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

August 31, 2016

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

SUBJECT: SEQUOYAH NUCLEAR PLANT UNIT 1– NRC SUPPLEMENTAL INSPECTION REPORT 05000327/2016008

Dear Mr. Shea:

On October 21, 2016, your staff reported an Initialing Events performance indicator that crossed a threshold from green to white. Based on your report, the NRC assigned a White performance indicator Action Matrix input to the Initiating Events cornerstone in the third quarter of 2015. In response to this Action Matrix input, the NRC informed you that a supplemental inspection under Inspection Procedure 95001, "Supplemental Inspection for One or Two White Inputs in a Strategic Performance Area," would be required.

On March 24, 2016, you informed the NRC that Sequoyah was ready for the supplemental inspection. On July 15, 2016, the NRC completed the supplemental inspection and discussed the results of this inspection with Mr. Schwarz and other members of your staff. The inspector documented the results of this inspection in the enclosed inspection report.

The NRC performed this supplemental inspection to determine if (1) the root and contributing causes for the significant issues were understood, (2) the extent of condition and extent of cause for the identified issues were understood, and (3) your completed or planned corrective actions were sufficient to address and prevent repetition of the root and contributing causes.

The inspector reviewed the root cause evaluation for the White performance indicator which concluded the causes and contributors from each of the individual inputs to the performance indicator involved sufficiently diverse aspects to not result in a single underlying common cause. The inspector determined that although there was no indication of a singular root cause for the events as a whole, the licensee performed individual cause analysis for each event which identified causes and corrective actions specific to each event. Furthermore, additional analysis was performed which identified consolidated causes that had applicability to multiple events and developed corrective actions to address these additional issues. The inspector determined that completed or planned corrective actions were sufficient to address the performance that led to the White performance indicator.

Although the 95001 Inspection has been completed successfully, because the PI for trips in 7000 critical hours has not returned to Green, in accordance with the guidance in Inspection Manual Chapter (IMC) 0305, "Operating Reactor Assessment Program," the NRC will continue to consider the White PI in assessing the plant performance.

NRC inspector documented one finding of very low safety significance (Green) in this report. This finding involved a violation of NRC requirements. Further, the inspector documented a licensee-identified violation which was determined to be of very low safety significance in this report. The NRC is treating these violations as non-cited violations (NCVs) consistent with Section 2.3.2.a of the Enforcement Policy.

If you contest the violations, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001; with copies to the Regional Administrator, Region II; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC resident inspector at the Sequoyah Nuclear Plant.

In accordance with Title 10 of the Code of Federal Regulations 2.390, "Public Inspections, Exemptions, Requests for Withholding," 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's Public Document Room or from the Publicly Available Records (PARS) component of the NRC's Agencywide Documents Access and Management System (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room)..

Sincerely,

/RA Daniel Rich Acting for/

Alan Blamey, Chief
Reactor Projects Branch 6
Division of Reactor Projects

Docket Nos. 50-327
License Nos. DPR-77

Enclosure: Inspection Report 05000327/2016008
w/Attachment: Supplemental Information

cc w/ encl. (See next page)

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☒ PUBLICLY AVAILABLE ☐ NON-PUBLICLY AVAILABLE ☐ SENSITIVE ☒ NON-SENSITIVE

ADAMS: ☒ Yes ACCESSION NUMBER:___ML16244A370________ ☒ SUNSI REVIEW COMPLETE ☒ FORM 665 ATTACHED

OFFICE	RII:DRP	RII:DRP					
SIGNATURE	CRK1	AJB3					
NAME	C. Kontz	A. Blamey					
DATE	8/9/2016	8/30/2016	9/ /2016	9/ /2016	9/ /2016	9/ /2016	9/ /2016
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\2016\008\SQ 2016008 95001.DOCX

U.S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket No.:	50-327
License No.:	DPR-77
Report No:	05000327/2016008
Licensee:	Tennessee Valley Authority (TVA)
Facility:	Sequoyah Nuclear Plant, Unit 1
Location:	Sequoyah Access Road Soddy-Daisy, TN 37379
Dates:	March 11-15, 2016
Inspectors:	C. Kontz, Senior Project Engineer
Approved by:	Alan Blamey, Chief Reactor Projects Branch 6 Division of Reactor Projects

SUMMARY

Inspection Report 05000327/2016008; 7/11/2016 – 7/15/2016; Sequoyah Nuclear Plant, Unit 1; Supplemental Inspection for One or Two White Inputs in a Strategic Performance Area, Event Followup

The report covered a one week period of inspection by a region-based inspector. Two violations 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 April 29, 2015. All violations of NRC requirements are dispositioned in accordance with the NRC's Enforcement Policy dated February 4, 2015. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 6.

Based on the results of this inspection, no significant weaknesses were identified. The inspectors concluded that, in general, the licensee properly determined: who and under what conditions issues were identified, how long issues existed and prior opportunities for identification, the plant risk specific consequences as applicable, and compliance concerns associated with issues. The inspectors also concluded that, in general, the licensee properly: evaluated issues using a systematic methodology to identify the root and contributing cause; conducted root cause evaluations to a level of detail commensurate with the significance of the problem; considered prior occurrences of the problem and of prior operating experience in root cause evaluations; addressed the extent of condition and extent of cause of the problem in root cause evaluations; and considered the safety culture components as described in IMC 0305 in root cause, extent of condition, and extent of cause evaluations. The inspectors also concluded that, in general, the licensee properly: specified appropriate corrective actions for each root and contributing cause or properly evaluated why no corrective actions are necessary, prioritized corrective actions with consideration of the risk significance and regulatory compliance, established a schedule for implementing and completing the corrective actions, and developed quantitative or qualitative measures of success in determining the effectiveness of the corrective actions to prevent recurrence.

Although the 95001 Inspection has been completed successfully, because the PI for trips in 7000 critical hours has not returned to Green, in accordance with the guidance in Inspection Manual Chapter (IMC) 0305, "Operating Reactor Assessment Program," the NRC will continue to consider the White PI in assessing the plant performance. The NRC will review the implementation and effectiveness of the licensee's corrective actions during future inspections.

A. NRC Identified and Self-Revealing Findings

Cornerstone: Initiating Events

Green. A self-revealing non-cited violation of Title 10 of the Code of Federal Regulations (10 CFR) 50, Appendix B, Criterion III, “Design Control”, was identified for the licensee’s failure to maintain design control of MSIV 1-FCV-1-22 controls. Specifically, inadequate design controls led to an under tightened electrical connection following replacement of the Unit 1 MSIV Control Room handswitch (1-HS-1-22A). The licensee entered the condition into the CAP as CR 1107656 and has corrected the loose wiring connection on 1-HS-1-22A.

The finding was determined to be more than minor because it adversely impacted the Design Control attribute of the initiating events 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. The inspectors reviewed IMC 0609, Attachment 4 and determined that the finding was of very low safety significance (Green) because it did not contribute to both the likelihood of a reactor trip and the likelihood that mitigating systems will not be available. No cross-cutting aspect was assigned to this finding because the performance deficiency is not reflective of current licensee performance. (Section 4OA3.4)

B. Licensee Identified Violations

Violations of very low safety or security significance or 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 corrective action program. These violations and corrective action tracking numbers are listed in Section 4OA7 of this report

REPORT DETAILS

4. OTHER ACTIVITIES

4OA3 Event Followup

.1 (Closed) Licensee Event Report (LER) 05000327/2015-001-00 and -01: Automatic Reactor Trip due to Negative Rate Trip as a Result of a Dropped Control Rod

a. Inspection Scope

On March 11, 2015 at 06:21 Eastern Daylight Time (EDT), SQN Unit 1 reactor automatically tripped due to a Negative Rate Trip as a result of Control Bank D Control Rod H-8 [EIIS Code AA] dropping into the core. Investigation revealed Control Rod H-8 dropped into the core approximately one second before the reactor trip. Control Rod H-8 is located in the center of the core and is one of nine control rods in control bank D. The dropped control rod caused a rapid decrease in power which was sensed by all four nuclear instrumentation system (NIS) power range channels. The reactor trip logic is two out of four channels.

Following initial LER submission the Licensee submitted a revision to reflect the results of the root cause analysis along with corrective actions to prevent recurrence. The inspector reviewed the LERs and related documents regarding the accuracy of the LERs, appropriateness of corrective actions, violations of requirements, and generic issues. The inspector verified that the event follow-up aspects of the event were previously inspected and documented in section 4OA3.2 Event Follow-up of IR 05000327/2015001 AND 05000328/2015001 (ML15124A270).

b. Findings

None - These LERs are closed.

.2 (Closed) Licensee Event Report (LER) 05000327/2015-002-00: Automatic Reactor Trips due to Improper Wire Termination in Main Generator Voltage Regulator Circuit

a. Inspection Scope

On July 24, 2015, at 1351 Eastern Daylight Time (EDT), Sequoyah Nuclear Plant (SQN) Unit 1 reactor automatically tripped following a turbine trip due to actuation of the generator backup relay. The relay actuation was a result of voltage variations on the main generator. A malfunctioning base adjuster follower card was identified on the main generator voltage regulator. After replacement of the base adjuster circuit card, Unit 1 was connected to the grid and ascending in power. On July 27, 2015, at 1040 EDT SQN Unit 1 reactor was at 82 percent power when it automatically tripped following a turbine trip due to actuation of the generator backup relay.

An improperly terminated wire was found in the voltage regulator logic drawer which caused high resistance. Prior to restart, the wire was properly connected. Following each of the reactor trips, all safety related equipment operated as designed, all control rods fully inserted as required, and auxiliary feedwater automatically initiated from the feedwater isolation signal as expected. The cause of each of the reactor trips was determined to be inadequate standards for multi-wire terminations and verification at the time of the original improper wire termination event in the mid-1990s.

The inspector reviewed the LER and related documents regarding the accuracy of the LER, appropriateness of corrective actions, violations of requirements, and generic issues. The inspector verified that the event follow-up aspects of the events were previously inspected and documented in section 4OA3.2/3, Event Follow-up of IR 05000327/2015003 AND 05000328/2015003 (ML15313A244).

b. Findings

None - This LER is closed.

.3 (Closed) Licensee Event Report (LER) 05000327/2015-003-00: Manual Reactor Trip due to Loss of Power to the Vital Inverter Power Board 1-11

a. Inspection Scope

On September 14, 2015, at 0426 Eastern Daylight Time, Sequoyah Nuclear Plant (SQN) Unit 1 reactor was manually tripped due to a loss of power to the Vital Instrument Power Board (VIPB) 1-11. Prior to the reactor trip, operators were in the process of realigning Vital Inverter 1-II for planned maintenance. During this evolution, VIPB 1-11 became de-energized. Operators entered Abnormal Operating Procedure AOP-P.03, "Loss of Unit 1 Vital Instrument Power Board" which required a manual reactor trip. Following the reactor trip, operators restored power to VIPB 1-II with the normal supply at 0550. All plant safety systems responded as designed, all control rods fully inserted, and auxiliary feedwater automatically initiated from the feedwater isolation signal as expected. It was determined that an Alternating Current (AC) output switch failed causing the loss of power to the VIPB 1-11. The direct cause of the switch failure was due to increased friction of bearing surfaces caused by lack of appropriate lubrication.

The inspector reviewed the LER and related documents regarding the accuracy of the LER, appropriateness of corrective actions, violations of requirements, and generic issues. The inspector verified that the event follow-up aspects of the event were previously inspected and documented in section 4OA3.4, Event Follow-up of IR 05000327/2015003 AND 05000328/2015003 (ML15313A244).

b. Findings

The enforcement aspects of this violation are discussed in Section 4OA7 in the LER closeout section under report Section 4OA3. This LER is closed.

.4 (Closed) Licensee Event Report (LER) 05000327/2015-004-00: Manual Reactor Trip due to Main Steam Isolation Valve Drifting in the Closed Direction

a. Inspection Scope

On November 23, 2015, at 0844 Eastern Standard Time, Sequoyah Nuclear Plant (SQN) Unit 1 reactor was manually tripped due to plant parameters indicating that the Loop 3 Main Steam Isolation Valve (MSIV) had started drifting in the closed direction. Prior to the reactor trip, the open light indication on the main control board for the Loop 3 MSIV was noted to be extinguished. The light bulb was replaced with no change in indication. At the same time, the Post Accident Monitoring panel indicator for the Loop 3 MSIV displayed full open; however, within two to three minutes, the panel provided dual indication. Subsequently, Operators noted that the reactor coolant system temperature and Loop 3 Steam Generator (SG) pressure were both rising, and the Loop 3 SG flow was lowering. These indications confirmed the Loop 3 MSIV was drifting closed.

Following the reactor trip, all plant safety systems operated as designed, all control rods fully inserted, and auxiliary feedwater automatically initiated from the feedwater isolation signal, as expected. Troubleshooting identified a loose termination associated with the Loop 3 MSIV handswitch that would result in a slow loss of air pressure and cause the MSIV to slowly drift in the closed direction. The direct cause was determined to be a loose electrical connection on the MSIV handswitch.

The inspector reviewed the LER and related documents regarding the accuracy of the LER, appropriateness of corrective actions, violations of requirements, and generic issues. The inspector verified that the event follow-up aspects of the event were previously inspected and documented in section 4OA3, Event Follow-up of IR 05000327/2015004 AND 05000328/2015004 (ML16043A261).

b. Findings

Introduction. A self-revealing Green non-cited violation of Title 10 of the Code of Federal Regulations (10 CFR) 50, Appendix B, Criterion III, “Design Control”, was identified for the licensee’s failure to maintain design control of MSIV 1-FCV-1-22 controls. Specifically, inadequate design controls led to under tightened electrical connection following replacement of the Unit 1 MSIV Control Room handswitch (1-HS-1-22A).

Description. On November 23, 2015, a manual reactor trip was initiated by operators as a result of a slowly closing loop # 3 main steam isolation valve. The event was reported to the NRC as event notification (EN) 51559, and documented in the licensee’s CAP as CR 1107656. The licensee’s root cause into the event identified the Unit 1 MSIV Control Room handswitch (1-HS-1-22A) having a loose connection (terminal lug and nut assembly - terminal E, wire 1B6) located in Panel 1-M-4 as the direct cases of 1-FCV-1-22A drifting shut and the manual reactor trip. In 1994, SQN electric shop technicians completed replacement of the MSIV 1-22 handswitch (1-HS-1-22A) under Work Order (WO) 93-000869-000. Review of WO 93- 000896-000 identified work planning and work steps were consistent with 1994 methodologies and requirements in that it provided references for fastener tightness.

The inspector reviewed the root cause analysis and identified since the handswitch replacement, the licensee had two opportunities to identify this condition when presented with unexpected/unexplained operation of 1-FCV-1-22. These were identified as contributing causes to the event and corrective actions were developed.

Analysis. The inspector determined that the licensee's failure to maintain design control of MSIV 1-FCV-1-22 controls as required by 10 CFR 50, Appendix B, Criterion III, "Design Control," was a performance deficiency. The finding was determined to be more than minor because it adversely impacted the Design Control attribute of the initiating events cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, it resulted in a slowly closing loop # 3 main steam isolation valve, which ultimately led to a manual reactor trip. The inspectors reviewed IMC 0609, Attachment 4 and determined that the finding was of very low safety significance (Green) because it did not contribute to both the likelihood of a reactor trip and the likelihood that mitigating systems will not be available. No cross-cutting aspect was assigned to this finding because the performance deficiency is not reflective of current licensee performance.

Enforcement. 10 CFR 50, Appendix B, Criterion III, "Design Control," requires, in part, that design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, and that design control measures shall be applied to items such as maintenance and repair. Contrary to the above, since February 15, 1994, TVA did not apply appropriate design control measures to Unit 1 MSIV Control Room handswitch (1-HS-1-22A). The licensee entered the condition into the CAP as CR 1107656 and has corrected the loose wiring connection on 1-HS-1-22A. Because this violation is of very low safety significance (Green) and has been entered into the CAP, this violation is being treated as an NCV consistent with Section 2.3.2.a of the NRC Enforcement Policy. (05000327/2016008-01: Failure to Maintain Design Control of MSIV Controls)

4OA4 Supplemental Inspections

.1 Inspection Scope

The supplemental inspection was performed in accordance with IP 95001 to assess the licensee's evaluation of a White PI, which affected the Initiating Events cornerstone objective in the Reactor Safety strategic performance area. The White PI is associated with having greater than three reactor trips in 7000 critical hours. On September 14, 2015, the indicator transitioned from Green to White when Unit 1 had a fourth cumulative reactor trip within the current 7000 critical hours. The licensee notified the NRC in March 2016 that they were ready for this supplemental inspection. The inspection objectives were to:

- Provide assurance that the root and contributing causes of risk-significant performance issues are understood.
- Provide assurance that the extent of condition and extent of cause of risk-significant performance issues are identified.
- Provide assurance that the licensee's corrective actions for risk-significant performance issues are sufficient to address the root and contributing causes and prevent recurrence.

The inspectors reviewed condition report (CR) 1096174, which documented the root cause evaluation (RCE) for Excessive Unplanned Scrams per 7000 hours for Unit 1, and its constituent CRs. CR 1096174 evaluated the RCEs conducted under the following CRs:

- CR 997605, Unit 1 automatic reactor trip due to control rod H-8 drop.
- CR 1062507, Unit 1 automatic reactor trip due to Automatic Voltage Regulator (AVR) excursions.
- CR 1063258, Unit 1 automatic reactor trip due to AVR placed in manual. SQN leadership decision-making RCA.
- CR 1081482, Unit 1 manual reactor trip due to loss of power to Vital Instrument Power Board (VIPB) 1-II.
- CR 1107656, Unit 1 manual reactor trip due to Loop 3 MSIV drifting toward the closed position.

.2 Evaluation of the Inspection Requirements

.01 Problem Identification

a. Determine that the evaluation documented who identified the issue (i.e., licensee-identified, self-revealing, or NRC-identified) and under what conditions the issue was identified.

The inspector concluded that the licensee's evaluations were generally effective in determining who identified the issues and under what conditions it was identified.

b. Determine that the evaluation documented how long the issue existed and prior opportunities for identification.

The inspector concluded that the licensee's evaluations were generally effective in documenting how long issues existed and prior opportunities for identification.

c. Determine that the evaluation documented the plant-specific risk consequences, as applicable, and compliance concerns associated with the issue.

The inspector concluded that the licensee's evaluations were generally effective in documenting the plant specific risk consequences, as applicable, and compliance concerns associated with the issue.

.02 Root Cause, Extent of Condition, and Extent of Cause Evaluation

a. Determine that the problem was evaluated using a systematic methodology to identify the root and contributing causes.

The inspector concluded that the licensee's evaluations were generally effective in evaluating problems using a systematic methodology to identify root and contributing causes. The inspectors noted that the licensee used varying methodologies in the RCEs commensurate with the issue being evaluated.

b. Determine that the root cause evaluation was conducted to a level of detail commensurate with the significance of the problem.

The inspector reviewed CR 1096174 which evaluated the individual RCEs for the events contributing to the White PI in the aggregate to determine the collective root cause of the scram events, any collective contributing causes, and the corrective actions required to improve performance and prevent recurrence. The evaluation concluded the causes and contributors from each of the events that contributed to the performance indicator involved sufficiently diverse aspects to not result in a single underlying common cause. The licensee performed further analysis to identify consolidated causes that had applicability to multiple events and developed corrective actions to address these additional issues.

The inspector determined that although there was no indication of a singular root cause for the events as a whole, the individual cause analysis performed for each event identified root causes and corrective actions specific to each event. The inspector concluded that based on the extensive work performed in aggregate for the issues, the licensee was evaluating problems to a level of detail that exceeded requirements for their significance.

c. Determine that the root cause evaluation included a consideration of prior occurrences of the problem and knowledge of prior operating experience.

The inspector concluded that the licensee's evaluations were generally effective in including consideration of prior occurrences of the problem and knowledge of prior operating experience.

d. Determine that the root cause evaluation addressed the extent of condition and the extent of cause of the problem.

The inspector concluded that the licensee's evaluations were generally effective in addressing the extent of condition and the extent of cause of problems.

e. Determine that the root cause, extent of condition, and extent of cause evaluations appropriately considered the safety culture components as described in IMC 0305.

The inspector concluded that the licensee's evaluations were generally effective in considering the safety culture components in root causes, extent of condition, and extent of cause. The inspectors noted that the licensee's evaluation process considers the applicability of each safety culture aspect and then addresses those that are determined to be applicable. The inspector also noted the licensee performed a separate analysis of leadership decision making to identify weaknesses in team leadership competencies.

.03 Corrective Actions

a. Determine that appropriate corrective actions are specified for each root and contributing cause or that the licensee has an adequate evaluation for why no corrective actions are necessary.

The inspector concluded that the licensee's evaluations were generally effective in specifying appropriate corrective actions for each root cause and contributing cause or adequately evaluating why no corrective actions were necessary.

b. Determine that corrective actions have been prioritized with consideration of risk significance and regulatory compliance.

The inspector concluded that the licensee's evaluations were generally effective in prioritizing corrective actions taking into consideration risk significance and regulatory compliance.

c. Determine that a schedule has been established for implementing and completing the corrective actions.

The inspector concluded that the licensee's evaluations were generally effective in establishing a schedule for implementing and completing corrective actions.

d. Determine that quantitative or qualitative measures of success have been developed for determining the effectiveness of the corrective actions to prevent recurrence.

The inspector concluded that the licensee's evaluations were generally effective in developing quantitative or qualitative measures of success for determining the effectiveness of the corrective actions to prevent recurrence.

4OA6 Meetings, Including Exit

Exit Meeting Summary

On July 15, 2016, the inspector presented the inspection results to Mr. Schwarz and other members of the licensee's staff. The inspectors asked whether the licensee considered any of the material examined during the inspection to be proprietary; none was identified.

4OA7 Licensee Identified Violations

The following licensee-identified violation of NRC requirements was determined to be of very low safety significance or Severity Level IV and met the NRC Enforcement Criteria for being dispositioned as a Non-Cited Violation.

10 CFR 50 Appendix B, Criterion XVI, Corrective Action, requires, in part, measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected. Contrary to the above, between July 2000 and Sept 14, 2015, the licensee failed to implement corrective actions which would have identified a condition adverse to quality on the output switch on the Spare Vital Inverter 0-II.

This issue was entered into the licensee's corrective action program as CR 1081482. The finding was screened using Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process for Findings At-Power," and was determined to be of very low safety significance (Green).

ATTACHMENT: SUPPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel:
C Schwarz, Site Vice President
J. Johnson, Program Manager Licensing
S. Bowman, Licensing Engineer
M. McBrearty, Licensing Manager

NRC personnel:
G. Smith, Senior Resident Inspector – Sequoyah Nuclear Plant

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened
None

Opened and Closed

05000327/2016008-01	NCV	Failure to Maintain Design Control of MSIV Controls (Section 4OA3.4)

Closed

05000327/2015-001-00,-01	LER	Automatic Reactor Trip due to Negative Rate Trip as a Result of a Dropped Control Rod (Section 4OA3.1)
05000327/2015-002-00	LER	Automatic Reactor Trips due to Improper Wire Termination in Main Generator Voltage Regulator Circuit (Section 4OA3.2)
05000327/2015-003-00	LER	Manual Reactor Trip due to Loss of Power to the Vital Inverter Power Board 1-11 (Section 4OA3.3)
05000327/2015-004-00	LER	Manual Reactor Trip due to Main Steam Isolation Valve Drifting in the Closed Direction (Section 4OA3.4)

LIST OF DOCUMENTS REVIEWED

RCA Packages
RCA 1096174 Rev 2 06/23/2016
RCA 1107656 Rev 4 06/09/2016
RCA 1081482 Rev 1 05/26/2016
RCA 1063258 Rev 2 06/10/2016
RCA 1062507 Rev 2 05/26/2016
RCA 997605 Rev 3 06/20/2016

LERs
05000327/2015-001-00,-01 Automatic Reactor Trip due to Negative Rate Trip as a Result of a Dropped Control Rod
05000327/2015-002-00 Automatic Reactor Trips due to Improper Wire Termination in Main Generator Voltage Regulator Circuit (Section 4OA3.2)
05000327/2015-003-00 Manual Reactor Trip due to Loss of Power to the Vital Inverter Power Board 1-11 (Section 4OA3.3)
05000327/2015-004-00 Manual Reactor Trip due to Main Steam Isolation Valve Drifting in the Closed Direction (Section 4OA3.3)

LIST OF ACRONYMS

ADAMS	Agencywide Document Access and Management System
CAPR	Corrective Action Prevent Recurrence
IMC	Inspection Manual Chapter
IP	Inspection Procedure
NAPS	North Anna Power Station
NRC	Nuclear Regulatory Commission
PI	Performance Indicator
RCE	Root Cause Evaluation