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

August 6, 2014

EA-14-112

Mr. Dennis Madison
Vice President - Vogtle
Southern Nuclear Operating Company, Inc.
Vogtle Electric Generating Plant
7821 River Road
Waynesboro, GA 30830

SUBJECT: VOGTLE ELECTRIC GENERATING PLANT - NRC INTEGRATED INSPECTION REPORT 05000424/2014003 AND 05000425/2014003, AND NOTICE OF VIOLATION

Dear Mr. Madison:

On June 30, 2014, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Vogtle Electric Generating Plant, Units 1 and 2. On July 25, 2014, the NRC inspectors discussed the results of this inspection with Mr. Tom Tynan and other members of the Vogtle staff. Inspectors documented the results of this inspection in the enclosed inspection report.

The enclosed inspection report discusses a finding of low to moderate safety significance (White). As described in Section 4OA2.3 of the enclosed inspection report, a calculation error resulted in the radiological threshold values for the RG1 (General Emergency) and RS1 (Site Area Emergency) emergency action levels to be sixty times greater than the appropriate values. This finding resulted in a potential safety concern for which appropriate immediate corrective actions were taken. The correct threshold values were provided to the appropriate operations staff decision makers which resolved the concern. The licensee took additional corrective actions, including performing a causal determination, processing formal changes to the station's emergency plan and associated implementing procedures, and performing extent of condition/cause reviews throughout the Southern Nuclear Operating Company fleet. Following the internal review process, the revised emergency plan and associated implementing procedure were provided to the NRC.

In a telephone conversation on July 3, 2014, Mr. Brian Bonser, Chief, Plant Support Branch, Division of Reactor Safety, Region II, informed Mr. Tynan of the details of the preliminary finding, the apparent violation, and advised Vogtle representatives that the finding satisfied the "old design issue" criteria contained in NRC Inspection Manual Chapter 0305, "Operating Reactor Assessment Program," Section 11.05, "Treatment of Items Associated with Enforcement Discretion," dated October 18, 2013. The intent of this section is to establish reactor oversight process (ROP) guidance that supports the objective of enforcement discretion, which is to encourage licensee initiatives to identify and resolve problems, especially issues that are not likely to be identified by routine efforts. Additionally, Mr. Bonser advised Mr. Tynan that based on the above, the NRC had sufficient information, including Vogtle's corrective actions, to make a final significance determination and enforcement decision without the need for a regulatory conference or a written response from you. Mr. Tynan indicated they did not believe that a regulatory conference or written response was necessary.

Based on the above, the NRC has concluded that the finding is appropriately characterized as White, a finding of low to moderate safety significance. Additionally, the NRC determined that the White finding meets the criteria specified in IMC 0305 for treatment as an "old design issue." The basis for the NRC's determination included the following: (1) the issue was licensee-identified through an extent of condition review prompted by Southern Co. fleet operating experience; (2) the issue was corrected within a reasonable time after discovery; (3) the issue was not likely to be previously identified by recent ongoing licensee efforts; and (4) the issue was not reflective of a current performance deficiency associated with existing programs, policy, or procedures. Therefore, in accordance with IMC 0305, the performance issue will not aggregate in the Action Matrix with other performance indicators and inspection findings. Note IMC 0305 specifies the need for an inspection in accordance with inspection procedure (IP) 95001 "Supplemental Inspection for One or Two White Inputs in a Strategic Performance Area," to review the licensee's root cause and corrective action plans even if the White finding meets the criteria for treatment as an old design issue. The White finding will remain open until IP 95001 is completed.

The NRC has also determined that the failure to maintain the effectiveness of your emergency plan is a violation of 10 CFR Part 50.54(q)(2), as cited in the attached Notice of Violation (Notice). The circumstances surrounding the violation are described in detail in the enclosed inspection report. In accordance with the NRC Enforcement Policy, the Notice is considered escalated enforcement action because it is associated with a White finding.

The NRC has concluded that the information regarding the reason of the violation, the corrective actions taken to correct the violation and prevent recurrence, and the date when full compliance was achieved is already adequately addressed on the docket in the enclosed inspection report. Therefore, you are not required to respond to this letter unless the description therein does not accurately reflect your corrective actions or your position.

NRC inspectors also documented three findings of very low safety significance (Green) identified during this inspection period. These findings involved violations of NRC requirements. 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 or significance 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 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 Vogtle Electric Generating Plant.

If you disagree with a cross-cutting aspect assignment in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region II; and the NRC resident inspector at the Vogtle Electric Generating Plant.

In accordance with Title 10 of the *Code of Federal Regulations* 2.390, “Public Inspections, Exemptions, Requests for Withholding,” of the NRC’s “Agency Rules of Practice and Procedure,” a copy of this letter, its enclosures, 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 the NRC’s Agencywide Document 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/

Joel T. Munday, Director
Division of Reactor Projects

Docket Nos.: 05000424, 05000425
License Nos.: NPF-68 and NPF-81

Enclosures:
1. Inspection Report 05000424/2014003 and 05000425/2014003
   w/Attachment: Supplemental Information

2. Notice of Violation

cc Distribution via ListServ

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SIGNATURE	Via email	Via email	Via email	Via email	Via email	Via email	Via email
NAME	MCain	TChandler	WPursley	A Nielsen	WLoo	CDykes	MSpeck
DATE	7/11/2014	7/24/2014	7/14/2014	7/14/2014	7/24/2014	7/24/2014	7/15/2014
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E-MAIL COPY?	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO

Letter to Dennis Madison from Joel T. Munday dated August 6, 2014.

SUBJECT: VOGTLE ELECTRIC GENERATING PLANT - NRC INTEGRATED INSPECTION REPORT 05000424/2014003 AND 05000425/2014003, AND NOTICE OF VIOLATION

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U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos.:	50-424, 50-425
License Nos.:	NPF-68, NPF-81
Report Nos.:	05000424/2014003 and 05000425/2014003
Licensee:	Southern Nuclear Operating Company, Inc. (SNC)
Facility:	Vogtle Electric Generating Plant, Units 1 and 2
Location:	Waynesboro, GA 30830
Dates:	April 1, 2014, through June 30, 2014
Inspectors:	M. Cain, Senior Resident Inspector T. Chandler, Resident Inspector A. Alen, Project Engineer W. Pursley, Health Physics Inspector (2RS1, 2RS2, 2RS4, 4OA1) A. Nielsen, Senior Health Physicist (2RS1, 2RS3, 4OA1) W. Loo, Senior Health Physicist (2RS1, 2RS3) C. Dykes, Health Physicist (2RS5) M. Speck, Senior Emergency Preparedness Inspector (4OA2.3) S. Sanchez, Senior Emergency Preparedness Inspector (4OA2.3) G. Ottenberg, Senior Reactor Inspector (4OA5)
Approved by:	Frank Ehrhardt, Chief Reactor Projects Branch 2 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000424/2014003, 05000425/2014003; 04/01/2014 – 06/30/2014; Vogtle Electric Generating Plant, Units 1 and 2; Maintenance Effectiveness, Radiological Hazard Assessment and Exposure Controls, Identification and Resolution of Problems, Event Follow-up

The report covered a 3-month period of inspection by resident inspectors and regional inspectors. There was one NRC-identified and three self-revealing violations identified and documented in this report. The significance of inspection findings are indicated by their color (i.e., greater than Green, or Green, White, Yellow, Red) and determined using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP) dated June 2, 2011. The cross-cutting aspects are determined using IMC 0310, "Aspects within the Cross-Cutting Areas" dated December 19, 2013. All violations of NRC requirements are dispositioned in accordance with the NRC's Enforcement Policy dated January 28, 2013. The NRC's program for overseeing the safe operations of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 5.

Cornerstone: Initiating Events

Green A self-revealing non-cited violation (NCV) of 10 CFR 50 Appendix B Criterion V, "Instructions, Procedures, and Drawings," was identified for failure to provide adequate work instructions in the maintenance procedure used for main steam isolation valve (MSIV) maintenance. Specifically, maintenance procedure 26854-C, "Main Steam Isolation Valve Actuator Maintenance," used to perform maintenance on Rockwell MSIV(s), did not provide adequate instructions for installing the lower manifold/cylinder O-ring during reassembly. This resulted in a 'pinched' O-ring on 1HV3006B, a subsequent failure of the O-ring causing the MSIV to fail closed, and a manual reactor trip. The licensee conducted a root cause investigation and entered the event into their corrective action program (condition report (CR) 800018). The licensee replaced the O-ring, performed an extent of condition evaluation for all other MSIVs, and revised the maintenance procedure to include specific instructions for the installation of the lower manifold/cylinder O-ring.

The finding was more than minor because it was associated with the procedure quality attribute of the reactor safety - initiating events cornerstone and it adversely affected the cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, the failure to provide an adequate procedure with adequate instructions for ensuring proper O-ring installation resulted in the failure of the Unit 1 loop 1 outboard MSIV hydraulic actuator causing the loop 1 MSIV to fail closed and a subsequent manual reactor trip due to lowering steam generator water level. Because the inspectors answered "No" to all of the IMC 0609 Appendix A (dated June 19, 2012) Exhibit 1, Section B, "Initiating Events Screening Questions," the inspectors concluded that the finding was of very low safety significance (Green). The inspectors determined the finding had a cross-cutting aspect of “resources” in the human performance area, because the maintenance procedure used to install manifold/cylinder O-ring did not provide adequate instructions for the proper installation of the O-ring. [H.1] (Section 1R12)

Green A self-revealing NCV of 10 CFR 50 Appendix B Criterion V, “Instructions, Procedures, and Drawings,” was identified for failure to provide adequate work instructions as well as failure to follow the maintenance procedure used to install flexible and rigid conduit. Specifically, the work instructions did not provide adequate directions and/or precautions to properly slope conduit during installation to prevent water intrusion into a valve positioner. The work instructions referenced maintenance procedure 25008-C, “Flexible and Rigid Conduit Installation.” The maintenance procedure referenced Vogtle design specification X3AR01 Section E-8, “Raceway Systems,” which provided sloping and tightness criteria for conduit installations. The licensee conducted a root cause investigation and entered the event into their corrective action program (CR 797929). The licensee repaired the improperly sloped conduit, replaced the positioner, and revised procedure 25008-C to specify standards for proper sloping of conduits.

The finding was more than minor because it was associated with the procedure quality and human performance attributes of the reactor safety - initiating events cornerstone and it adversely affected the cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, the failure to provide adequate work instructions as well as failure to follow procedure 25008-C, “Flexible and Rigid Conduit Installation,” resulted in the Unit 2 loop 3 main feedwater regulating valve (MFRV) positioner failing closed, causing a subsequent automatic reactor trip due to low-low steam generator (SG) water level. Because the inspectors answered “No” to all of the IMC 0609 Appendix A (dated June 19, 2012) Exhibit 1, Section B, “Initiating Events Screening Questions,” the inspectors concluded that the finding was of very low safety significance (Green). The inspectors determined that the finding had a cross-cutting aspect of “procedure adherence” in the human performance area because the maintenance electricians did not follow Vogtle design specification procedures or drawings resulting in the improper sloping of the MFRV flexible conduit [H.8] (Section 40A3)

Cornerstone: Occupational Radiation Safety

Green A self-revealing NCV of Technical Specification (TS) 5.7.1, “High Radiation Area”, was identified for an entry into a high radiation area (HRA) without meeting the entry requirements as specified therein. Specifically, on March 17, 2014, an operator was authorized to enter an HRA on Unit 1 under conditions where dose rates were known to be changing. This allowed the operator entry into an HRA without knowledge of actual radiological conditions. He was not provided with a radiation monitoring device that continuously indicated dose rates in the area, nor was he accompanied by an individual qualified in radiation protection procedures with a radiation monitoring device providing positive control over his activities. Upon discovery of the condition, the licensee secured access to the area, performed follow-up surveys and convened a human performance review board to examine causal factors and identify corrective actions. The licensee entered this issue into the corrective action program as CR

787908.

This finding was more than minor because it was associated with the occupational radiation safety cornerstone attribute of human performance and adversely affects the cornerstone objective of ensuring adequate protection of worker health and safety from exposure to radiation from radioactive material during routine civilian nuclear reactor operation. Specifically, workers permitted entry into HRAs with inadequate knowledge of current radiological conditions could receive unintended occupational exposures. The finding was evaluated using IMC 0609, Appendix C, "Occupational Radiation Safety Significance Determination Process (SDP)", dated August 19, 2008. The finding was not related to As Low As Reasonably Achievable (ALARA) planning, nor did it involve an overexposure or substantial potential for overexposure and the ability to assess dose was not compromised. Therefore, the finding was determined to be of very low safety significance (Green). This finding had a cross-cutting aspect of "avoid complacency" in the human performance area because health physics (HP) personnel failed to verify plant conditions through available means when an evolution was in progress that was known to increase area dose rates prior to authorizing entry into an HRA. [H.12] (Section 2RS1)

<u>Cornerstone: Emergency Preparedness</u>

<u>White</u>: A finding and associated violation of 10 CFR 50.54(q)(2) was identified by the licensee for the failure to follow and maintain the effectiveness of emergency plans which use a standard emergency classification and action level scheme. Specifically, the licensee's emergency plan emergency action level (EAL) Category R – Abnormal Radiological RG1 (General Emergency) and RS1(Site Area Emergency) specified threshold values which were sixty times too high due to a calculation error. As immediate corrective action, the licensee provided the corrected threshold values to appropriate management and decision-makers (shift managers/emergency directors). The licensee entered this issue into the corrective action program as CR 648248.

The performance deficiency was determined to be more than minor because it was associated with the emergency preparedness cornerstone attribute of procedure quality. It impacted the cornerstone objective because it was associated with inappropriate EAL and emergency plan changes and their adequacy to protect the health and safety of the public in the event of a radiological emergency. Specifically, the licensee's ability to declare a Site Area Emergency and General Emergency based on effluent radiation monitor values was degraded in that event classification using these radiation monitors would be delayed. The finding was assessed for significance in accordance with NRC Manual Chapter 0609, Appendix B, "Emergency Preparedness Significance Determination Process," which states, "Failure to comply means that a program is noncompliant with a Regulatory requirement." The inspector determined that the issue of concern constituted a degraded rather than lost risk-significant planning standard (RSPS). The issue of concern was similar to the example in Table 5.4.1 (Degraded RSPS) and was determined to be of low to moderate safety significance (White). The violation was determined to meet the IMC 0305 criteria for enforcement discretion as an old design issue. A cross-cutting aspect was not assigned based on the elapsed time since the performance deficiency occurred and because the inspectors determined it was not reflective of current licensee performance. (Section 4OA2)

REPORT DETAILS

Summary of Plant Status

Unit 1 started the reporting period shut down for a planned refueling outage. Operators restarted the unit on April 11, 2014, and attained 100 percent rated thermal power (RTP) on April 12, 2014. Operators manually tripped the unit on April 12, 2014, due to a failure of the loop 1 main steam isolation valve (MSIV) failing closed at 100 percent RTP. Operators restarted the unit on April 13, 2014 and attained 100 percent RTP on April 27, 2014. The unit operated at essentially RTP for the rest of the inspection period.

Unit 2 started the report period at full RTP. The unit automatically tripped from 100 percent RTP on April 8, 2014, due to low level in the loop 3 steam generator caused by the main feedwater regulator valve (MFRV) failing closed. Operators restarted the unit on April 10, 2014, and attained 100 percent power on April 11, 2014.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

a. Inspection Scope

.1 Summer Readiness of Offsite and Alternate AC Power System

Because the licensee implemented modifications to the high and low voltage switchyards, the inspectors reviewed the licensee's procedures for operation and continued availability of offsite and onsite alternate AC power systems. The inspectors also reviewed the communications protocols between the transmission system operator and the licensee to verify that the appropriate information is exchanged when issues arise that could affect the offsite power system.

The inspectors reviewed the material condition of offsite and onsite alternate AC power systems (including switchyard and transformers) by performing a walkdown of the switchyard. The inspectors reviewed outstanding work orders and assessed corrective actions for any degraded conditions that impacted plant risk or required compensatory actions. Documents reviewed are listed in the Attachment.

.2 Seasonal Extreme Weather Conditions

The inspectors conducted a detailed review of the station's adverse weather procedures written for extreme high temperatures. The inspectors verified that weather related equipment deficiencies identified during the previous year had been placed into the work control process and/or corrected before the onset of seasonal extremes. The inspectors evaluated the licensee's implementation of adverse weather preparation procedures and compensatory measures before the onset of seasonal extreme weather conditions. Documents reviewed are listed in the Attachment.

The inspectors evaluated the following risk-significant systems:

- Unit 2 nuclear service cooling water (NSCW) system (both trains)
- Unit 1 emergency diesel generator (EDG) system (both trains)

b. Findings

No findings were identified.

1R04 Equipment Alignment (71111.04)

a. Inspection Scope

Partial Walkdown

The inspectors verified that critical portions of the selected systems were correctly aligned by performing partial walkdowns. The inspectors selected systems for assessment because they were a redundant or backup system or train, were important for mitigating risk for the current plant conditions, had been recently realigned, or were a single-train system. The inspectors determined the correct system lineup by reviewing plant procedures and drawings. Documents reviewed are listed in the Attachment.

The inspectors selected the following four systems or trains to inspect:

- Unit 2 train “B” EDG while the train “A” EDG was out of service due to a planned maintenance outage
- Unit 2 train “B” motor-driven auxiliary feedwater system and the train “C” turbine-driven auxiliary feedwater system during the train “A” EDG planned maintenance outage
- Unit 2 train “A” EDG during the train “B” EDG planned maintenance outage
- Unit 2 train “A” motor-driven auxiliary feedwater system and the train “C” turbine-driven auxiliary feedwater system during the train “B” EDG planned maintenance outage

b. Findings

No findings were identified.

1R05 Fire Protection (71111.05AQ)

a. Inspection Scope

Quarterly Inspection

The inspectors evaluated the adequacy of selected fire plans by comparing the fire plans to the defined hazards and defense-in-depth features specified in the fire protection program. In evaluating the fire plans, the inspectors assessed the following items:

- control of transient combustibles and ignition sources
- fire detection systems
- water-based fire suppression systems
- gaseous fire suppression systems
- manual firefighting equipment and capability
- passive fire protection features
- compensatory measures and fire watches
- issues related to fire protection contained in the licensee's corrective action program

The inspectors toured the following five fire areas to assess material condition and operational status of fire protection equipment. Documents reviewed are listed in the Attachment.

- Unit 2 component cooling water (CCW) heat exchanger rooms, fire zones 54, 55, 148, 23, 172, and 147
- Unit 1 centrifugal charging pump (CCP) rooms and the level "C" pipe penetration area in the Unit 1 auxiliary building, fire zones 14B, 19, 20, and 21
- Unit 2 control building level "A" west and east penetration areas, fire zones 87, 88, 89, 90 93, 102, 158 and 159.
- Unit 1 "B" train EDG building, fire zones 162 and 164
- Unit 2 auxiliary feedwater pump house, fire zones 155, 156, 157A and 157B

b. Findings

No findings were identified.

1R06 Flood Protection Measures (71111.06)

a. Inspection Scope

.1 Internal Flooding

The inspectors reviewed related flood analysis documents and walked down the area listed below containing risk-significant structures, systems, and components susceptible to flooding. The inspectors verified that plant design features and plant procedures for flood mitigation were consistent with design requirements and internal flooding analysis assumptions. The inspectors also assessed the condition of flood protection barriers and drain systems. In addition, the inspectors verified the licensee was identifying and properly addressing issues using the corrective action program. Documents reviewed are listed in the Attachment.

- Unit 1 residual heat removal (RHR) and containment spray (CS) pump rooms (both trains) in auxiliary building

1R11 Licensed Operator Requalification Program and Licensed Operator Performance (71111.11)

a. Inspection Scope

.1 Resident Inspector Quarterly Review of Licensed Operator Requalification

The inspectors observed an evaluated simulator scenario administered to an operating crew conducted in accordance with the licensee's accredited requalification training program.

The inspectors assessed the following:

- licensed operator performance
- the ability of the licensee to administer the scenario and evaluate the operators
- the quality of the post-scenario critique
- simulator performance

Documents reviewed are listed in the Attachment.

.2 Resident Inspector Quarterly Review of Licensed Operator Performance

The inspectors observed licensed operator performance in the main control room on April 9, 2014, while operators were starting up the Unit 2 reactor.

The inspectors assessed the following:

- use of plant procedures
- control board manipulations
- communications between crew members
- use and interpretation of instruments, indications, and alarms
- use of human error prevention techniques
- documentation of activities
- management and supervision

Documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

1R12 Maintenance Effectiveness (71111.12)

a. Inspection Scope

The inspectors assessed the licensee's treatment of the two issues listed below in order to verify the licensee appropriately addressed equipment problems within the scope of the maintenance rule (10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants".) The inspectors reviewed procedures and records in order to evaluate the licensee's identification, assessment, and characterization of the problems as well as their corrective actions for returning the equipment to a satisfactory condition. The inspectors also interviewed system engineers and the maintenance rule coordinator to assess the accuracy of performance deficiencies and extent of condition. Documents reviewed are listed in the Attachment.

- Unit 2, system 1305, 2HV5230 hydraulic leak
- Unit 1, system 1301, 1HV3006B maintenance preventable functional failure (MPFF)

b. Findings

Introduction: A Green, self-revealing NCV of 10 CFR 50 Appendix B Criterion V, "Instructions, Procedures, and Drawings," was identified for failure to provide adequate work instructions in the maintenance procedure used to reassemble Rockwell MSIVs. Specifically, maintenance procedure 26854-C, "Main Steam Isolation Valve Actuator Maintenance," which is used to perform maintenance on Rockwell MSIVs, did not provide adequate instructions for installing of the lower manifold/cylinder O-ring during reassembly using threaded guide rods to align the mating surfaces.

Description: On April 12, 2014, Unit 1 was in Mode 1 ascending in power after the 1R18 refueling outage. At approximately 20:08, control room operators received an MSIV actuator trouble alarm followed by the MSIV not fully open indication. Control room operators identified lowering loop 1 steam generator (SG) #1 level and steam flow and manually tripped Unit 1 at about 28 percent reactor power. Upon further investigation, operators discovered a severe leak on the loop 1 outboard MSIV hydraulic actuator, which had caused the valve to close. Operators stabilized the plant in Mode 3 and all safety related equipment responded as expected. The licensee assembled an issue response team (IRT) and a root cause team to investigate the cause of the hydraulic leak and subsequent manual reactor trip and to determine the required corrective actions. Further investigation revealed that the manifold to cylinder O-ring on the valve actuator had failed catastrophically due to being pinched during actuator reassembly in 2012. Further research by the root cause team revealed that maintenance personnel relied on "skill of the craft" to install the O-ring and used a hoist to align the cylinder with the manifold body. Use of the hoist resulted in rotational and/or oscillatory movement of the mating surfaces, pinching the O-ring. The maintenance procedure that the mechanics used to reassemble the actuator did not contain adequate instructions for installing the manifold/cylinder O-ring. Specifically, maintenance procedure 26854-C, "Main Steam Isolation Valve Actuator Maintenance," which is used to perform maintenance on Rockwell MSIVs, did not provide adequate instructions for installing the lower manifold/cylinder O-ring during reassembly using threaded guide rods to align the mating surfaces. The licensee revised the maintenance procedure, replaced the O-ring, and conducted an extent of condition evaluation of all other MSIV actuators. The licensee entered this issue into their corrective action program as CR 800018.

Analysis: The failure to provide adequate procedures required by 10 CFR 50 Appendix B Criterion V was a performance deficiency. The inspectors determined that the performance deficiency was more than minor because it was associated with the procedure quality attribute of the initiating events cornerstone and it adversely affected the cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, the failure to provide adequate instructions for the installation of the manifold to cylinder O-ring resulted in failure of the loop 1 MSIV and a subsequent manual reactor trip due to lowering SG water level and steam flow. The inspectors evaluated the finding using IMC 0609, Appendix A, “The Significance Determination Process (SDP) for Findings At-Power,” dated June 19, 2012. Because the inspectors answered “No” to all the Exhibit 1, Section B, “Initiating Events Screening Questions,” the inspectors determined that the finding was of very low safety significance (Green). The inspectors determined the finding had a cross-cutting aspect of “resources” in the human performance area, because the maintenance procedure used to install manifold/cylinder O-ring did not provide adequate instructions for the proper installation of the O-ring. [H.1]

Enforcement: 10 CFR 50 Appendix B Criterion V, “Instructions, Procedures, and Drawings,” requires, in part, that procedures shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Contrary to the above, the maintenance procedure used to reassemble the MSIV hydraulic actuator did not provide adequate instructions for the proper alignment of the manifold to cylinder mating surfaces resulting in a pinched O-ring and subsequent MSIV actuator failure. Specifically, maintenance procedure 26854-C, “Main Steam Isolation Valve Actuator Maintenance,” which is used to perform maintenance on Rockwell MSIVs, did not provide adequate instructions for installing the lower manifold/cylinder O-ring during reassembly. To restore compliance, the licensee revised the maintenance procedure, replaced the O-ring, and conducted an extent of condition evaluation of all other MSIV actuators. This violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy. The violation was entered into the licensee’s corrective action program as CR 800018. (NCV 05000424/2014003-01, “Inadequate Maintenance Procedure Results in a Failed MSIV and a Manual Reactor Trip”)

1R13 Maintenance Risk Assessments and Emergent Work Control (71111.13)

a. Inspection Scope

The inspectors reviewed the five maintenance activities listed below to verify that the licensee assessed and managed plant risk as required by 10 CFR 50.65(a)(4) and licensee procedures. The inspectors assessed the adequacy of the licensee’s risk assessments and implementation of risk management actions. The inspectors also verified that the licensee was identifying and resolving problems with assessing and managing maintenance-related risk using the corrective action program. Additionally, for maintenance resulting from unforeseen situations, the inspectors assessed the effectiveness of the licensee’s planning and control of emergent work activities. Documents reviewed are listed in the Attachment.

- Unit 2, week of May 5, 2014, Yellow risk condition associated with the extended allowed outage time (AOT) of the Unit 2 “A” EDG
- Unit 2, week of May 12, 2014, Orange risk condition associated with the extended AOT of the Unit 2 “A” EDG
- Unit 1, week of May 19, Yellow risk condition associated with the extended AOT of the Unit 1 “A” NSCW cooling tower fan #3
- Unit 1, week of June 2, 2014, during a planned maintenance outage of “1A” CCW pump in conjunction with an unplanned inoperability of the Unit 1A control room emergency fan system (CREFS)
- Unit 2, week of June 16, 2014, Yellow risk condition associated with the extended AOT of the Unit 2 “B” EDG

b. Findings

No findings were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors selected the five operability determinations or functionality evaluations listed below for review based on the risk-significance of the associated components and systems. The inspectors reviewed the technical adequacy of the determinations to ensure that technical specification operability was properly justified and the components or systems remained capable of performing their design functions. To verify whether components or systems were operable, the inspectors compared the operability and design criteria in the appropriate sections of the technical specification and updated final safety analysis report to the licensee’s evaluations. Where compensatory measures were required to maintain operability, the inspectors determined whether the measures in place would function as intended and were properly controlled. Additionally, the inspectors reviewed a sample of corrective action documents to verify the licensee was identifying and correcting any deficiencies associated with operability evaluations. Documents reviewed are listed in the Attachment.

- CR 776584, Unknown chemical buildup on top of upper motor windings
- CR 808990, “2B” EDG jacket water leak
- CR 805473/CAR 210188, 1HV3036A MSIV control board red light flickering
- CR 807567/CAR 210214, Unit 2 turbine driven auxiliary feedwater pump (TDAFW) controller output reading low
- CR 607966, U1 CCW Pump “1A” inboard bearing over 160 degrees Fahrenheit

b. Findings

No findings were identified.

1R18 Plant Modifications (71111.18)

a. Inspection Scope

The inspectors verified that the two plant modifications listed below did not affect the safety functions of important safety systems. The inspectors confirmed the modifications did not degrade the design bases, licensing bases, and performance capability of risk significant structures, systems, and components. The inspectors also verified modifications performed during plant configurations involving increased risk did not place the plant in an unsafe condition. Additionally, the inspectors evaluated whether system operability and availability, configuration control, post-installation test activities, and changes to documents, such as drawings, procedures, and operator training materials, complied with licensee standards and NRC requirements. In addition, the inspectors reviewed a sample of related corrective action documents to verify the licensee was identifying and correcting any deficiencies associated with modifications. Documents reviewed are listed in the Attachment.

- SNC417397, Temporary modification to install accelerometers and a pressure transducer on chemical volume control system (CVCS) letdown lines, Unit 1
- DCP 98-VAN0055, Replace alternate radwaste building (ARB) with radwaste processing facility (RPF)

b. Findings

No findings were identified.

1R19 Post-Maintenance Testing (71111.19)

a. Inspection Scope

The inspectors either observed post-maintenance testing or reviewed the test results for the six maintenance activities listed below to verify the work performed was completed correctly and the test activities were adequate to verify system operability and functional capability.

- Maintenance Work Order (MWO) SNC137725 – Replacement of “1E” D26 relays MCC21805S3ABE
- MWO SNC413540 – 2PV3020 Replace A/B solenoid
- MWOs SNC408041 – (1A NSCW Fan 2) – Replace agastat relay and SNC383989 – (1A NSCW Fan 2) – Replace rubber bushings on fan couplings
- MWO SNC525486 – Unit “2A” EDG Undervoltage relay calibration
- MWO SNC516991 – Unit 1 delta T/Tavg loop 3 protection channel operational test and calibration
- MWO SNC488414 - Unit 2 delta T/Tavg loop 1 protection channel I 2T-411 operational test and calibration

The inspectors evaluated these activities for the following:

- Acceptance criteria were clear and demonstrated operational readiness.
- Effects of testing on the plant were adequately addressed.
- Test instrumentation was appropriate.
- Tests were performed in accordance with approved procedures.
- Equipment was returned to its operational status following testing.
- Test documentation was properly evaluated.

Additionally, the inspectors reviewed a sample of corrective action documents to verify the licensee was identifying and correcting any deficiencies associated with post-maintenance testing. Documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

1R20 Refueling and Other Outage Activities (71111.20)

a. Inspection Scope

For the Unit 1 refueling outage which continued from April 1 2014, through April 27 2014, the inspectors evaluated the following outage activities:

- outage planning
- heatup, and startup
- reactor coolant system instrumentation and electrical power configuration
- reactivity and inventory control
- decay heat removal and spent fuel pool cooling system operation
- containment closure

The inspectors verified that the licensee:

- considered risk in developing the outage schedule
- controlled plant configuration in accordance with administrative risk reduction methodologies
- developed work schedules to manage fatigue
- developed mitigation strategies for loss of key safety functions
- adhered to operating license and technical specification requirements

Inspectors verified that safety-related and risk-significant structures, systems, and components not accessible during power operations were maintained in an operable condition. The inspectors also reviewed a sample of related corrective action documents to verify the licensee was identifying and correcting any deficiencies associated with outage activities. Documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors reviewed the seven surveillance tests listed below and either observed the test or reviewed test results to verify testing adequately demonstrated equipment operability and met technical specification and licensee procedural requirements. The inspectors evaluated the test activities to assess for preconditioning of equipment, procedure adherence, and equipment alignment following completion of the surveillance. Additionally, the inspectors reviewed a sample of related corrective action documents to verify the licensee was identifying and correcting any deficiencies associated with surveillance testing. Documents reviewed are listed in the Attachment.

Routine Surveillance Tests

- 14802A-2 Rev. 5, Train “A” NSCW Pump / Check Valve IST and Response Time Test
- 24568-2 Rev. 38, RCP 1 Train “A”, Reactor Trip Relays Under Frequency (281-A), Under Voltage (227-A), Timing (262R-A) Trip Actuating Device Operational Test and Channel Calibration and 24565-2, Rev. 37, RCP 2 Train “A”, Reactor Trip Relays Under Frequency (281-A), Under Voltage (227-A), Timing (262R-A) Trip Actuating Device Operational Test and Channel Calibration
- 24449-2 Rev. 9, Diesel Generator Power Out Train 2Q-2791 Channel Calibration
- 21118-2 Rev. 3.2, Centrifugal Charging Pump (CCP) Train “A” Safety Grade Charging Flow Loop 2F-0138 Channel Calibration

Reactor Coolant System Leak Detection

- 14905-1 Rev. 69.0, RCS Leakage Calculation (Inventory Balance)
- 14905-2 Rev. 53.0, RCS Leakage Calculation (Inventory Balance)

In-Service Tests (IST)

- 14804B-1 Rev. 5.0, Safety Injection Pump “B” Inservice and Response Time Tests

b. Findings

No findings were identified.

Cornerstone: Emergency Preparedness

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope

The inspectors observed the emergency preparedness drill conducted on May 21, 2014. The inspectors observed licensee activities in the simulator and alternate technical support center to evaluate implementation of the emergency plan, including event classification, notification, and protective action recommendations. The inspectors evaluated the licensee's performance against criteria established in the licensee's procedures. Additionally, the inspectors attended the post-exercise critique to assess the licensee's effectiveness in identifying emergency preparedness weaknesses and verified the identified weaknesses were entered in the corrective action program.

b. Findings

No findings were identified.

2. RADIATION SAFETY

2RS1 Radiological Hazard Assessment and Exposure Controls

a. Inspection Scope

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

Hazard Control and Work Practices The inspectors evaluated access barrier effectiveness for selected Unit 1 and Unit 2 locked high radiation area (LHRA) and very high radiation area (VHRA) locations. Changes to procedural guidance for LHRA and VHRA controls were discussed with HP supervisors. Controls and their implementation for storage of irradiated material within the spent fuel pool were reviewed and discussed in detail. Established radiological controls (including airborne controls) were evaluated for selected Unit 1 refueling outage 18 (1R18) tasks including detensioning of the reactor head, reactor head lift, upper internals lift, and scaffold building in Unit 1 containment. 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.

Occupational workers' adherence to selected RWPs and HP technician (HPT) proficiency in providing job coverage were evaluated through direct observations and interviews with licensee staff. Electronic dosimeter (ED) alarm set points and worker stay times were evaluated against area radiation survey results for detensioning of the reactor head, reactor head lift, upper internals lift, and scaffold building in Unit 1 containment. ED alarm logs were reviewed and worker response to dose and dose rate alarms during selected work activities was evaluated. 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.

Control of Radioactive Material The inspectors observed surveys of material and personnel being released from the RCA using small article monitor (SAM), personnel contamination monitor (PCM), and portal monitor (PM) instruments. The inspectors reviewed selected calibration records for selected release point survey instruments and discussed equipment sensitivity, alarm setpoints, and release program guidance with licensee staff. The inspectors compared recent 10 CFR Part 61 results for the dry active waste (DAW) radioactive waste stream with radionuclides used in calibration sources to evaluate the appropriateness and accuracy of release survey instrumentation. The inspectors also reviewed records of leak tests on selected sealed sources and discussed nationally tracked source transactions with licensee staff.

Problem Identification and Resolution CRs associated with radiological hazard assessment and control were reviewed and assessed. The inspectors evaluated the licensee's ability to identify and resolve the issues in accordance with procedure NMP-GM-002, "Corrective Action Program," Version (Ver.) 12.1. The inspectors also evaluated the scope of the licensee's internal audit program and reviewed recent assessment results.

Radiation protection activities were evaluated against the requirements of Updated Final Safety Analysis Report (UFSAR) Section 12; TS Sections 5.4 and 5.7; 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 report Attachment.

b. Findings

Introduction: A Green, self-revealing, NCV of TS 5.7.1, "High Radiation Area," was identified for permitting an individual entry into a HRA without meeting the entry requirements as specified in TS 5.7.1.b.

Description: On March 17, 2014, with the Unit 1 reactor shutdown for refueling, an operator was performing troubleshooting of Unit 1 containment sumps leakage. A planned reactor coolant system crud burst was in progress on Unit 1. As a result of the crud burst radiation levels in parts of the auxiliary building were elevated and areas were posted as high radiation areas. The operator observed a “Danger High Radiation Area - HP Escort Required for Entry - Alarming Dosimetry,” posting at the entrance to the encapsulation vessel room and returned to the HP control point for further instructions. The operator was briefed by an HP technician using a survey performed for the area on March 6, 2014, that did not reflect the current postings or current radiological conditions. The operator was informed by the HP technician that he could enter the area without an HP escort because he was using an alarming ED. In the follow-up investigation the HP technician stated that he was not aware the crud burst had started. Upon entry into the encapsulation vessel room, the operator received a dose rate alarm on his ED. He stopped immediately and exited the area. The worker’s ED alarm setpoint was 250 millirem per hour (mrem/hr) and the highest exposure rate seen by the ED was 262 mrem/hr. Dose rates in the area were as high as 300 mrem/hr on contact and 193 mrem/hr at 30 cm based on a follow-up survey. The licensee entered this issue into their corrective action program as CR 787908 and took immediate corrective actions which included securing access to the area, performing follow-up surveys and convening a human performance review board to examine causal factors for the purpose of determining corrective actions.

<u>Analysis</u>: The inspectors determined that entry into a HRA without meeting the entry requirements specified in T.S. 5.7.1 was a performance deficiency. This finding was more than minor because it was associated with the occupational radiation safety cornerstone attribute of human performance and adversely affects the cornerstone objective of ensuring adequate protection of worker health and safety from exposure to radiation from radioactive material during routine civilian nuclear reactor operation. Specifically, workers permitted entry into HRAs with inadequate knowledge of current radiological conditions could receive unintended occupational exposures. The finding was evaluated using IMC 0609, Appendix C, “Occupational Radiation Safety Significance Determination Process (SDP)”, dated August 19, 2008. The finding was not related to ALARA planning, nor did it involve an overexposure or substantial potential for overexposure, and the ability to assess dose was not compromised. Therefore, the inspectors determined the finding to be of very low safety significance (Green). The inspectors noted that the operator responded properly to the ED dose rate alarm thereby limiting his potential for unintended exposure. This finding had a cross-cutting aspect of “avoid complacency” in the human performance area because HP personnel failed to verify plant conditions through available means when an evolution was in progress that was known to increase area dose rates prior to authorizing entry into an HRA. [H.12]

<u>Enforcement</u>: Technical Specification 5.7.1, “High Radiation Area”, requires in part, individuals entering HRAs meet one or more of the following criteria: a) be provided with a radiation monitoring device that continuously indicates radiation dose rate in the area; b) a radiation monitoring device that continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel are aware of them or c) An individual qualified in radiation protection procedures with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by health physics supervision in the RWP. Contrary to the above, on March 17, 2014, a worker entered a HRA without a

device that continuously indicated dose rates in the area (survey meter), knowledge of the actual radiological conditions in the area and no trained escort with a survey meter. This violation is being treated as an NCV, consistent with Section 2.3.2 of the Enforcement Policy. The violation was entered into the licensee's corrective action program as CR 787908. (NCV 05000424, 2014003-02, "Unauthorized Entry into a High Radiation Area.")

2RS2 Occupational ALARA Planning and Controls

a. Inspection Scope

Work Planning and Exposure Tracking The inspectors reviewed planned work activities and their collective exposure estimates for the current 1R18 outage. The inspectors reviewed ALARA planning packages for the following high collective exposure tasks: install/remove scaffolding, thermocouple work, mechanical stress improvement project (MSIP) work in containment and interference removal work in Unit 1 containment annulus. For the selected tasks, the inspectors reviewed established dose goals, discussed assumptions regarding the bases for the current estimates with responsible ALARA planners and walked down a mock-up of the reactor cavity annulus. The inspectors evaluated the incorporation of exposure reduction initiatives and operating experience, including historical post-job reviews, into RWP requirements. Day-to-day collective dose data for the selected tasks were compared with established dose estimates and evaluated against procedural criteria (work-in-progress review limits) for additional ALARA review. Where applicable, the inspectors discussed changes to established estimates with ALARA planners and evaluated them against work scope changes or unanticipated elevated dose rates.

Source Term Reduction and Control The inspectors reviewed the collective exposure three-year rolling average from 2010 – 2012 and reviewed historical collective exposure trends from 1988 – 2014. The inspectors evaluated historical dose rate trends for reactor coolant system piping and compared them to current 1R18 data. The crud burst evolution during the first week of the 1R18 outage and source term reduction initiatives were reviewed and discussed with chemistry and HP staff.

Radiation Worker Performance The inspectors observed radiation worker performance for job evolutions such as the MSIP interference removal, installation of shielding and work in and around the reactor cavity. The inspectors observed ALARA briefings for multiple MSIP jobs and emerging jobs such as Unit 1 bullet nose repair and radiation worker performance was also evaluated as part of IP 71124.01. While observing job tasks, the inspectors evaluated the use of remote technologies to reduce dose including teledosimetry and remote visual monitoring.

Problem Identification and Resolution The inspectors reviewed and discussed selected corrective action program documents associated with ALARA program implementation. The inspectors evaluated the licensee's ability to identify and resolve the issues in accordance with licensee procedure NMP-GM-002, "Corrective Action Program", Ver. 12.1. 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 UFSAR Section 12, TS Section 5.4, 10 CFR Part 20, and approved licensee procedures. Records reviewed are listed in the report Attachment.

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 the 1R18 refueling outage. The inspectors observed the use of portable air filtration units for work in contaminated areas of the containment building and reviewed filtration unit testing records. The inspectors evaluated the effectiveness of continuous air monitors and air samplers placed in work area “breathing zones” to provide indication of increasing airborne levels.

Respiratory Protection Equipment The inspectors reviewed the use of respiratory protection devices to limit the intake of radioactive material. This included review of devices used for routine tasks and devices stored for use in emergency situations. The inspectors reviewed ALARA evaluations for the use of respiratory protection devices during work associated with steam generator (S/G) eddy current testing. Selected self-contained breathing apparatus (SCBA) units and negative pressure respirators (NPR)s staged for routine and emergency use in the main control room and other locations were inspected for material condition, SCBA bottle air pressure, number of units, and number of spare masks and air bottles available. The inspectors reviewed maintenance records for selected SCBA units for the past two years and evaluated SCBA and NPR compliance with National Institute for Occupational Safety and Health certification requirements. The inspectors also reviewed records of air quality testing for supplied-air devices and SCBA bottles.

The inspectors observed the use of powered air-purifying hoods during work on the S/G platforms and in the upper cavity. The inspectors discussed training for various types of respiratory protection devices with HP staff and interviewed radworkers and control room operators on use of the devices. The inspectors reviewed respirator qualification records (including medical qualifications) for several main control room operators and emergency responder personnel in the maintenance department.

Problem Identification and Resolution The inspectors reviewed CRs associated with airborne radioactivity mitigation and respiratory protection. The inspectors evaluated the licensee’s ability to identify and resolve the issues in accordance with licensee procedures. The inspectors also reviewed recent self-assessment results.

Licensee activities associated with the use of engineering controls and respiratory protection equipment were reviewed against TS Section 5.4; 10 CFR Part 20; Regulatory Guide 8.15, “Acceptable Programs for Respiratory Protection,” and applicable licensee procedures. Documents reviewed are listed in the report Attachment.

b. Findings

No findings were identified.

2RS4 Occupational Dose Assessment

a. Inspection Scope

External Dosimetry The inspectors reviewed the licensee's national voluntary accreditation program (NVLAP) certification data for accreditation for the current year for ionizing radiation dosimetry. The inspectors reviewed program procedures for processing EDs and onsite storage of optically stimulated luminescent dosimeters (OSLD)s. Comparisons between ED and OSLD results, including correction factors, were discussed in detail. The inspectors also reviewed dosimetry occurrence reports regarding alarming dosimeters.

Internal Dosimetry Inspectors reviewed and discussed the *in vivo* bioassay program with the licensee. Inspectors reviewed procedures that addressed methods for determining internal or external contamination, releasing contaminated individuals, the assignment of dose, and the frequency of measurements depending on the nuclides. Inspectors reviewed and evaluated a sample of whole body counter (WBC) records selected from September 2012 through February 2014. There were no internal dose assessments for internal exposure greater than 10 millirem committed effective dose equivalent to review.

The inspectors evaluated the licensee's program for *in vitro* monitoring, however, no dose assessments had been performed using this method since the last inspection.

Special Dosimetric Situations The inspectors reviewed records for declared pregnant workers (DPW)s from September 2012 through February 2014 and discussed guidance for monitoring and instructing DPWs. Inspectors reviewed and witnessed the licensee's practices for monitoring external dose in areas of expected dose rate gradients, including the use of multi-badging and extremity dosimetry. The inspectors evaluated the licensee's neutron dosimetry program including instrumentation which was evaluated under procedure 71124.05. In addition, the inspectors evaluated the adequacy of procedures and processes for assessing shallow dose.

Problem Identification and Resolution The inspectors reviewed and discussed licensee corrective action program documents associated with occupational dose assessment. Inspectors evaluated the licensee's ability to identify and resolve the identified issues in accordance with procedure NPM-GM-002, "Corrective Action Program", Ver. 12.1. The inspectors also discussed the scope of the licensee's internal audit program and reviewed recent assessment results.

Health physics program occupational dose assessment activities were evaluated against the requirements of UFSAR Section 12; TS Section 5.4; 10 CFR Parts 19 and 20; and approved licensee procedures. Records reviewed are listed in Section 2RS01, 2RS02, and 2RS04 of the report Attachment.

b. Findings

No findings were identified.

2RS5 Radiation Monitoring Instrumentation

a. Inspection Scope

Radiation Monitoring Instrumentation: During walk-downs of the auxiliary building, radwaste processing building, fuel handling building and the RCA exit points, the inspectors observed installed and portable radiation detection equipment. These included area radiation monitors (ARM)s, continuous air monitors (CAMs), PCMs, SAMs, PMs, and liquid and gaseous effluent monitors, a WBC, count room equipment, and portable survey instruments. The inspectors observed the physical location of the components, noted the material condition, noted flow measurement devices, input and output of flow to monitors and compared sensitivity ranges with UFSAR requirements. In addition to equipment walkdowns, the inspectors observed source checks and alarm setpoint testing of various portable and fixed detection instruments including ion chambers, a telepole, GEM™-5s, ARGOS™-ABs, and SAMs. Material condition of source check devices, device operation, and establishment of source check acceptance ranges were also discussed with calibration lab personnel.

Calibration and Testing: The inspectors reviewed the last two calibration records for selected ARMs, PCMs, PMs, SAMs, and containment high-range ARMs and the most recent calibration record for a WBC. Inspectors reviewed records of survey instrument function/source checks and observed and discussed performance of required checks with calibration lab personnel. Calibration source documentation was reviewed for the ARM high-range calibrator and the Cs-137 (J.L. Shepherd) source used for portable instrument checks. Calibration stickers on portable survey instruments were reviewed and inspections of storage areas for 'ready-to-use' equipment were completed during walkdowns. The inspectors reviewed alarm setpoint values for selected ARMs, PCMs, PMs, SAMs, and effluent monitors. The inspectors also reviewed count room quality control records for germanium detectors and liquid scintillator counters.

Problem Identification and Resolution: The inspectors reviewed selected CAP reports in the area of radiological instrumentation. The inspectors evaluated the licensee's ability to identify and resolve the issues in accordance with procedure NMP-GM-002-001, "Corrective Action Program Instructions", Ver. 31.1.

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 reviewed during the inspection are listed in the report Attachment.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

a. Inspection Scope

The inspectors reviewed a sample of the performance indicator (PI) data, submitted by the licensee, for the Unit 1 and Unit 2 PIs listed below. The inspectors reviewed plant records compiled between April 2013 and March 2014 to verify the accuracy and completeness of the data reported for the station. The inspectors verified that the PI data complied with guidance contained in Nuclear Energy Institute 99-02, "Regulatory Assessment Performance Indicator Guideline," and licensee procedures. The inspectors verified the accuracy of reported data that were used to calculate the value of each PI. In addition, the inspectors reviewed a sample of related corrective action documents to verify the licensee was identifying and correcting any deficiencies associated with PI data. Documents reviewed are listed in the Attachment.

Cornerstone: Barrier Integrity

- reactor coolant system leak rate
- reactor coolant system specific activity

Cornerstone: Occupational Radiation Safety

The inspectors reviewed the occupational exposure control effectiveness PI results for the occupational radiation safety cornerstone from January 2013 through December 2013. For the assessment period, the inspectors reviewed ED alarm logs and CRs related to controls for exposure significant areas. Documents reviewed are listed in the report Attachment.

Cornerstone: Public Radiation Safety:

The inspectors reviewed the radiological control effluent release occurrences PI results for the public radiation safety cornerstone from January 2013 through December 2013. The inspectors reviewed cumulative and projected doses to the public contained in liquid and gaseous release permits and CRs related to radiological effluent technical specifications/offsite dose calculation manual issues. The inspectors also reviewed licensee procedural guidance for collecting and documenting PI data. Documents reviewed are listed in the report Attachment.

b. Findings

No findings were identified.

4OA2 Identification and Resolution of Problems (71152)

.1 Routine Review

The inspectors screened items entered into the licensee's corrective action program in order to identify repetitive equipment failures or specific human performance issues for follow-up. The inspectors reviewed condition reports, attended screening meetings, or accessed the licensee's computerized corrective action database.

.2 Semi-Annual Trend Review

a. Inspection Scope

The inspectors reviewed issues entered in the licensee's corrective action program and associated documents to identify trends that could indicate the existence of a more significant safety issue. The inspectors focused their review on equipment issues, but also considered the results of inspector daily condition report screenings, licensee trending efforts, and licensee human performance results. The review nominally considered the 6-month period of January 2014 through June 2014 although some examples extended beyond those dates when the scope of the trend warranted. The inspectors compared their results with the licensee's analysis of trends. Additionally, the inspectors reviewed the adequacy of corrective actions associated with a sample of the issues identified in the licensee's trend reports. The inspectors also reviewed corrective action documents that were processed by the licensee to identify potential adverse trends in the condition of structures, systems, and/or components as evidenced by acceptance of long-standing non-conforming or degraded conditions. Documents reviewed are listed in the Attachment.

b. Findings and Observations

No findings were identified.

.3 Annual Follow-up of Selected Samples

a. Inspection Scope

The inspectors conducted a detailed review of condition report CR 648248, "Calculation Error Affects Emergency Action Level (EAL) Setpoints."

The inspectors evaluated the following attributes of the licensee's actions:

- complete and accurate identification of the problem in a timely manner
- evaluation and disposition of operability/reportability issues
- consideration of extent of condition, generic implications, common cause, and previous occurrences
- classification and prioritization of the problem
- identification of root and contributing causes of the problem

- identification of any additional condition reports
- completion of corrective actions in a timely manner

Documents reviewed are listed in the Attachment.

b. Findings

Introduction: A White finding and associated violation of 10 CFR 50.54(q)(2) was identified by the licensee for the failure to follow and maintain the effectiveness of emergency plans which meet the requirements of 10 CFR 50.47(b)(4). Specifically, the licensee's emergency classification scheme action levels for Category R – Abnormal Radiological General Emergency Action Level RG1 and Site Area Emergency Action Level RS1 contained declaration threshold values which were significantly higher than appropriate due to a calculation error.

Description: In March 2005 Southern Co. corporate engineering calculation, X6CNA14, V3.0, was developed to estimate dose rates as a function of radiological releases correlated to radiation monitor values. The calculation provided radiation monitor threshold values for General Emergency (i.e. exceeding 1000 mrem TEDE/5000 mrem thyroid CDE beyond the site boundary) and Site Area Emergency (i.e. exceeding 100 mR TEDE/500 mrem thyroid CDE beyond the site boundary). The calculation was a manual calculation using a spreadsheet program; however, a unit conversion (Sieverts/second to mrem/hour) was made incorrectly and not detected during the review process. The error resulted in threshold values sixty times greater than appropriate. In 2005, Vogtle Electric Generating Plant submitted a license amendment request to the NRC to change their EAL scheme to one based on NEI 99-01, "Development of Emergency Action Levels for Non-Passive Reactors," Rev. 4 guidelines. The request included EAL threshold values for RG1 and RS1 which were based on the errant calculation. The NRC approved the amendment and the licensee implemented the EAL scheme by issuing Revision 29 of Vogtle procedure 91001-C, "Emergency Classification and Implementing Instructions," on March 20, 2008. The non-conservative threshold values were contained in this implementing procedure.

During an extent of condition review prompted by Southern Co. fleet operating experience, calculation X6CNA14, V3.0 was found to contain the calculation error. On May 31, 2013, this issue was placed in the licensee's corrective action program as CR 648248. The licensee took immediate corrective actions, which included providing corrected threshold values to appropriate management and decision-makers (shift managers/emergency directors). In addition, the licensee performed an enhanced apparent cause determination per the licensee's procedures, processed formal changes to the station emergency plan and associated implementing procedures, and performed additional extent of condition/cause reviews throughout the Southern Co. fleet. NRC regional inspectors were advised of the issue and intended plan-of-action. Following extensive review, the revised emergency plan and associated implementing procedure were provided to the NRC in September 2013.

These discrepant threshold values degraded the licensee's ability to make timely and accurate General Emergency and Site Area Emergency classifications based on the abnormal radiological initiating condition, in that decision-makers would have to rely on other means to classify the event (e.g. dose assessments or field monitoring data) and that could delay such a declaration.

Analysis: The inspectors concluded that the failure to maintain the effectiveness of an emergency plan to meet the requirements of 10 CFR 50.47(b)(4) and Part 50 Appendix E to have a standardized EAL scheme in use based on facility system and effluent parameters, was a performance deficiency. The performance deficiency was determined to be more than minor because it was associated with the emergency preparedness cornerstone attribute of procedure quality. It impacted the cornerstone objective because it was associated with inappropriate EAL and emergency plan changes and their adequacy to protect the health and safety of the public in the event of a radiological emergency. Specifically, the licensee's ability to declare a Site Area Emergency and General Emergency based on effluent radiation monitor values was degraded in that event classification using these radiation monitors would be delayed. The finding was assessed for significance in accordance with NRC Manual Chapter 0609, Appendix B, "Emergency Preparedness Significance Determination Process," which states, "Failure to comply means that a program is noncompliant with a Regulatory requirement." The inspector determined the licensee was noncompliant with 10 CFR 50.54(q), 50.47(b)(4), and Appendix E, Section IV.B in that, due to a calculation error, the abnormal radiological initiating conditions RG1(General Emergency) and RS1 (Site Area Emergency) emergency action levels contained classification threshold values sixty times greater than the appropriate value. This would require use of other means (dose assessment or actual field readings) to determine whether a Site Area Emergency or General Emergency threshold had been exceeded which could delay the declaration. The inspector determined that the situation constituted a degraded rather than lost risk-significant planning standard (RSPS). The issue of concern was similar to the example in Table 5.4.1 (Degraded RSPS) and was determined to be of low to moderate safety significance (White). The licensee took immediate corrective actions providing corrected threshold values to appropriate management and decision-makers (shift managers/emergency directors). These and additional corrective actions were placed in the licensee's corrective action program as CR 648248. A cross-cutting aspect was not assigned based on the elapsed time since the performance deficiency occurred and because the inspectors determined it was not reflective of current licensee performance.

Enforcement: 10 CFR 50.54(q)(2), requires that a holder of a nuclear power reactor operating license under this part, shall follow and maintain the effectiveness of emergency plans which meet the standards in 10 CFR 50.47(b), and the requirements in Appendix E of this part.

10 CFR 50.47(b)(4), requires a standard emergency classification and action level scheme, the bases of which include facility and system effluent parameters in use by the nuclear facility licensee, and state and local response calls for reliance on information by facility licensees for determinations of minimum initial offsite response measures.
10 CFR Part 50, Appendix E, Section IV.B., "Assessment Actions," requires that means to be used for determining the magnitude of, and for continuously assessing the impact

of, the release of radioactive materials shall be described, including emergency action levels that are to be used as criteria for determining the need for notification and participation of local and state agencies, the Commission, and other federal agencies. The emergency action levels shall be based on in-plant conditions and instrumentation in addition to onsite and offsite monitoring.

Contrary to the above, from March 2008 to May 2013, the licensee failed to maintain the effectiveness of its emergency plan. The licensee failed to maintain a standard emergency classification scheme which included facility effluent parameters. Specifically, the emergency classifications RG1 (General Emergency) and RS1 (Site Area Emergency) contained effluent radiation monitor threshold values significantly greater than appropriate. These monitors were being relied upon to determine the magnitude and for continuously assessing the impact of the release of radioactive materials, as well as providing criteria for determining the need for notification and participation of local and state agencies. Following review by a Significance Enforcement Review Panel and NRC management, the violation was determined to meet IMC 0305, Section 11.05, criteria for discretion as an old design issue. Specifically, the issue was licensee-identified through an extent-of-condition review of internal operating experience, the issue was immediately corrected by the licensee, the issue was not likely to be previously identified during normal operations, routine testing, or maintenance, and the issue is not reflective of current licensee performance. As such, this finding will not be used as an input in the assessment process or NRC Action Matrix. This finding has been identified as a cited violation 05000424, -425/2014003-03, "Calculation Error Results in Significantly non-Conservative EAL Threshold Values." This is a violation of 10 CFR 50.54(q)(2) and a Notice of Violation is enclosed. (Enclosure 2)

.4 Operator Work-Around Annual Review

a. Inspection Scope

The inspectors performed a detailed review of the licensee's operator work-around, operator burden, and control room deficiency lists for the station in effect on June 16, 2014 to verify that the licensee identified operator workarounds at an appropriate threshold and entered them in the corrective action program. The inspectors verified that the licensee identified the full extent of issues, performed appropriate evaluations, and planned appropriate corrective actions. The inspectors also reviewed compensatory actions and their cumulative effects on plant operation. Documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

4OA3 Event Follow-up

.1 (Closed) Licensee Event Report 05000425/2014-001-00: Automatic Reactor Trip Due to Low Steam Generator Level

a. Inspection Scope

On April 08, 2014, with Unit 2 in Mode 1, 100 percent reactor power, at approximately 04:28, operators received unexpected annunciators, “Digital Feedwater Trouble Alarm” for all four steam generators. Upon further investigation, operators noted loop 3 steam generator water level was lowering at rapid rate. The operator at the controls (OATC) took manual control of the loop 3 MFRV and attempted to raise water level. Water level continued to decrease to the SG low-low level reactor trip setpoint and an automatic reactor trip occurred as expected. The inspectors reviewed the licensee event report (LER), the associated condition report and root cause determination, and subsequent action items for potential performance deficiencies and/or violations of regulatory requirements. Additionally, discussions were held with operations, engineering and licensing staff members to understand the details surrounding this issue. This condition was documented in the licensee’s corrective action program as CR 797929. This LER is closed.

b. Findings

Introduction: A Green, self-revealing non-cited violation (NCV) of 10 CFR 50 Appendix B Criterion V, “Instructions, Procedures, and Drawings,” was identified for failure to provide adequate work instructions as well as failure to follow the maintenance procedure used to install flexible and rigid conduit. Specifically, the work instructions did not provide adequate instructions and/or precautions to properly slope conduit during installation to prevent water intrusion into a valve positioner. The work instructions referenced maintenance procedure 25008-C, “Flexible and Rigid Conduit Installation.” The maintenance procedure referenced Vogtle design specification X3AR01 Section E-8, “Raceway Systems,” which provided sloping and tightness criteria for conduit installations.

Description: On April 08, 2014, with Unit 2 in Mode 1, 100 percent reactor power, at approximately 04:28, operators received unexpected annunciators, “Digital Feedwater Trouble Alarm,” for all four steam generators. Upon further investigation, operators noted loop 3 steam generator water level was lowering at rapid rate. The operator at the controls (OATC) took manual control of the loop 3 MFRV and attempted to raise water level. Water level continued to decrease to the SG low-low level reactor trip setpoint and an automatic reactor trip occurred as expected. The plant was stabilized in Mode 3 and all safety related equipment responded as expected. Loop 3 SG water level was restored using auxiliary feedwater. The licensee subsequently assembled an issue response team (IRT) and a root cause team to investigate the cause of the automatic reactor trip due to the failure of the loop 3 MFRV and to determine the required corrective actions. Further investigation revealed water had entered the loop 3 MFRV positioner junction box through a conduit penetration from a leaking valve located approximately twenty feet above the junction box. The water had shorted out the valve positioner and caused the MFRV to go shut. The licensee had identified the leak one month before the incident and had entered it into their corrective action program, but had not yet entered it into the work control process. The licensee determined the conduit connection was loose and not installed per design specification drawing AX2D94V077-3, “Digital Feedwater Flow Controller Instrument Support Details,” Rev. 1.0. The specification drawing shows the conduit being routed to the underside of the junction versus the top where it was installed. A combination of the loose conduit connection combined with improper conduit installation resulted in the leaking water entering the positioner junction box shorting the MFRV positioner and causing the MFRV to close. Further research by the root cause team revealed that during digital feedwater design modification installation, the work instructions used by the maintenance technician to install the flexible conduit was inadequate. Specifically, the work instructions did not contain sufficient detail to properly slope the conduit to prevent water intrusion. The work instructions referenced maintenance procedure 25008-C, “Flexible and Rigid Conduit Installation.” The maintenance procedure directed the use of specification X3AR01 Section E-8, “Raceway Systems,” which contained proper sloping and tightness criteria. The licensee replaced the positioner, revised the procedure, and rerouted the conduit per design specification. The licensee entered this issue into their corrective action program as CR 797929.

Analysis: The failure to provide adequate work instructions as well as the failure to follow maintenance procedure 25008-C as required by 10 CFR 50 Appendix B Criterion V was a performance deficiency. The inspectors determined that the finding was more than minor because it was associated with the procedure quality and human performance attributes of the initiating events cornerstone and it adversely affected the cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, the failure to provide adequate work instructions resulted in a failure of the loop 3 MFRV and a subsequent automatic reactor trip due to low-low SG water level. Using IMC 0609, Attachment 4, “Initial Characterization of Findings” dated June 19, 2012, the inspectors determined that the finding affected the initiating events cornerstone. The inspectors evaluated the finding using IMC 0609, Appendix A, “The Significance Determination Process (SDP) for Findings At-Power,” dated June 19, 2012. Because the inspectors answered “No” to all the Exhibit 1, Section B, “Initiating Events Screening Questions,” the inspectors determined that the finding was of very low safety significance (Green). The inspectors determined that the finding had a cross-cutting aspect of “procedure adherence” in the human performance area because the maintenance electricians did not follow Vogtle design specification procedures or drawings resulting in the improper sloping of the MFRV flexible conduit. [H.8]

Enforcement: 10 CFR 50 Appendix B Criterion V requires, in part, that procedures shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Contrary to the above, the maintenance procedure used to install the flexible conduit during the Unit 2 digital feedwater design change modification installation did not provide appropriate instructions for the sloping and tightening of the conduit thus preventing water intrusion into the loop 3 MFRV positioner junction box. Specifically, maintenance procedure 25008-C, “Flexible and Rigid Conduit Installation,” which is used to install conduit, did not provide adequate instructions and/or precautions to properly slope and tighten conduit such that water intrusion is avoided. To restore compliance, the licensee replaced the positioner, revised the procedure, and rerouted the conduit per the design specification. This violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy. The violation was entered into the licensee's corrective action program as CR 797929. (NCV 05000425/2014003-04, "Inadequate Maintenance Procedures and Usage Results in a Failed MFRV and an Automatic Reactor Trip")

.2 (Closed) Licensee Event Report 05000424/2014-002-00: Manual Reactor Trip Due to Main Steam Isolation Valve Failure

a. Inspection Scope

On April 12, 2014 Unit 1 was in Mode 1 ascending in power after the 1R18 refueling outage. At approximately 20:08, control room operators received an MSIV actuator trouble alarm followed by the MSIV not fully open indication. Control room operators identified lowering loop 1 steam generator (SG) #1 level and steam flow and manually tripped Unit 1 at about 28 percent reactor power. The inspectors reviewed the LER, the associated condition report and root cause determination, and subsequent action items. This condition was documented in the licensee's corrective action program as CR 800018. This LER is closed.

b. Findings

The enforcement aspects associated with this event are discussed in Section 1R12 of this integrated inspection report.

4OA5 Other Activities

.1 (Closed) Unresolved Item 05000425/2013007-02: Failure to Identify and Correct Potential Emergency Diesel Generator "2B" Inoperability Following Failed Surveillance Testing

a. Inspection Scope

During the component design bases inspection documented in NRC Inspection Report 05000424, 425/2013007 (ADAMS ML13269A419), the team identified an unresolved item (URI) regarding the discovery of a condition that could have potentially resulted in an inoperable condition of the "2B" EDG due to an intermittently misaligned mechanically operated cell (MOC) switch. Since the licensee had not recognized the potential operability impact on the "2B" EDG during their investigations of EDG surveillance test failures on December 13, 2011, and June 25, 2012, additional NRC inspection of the specific alignment of the affected MOC switch contacts, and of the licensee's evaluation of operability of the "2B" EDG, prior to the MOC switch being adjusted, was necessary to determine if the issue of concern was minor or more than minor. On June 16, 2014, NRC inspection of the MOC switch contacts was performed to determine if the proper functioning of the "2B" EDG, during emergency mode of operation, would have been affected. Based on this additional review, this URI is now closed.

b. Findings

No findings were identified. However, the inspectors identified a minor performance deficiency and associated minor violation of 10 CFR 50, Appendix B, Criterion XVI, “Corrective Action.” In accordance with IMC 0612, “Power Reactor Inspection Reports,” dated January 24, 2013, minor violations are not routinely documented in inspection reports. However, they may be documented to discuss inspection activities and conclusions for closing a URI.

The inspectors determined that the licensee’s failure to promptly identify and correct a misaligned MOC switch associated with the “2B” EDG output breaker following a surveillance test failure on December 13, 2011, was contrary to 10 CFR 50, Appendix B, Criterion XVI, and was a performance deficiency. This failure led to a small amount of additional unavailability to troubleshoot the issue following an additional failure on June 25, 2012. Following additional NRC inspection on June 16, 2014, the inspectors determined the actual radial alignment of the MOC switch contacts would have supported the proper functioning of the EDG if it had been called upon during an event. Using IMC 0612, Appendix B, “Issue Screening,” dated September 7, 2012, the inspectors determined the issue was of minor significance because, if left uncorrected, would not have led to a more significant safety concern. The licensee corrected the condition of the misaligned MOC switch following the second failure on June 25, 2012. Because this issue was entered into the licensee’s corrective action program as CR 687752, and was of minor significance, the failure to comply with 10 CFR 50, Appendix B, Criterion XVI, “Corrective Action,” constituted a minor violation that is not subject to enforcement action in accordance with the NRC’s Enforcement Policy.

4OA6 Meetings, Including Exit

.1 Exit Meeting

On July 25, 2014, the resident inspectors presented the inspection results to Mr. T. Tynan and other members of the licensee’s staff. The inspectors confirmed that proprietary information was not provided or examined during the inspection.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel:
R. Barringer, Security Manager
R. Collins, Chemistry Manager
H. Cooper, Engineering Programs Supervisor
J. Dixon, Corporate Fleet Area Manager, Health Physics
G. Gunn, Licensing Supervisor
M. Hayden, EP Manager
R. Hons, Training Manager
M. Johnson, Health Physics Manager
K. Morrow, Licensing
F. Pournia, Engineering Director
J. Robinson, Engineering Programs Manager
I. Sarygin, Sr. Engineer
G. Saxon, Plant Manager
J. Thomas, Work Management Director
T. Thompson, Systems Engineering Manager
T. Tynan, Site Vice-President
K. Walden, Licensing Engineer
S. Waldrup, Licensing Director

NRC personnel:
F. Ehrhardt, Chief, Region II Reactor Projects Branch 2

LIST OF ITEMS OPENED AND CLOSED

Opened

05000424,425/2014003-03	VIO	Calculation Error Results in Significantly Non-Conservative EAL Threshold Values (Section 4OA2.3)

Open and Closed

05000424/2014003-01	NCV	Inadequate Maintenance Procedure Results in a Failed MSIV and a Manual Reactor Trip (Section 1R12)
05000424/2014003-02	NCV	Unauthorized Entry into a High Radiation Area (Section 2RS1)
05000425/2014003-04	NCV	Inadequate Maintenance Procedures and Usage Results in a Failed MFRV and an Automatic Reactor Trip (Section 4OA3.1)

Closed

05000425/2014-001-00	LER	Automatic Reactor Trip Due to Low Steam Generator Level (Section 4OA3.1)
05000424/2014-002-00	LER	Manual Reactor Trip due to Main Steam Isolation Valve Failure (Section 4OA3.2)
05000425/2013007-02	URI	Failure to Identify and Correct Potential Emergency Diesel Generator 2B Inoperability Following Failed Surveillance Testing (Section 4OA5)

LIST OF DOCUMENTS REVIEWED

Section 1R01: Adverse Weather Protection
Procedures
11889-C Rev. 21, Severe Weather Checklist
VNP-CMS-710-00-PR-00001 Rev. 0, CB&I Health, Safety and Environmental Management System (Units 3&4)
VNP-CMS-710-03-PR-00400 Rev. 0, CB&I Emergency Preparedness Plan (Units 3&4)
14230-1, Rev. 23.0, Offsite AC Circuit Verification and Capacity/Capability Evaluation
14230-2, Rev. 22.0, Offsite AC Circuit Verification and Capacity/Capability Evaluation
18017-C, Rev. 9.6, Abnormal Grid Disturbances/Loss of Grid
13830-1, Rev. 69.0, Main Generator Operation
13830-2, Rev. 55.0, Main Generator Operation

Section 1R04: Equipment Alignment
Procedures
11145-2 Rev. 12.2, Diesel Generator Alignment
11146-2 Rev. 7.1, Diesel Generator Fuel Oil Transfer System Alignment
11610-2 Rev. 21.3, Auxiliary Feedwater System Alignment

Drawings
2X4DB170-1 Rev. 42.0, P&I Diagram Diesel Generator System Train A – System No. 2403
2X4DB170-2 Rev. 47.0, P&I Diagram Diesel Generator System Train B – System No. 2403
2X4DB161-1, P&I Diagram Rev. 36.0, Auxiliary Feedwater System Condensate Storage & Degasifier System, System No. 1302
2X4DB161-2, P&I Diagram Rev. 24.0, Auxiliary Feedwater System, System No. 1302
2X4DB161-3, P&I Diagram Rev. 38.0, Auxiliary Feedwater Pump System, (Aux Feedwater Pump Turbine Driver) System No. 1302
2X4DB168-3, P&I Diagram Rev. 37.0, Condensate and Feedwater System, System No. 1305

Section 1R05: Fire Protection
Procedures
92754-2 Rev. 0.2, Zone 54 – Auxiliary Building – Level 2 Train “A” CCW HX Fire Fighting Preplan

92755-2 Rev. 0.2, Zone 55 – Auxiliary Building – Level 2 Train “B” CCW HX Fire Fighting Preplan
92848-2 Rev. 0.2, Zone 148 – Auxiliary Building – Level 2 Fire Fighting Preplan
92723-2 Rev. 2.1, Zone 23 – Auxiliary Building – Electrical Chase Rooms Fire Fighting Preplan
92872-2 Rev. 1.2, Zone 172 – Auxiliary Building – Level 2 Fire Fighting Preplan
92847-2 Rev. 1.2, Zone 147 – Auxiliary Building – Level 2 Fire Fighting Preplan
92714B-1 Rev. 2.2, Zone 14B – Auxiliary Building – Level C Fire Fighting Preplan
92719-1 Rev. 4.1, Zone 19 – Auxiliary Building – CVCS Centrifugal Charging Pump Rooms Fire Fighting Preplan
92720-1 Rev. 4.1, Zone 20 – Auxiliary Building – CVCS Pump Rm Train A Fire Fighting Preplan
92721-1 Rev. 5.1, Zone 21 – Auxiliary Building – CVCS NCP Room Fire Fighting Preplan
92789-2 Rev. 3.1, Zone 89 – Control Building – Level A Fire Fighting Preplan
92790-2 Rev. 2.2, Zone 90 – Control Building – Level A Fire Fighting Preplan
92859-2 Rev. 1.2, Zone 159 – Control Building – Level A Fire Fighting Preplan
92787-2 Rev. 2.2, Zone 87 – Control Building – Level A Fire Fighting Preplan
92788-2 Rev. 2.2, Zone 88 – Control Building – Level A Fire Fighting Preplan
92793-2 Rev. 3.2, Zone 93 – Control Building – Level A Fire Fighting Preplan
92802-2 Rev. 2.2, Zone 102 – Control Building – Level A Fire Fighting Preplan
92858-2 Rev. 1.2, Zone 158 – Control Building – Level A Fire Fighting Preplan
92862-1 Rev. 2.2, Zone 162 – Diesel Generator Building Fire Fighting Preplan
92864-1 Rev. 2.2, Zone 164 – Diesel Generator Building – Train B DFO Tank Fire Fighting
92855-2, Rev. 0.2, Zone 155 – Auxiliary Feedwater Pumphouse – Train B Fire Fighting Preplan
92856-2, Rev. 0.2, Zone 156 – Auxiliary Feedwater Pumphouse Fire Fighting Preplan
92857A-2, Rev. 0.2, Zone 157A – Auxiliary Feedwater Pumphouse – Train C Fire Fighting Preplan
92857B-2, Rev. 0.2, Zone 157B – Auxiliary Feedwater Pumphouse – Train C Fire Fighting Preplan

Section 1R06: Internal Flooding
Procedures
13219-1 Rev. 35, Auxiliary and Containment Buildings and Miscellaneous Drain Systems

Other
X6CXC-27 Rev.8, Flooding Analysis Auxiliary Building Level D
AX1D94A56 Rev. 2.0, Auxiliary Building Units 1 & 2 Door Schedule Level D
CCN-V-07-0011 Rev. 8.0, Flooding - Auxiliary Building Level D

Drawings
AX1D08A02-2, Rev. 6.0, Auxiliary Building Floor Plan El. 119 Level D

Section 1R11: Licensed Operator Requalification Program
Procedures
12003-C Rev. 53, Reactor Startup Mode 3 to Mode 2
12004-C Rev.107.2, Power Operation Mode 1
NMP-OS-007-001 Rev. 14.3, Conduct of Operations Standards and Expectations

Other
Simulator scenario V-RQ-SE-12702, Loss of Grid/Natural Circulation Cooldown
Simulator scenario V-RQ-SE-14300, Performance Improvement Exercise
Simulator scenario V-RQ-SE-14301, Large Break LOCA Response
Simulator scenario V-RQ-SE-14302, SGTL/SGTR/Recovery
Simulator scenario V-RQ-SE-14303, Control Room Evacuation

Section 1R12: Maintenance Rule Effectiveness
Condition Reports and Action Items
807906, MPFF documented for Unit 2, System 1305, 2HV5230
795933, Unexpected control room annunciator ALB16-D04, MFIV Loop 4 low hydraulic pressure

Section 1R15: Operability Evaluations
Condition Reports
776584, Unknown chemical buildup on top of upper motor windings
808990, 2B DG Jacket Water Leak
805473/CAR 210188, 1HV3036A MSIV control board ‘red’ light flickering
807567/CAR 210214, Unit 2 turbine driven auxiliary feedwater pump (TDAFW) controller output reading low
CR 607966, U1 CCW Pump 1A inboard bearing over 160F

Other Records
TE 776816, IDO request for 1A RHR pump
TE 776799, “OBDN resolution” 1A RHR IDO comp action
TE 776807, “OBDN resolution” 1A RHR IDO comp action
EMI Diagnostics Report for Plant Vogtle 1 and 2 Electric Generating Plant, by Doble Global Power Services, PO# SNG10075822 dated 3/10/2014 initiative
CAR 210245, IDO - 2B DG Jacket Water Leak
MWO SNC572497, 2B DG Jacket Water Leak
CAR 210188, IDO - 1HV3036A MSIV
CAR 210214, IDO - Unit 2 turbine driven auxiliary feedwater pump (TDAFW) controller
MWO SNC525698, Troubleshoot Unit 2 TDAFW controller output
TE 767342, IDO revision for CR 607966

Section 1R18: Plant Modifications
Procedures
NMP-AD-010 Rev. 13.0, 10 CFR 50.59 Screening/Evaluation
NMP-ES-054-001 Rev. 2.0, Temporary Modification Processing

Work Orders
SNC417397, Temporary modification to install accelerometers and a pressure transducer on CVCS letdown lines, Unit 1
1081013501, Accelerometer Installation at the CVCS letdown flow orifices and line 1-1208-255-3”, 6/13/2008
DCP 98-VAN0055, Replace the Alternate Radwaste Building (ARB) with the Radwaste Processing Facility (RPF)

Drawings
AX3D-CH-T01J, Wiring Diagram Alternate Radwaste Building and ABB Control Room Misc Devices
AX3D-BC-G20C, Elementary Diagram Alternate Radwaste Building Cabling Block Diagram Rad Monitors, HVAC, Bridge Crane
AX3DH469-1, Wiring Diagram Alternate Radwaste Building Control Room Conduit and Lighting and Communications Plans Sheet 001

Corrective Action Documents
Condition Report (CR) 2008106194, Walk down of Unit 1 containment for increased leakage discovered upstream of letdown orifice isolation valve 1HV8149A, 6/1/2008
Technical Evaluation (TE) 34658, Corrective Action to establish a replacement interval for the letdown flow orifices, 5/7/2009
TE 14379, Corrective action to replace the Unit 1 letdown flow orifices with a butt weld connection orifice during the refueling outage 15, 9/11/2008
Enhanced Apparent Cause Determination (EACD) 194554, Station personnel failed to implement the corrective action program to resolve an uncontrolled change in which area radiation monitors were permanently removed from the Alternate Radwaste Building (ARB).
Technical Evaluation (TE) 363628, Revise procedure NMP-GM-002—001 Attachment 1 to provide guidance for screening CRs that include design document aspects and configuration control issues.
TE 366691, Generate an LDCR to update the FSAR to reflect the ARE-16851, ARE-16852, ARE-16853, ARE-16854 as being no longer in service.
TE 366715, Complete and approve an ABN to update any associated documents to reflect the ARE-16851, ARE-16852, ARE-16853, ARE-16854 as being no longer in service.
TE 367763, Properly label as abandoned in place or remove all remnants of the ARB rad monitor system that is no longer in use.

Other
VEGP-FSAR-11, Radioactive Waste Management
VEGP-FSAR-12, Radiation Protection
ABN-V03007, Incorporate PDMS changes per DEC DBN-V03007
LDCR No. 2012017, Update the FSAR to reflect the ARE-16851, ARE-16852, ARE-16853 and ARE-16854 as being no longer is Service.

Section 1R19: Post Maintenance Testing
Procedures
14825-2 Rev. 94, Quarterly Inservice Valve Test
14825-2 Rev. 95, Quarterly Inservice Valve Test
14430-1 Rev. 11.0, NSCW Cooling Tower Fans Monthly Test
24449-2 Rev. 9, Diesel Generator Power Out Train 2Q-2791 Channel Calibration
24812-1 Rev. 44, Unit 1 Delta T/Tavg loop 3 protection channel III 1T 431 operational test and calibration
24810-2 Rev. 36, Unit 2 Delta T/Tavg loop 1 protection channel I 2T-411 operational test and calibration

Work Orders
SNC137725 – Replacement of 1E D26 Relays MCC21805S3ABE
SNC413540 – 2PV3020 Replace A/B Solenoid
SNC527135 – Quarterly Steam Generator Atmospheric Relief Valve Inservice Valve Test
SNC507135 – Manually stroke 2PV3020 from the local control station and perform ARV fail safe test per 14825-2
SNC408041 – (1A NSCW Fan 2) – Replace Agastat Relay
SNC383989 – (1A NSCW Fan 2) – Replace Rubber Bushings on Fan Couplings
SNC525486 – Unit 2A EDG Undervoltage Relay Calibration
SNC516991 – Unit 1 Delta T/Tavg loop 3 protection channel operational test and calibration
SNC488414 - Unit 2 Delta T/Tavg loop 1 protection channel I 2T-411 operational test and calibration

Other Records
Unit 2 operator logs for 4/14/14
Unit 2 operator logs for 4/26/14
Unit 2 ARV 3020 system outage fragnet
Unit 1 operator logs for 5/12/14
1A NSCW Fan 2 system outage fragnet

Section 1R22: Surveillance Testing
Procedures
14802A-2 Rev. 5, Train A NSCW Pump / Check Valve IST and Response Time Test
24568-2 Rev. 38, RCP 1 Train A, Reactor Trip Relays Under Frequency (281-A), Under Voltage (227-A), Timing (262R-A) Trip Actuating Device Operational Test and Channel Calibration
24565-2, Rev. 37, RCP 2 Train A, Reactor Trip Relays Under Frequency (281-A), Under Voltage (227-A), Timing (262R-A) Trip Actuating Device Operational Test and Channel Calibration
14804B-1 Rev. 5.0, Safety Injection Pump B Inservice and Response Time Tests
24449-2 Rev. 9, Diesel Generator Power Out Train 2Q-2791 Channel Calibration
21118-2 Rev. 3.2, Centrifugal Charging Pump (CCP) Train A Safety Grade Charging Flow Loop 2F-0138 Channel Calibration
14905-1 Rev. 69.0, RCS Leakage Calculation (Inventory Balance)
14905-2 Rev. 53.0, RCS Leakage Calculation (Inventory Balance)

Work Orders
SNC523019 – Quarterly train A NSCW pump 21202P4005 discharge MOV and check valve inservice test
SNC523018 – Quarterly train A NSCW pump 21202P4003 discharge MOV and check valve inservice test
SNC523447– Quarterly train A NSCW pump 21202P4001 discharge MOV and check valve inservice test
SNC528899, Quarterly train A RCP #1 under voltage and under frequency relays TADOT
SNC405763, 18-month train A RCP #2 under voltage and under frequency relays TADOT
SNC457082, 18M staggered test basis (train B) safety injection pump response time test
SNC520727, Quarterly (train B) safety injection pump and discharge check valve inservice test
SNC525486, Unit 2A EDG Undervoltage Relay Calibration
SNC412442, Centrifugal Charging Pump (CCP) Train A Safety Grade Charging Flow Loop 2F-

0138 Channel Calibration

Section 2RS1: Radiological Hazard Assessment and Exposure Controls

Procedures, Guidance Documents, and Manuals
00008-C, Plant Lock and Key Control, Ver. 16.2
11882-1, Outside Area Rounds Sheets, Ver. 90.2
43014-C, Special Radiological Controls, Ver. 43.5
43021-C, Health Physics Central Monitoring Station Expectation and Guidelines, Rev. 4.4
43022-C, Health Physics Central Monitoring Station, Ver. 5.2
43032-C, Reactor Head and Upper Internals Movement, Ver. 3.2
46100-C, 10 CFR 61 Waste Classification Sampling Program, Ver. 9
46111-C, Storage of Radwaste in Outdoor Process Shields, Ver. 6.1
47009-C, Operation and Use of Portable Ventilation Units, Ver. 22.3
93610-C, Conduct of Special Nuclear Material Control and Accountability, Ver. 11.1
93641-C, Development and Implementation of the Fuel Shuffle Sequence Plan, Ver, 21.1
93780-C, HI-TRAC Contamination Survey, Ver. 1.0
93781-C, HI –TRAC Surface Dose Rates, Ver. 1.0
93782-C, HI-STORM Surface Dose Rates, Ver. 1.0
NMP-GM-002, Corrective Action Program, Ver. 12.1
NMP-GM-002-001, Corrective Action Program Instructions, Ver. 31.1
NMP-HP-109, Investigation, Evaluation and Management of Damaged, Lost, Malfunctioning or Alarming Dosimetry, Ver. 1.1
NMP-HP-202, Radiological Controls for Highly Radioactive Objects, Ver. 1.0
NMP-HP-206, Issuance, Use and Control of Radiation Work Permits, Ver. 3.0
NMP-HP-207, Selection and Use of Protective Clothing, Ver. 1.0
NMP-HP-218, Health Physics Stop Work Authority and Guidance on Response, Ver. 1.0
NMP-HP-300, Radiation and Contamination Surveys, Ver. 2.1
NMP-HP-301, Airborne Radioactivity Sampling and Evaluation, Ver. 2.2
NMP-HP-302, Restricted Area Classification, Postings, and Access Control, Ver. 6.0
NMP-HP-302-001, Radiological Key Control, Ver. 2.1
NMP-HP-303, Personnel Decontamination, Ver. 2.2
NMP-HP-304, Decontamination of Areas, Tools and Equipment, Ver. 1.0
NMP-HP-305, Alpha Radiation Monitoring, Ver. 4.0
NMP-HP-400, Control and Accountability of Radioactive Sources, Ver. 2.0
NMP-HP-403, Control and Monitoring of Materials in Radiation Controlled Areas, Ver. 1.0
NMP-HP-404, Release of Materials from the RCA and Protected Areas, Ver. 1.0
Health Physics Work Plan, Rx Cavity Decon

Records and Data
46100-C, 10 CFR 61 Waste Classification Sampling Program, Ver. 9, Dated 06/12/12
Air Sampler Calibration, Sheet 1 of 3, Data Sheet 1, Air Sampler Calibration Form, Instrument Nos. VEGP-HP-1368, Model No. RAS-1, Dated 03/13/14; VEGP-HP-1369, Model No. RAS-1, Dated 01/09/14; and VEGP-HP-1371, Model No. RAS-1, Dated 12/26/13
Airborne Radioactivity Sampling and Evaluation, Data Sheet 1, Air Sample Record (Particulate and Iodine), Air Sample Nos.: 14-0132, U1 CTMT/220' (Pulling NI Covers in Upper RX Cavity), Dated 03/16/14; 14-0133, U1 EH (Equipment Hatch Routine), Dated 03/16/14; and 14-0157, U1 CTMT/171' (Routine VEGP-HP-1368), Dated 03/20/14
National Source Tracking System, Annual Inventory Reconciliation Report, Vogtle 1, Dated 01/17/14

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<u>CAP Documents</u>
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<u>Section 2RS2: Occupational ALARA Planning and Controls</u>
<u>Procedures, Guidance Documents, and Manuals</u>
16035-1, "Chemistry Operations Interface for RCS Chemistry Control During Scheduled Plant Shutdowns", Ver. 15.2
NMP-AD-035, "ALARA Program", Ver. 1.3
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41006-C, Temporary Shielding, Ver 29.2

NMP-HP-202, Radiological Controls for Highly Radioactive Objects, Ver. 1.0
NMP-HP-206, Issuance, Use and Control of Radiation Work Permits, Ver 3.0

Records and Data
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1R18 Outage Dose Summary Report, dated 03/20/2014
HP Duty Foreman's Checklist – Daily Report Items, dated 03/19/2014
1R18 Temporary Shielding Worksheet, dated 11/21/2013
Plant Vogtle Radiological Information Survey Nos. 168301, Under Vessel Annulus Area (Pre-Shielding), Dated 03/17/2014, 168313, Under Vessel Annulus Area (Pre-Shielding), Dated 03/18/2014, 168345, Under Vessel Annulus Area (Post Shielding), Dated 03/18/2014, 168362, Under Vessel Annulus Area (Post Shielding), dated 03/17/2014, 169356, Reactor Cavity Area for Core Exit Thermocouple Bullet Nose Manual Alignment, dated 04/02/2014, 169077, Lower Reactor Cavity Area for Core Exit Thermocouple Bullet Nose Manual Alignment, dated 03/29/2014,
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Shutdown Chemistry Review: Vogtle Unit 1 Fuel Cycle 17, dated 11/27/2012
U1 EPRI Shutdown Survey Points Trend Graph for Refueling Outages 1R1 – 1R17
U1 S/G Channel Head Dose Rate Trend Graph for Refueling Outages 1R1 – 1R17
U2 EPRI Shutdown Survey Points Trend Graph for Refueling Outages 2R1 – 2R16
U2 S/G Channel Head Dose Rate Trend Graph for Refueling Outages 2R1 – 2R16
VEGP ALARA Strategic Plan 2013 – 2018
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NOSCPA-HP-2012-04, Health Physics Fleet Performance Summary Report, dated 11/26/2012
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NOSCPA-HP-2013-13, Health Physics Fleet Performance Summary Report, dated 12/04/2013
ALARA Committee Meeting Minutes Fourth Quarter 2013
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2R16 ALARA Report
Plant ALARA Review Committee (PARC) “Called Monthly Meeting,” dated 03/21/14
PARC “Called Monthly Meeting,” dated 03/05/14
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<u>CAP Documents</u>
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CR 643120
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<u>Section 2RS3: In-Plant Airborne Radioactivity Control and Mitigation Adam 4/9/14</u>
<u>Procedures, Guidance Documents, and Manuals</u>
47020-C, DOP Testing of HEPA Filters, Ver. 5.2
47004-C, Breathing Air Analysis, Rev. 16
47001-C, Selection and Use of Respiratory Protection Equipment Used for Radiological Purposes, Ver. 19.2
47005-C, Inspection, Repair, and Storage of Respiratory Protection Equipment, Rev. 15
NMP-GM-002-001, Corrective Action Program, Ver. 31.1

<u>Records and Data Reviewed</u>
SCBA Maintenance Records, Kit 58 and HP-0060, January 2012 – December 2013
Respirator Use Evaluation Worksheets, 10/9/13
DOP Test Log Sheets, 3/10/14, 3/20/14, 3/24/14
Breathing Air Analysis Results, Scott Revolve 5016 Compressor, 6/1/13, 8/26/13, 12/6/13, 2/18/14
Breathing Air Analysis Results, U2 Containment Breathing Air, 3/14/13
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Breathing Air Analysis Results, U1 Service Air Compressors 2 & 3, 2/18/14
Breathing Air Analysis Results, U1 Service Air, 8/26/13
Breathing Air Analysis Results, U1 Equipment Hatch Compressor, 3/21/14
Laboratory Report Compressed Air/Gas Quality Testing, Scott Revolve 5016 Compressor, 2/12/13
Laboratory Report Compressed Air/Gas Quality Testing, Hypress FTB Compressor, 2/12/13
List of Maintenance Personnel with SCBA Qualification Assigned, 2/28/14
List of Operations Personnel with SCBA Qualification Assigned, 2/28/14

<u>CAP Documents</u>
Fleet-HP-2013, Nuclear Oversight Audit of Health Physics, 7/15/13
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Section 2RS4: Occupational Dose Assessment

Procedures, Guidance Documents, and Manuals

NMP-HP-107-001, “Instructions for Retrieving, Printing and Updating Individual Radiation Exposure Records”, Ver.1.0
NMP-HP-105, “Comparisons of OSLD and ED Dosimetry Results”, Ver. 1.1
NMP-HP-106, “Investigating of Exposures Exceeding Fleet Administrative Limits”, Ver. 1.0
NMP-HP-103, “Skin Dose Assessment”, Ver. 1.1
NMP-HP-100, “Bioassay Program”, Ver. 1.1
NMP-HP-101, “In-Vivo Bioassay and Internal Dose Assessment”, Ver. 3.0
NMP-HP-102, “In-Vitro Bioassay,” Ver. 1.1
NMP-HP-201, “Personnel Dosimetry Program,” Ver. 1.1
NMP-HP-204, “Use and Calibration of Whole Body Counters,” Ver. 1.3

Records and Data

NVLAP Certification of Accreditation to ISO/IEC 17025:2005, for Lab Code:100551-0, dated 02/13/2013.
Vogtle Alpha Plant Characterization Study 2011 Update
Canberra Report of Performance Testing Results for Nuclear Enterprises (NE) Model SPM 904B/906 Personnel Portal Monitor, May 18, 2012
Personnel Contamination Events/Personnel Contamination Reports (PCE/PCR) Logs, 2/2012 - 3/2014
EDE & NRC Form 5 Calculations for Steam Generator Multibadging Jobs entry made on 3/25/14; Multibadge RCA Authorization/Worksheets
NMP-HP-109 Data Sheets, Investigation of Lost, Damaged or Malfunctioning Personnel Dosimetry, for occurrence on 3/18/2014
NMP-HP-109 Data Sheet 2, Investigation of Lost, Damaged or Malfunctioning Personnel Dosimetry, for occurrence on 6/12/2013

CAP Documents

CR 541097
CR 585435
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CR 746418
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Section 2RS5: Radiation Monitoring Instrumentation

Procedures, Guidance Documents, and Manuals

43802-C, “Calibration of Gamma Standards”, Ver. 12.4
NMP-HP-700, “Radiation Protection Instrumentation Program,” Ver. 1.0
NMP-HP-701, “Daily Instrumentation Source Checks,” Ver. 1.3
NMP-HP-719, “Operation and Calibration of the CANBERRA ARGOS-5AB Exit Monitor”, Ver. 2.0
NMP-HP-718, “Operation and Calibration of the CANBERRA GEM-5 Gamma Exit Monitor”, Ver. 1.0
NMP-HP-709, “Calibration of the Small Article Monitor (SAM-12)”, Ver. 1.0
NMP-HP-708, “Operation and Calibration of the MGPI Telepole Instrument”, Ver. 3.0
43693-C, “Operation and Use of the JL Shepard Model 89-400 Calibrator”, Ver. 2.2

Records and Data
Work Order SNC551063, RMSOOS 1-RE003 Out of Service
Work Order SNC405774, SGBD to MN Cond Rad Mon Ch CAL 1RE0021-18M, 2/28/13
Work Order SNC405890, Plant Vent Post Accident COT 1RE12444C-18M, 8/20/12
System Health Report, Unit 1 1609-Rad Monitoring System, 7/1/2013-9/30/2013
Fleet-HP-2013, Nuclear Oversight Audit of Health Physics, July 15, 2013
43689-C Data Sheet 1, Calibration of the Small Article Monitor, Rev 7, for SAM-11 VEGP #1151, 5/23/2012 & 5/22/2014
NMP-HP-708 Data Sheet1, Telepole Gamma Calibration, SN# VEG-HP-1511 3/04/14
43635-C Data Sheet 2, AMS High Voltage and Flow Calibration, SN# VEGP-1450, 3-13-14 & 3-6-13
NMP-HP-703 Data Sheet 1, Calibration Sheet, RO-20 SN# VEGP-HP-1017 03-4-14 & 03-5-13
43658-C Data Sheet 1, Air Sampler Calibration Sheet, SN# VEGP-HP-1372 11-26-13 & 11-28-13
NMP-HP-719, "Operation and Calibration of the CANBERRA ARGOS-5AB Exit Monitor" Data Sheet 1, ARGOS 5AB Calibration Certificate, 12-3-13

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TE 710894
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CR 785178

Section 4OA1: Performance Indicator (PI) Verification
Procedures, Guidance Documents, and Manuals
00163-C, NRC Performance Indicator and Monthly Operating Report Preparation and Submittal, Ver. 14.6

Records and Data
Liquid Effluent Release Permits L-20131221-239-B and L-20140227-035-B
Gaseous Effluent Release Permits G-20131231-002-B and G-20140222-045-B

CAP Documents
CR 617317
CR 654735
CR 700923
CR 723420

Section 4OA2: Identification and Resolution of Problems
Condition Reports:
CR 648248; Calculation Error Affects EAL Setpoints for AS1 and AG1
CR 648345; Revise Emergency Plan and EPIP to correct EAL RS1 and RG1 error
CR 650353; Perform Apparent Cause Determination on Calculation Error

Documents:
Southern Co. letter NL-13-1979 to NRC, Emergency Plan Revision 60, dated September 24, 2013
Apparent Cause Determination Report, Calculation Errors Resulted in Incorrect EAL Setpoints, July 1, 2013
Documentation of Engineering Judgment DOEJ-VXSNC648248-M001, Corrected Emergency Action Level Set Points for RS1 and RG1 for Plant Vogtle, 5/31/2013

Procedures:
91001-C, Emergency Classification and Implementing Instructions, Rev. 29
NMP-GM-002-001, Corrective Action Program Instructions, Ver. 31.1
NMP-GM-002-007, Apparent Cause Determination Instruction, Ver. 10

Section 4OA5: Other Activities
Condition Reports
CR 687752, 2B EDG Operability Assessment – CAR 195200

NOTICE OF VIOLATION

Southern Nuclear Operating Company, Inc
Vogtle Electric Generating Plant

Docket No. 50-424, 50-425
License No. NPF-68, NFP-81
EA-14-112

During an NRC inspection completed on June 30, 2014, one violation of NRC requirements was identified. In accordance with the NRC Enforcement Policy, the violation is listed below:

> 10 CFR Part 50.54(q)(2), requires that a holder of a nuclear power reactor operating license under this part, shall follow and maintain the effectiveness of emergency plans which meet the requirements in Appendix E of this part and the standards in 10 CFR 50.47(b)
>
> 10 CFR 50.47(b)(4), requires a standard emergency classification and action level scheme, the bases of which include facility and system effluent parameters, is in use by the nuclear facility licensee, and State and local response plans call for reliance on information provided by facility licensees for determinations of minimum initial offsite response measures.
>
> 10 CFR Part 50, Appendix E, Section IV.B., Assessment Actions, requires the means to be used for determining the magnitude of, and for continuously assessing the impact of, the release of radioactive materials shall be described, including emergency action levels that are to be used as criteria for determining the need for notification and participation of local and State agencies, and the Commission. The emergency action levels shall be based on in-plant conditions and instrumentation in addition to onsite and offsite monitoring.
>
> Contrary to the above, from March 20, 2008, until May 31, 2013, the licensee failed to maintain the effectiveness of their emergency plan. Specifically, the licensee failed to maintain a standard emergency classification scheme which included facility effluent parameters in that effluent parameter classification threshold values for RG1 (General Emergency) and RS1 (Site Area Emergency) were significantly non-conservative. These monitors were being relied upon to continuously assess the impact of the release of radioactive materials as well as provide criteria for determining the need for notification and participation of local and State agencies.

This violation is associated with a White SDP finding.

The NRC has concluded that information regarding: 1) the reason for the violation; 2) the actions planned or already taken to correct the violation and prevent recurrence; and, 3) the date when full compliance was achieved, is already adequately addressed on the docket in Inspection Report No. 05000424/2014003 and 05000425/2014003. However, you are required to submit a written statement or explanation pursuant to 10 CFR 2.201 if the description therein does not accurately reflect your corrective actions or your position. In that case, or if you choose to respond, clearly mark your response as a "Reply to a Notice of Violation, EA-14-112," and send it 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, within 30 days of the date of the letter transmitting this Notice.

If you choose to respond, 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. Therefore, to the extent possible, the response should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the Public without redaction.

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

Dated this 6th day of August, 2014.