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

February 13, 2013

Mr. Joseph W. Shea
Vice President, Nuclear Licensing
Tennessee Valley Authority
1101 Market Street, LP 3D-C
Chattanooga, TN 37402-2801

SUBJECT: SEQUOYAH NUCLEAR PLANT - NRC INTEGRATED INSPECTION REPORT 05000327/2012005, 05000328/2012005

Dear Mr. Shea:

On December 31, 2012, 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 11, 2013 with Mr. P. Simmons and other members of his 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.

Based on the results of the inspection, the NRC has identified an issue that was evaluated under the risk significance determination process as having very low safety significance (Green). The NRC has also determined that one violation is associated with this issue. The violation is being cited because Tennessee Valley Authority's Sequoyah Nuclear Station failed to restore compliance with NRC requirements within a reasonable time after a previous violation was identified in NRC Inspection Report 05000327(328)/2011002 (issued May 12, 2011). This is consistent with the NRC Enforcement Policy, Section 2.3.2, which states in part that a cited violation will be considered if the licensee fails to restore compliance within a reasonable time after a violation is identified. Further, the NRC Enforcement Manual, Section 3.1.2.A.1.b).1)., defines restoring compliance to include those actions taken to stop an ongoing violation from continuing.

You are required to respond to this letter and should follow the instructions specified in the enclosed Notice when preparing your response. If you have additional information that you believe the NRC should consider, you may provide it in your response to the Notice. The NRC review of your response to the Notice will also determine whether further enforcement action is necessary to ensure compliance with regulatory requirements.

Additionally, based on the results of this inspection, the inspectors identified ten issues of very low safety significance. Five NRC-identified findings of very low safety significance (Green) were identified during this inspection. All of these findings were determined to involve violations of NRC requirements. One SL IV traditional enforcement violation was also identified. Further, four licensee-identified violations which were determined to be of very low safety significance are listed in this report. Because of the very low safety significance and because they are entered into your Corrective Action Program (CAP), the NRC is treating these findings as non-cited violations (NCVs) consistent with the NRC Enforcement Policy. If you contest any of these NCVs, 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 cross-cutting aspect assigned to 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, RII, and the NRC Senior Resident Inspector at 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 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/

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: 1. Notice of Violation
2. Inspection Report 05000327/2012-005, 05000328/2012-005
w/Attachment: Supplemental Information

cc w/encl: (See page 3)

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Sincerely,

/RA/
Scott M. Shaeffer, Chief
Reactor Projects Branch 6
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Docket Nos.: 50-327, 50-328
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cc w/encl: (See page 3)

X PUBLICLY AVAILABLE ☐ NON-PUBLICLY AVAILABLE ☐ SENSITIVE X NON-SENSITIVE
ADAMS: ☐ YesACCESSION NUMBER:_______________________ X SUNSI REVIEW COMPLETE ☐ FORM 665 ATTACHED

OFFICE	RII:DRP	RII:DRS	RII:DRP	RII:DRS	RII:DRS	RII:DRS	RII:DRS
SIGNATURE	Via email	Via email	Via email	Via email	Via email	Via email	Via email
NAME	RLewis	WDeschaine	RMonk	KMiller	MSpeck	RHamilton	LMahlahla
DATE	02/13/2013	02/13/2013	02/13/2013	02/13/2013	02/13/2013	02/13/2013	02/13/2013
E-MAIL COPY?	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO

OFFICE	RII:DRS	RII:DRS	RII:DRS	NRR	RII:DRS	RII:DRS	RII:DRS
SIGNATURE	Via email	Via email	Via email	Via email	Via email	Via email	Via email
NAME	CDykes	RKellner	JRivera	JLaughlin	RCarrion	BCollins	CFletcher
DATE	02/13/2013	02/13/2013	02/13/2013	02/12/2013	02/13/2013	02/12/2013	02/13/2013
E-MAIL COPY?	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO

OFFICE	RII:DRS	RII:DRP	RII:DRP				
SIGNATURE	Via email	JDH /RA/	SMS /RA/				
NAME	JShehee	JHamman	SShaeffer				
DATE	02/12/2013	02/13/2013	02/12/2013				
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\2012\SEQ 12-05\SEQUOYAH 2012 005.DOCX

cc w/encl:
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Site Vice President
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Ann Harris
341 Swing Loop
Rockwood, TN 37854

Letter to J.W. Shea from Scott Shaeffer dated February 13, 2013

SUBJECT: SEQUOYAH NUCLEAR PLANT - NRC INTEGRATED INSPECTION REPORT 05000327/2012005, 05000328/2012005

Distribution w/encl:
C. Evans, RII
L. Douglas, RII
OE Mail
RIDSNRRDIRS
PUBLIC
RidsNrrPMSequoyah Resource

NOTICE OF VIOLATION

Tennessee Valley Authority
Sequoyah Nuclear Plant - Units 1 and 2
Soddy-Daisy, TN

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

During an NRC inspection conducted on October 1 – December 31, 2012, a violation of NRC requirements was identified. In accordance with the NRC Enforcement Policy, the violation is listed below:

10 CFR 50, Appendix B, Criterion III, "Design Control," states that measures shall be established for the review for suitability of application of materials, parts, and equipment that are essential to the safety related functions of the structures, systems, and components (SSCs). The design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program. Where a test program is used to verify the adequacy of a specific design feature in lieu of other verifying or checking processes, it shall include suitable qualifications testing of a prototype unit under the most adverse design conditions.

Contrary to the above, measures used to review the suitability of application of materials, parts, and equipment essential to the safety related functions of molded case circuit breakers and measures to provide for the verification of checking the adequacy of design, such as, calculational methods, performing a suitable test program, including qualification testing of a prototype unit under the most adverse design conditions, were not adequate in that:

1. Licensee Drawing 1,2-45N706 (Series) contains a note that, "The circuit breaker restraint bars may be modified by adding extra scotch 13 or 23 tape, approximately 1/4 inch thick and 1 inch wide, to secure the newer styles of Heinemann breakers."

   - The licensee failed to perform an adequate review for suitability of application of parts and material used to modify dimensional critical characteristics in molded case circuit breaker application; further, the licensee failed to verify the adequacy of design for the modification and the effects on essential safety related functions of the circuit breakers.
   - The inspectors observed that the licensee's application of the tape repair was inconsistent and not governed by suitable acceptance criteria. Several instances of tape only on the lower restraint bar, with a gap at the top restraint bar were observed, along with instances of tape modification to both top and bottom restraint bars.

2. The inspectors identified additional instances in the field where the licensee had failed to perform an adequate review for suitability of application of parts and material used to modify dimensional critical characteristics in molded case circuit breaker application, and for which the adequacy of design had not been properly verified or checked:

   - The licensee had installed maintenance aids (foam weather-stripping and vertical support rods applied to the restraint bars) which were not on drawings, nor listed in the original seismic testing documents.

- The licensee had installed circuit breakers which were less conservatively seated across the rear restraint bars than those originally tested in that they only made contact from the vertical centerline to one side, which might allow the breaker to roll out in a seismic event.
- The inspectors observed several instances where molded case circuit breakers which had been aligned to protrude through the face-plate did not exhibit seating upon the weather-stripping as neighboring breakers were.

This violation is associated with a Green Significance Determination Process finding.

Pursuant to the provisions of 10 CFR 2.201, Tennessee Valley Authority is hereby required to submit a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001 with a copy to the Regional Administrator, Region II, and a copy to the NRC Resident Inspector at the facility that is the subject of this Notice, within 30 days of the date of the letter transmitting this Notice of Violation (Notice). This reply should be clearly marked as a "Reply to a Notice of Violation" and should include for each violation: (1) the reason for the violation, or, if contested, the basis for disputing the violation or severity level, (2) the corrective steps that have been taken and the results achieved, (3) the corrective steps that will be taken, and (4) the date when full compliance will be achieved. Your response may reference or include previous docketed correspondence, if the correspondence adequately addresses the required response. If an adequate reply is not received within the time specified in this Notice, an order or a Demand for Information may be issued as to why the license should not be modified, suspended, or revoked, or why such other action as may be proper should not be taken. Where good cause is shown, consideration will be given to extending the response time.

If you contest this enforcement action, you should also provide a copy of your response, with the basis for your denial, to the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001.

Because your response will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's document system (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html, to the extent possible, it should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the public without redaction. If personal privacy or proprietary information is necessary to provide an acceptable response, then please provide a bracketed copy of your response that identifies the information that should be protected and a redacted copy of your response that delete such information. If you request withholding of such material, you must specifically identify the portions of your response that you seek to have withheld and provide in detail the bases for your claim of withholding (e.g., explain why the disclosure of information will create an unwarranted invasion of personal privacy or provide the information required by 10 CFR 2.390(b) to support a request for withholding confidential commercial or financial information). If safeguards information is necessary to provide an acceptable response, please provide the level of protection described in 10 CFR 73.21.

In accordance with 10 CFR 19.11, you may be required to post this Notice within two working days.

Dated this 13th day of February, 2013

U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos.:	50-327, 50-328
License Nos.:	DPR-77, DPR-79
Report Nos.:	05000327/2012-005, 05000328/2012-005
Licensee:	Tennessee Valley Authority (TVA)
Facility:	Sequoyah Nuclear Plant, Units 1 and 2
Location:	Sequoyah Access Road Soddy-Daisy, TN 37379
Dates:	October 1 – December 31, 2012
Inspectors:	R .Lewis, Senior Resident Inspector (Acting) W. Deschaine, Resident Inspector R. Monk, Senior Resident Inspector (1R12, 1R15) K. Miller, Resident Inspector (1R12, 1R15, 4OA2) M. Speck, Senior Emergency Preparedness Inspector (1R19, 1R20) R. Hamilton, Sr. Health Physicist (2RS5, 4OA1, 4OA5, 4OA6) L. Mahlahla, Sr. Health Physicist (2RS1, 4OA1, 4OA5, 4OA7) C. Dykes, Health Physicist (2RS3, 2RS4, 4OA5) R. Kellner, Health Physicist (2RS2, 4OA5) J. Rivera, Health Physicist (2RS2) J. Laughlin, Emergency Preparedness Inspector (1EP4) R. Carrion, Senior Reactor Inspector (4OA5.5) B. Collins, Reactor Inspector (1R08, 4OA5.5) C. Fletcher, Senior Reactor Inspector (4OA5.5) J. Shehee, Senior Security Inspector (4OA5.5)
Approved by:	Scott M. Shaeffer, Chief Reactor Projects Branch 6 Division of Reactor Projects

Enclosure 2

SUMMARY OF FINDINGS

IR 05000327/2012-005, 05000328/2012-005; 10/1-12/31/2012; Sequoyah Nuclear Plant, Units 1 and 2; (i.e. Adverse Weather Protection, Fire Protection, Inservice Inspection Activities, Refueling and Outage Activities, and Problem Identification and Resolution.)

The report covered a three-month period of inspection by resident inspectors and announced inspections by regional inspectors. Ten findings were identified, of which one notice of violation (NOV) of NRC requirements, four NCVs of NRC requirements, and one SL IV traditional enforcement violation 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) dated June 2, 2011. Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. Cross-cutting aspects are determined using IMC 0310, “Components Within the Cross-Cutting Areas” dated October 28, 2011. All violations of NRC requirements are dispositioned in accordance with the NRCs Enforcement Policy dated June 7, 2012. 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. <u>NRC-Identified and Self-Revealing Findings</u>

Cornerstone: Mitigating Systems

- Green. A NRC-identified Green non-cited violation (NCV) of Unit 1 and 2 Technical Specification 6.8.1.a for the licensee’s failure to follow station procedures to adequately implement freeze protection requirements. Specifically, inspectors found a number of requirements improperly executed with no specific follow-up of those requirements contained within periodic instructions used to verify program implementation. The licensee placed the issue into the CAP and corrected the identified deficiencies.

  The inspectors determined that the failure to adequately implement all requirements of the licensee’s freeze protection program procedures was a performance deficiency. The inspectors determined that the performance deficiency was more than minor because it was associated with the Mitigating System Cornerstone attribute of Protection against External Factors and adversely affected the cornerstone objective in that specific measures required for freeze protection were not properly implemented and station procedures did not maintain those expected conditions. The inspectors determined the finding was of very low safety significance (Green) as the site had not experienced significant freeze conditions yet this season. The cause of this finding was related to the cross-cutting aspect of ensuring personnel training is adequate to assure nuclear safety [H.2(b)] (Section 1R01)

- Green. The inspectors identified a Green noncited violation of Units 1 & 2 Technical Specification 6.8.1.f for the licensee’s failure to implement procedures required for fire protection program implementation. The inspectors found multiple examples of where fire watches were not conducted in accordance with procedure NPG-SPP-18.4.6, Control of Fire Protection Impairments, Revision 1, when required. The licensee entered this issue into the CAP program as PERs 635934 and 635934.

Failure of the licensee to implement the requirements of procedure NPG-SPP-18.4.6, Control of Fire Protection Impairments, Revision 1, was a performance deficiency. The performance deficiency was determined to be more than minor because it was associated with the protection against external events (fire) attribute of the mitigating systems cornerstone and adversely 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 perform compensatory measures (fire watches), could have potentially compromised the ability to safely shutdown the plant in the event of a fire in any of the fire zones where the fire watches were required. The significance of this finding was evaluated in accordance with the IMC 0609 Attachment 4, Phase 1- Initial Screening and Characterization of Findings, which required further evaluation in accordance with Appendix F, Attachment 01, Part 1, Fire Protection SDP Phase 1 Worksheet. The finding was assigned to the fire prevention and administrative controls category and represented a low degradation level. The inspectors concluded that the finding was of very low safety significance (Green) based on a qualitative screening and the low degradation rating. The finding was determined to have a cross-cutting aspect in the Work Practices component of the Human Performance cross-cutting area [H.4(c)] since the licensee failed to ensure that there was adequate supervisory and management oversight of fire watches. (Section 1R05).

- Green. The inspectors identified a Green noncited violation of Units 1 & 2 Technical Specification 6.8.1.f for the licensee's failure to establish adequate procedures required for fire protection program implementation. Specifically, NPG-SPP-18.4.6, Control of Fire Protection, Revision 1 Impairments was determined to be inadequate because it did not provide any guidance on what a fire watch was supposed to do when they came to a protected door. The licensee entered this issue into the CAP program as PER 652672.

  Failure of the licensee to establish adequate procedures required for fire protection program implementation was a performance deficiency. The performance deficiency was determined to be more than minor because it was associated with the protection against external events (fire) attribute of the mitigating systems cornerstone and adversely 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 establish adequate procedures required for fire protection program implementation caused compensatory measures (fire watches) to not be adequately completed and could have potentially compromised the ability to safely shutdown the plant in the event of a fire in any of the fire zones where the fire watches were required. The significance of this finding was evaluated in accordance with the IMC 0609 Attachment 4, Phase 1- Initial Screening and Characterization of Findings, which required further evaluation in accordance with Appendix F, Attachment 01, Part 1, Fire Protection SDP Phase 1 Worksheet. The finding was assigned to the fire prevention and administrative controls category and represented a low degradation level. The inspectors concluded that the finding was of very low safety significance (Green) based on a qualitative screening and the low degradation rating. The finding was determined to have a cross-cutting aspect in the

Work Practices component of the Human Performance cross-cutting area [H.2(c)] for failure to provide adequate procedures for individuals conducting fire watches. (Section 1R05).

- Green. The inspectors identified a violation with several examples of 10 CFR 50, Appendix B, Criterion III, "Design Control," for failure to implement design control measures that review for suitability of application of materials, parts, and equipment that are essential to the safety-related functions of the structures, systems, and components and that provide for verifying or checking the adequacy of design such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program, including qualification testing of a prototype unit under the most adverse design conditions. The licensee entered this issue into the CAP as PER 668367.

  Failure of the licensee to ensure measures used to review the suitability of application of materials, parts, and equipment essential to the safety-related functions of molded case circuit breakers, and measures to provide for the verification of checking the adequacy of design were in place was a performance deficiency. This performance deficiency was more than minor because it affected the design control attribute of the mitigating systems cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, adequate measures were not implemented to ensure the station 120-VAC vital instrumentation boards had properly maintained their seismic qualification for their application. The inspectors assessed this finding for significance in accordance with NRC Manual Chapter 0609, Appendix A, Exhibit 2, Significance Determination Process (SDP) for Findings At-Power – Mitigating Systems Screening Questions, and determined that it was of very low safety significance (Green) as the devices in question had been intrinsically qualified for this application as part of a complete panel test by the original vendor and the licensee determined that the SSC maintained its operability or functionality despite the identified non-conformances. The inspectors evaluated this finding and violation of NRC requirements in accordance with the NRC Enforcement Policy, Section 2.3.2, and found two conditions to not be met requiring a Notice of Violation be issued. First, inspectors found the licensee failed to restore compliance within a reasonable time after the original violation (05000327.328/2011002-01) was identified. The NRC Enforcement Manual, Section 3.1.2.A.1.b).1), further defines restoring compliance to include those actions taken to stop an ongoing violation from continuing. Second, the inspectors determined that the identified non-conformances represented a repetitive violation as a result of inadequate corrective action and that identification was by the NRC inspector. The lack of rigor in addressing the root of the prior violation which resulted in the inadequate corrective action further led the inspectors to identify a crosscutting aspect in the CAP component of the Problem Identification and Resolution area [P.1(c)]. (Section 4OA2.2)

Cornerstone: Barrier Integrity

- Green. The inspectors identified a Green NCV of 10 CFR Part 50.55a, "Codes and Standards," involving the licensee's failure to properly apply Subsection IWE of ASME Section XI for conducting general visual examinations of the metal-to-metal pipe plugs installed in the containment liner channel weld leak chase test connections that provide a moisture barrier to the containment liner seam welds. Following the inspectors' identification of this issue, the licensee conducted the visual examinations on all eight of the leak chase test connection upper cavities. These visual examinations revealed significant corrosion of the upper cavities, including one through-wall hole in the tubing leading down to the leak chase channels. Upon further inspection of the channels using a boroscope, the licensee noted water in the channels and corresponding corrosion. No through-wall condition was noted in any leak chase channel, and corrosion was limited to a very small percentage of the liner plate thickness. The licensee adequately evaluated the deficiencies prior to entering Mode 4 (Hot Shutdown) to ensure the integrity of containment was maintained. The issue was entered into the licensee's CAP as problem evaluation report (PER) 636215.

  The failure to conduct a general visual examination of 100 percent of the moisture barriers intended to prevent intrusion of moisture against inaccessible areas of the containment liner at metal-to-metal interfaces which are not seal welded, was a performance deficiency that was within the licensee's ability to foresee and correct. This finding was of more than minor significance because the failure to conduct required visual examinations and identify the degraded moisture barriers which allowed the intrusion of water into the liner leak chase channel, if left uncorrected, would have resulted in more significant corrosion degradation of the containment liner or associated liner welds. The finding was associated with the design control attribute of the Barrier Integrity Cornerstone and affected the cornerstone objective of providing reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Specifically, visual examinations of the containment metal liner provide assurance that the liner remains capable of performing its intended safety function. The inspectors used IMC 0609, "Significance Determination Process," Attachment 0609.04, "Phase 1 – Initial Screening and Characterization of Findings," and determined that the finding was of low safety significance (Green) because it did not represent an actual open pathway in the physical integrity of the reactor containment.

  The inspectors identified a cross-cutting aspect in the Operating Experience component of the CAP cross-cutting area (P.2(b)). Specifically, the licensee failed to apply available Operating Experience from four other relevant industry issues to assure plant performance. (Section 1R08)

Cornerstone: Miscellaneous

- Severity Level IV. The NRC identified a Severity Level IV noncited violation of 10 CFR 50.36(c)(5) for failure to submit the Technical Specification (TS) required U1R18 Steam Generator report within 180 days after the initial entry into Mode 4 following completion of an inspection performed in accordance with the Specification 6.8.4.k, “Steam Generator (SG) Program. The licensee entered this into their CAP as PER 648658 and as a corrective action submitted the report on December 17th 2012 to the NRC.

  The inspectors concluded that the failure of the licensee to submit a TS required report was a performance deficiency. The inspectors evaluated this performance deficiency using the traditional enforcement process because the failure to submit a required report affected the NRC’s ability to perform its regulatory function. Consistent with the guidance in Section 2.2.2 and Section 6.9.d of the NRC Enforcement Policy, the inspectors concluded the finding was a Severity Level IV violation because the licensee failed to make a TS required report that resulted in no or relatively inappreciable potential safety consequences. In accordance with section 07.03.c of NRC Inspection Manual Chapter 0612 cross-cutting aspects are not assigned to traditional enforcement violations. (Section 1R20).

B. Licensee-Identified Violations

Violations of very low safety significance or of Severity Level IV that were identified by the licensee have been reviewed by the NRC. Corrective actions taken or planned by the licensee have been entered into the licensee’s CAP. These violations and corrective action tracking numbers are listed in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status:

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

Unit 2 operated at or near 100 percent rated thermal power (RTP) until October 15, 2012, when Unit 2 was shut down for a planned refueling outage. The unit remained shutdown for the rest of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection

a. Inspection Scope

The inspectors reviewed the state of the licensee's preparations for freezing weather following several weeks of sub-freezing evenings, implemented through work order containing the licensee's implementing procedure, M&AI-27, Freeze Protection, Revision 12. This maintenance procedure includes installing freeze protection for safety related systems and components that could be affected by adverse weather. The inspectors performed accompanied walk-downs of completed portions of the procedure, observed operations performance of periodic instructions which are intended to validate the effectiveness of the licensee's freeze protection preparations, observed plant conditions, and performed independent walk-downs of noted discrepancies. Documents reviewed are listed in the Attachment. This activity constituted one inspection sample.

b. Findings

Introduction. The inspectors identified a Green non-cited violation of Units 1&2 Technical Specification 6.8.1.a for the licensee's failure to follow station procedures to adequately implement freeze protection requirements. Specifically, inspectors found a number of requirements improperly executed with no specific follow-up of those requirements contained within periodic instructions used to verify program implementation.

Description. The site's freeze protection program was governed by NPG-SPP-10.14, Freeze Protection, Revision (Rev.) 0. It was implemented through M&AI-27, Freeze Protection, Rev. 12, which itself was implemented through a work order (WO). This WO was still open and coded as a priority 4 (complete as resources allow) activity as of December 29, having commenced in September ahead of potential freezing weather. Through the winter months, execution of the site's freeze protection program was through each shift a completion of periodic instructions. In particular, 1-PI-OPS-000-021.1, Control Room Operator Main Control Room Duty Station Shift Relief and System Status Checklists, Rev. 69 drove the performance of more specific verification periodic instructions (listed under documents reviewed) which validated the adequacy of the site's freeze protection relative to prevailing weather conditions.

The inspectors found that specific procedural steps in the implementing M&AI regarding placement of "hi-lo" thermometers were not satisfied although completion and verification sign-offs were initialed. The procedural acceptance criteria (step 5.7) called for these thermometers to be placed in areas of high temperature concern, such as near motor-operated valves in the vicinity of temporary space heaters, as well as in areas for which low temperatures could affect operation, such as near steam sensing lines in the vicinity of room pressure-relief openings. The inspectors observed no thermometers posted suitable to the described protections. Review of thermometer readings was not observed to be part of any of the more specific verification periodic instructions, which would appear to reinforce the lack of rigor in placement. Additionally, the inspectors found heater placement in accordance with step 6.1.3[2] was not satisfied although completion and verification sign-offs were initialed. The particular heater was specified to point up towards the pressure relief openings previously mentioned in an apparent effort to mix warm air with the incoming cold air. Contributing to its mispositioning, the inspectors observed that the heater was not tied down in accordance with station procedures. Next, the inspectors observed that as Unit 2 had been shut down for nearly the entire quarter for an extended outage, a damper hood in the main steam valve vault room which was to be closed, and verified closed by periodic instruction, had remained open and locked on its hasp for the entirety of the outage. Finally, the inspectors found that temporary equipment tagging requirements specified by implementing procedures were not performed as required. The latest revision to M&AI-27 lists its revision intent as that of clarifying that temporary equipment shall be tagged as required. The revision not referred to a cancelled procedure, though the requirements for the site still exist within governing procedures. Temporary equipment control did not conform to those requirements at the time of inspection.

Analysis. The licensee's failure to adequately implement freeze protection requirements per station procedures was a performance deficiency. The performance deficiency was determined to be more than minor as it was associated with the protection against external factors attribute of the initiating events cornerstone and adversely 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, the failure to implement the above listed requirements could limit the sites ability to detect, respond to, or mitigate the consequences of an accident. The inspectors evaluated the significance of the finding using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process," Attachment 4, "Phase 1 – Initial Screening and Characterization of Findings," and determined the finding to be of very low safety significance (i.e. Green) because the finding did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions will not be available. More specifically, the site had not experienced freezing weather conditions of sufficient magnitude to challenge plant systems. The finding was determined to have a cross-cutting aspect in the Resources component of the Human Performance area since the licensee failed to ensure that personnel training was adequate to reinforce implementation procedure expectations [H.2(b)].

Enforcement. Units 1 and 2 TS 6.8.1.a requires, in part, that written procedures be established, implemented, and maintained covering the activities specified in Appendix A, “Typical Procedures for Pressurized Water Reactors and Boiling Water Reactors,” of Regulatory Guide (RG) 1.33, “Quality Assurance Program Requirements (Operations),” Revision 2, dated February 1978. RG 1.33, Appendix A, Section 6, “Procedures for Combating Emergencies and Other Significant Events,” required procedures for acts of nature, including freezing conditions. The licensee relies upon station procedures and associated implementing work orders to protect against freezing conditions. Contrary to the requirements of the governing procedures, the licensee failed to implement requirements specified by those procedures. Because the finding was of very low safety significance and has been entered into the licensee’s corrective action program as PER 665663, this violation is being treated as an NCV, consistent with the NRC Enforcement Policy: NCV 05000327,328/2012005-01, “Failure to Implement Freeze Protection Program Requirements.”

1R04 Equipment Alignment

.1 Partial System Walkdown

a. Inspection Scope

The inspectors performed partial walkdowns 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 them into the CAP. Documents reviewed are listed in the Attachment. The inspectors completed 2 samples.

- Both trains of Spent Fuel Pool Cooling during Unit 2 Refueling outage
- 1A Containment Spray System while 1B underwent Section XI testing

.2 Complete System Walkdown

a. Inspection Scope

The inspectors performed a complete system walkdown of the Flood Mode Boration 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. The inspectors completed 1 sample.

b. Findings

No findings were identified.

1R05 Fire Protection

.1 Fire Protection Tours

a. Inspection Scope

The inspectors conducted a tour of the six areas important to safety listed below to assess the material condition and operational status of fire protection features. The inspectors evaluated whether: combustibles and ignition sources were controlled in accordance with the licensee's administrative procedures; fire detection and suppression equipment was available for use; passive fire barriers were maintained in good material condition; and compensatory measures for out-of-service, degraded, or inoperable fire protection equipment were implemented in accordance with the licensee's fire plan. Documents reviewed are listed in the Attachment. The inspectors completed 6 samples.

- Control Building Elevation 706 (Cable Spreading Room)
- Control Building Elevation 685 (Auxiliary Instrument Rooms)
- Unit 2 Annulus
- Auxiliary Building Elevation 690 Unit 1 & 2
- Auxiliary Building Elevation 743 (Spent Fuel Pool)
- Auxiliary Building Elevation 669 (ERCW tunnel)

b. Findings

.2 Introduction. The inspectors identified a Green NCV of Units 1 & 2 Technical Specification 6.8.1.f for the licensee's failure to implement procedures required for fire protection program implementation. The inspectors found multiple examples of where fire watches were not conducted in accordance with procedure NPG-SPP-18.4.6, Control of Fire Protection Impairments, Revision 1, when required.

Description. On October 31, 2012, the inspectors performed walkdowns of the Control Building Elevation 706 (Cable Spreading Room) and Elevation 685 (Auxiliary Instrument Rooms). The inspectors reviewed the fire protection impairments (fire watches) for the rooms and interviewed the individuals that were conducting the fire watches. The inspectors noted that the individuals interviewed were maintaining multiple fire watch route sheets (Form NPG-SPP-18.4.6-2) on a clipboard over several shifts. After reviewing NPG-SPP-18.4.6, Control of Fire Protection Impairments, Revision 1 the inspectors identified to the licensee that this was not being done in accordance with their procedure. Appendix A, Section 3.2.C of NPG-SPP-18.4.6, Control of Fire Protection Impairments, states that Compensatory Fire Watches are to complete form NPG-SPP-18.4.6-2 by entering the time and initials as each area is patrolled, and return the completed form to the Fire Protection Foreman/designee at the end of the shift. This issue was entered into the licensee's CAP as PER 635934. The inspectors also reviewed documentation for open fire protection impairments for the month of October. Multiple discrepancies were identified which include: examples exist where the duration of time between documented hourly fire watches exceed 1 hour, examples existed where dates were not filled in correctly or were not filled in at all, examples existed where times were filled in, but there were no corresponding initials, and examples existed where verification signatures were not present. This issue was entered into the licensee's CAP as PER 637101. For each discrepancy identified, the licensee concluded no actual fire did occur based on the subsequent fire watch's documentation. The inspectors determined that the licensee had missed multiple hourly fire watches when required as compensatory actions for impaired fire protection systems.

Analysis. The licensee's failure to implement the requirements of procedure NPG-SPP-18.4.6, Control of Fire Protection Impairments, Revision 1, was a performance deficiency. The performance deficiency was determined to be more than minor because it was associated with the protection against external events (fire) attribute of the mitigating systems cornerstone and adversely 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 perform compensatory measures (fire watches), could have potentially compromised the ability to safely shutdown the plant in the event of a fire in any of the fire zones where the fire watches were required. The significance of this finding was evaluated in accordance with the IMC 0609 Attachment 4, Phase 1- Initial Screening and Characterization of Findings, which required further evaluation in accordance with Appendix F, Attachment 01, Part 1, Fire Protection SDP Phase 1 Worksheet. The finding was assigned to the fire prevention and administrative controls category and represented a low degradation level. The inspectors concluded that the finding was of very low safety significance (Green) based on a qualitative screening and the low degradation rating. The cause of this finding was directly related to the cross-cutting aspect of oversight of work activities in the Work Practices component of the Human Performance area, because the licensee failed to ensure that there was adequate supervisory and management oversight of fire watches [H.4(c)].

Enforcement. Units 1 and 2 TS 6.8.1.f requires, in part, that written procedures be established, implemented, and maintained covering the activities involved with Fire Protection Program implementation. Procedure NPG-SPP-18.4.6, Control of Fire Protection Impairments, Revision 1, is the implementing/controlling process for all Fire Protection impairments. Contrary to the above, during the month of October 2012, the licensee failed to implement written procedures for fire protection program implementation. Specifically, the requirement for an individual assigned to traverse a fire impairment area at specified time intervals with an allowable margin of plus or minus 25 percent was not met. Because the finding was of very low safety significance and has been entered into the licensee's CAP as PERs 635934 and 635934, this violation is being treated as an NCV, consistent with Section 2.3.2 of the Enforcement Policy: NCV 05000327,328/2012005-02, "Failure to Implement Fire Protection Impairment Requirements."

.3 Introduction. The inspectors identified a Green NCV of Units 1 & 2 Technical Specification 6.8.1.f for the licensee's failure to establish adequate procedures required for fire protection program implementation. Specifically, NPG-SPP-18.4.6, Control of Fire Protection, Revision 1 Impairments was determined to be inadequate because it did not provide any guidance on what a fire watch was supposed to do when they came to a protected door.

Description. On November 29, 2012, the inspectors performed a walkdown of the Emergency Diesel Building and observed that Fire Watches were not physically entering spaces as part of the required fire watch rounds (e.g., 1B Emergency Diesel room.) Instead of physically entering the spaces the individuals conducting the fire watch would place their hand on the door to check for heat. This action was due to the fire watch encountering an operations Protected Device on the Fire Watch Route. The inspectors questioned whether this practice is within procedural adherence and also questioned if this was a one-time incident with this Fire Watch at this location. After reviewing the requirements of the NPG-SPP-18.4.6 Control of Fire Protection Impairments, Revision 1, the inspectors identified to the licensee that there were no provisions for this type of incident, thus the procedure was inadequate contrary to TS. This issue was entered into the licensee's CAP as PER 652672.

Analysis. The licensee's failure to establish adequate procedures required for fire protection program implementation was a performance deficiency. The performance deficiency was determined to be more than minor because it was associated with the protection against external events (fire) attribute of the mitigating systems cornerstone and adversely 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 establish adequate procedures required for fire protection program implementation caused compensatory measures (fire watches) to not be adequately completed and could have potentially compromised the ability to safely shutdown the plant in the event of a fire in any of the fire zones where the fire watches were required. The significance of this finding was evaluated in accordance with the IMC 0609 Attachment 4, Phase 1- Initial Screening and Characterization of Findings, which required further evaluation in accordance with Appendix F, Attachment 01, Part 1, Fire Protection SDP Phase 1 Worksheet. The finding was assigned to the fire prevention and administrative controls category and represented a low degradation level. The inspectors concluded that the finding was of very low safety significance (Green) based on a qualitative screening and the low degradation rating. The finding was determined to have a cross-cutting aspect in the Work Practices component of the Human Performance cross-cutting area [H.2(c)] for failure to provide adequate procedures for individuals conducting fire watches.

Enforcement. Units 1 and 2 TS 6.8.1.f requires, in part, that written procedures be established, implemented, and maintained covering the activities involved with Fire Protection Program implementation. Procedure NPG-SPP-18.4.6, Control of Fire Protection Impairments, Revision 1, is the implementing/controlling process for all Fire Protection impairments. Contrary to the above, during the month of November 2012, the licensee failed to establish written procedures for fire protection program implementation. Specifically, NPG-SPP-18.4.6, Control of Fire Protection Impairments was determined to be inadequate because it did not provide any guidance on what a fire watch was supposed to do when they came to a protected door. Because the finding was of very low safety significance and has been entered into the licensee's CAP as PER 652672, this violation is being treated as an NCV, consistent with Section 2.3.2 of the Enforcement Policy: NCV 05000327,328/2012005-03, "Failure to Establish Adequate Procedures for Fire Protection Impairment Requirements."

1R06 Flood Protection Measures

.1 Internal Flooding

a. Inspection Scope

The inspectors reviewed one internal flood protection measures sample for the 6.9kV Shutdown Board Rooms internal flood design to verify that flood mitigation plans were consistent with the design requirements and risk analysis assumptions and that equipment essential for reactor shutdown was properly protected from a flood caused by pipe breaks in the 6.9kV Shutdown Board 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 6.9kV Shutdown Board Rooms 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. The inspectors completed one sample.

b. Findings

No findings were identified.

1R08 Inservice Inspection Activities (71111.08P)

a. Inspection Scope

Non-Destructive Examination (NDE) Activities and Welding Activities: From October 29, 2012, through December 7, 2012, the inspectors conducted an on-site review of the implementation of the licensee's Inservice Inspection (ISI) Program for monitoring degradation of the reactor coolant system (RCS), steam generator (SG) tubes, risk-significant piping and components and containment systems. The inspectors' activities included a review of NDEs to evaluate compliance with the applicable edition of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC), Section XI (Code of record: 2001 Edition through 2003 Addenda), and to verify that indications and defects (if present) were appropriately evaluated and dispositioned in accordance with the requirements of the ASME Code, Section XI, acceptance standards. As it is typically a part of this inspection procedure (IP 71111.08) to perform a review of NDE and welding activities, the NDE and welding associated with the Replacement Steam Generators (RSGs) was reviewed during this inspection. For that reason, the NDE/Welding samples required by Section 02.03.a and 02.03.c of IP 50001 are described here, rather than in section 4OA5.

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

- Radiographic Testing (RT)
  - RT on weld FW-1, Main Steam Elbow-to-Nozzle, 31" ID, ASME Class 2
  - RT on weld FW-1, Feedwater Elbow-to-Nozzle, 16" ID, ASME Class 2

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

- Visual Examination (VE) of the Unit 2 Reactor Pressure Vessel Bottom-Mounted Instrumentation Nozzles, ASME Class 1
- Radiographic Testing (RT)
  - RT on weld FW-1, Steel Containment Vessel (SCV) Liner Plate weld, ASME Class MC
- Ultrasonic Examination (UT)
  - UT on weld #2-RC-2-1, RCS Hot Leg Elbow-to-RSG A Nozzle Safe End, 31" ID, ASME Class 1
  - UT on weld #2-RC-3-1, RCS Crossover Leg Elbow-to-RSG A Nozzle Safe End, 31" ID, ASME Class 1
  - UT on weld #2-RC-10-1, RCS Hot Leg Elbow-to-RSG B Nozzle Safe End, 31" ID, ASME Class 1

- o UT on weld #2-RC-11-1, RCS Crossover Leg Elbow-to-RSG B Nozzle Safe End, 31” ID, ASME Class 1
- o UT on weld #2-RC-18-1, RCS Hot Leg Elbow-to-RSG C Nozzle Safe End, 31” ID, ASME Class 1
- o UT on weld #2-RC-19-1, RCS Crossover Leg Elbow-to-RSG C Nozzle Safe End, 31” ID, ASME Class 1
- o UT on weld #2-RC-26-1, RCS Hot Leg Elbow-to-RSG D Nozzle Safe End, 31” ID, ASME Class 1
- o UT on weld #2-RC-27-1, RCS Crossover Leg Elbow-to-RSG D Nozzle Safe End, 31” ID, ASME Class 1

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

The inspectors observed the following pressure boundary welds completed for risk-significant systems during the Unit 2 refueling outage to evaluate if the licensee applied the preservice non-destructive examinations and acceptance criteria required by the Construction Code. In addition, the inspectors reviewed the welding procedure specification, welder qualifications, welding material certification and supporting weld procedure qualification records, to evaluate if the weld procedures were qualified in accordance with the requirements of Construction Code and the ASME Code Section IX.

- Weld #2-RC-2-1, RCS Hot Leg Elbow-to-RSG A Nozzle Safe End, 31” ID, ASME Class 1
- Weld #2-RC-3-1, RCS Crossover Leg Elbow-to-RSG A Nozzle Safe End, 31” ID, ASME Class 1
- Weld #2-RC-10-1, RCS Hot Leg Elbow-to-RSG B Nozzle Safe End, 31” ID, ASME Class 1
- Weld #2-RC-11-1, RCS Crossover Leg Elbow-to-RSG B Nozzle Safe End, 31” ID, ASME Class 1
- Weld #2-RC-18-1, RCS Hot Leg Elbow-to-RSG C Nozzle Safe End, 31” ID, ASME Class 1
- Weld #2-RC-19-1, RCS Crossover Leg Elbow-to-RSG C Nozzle Safe End, 31” ID, ASME Class 1
- Weld #2-RC-26-1, RCS Hot Leg Elbow-to-RSG D Nozzle Safe End, 31” ID, ASME Class 1
- Weld #2-RC-27-1, RCS Crossover Leg Elbow-to-RSG D Nozzle Safe End, 31” ID, ASME Class 1
- Weld #2-MS-31, MS Pipe-to-Pipe RSG A, 32” OD, ASME Class 2
- Weld #2FD-21C, FW Elbow-to-Pipe RSG D, 16” OD, ASME Class 2

PWR Vessel Upper Head Penetration (VUHP) Inspection Activities: For the Unit 2 vessel head, a bare metal visual examination was not required this outage pursuant to 10 CFR 50.55a. The licensee did not perform any inspections or repairs on the VUHP this outage. Therefore, no NRC review was completed for this inspection procedure attribute.

Boric Acid Corrosion Control (BACC) Inspection Activities: The inspectors reviewed the licensee's BACC program activities to ensure implementation with commitments made in response to NRC Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary," and applicable industry guidance documents. Specifically, the inspectors performed an on-site record review of procedures and the results of the licensee's containment walk-down inspections performed during the current fall refueling outage (2R18). The inspectors also interviewed the BACC program owner, conducted an independent walkdown of containment to evaluate compliance with licensee's BACC program requirements, and verified that degraded or non-conforming conditions, such as boric acid leaks, were properly identified and corrected in accordance with the licensee's BACC and corrective action programs.

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

- PER 458020, Wet Active Boric Acid Contaminated Leak
- PER 516195, Boric Acid Buildup on 0-VLV-078-0545
- PER 558612, Significant Boric Acid Buildup on CCP Seals

The inspectors reviewed the following licensee evaluations of RCS components with boric acid deposits to evaluate if degraded components were documented in the corrective action system. The inspectors also evaluated the corrective actions for any degraded RCS components against the component ASME Code Section XI, and other licensee-committed documents:

- PER 551465, SQN-1-VLV-074-0521, RHR Pump 1B-B Discharge Isolation Valve
- PER 601130, 2-FCV-063-0078, SIS Accumulator Tank 3 Check Valve Isolation Valve
- PER 601133, 2-ISV-063-0804, SIS Test Header Isolation Valve

Steam Generator Tube Inspection Activities: During this refueling outage, all four of the Unit 2 SGs were replaced. These inspection activities are described in Section 4OA5 of this report. As it is typically a part of this inspection procedure (IP 71111.08) to perform a review of SG tube inspection activities, the pre-service eddy current examinations performed on the RSGs were reviewed during this inspection. For that reason, the review of baseline eddy current examination of the new SG tubes required by Section 02.03.a.5 of IP 50001 is described here, rather than in section 4OA5.

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

- Reviewed 5 samples of ET data
- Reviewed the SG tube ET examination scope
- Evaluated if the ET equipment and techniques used by the licensee to acquire data from the SG tubes were qualified or validated to detect the known/expected types of SG tube degradation in accordance with Appendix H, Performance Demonstration for Eddy Current Examination, of EPRI Pressurized Water Reactor Steam Generator Examination Guidelines, Revision 7
- Reviewed the licensee's secondary side SG Foreign Object Search and Removal (FOSAR) activities.
- Reviewed ET personnel qualifications

Identification and Resolution of Problems: The inspectors performed a review of a sample of ISI-related problems which were identified by the licensee and entered into the CAP as PERs. The inspectors reviewed the PERs to confirm that the licensee had appropriately described the scope of the problem, and had initiated corrective actions. The review also included the licensee's consideration and assessment of operating experience events applicable to the plant. The inspectors performed this review to ensure compliance with 10CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requirements. Documents reviewed are listed in the Attachment.

b. Findings

Introduction. The inspectors identified a Green NCV of 10 CFR Part 50.55a, "Codes and Standards," involving the licensee's failure to properly apply Subsection IWE of ASME Section XI for conducting general visual examinations of the metal-to-metal pipe plugs installed in the containment liner channel weld leak chase test connections that provide a moisture barrier to the containment liner seam welds.

Description. While conducting an independent walkdown of the containment to evaluate compliance with licensee's BACC program requirements, the inspectors observed several metal covers that were installed flush with the concrete containment basement floor. Following subsequent inquiries with the licensee on the function of these covers, the inspectors learned that underneath each cover plate was an access (junction) box that housed the test connections for the containment liner channel weld leak chase system. These cover plates were each attached by several small screws, placed around the edges of the plate. The plate was intended to be painted over, consistent with the paint on the rest of the containment floor. There was intended to be a neoprene rubber gasket under the cover, providing a seal between the containment floor and the inspection port cavity. There were eight such cover plates throughout the containment. The containment liner channel weld leak chase system consisted of three-inch wide channel steel that was welded continuously over the entire bottom liner seam welds located under the three-foot thick concrete base mat of the containment. The channels were subdivided into eight zones and in each zone a test connection was installed. These test connections consist of a 3/4-inch carbon steel tube that penetrated through the back of the channel steel and was seal-welded to the channel steel. The opposite end of the tube extended up through the base mat concrete and terminated in the aforementioned junction boxes. A carbon steel threaded pipe cap was installed onto the upper end of the tube. The purpose of the test connections was to perform pressure tests of the inaccessible liner seam welds after the concrete base mat was originally installed during plant construction in order to ensure the leak tight integrity of the liner. The pipe caps were installed following these pressure tests along with the cover plates to the top of the junction boxes. The cover plates served to house and protect the test connections from traffic during and after initial containment construction; however, the combination of the painted cover plates, neoprene rubber gasket, and pipe caps combined to serve as a moisture barrier to prevent the intrusion of water into the leak chase channel weld area.

In response to the inspectors' follow-up questions, the licensee indicated that they had no program in place to inspect any portion of these leak chase test connections for evidence of moisture intrusion that could have reached the containment liner. The inspectors determined that ISI inspection requirements for moisture barriers found in ASME Section XI, Subsection IWE, "Requirements for Class MC and Metallic Liners of Class CC Components of Light-Water Cooled Plants," were applicable to this configuration. Specifically, Table IWE-2500-1, Category E-A, "Containment Surfaces," Item E1.30, "Moisture Barriers," requires a general visual examination of 100 percent of moisture barriers. The reference to moisture barriers is further defined in Note (3) of this table, which states, in part; "Examination shall include moisture barrier materials intended to prevent intrusion of moisture against inaccessible areas of the pressure retaining metal containment shell or liner at concrete-to-metal interfaces and at metal-at-metal interfaces which are not seal welded." Because neither the cover plate nor the tube cap was seal welded, and leakage past these components would allow the intrusion of water to the inaccessible liner seam welds, each represented a moisture barrier and was required to be inspected in accordance with Subsection IWE of ASME Section XI.

As part of their follow-up actions to evaluate the condition identified by the inspector initially, the licensee initiated actions to conduct the required ISI general visual examinations of all eight leak chase test connections. All eight access covers were found intact and properly aligned. Inspection of the junction boxes revealed water had infiltrated into the junction boxes, as evidenced by general corrosion. In one case, the tubing leading down to the leak chase channel had a through-wall hole as a result of corrosion. Water and corresponding corrosion were noted in the leak chase channels. The licensee took action to remove as much of the water as possible from the leak chase channels, and the conditions in all eight junction boxes/leak chase channels were further evaluated to verify that containment integrity had been maintained and would continue to be maintained through the expected life of the plant, including the years associated with a renewed operating license. The licensee intended to seal-weld the cap over the tubing in the junction boxes, and the junction boxes were added to the Augmented ISI plan such that the required inspections would be performed in the future. The inspectors determined that the licensee had taken adequate immediate corrective actions to address the deficiencies identified and to ensure the leak-tight integrity of the containment. The licensee initiated PER 636215 to address the issues associated with this problem and at the end of the inspection period the licensee's causal evaluation had not been completed.

Analysis. The failure to conduct a general visual examination of 100 percent of the moisture barriers intended to prevent intrusion of moisture against inaccessible areas of the containment liner was a performance deficiency that was within the licensee's ability to foresee and correct. The inspectors determined that this finding was of more than minor significance because the failure to conduct required visual examinations and identify the degraded moisture barriers which allowed the intrusion of water into the liner leak chase channel, if left uncorrected, would have resulted in more significant corrosion degradation of the containment liner or associated liner welds. This finding was associated with the Design Control attribute of the Barrier Integrity Cornerstone and affected the cornerstone objective of providing reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Specifically, visual examinations of the containment metal liner provide assurance that the liner remains capable of performing its intended safety function. The inspectors used Inspection Manual Chapter 0609, "Significance Determination Process," Attachment 0609.04, "Phase 1 – Initial Screening and Characterization of Findings," and determined that the finding was of low safety significance (Green) because it did not represent an actual open pathway in the physical integrity of the reactor containment.

The inspectors reviewed this performance deficiency for cross-cutting aspects as required by Manual Chapter 0310, "Components With Cross-Cutting Aspects." The inspectors determined that there was available Operating Experience from four other relevant industry issues which the licensee did not use. Therefore, the finding was assigned a cross-cutting aspect in the Operating Experience component of the CAP cross-cutting area (P.2(b)).

Enforcement. 10 CFR Part 50.55a, "Codes and Standards," as modified by NRC Final Rule-Making published in the Federal Register dated August 8, 1996, and October 1, 2004, states, in part, that the examination of metal liners in concrete containments shall satisfy the requirements of ASME Section XI, Subsection IWE of the 1992 Edition with the 1992 Addenda or the 1998 Edition through the latest edition and addenda incorporated by reference in paragraph 10 CFR 50.55a(b)(2). The 1992 Edition with the 1992 Addenda of ASME Section XI, Subsection IWE; as well as the current 2001 Edition with the 2003 Addenda required examination of moisture barriers in concrete containments. Specifically, Table IWE-2500-1, Category E-A, "Containment Surfaces," Item E1.30, "Moisture Barriers," required a general visual examination of 100 percent of moisture barriers that is further defined in Note (3), which states, in part; "Examination shall include moisture barrier materials intended to prevent intrusion of moisture against inaccessible areas of the pressure retaining metal containment shell or liner at concrete-to-metal interfaces and at metal-at-metal interfaces which are not seal welded."

Contrary to the above, since initial 10 CFR 50.55a, Subsection IWE requirements were established (in 1996) until present, the licensee had failed to perform visual examinations of the non-seal-welded cover plates and non-seal-welded, threaded tube cap at the top of the leak chase channel test connections, and thereby failed to identify defective areas in the moisture barrier, and failed to correct the defects. Because this finding is of very low safety significance and has been entered into the licensee's CAP as PER 636215, this violation is being treated as a non-cited violation, consistent with Section 2.3.2 of the NRC Enforcement Policy: NCV 05000328/2012-005-04, Failure to Perform ISI General Visual Examinations of Containment Moisture Barrier Associated with Containment Liner Leak Chase Test Connection Threaded Pipe Plugs.

1R11 Licensed Operator Requalification Program and Licensed Operator Performance

.1 Quarterly Review

a. Inspection Scope

The inspectors performed one licensed operator requalification program review. The inspectors observed a simulator session on October 11, 2012. The training scenario involved Just-In-Time Training for Pre-Refueling Outage risk significant activities such as placing the RHR system in service. The inspectors observed crew performance in terms of: communications; ability to take timely and proper actions; prioritizing, interpreting and verifying alarms; correct use and implementation of procedures, including the alarm response procedures; timely control board operation and manipulation, including high risk operator actions; oversight and direction provided by shift manager, including the ability to identify and implement appropriate Technical Specification (TS) action; and, group dynamics involved in crew performance. The inspectors also observed the evaluators' critique and reviewed simulator fidelity to verify that it matched actual plant response. Documents reviewed are listed in the Attachment. This activity constituted one inspection sample.

b. Findings

No findings were identified

.2 Quarterly Review of Licensed Operator Performance

a. Inspection Scope

The inspectors observed and assessed licensed operator performance in the main control room during periods of heightened activity or risk. The inspectors reviewed various licensee policies and procedures such as OPDP-1, Conduct of Operations, NPG-SPP-10.0, Plant Operations, and 0-GO-5, Normal Power Operation. The inspectors utilized activities such as post-maintenance testing, surveillance testing, unplanned transients, infrequent plant evolutions, plant startups and shutdowns, reactor power and turbine load changes, and refueling and other outage activities to focus on the following conduct of operations as appropriate:

- Operator compliance and use of procedures
- Control board manipulations
- Communication between crew members

- Use and interpretation of plant instruments, indications and alarms
- Use of human error prevention techniques
- Documentation of activities, including initials and sign-offs in procedures
- Supervision of activities, including risk and reactivity management
- Pre-job briefs

Specifically, the inspectors observed licensed operator performance during the following activities:

- Unit 2 shutdown & placing RHR inservice for shutdown cooling
- Unit 2 refueling and other outage activities, including midloop operations
- Unit 2 mode changes coming out of the refueling outage

Documents reviewed are listed in the Attachment. This activity constituted one inspection sample.

b. Findings

No findings were identified

.3 Licensed Operator Requalification

a. Inspection Scope

Annual Review of Licensee Requalification Examination Results: On September 27, 2012, the licensee completed the comprehensive biennial requalification written examinations and the annual requalification operating examinations 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 the individual operating examinations and the crew simulator operating examinations in accordance with Inspection Procedure (IP) 71111.11, “Licensed Operator Requalification Program.” These results were compared to the thresholds established in Inspection Manual Chapter (IMC) 0609, “Significance Determination Process,” 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 five samples.

- Reviewed the technical basis for revision of the hydrogen recombiners a(1) plan to add all three remaining Unit 1 and Unit 2 trains of hydrogen recombiners to the current existing Unit 1 Train A recombiner a(1) plan
- Reviewed a(1) Plan, Revision 0, for 6.9 kV Shutdown Board (SDBD) Room Chillers (Train A and B)
- Repeat Failure of 2-MVOP-74-033 Spiral Ring to Maintain Position of the Hand wheel Clutch Gear
- System 30B, Containment Vacuum Relief CDE 2648, zone switch
- System 30K, SDBR A chiller CDE 2654, thermal expansion valve

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 8, and Instruction 0-TI-DSM-000-007.1, “Risk Assessment Guidelines,” Revision 9. Documents reviewed are listed in the Attachment. The inspectors completed two samples.

- Unit 2 Refueling Outage risk review
- Infrequently Performed High Risk Review during 2-SI-OPS-082-026.B

b. Findings

No findings were identified.

1R15 Operability Determinations and Functionality Assessments

a. Inspection Scope

For the eight 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.

- PER 523862 – Emergency Diesel Generator 1B-B air starting motor air line lubricator (one of four total) did not provide lubricant as expected to two of the starting motors (8 total)
- PER 486952 – Emergency Diesel Generator 2B-B air starting motor air line lubricator (one of four total) did not provide lubricant as expected to two of the starting motors (8 total)
- PER 549340 – Vital Battery Charger III Preventive Maintenance identified output voltage change between no-load and full-load condition was greater than expected
- PER 366228-A-A Fire/Flood Mode Pump long term degrading insulation
- PER 653651, HH-52B has 8" of water
- Operating Experience Smart Sample (OpESS) 2012/02, Technical Specification Interpretation and Operability Determination, Revision 1
- PER 655601, Found One Bad Hydrogen Igniter
- PER 659558, Through wall hole found in Channel Test Connection Box in Raceway
- PERs 604850 and 610621, Penetration Sealant Outside Temperature Acceptance Range

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 activities 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 113945604, Unit 2 Annulus Door, Dome Integrity
- WOs 112619112, 113387593, U2C18 ECCS maintenance and testing
- WO 114099131, 2B-B MDAFW Comprehensive Test
- 2-STI-088-156.0, Primary Containment Vessel Post-Modification Pressure/New Weld Leakage Inspection Test, Revision 0

b. Findings

No findings were identified.

1R20 Refueling and Outage Activities

.1 Unit 2 Refueling Outage Cycle 18

a. Inspection Scope

For the Unit 2 refueling outage that began on October 15, the inspectors evaluated licensee activities to verify that the licensee considered risk in developing outage schedules, followed risk reduction methods developed to control plant configuration, developed mitigation strategies for the loss of key safety functions, and adhered to operating license and TS requirements that ensure defense-in-depth. The inspectors also walked down portions of Unit 2 not normally accessible during at-power operations to verify that safety-related and risk-significant SSCs were maintained in an operable condition. Specifically, between October 15 and December 31, the inspectors performed inspections and reviews of the following outage activities. Documents reviewed are listed in the Attachment. This inspection satisfied one inspection sample for Refueling Activities.

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

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

- Licensee Control of Outage Activities. On a daily basis, the inspectors attended the licensee outage turnover meeting, reviewed PERs, and reviewed the defense-in-depth status sheets to verify that status control was commensurate with the outage safety plan and in compliance with the applicable TS when taking equipment out of service. The inspectors further toured the main control room and areas of the plant daily to ensure that the following key safety functions were maintained in accordance with the outage safety plan and TS: electrical power, decay heat removal, spent fuel cooling, inventory control, reactivity control, and containment closure. The inspectors also observed a tagout of 2-MVOP-074-0012-A, the A train RHR mini-flow valve for “as found” MOVATS testing to verify that the equipment was appropriately configured to safely support the work and testing. To ensure that RCS level instrumentation was properly installed and configured to give accurate information, the inspectors reviewed the installation of the Mansell level monitoring system. Specifically, the inspectors discussed the system with engineering, walked it down to verify that it was installed in accordance with procedures and adequately protected from inadvertent damage, verified that Mansell indication properly overlapped with pressurizer level instruments during pressurizer draindown, verified that operators properly set level alarms to procedurally required setpoints, and verified that the system consistently tracked RCS level while lowering to reduced inventory conditions. The inspectors also observed operators compare the Mansell indications with locally-installed ultrasonic level indicators during entry into mid-loop conditions.

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

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

- Heatup and Startup Activities. The inspectors toured the containment prior to reactor startup to verify that debris that could affect the performance of the containment sump had not been left in the containment. The inspectors reviewed the licensee's mode-change checklists to verify that appropriate prerequisites were met prior to changing TS modes. To verify RCS integrity and containment integrity, the inspectors further reviewed the licensee's RCS leakage calculations and containment isolation valve lineups.

b. Findings

Introduction. The NRC identified a Severity Level IV noncited violation of 10 CFR 50.36(c)(5) for failure to submit the Technical Specification (TS) required U1R18 Steam Generator report within 180 days after the initial entry into Mode 4.

Discussion. Technical Specification 6.9.1.16, "Steam Generator Tube Inspection Report" required that a report shall be submitted within 180 days after the initial entry into Mode 4 following completion of an inspection performed in accordance with the Specification 6.8.4.k, "Steam Generator (SG) Program". Sequoyah Unit 1 entered Mode 4 on March 28, 2012 at 1236 after the spring refueling outage. During this outage the licensee conducted inspections in accordance with TS 6.8.4.k, "Steam Generator (SG) Program. The U1R18 180 day report was due to the NRC on September 24th. On November 27, 2012, the NRC notified the licensee that they had failed to submit the required report. The licensee entered this into their CAP as PER 648658 and as a corrective action submitted the report on December 17, 2012, to the NRC.

Analysis. The inspectors concluded that the failure of the licensee to submit a TS required report was a performance deficiency. The inspectors evaluated this performance deficiency using the traditional enforcement process because the failure to submit a required report affected the NRC's ability to perform its regulatory function. Consistent with the guidance in Section 2.2.2 and Section 6.9.d of the NRC Enforcement Policy, the inspectors concluded the finding was a Severity Level IV violation because the licensee failed to make a Technical Specification required report that resulted in no or relatively inappreciable potential safety consequences. In accordance with section 07.03.c of NRC Inspection Manual Chapter 0612 cross-cutting aspects are not assigned to traditional enforcement violations.

Enforcement. 10 CFR 50.36(c)(5) requires, in part, that each licensee shall submit any reports to the Commission pursuant to approved technical specifications as specified in 10 CFR 50.4. Sequoyah Unit 1 Technical Specification 6.9.1.16, "Steam Generator Tube Inspection Report" requires that a report shall be submitted within 180 days after the initial entry into Mode 4 following completion of an inspection performed in accordance with the Specification 6.8.4.k, "Steam Generator (SG) Program. Contrary to the above, the licensee failed to submit the required U1R18 Steam Generator report within 180 days after the initial entry into Mode 4. This is a Severity Level IV noncited violation consistent with Section 2.2.2 and Section 6.9.d of the NRC Enforcement Policy. Because this finding is of very low safety significance and has been entered into the

CAP as PER 648658, this violation is being treated as a noncited violation consistent with Section 2.3.2 of the NRC Enforcement Policy: NCV 05000327/2012005-06, "Failure to Submit a Technical Specifications Required Report."

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.

Routine Surveillance Tests:
- 2-SI-OPS-088-001.0, Phase A Isolation Test, Revision 20
- 2-SI-OPS-082-026.B, Loss of Offsite Power with Safety Injection D/G 2B-B Test, Revision 40
- 0-SI-SXV-068-201.0, Pressurizer PORV Operability Test, Rev. 1
- 0-SI-SXV-074-203.2, Full Stroking of RHR Valves FCV-74-1 and FCV-74-2, Rev. 1

Ice Condenser Surveillance Test:
- 0-SI-MIN-061-105.0, Ice Condenser – Ice Weighing, Revision 10 (As Left)

Containment Isolation Valve (CIV) Surveillance Tests:
- 0-SI-SLT-030-258.1, Containment Isolation Valve Local Leak Rate Test Purge Air, Revision 8
- 0-SI-SLT-072-258.1, Containment Spray and RHR Spray Header Valve Water Inventory Test, Revision 10

b. Findings

No findings were identified.

Cornerstone: Emergency Preparedness

1EP4 Emergency Action Level and Emergency Plan Changes

a. Inspection Scope

The NSIR headquarters staff performed an in-office review of the latest revisions of various Emergency Plan Implementing Procedures (EPIPs) and the Emergency Plan located under ADAMS accession numbers ML121910329, ML12199A022 ML12270A362, ML12296A649, and ML12307A285, as listed in the Attachment.

The licensee determined that in accordance with 10 CFR 50.54(q), the changes made in the revisions resulted in no reduction 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 NRC review was not documented in a safety evaluation report and did not constitute approval of licensee-generated changes; therefore, these revisions are subject to future inspection. Documents reviewed are listed in the Attachment. This inspection activity satisfied one inspection sample for the emergency action level and emergency plan changes on an annual basis.

b. Findings

No findings were identified.

2. RADIATION SAFETY (RS)

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

2RS1 Radiological Hazard Assessment and Exposure Controls (71124.01)

a. Inspection Scope

Hazard Assessment and Instructions to workers: During facility tours, the inspectors directly observed labeled radioactive material and postings for radiation areas and High Radiation Areas (HRAs) established within the Radiologically Controlled Area (RCA) of the U-2 containment, Auxiliary Building and Dry Active Waste Storage Facility. The inspector directly observed conduct of licensee radiation surveys for selected RCA areas. The inspectors reviewed and verified survey records for several plant areas including surveys for alpha emitters, airborne radioactivity, and gamma surveys with a range of dose rate gradients. The inspectors reviewed several radiation work permit (RWP) details to assess communication of radiological control requirements and current radiological conditions to workers. The inspectors reviewed selected Electronic Dosimeter (ED) dose and dose rate alarms, to verify workers properly responded to the alarms and that the licensee's review of the events was appropriate. The inspectors observed jobs in radiologically risk-significant areas including high radiation areas (HRAs) and areas with, or with the potential for airborne activity.

Contamination and Radioactive Material Control: The inspectors observed the release of potentially contaminated items from the RCA and from contaminated areas (i.e. U2 containment). The inspectors also reviewed the procedural requirements for, and equipment used to perform, the radiation surveys for release. During plant walk downs, the inspectors evaluated radioactive material storage areas and containers, including satellite RCAs and yard areas, assessing material condition, posting/labeling, and control of materials/areas. In addition, the inspectors reviewed the sealed source inventory and verified labeling, storage conditions, and leak testing of selected sources.

Radiological Hazards Control and Work Coverage: The inspectors evaluated licensee performance in controlling worker access to radiologically significant areas and monitoring jobs in-progress during the week of the onsite inspection. The inspectors also reviewed the procedural guidance for multi- and extremity badging. For HRA tasks involving significant dose rate gradients, the inspectors evaluated the use and placement of whole body and extremity dosimetry to monitor worker exposure. The inspectors reviewed RWPs for use in airborne areas, ensuring the prescribed controls were appropriate for the conditions as identified in radiological surveys and air samples. ED alarm set points and worker stay times were evaluated against area radiation survey results for containment and auxiliary building activities.

Risk Significant High Radiation Areas and Very High Radiation Area Controls: The inspectors evaluated access barrier effectiveness for selected Locked High Radiation Area (LHRA) and Very High Radiation Area (VHRA) locations. Changes to procedural guidance for LHRA and VHRA controls were discussed with RP supervisors.

During plant walk downs of the Unit 2 Containment and Auxiliary Building, the inspectors verified the posting/locking of LHRA/VHRA areas. Established radiological controls (including airborne controls) were evaluated for selected tasks including work in auxiliary building HRAs, and radwaste processing and storage. In addition, licensee controls for areas where dose rates could change significantly as a result of plant shutdown and refueling operations were reviewed and discussed.

Radiation Worker Performance and Radiation Protection Technician Proficiency: The inspectors observed radiation worker performance through direct observation. Jobs observed included routine waste packaging activities in the auxiliary building and routine survey activities in the Auxiliary Building in high radiation and contaminated areas. The inspectors also observed HPTs providing pre-job/RWP briefings, releasing material from the RCA, and providing field coverage of jobs. Occupational workers' adherence to selected RWPs and RP technician proficiency in providing job coverage were evaluated through direct observations and interviews with licensee staff. Electronic dosimeter (ED) alarm set points and worker stay times were evaluated against area radiation survey results for reviewed RWPs.

Problem Identification and Resolution: PERs associated with radiological hazard assessment and control were reviewed and assessed. The inspectors evaluated the licensee's ability to identify, characterize, prioritize, and resolve the identified issues in accordance with procedure NPG-SPP-03.1, Corrective Action Program, (CAP) Rev. 4. The inspectors also evaluated the scope of the licensee's internal audit program and reviewed recent assessment results.

RP activities were evaluated against the requirements of UFSAR Section 12; TS Sections 6.12; 10 CFR Parts 19 and 20; and approved licensee procedures. Licensee programs for monitoring materials and personnel released from the RCA were evaluated against 10 CFR Part 20 and IE Circular 81-07, Control of Radioactively Contaminated Material. Documents reviewed are listed in the Attachment.

The inspectors completed all specified line-items detailed in Inspection Procedure (IP) 71124.01 (sample size of 1).

b. Findings

A licensee identified violation is documented in Section 4OA7

2RS2 Occupational ALARA Planning and Controls

a. Inspection Scope

Work Planning and Exposure Tracking: The inspectors reviewed work activities and their collective exposure estimates associated with the previous U1R18 refueling outage, and the current U2R18 Steam Generator Replacement (SGR)/refueling outage. ALARA planning packages (ALARA Plans) were reviewed for the following high collective exposure tasks: Refueling operations (U1R18 and U2R18); Plant Services (Shielding, Upper & Lower Decon) (U1R18); S/G Primary and Secondary (U1R18); Scaffold/Insulation (U1R18 and U2R18); MOVATS/Appendix R work (U2R18); RP Surveillance (U2R18); SGR Scaffolding (U2R18); SGR RCS System/pipe End Decon (U2R18); and SGR Structural Interferences, Pipe Supports, and Replace SG (U2R18). For the selected tasks, the inspectors reviewed the assumptions and basis for the dose rate and man-hour estimates. The inspectors discussed with ALARA staff the means by which wrench-hours were derived from the work order hours provided by craft supervision to ALARA staff. The inspectors verified the licensee had established several means to track and trend doses for ongoing work activities. The inspectors evaluated the incorporation of exposure reduction initiatives and operating experience, including historical post-job reviews, into RWP requirements. Collective dose data for selected tasks were compared with established dose estimates and evaluated against procedural criteria (trigger points) for additional ALARA review. Where applicable, changes to established estimates were discussed with ALARA planners and evaluated against work scope changes or unanticipated elevated dose rate. For ALARA Plans from U1C18, the inspectors compared the results achieved in terms of actual dose vs. planned dose and actual hours vs. estimated hours, reviewed in-progress and post-job ALARA reviews, and discussed the job planning, performance, and reviews with ALARA staff. For ALARA Plans associated with U2C18, the inspectors reviewed dose-to-date on select jobs, comparing estimates with actuals, and observed development of selected in-progress reviews.

Source Term Reduction and Control: The inspectors reviewed the collective exposure three-year rolling average (TYRA) from 2009 - 2011 and reviewed historical outage collective exposure trends. Through interviews with licensee staff and document review, the inspectors assessed the licensee’s current activities and future plans related to source term reduction, including elevated zinc injection on U2 after Steam Generator Replacement, shutdown chemistry, planned ultrasonic fuel cleaning, and response to fuel defects during previous operating cycles.

Radiation Worker Performance: Radiation worker performance was also observed and evaluated as part of IP 71124.01 and is documented in section 2RS1. While observing job tasks, the inspectors evaluated the use of remote technologies to reduce dose including teledosimetry and remote visual monitoring. Jobs observed were associated with the refueling and Steam Generator Replacement outage.

Problem Identification & Resolution: Licensee CAP documents associated with ALARA planning and controls were reviewed and assessed. This included review of selected Action Requests (ARs), self-assessments, and audits. The inspectors evaluated the licensee’s ability to identify, characterize, prioritize, and resolve the identified issues in accordance with procedure NPG-SPP-03.1, Corrective Action Program, Rev. 0004. The inspectors also evaluated the scope and frequency of the licensee’s self-assessment program and reviewed recent assessment results.

ALARA program activities were evaluated against the requirements of FSAR Section 12, Radiation Protection; TS Section 6.8, Procedures and Programs; 10 CFR Part 20; and approved licensee procedures. Records reviewed are listed in Sections 2RS1 and 2RS2 of the report Attachment.

The inspectors completed all specified line-items detailed in IP 71124.02 (sample size of 1).

b. Findings

No findings were identified.

2RS3 In-Plant Airborne Radioactivity Control and Mitigation

a. Inspection Scope

Engineering Controls: The inspectors reviewed the use of temporary and permanent engineering controls to mitigate airborne radioactivity during Unit 2 Refueling Outage 18 including the Auxiliary Building Gas Treatment System (ABGTS) and controls used for the steam generator replacement. In addition, during observations of jobs in-progress and containment walk-downs, inspectors observed the placement and use of HEPA negative pressure units, and air sampling equipment.

Use of Respiratory Protection Devices & Self-Contained Breathing Apparatus for Emergency Use: Inspectors reviewed the use of respiratory protection devices to limit the intake of radioactive material, including devices used for routine tasks and devices stored for use in emergency situations. Inspectors observed the physical condition of Self-Contained Breathing Apparatus (SCBA) units, negative pressure respirators (NPR)s, powered air purifying respirators (PAPRs) and device components staged for routine and emergency use throughout the plant. SCBA bottle air pressure, the number of units, and the number of spare masks and air bottles available was also evaluated by inspectors. The inspectors reviewed maintenance records for selected SCBA units for the past year and evaluated SCBA and NPR compliance with National Institute for Occupational Safety and Health certification requirements. The inspectors also reviewed records of Grade D (or better) air quality testing for supplied-air devices and SCBA bottles. In addition, the inspectors walked-down the compressor used for filling SCBA bottles. The inspectors reviewed the status and surveillance records of SCBAs staged for in-plant use during emergencies through review of records and walk-down of SCBA staged in the control room and selected locations.

The inspectors verified the licensee had procedures in place to ensure that the use of respiratory protection devices was ALARA when engineering controls were not practicable. Control room operators and fire brigade were interviewed on the use of the devices including SCBA bottle change-out and use of corrective lens inserts. Respirator qualification records were reviewed and cross checked against for several Main Control Room operators. In addition, qualifications for individuals responsible for testing and repairing SCBA vital components were evaluated through review of training records. Selected maintenance records for SCBA units and air cylinder hydrostatic testing documentation were reviewed.

The inspectors verified that the licensee has procedural requirements in place for evaluating air samples for the presence of alpha emitters and reviewed airborne radioactivity and contamination survey records for selected plant areas to ensure air samples are screened and evaluated per the procedure requirements.

The inspectors walked-down the respirator issue and storage locations and verified that the equipment was appropriately stored and maintained. Records of monthly and quarterly inventory and inspection of the equipment were also reviewed by the inspectors. The inspectors discussed the process for issuing respirators, and verified that selected individuals qualified for respirator and/or self-contained breathing apparatus (SCBA) use had completed the required training, fit-test, and medical evaluation.

Problem Identification and Resolution: Licensee CAP documents associated with the control and mitigation of in-plant radioactivity were reviewed and assessed. This included review of selected ARs related to use of respiratory protection devices including SCBA. The inspectors evaluated the licensee's ability to identify, characterize, prioritize, and resolve the identified issues in accordance with procedure NPG-SPP-03.1, Corrective Action Program, Rev. 04. The inspectors also evaluated the scope of the licensee's internal audit program and reviewed recent assessment results. Documents reviewed are listed in the Attachment.

Radiation protection activities were evaluated against the requirements UFSAR Section 12; 10 CFR Parts 19 and 20; and approved licensee procedures. Documents reviewed are listed in the Attachment.

The inspectors completed all specified line-items detailed in IP 71124.03 (sample size of 1).

b. Findings

No findings were identified.

2RS4 Occupational Dose Assessment

a. Inspection Scope

External Dosimetry: The inspectors reviewed National Voluntary Laboratory Accreditation Program (NVLAP) certification data and discussed program guidance for storage, processing, and evaluation of results for active and passive personnel dosimeters currently in use. Comparisons between ED and TLD data were discussed in detail. The inspectors reviewed ED alarm logs and reviewed licensee's dosimeter incident reports and assessment actions for selected alarm events.

Internal Dosimetry: Program guidance and assessment results for internally deposited radionuclides were reviewed. The inspectors reviewed selected Whole Body Count (in vivo) analyses from January 2012 to September 2012. Capabilities for collection and analysis of special bioassay samples were discussed with licensee staff, there were no dose assessments based on biological samples to review.

Special Dosimetric Situations: The inspectors evaluated the licensee's use of multi-badging, extremity dosimetry, and dosimeter relocation within non-uniform dose rate fields and reviewed assessments performed the previous Unit 1 outage for Steam Generator jumpers. Worker monitoring in neutron areas was discussed with licensee staff. The inspectors also reviewed records of monitoring for declared pregnant workers from September 2011 to September 2012 and discussed monitoring guidance with dosimetry staff. In addition, shallow dose assessments for selected Personnel Contamination Events occurring between September 2011 and September 2012 were reviewed and discussed.

Problem Identification and Resolution: The inspectors reviewed and discussed selected CAP documents associated with occupational dose assessment. The inspectors evaluated the licensee's ability to identify and resolve the issues in accordance with procedure NPG-SPP-03.1, "Corrective Action Program", Rev. 04. The inspectors also discussed the scope of the licensee's internal audit program and reviewed recent assessment results.

Occupational dose assessment activities were evaluated against the requirements of UFSAR Section 12; TS Section 6; 10 CFR Parts 19 and 20; and approved licensee procedures. Records reviewed are listed in the Attachment.

The inspectors completed 1 sample as required by IP 71124.04.

b. Findings

No findings were identified.

2RS5 Radiation Monitoring Instrumentation

a. Inspection Scope

Radiation Monitoring Instrumentation: During walk-downs of the auxiliary building and the RCA exit point, the inspectors observed installed radiation detection equipment. These included area radiation monitors (ARMs), liquid and gaseous effluent monitors, personnel contamination monitors (PCMs), small article monitors (SAMs), and portal monitors (PMs). The inspectors observed the physical location of the components and noted their material condition.

In addition to equipment walk-downs, the inspectors reviewed source checks of various portable and fixed detection instruments, including ion chambers, teletectors, PCMs, SAMs, PMs, and a whole body counter (WBC). The inspectors reviewed calibration records and evaluated alarm setpoint values for PCMs, PMs, effluent monitors, an ARM, a SAM, and a WBC. This included a sampling of instruments used for post-accident monitoring such as a containment high-range radiation monitor and effluent monitors for noble gas and iodine. The radioactive source used to calibrate an effluent monitor was evaluated for traceability to national standards. Calibration stickers on portable survey instruments were noted during inspection of the storage area for ready-to-use equipment. The most recent 10 CFR Part 61 analysis for dry active waste (DAW) was reviewed to determine if calibration and check sources are representative of the plant source term. The inspectors also reviewed countroom calibration records for a gamma spectroscopy germanium detector and a liquid scintillation detector.

Effectiveness and reliability of selected radiation detection instruments were reviewed against details documented in the following: 10 CFR Part 20; NUREG-0737, Clarification of TMI Action Plan Requirements; UFSAR Chapters 11 and 12; and applicable licensee procedures. Documents are listed in the Attachment.

Problem Identification and Resolution: The inspectors reviewed selected PER reports in the area of radiological instrumentation. The inspectors evaluated the licensee’s ability to identify and resolve the issues in accordance with procedure NPG-SPP-03.1, Corrective Action Program, Rev. 4. Documents reviewed are listed in the Attachment.

The inspectors completed one (1) sample as required by IP 71124.05.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, Emergency Preparedness, Public Radiation Safety, Occupational Radiation Safety, and Security

4OA1 Performance Indicator (PI) Verification

a. Inspection Scope

Occupational Radiation Safety Cornerstone: The inspectors reviewed Performance Indicator (PI) data collected from September 16, 2012, through October 1, 2012, for the Occupational Exposure Control Effectiveness PI. For the reviewed period, the inspectors assessed PER records to determine whether HRA, VHRA or unplanned exposures, resulting in TS or 10 CFR 20 non-conformances, had occurred during the review period. The inspectors reviewed radiologically controlled area exit transactions with exposures greater than 100 mrem to determine if they were consistent with the requirements of the RWP. The reviewed data were assessed against guidance contained in Nuclear Energy Institute (NEI) 99-02, "Regulatory Assessment Indicator Guideline," Rev. 6. Documents reviewed are listed in the Attachment.

Public Radiation Safety Cornerstone: The inspectors reviewed the Radiological Control Effluent Release Occurrences PI results for the Public Radiation Safety Cornerstone from September 16, 2011, through October 1, 2012. For the assessment period, the inspectors reviewed cumulative and projected doses to the public and PER documents related to Radiological Effluent Technical Specifications/Offsite Dose Calculation Manual issues. Documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

4OA2 Problem Identification and Resolution

.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: Heinemann Breakers

a. Inspection Scope

On May 12, 2011, the NRC issued a NCV to Sequoyah Nuclear Plant for failure to adequately qualify molded-case circuit breakers to safety-related application through commercial grade dedication. Specifically, the licensee was cited for failure to assure that appropriate quality standards were specified and included in design documents and that deviations from such standards were controlled so as to ensure that the molded case circuit breakers utilized in the station's 120VAC vital instrument power boards were properly seismically qualified for their application. As the licensee was to correct breaker alignment issues for Unit 2 during U2R18 outage as specified in TVA's July 17, 2012, letter (ML12202A010), the inspectors reviewed licensee actions associated with the PERs referenced in the NCV to determine if the licensee had adequately addressed all relevant issues. Inspectors interviewed maintenance, engineering and operations personnel, reviewed documents, performed work observations and field walkdowns. Documents reviewed are listed in the Attachment.

Background: By letter dated April 24, 2009, (ML091140140), the U.S. Nuclear Regulatory Commission (NRC issued Inspection Report number 05000390/2009002 concerning the issuance of an Unresolved Item (URI 05000390/2009002-03: Acceptability of Seismic Qualification for 120VAC Vital Instrumentation Board Circuit Breakers) which noted inspector concerns with the qualification seismic response spectrum, inconsistencies in breaker mounting within the 120VAC vital instrumentation boards relative to the original documented seismic testing breaker mounting, and with the qualification test mounting. Somewhat later, by letter dated August 5, 2010, (ML102170465), the NRC issued inspection report number 05000391/2010603 to the Watts Bar Unit 2 Completion Project in which a Notice of Violation was cited for the licensee's failure to adequately evaluate and qualify molded case circuit breakers.

Tennessee Valley Authority denied the violation in a letter dated September 7, 2010, (ML102520435) and provided additional information in a letter dated October 15, 2010, (ML102880493). By letter dated October 19, 2010, (ML102920665), the NRC informed TVA that after review and consideration of TVA's response, the NRC had concluded that the violation had occurred as stated in NOV 05000391/2010603-08, and that TVA was required to provide an additional response. In a letter dated November 24, 2010, (ML103300217), TVA admitted that the violation occurred and described associated corrective actions to achieve compliance. By letter dated January 28, 2011, (ML110280456), the NRC informed TVA that a green non-cited violation of 10 CFR 50, Appendix B, Criterion III, Design Control, at Watts Bar Unit 1 was identified for the failure to assure that appropriate quality standards were specified and included in design documents and that deviations from such standards were controlled. Specifically, TVA failed to ensure that the Heinemann circuit breakers utilized in the station 120VAC vital instrument power boards were properly seismically qualified for their application.

By letter dated March 4, 2011, (ML102920665) the NRC informed TVA that the corrective actions prescribed in their letter of November 24 did not appear adequate to achieve full compliance and requested an amended response to the violation describing corrective actions that are sufficient to achieve full compliance. By letter dated May 12, 2011, (ML111320467) the NRC informed TVA that a green non-cited violation of 10 CFR 50, Appendix B, Criterion III, Design Control, at Sequoyah Nuclear Plant was identified for failure to assure that appropriate quality standards were specified and included in design documents and that deviations from such standards were controlled. On June 9, 2011, (ML11164A141), TVA provided a revised response to their letter of November 24 as requested by the NRC's letter of March 4.

The NRC entered into a Task Interface Agreement between Region II and the Headquarters offices for the balance of 2011 and early 2012, working through the licensing basis requirements associated with the Heinemann circuit breaker mounting and testing. By letter dated May 18, 2012, (ML12139A052) the NRC informed TVA that the corrective actions for the violation described in their June 9, 2011, revision to the November 24, 2010, letter were inadequate to achieve full compliance and that TVA is required to submit a supplemental response to the subject NOV within 30 days (i.e. June 18, 2012). By letter dated June 18, 2012, (ML12172A412) TVA requested an extension for the required submittal of the supplemental response to the NOV until July 20, 2012. By letter dated June 22, 2012, (ML12174A139) the NRC agreed to extend the date for the TVA response an additional 30 days from the original due date to July 17, 2012.

By letter dated July 17, 2012, (ML12202A010) TVA provided its revised response in reply to NOV 05000391/2010603-08, Failure to Adequately Evaluate and Qualify Molded Case Circuit Breakers, which included reasons for the violation, corrective steps taken and results achieved, short-term operability and long-term corrective actions at Watts Bar and at Sequoyah.

b. Findings and Observations

Introduction. On October 23, 2012, the inspectors identified a violation with several examples of 10 CFR 50, Appendix B, Criterion III, "Design Control," for failure to implement design control measures that review for suitability of application of materials, parts, and equipment that are essential to the safety-related functions of the structures, systems, and components and that provide for verifying or checking the adequacy of design such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program, including qualification testing of a prototype unit under the most adverse design conditions.

Description. Inspectors made a field observation on the evening of October 23, 2012, of the licensee's actions under work order 113707839 to correct fit-up issues with breakers 22, 23, and 24 in vital instrumentation power board 2-II as part of the necessary actions for PER 286156, referenced in Sequoyah Nuclear Station's NCV 05000327,328/ 2011002-01. Work order 111632799 was originally written to align the breakers in this panel and was completed, but the listed breakers no longer appeared flush, necessitating rework. The inspectors identified several non-conformances beyond the existing fit-up issue at hand relative to the original seismic qualification, non-conformances which were not identified in the licensee's CAP, nor addressed in the existing functional evaluation.

The inspectors observed several breakers in the panel under observation (breakers 14 and 36) exhibiting some fashion of tape modification which was apparently intended to take up space between the front face panel and the rear restraint bars – space which existed because the procured and installed breakers did not have sufficient dimensional characteristics so as to be rigidly retained in compression between the two structural members (front panel face-piece and rear restraining bars). For the two observations in question, tape was only applied to the lower restraint and the inspectors observed a gap at the top of the breaker where the breaker was not seated against the upper restraint bar as per the Westinghouse 1975 Seismic Test Procedure No. CO-33419. The licensee provided TVA drawings 1,2 45N706 (Series) containing a note that, "The circuit breaker restraint bars may be modified by adding extra scotch 13 or 23 tape, approximately 1/4 inch thick and 1 inch wide, to secure the newer style Heinemann breakers." The inspectors observed the tape repair in other cabinets, and, in some cases, observed that tape was applied to both the upper and the lower restraint bars.

Maintenance technicians questioned by the inspectors described the tape repair as one of balling up some electrical tape to a sufficient thickness, and then taping the ball to the restraint bar per the drawing. The licensee was unable to provide neither any testing or analysis which justified the adequacy of the tape repair, nor any specific acceptance criteria for its application per the drawing note. The licensee had performed a "push test" to demonstrate panel operability in response to the 2011 non-cited violation, as had Watts Bar; but unlike Watts Bar, Sequoyah Nuclear Station had only performed the push test on the lower center line of the breaker, which would not challenge the lack of restraint observed at the top of the breaker. Discussions with engineering personnel indicated that given the acceptability of the repair by the drawing note, breakers requiring the tape repair may remain in the panel through their expected 10 year service interval (i.e. most likely would not have been addressed by the fall 2013 U1R19 refueling outage for Unit 1 full compliance, or the spring 2014 U2R19 outage for Unit 2 full compliance). This undocumented non-conformance would have been in contradiction to the licensee's July 17, 2012, commitment letter.

The inspectors identified additional instances in the field where the licensee had not properly accounted for, nor addressed, non-conformances which existed because of a loss of configuration control of the 120-VAC vital instrument boards to their original qualification, undermining the direct application of those results to the current configuration. In some instances, the licensee had installed circuit breakers which varied in form and fit so as to be less conservatively seated across the rear restraint bars than those originally tested, in that they only made contact from the vertical centerline to one side, which might allow the breaker to roll out of its retention and introduce acceleration effects which could be higher than the original acceleration input to the shake table. As with the tape repair, these breakers would likely not have been replaced prior to the expiration of their service interval. This condition was not captured in the licensee's CAP nor addressed in the existing functional evaluation. In all panels, maintenance aids (foam weather-stripping and vertical support rods applied to the restrain bars) which were not on design drawings nor listed for the original seismic qualification testing, would be addressed by the licensee's revised testing and modification but were not in the licensee's CAP nor contained within the existing functional evaluation as required by licensee procedures. Finally, the inspectors observed several instances where molded case circuit breakers which had been aligned to protrude through the face-plate did not exhibit seating upon the weather-stripping as neighboring breakers were, resulting in a possible gap in the intended retention with no documented acceptance criteria. Once again, this non-conformance was not addressed in the licensee's CAP or in the existing functional evaluation.

Analysis. The inspectors determined that the licensee's failure to ensure measures used to review the suitability of application of materials, parts, and equipment essential to the safety-related functions of molded case circuit breakers, and measures to provide for the verification of checking the adequacy of design were in place was a performance deficiency. This performance deficiency was more than minor because it affected the design control attribute of the mitigating systems cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, adequate measures were not implemented to ensure the station 120-VAC vital instrumentation boards had properly maintained their seismic qualification for their application. The inspectors assessed this finding for significance in accordance with NRC Manual Chapter 0609, Appendix A, Exhibit 2, Significance Determination Process (SDP) for Findings At-Power – Mitigating Systems Screening Questions, and determined that it was of very low safety significance (Green) as the devices in question had been intrinsically qualified for this application as part of a complete panel test by the original vendor and the licensee determined that the SSC maintained its operability or functionality despite the identified non-conformances. The inspectors evaluated this finding and violation of NRC requirements in accordance with the NRC Enforcement Policy, Section 2.3.2, and found two conditions to not be met requiring a Notice of Violation be issued. First, inspectors found the licensee failed to restore compliance within a reasonable time after the original violation (05000327.328/2011002-01) was identified. The NRC Enforcement Manual, Section 3.1.2.A.1.b).1), further defines restoring compliance to include those actions taken to stop an ongoing violation from continuing. Second, the inspectors determined that the identified non-conformances represented a repetitive violation as a result of inadequate corrective action and that identification was by the NRC inspector. The lack of rigor in addressing the root of the prior violation which resulted in the inadequate corrective action further led the inspectors to identify a crosscutting aspect in the CAP component of the Problem Identification and Resolution area [P.1(c)].

Enforcement. 10 CFR 50, Appendix B, Criterion III, Design Control, states, in part, that measures shall be established for the review for suitability of application of materials, parts, and equipment that are essential to the safety related functions of the structures, systems, and components (SSCs) and that the design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program. Where a test program is used to verify the adequacy of a specific design feature in lieu of other verifying or checking processes, it shall include suitable qualifications testing of a prototype unit under the most adverse design conditions. Contrary to the above, modifications had been made to the station's 120-VAC panels for which suitability reviews and verification or checking the adequacy of design could not be demonstrated. This finding was entered into the licensee's corrective action program as PER 668367 and the licensee planned to perform an expanded push-test to certain breakers for which the tape repair had been applied. This is identified as violation (VIO) 05000327,328/2012005-05, Failure to Adequately Evaluate and Qualify Molded Case Circuit Breakers.

.3 Selected Issue Follow-up: PER 506338, Motor Operated Valve Critical Joint Bolting

a. Inspection Scope

On February 15, 2012, the licensee identified (PER 506338) a broken bolt discovered in the housing of a motor operated valve (1-MVOP-063-156-A, SIS PMP OUT RCS LP 1&3 HL) undergoing MOVATS testing for the U1R18 refueling outage. The bolt served as one of six retainers for the spring pack cartridge, which Limitorque classifies as a critical component. The bolts were to be 5/16 by 1-1/2 inches. Technicians were to verify the proper spring pack was installed, which required disassembly and removal of the spring pack cartridge. Six dissimilar fasteners were found, one of which proved to be a 3/8 by 3/4 inches non-quality fastener used to fill the cartridge hole. The previous fastener had been broken and blocked the threaded hole in the housing from inserting a proper fastener. Upon investigation, the spring pack was replaced in April 2009 under work order 08-776288-000. Of the five retaining bolts also removed, three were incorrect to the application: two were 1-1/4 inches long (with approximately 1/4 inch of thread engagement) and one of these was marked as 316 SS which is not the correct material, one was 1-1/8 inches long (with only 1/8 inch thread engagement). Two were of the correct length and type. It was also reported that the limit switch cover had bolts of differing lengths and types installed in it as well.

The inspectors reviewed the appropriateness of the assigned significance, the scope and depth of the causal analysis, and the timeliness of the resolution. The inspectors assessed whether the evaluation identified likely causes for the issues and identified appropriate corrective actions to address the identified causes. The inspectors also conducted a review of the corrective actions to verify that appropriate measures were in place to prevent reoccurrence of the issue. In addition, the inspectors assessed whether the licensee's evaluation considered extent of condition, generic implications, common cause, and previous occurrences. The inspectors reviewed the potential impact on nuclear safety and risk to verify that the licensee had taken corrective actions commensurate with the significance of the issue. The inspectors evaluated these actions against the requirements of the licensee's CAP and performance attributes contained in IP 71152, Section 03.06.

These activities constitute completion of one in-depth problem identification and resolution sample as defined in Inspection Procedure 71152-05.

b. Findings and Observations

Over the course of the U2R18 Refueling Outage, contract personnel performing motor operated valve work on behalf of the licensee identified additional instances where non-conforming bolting was utilized, either within or outside of approved maintenance instructions, for critical joint bolting on safety-related valves. As a result, one licensee identified violation is listed in Section 4OA7 of this report.

.4 Semi-Annual Trend Review

a. Inspection Scope

As required by Inspection Procedure 71152, the inspectors performed a review of the licensee's CAP and 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 through December 2012, 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 2012. This inspection satisfied one inspection sample for Semi-annual Trend Review.

b. Findings and Observations

No findings were identified. In general, the licensee had identified trends and appropriately addressed them in their CAP. The inspectors evaluated the licensee trending methodology and observed that the licensee had performed a detailed review. The licensee routinely reviewed cause codes, involved organizations, key words, and system links to identify potential trends in their data. The inspectors compared the licensee process results with the results of the inspectors' daily screening.

The inspectors identified a trend which included three issues that involved deficiencies in the way the site has been treating components and procedures associated with external events such as high winds, earthquakes, and freeze protection.

- PER 515684 - NRC-identified NCV of Units 1 & 2 TS 6.8.1.a for failure to adequately implement procedure AOP-N.02, "Tornado Watch/Warning." Specifically, on March 2, 2012, AOP-N.02 was entered due to a tornado watch/warning, and loose material in the Switchyard/Transformer Yard was not secured or removed as required by the procedure.

- PERs 264271, 266599, 286156, 319161, and 631346 - NRC-identified NCV 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.

- PER 665663 - NRC-identified NCV of Units 1&2 Technical Specification 6.8.1.a for the licensee's failure to follow station procedures to adequately implement freeze protection requirements. Specifically, inspectors found a number of requirements improperly executed with no specific follow-up of those requirements contained within periodic instructions used to verify program implementation.

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 (Closed) Temporary Instruction 2515/188 – Inspection of Near-Term Task Force Recommendation 2.3 Seismic Walkdowns

a. Inspection Scope

The inspectors accompanied the licensee on their seismic walkdowns of the Emergency Diesel Generator Building on July 16, 2012, and the 6.9kV Shutdown Board Rooms on August 16, 2012, and verified that the licensee confirmed that the following seismic features associated with the Emergency Diesel Generators and the 6.9kV Shutdown Boards were free of potential adverse seismic conditions.

- Anchorage was free of bent, broken, missing, or loose hardware.
- Anchorage was free of corrosion that is more than mild surface oxidation.
- Anchorage was free of visible cracks in the concrete near the anchors.
- Anchorage configuration was consistent with plant documentation.

- SSCs will not be damaged from impact by nearby equipment or structures.
- Overhead equipment, distribution systems, ceiling tiles and lighting, and masonry block walls are secure and not likely to collapse onto the equipment.
- Attached lines have adequate flexibility to avoid damage.
- The area appears to be free of potentially adverse seismic interactions that could cause flooding or spray in the area.
- The area appears to be free of potentially adverse seismic interactions that could cause a fire in the area.
- The area appears to be free of potentially adverse seismic interactions associated with housekeeping practices, storage of portable equipment, and temporary installations (e.g., scaffolding, lead shielding).

On August 17, 2012, the inspectors independently performed their walkdown and verified that the ERCW system, including the pumps, motors, strainers, breakers, pipes and associated valves, all located in the ERCW building were free of potential seismic conditions.

- Anchorage was free of bent, broken, missing, or loose hardware.
- Anchorage was free of corrosion that is more than mild surface oxidation.
- Anchorage was free of visible cracks in the concrete near the anchors.
- Anchorage configuration was consistent with plant documentation.
- SSCs will not be damaged from impact by nearby equipment or structures.
- Overhead equipment, distribution systems, ceiling tiles and lighting, and masonry block walls are secure and not likely to collapse onto the equipment.
- Attached lines have adequate flexibility to avoid damage.
- The area appears to be free of potentially adverse seismic interactions that could cause flooding or spray in the area.
- The area appears to be free of potentially adverse seismic interactions that could cause a fire in the area.
- The area appears to be free of potentially adverse seismic interactions associated with housekeeping practices, storage of portable equipment, and temporary installations (e.g., scaffolding, lead shielding).

Observations made during the walkdown that could not be determined to be acceptable were entered into the licensee's corrective action program for evaluation.

Additionally, inspectors verified that items that could allow the spent fuel pool to drain down rapidly were added to the SWEL and these items were walked down by the licensee.

b. Findings

No findings were identified.

.3 (Discussed) NRC Temporary Instruction (TI) 2515/187, Inspection of Near-Term Task Force Recommendation 2.3 Flooding Walkdowns

a. Inspection Scope

Inspectors conducted independent walkdowns to verify that the licensee completed the actions associated with the flood protection feature specified in paragraph 03.02.a.2 of this TI. Inspectors are performing walkdowns at all sites in response to a letter from the NRC to licensees, entitled "Request for Information Pursuant to Title 10 of the *Code of Federal Regulations* 50.54(f) Regarding Recommendations 2.1, 2.3, and 9.3, of the Near-Term Task Force Review of Insights from the Fukushima Dai-Ichi Accident," dated March 12, 2012 (ADAMS Accession No. ML12053A340).

Enclosure 4 of the letter requested licensees to perform external flooding walkdowns using an NRC endorsed walkdown methodology (ADAMS Accession No. ML12056A050). Nuclear Energy Industry (NEI) document 1207 titled, "Guidelines for Performing Verification Walkdowns of Plant Protection Features," (ADAMS Accession No. ML12173A215) provided the NRC endorsed methodology for assessing external flood protection and mitigation capabilities to verify that plant features, credited in the CLB for protection and mitigation from external flood events, and are available, functional, and properly maintained.

b. Findings

Findings or violations associated with the flooding walkdowns, if any, will be documented in future reports.

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

a. Inspection Scope

The inspectors interviewed personnel and reviewed the licensee's documentation regarding storing spent fuel to verify that these independent spent fuel storage installation (ISFSI) related programs and procedures fulfill the commitments and requirements specified in the Safety Analysis Report (SAR), Certificate of Compliance (CoC), 10 CFR Part 72, the Technical Specifications (TS), and any related 10 CFR 72.48 or 72.212(b) evaluations. In addition, the inspectors reviewed seventeen specific 10 CFR 72.48 screening reviews for ISFSI procedure changes over calendar year 2012 and verified that all changes were consistent with the license and CoC, and did not reduce program effectiveness. Inspectors reviewed the licensee's annual special nuclear material inventory, conducted during April 2012, to verify that requirements of 10 CFR 72.72(b) are satisfied. Inspectors observed a daily ISFSI periodic instruction performance by operations and walked down the entire pad to verify material condition attributes satisfied design assumptions. No ISFSI campaign was conducted during the reporting period. Documents reviewed are listed in the Attachment. The inspectors completed one sample

b. Findings

No findings were identified.

.5 Unit 2 Steam Generator Replacement Project (SGRP)

Background: Based on past Steam Generator inspection results, the licensee determined that the Inconel alloy tubing experienced degradation from Primary Water Stress Corrosion Cracking (PWSCC) and Outside Diameter Stress Corrosion Cracking (ODSCC). The licensee concluded that continued tube degradation would lead to higher maintenance costs associated with inspection and repairs, increased potential for tube leakage, and eventually to reduced operating efficiency as additional tubes were plugged. The licensee determined that it was conservative and prudent to replace the four Westinghouse Model 51 steam generators to support continued operations of Unit 2. The licensee opted to replace the old steam generators (OSGs) of Unit 2 with Westinghouse Model AG57+ replacement steam generators (RSGs) to provide safe and reliable service, as well as increased thermal-hydraulic operating efficiency. The NRC inspectors used IP 50001, Steam Generator Replacement Inspection, to inspect the activities associated with this project.

Inspection Overview: This inspection report documents completion of inspections required by IP 50001, Steam Generator Replacement Inspection, some of which were fulfilled by completion of baseline inspection procedures. The table below identifies and correlates specific IP 50001 inspection requirements examined during this inspection period with the corresponding sections of this report.

IP 50001 Section	**Inspection Scope**	**Section(s) of this Report**
	Design and Planning Inspections	
02.02.a.1	Design changes and modifications to SSCs described in the UFSAR reviewed for compliance with 10 CFR 50.59.	4OA5.5.1
02.02.a.2	Key design aspects and modifications for the replacement SGs and other Mods associated with SG replacement.	4OA5.5.1
02.02.b	Engineering design, modification, and analysis associated with SG lifting and rigging.	4OA5.5.2
02.02.c	Radiation protection program controls, planning, and preparation.	4OA5.5.3, 2RS1, 2RS2, 2RS3, 2RS4, and 2RS5
02.02.d.1	Security considerations associated with vital and protected area barriers that may be affected during replacement activities	4OA5.5.4
02.02.d.2	Controls and plans to minimize any adverse impact on the operating unit and common systems	4OA5.5.4

	Steam Generator Removal and Replacement Inspections	
02.03.a.1	Special procedures for welding and Non-Destructive Examination (NDE)	4OA5.5.5 and 1RO8
02.03.a.2	Training and qualifications for welding and NDE personnel.	4OA5.5.5 and 1RO8
02.03.a.3	NDE results and work packages for selected welds.	4OA5.5.5 and 1RO8
02.03.a.4	Completion of pre-service NDE requirements for welds.	4OA5.5.5 and 1RO8
02.03.a.5	Completion of baseline eddy current examination of new SG tubes.	4OA5.5.5 and 1RO8
02.03.b	Activities associated with lifting and rigging.	4OA5.5.6
02.03.c	Old and new SG cutting, movement, and reconnection inside and outside containment.	4OA5.5.7
02.03.d	Major structural modifications to facilitate SG replacement.	4OA5.5.8
02.03.e	Restoration of temporary containment opening and containment leakage testing.	4OA5.5.9
02.03.f.1	Establishment of operating conditions including defueling, RCS drain down, system isolation and safety tagging	1R20 and 4OA5.5.10
02.03.f.2	Implementation of radiation protection controls.	4OA5.5.10, 2RS1, 2RS2, 2RS3, 2RS4, and 2RS5
02.03.f.3	Controls for excluding foreign materials in the primary and secondary side of the SGs and in related RCS openings	1R20 and 4OA5.5.10
02.03.f.4	Installation, use, and removal of temporary services	1R20 and 4OA5.5.10
02.03.g	Radiological safety plans for temporary storage or disposal of retired SGs and components.	4OA5.5.11, 2RS1, 2RS2, 2RS3, 2RS4, and 2RS5
	Post-Installation Verification and Testing Inspections	
02.04.1	Containment leak testing	1R19, 1R20 and 4OA5.5.12
02.04.2	Post-installation inspections and verifications program and implementation	4OA5.5.12
02.04.3	RCS leakage testing	4OA5.5.12

02.04.4	SG secondary side leakage testing	4OA5.5.12
02.04.5	Calibration and testing of instrumentation	1R22 and 4OA5.5.12
02.04.6	Procedures for equipment performance testing	1R19 and 4OA5.5.12

.5.1 Design Changes and Modifications to Systems, Structures, and Components – 10 CFR 50.59 Review

a. Inspection Scope

As required by IP 50001, Sections 02.02.a.1 and 02.02.a.2, the inspectors reviewed design change notice (DCN) D-22471A, Implement Reactor Building Structural Modifications Required to Support Steam Generator Replacement, for key design aspects and modifications of the replacement steam generators and verified that changes to the facility as described in the UFSAR were reviewed and documented in accordance with 10 CFR 50.59. To complete these reviews, the inspectors used IP 71111.17 as guidance, as suggested in IP 50001.

b. Findings

No findings were identified.

.5.2 Review of Engineering Design, Modification, Testing, and Analysis Associated with SG Lifting and Rigging

a. Inspection Scope

As required by IP 50001, Section 02.02.b, the inspectors reviewed the adequacy of the SGRP lifting programs as described in 39866-EP-C-004, Revision 0, assuring that it was prepared in accordance with regulatory requirements, appropriate industrial codes, and standards; and verified that the maximum anticipated loads to be lifted would not exceed the capacity of the lifting equipment and supporting structures.

The inspectors reviewed the adequacy of the haul route evaluation, placement of temporary protection for plant commodities and haul route upgrades required to prepare the haul route for load testing and transport of the steam generators as described in 39866-EP-C-006, Revision 2. The inspectors verified that they had been prepared in accordance with regulatory requirements, appropriate industrial codes, and standards and also discussed the transport path load testing with SGRP Engineering personnel.

b. Findings

No findings were identified.

.5.3 Review Radiation Protection Program Controls, Planning, and Preparations

a. Inspection Scope

As required by IP 50001, Sections 02.02.c, the inspectors reviewed radiation protection program controls, planning, and preparation in the following areas utilizing applicable portions of baseline inspection procedures IP 71124.01, 71124.02, 71124.03, 71124.04, and 71124.06 as guidance:

- As Low As Reasonably Achievable (ALARA) planning.
- Dose estimates and dose tracking.
- Exposure controls including temporary shielding.
- Contamination controls.
- Radioactive material management.
- Radiological work plans and controls.
- Emergency contingencies.
- Project staffing and training plans.
- Airborne radioactivity effluent controls.

These areas are documented in Sections 2RS1, 2RS2, 2RS3, 2RS4, and 2RS5 of this report.

b. Findings

No findings were identified.

.5.4 Security Considerations Associated with Vital and Protected Area Barriers and Plans to Minimize Adverse Impacts on the Operating Unit and Common Systems

a. Inspection Scope

As required by IP 50001, Sections 02.02.d.1 and 02.02.d.2, security inspectors reviewed the concept of operations with licensee representatives and determined that the licensee would have appropriate measures in place by the time that the actual outage work would begin. The inspectors walked down areas associated with vital and protected area barriers that could be affected during replacement activities and concluded that they were operational and that the licensee was in compliance with the security requirements. The inspectors also reviewed licensee controls and plans to minimize any adverse impact on the operating unit and common systems and determined that they were adequate and appropriate. These activities were completed prior to the SGR outage.

b. Findings

No findings were identified.

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

a. Inspection Scope

As required by IP 50001, Section 02.03.a, the inspectors reviewed welding and non-destructive examination (NDE) activities associated with the SGRP. These items and activities were inspected in accordance with IP 71111.08 and are documented in Section 1R08 of this report.

b. Findings

No findings were identified.

.5.6 Activities Associated with Lifting and Rigging

a. Inspection Scope

As required by IP 50001, Section 02.03.b, the inspectors examined the SGRP lifting equipment necessary to perform steam generator rigging and transport; design evaluation/erection/use and disassembly of the Outside Lift System (OLS); removal of sections of the shield building dome concrete, steel containment vessel, and steam generator compartment concrete; and load drop protection.

b. Findings

No findings were identified.

.5.7 Old and New SG Cutting, Movement, and Reconnection Inside and Outside Containment

a. Inspection Scope

As required by IP 50001, Section 02.03.c, the inspectors observed various portions of the process of the old steam generators (OSG) being lifted from the steam generator (SG) cubicle through the temporary penetrations in the steel containment vessel and shield building to the hydraulic trailer transporter. The inspectors also observed various portions of the sequence of the replacement steam generators (RSG) being transferred from the hydraulic trailer transporter, upended, lifted, and positioned into their respective cubicles. During these observations the inspectors performed visual inspections of the Outside Lift System and the hydraulic trailer transporter.

b. Findings

No findings were identified.

.5.8 Major Structural Modifications to Facilitate SG Replacement

a. Inspection Scope

As required by IP 50001, Section 02.03.d, the inspectors reviewed the licensee's 50.59 evaluations, Engineering Packages, Engineering Change Requests, associated drawings and Design Change Notices, and other documents to verify that the work had been properly planned and would be executed in compliance to applicable codes and standards and industry practices. The inspectors observed several structural modification activities, including removal of sections of the dome of the Shield Building, removal of sections of the SCV, removal of the top of SG cubicles and their restoration, to verify that the work was done by approved procedures and complied with applicable codes and standards and industry practices.

b. Findings

A licensee identified violation is documented in Section 4OA7.

.5.9 Containment Access and Restoration of Temporary Containment Opening and Containment Leakage Testing

a. Inspection Scope

As required by IP 50001, Section 02.03.e, the inspectors reviewed containment restoration activities associated with the two temporary construction openings in both the shield building dome and steel containment vessel, which were approximately 20 feet by 40 feet, for the Unit 2 SGRP.

The inspectors reviewed drawings and procedures for the installation of concrete reinforcing steel and Bar-Lock splices and procedures for the control of concrete placement activities. The inspectors observed installation of concrete reinforcing steel and installation of Bar-Lock splices to determine if the work was completed in accordance with requirements shown on design drawings. The inspectors reviewed results of quality control acceptance testing performed on materials (cement, fine and coarse aggregate, water, and admixtures) selected for batching of the concrete and results of qualification testing for the Bar-Lock splices. The inspectors also reviewed the concrete mix data to ensure that selected trial mix met concrete design strength requirements, and that Quality Control (QC) acceptance criteria specified in the procedures for the concrete were based on the trial mixes.

The inspectors reviewed drawings and observed work activities for restoration of the steam generator cubicle roofs after replacement of the Unit 2 steam generators to verify that the acceptance criteria for the restoration of the cubicle roofs, which were specified on the design drawings, were met.

The inspectors observed concrete placement during the restoration of the Shield Building dome and related concrete testing and sampling activities by the independent laboratory to assure compliance with American Concrete Institute (ACI) and American Society for the Testing of Materials (ASTM) codes.

The inspectors also reviewed the procedure used for the local leak rate test (LLRT) of the SCV and observed its implementation to assure ASME code compliance.

b. Findings

No findings were identified.

.5.10 Operating Conditions throughout the SGRP Process

a. Inspection Scope

As required by IP 50001, Section 02.03.f, the inspectors routinely inspected the following activities as they occurred throughout this inspection period:

- Establishment of operating conditions including defueling, RCS drain down, and system isolation and safety tagging/blocking.
- Implementation of radiation protection controls.
- Implementation of controls for excluding foreign materials in the primary and secondary side of the SGs and in the related RCS openings.
- Installation, use, and removal of temporary services directly related to steam generator replacement activities.

b. Findings

No findings were identified.

.5.11 Radiological Safety Plans for Temporary Storage of Retired SGs and Components

a. Inspection Scope

As required by IP 50001, Section 02.03.g, the inspectors reviewed the implementation of the radiation controls as listed in Section 2RS1 of this report, which included plans for the temporary storage of the retired SGs and their components.

b. Findings

No findings were identified.

.5.12 SG Post-Installation Verification and Testing

a. Inspection Scope

As required by IP 50001, Section 02.04, the inspectors performed selective reviews and inspections, consistent with the safety significance, of the following areas: containment leak testing; the licensee's post-installation inspections and verifications program and its implementation; RCS leakage testing and review the test results; SG secondary side leakage testing and review the test results; calibration and testing of instrumentation for both the primary (RCS) and secondary side (FW and MS) systems affected by the SG replacement; and procedures for equipment performance testing required to confirm the design and to establish baseline measurements, to include post-installation and power ascension. These items and activities were inspected in accordance with IP 71111.19, .20, .22 and are documented in Section 1R19, 1R20, and 1R22 of this report.

b. Findings

No findings were identified.

4OA6 Meetings

.1 Exit Meeting Summary

On January 11, 2013, 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.

An exit meeting for the ISI portion was conducted on November 2, 2012, with J. Carlin of licensee management and other members of his staff. The licensee did not identify any material provided to the inspector to be proprietary.

On December 7, 2012, regional inspectors discussed results of the onsite radiation protection inspections with Mr. A. Day, Radiation Protection Manager, and other responsible staff. The inspectors noted that one proprietary document was reviewed during the course of the inspection that would be properly disposed of when no longer needed.

Due to the complexity and length of the SGRP, several status meetings were conducted with licensee management through the course of the evolution.

4OA7 Licensee-identified Violations

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

.1 10 CFR 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” states, in part, that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. The Quality Execution Procedure governing the licensee’s contractor for the steam generator replacement project, 11.01 paragraph 2.5.1a requires that work be performed in strict accordance with the Work Instructions provided in the applicable Work Package. Contrary to these requirements, in work packages 3575A/D for the 2-1 and 2-4 steam generator enclosure plugs, multiple steps for the welding of tapered shims were not performed as indicated in the work packages. The licensee found the omission in document review and closure, and performed rework to establish the requisite number of shims for the seismic support requirements of the subject enclosure plugs. The inspectors determined that the violation was not greater than of very low safety significance as it was identified and corrected with the reactor defueled in Mode 6. The issue is documented in the licensee’s CAP as the contractor’s Nonconformance Report (NCR) 1163, and TVA PER 655762.

.2 10 CFR 50, Appendix B, Criterion II, “Quality Assurance Program,” states, in part, that the quality assurance program shall provide control over activities affecting the quality of the identified structures, systems, and components, to an extent consistent with their importance to safety and that activities affecting quality shall be accomplished under suitably controlled conditions to include the use of appropriate equipment. Contrary to these requirements, the licensee’s quality assurance controls failed to ensure that critical fasteners utilized in reassembly of the site’s motor-operated valves, specifically those utilized in safety-related service, were of the appropriate grade, class, or type, to meet design output requirements. The inspectors determined that the violation was not greater than of very low safety significance as it was determined that fasteners utilized would permit the valves to perform their function, although they were not acceptable per design. The issue is documented in the licensee’s CAP as TVA PERs 506338, 518423, 622076, 623383, 624373, 626982, and 644661.

.3 Facility operating license DPR-79 condition 2.C.(13) states that TVA shall implement and maintain in effect all provisions of the approved fire protection program referenced in Sequoyah Nuclear Plant’s Final Safety Analysis Report and as approved in NRC Safety Evaluation Reports contained in NUREG-0011, Supplements 1, 2, and 5, NUREG-1232, Volume 2, NRC letters dated May 29, and October 6, 1986, and the Safety Evaluation issued on August 12, 1997, for License Amendment No. 218. Contrary to the above, on June 28, 2011, the licensee did not implement and maintain in effect all provisions of the approved fire protection program. Specifically, Sequoyah’s Fire Protection Report Part II, Limiting Condition for Operation (LCO) 3.3.3.8.a.1 & 3.7.11.2.a.1 require a fire watch to be established when the required number of operable fire detection instruments and the required number of spray and/or sprinkler systems are inoperable. On January 3, 2012, the licensee discovered that standing Fire Protection Impairment Permit (FPIP) FOR110249 and the associated Fire Protection Report Part II action statement had been incorrectly entered. The licensee entered this issue into the corrective action program as PER 485817. The finding was screened using Inspection Manual Chapter 0609, Appendix F – Fire Protection Significance Determination Process, and was determined to be of very low safety significance (Green).

.4 In place of the controls required 20.1601(a) and b of 10 CFR Part 20, TS 6.12.2 requires that entryways into HRAs with dose rates exceeding 1 rem/hour at 30 cm be provided with a locked or continually guarded door or gate that prevents unauthorized entry. Contrary to this, on December 16, 2011, during the dewatering of a spent resin liner, the entryway into the Auxiliary Building Railroad Bay was controlled using remote monitoring and surveillance. During the resin transfer an individual was assigned as the access controller and a camera was placed in the area to monitor the LHRA. The access controller monitored the LHRA using the camera in Laundry/Trash Sorting area adjacent to the Railroad Bay behind a closed door. The dose rates on the top of the liner were 1500 mrem/hr at 30 cm. Although no unauthorized entry occurred during this time period, workers could have potentially entered the area from the Auxiliary Building Door. Although this event involved the failure to maintain proper control of a LHRA, this finding is of very low safety significance because there was neither evidence of unauthorized worker entry into the affected areas nor any unexpected radiation exposures to licensee personnel.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel
J. Barrick, Site ISI Engineer
B. Bashom, Operations Corrective Action Program
J. Carlin, Site Vice President
S. Carter, SGR Project Manager / Engineering
S. Connors, Operations Manager
J. Cross, Chemistry Manager
A. Day, Radiation Protection Manager
C. Dieckmann, Manager, Maintenance
M. Gillman, SGRP Manager
J. Hamilton, Steam Generator Replacement Corrective Action Program
T. Hamilton, Site Welding Engineer
J. Johnson, Licensing
J. Johnson, Program Manager
J. Johnston, Site Licensing
A. Keyser, Corporate ISI Program Owner
A. Little, Site Security Manager
K. Loomis, Site Assistant BACCP Owner
D. Love, WC PI Coordinator
T. Marshall, Director Safety and Licensing
S. McCamy, Quality Assurance Manager
B. Meers, Steam Generator Team LIII
P. Noe, Site Engineering Director
L. Perkins, Corrective Action Program Manger
W. Pierce, Assistant Engineering Director
P. Pratt, Work Control Manager
R. Proffitt, Licensing Manager
P. Simmons, Plant Manager
B. Sisson, SGT Senior Project Director
K. Smith, Director of Training
D. Sutton, Licensing
C. Thompson, Health Physicist, ALARA
T. Vander Wart, Assistant Operations Superintendent

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

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000327,328/2012005-05	VIO	Failure to Adequately Evaluate and Qualify Molded Case Circuit Breakers (Section 4OA2.2)

Opened and Closed

05000327,328/2012005-01	NCV	Failure to Implement Freeze Protection Program Requirements (Section 1R01)
05000327,328/2012005-02	NCV	Failure to Implement Fire Protection Impairment Requirements (Section 1R05)
05000327,328/2012005-03	NCV	Failure to Establish Adequate Procedures for Fire Protection Impairment Requirements (Section 1R05)
05000328/2012005-04	NCV	Failure to Perform ISI General Visual Examinations of Containment Moisture Barrier Associated with Containment Liner Leak Chase Test Connection Threaded Pipe Plugs (Section 1R08)
05000327/2012005-06	SL-IV	Failure to Submit a Technical Specifications Required Report (Section 1R20)

Closed

2515/188	TI	Inspection of Near-Term Task Force Recommendation 2.3 Seismic Walkdowns (Section 4OA5.2)

Discussed

2515/187	TI	Inspection of Near-Term Task Force Recommendation 2.3 Flooding Walkdowns (Section 4OA5.1)

LIST OF DOCUMENTS REVIEWED

<u>Section R01: Adverse Weather Protection</u>
<u>Procedures</u>
NPG-SPP-10.14, Freeze Protection, Revision 0
M&AI-27, Freeze Protection, Revision 12
1-PI-OPS-000-021.1, Control Room Operator MCR Duty Station Shift Relief and System Status Checklists, Revision 69
0-PI-OPS-000-006.0, Freeze Protection Revision, 53
0-PI-MIN-000-706.0, Freeze Protection Insulation Inspection, Revision 9
1/2-PI-EFT-234-706.0, Freeze Protection Heat Trace Functional Test, Revision 37
0-TI-DXX-000-013.0 Temporary Equipment Control, Rev 7

<u>PERs</u>
665633, NRC Identified Freeze Protection Issues

<u>Work Orders</u>
113407911, Implement M&AI-27 for the 2012/2013 Winter Season

<u>Section R04: Equipment Alignment</u>
<u>Partial System Walkdowns</u>
<u>Procedures</u>
0-SO-72-1, Containment Spray Systems, Revision 42
1-SI-SXP-072-201.B, Containment Spray Pump 1B Section XI Test, Revision 15
0-SO-78-1, Spent Fuel Pit Cooling System, Revision 59
0-SO-67-2, Layup of Containment Spray Heat Exchanger, Revision 24

<u>Other documents</u>
47W610-72-1, Control Diagram Containment Spray System
1-47W611-72-1, Logic Diagram Containment Spray System
Technical Specification 3.6.2.1 – Containment Spray System
0-47W855-1, 0-47W859-1, 0-47W454-3; Spent Fuel Pit Cooling System Drawings
47W812-1

<u>Complete System Walkdown</u>
<u>Procedures</u>
1-PI-SFT-084-001.0, Function Test of Flood Mode Boration Makeup System, Revision 10
2-PI-SFT-084-001.0, Function Test of Flood Mode Boration Makeup System, Revision 11
0-TI-SXX-000-005.0, Augmented Pump and Valve Test Program, Revision 9
0-SO-84-1, Flood Mode Boration makeup System, Revision 11
AOP-N.03, External Flooding, Revision 42

<u>Other documents</u>
FSAR Section 9.3.5, Auxiliary Charging System
SQN-DC-V-12.1, Design Criteria – Flood Protection, Revision 10
SQN-DC-V-12.6, Flood Mode Boration Makeup System, Revision 2
1-47W809-1, Flow Diagram CVCS
2-47W809-1, Flow Diagram CVCS
0-47W809-7, Flood Mode Boration Makeup System

0-47W803-2, Flow Diagram – Auxiliary Feedwater
45N756-1, 480V Board Diagram
45N756-5, 480V Board Diagram
45N756-6, 480V Board Diagram
Technical Requirements Manual (TRM) 3.7.6, Flood Protection

Section R05: Fire Protection

Procedures
NPG-SPP-18.4.6, Control of Fire Protection Impairments, Revision 1
NPG-SPP-18.4.7, Control of Transient Combustibles, Rev. 0
SQN-FPR-Part-II, SQN Fire Protection Report Part II – Fire Protection Plan, Revision 28
NPG-SPP-09.5, Temporary Alterations, Revision 2
NPG-SPP-09.17, Temporary Equipment Control, Revision 1

WOs
113547924, Installation of Temporary cables to support CCTV and audio system

PERs
635934, Inadequate procedural adherence with fire watches
635937, NPG-SPP-18.4.6, Control of Fire Impairments may be non-conservative
649975, QA identified a REPEAT procedure use and adherence issue that was identified by the NRC
658308, SGR Fire Watch was inattentive to duty and had contraband in the RCA
652672, NRC question about conduct of compensatory fire watch

Other documents
AUX-0-690-01, Auxiliary Building – El. 690 (Unit 1 Side), Revision 7
AUX-0-690-2, Auxiliary Building – El. 690 (Unit 2 Side), Revision 7
AUX-0-734-03, Auxiliary Building – El. 734, Revision 7
AUX-0-669-04, Auxiliary Building – El. 669 (ERCW Tunnels), Revision 5
RXB-0-679-02, Reactor Building – El. 679, Revision 3
RXB-0-701-02, Reactor Building Annulus Area – El. 701 & 721, Revision 3
RXB-0-734-02, Annulus – El. 740, 759, & 778, Revision 3
CON-0-706-00, Control Building – El. 706, Revision 5
CON-0-685-00, Control Building – El. 685, Revision 5
Technical Evaluation for WO 113048354, Revision 0
Fire Protection Impairment Permit FOR # 120932
Fire Protection Impairment Permit FOR # 120506
Fire Protection Impairment Permit FOR # 120507

Section R06: Flood Protection Measures

Procedures
1-AR-M15-B, Annunciator Response for Miscellaneous 1-XA-55-15B, Revision 29
0-AR-M-29, Annunciator Response for Fire Detection System, Revision 9
AOP-N.05, Earthquake, Revision 16
1-AR-M15-B, Annunciator Response for Miscellaneous 1-XA-55-15B, Revision 32

Work Orders
114181427, TDAFW Pump room drains need cleaning

PERs
653722, Clarification to SR 651936 – U2 TDAFW Drain Clogged
651941, TDAFW Pump room drains need cleaning
345431, Potential internal flooding vulnerability
165889, inadequate procedure for fire header ruptures
542835, FE required revision
539250, Internal flooding vulnerability
344249, Internal flooding (Japan IER L1 11-1)

Other documents
TVA letter to NRC dated May 4, 2007. TVA response to GL 2007-01

Section R08: Inservice Inspection Activities
Corrective Actions
PER 458020, Wet Active Boric Acid Contaminated Leak
PER 516195, Boric Acid Buildup on 0-VLV-078-0545
PER 551465, Boric Acid on SQN-1-VLV-074-0521, RHR Pump 1B-B Discharge Isolation Valve
PER 558612, Significant Boric Acid Buildup on CCP Seals
PER 601130, Boric Acid on 2-FCV-063-0078, SIS Accumulator Tank 3 Check Valve Isolation Valve
PER 601133, Boric Acid on 2-ISV-063-0804, SIS Test Header Isolation Valve
PER 636215, NRC Potential Green NCV
PER 659558, Through-wall Hole in Channel Test Connection Box Tubing

Drawings
48N401, Drawing (excerpt): B-B Typical [Liner] Plate Joint and Test Connection

Procedures
N-UT-33, Manual Ultrasonic Examination of Static and Centrifugally Cast Stainless Steel Piping Welds
N-UT-64, Generic Procedure for the Ultrasonic Examination of Austenitic Pipe Welds
N-UT-76, Generic Procedure for the Ultrasonic Examination of Ferritic Pipe Welds

Other Documents
3065A-WDC-001, Steam Generating Team Weld Data Card
3065A-WDC-002, Steam Generating Team Weld Data Card
3065B-WDC-001, Steam Generating Team Weld Data Card
3065B-WDC-002, Steam Generating Team Weld Data Card
3065C-WDC-001, Steam Generating Team Weld Data Card
3065C-WDC-002, Steam Generating Team Weld Data Card
3065D-WDC-001, Steam Generating Team Weld Data Card
3065D-WDC-002, Steam Generating Team Weld Data Card
3080A-WDC-002, Steam Generating Team Weld Data Card
3085D-WDC-001, Steam Generating Team Weld Data Card
55-PQ4144-01, Framatome Welding Procedure Qualification Record, Rev. 1

55-PQ7032-003, AREVA Welding Procedure Qualification Record, Rev. 3
55-PQ7037-06, Framatome Welding Procedure Qualification Record, Rev. 6
55-PQ7038-05, Framatome Welding Procedure Qualification Record, Rev. 5
55-PQ7225-00, AREVA Welding Procedure Qualification Record, Rev. 0
55-PQ7226-01, Framatome Welding Procedure Qualification Record, Rev. 1
55-PQ7227-00, AREVA Welding Procedure Qualification Record, Rev. 0
55-PQ7228-00, Framatome Welding Procedure Qualification Record, Rev. 0
55-PQ7232-03, Framatome Welding Procedure Qualification Record, Rev. 3
55-PQ7233-01, AREVA Welding Procedure Qualification Record, Rev. 1
GT/1.1-1, Steam Generating Team Welding Procedure Specification, Rev. 01
GT/1.1-2, Steam Generating Team Welding Procedure Specification, Rev. 01
GT/1.8-1, Steam Generating Team Welding Procedure Specification, Rev. 02
GT/8.8-1, Steam Generating Team Welding Procedure Specification, Rev. 04
GTM/1.1-2, Steam Generating Team Welding Procedure Specification, Rev. 01
GTM/8.8-1, Steam Generating Team Welding Procedure Specification, Rev. 02
GT-SM/1.1-1, Steam Generating Team Welding Procedure Specification, Rev. 01
GT-SM/1.1-2, Steam Generating Team Welding Procedure Specification, Rev. 01
GT-SM/1.1-Q6, Morrison Knudsen Corp. Welding Procedure Qualification Record, Rev. 1
GT-SM/1.3-4, Steam Generating Team Welding Procedure Specification, Rev. 03
GT-SM/1.8-Q4, Morrison Knudsen Corp. Welding Procedure Qualification Record, Rev. 3
PQ7220, Framatome Welding Procedure Qualification Record, Rev. 0
PQ7221, Framatome Welding Procedure Qualification Record, Rev. 0
PQ7222, Framatome Welding Procedure Qualification Record, Rev. 0
PQ7225, AREVA Welding Procedure Qualification Record, Rev. 0
QEP 12.06-2, Radiographic Examination Report – WO3080D, FW-1
QEP 12.06-2, Radiographic Examination Report – WO3085D, FW-1
QEP 12.06-2, Radiographic Examination Report – WO3720A, FW-1
R-0028, Reactor Pressure Vessel Lower Head Remote Visual Enhanced Bottom-Mounted Instrumentation (BMI) Penetration Examination
R-0043, TVA UT Calibration/Examination Report
R-0044, TVA UT Calibration/Examination Report
R-0045, TVA UT Calibration/Examination Report
R-0046, TVA UT Calibration/Examination Report
R-0092, TVA UT Calibration/Examination Report
R-0093, TVA UT Calibration/Examination Report
R-0095, TVA UT Calibration/Examination Report
R-0101, TVA UT Calibration/Examination Report
SM/1.1-1, Steam Generating Team Welding Procedure Specification, Rev. 04
SM/1.1-2, Steam Generating Team Welding Procedure Specification, Rev. 02
SM/1.3-1, Steam Generating Team Welding Procedure Specification, Rev. 00
SM/1.3-4, Steam Generating Team Welding Procedure Specification, Rev. 02
SM/1.8-1, Steam Generating Team Welding Procedure Specification, Rev. 00
SM/8.8-1, Steam Generating Team Welding Procedure Specification, Rev. 01
Steam Generating Team Welder Performance Qualification Record (Beach)
Steam Generating Team Welder Performance Qualification Record (Binkley)
Steam Generating Team Welder Performance Qualification Record (Burk)
Steam Generating Team Welder Performance Qualification Record (Clark)
Steam Generating Team Welder Performance Qualification Record (Culeton)

Steam Generating Team Welder Performance Qualification Record (Dorough)
Steam Generating Team Welder Performance Qualification Record (McElyea)
Steam Generating Team Welder Performance Qualification Record (Payne)
Steam Generating Team Welder Performance Qualification Record (Stewart)
Steam Generating Team Welder Performance Qualification Record (Ward)

Section R12: Maintenance Effectiveness
Procedures
TI-4, Maintenance Rule Performance Indicator Monitoring, Trending, and Reporting – 10 CFR 50.65, Revision 24

Work Orders
07-772135-000, EQ Maintenance and Inspection for SQN-2-MVOP-074-0033
111463684, Perform the "As Found" and "As Left" MOV Test to Support WO 112731542
112731542, EQ Maintenance and Inspection, Spring Pack Change Out
PERs
145361, 2-MVOP-74-33-A Snap Ring Failure, 5/17/2008
633120, Repeat Issue for 2-MVOP-74-33 Spiral Ring Failure to Maintain Position of Handwheel Clutch Gear, 10/29/2012
635935, QA Identified: Procedure Use and Adherence Issues on 2FCV-74-33 under WO 111463684

Other documents
System 30B, Containment Vacuum Relief CDE 2648 for zone switch failure
System 30K, SDBR A chiller CDE 2654 for thermal expansion valve
A(1) Plan, Revision 0, for 6.9 kV Shutdown Board (SDBD) Room Chillers (Train A and B)
Technical basis document for revision of the hydrogen recombiners a(1) plan to add all three remaining Unit 1 and Unit 2 trains of hydrogen recombiners to the current existing Unit 1 Train A recombiner a(1) plan
VTM-L200-0010, Limitorque Corp (Division of FlowServe Corp.) Type SMB, SB, and HVC Series Valve Operators, Rev. 16

Section R13: Maintenance Risk Assessments and Emergent Work Evaluation
Procedures
0-TI-DSM-000-007.1, Risk Assessment Guidelines, Revision 9
NPG-SPP-07.3, Work Activity Risk Management Process, Revision 11
NPG-SPP-07.2.4, Forced Outage or Short Duration Planned Outage Management, Revision 11
NPG-SPP-07.2, Outage Management, Revision 3
GOI-6, Apparatus Operations, Revision 151
NPG-SPP-07.1, On-Line Work Management, Revision 8

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
1-SI-EIV-268-305.A, Hydrogen Mitigation System Operability Current Check, Revision 0

Work Orders
112535580, Excessive amounts of water found in manhole

PERs
653651, HH-52B has 8” of water
655601, Found One Bad Hydrogen Igniter
523862, Emergency Diesel Generator 1B-B air starting motor air line lubricator (one of four total) did not provide lubricant as expected to two of the starting motors (8 total)
486952, Emergency Diesel Generator 2B-B air starting motor air line lubricator (one of four total) did not provide lubricant as expected to two of the starting motors (8 total)
549340, Vital Battery Charger III Preventive Maintenance identified output voltage change between no-load and full-load condition was greater than expected
366228, A-A Fire/Flood Mode Pump long term degrading insulation
659558, Through wall hole found in Channel Test Connection Box in Raceway
505606, Testing Frequency for 1,2-FCV-63-47 Does Not Conform to ASME OM Code Requirement (Lower Tier)

Other documents
0-47W611-67-6
0-45N771-1
TVA SQN Technical Specification Bases Change 12-02, dated 3/26/2012
TVA SQN Technical Specification Bases Change 12-04, dated 10/12/2012
TVA SQN Technical Specification Bases Change 12-05, dated 10/10/2012
TVA SQN Technical Specification Bases Change 12-06, dated 10/31/2012
TVA SQN Narrative Log Summary of All Operability Entries 1/1/2012 – 11/20/2012.
TVA SQN ESOMS Query Results for “Measures,” “Comp Action,” and “Bases,” dated 11/21/2012

Section R19: Post Maintenance Testing
Procedures
MMDP-1, Maintenance Management System, Revision 20
MMDP-3, Guidelines for Planning and Execution of Troubleshooting Activities, Revision 6
NPG-SPP-6.5, Foreign Material Control, Revision 0
NPG-SPP-6.1, Work Order Process Initiation, Revision 0
NPG-SPP-06.3, Pre-/Post-Maintenance Testing, Revision 0
NPG-SPP-06.9, Testing Programs, Revision 0
NPG-SPP-06.9.1, Conduct of Testing, Revision 1
NPG-SPP-06.9.3, Post-Modification Testing, Revision 0
2-SI-SXP-062-203.0, Centrifugal Charging Pumps 2A-A and 2B-B Comprehensive Pump Test and Check Valve Test, Rev. 5
2-STI-088-156.0, Primary Containment Vessel Post-Modification Pressure/New Weld Leakage Inspection Test, Revision 0
N-VT-4, System Pressure Test Visual Examination Procedure, Rev. 25
N-VT-15, Visual Examination of Class MC and Metallic Liners of Class CC Components of Light-Water Cooled Plants, Rev. 10

Work Orders
112619112, Centrifugal Charging Pump 2A-A Discharge piping
113387593, Centrifugal Charging Pump 2A-A Discharge piping
113945604, Unit 2 Annulus Door, Dome Integrity
114099131, 2B-B MDAFW Comprehensive Test
112926425, Perform U2 Cont LRT as PMT for DCN D22471A

Other documents
0-SO-65-1, Emergency Gas Treatment System Air Cleanup and Annulus Vacuum, Rev. 26
Visual Acuity Examination and Certification Records for C.G. Perron, J.A. Fergerson, M.C. Welch, and D.L. Allen

Section R20: Refueling and Outage Activities
Procedures
FHI-3, Movement of Fuel, Revision 65
0-GO-15, Containment Closure Control, Revision 34
0-GO-13, Reactor Coolant System Drain and Fill Operations, Revision 71
NPG-SPP-08.1, Nuclear Fuel Management, Revision 00
0-PI-OPS-000-011.0, "Containment Access Control During Modes 1-4, Revision 1
NPG-SPP-07.2.3, Plant Startup Review/Checklists, Revision 2
NPG-SPP-03.21, Fatigue Management and Work Hour Limits, Revision 8

PERs/SRs
SR 655713, QA Identified operations personnel standing watch in online unit while on outage work hours
648658, U1R18 Steam Generator 180 Day Report

Section R22: Surveillance Testing
Procedures
0-SI-SLT-030-258.1, Containment Isolation Valve Local Leak Rate Test Purge Air, Revision 8
0-SI-MIN-061-105.0, Ice Condenser – Ice Weighing, Revision 10
2-SI-OPS-088-001.0, Phase A Isolation Test, Revision 20
2-SI-OPS-082-026.B, Loss of Offsite Power with Safety Injection D/G 2B-B Test, Revision 40
0-SI-SLT-072-258.1, Containment Spray and RHR Spray Header Valve Water Inventory Test, Revision 10, Appendix D, Penetration X-49B
0-SI-SXV-068-201.0, Pressurizer PORV Operability Test, Rev. 1
0-SI-SXV-074-203.2, Full Stroking of RHR Valves FCV-74-1 and FCV-74-2, Rev. 1

Work Orders
112611722, 2-SI-OPS-088-001.0 Phase A Isolation Test
112611707, 0-SI-SXV-068-201.0 U2 Stroke Pzr PORVs
112610711, 2-SI-OPS-082-026.B DG 2B Loss of Offsite Pwr with SI
112612222, 0-SI-SLT-072.1 U2 Cntmt and RHR Spray Header Valve Water Inventory Test
112618932, 0-SI-SXV-074-203.2 U2 Stoke FCV-74-1 and FCV-74-2
112611172, 0-SI-MIN-061-105.0 U2 Ice Condenser – Ice Weighing (As Left)

Section 1EP4: Emergency Action Level and Emergency Plan Changes

Change Packages

EPIP -1, “Emergency Plan Classification Matrix,” Revision 46
Tennessee Valley Authority, Radiological Emergency Plan, Revision 97 and 98
EPIP-1, “Emergency Plan Classification Matrix,” Revision 47
EPIP-2, “Notification of Unusual Event,” Revision 31
EPIP-3, “Alert,” Revision 34
EPIP-4, “Site Area Emergency,” Revision 34
EPIP-5, “General Emergency,” Revision 42
CECC EPIP-8, “Dose Assessment Staff Activities During Nuclear Plant Radiological Emergencies,” Revision 37

Section 2RS1: Radiological Hazard Assessment and Exposure Controls

Procedures, Guidance Documents, and Manuals

RCI-15, Radiological Postings Rev. 0023
RCI-21, Control of Radioactive Materials, Rev. 0016
RCI-22, Contamination Control Rev. 0022
RCI-24, Control of Very High Radiation Areas Rev. 0012
RCI-29, Control of Radiation Protection Keys, Rev. 0012
RCI-28, Control of Locked High Radiation Areas Rev. 0013
RCI-29, Control of Radiation Protection Keys Rev. 0012
RCI-201, Radiation and Contamination Surveys, Rev. 0011
RCI-202, Airborne Radioactivity Surveys Rev. 006
RCI-204, Radiological Surveys of Equipment and Materials Leaving the RCA
RCI-209, Radiological Surveys of Personnel Leaving the RCA or Protected Area
RCI-403, Surveys during Cavity Drain Down, Rev. 001
RCI-404, Radiation Protection Requirements for Remote Job Coverage
RCI-414, Access Control Gates and Turnstiles Rev. 0000
RCDP-1, Conduct or Radiological Control Rev. 004
NPG-SPP-05.1.1, Alpha Radiation Monitoring Program, Rev. 00
NPG-SPP-05.1 Radiological Controls Rev. 002
O-SI-RCI-000-056.0, Byproduct Material Inventory and Sealed Source Leak Test Rev. 0013

Records and Data

PM#: P0615, Work Order #112094227, Sealed Source Leak Test 1/19/2012
Radiological Hazard Assessment and Exposure Controls Self-Assessment Report, June 2012
RWP # 11002045, Locked High Radiation Areas, SFP Filters, and Ion Exchange Changeout
RWP # 11037004, Steam Generator Full Jump for Installing/Removing Nozzle Dams
RWP # 11047135, U2 Reactor Cavity – LHRA, Inspection during Reactor Head lift.
RWP # 12000080, RCA Areas Steam Generator Pre-Outage Activities
RWP # 12017501, Sort Trash and Prepare Shipment during U1R18
RWP # 12017572, High Radiation Areas – MOVATs Valve maintenance and testing in support of U2R18 Refueling Outage.
RWP # 12027110, U2 Lower Containment- Reactor Coolant Pump Maintenance and RCP Motor Replacement for U2R18
RWP # 12027570, U-2 Lower Containment Excess Letdown Heat Exchanger Room
RWP # 12034004, Steam Generator, Full Jump For installing Removing Nozzle Dams

RWP # 12038050, U2 SGR Upper/Lower Containment, Cut, Weld, Prep/ Modify for Old/New Steam Generator Activities
RWP # 12038070, U2 SGR, Cut/Prep/Weld SCV Liner Plate
RWP # 12038790, U2. SGR LHR, Primary Platform Work and Associated Activities
RWP # 12044135, U-1 Reactor Cavity, Inspection During Reactor Head Lift and Head Set on Flange
RWP # 12047138, Clean Reactor Head Stud Bolts and Associated Work
RWP # 09047200, U-2 Upper Containment –MODs, Replace RP Cables From Reactor Head Vertical Panel To Devices on the Top of the Reactor
RWP # 09037020, U-2 Lower Containment Steam Generators 1-4, Full Jump for Installing and Removing Nozzle Dams
RWP # 1004413, U-1 Upper CTMT/RX Cavity: Remove/Replace CETNAS to Include Cleaning and Associated Work
RWP#09047062, U-2 Upper Containment: I&C- Remove and Install Thermocouples and Loose Parts Monitors
RWP 1004413, U-1 Upper CTMT/RX Cavity: LHRA –Remove/Replace CRDM Duct work, Flex Boots and Supports
RWP # 10044120, U-1 Upper CTMT/RX Cavity/Pressurizer: Install Temporary Shielding
RWP # 10044062, U-1 Upper CTMT: I&C – Remove/Install Thermocouples Connections and Loose Parts Monitors
RWP# 09047143, U-2 Reactor Cavity LHRA-Retrieve Stud Hole Plug
RWP # 09047159, U-2 RX Head Stand Area, Remove RX Head O-RINGS; Perform Manual Cleaning and Honing of Surface; QA/QC Inspections; Inspect RX Vessel Flange in Reactor Cavity
RWP#09047062, U-2 Upper Containment, Remove/Install Thermocouples and Loose Parts Monitors
PER 206522
PER 321906
PER 361203
PER 375694
PER 385541
PER 398297
PER 387916
PER 433228
PER 513469
PER 517994
PER 517997
PER 518630
PER 518641
PER 541215
PER 543370
PER 520428
PER 610583
Survey # SQN-M-20121015-2, U2# 1 RCP Seal Platform 10/15/2012, 10/16/2012, 10/17/2012
Survey # SQN-M-20121017-4, U2 #1, RCP Seal Platform 10/17/2012
Survey # SQN-O-20121023-29, Auxiliary Building Cavity Drain Down 10/23/2012
Survey # SQN-O-20121020-22, R211 RCP#1 South Side Main Flange Bolts
Survey # SQN-O-20121020-35, R211 RCP#1 Seal Removal, 10/20/2012

Survey # SQN-O-20121020-40, R211 RCP#1 Motor To Pump Mounting Flange, 10/20/2012
Survey # SQN-O-20121106-32, R223 S/G Platform-Steam Generator Removal11/6/2012
Survey # SQN-O-2012117-9, Steam Generator Primary Platform Survey for Tripod Installation 11/6/2012
Survey #, SQN-O-20120907-5, ISFSI Weekly Perimeter Survey 9/7/2012
Survey # SQN-M-12111-2, Cask Loading Area (Truck Bay) 12/18/2011
Survey # SQN-O-121711-6, Dewatering Survey on Top of Trailer, 12/17/2011
Survey # SQN-O-121811-6, Top of Liner on Fill Hear Prior to Disassembly, 12/18/2011
Survey # SQN-M-20121019-9, U-2 Excess Letdown HX – 3D, 10/18/2012
Survey # 10112003, U-1 Lower Containment Seal Table (air sample), 1/4/2012
Survey # 117012008, Update Air Sampling Record (air sample, 11/7/2012
Survey # 090612006, U-2 Lower Containment (air sample), 9/6/2012
Survey # 62812006, U-2 Raceway (air sample), 6/28/2012

Section 2RS2: Occupational ALARA Planning and Controls

Procedures, Guidance Documents, and Manuals

CHEM-002, Sequoyah Nuclear Plant Primary Water Chemistry Program Strategic Plan, Rev. 6
NPG-SPP-05.2, ALARA Program, Rev. 3
NPG-SPP-05.1, Radiological Controls, Rev. 2
NPG-SPP-05.16, Radiological Controls for Performance of Radiography Operations, Rev. 2
NPG-SPP-05.2.1, Operational ALARA Planning and Controls, Rev.
NPG-SPP-05.2.3, Outage Exposure Estimating and Tracking, Rev. 0
NPG-SPP-05.2.4, ALARA Suggestion Program, Rev. 0
NPG-SPP-05.9, Radiological Control and Radioactive Material Shipment Augmented Quality Assurance Program, Rev. 0
RCI-10, ALARA Program, Rev. 34
RCI-14, Radiation Work Permit (RWP) Program, Rev. 53
RCI-19, Temporary Shielding Program, Rev. 11
RCI-106, Radiation Protection Standards and Expectations, Rev. 2
RCI-302, Reactor Coolant System Monitoring Program, Rev. 0

Reports, Records, and Data

Annual Dose Report, CY 10 (RWP Total Dose, Hours and Dose Rate), 9/18/12
Annual Dose Report, CY 11 (RWP Total Dose, Hours and Dose Rate), 9/18/12
Change Management Presentation, Increase SQN-2 RCS Zinc Concentration after Steam Generator Replacement, 9/28/12
Design Change Notice (DCN), DCN S08492A, Storage of lead blankets inside containment Raceway during power operation, 12/13/93
Graphs: Trend of installed Electronic Dosimeter dose rates during crud burst and outage (hourly readings) U1R13 through U1R16
Graphs: Trend of installed Electronic Dosimeter dose rates during crud burst and outage (hourly readings) U2R13 through U2R16
Graphs: Installed Electronic Dosimeter dose rates during crud burst and outage (hourly readings) U1R18, 2/27 – 3/23/12
Graphs: U1 and U2 Co-58 & Co-60 Activity, 10/2010 through 9/2012, 10/3/12
Graph: U1R18 Radiation Protection Status Report (Dose vs. Goal), 9/14/12
Radiation Work Permit (RWP) 12017558, Appendix R Conduit Installation, Auxiliary Building HRA (for ALARA Plan 2012-064)

RWP 12027010, Routine Plant Maintenance, Lower Containment All Areas (for ALARA Plan 2012-062)
RWP 12017572, MOVATS Valve Maintenance, Auxiliary Building HRA (for ALARA Plan 2012-064)
RWP 12027072, MOVATS Valve Maintenance, Lower Containment HRA (for ALARA Plan 2012-064)
RWP 12038010, Special Plant Maintenance, All of Containment (for ALARA Plan 2012-067)
RWP 12038760, Special Plant Maintenance, Lower Containment All Areas (for ALARA Plan 2012-068)
RWP 12038790, Special Plant Maintenance, Lower Containment All Areas (for ALARA Plan 2012-070)
RWP 12047135, Refueling, Upper Containment All Areas (for ALARA Plan 2012-050)
Sequoyah Nuclear Plant (SQN) - ALARA Committee Meeting Minutes for the following dates; 1/6/12, 1/20/12, 1/23/12, 1/24/12, 1/25/12, 1/27/12, 2/14/12, 3/15/12, 4/10/12, 5/8/12, 6/12/12, 6/18/12, 6/28/12, 7/18/12, 7/26/12, 8/14/12, 8/22/12, 8/23/12, 8/29/12, 9/5/12, and 9/11/12.
Sequoyah Nuclear Plant Annual Radionuclide Trending and Assessment Report for 2011, 6/30/12
SQN - ALARA Committee Meeting Agenda, 12/5/12
Sequoyah Unit 1 Forced Outage Report - May 2012
SQN ALARA Plan 2012-10, Refueling Operations U1R18
SQN ALARA Plan 2012-10, Post job review. 4/18/12
SQN ALARA Plan 2012-12, S/G Primary and Secondary Maintenance and Inspections
SQN ALARA Plan 2012-12, Post job review. 3/17/12
SQN ALARA Plan 2012-19, Plant Services (Shielding, Upper & Lower Decon)
SQN ALARA Plan 2012-19, Post job review, 4/25/12
SQN ALARA Plan 2012-21, RP Surveillance
SQN ALARA Plan 2012-21, Post job review, 4/25/12
SQN ALARA Plan 2012-22, Modifications Support (Scaffolding and Insulation)
SQN ALARA Plan 2012-22, Post job review, 4/25/12
SQN ALARA Plan 2012-64, U2R18 MODS/MOVATS/Appendix R (Motor Operated Valve maintenance and testing)
SQN ALARA Plan 2012-64, Post job review, 12/4/12
SQN ALARA Plan 2012-050, Refueling Operations U2R18
SQN ALARA Plan 2012-062, Modifications Support Scaffold/Insulation U2R18
SQN ALARA Plan 2012-067, SGR Scaffolding (U2R18)
SQN ALARA Plan 2012-068, SGR RCS System and pipe End Decon (U2R18)
SQN ALARA Plan 2012-070, SGR Structural Interferences, Pipe Supports, and Replace SG (U2R18)
SQN Collective Radiation Exposure Reduction Initiatives 2011-2016
SQN Collective Radiation Exposure Gap Analysis 2010-2016
SQN FY 2012-2016 Radiation Protection Business Plan Input
Spreadsheet SQN FY 2010 Personnel Exposure Goals, including exposure by section and department.
Spreadsheet SQN FY 2011 Personnel Exposure Goals, including exposure by section and department.
Spreadsheet SQN FY 2012 Personnel Exposure Goals, including exposure by section and department.

Temporary Shielding Request (TSR) TSR 97-0021, Install shielding on ERCW radiation monitors 0-RM-90-133/134 and 0-90-RM-140/141
TSR 12-51, Install Reactor Head shielding package
TSR 12-53, Install 2 lead blankets on Reactor Cavity Drain Tank.
TSR 12-59, Install steel plate in overhead of 669' Aux building Hot Tool Room to shield resin transfer line.
TSR 12-62, Install tungsten mat shielding on refuel bridge walkway.
U1R18 Outage Report
U2R18 Radiation Protection Status Reports (Daily dose status of various ALARA Plans): 10/23/12, 10/24/12, 10/25/12, 11/3/12, and 12/3/12
2012-2016 SQN Business Plan (Includes Collective Radiation Exposure Target, Gap Analysis, and exposure reduction initiatives)

<u>CAP Documents</u>
Assessment: SQN-RP-S-11-96, Evaluation of RP Procedures against INPO 05-008 Revision 1, 3/29 through 4/8/11
Assessment: SQN-RP-F-12-01, INPO SOER 01-1, Unplanned Radiation Exposures, 6/18 through 6/27/12
Assessment: SQN-RP-S-12-44, Occupational ALARA Planning and Controls, 6/9/12 through 6/15/12PER 215405, FY2010 Dose Goals are not Reflective of the Actual Work Being Performed
PER 389287
PER 519286
PER 519292
PER 519303
PER 568255
PER 588365
PER 597354
PER 600909
PER 648588
PER 650723

<u>Section 2RS3: In-Plant Airborne Radioactivity Control And Mitigation</u>

<u>Procedures and Guidance Documents</u>
UFSAR Chapter 11& 12
0-PI-FPU-049-401.M, Self Contained Breathing Apparatus, Revision 29
0-PI-RCI-033-001.0, Periodic Monitoring of Service Air System for Use as Breathing Air, Rev 8
0-PI-FPU-049-401.Y, Testing and Overhaul of MSA SCBA Regulators, Rev 3
RC1-04.02, Cleaning/Sanitizing, Maintenance, Inspection, Storage and Inventory or Respiratory Protection Devices, Rev 4
NPG-SPP-05.10, Radiological Respiratory Protection Program, Rev 1

<u>Records and Data Reviewed</u>
SCBA Regulator Tracking Matrix, SERTA updated 10/23/12
Surveillance Task Sheet WO #113235685, Inspections of MSA SCBAs 9/22/12
Posi3 USB Test results for cylinders: AH052- 8/09/12, AH045- 8/10/12, and AH035- 8/09/12
Radiation Protection White Paper U2R18 Steam Generator Replacement Sequoyah Nuclear Plant

Personnel Contamination Log, 1/2012-10/2012
MSA Certification Records, Current
Breathing Air System Air Quality Analysis certifications for specified kits
Surveillance Task Sheet WO# 112072807, Main control room Intake Train B Rad Mon 0-R-90-126CC, 5/8/12
Surveillance Task Sheet WO# 111595373, Fuel Pool Area Rad Mon (Train A) 0-R-90-102 CC, 7/19/11
09-771276-000, CAL PM, SQN-0-PDI-030-0481 Aux BLDG Fuel Handling EXH HEPA Filter Diff Press, 9-8-10

Corrective Action Program (CAP) Documents
PER 324157
PER 403860
PER 599748
PER 604476

Section 2RS4: Occupational Dose Assessment
Procedures and Guidance Documents
RCDP-10 Personnel Contamination Reporting, Rev 05
NPG-SPP-05.1.1 Alpha Radiation Monitoring Program, Rev0
RCTP-106 Special Dosimetry Operations, Rev1
L17 110928 801 TVA QA-NPG-SQN-TEDS Site Audit Report, 8/28/11
NPG-SPP-05-1, Radiological Controls, Rev 02
RCI-209, Radiological Surveys of Personnel Leaving the RCA or Protected Area, Rev 01
RCI-14 Radiation Work Permit Program, Rev 053
White Paper, Passive Detection of Internally Deposited Radioactivity, 09/15/2010

Records and Data Reviewed
Whole body count Investigation 03/09/12
Radiological Incident Report 3/2/12
Radiological Incident Report 3/25/12
Radiological Incident Report 3/8/12
U1R18 Outage Personnel Contamination Trend Report, 5/7/12
ALARA Plan 2012-011
NVLAP Accreditation Certificates 4/01/2012-03/31/2013
Occupational Dose Assessment Snapshot Self-Assessment Report 8/6/12-8/9/12
RCI-14 Att 4 TEDE ALARA Worksheet, RWP 12038032 U2 SGR Lower CTMT Empty Contaminated Vacuums, 10/25/12

Corrective Action Program (CAP) Documents
PER 447959
PER 404383
PER 387248
PER 501249
PER 447959
PER 459806
PER 471313
PER 276155

Section RS5: Radiation Monitoring Instrumentation

Procedures and Guidance Documents

RCDP-8, Radiological Instrumentation/ Equipment Controls, Rev. 4
NPG-SPP-06.7, Instrumentation Setpoint, Scaling and Calibration Program, Rev.1
0-PI-CEM-016-001.4, Shield Building Radiation Monitor Sampling Methods, Rev. 30
1-PI-CEM-043-487.0, Sentry Post Accident Sampling System Operability Verification and Calibration, Rev.25
RCI-5.100, Operation of Laboratory Counter/Scalers, Rev. 4
RCI-5.101, Calibration and System Reliability of Laboratory Counters/Scalers, Rev.3
RCI-05-301, Operational Checks for Portal Monitors, Rev. 6
RCI-05.303, Calibration, Response Check and Operation of the Thermo Electron Small Article Monitor (SAM-11), Rev. 4
RCI-05.305, Calibration, Response Check, And Operation of the Canberra ARGOS-5AB Personnel Contamination Monitor, Rev.6
RCI-05.306, Calibration, Response Check, and Operation of the Canberra Cronos-11 Contamination Monitor, Rev.1
RCI-05.401, Instrument Response Checks \Utilizing the Shepherd Calibrator, Rev.3

Records and Data Reviewed

0-SI-ICC-090-101.B, Calibration of Auxiliary Building Gaseous Radiation Monitor 0-R-90-101B and Exhaust Vent Flow Monitor, 9/14/09 and 5/5/11
0-SI-ICC-090-118.0, Channel Calibration of Waste Disposal Systems Gas Effluent Radiation Monitor 0-R-90-118, 6/26/09, 4/15/11 and 6/7/12
0-SI-ICC-090-122.0, Channel Calibration of Waste Disposal Systems Liquid Effluent Radiation Monitor 0-R-90-122, 12/16/08 and 9/29/10
0-SI-ICC-090-212.0, Calibration Station Sump Effluent Radiation Monitor 0-R-90-212-CC, 1/4/10 and 4/7/11
1-SI-ICC-090-106.0, Channel Calibration of Containment Building Lower Compartment Air Monitor 1-R-90-106, 9/22/10 and 5/9/12
1-SI-ICC-090-112.0, Channel Calibration of Containment Building Upper Compartment Air Monitor, 1-R-90-112, 4/29/10 and 7/6/11
1-SO-ICC-090-273.0, Lower Containment Post Accident High Range Area Monitor 1-R-90-273, 10/7/10 and 3/15/12
Results of Radiochemistry Cross Check Program Sequoyah Nuclear Plant 1st Qtr 2010, 4th Qtr 2011 and 1st Quarter 2012
Quality Control Chart for Liquid Scintillation Counter 60450, 6/22/12 through 11/19/12
Quality Control Charts for ADC #3 and ADC#4, 6/22/12 through 11/19/12
Calibration of the Western Area Radiological Laboratory (WARL) High Level Cs-137 Source Range (used to calibrate survey instrumentation for the various TVA sites), 9/6/12

Corrective Action Program (CAP) Documents

Assessment SQN-RP-S-11-40, Radiation Monitoring, 3/28/11 to 4/1/11
Assessment SQN-RP-S-12-48, Radiation Monitoring Instrumentation, 9/1-24/12
Self Assessment CRP-TPR-F-10-002, External Measurement Quality Assurance Results for Portable Radiation Survey Instrumentation (WARL), 7/14/11
PER 475776
PER 476639
PER 480583

PER 497751
PER 529444
PER 530812
PER 550011
PER 593379
PER 630047

Section 4OA1: Performance Indicator Verification
Records and Data Reviewed
Individual Employee Access Records with Exit Transactions greater than 100 mrem, 6/1/2011-11/6/2012
HIS-20, Dose Rate Alarm Data
0-SI-CEM-030-415.0, Gas Radwaste Release Permits (Monthly) 1/3/12, 5/4/12, and 8/3/12
0-SI-CEM-077-400.1, Liquid Waste Effluent Batch Release Permits 9/21/11, 1/2/12, and 7/7/12

PERS Documents
PER 379547
PER 497135
PER 589561
PER 578611

Section 4OA2: Identification and Resolution of Problems
Procedures
NPG-SPP-03.1, Corrective Action Program, Revision 1

Work Orders
113707839, Breakers 22, 23, and 24 in VIPB 2-II not Fitting Up Properly
111632799, Rework and Adjust Breaker Mounting Alignment Issues on VIPB 2-II
08-776288-000, Replace Spring Pack and Limiter Plate for SQN-1-MVOP-063-0156-A
111463339, FCV-63-156 Replace Spring Pack to Satisfy Limitorque Requirements
111752332, FCV-63-156-A Perform AF/AL MOVATs and Reduce TS Setting
113218662, Drill Out Broken Bolt in Spring Pack Cover in Operator, Ref. SR 506257, NPG-SPP-06.6 May Apply

Drawings
1,2-45N706 (Series), Wiring Diagram 120VAC Vital Instrument Power Boards

PERs
286156, Vital Instrument Power Board Breaker Mounting Issue (Lower Tier)
264271, Breaker Mounting in 120VAC Vital Instrument Power Boards
506338, Broken Bolt Discovered in MOV Housing During MOVATs Testing LOWER TIER, 2/15/2012
506995, WO 113218662, Potential Rework Per NPG-SPP-06.6, 2/15/2012
518423, Dissimilar Limit Cover Bolting, 3/8/2012

Other documents
PER 264271 Functional Evaluation, through Rev. 1 (which Incorporates PER 286156)

<u>Section 4OA5: Other Activities</u>

<u>SGT/ AREVA Documents</u>

<u>Calculations</u>

39866-CALC-C-009, Evaluation of Safety-Related Buried Commodities in the Vicinity of Heavy Lift Load Path for Postulated Load Drop from the OLS, Revision 0
39866-CALC-C-013, Evaluation of Interim Configuration of Unit 2 Steam Generator Compartments during SGRO, Revision 0
39866-CALC-C-015, Evaluation of Shield Building Dome for Core-Boring and Temporary Sealing during Power Operation, Revision 0

<u>Drawings</u>

39866-EP-C-004-006, OLS Arrangement and Rigging, Revision 1
39866-EP-C-008-001, Liebherr LR 1400/1 Crane Location and Safe Usage, Revision 4
2-2-H01-0382-01, Mechanical Main Steam Pipe Supports, Rev. 0
DCA D22482-1304-CN, OSG 2-3 Main Steam Piping Removal – Temporary Removal of 2-MSH-382, Rev. 3

<u>Engineering Change Requests (ECRs)</u>

ECR 002, Allow the Use of Plywood to Achieve Level Surface of the Concrete Piers, Revision 0
ECR 003, Incorporate Changes Requested by Construction, Revision 0
ECR 005, Incorporate Changes Requested by Construction – Shield Building Hatch Plates, Revision 0
ECR 006, N-1 Mobile Crane Changed to Grove RT890E, Revision 0
ECR 007, N-1 Mobile Crane Changed to Grove RT890E, Revision 0
ECR 009, Update DCA D22406-015 and DCA D22406-016, Revision 0
ECR 012, Evaluate 6” PVC Pipe within OLS Erection Area and Update OLS Arrangement and Rigging Drawing, Revision 1
ECR 013, Revise Power Supply for OLS and Mammoet Facility, Revision 1
ECR 014, Revise Drawings 39866-EP-C-004-001 Sheets 1 & 2, Revision 1
ECR 015, Update Drawing 39866-EP-C-006 to Include Evaluation of Previously Unidentified Commodities and Other Minor Changes, Revision 1
ECR 018, Construction Elevator Upgrades, Utility Tower Installation, and Fill Structure Removal, Revision 1
ECR 019, Enclosure Roof Plug Cut Modification Cut Angle Correction, Revision 1
ECR 020, Adding Reinforcement to the SCV Cutouts and Connection Restrictions to the Superstructure, Revision 1
ECR 024, Revise Source Breaker for SGR Temporary Power, Revision 1
ECR 026, Crawler Crane Substitution, Revision 1
ECR 027, Adding Second Scuttle Hole and Clarifications to the Formwork Drawing, Revision 1
ECR 028, Truss Frame and Anchor Bolts, Revision 1
ECR 029, Elevator Anchorage Modification, Revision 1
ECR 030, Installation of the Temporary SCV Supports and Plate Adjustment Hardware, Revision 2
ECR 031, Additional Facilities and CAF HPFP System Connection Revision, Revision 1
ECR 034, Weld Length Change on Drawing 39866-EP-C-002, Revision 1
ECR 036, SG Downending Device with Bumper Blocks Installed, Revision 1
ECR 037, Core-Drilling, Hydro-Demolition Initiation Modes Upgrade, Revision 2
ECR 039, Haul Route Load Drop Protection, Revision 1

ECR 040, Fan Room Access Provisions, Revision 2
ECR 042, Update Mammoet Drawings and RSG Insulation Weight, Revision 2
ECR 043, Revise Site Layout Plan Temporary Facilities Sketch, Revision 2
ECR 044, LR1400 Load Handling Areas, Revision 2
ECR 045, Revision to the Installation of the Temporary SCV Supports and Plate Adjustment Hardware and Other Miscellaneous Changes, Revision 2
ECR 046, Anchor Bolts for the Debris/Water Collection System to be Installed on the Shield Building Dome, Revision 2
ECR 048, SCV Stiffener Weld Re-Design, Revision 2
ECR 049, Superstructure Connection, Revision 2
ECR 050, Update Mammoet OLS Assembly Drawings, Revision 2
ECR 051, SCV Stiffener Weld Re-Design, Revision 2
ECR 052, SG Enclosure Roof Cut Concrete Temporary Cribbing on New Location, Revision 2
ECR 053, Superstructure Truss T1 Connection, Revision 2
ECR 054, SCV Stiffener Weld Re-Design, Revision 2
ECR 055, Rigging Frame Bearing Plate, Revision 2
ECR 056, Hoop Stiffener Notch, Revision 2
ECR 058, SCV Stiffeners, Revision 2
ECR 059, Revision to the Installation of the Temporary SCV Supports, Revision 2
ECR 060, Knife Plate for SCV Gusset, Revision 2
ECR 062, Revise Site Layout Plan Temporary Facilities Sketch, Revision 2
ECR 063, SCV Rigging Lug Weld Detail, Revision 2
ECR 064, SCV Rigging Plates, Revision 2
ECR 065, 50.59 Evaluation Revised to Revision 2, Revision 2
ECR 067, SCV Rigging Lug Weld Detail, Revision 2

Engineering Packages (EPs)
39866-EP-C-001, Temporary Structures and Commodities for Reactor Building Modifications, Revision 0
39866-EP-C-002, Temporary Structures and Commodities for the Removal and Restoration of the Shield Building Concrete Dome and Steel Containment Vessel, Revision 0
39866-EP-C-004, Rigging and Transport, Revision 0
39866-EP-C-005, Temporary Facilities, Revision 1
39866-EP-C-006, Haul Route and RSG Offload, Revision 2
39866-EP-P-007, Temporary RCS and RG Supports, Revision 0
39866-EP-C-008, Supplemental Crane, Revision 0
39866-EP-C-009, Fan Room Access Provisions, Revision 0

Procedures
39866-SPEC-C-004, Ready-Mix Concrete, Revision 3
39866-SPEC-C-006, Purchase of Bar-Lock Rebar Couplers, Revision 0
MD 16.01, Issues Identification and Resolution Program and Lessons Learned Program, Rev. 4
QEP 11.01, Work Packages, Rev. 8

Temporary Alteration Control Forms (TACFs)
TACF 2-10-015-303, Polar Crane Breaker 2-BKRB-303-DQ/4C1
TACF 2-11-003-410, Temporary Defeat of Interlocks for Doors A122 and A123, Revision 1
TACF 2-11-004-410, Temporary ABSCE Door Installation, Operation, and Removal, Revision 5

Work Packages
WP 1032 (WO 112452000), Install Outside Lift System (OLS)
WP 2095 (WO 112498042), Haul Route Upgrades, Temporary Storage to Containment
WP 2580A (WO 112455790), Transport OSG 1 from Downending Area to OSGSF
WP 3080C-1 (WO 112800988) Add Modification for 2-MSH-382

Other Documents
AREVA Document 32-9129996-000, Original and Replacement SG Comparison for Sequoyah Unit 1 and Unit 2, dated November 19, 2010
AREVA Report 77-9142036-000, Replacement Steam Generator Report for Tennessee Valley Authority (TVA) Sequoyah Unit 2, Revision 0
RDR/10-09-12/02, Rebar Receipt Report, dated 10 October 2012
Report of Concrete Compressive Strength Testing of Mix Design by University of Tennessee at Chattanooga Civil Engineering Materials Research Laboratory, dated May 1, 2012.
S&ME Certified Materials Test Report for No. 8 Mechanical Rebar Sister Splices, dated November 28, 2012
S&ME Concrete Testing and Inspection Technician Personnel Qualifications
S&ME Certified Materials Test Report for Materials Used to Make Structural Concrete for Shield Building Restoration, July 20, 2012.
SGT Letter to TVA with respect to Concrete Test Batching for Shield Building, Revision 1, dated September 11, 2012
Technical Report SQN2-SGR-TR1, SG Rigging and Handling, Revision 3
Technical Report SQN2-SGR-TR3, Sequoyah Unit 2 Steam Generator Replacement Alternate Rebar Splice – Bar-Lock Mechanical Splices, Revision 01

TVA Documents
Calculations
CDQ0026612012000011, Steam Generator Enclosure Roof Modifications, Revision 1
SCG-1-40, Reactor Building Steam Generator Compartment, Final Design, Revision 5
SCG-1S-803, Evaluation of Shield Building Dome Access Openings, Revision 4
SCG-1S-805, Steam Generator Enclosure Modifications, Revision 0
SCG-1S-806, Steam Generator Enclosure Modifications – Design of Roof Support Frames, Revision 1
SCG-1S-807, SQN Unit 2 Shield Building Dome Analysis, Revision 0
SCG-1S-811, Unit 2 Shield Building Dome – Permanent Formplate, Revision 1
SCG-1S-815, Feedwater Elbow Rigging Point Evaluation, Revision 0
SCG-1S-817, Shield Building Dome Construction Platform Anchorage, Revision 0
SQNAPS2-002, SQN Containment Structure Heat Sinks, Revision 0
SQN-MEB-HE-002, Kevlar Slings for the Reactor Building Plugs and Missile Shields, Revision 0

Design Change Notices (DCNs)
D-20673A, Containment Structural Modifications
D-22406A, Modifications to Shield Building Roof, Annulus Ladder and Steel Containment Building Supporting Steam Generator Replacement
D-22437A, Modify Steam Generator Manway Platforms to Improve Accessibility for RSGs
D-22460A, Install Reflective Metal Insulation on the RSGs and Portions of the Associated MS, FW, Blowdown and Reactor Coolant Piping Systems which Interface with the RSGs

D-22470A, Install Foundation for the Outside Lift System (OLS) Crane which Will Be Used to Support SGR
D-22471A, Implement Reactor Building Structural Modifications Required to Support SGR
D-22477A, Temporary Interference Removal for SGR - Electrical
D-22478A, SG Vessel Replacement Modification
D-22479A, SG and RCS Support Modification
D-22480A, Temporary SGR Interference Removal (Civil and Mechanical)
D-22481A, Small Bore Piping Modification
D-22482A, Large Bore Piping Modification
D-22502A, Replace Main Steam Flow Transmitters Due to Unit 2 SGR
D-22507A, Modify the SG Level Transmitters As Needed to Support SG Replacement
D-22508A, Modify No. 7 Heater Drain Tank Pumps and Changes Due to RSG
D-22515A, Construct SQN Unit 2 OSGSF
D-22516A, Modify Spare Penetration for SGR Temp Power Feed
D-22571A, Provide Connection for Electrical, Fire Protection and Fire Detection between a New Decontamination Facility Building and Existing Plant Systems
D-22585A, Modify Pipe Supports on RCL Branch Lines Due to Reanalysis Resulting from the RSG
D-22598A, Added Additional Equipment for Security Concerns
D-22731A, Modify the RCS Flow Transmitters As Needed to Support RSG and Rx Fuel Change

<u>Drawings</u>
1, 2-44W269, Plug, Gate and Missile Shield Handling with Kevlar Slings – Arrangement and Details Sheet 1, Revision 1
41N718-1, Concrete Dome Outline, Revision 5
41N719-1, Concrete Dome Reinforcement, Revision 2
41N719-2, Concrete Dome Reinforcement, Revision 1

<u>Procedures</u>
CTP-DCS-300.4, HI-STORM and HI-TRAC Site Transportation, Revision 13
Special Test Instruction 2-STI-088-156.0, Primary Containment Vessel Post-Modification Pressure/New Weld Leakage Inspection Test. Revision 0
TVA Safety Procedure 802, Requirements for the Safe Operation of Cranes, Revision 9
Specifications
General Engineering Specification G-34, Requirements for Repair of Concrete during Construction, Modification, and Maintenance, Revision 7

<u>Other Documents</u>
10 CFR 72.48 Evaluation of CTP-DCS-300.4, Revision 10, to allow relocating loaded HI-STORMs from one location to another on the ISFSI pad.
50.59 Evaluation for DCN D22471A, Documents the Basis for Change to the Number of Shims Proposed to be Installed per Topical Report 24370-TR-C-003-A, Revision 4
Bechtel Calculation 24370-C-022, Steam Generator Post-Impact Response from Postulated Drop Above Containment, Revision 0, dated March 21, 2003
Design Standard DS-C1.7.1, General Anchorage to Concrete, Revision 11
Engineering Package No 39866-EP-006, Rev 2, Haul Route and RSG Offload
HOLTEC Response to Request for Technical Information (RRTI) 1058-006R0, Moving Cask on Pad

National Ready Mixed Concrete Association (NRMCA) Certificate of Conformance for Concrete Production Facilities for Sequatchie Concrete Service, Inc., Plant ID # 834072, Certification # 13744, expiration November 10, 2013.
NRMCA Fleet Inspection Reporting Form- Truck Mixers, Inspection Record of Delivery Fleet
NPG-SPP-09.6, Master Equipment List (MEL), effective date 07-20-2012
Personnel training documents for Liebherr Tower Crane
PER 655762, Enclosure Plugs 2-1 and 2-4 Sides
PIC P23152, Sheet 6, SG Enclosure Roof Support Frame – Shim & Bearing Plate Details
Relief Request 2-APPJ-1, Request to Employ Alternative Testing to IWE-5221 Requirements in ASME Boiler and Pressure Vessel Code, Section XI (2001 Edition with 2003 Addenda)
Report 77-9142036-000, Replacement Steam Generator Report for Tennessee Valley Authority (TVA) Sequoyah Unit 2, dated September 2011.
SEQ High Hazard Lift Plan, dated 7/31/12
SQN2-SGR-TR1, Rev 3, Sequoyah Unit 2 SGR Rigging and Heavy Load Handling Technical Report
Technical Instruction 0-TI-SXX-000-016.0, Breaching the Shield Building. ABSCE, or Control Room Boundaries, Revision 27

Reference Documents

American Concrete Institute (ACI) Standards
ACI 117-10, Standard Specification for Tolerances for Concrete for Tolerances for Concrete Construction and Materials
ACI 211.1-91, Standard Practice for Selecting Proportions for Normal, Heavyweight, and Mass Concrete
ACI 301-10, Specifications for Structural Concrete
ACI 304R-00, Guide for Measuring, Mixing, Transporting, and Placing Concrete
ACI 306R-10, Guide to Cold Weather Concreting
ACI 308-01, Guide to Curing Concrete
ACI 309R-05, Guide for Consolidation of Concrete
ACI 318-11, Building Code Requirements for Reinforced Concrete
ACI 347R-04, Guide to Formwork for Concrete
ACI 349-01, Code Requirements for Nuclear Safety-Related Concrete Structures

American Society for the Testing of Materials (ASTM)
C31-09, Standard Practice for Making and Curing Concrete Test Specimens in the Field
C39-12, Standard Test Method for Compressive Strength of Cylindrical Concrete Specimens
C94-09, Standard Specification for Ready-Mixed Concrete
C138-10, Standard Test Method for Density (Unit Weight), Yield, and Air Content (Gravimetric) of Concrete
C143-05, Standard Test Method for Slump of Hydraulic-Cement Concrete
C172-07, Standard Practice for Sampling Freshly Mixed Concrete

American Society of Mechanical Engineers (ASME)
Boiler and Pressure Vessel Code, 2001 Edition, 2003 Addenda, Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components

American National Standard Institute (ANSI) Documents
ANSI N45.2.5 – 1974, Supplementary Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants
ANSI N45.2.11 – 1974, Quality Assurance Requirements for the Design of Nuclear Power Plants

Integrated Report Documents

Procedures
CTP-SWD-100, Seismic Walkdowns EPRI 1025286, Revision 0

PERs
579192, 1B Diesel air tank engine 1 Bolting on the base is not fully seated
584475, cover loose on conduit housing
581395, D/G Battery corrosion
584472, L-B ERCW Pump
585201, Vital Battery Room wall crack
587186, Battery Room wall missing bolt
587046, P-B ERCW Pump seal leak
588145, 1A 480V Shut Down Bd. Chiller
588588, 2-FCV-61-110: Dust cover for valve
590084, Temporary cooling fans not seismically secured
590172, 2B MG Set missing shroud bolt
590174, Flexible Conduit above Hydrogen Recombiner Is pulled out at both ends from solid conduit
590175, Missing an anchor bolt on EBGTS conduit support above 2B MG Set Room
583600, Trapeze hanger on DG
591742, 2B CCP flex conduit pulled loose
592319, 2B Aux Feed Water Pump, seal leakoff line
592842, 2A CCS Pump flex conduit
592845, B SD Board Chiller Conduit panel screws
592851, Loose wire on light next to B SD Board Chiller
596884, Scaffolding not tied down and anchored properly in 6.9kV Shutdown Board Rooms
647403, First Aid: Security Officer Knocked to the Ground While Closing Gate 56
643432, Unattended Rod Room El 734

Work Orders
113507183, 0-PI-SQS-000-676.W Protected and Vital Area Weekly Inspection

Other documents
NPG-SPP-05.8, Special Nuclear Material Control, Rev. 3
0-PI-NUC-000-004.0, Annual Special Nuclear Material Inventories (Formerly 0-TI-NUC-000-004.0, Rev. 5), Rev. 0
TI-45, Physical Verification of Core Load Prior to Vessel Closure, Rev. 29
CTP-DCS-300.1, Spent Fuel Cask Loading Verification (Formerly 0-SF-DCS-079-003.1, Rev. 4), Rev 1
Annual SNM Inventory from Kathryn Allen to James Keck, dated April 2012

0-PI-SQS-000-676.W, Protected and Vital Area Weekly Inspection, dated 12/13/2012
NPG-SPP-03.21, Attachment 1, Fatigue Management and Work Hour Limits Evaluation Associated with PER 647403, dated 11/26/2012
TASK 25-27C, Security Annual Tactical Qualification, Rev. 20

Section 4OA7: Licensee-Identified Violations
Procedures
0-SI-FPU-013-630.0, Fire Detection Panel 0-L-630 Test, Revision 7
NPG-SPP-18.4.6, Control of Fire Protection Impairments, Revision 0
SPP-10.9-1, Control of Fire Protection Impairments, Date Issued 1/29/1999

Work Orders
111463725, Implement Limitorque Maintenance Update 88-2 and 90-1 to Reduce The Potential for Hydraulic Lock
114022327, Replace MVOP Actuator Housing Screws

PERs
485817, Incorrect Fire Operations Impairment
655762, NCR 1163 against WP-3575A/D Failure to Install Shims for Enclosure Plugs 2-1 and 2-4 Sides, 12/13/2012
506338, Broken Bolt Discovered In MOV Housing During MOVATs Testing, 2/15/2012
516521, Missing Limit Switch Cover Bolts on 1-MVOP-72-22, 3/5/2012
518423, Dissimilar Limit Cover Bolting Found During EOC for PER 506338, 3/8/2012
518572, Actuator Mounting Bolt Thread Engagement Deficiency 1-MVOP-70-133, 3/8/2012
522640, Actuator Yoke Bolt (One Only) Engagement of About 2 Threads, 3/16/2012
523741, Bolt Found Too Long in Limit Cover, 3/19/2012
622076, As Found Mounting Bolts Do Not Match Drawing, 10/11/2012
623383, Wrong Bolts Were Found in 2-FCV-72-34 Limit Switch Housing Cover, 10/13/2012
624265, Wrong Bolts Found in 2-FCV-72-34, 10/15/2012
624373, Questionable Bolts Installed in 2-FCV-72-13, 10/15/2012
626982, Review MOV Maintenance Practices, 10/19/2012
627418, Actuator to Valve Bolting Material for 2-MVOP-003-0116B, 10/20/2012
631981, Actuator to Valve Bolting for 2-MVOP-062-0132, 10/27/2012
644661, Procedure and Training Deficiency Leads to Installation of Wrong Bolting on Safety-Related MOVs, 11/19/2012

Other documents
SGT Nonconformance Report 1163, Enclosure plugs 2-1 and 2-4 sides
SGT Deficiency Report 42, Installation of Enclosure Plugs
Safety Evaluation of TVA Topical Report 24370-TR-C-003-A
Technical Report No. SQN2-SGR-TR2
SQN Fire Protection Report Part II, Limiting Condition for Operation (LCOs):
3.3.3.8.a.1 – Fire Detection Instrumentation
3.7.11.2.a.1 – Spray and/or Sprinkler Systems
Apparent Cause Evaluation for PER 506338, 3/10/2012
Apparent Cause Evaluation for PER 626982, 11/16/2012