintenance involving the replacement of a safety-related 480V breaker. This resulted in the normal feeder breaker for the safety related 1A2 reactor motor operated valve (MOV) board unexpectedly tripping open when energized following maintenance, causing a loss of power to the board. The licensee entered this issue into their corrective actions program as PER 320274. Licensee corrective actions included revising the applicable breaker maintenance procedure, and reinforcing expectations regarding peer checking and procedure use and adherence.

  The finding was determined to be greater than minor because it was associated with the procedure quality 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 loss of power to the 1A2 reactor MOV board resulted in the inoperability of its associated MOVs affecting two trains of AFW, one train of containment spray (CS), feedwater isolation valves, and containment isolation valves. 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 Resources component. The work package was not adequate to assure nuclear safety due to the complexity and ambiguity associated with the procedure step which involved the jumper installation requirement. [H.2(c)]. (Section 1R15.2)

- Green. A self-revealing non-cited violation of 10 CFR 50 Appendix B Criterion V, "Instructions, Procedures, and Drawings," was identified for the failure to provide adequate procedures for maintenance on the Unit 1 loop 3 feedwater regulating valve (FRV). The applicable procedures did not contain adequate guidance to ensure that the valve's operability was not adversely affected during reassembly. As a result, the FRV was placed in a condition where it was unable to perform its function of main feedwater isolation as required by TS LCO 3.7.1.6. The licensee entered this issue into their corrective actions program as PERs 284451 and 314771. Corrective actions included revision to the applicable work procedure to ensure no inadvertent valve stem rotation during reassembly.

  The finding was determined to be greater than minor because it was associated with the procedure quality 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, as a result of the maintenance activity, the FRV was placed in a condition where it was unable to perform its required function of main feedwater isolation. 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) since the finding 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 Resources component, in that a complete work package which was adequate to assure nuclear safety was not provided for this maintenance activity. The work procedures did not include guidance to ensure that the operability of the FRV was not adversely affected. [H.2(c)]. (Section 4OA2.2)

REPORT DETAILS

Summary of Plant Status:

Unit 1 operated at or near 100 percent rated thermal power (RTP) for the entire inspection period.

Unit 2 operated at or near 100 percent rated thermal power (RTP) for the entire inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection

.1 Tornado Warning

a. Inspection Scope

The inspectors observed the licensee's response to a tornado warning on February 28, 2011. The inspectors reviewed licensee Procedure AOP-N.02, Tornado Watch/Warning, Revision 26, to assess its effectiveness in limiting the risk of tornado-related initiating events and adequately protecting mitigating systems from the effects of a tornado. The inspectors also verified the licensee's performance of required actions. In addition, the inspectors verified that no loose debris was in the 500kV and 161kV Switchyards which would serve as missile hazards during a tornado. This activity constituted one inspection sample.

b. Findings

No findings were identified.

1R04 Equipment Alignment

.2 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 1 B-train Motor-Driven Auxiliary Feedwater System
- Unit 1 B-train Containment Spray System during A-train planned maintenance
- Unit 1 B-train ABGTS and Containment Spray Systems During Elevation 714' Penetration Room Cooler 1A Planned Maintenance

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 669, Unit 2 side
- Auxiliary Building Elevation 690 (Corridor)
- Control Building Elevation 732 (Mechanical Equipment Room and Relay Room)

b. Findings

No findings were identified.

1R06 Flood Protection Measures

.1 Annual Review of Cables Located in Underground Bunkers/Manholes

a. Inspection Scope

The inspectors conducted a review of licensee inspections of safety-related cables located in underground bunkers/manholes subject to flooding. Specifically, inspectors reviewed maintenance records of inspections of 500kV and 161kV Switchyard Underground Cable Tunnels to determine if water was present and, if found, whether it would affect safety-related system operation. In addition, the inspectors reviewed the licensee's corrective action program to ensure that the licensee was identifying underground cabling issues and that they were properly addressed for resolution. Documents reviewed are listed in the Attachment. The inspectors completed one sample.

b. Findings

No findings were identified.

1R07 Heat Sink Performance

a. Inspection Scope

The inspectors observed thermal performance testing of the 1A Centrifugal Charging Pump Lube Oil Cooler and reviewed the results of thermal performance testing of the 1B, 2A, and 2B Centrifugal Charging Pump Lube Oil Coolers to determine whether there were any previously undetected adverse performance trends, whether the acceptance criteria and results appropriately considered differences between testing conditions and design conditions; whether test results were appropriately categorized against pre-established acceptance criteria; and whether the frequency of testing was sufficient to detect degradation prior to loss of heat removal capability below design basis values, and whether tube plugging guidance addressed the evaluation of remaining performance margin prior to plugging. The inspectors also reviewed work documents detailing observations and results of the last internal inspection of the heat exchangers. Documents reviewed are listed in the Attachment. The inspectors completed one sample.

b. Findings

No findings were identified.

1R11 Licensed Operator Requalification Program

a. Inspection Scope

The inspectors performed one licensed operator requalification program review. The inspectors observed a simulator session on January 31, 2011. The training scenario involved a steam generator tube leak degrading into a steam generator tube rupture accident. Additional anomalies included a turbine impulse pressure transmitter failure, a main feedwater pump trip, and a failure of selected B-train engineered safeguards feature equipment to start automatically. The inspectors observed crew performance in terms of: communications; ability to take timely and proper actions; prioritizing, interpreting and verifying alarms; correct use and implementation of procedures, including the alarm response procedures; timely control board operation and manipulation, including high risk operator actions; oversight and direction provided by shift manager, including the ability to identify and implement appropriate Technical Specification (TS) action; and, group dynamics involved in crew performance. The inspectors also observed the evaluators' critique and reviewed simulator fidelity to verify that it matched actual plant response. Documents reviewed are listed in the Attachment to this report. 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 inspectors completed two samples.

- Main control room/Electric board room chillers maintenance rule a(1) plan
- PER 307220, Reactor Power Transient Due to Main Steam Relief 2B2 High Pressure Steam Isolation Valve Shutting

b. Findings

No findings were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope

The inspectors reviewed the following activity 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. This inspection satisfied one inspection sample for Maintenance Risk Assessment and Emergent Work Control.

- Scheduled Maintenance of Motor-driven AFW Pump 2A Motor with Unplanned Unavailability of ERCW Pump M-B and Containment Lower Compartment Cooling Unit 2A

b. Findings

No findings were identified.

1R15 Operability Evaluations

a. Inspection Scope

For the nine 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 nine samples.

- SR 316704, PER 317979, Operability of P-9 Interlock During Reactor Startup
- PER 320274, Reactor MOV Board 1A2-A Normal Supply Breaker Failure
- SR 316717 and 316777, PER 321880, 1A Diesel Generator High Vibrations and Loading Variations
- PER 317995, Missing Pipe Support Clamp on Safety Injection high point vent piping
- PER 264271, 120V vital instrument power board circuit breaker seismic qualification
- PER 324530, 1A Diesel Generator Load Swings
- PER 330332, B Main control room chiller
- PER 332950, Unit 1 RHR discharge piping pressurization
- PER 339317, Unit 1 urgent rod control failure

b. Findings

.1 Failure to Adequately Qualify Molded-Case Circuit Breakers to Safety-Related Application Through Commercial Grade Dedication

Introduction: The inspectors identified a Green non-cited violation of 10 CFR 50, Appendix B, Criterion III, Design Control, for the licensee's failure to assure that appropriate quality standards were specified and included in design documents and that deviations from such standards were controlled. Specifically, the licensee failed to ensure that the molded case circuit breakers utilized in the station 120VAC vital instrument power boards were properly seismically qualified for their application.

Description: On October 10, 2010, the licensee initiated PER 264271 to document that a significant number of breakers in the station 120VAC vital instrument power boards were found to be in a configuration which was not adequately representative of the mounting configuration associated with the applicable seismic qualification testing of the devices. The molded bezel surrounding the operating handle of the breakers in question was not surrounded by the corresponding cut-out in the front panel of the board, such that the front face of the breaker was not in full contact with the back of the front panel.

The inspectors noted that this PER was initially not classified appropriately as a condition adverse to quality, and that no functional evaluation was initiated. In response to the inspectors' questions, the licensee re-classified the PER and performed a functional evaluation which concluded that reasonable assurance exists that the breakers remain capable of performing their required safety function, but that the identified condition was a non-conforming condition since the breaker alignment did not conform to the as-tested configuration. The licensee's cause evaluation determined that the applicable maintenance procedures associated with breaker replacement failed to include adequate requirements to ensure proper installation. Corrective actions were issued to revise the applicable procedures and to perform the re-alignment of the breakers in question.

Seismic qualification testing was performed by a third party vendor as part of the commercial grade dedication process for replacement Heinemann circuit breakers in the station vital instrument power boards. The inspectors noted that the applicable testing procedure stated that "the test specimen shall be mounted on the seismic simulation table in a manner that supplicates the normal in-plant mounting." In these tests, the breaker was bolted to the test platform. The actual in-plant configuration is such that the front of the breakers is press-fit to the back of the front switchboard panel. In many cases, as noted above, gaps were identified to exist between the front of a number of breakers and the switchboard front panels.

IEEE Standard 344 (1975) required, in part, that the test mounting dynamically simulate the plant-specific mounting and that the test accelerations adequately bound the required response spectrum (RRS) for the application. The licensee translated maximum accelerations seen on the panel itself as bounding the subcomponent accelerations without adequately demonstrating the rigidity of mounting necessary to support that assumption. As the mounting configuration of the devices to the test platform did not mimic the actual installed mounting, the licensee had responsibility to ensure, by analysis, that the test accelerations adequately bounded the RRS. The licensee failed to ensure such analysis was conducted. Specifically, calculation SCG4M00556, Seismic Qualification of 120V A.C. Vital Instrument Power Boards and Modifications to The Boards, revision 6, failed to fully establish that the method of support of the breakers within the board was a rigid mounting system, that the replacement breaker seismic qualification test mounting represents a suitable mounting method, or that the test accelerations to which the device was subjected were, in fact, bounding.

Analysis: Failure to adequately qualify commercial grade molded-case circuit breakers to their safety-related application was a performance deficiency. The finding was determined to be greater than minor because it was associated with the design control 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 ensure that the 120VAC vital instrumentation board components received proper seismic qualification had the potential to affect the ability of the safety-related equipment to perform its required function under design basis conditions. Using Inspection IMC 0609, “Significance Determination Process,” Attachment 4, “Phase 1 - Initial Screening and Characterization of Findings,” the finding was determined to have very low safety significance (Green) because it did not represent an actual loss of safety function.

No cross-cutting aspect was identified, since the issue was determined to not reflect current licensee performance.

Enforcement: 10 CFR 50, Appendix B, Criterion III, Design Control, required, in part, that design control measures shall include provisions to assure that appropriate quality standards are specified and included in design documents and that deviations from such standards are controlled. Contrary to the above, on October 20, 2003, the licensee’s design control measures failed to assure that appropriate quality standards are specified and included in design documents. Specifically, the licensee failed to demonstrate the necessary conditions to support the commercial grade dedication and seismic qualification of molded case circuit breakers to safety-related application within the station 120VAC vital instrumentation boards. Because this violation was determined to be of very low safety significance and has been entered into the licensee's corrective action program as PERs 264271, 266599, 286156, and 319161, it is being treated as an NCV consistent with the NRC Enforcement Policy: NCV 05000327,328/2011002-01, “Failure to Adequately Qualify Molded-Case Circuit Breakers to Safety-Related Application Through Commercial Grade Dedication.”

.2 Loss of 480-V Motor Control Center Due To Inadequate Maintenance Procedures for Breaker Replacement

Introduction. A Green self-revealing NCV of Unit 1 TS 6.8, “Procedures & Programs,” was identified for the licensee’s failure to provide adequate procedures for maintenance involving the replacement of a safety-related 480V breaker. This resulted in the normal feeder breaker for the safety related 1A2 reactor motor operated valve (MOV) board tripping opening when energized from the bus following maintenance, causing a loss of power to the board.

Description. On February 3, 2011, the licensee implemented WO 08-776742-000, which included the replacement of the normal feeder breaker, 1-BCTB-201-DK/8B-A, for the 1A2 reactor MOV board. The MOV board supplies power to MOVs in several standby safety systems, including the essential raw cooling water (ERCW), auxiliary feedwater (AFW), component cooling system (CCS), containment spray (CS), and residual heat removal (RHR) systems. The board was transferred to its alternate power supply during the replacement of the normal supply breaker in order to maintain system operability throughout the maintenance activity. Following completion of work activities, including post-maintenance testing verifications, the operators attempted to restore power to the MOV board from the normal supply. The MOV board unexpectedly lost power during this evolution. The operators restored power after several minutes using the alternate power source. This resulted in an unexpected loss of power to safety-related loads affecting operability of several safety-related systems.

The licensee entered this issue into their CAP as PER 320274 and performed an apparent cause evaluation. The cause of the breaker trip was determined to be the absence of a required permanent jumper in the Westector current sensing circuit of the replacement breaker. This jumper was determined to not have been installed due to the applicable maintenance procedure having inadequately addressed whether this jumper needed to be installed in the replacement breaker. The licensee cause determination determined that this procedure step was complex and ambiguous, and failed to clearly indicate the need for the jumper installation; additionally no independent verification was directed. The maintenance technician performing the activity did not install the jumper, and did not request a peer check at this step.

Analysis. The licensee's failure to satisfactorily configure a safety-related 480V breaker prior to installation was a performance deficiency. The finding was determined to be greater than minor because it was associated with the procedure quality 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 loss of power to the 1A2 reactor MOV board resulted in the inoperability of its associated MOVs and the 'A' train of their respective safety systems. The affected systems included two of three trains of AFW required by LCO 3.7.1.2, one train of CS required by LCO 3.6.2.1, feedwater isolation valves required by LCO 3.7.1.6, and containment isolation valves required by LCO 3.6.3. 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 Resources component. The work package was not adequate to assure nuclear safety due to the complexity and ambiguity associated with the procedure step which involved the jumper installation requirement. [H.2(c)].

Enforcement. Unit 1 TS 6.8.1.a required, in part, that written procedures be established, implemented, and maintained covering the activities specified in Appendix A, “Typical Procedures for Pressurized Water Reactors and Boiling Water Reactors,” of Regulatory Guide (RG) 1.33, “Quality Assurance Program Requirements (Operations),” Revision 2, dated February 1978. RG 1.33 Appendix A Section 9.a, “Procedures for Performing Maintenance,” required, in part, that maintenance that can affect the performance of safety-related equipment should be properly pre-planned and performed in accordance with written procedures, documented instructions, or drawings appropriate to the circumstances. Contrary to the above, on February 3, 2011, the licensee failed to establish and implement adequate written procedures appropriate to the circumstances for maintenance that could affect the performance of safety-related equipment. Specifically, the maintenance instructions associated with the replacement of the 1A2 reactor MOV board normal supply breaker, 1-BCTB-201-DK/8B-A, were not adequate to satisfactorily configure the breaker prior to installation. Because the finding was of very low safety significance and has been entered into the licensee’s CAP as PER 320274, this violation is being treated as an NCV, consistent with the NRC Enforcement Policy: NCV 05000327/2011002-02, “Loss of 480-V Motor Control Center Due To Inadequate Breaker Replacement Maintenance.”

1R18 Plant Modifications

.1 Temporary Modifications

a. Inspection Scope

The inspectors reviewed the temporary modifications 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 1-10-020-003, 1A-A Auxiliary Feedwater Pump Motor
- WO 111940742, B-train main control room chiller temperature switch

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 two samples.

b. Findings

No findings were identified.

1R19 Post-Maintenance Testing

a. Inspection Scope

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

- WO 111847313, Unit 1 Loop 3 OT/ΔT Trip/Runback circuit failure
- WO 111814032, MSR 2B2 bypass control valve failure resulting in thermal power transient
- WO 111947892, 1A Diesel generator governor motor operated potentiometer
- WO 112048990, Unit 1 rod control power cabinet 2BD urgent failure

b. Findings

No findings were identified.

1R22 Surveillance Testing

a. Inspection Scope

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

In-Service Tests:

- 1-SI-SXP-062-201.A, Centrifugal Charging Pump 1A-A Performance Test, Rev. 16
- 1-SI-SXP-072-201.A, Containment Spray Pump 1A-A Performance Test, Rev. 17
- 0-SI-SXV-072-215.A, Closing Test of Containment Spray Check Valves 72-506 and 72-528, Rev. 1

RCS leakage test:

- 0-SI-OPS-068-137.0, Reactor Coolant System Water Inventory, Rev. 23

Routine Surveillance Tests:

- 2-SI-OPS-082-024.A, 2A-A D/G 24 Hour Run and Load Rejection Testing, Rev. 18
- 0-SI-NUC-000-007.0, Measurement of the At-Power Moderator Temperature Coefficient, Rev. 16 (Unit 2 performance)

- 1-TI-CEM-000-016.31, Primary Sampling – Reactor Coolant, Refueling Canal and Transfer Canal, Rev. 6

b. Findings

No findings were identified.

1EP6 Drill Evaluation

a. Inspection Scope

Resident inspectors evaluated the conduct of routine licensee emergency drill on March 15, 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 45. 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. The inspectors completed one sample.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification

a. Inspection Scope

The inspectors sampled licensee submittals for the 2 PIs listed below for the period from January 1, 2010, through December 31, 2010, 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: Barrier Integrity

- Reactor Coolant System Activity
- Reactor Coolant System Leakage

The inspectors reviewed portions of the operations and chemistry logs to verify whether the licensee had accurately determined and reported the Reactor Coolant System (RCS) activity and leakage during the previous four quarters for both units. The inspectors also observed the performance of procedures 0-SI-OPS-068-137.0, RCS Water Inventory, and 1-TI-CEM-000-016.31, Primary Sampling – Reactor Coolant, Refueling Canal and Transfer Canal. Documents reviewed are listed in 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 Selected Issue Follow-up: Unit 1 Loop 3 Feedwater Regulating Valve Failure

a. Inspection Scope

On November 15, 2010, Unit 1 was in Mode 2 while conducting a plant startup from a planned refueling outage. When the main feedwater system was being placed in service, operators observed loop 3 steam generator level rising unexpectedly. This indication led to the discovery that the loop 3 feedwater regulating valve (FRV) was in an inoperable condition due to its inability to close as required by TS LCO 3.7.1.6 for main feedwater isolation.

The inspectors reviewed the licensee's actions in response to this condition, including actions to determine and correct the cause of the failure. The inspectors reviewed PERs 284451 and 314771 dealing with this event, interviewed engineering and operations personnel, and reviewed the licensee's corrective actions.

b. Findings and Observations

The Unit 1 loop 3 FRV was declared inoperable by operators upon discovery of the intermediate valve position, and was subsequently isolated on November 16, 2010, to comply with action b of LCO 3.7.1.6. The licensee entered the discovery of the condition on the FRV into the CAP as PER 284451 on November 16, 2011. Corrective maintenance was performed on the FRV to restore it to an operable condition.

The inspectors noted that no cause evaluation was performed in response to PER 284451, nor had there been any evaluation performed for potential reportability of the condition. The inspectors questioned the lack of a cause evaluation for the identified condition on the basis that an extent of cause could potentially apply to other similar components, also the condition could potentially be reportable depending in part on whether firm evidence that inoperability existed prior to discovery could be identified. In response, the licensee initiated PER 314771 on January 28, 2011, and performed an apparent cause evaluation.

The inspectors determined that the licensee's apparent cause evaluation associated with PER 314771 was thorough and that the corrective actions appeared to be adequate to address the identified causes. The licensee's investigation determined that the 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. The licensee identified that the applicable maintenance procedures did not contain guidance to verify no stem rotation either during or following the outage maintenance. The licensee's corrective actions included revision to 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.

On March 31, 2011, the inspectors observed that no evaluation for potential reportability had been conducted for the identified condition. In response, the licensee initiated PER 349161 on April 4, 2011, to perform an evaluation of past operability and reportability. That evaluation resulted in LER 05000327/2011-002-00, "Feedwater Regulator Valve Inoperable," being submitted on April 22, 2011.

One finding was identified, as detailed below.

<u>Introduction</u>. A self-revealing Green non-cited violation of 10 CFR 50 Appendix B Criterion V, "Instructions, Procedures, and Drawings," was identified for the failure to provide adequate procedures for maintenance on the Unit 1 loop 3 feedwater regulating valve (FRV). The applicable procedures did not contain adequate guidance to ensure that the valve's operability was not adversely affected following the completion of maintenance. As a result, the Unit 1 loop 3 FRV was left in a condition where it was unable to perform its function of main feedwater isolation as required by TS LCO 3.7.1.6.

<u>Description</u>. On October 11, 2010, the licensee performed WO 09-777032-000 during a refueling outage on the Unit 1 loop 3 FRV actuator, which included diaphragm replacement. While in Mode 2 during plant startup on November 15, 2010, the FRV was discovered to be in an inoperable condition when operators observed loop 3 steam generator level rising when the main feedwater system was being placed in service. Unit 1 TS LCO 3.7.1.6 requires that each FRV be operable – i.e. capable of being closed to isolate main feedwater – in Modes 1, 2, and 3. The loop 3 FRV was declared inoperable upon discovery of the condition (due to its inability to close), and was subsequently isolated on November 16, 2010, to comply with action b of LCO 3.7.1.6.

The licensee's cause evaluation determined that the 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. This resulted in the FRV being left in an inoperable condition following the maintenance activity. It was identified that the applicable maintenance procedures did not contain guidance to verify no stem rotation during the outage maintenance, nor did they contain steps to adequately verify the valve's ability to achieve its required closed position following the maintenance.

The inspectors reviewed licensee Procedure MMDP-1, "Maintenance Management System," Revision 19, Section 3.2.3, which contained guidance pertaining to the level of detail of work order content. In particular, "Work orders are to be complete and accurate…Completeness and accuracy for the content of a work order package includes the following considerations: …Level of detail of work instructions is right for the task and for the performers." Also, "The work order package should contain sufficient controls and instructions to perform the activity in a safe, quality manner without unanticipated impact on the plant and without the introduction of latent problems into the equipment."

<u>Analysis</u>. The licensee's failure to satisfactorily accomplish maintenance on the Unit 1 loop 3 FRV such that the component was restored to an operable condition upon completion was a performance deficiency. The finding was determined to be greater than minor because it was associated with the procedure quality 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, as a result of the maintenance activity, the FRV was placed in a condition where it was unable to perform its required function of main feedwater isolation. 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) since the finding 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 Resources component, in that a complete work package which was adequate to assure nuclear safety was not provided for this maintenance activity. The work procedures did not include guidance to ensure that the operability of the FRV was not adversely affected. [H.2(c)].

<u>Enforcement</u>. 10 CFR 50 Appendix B, Criterion V, required, in part, that activities affecting quality shall be prescribed by documented procedures of a type appropriate to the circumstances, and that procedures shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Contrary to this, on October 11, 2010, the licensee failed to provide adequate procedures which included appropriate acceptance criteria for determining that activities affecting quality have been satisfactorily accomplished. Specifically, the procedures associated with Unit 1 loop 3 FRV maintenance did not

contain adequate guidance to ensure that the valve's operability was not adversely affected by the activity. 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 314771, it is being treated as an NCV consistent with the NRC Enforcement Policy: NCV 05000327/2011002-03, "Inadequate Maintenance Procedures Result in Inoperable Feedwater Regulating Valve."

4OA3 Event Follow-up

.1 LCO 3.0.3 Entry Due To Both Trains of CRACS Inoperable

a. Inspection Scope

On February 15, 2011, the inspectors responded to a condition which placed both Units in TS LCO 3.0.3 and 3.0.5 shutdown actions due to both trains of CRACS being inoperable. At 7:05 a.m. on February 15, 2011, the B-train main control room (MCR) chiller was found tripped, making the B-train control room air conditioning system (CRACS) to be inoperable. TS LCO 3.7.15 for both Units 1 and 2 requires two independent CRACS trains to be operable. The A-train CRACS was also inoperable at this time due to being previously tagged out for planned maintenance. Therefore, TS LCO 3.0.3 required both Units to initiate action within one hour to place both units in hot standby within the next 6 hours. The 1A EDG had also been previously declared inoperable due to load swings which had been observed during a recent surveillance test. Since the 1A EDG is the emergency backup power supply for the A-train of CRACS, TS LCO 3.0.5 was also applicable for both Units at this time. With the emergency backup power supply for the A-train CRACS inoperable, TS LCO 3.0.5 required that action be initiated within 2 hours to place both Units 1 and 2 in hot standby within the next 6 hours.

The A-train MCR chiller was recovered and restored at 10:05 a.m. on February 15, 2011. The 1A EDG was declared operable at 10:30 a.m. on February 15, 2011, which allowed both Units to exit TS LCO 3.0.3 and 3.0.5.

The inspectors discussed this event 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. This event was entered into the licensee's CAP as PERs 323780, 323782, 323785, 324814, 325326, and 325906.

b. Findings

No findings were identified.

.2 Unit 1 RHR Discharge Header Pressurization

a. Inspection Scope

On March 4, 2011, the inspectors responded to an abnormal condition involving the pressurization of the Unit 1 RHR discharge header. Operators responded to an alarm in the main control room, and observed RHR system discharge header pressure to be abnormally high.

The inspectors discussed the event 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, including taking actions to manually depressurize the RHR system as needed to compensate for the pressurization and maintain the system in an operable condition. This event was documented in the licensee's CAP as PERs 332950 and 334162. The licensee determined that the most likely source of system pressurization is RCS check valve leakage. The inspectors verified that the licensee implemented adequate actions to ensure that applicable TS limits for RCS leakage and check valve leakage are not exceeded.

b. Findings

No findings were identified.

.3 (Closed) Licensee Event Report (LER) 05000327/2010-003-00, Unit 1 Manual Reactor Trip as a Result of a Fire in the Main Generator Neutral Bushing Bus Duct Housing

a. Inspection Scope

On December 20, 2010, Unit 1 was manually tripped as a result of a fire associated with the main generator. Operations personnel entered the applicable abnormal operating procedure for plant fires. The safety systems performed as designed for the reactor trip. The fire brigade extinguished the fire. The event was documented in the licensee corrective action program as PER 299269.

The inspectors reviewed the LER, PER and Root Cause Evaluation Report to verify that the cause of the fire was identified and that corrective actions were appropriate. The cause of the fire was determined to be cyclic vibration induced failure of the main generator neutral bushing which allowed a hydrogen leak that auto ignited. The inspectors concluded that the licensee's corrective actions were appropriate, including developing preventative maintenance procedures for refurbishment/replacement of the bushings and procedures for periodic inspection of the bushings.

The inspectors discussed the trip with operations, engineering, and licensee management personnel to gain an understanding of the event and assess follow-up actions. The inspectors reviewed operator actions taken to determine whether they were in accordance with licensee procedures and TS, and reviewed unit and system indications to verify whether actions and system responses were as expected and designed. The inspectors verified that timely notifications were made in accordance with 10 CFR 50.72, that licensee staff properly implemented the appropriate plant procedures, and that plant equipment performed as required. This LER is closed.

b. Findings

No findings were identified.

.4 (Closed) Licensee Event Report (LER) 05000327/2010-002-00, Manual Reactor Trip as a Result of a Failure of Feedwater Control to Maintain Steam Generator Levels

a. Inspection Scope

On November 16, 2010, Unit 1 was manually tripped due to imminent loss of steam generator water level following a manual turbine trip while operating at approximately 26% 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 events were reported to the NRC as event notification (EN) 46424 and documented in the licensee corrective action program as PERs 285349, 287687, 287695, and 290089.

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 root cause report to assess the detail and thoroughness of the evaluation and the adequacy of the proposed corrective actions.

The inspectors reviewed the LER and associated PERs to verify that the cause of the reactor trips was identified and whether corrective actions were appropriate. The cause of the lack of steam generator water level control following the turbine trip was determined to be due to erratic behavior of the steam dumps. The licensee’s root cause evaluation identified the root and contributing causes to be the failure to adequately connect the steam dump load reject controller to its power source when the controller was relocated as part of a maintenance activity which implemented a plant modification, and the failure to have identified and performed post-modification testing to verify the operational condition of the controller following the maintenance. The inspectors concluded that the licensee’s corrective actions to this event were appropriate, including revision to post-modification testing procedures to require an additional PMT review for large/complex modifications, as well as revision to applicable maintenance procedures to require verification for plug-in type connections. The inspectors also verified that timely notifications were made in accordance with 10 CFR 50.72, that licensee staff properly implemented the appropriate plant procedures, and that available plant equipment performed as required during the event.

One finding was identified as discussed below. This LER is closed.

b. Findings

Introduction. A Green self-revealing finding was identified for the licensee's failure to perform adequate post-maintenance testing, as specified by procedures SPP-8.3, Post-Modification Testing, revision 10, and NPG-SPP-06.3, Pre-/Post-Maintenance Testing, revision 0, in conjunction with a work order which implemented a plant modification on Unit 1 and included the relocation of the steam dump load reject controller. This resulted in a manual trip of Unit 1 when the steam dump load reject controller was unable to function properly following a turbine trip from 26 percent RTP due to not being properly plugged in.

Description. On November 16, 2010, with Unit 1 operating at approximately 26 percent RTP during startup from a refueling outage, a moisture separator reheater (MSR) shell side relief valve lifted. Operators responded by tripping the turbine in accordance with plant procedures. Following the turbine trip, operators were unable to maintain adequate control of steam generator levels, and responded in accordance with plant procedures by manually tripping the reactor due to continued lowering of steam generator water level.

This event was entered into the licensee's corrective action program as PERs 285349, 287687, 287695, and 290089. The licensee performed a root cause evaluation, which determined that the lack of steam generator water level control following the turbine trip was due to erratic behavior of the steam dumps. The steam dump load reject controller power plug was subsequently found by the licensee to be not fully connected. This controller had been relocated by WO 07-780032-016, which was performed on October 12, 2010, during the refueling outage. This WO implemented a digital feedwater control upgrade plant modification on Unit 1, which involved the relocation of the steam dump load reject controller to a different location. The controller was a rail-mounted module, and the location of the plug involved limited access and visibility. The licensee identified that there were no steps in the WO instructions to verify the plug was fully engaged, and no method of verifying the operational status of the device following the activity was provided.

The licensee also identified that the load reject controller calibration procedure was scheduled to be performed during the outage, but was not performed due to resource and scheduling issues. The inspectors determined that this procedure would likely have identified the problem with the controller. This activity was deferred to be performed on-line; however, the deferral request was not officially approved.

The inspectors reviewed the UFSAR and noted that following a turbine trip from an initial power level below 50 percent RTP, the reactor should not be required to be tripped, but instead the reactor plant is designed to be maintained in a stable and controlled manner by plant systems, including the steam dumps via the load reject controller.

The inspectors reviewed licensee procedure SPP-8.3, Post-Modification Testing, revision 10, and noted that it contained the following requirements. Section 2.0.B stated that “the test shall verify that no undesirable affects or new deficiencies are created.” Section 2.0.K stated, “Testing must not be restricted to obvious functions. Seemingly minor changes sometimes result in considerable reductions in system performance.” The inspectors also reviewed licensee procedures NPG-SPP-06.3, Pre-/Post-Maintenance Testing, revision 0, and NPG-SPP-06.1, Work Order Process, revision 0, Appendix A, Minor Maintenance. These procedures both contained the following requirement: “Post-maintenance testing shall be based on the extent of maintenance performed. The PMT shall be sufficiently comprehensive to ensure that the maintenance performed does not adversely affect the equipment’s ability to perform its intended function, …and that no new or related problems were created by the maintenance activity.”

The inspectors concluded that the failure to include verification of the operational condition of the steam dump load reject controller in the post-maintenance testing associated with WO 07-780032-016 constituted a failure to meet the above site standards.

Analysis. The licensee’s failure to perform adequate post-maintenance testing, as specified by procedures SPP-8.3, Post-Modification Testing, revision 10, and NPG-SPP-06.3, Pre-/Post-Maintenance Testing, revision 0, following maintenance which relocated the steam dump load reject controller was a performance deficiency. The finding was determined to be greater than minor because it was associated with the equipment performance attribute of the initiating events cornerstone and affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, it resulted in the controller being in a condition where it was unable to function properly, which led to a reactor trip. 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) because it did not contribute to both the likelihood of a reactor trip and the likelihood that mitigating systems will not be available.

The cause of this finding was determined to have a cross-cutting aspect in the area of Human Performance associated with the Work Control component. The work planning processes failed to identify the need to include steps to verify the operational status of the controller following completion of the activity, considering the physical conditions and requirements associated with relocating the device. [H.3(a)].

Enforcement. Enforcement action does not apply because the performance deficiency did not involve a violation of regulatory requirements. No violation of NRC requirements was identified since the subject steam dump load reject controller was not a safety-related component. Because this finding has been entered into the licensee’s corrective action program as PER 285349, and has very low safety significance, it is identified as FIN 05000327/2011002-04, “Reactor Trip due to Unplugged Steam Dump Load Reject Controller.”

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.

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

a. Inspection Scope

The inspectors reviewed the sixth 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 April 7, 2010, the resident inspectors presented the inspection results to Mr. M. Skaggs and other members of his staff, who acknowledged the findings. The inspectors asked the licensee whether any of the material examined during the inspection should be considered proprietary. No proprietary information was identified.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel
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. Foster, Performance Improvement Manager
J. Furr, Quality Assurance Manager
Z. Kitts, Licensing
R. Krich, Licensing Vice President
K. Langdon, Plant Manager
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
C. Ware, Training Director
K. Wilkes, Operations Support Superintendent
J. Williams, Site Engineering Director
S. Young, Site Security Manager

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

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000327,328/2011002-01	NCV	Failure to Adequately Qualify Molded-Case Circuit Breakers to Safety-Related Application Through Commercial Grade Dedication (Section 1R15.1)
05000327/2011002-02	NCV	Loss of 480-V Motor Control Center Due To Inadequate Breaker Replacement Maintenance (Section 1R15.2)

05000327/2011002-03	NCV	Inadequate Maintenance Procedures Result in Inoperable Feedwater Regulating Valve (Section 4OA2.2)
05000327/2011002-04	FIN	Reactor Trip due to Unplugged Steam Dump Load Reject Controller (Section 4OA3.4)

Closed

05000327/2010-003-00	LER	Unit 1 Manual Reactor Trip as a Result of a Fire in the Main Generator Neutral Bushing Bus Duct Housing (Section 4OA3.3)
05000327/2010-002-00	LER	Manual Reactor Trip as a Result of a Failure of Feedwater Control to Maintain Steam Generator Levels (Section 4OA3.4)

LIST OF DOCUMENTS REVIEWED

Section R01: Adverse Weather Protection
AOP-N.02, Tornado Watch/Warning, revision 26

Section R04: Equipment Alignment
0-SI-OPS-072-034.0, Containment Spray System Valve Position Verification, Rev. 5
0-SO-72-1 Att. 3, Containment Spray System Power Checklist, Rev. 10
0-SO-72-1 Att. 9, Containment Spray System Valve Checklist, Rev. 11
Containment Spray Student Handout, Rev.1
1-SO-3-2 Att. 1, Auxiliary Feedwater System Power Checklist, Rev. 11
1-SI-OPS-003-005.M, Auxiliary Feedwater Valve Position Verification, Rev. 3
1-SO-3-2, Auxiliary Feedwater System, Rev. 44
1-SO-3-2 Att. 2, Auxiliary Feedwater System Valve Checklist, Rev. 16

Section R05: Fire Protection
FPDP-1, Conduct of Fire Protection, Revision 1
0-PI-FPU-317-299.W, Att. 8, Shift Check List
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

Section R06: Flood Protection Measures
WO 11108121224, Check Standing Water Level in Manholes/Handholes
TVA letter to NRC dated May 4, 2007, TVA response to GL 2007-01
SRs 305911, 312734, 308869, 304738, 314815
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FHI-3, Movement of Fuel, Revision 60
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Section R22: Surveillance Testing
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EPIP-1, Emergency Plan Classification Matrix, revision 45

Section 4OA1: Performance Indicator Verification
SPP-3.4, Performance Indicator Program, Revision 10
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OPDP-8, Limiting Conditions for Operation Tracking, revision 5
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NPG-SPP-01.2, Administration of Site Technical Procedures, Revision 1
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LIST OF ACRONYMS

CAP	Corrective Action Program
CFR	Code of Federal Regulations
ED	electronic dosimeter
HPT	Health Physics Technician
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
RCA	radiologically controlled area
Rev.	Revision

RS	Radiation Safety
RWP	radiation work permit
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