UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
1600 EAST LAMAR BOULEVARD
ARLINGTON, TEXAS 76011-4511

August 2, 2018

Mr. Eric Larson, Site Vice President
Entergy Operations, Inc.
Grand Gulf Nuclear Station
P.O. Box 756
Port Gibson, MS 39150

SUBJECT: GRAND GULF NUCLEAR STATION – NRC INTEGRATED INSPECTION REPORT 05000416/2018002

Dear Mr. Larson:

On June 30, 2018, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Grand Gulf Nuclear Station. On July 24, 2018, the NRC inspectors discussed the results of this inspection with Mr. B. Franssen, General Manager of Plant Operations, and other members of your staff. The results of this inspection are documented in the enclosed report.

NRC inspectors documented eight findings of very low safety significance (Green) in this report. All of 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 IV; the Director, Office of Enforcement; and the NRC resident inspector at the Grand Gulf Nuclear Station.

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 U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region IV; and the NRC resident inspector at the Grand Gulf Nuclear Station.

This letter, its enclosure, and your response (if any) will be made available for public inspection and copying at http://www.nrc.gov/reading-rm/adams.html and at the NRC Public Document Room in accordance with 10 CFR 2.390, "Public Inspections, Exemptions, Requests for Withholding."

Sincerely,

/RA/

Jason W. Kozal, Chief
Project Branch C
Division of Reactor Projects

Docket No. 50-416
License No. NPF-29

Enclosure:
Inspection Report 05000416/2018002
w/ Attachments:
1. Documents Reviewed
2. Occupational Radiation Safety Inspection
   Request for Information

U.S. NUCLEAR REGULATORY COMMISSION
Inspection Report

Docket Number:	05000416
License Number:	NPF-29
Report Number:	05000416/2018002
Enterprise Identifier:	I-2018-002-0008
Licensee:	Entergy Operations, Inc.
Facility:	Grand Gulf Nuclear Station, Unit 1
Location:	Port Gibson, Mississippi
Inspection Dates:	April 1, 2018 to June 30, 2018
Inspectors:	T. Steadham, Senior Resident Inspector C. Roettgen, Acting Senior Resident Inspector N. Day, Acting Senior Resident Inspector R. Azua, Senior Reactor Inspector A. Athar, Acting Resident Inspector N. Greene, Ph.D., Senior Health Physicist J. O'Donnell, CHP, Health Physicist C. Newport, Senior Resident Inspector, Diablo Canyon NPP M. Bloodgood, Emergency Response Coordinator
Approved By:	Jason W. Kozal Chief, Project Branch C Division of Reactor Projects

Enclosure

SUMMARY

The U.S. Nuclear Regulatory Commission (NRC) continued monitoring the licensee's performance by conducting an integrated inspection at Grand Gulf Nuclear Station Unit 1 in accordance with the Reactor Oversight Process. The Reactor Oversight Process is the NRC's program for overseeing the safe operation of commercial nuclear power reactors. Refer to https://www.nrc.gov/reactors/operating/oversight.html for more information. NRC and self-revealed violations and additional items are summarized in the tables below.

List of Findings and Violations

Failure to Institute Effective Corrective Action to Preclude Repetition.			
Cornerstone	Significance	Cross-cutting Aspect	Report Section
Mitigating Systems	Green NCV 05000416/2018002-01 Closed	[P.2] – Problem Identification and Resolution, Evaluation	71111.01 – Adverse Weather Protection
An NRC-identified, Green non-cited violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," was identified when the licensee failed to institute effective corrective actions to preclude repetition of a significant condition adverse to quality. Specifically, the licensee left a secondary containment personnel hatch in an open configuration for approximately 30 minutes while performing a roof inspection, which rendered secondary containment inoperable. This issue had also previously occurred in 2016, but corrective actions to prevent it from occurring again were ineffective.			

Failure to Follow ASME Requirements for Maintaining Inservice Inspection (ISI) Cycles and Perform ASME Required Inservice Inspections within the Scheduled ISI Cycle			
Cornerstone	Significance	Cross-cutting Aspect	Report Section
Mitigating Systems	Green NCV 05000416/2018002-02 Closed	[H.5] – Human Performance, Work Management	71111.08 – Inservice Inspection Activities
The inspector identified 15 examples of a Green non-cited violation of 10 CFR 50.55(a)(g)(4)(ii), which requires that inservice examination of components classified as American Society of Mechanical Engineers (ASME), Section XI, Code Class 1, Class 2, and Class 3, be conducted during successive 120-month inspection intervals, and requires compliance with the requirements of the latest edition and addenda of the ASME Code (and all its paragraphs) applicable to the specific interval, including maintaining the 120-month inspection interval in accordance with the ASME Code, Section XI, Paragraph IWA-2430. Specifically, the licensee inappropriately adjusted its second inservice inspection 120-month cycle, and failed to perform VT-3 and MT examinations of 15 Class 1, Class 2, and Class 3 components, including the high pressure core spray pump attachment weld and reinforcing band, before the third inservice inspection cycle expired on November 30, 2017, as required by 10 CFR 50.55(a)(g)(4)(ii).			

Failure to Adequately Test NUS Temperature Switch			
Cornerstone	Significance	Cross-cutting Aspect	Report Section
Mitigating Systems	Green NCV 05000416/2018002-03 Closed	None	71111.19 – Post Maintenance Testing
A self-revealed, Green non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," was identified when the reactor core isolation cooling (RCIC) system automatically isolated due to an inadvertent high temperature input from the leakage detection system. Specifically, the licensee failed to fully test a modification that installed a new type of temperature switch, and the system inappropriately isolated the RCIC system when a loss and subsequent restoration of power occurred.			

High Radiation Area Boundary Violation			
Cornerstone	Significance	Cross-cutting Aspect	Report Section
Occupational Radiation Safety	Green NCV 05000416/2018002-04 Closed	[H.8] – Human Performance, Procedure Adherence	71124.01 – Radiological Hazard Assessment
A self-revealed, Green non-cited violation of Technical Specification 5.7.1 was identified when an individual received a dose rate alarm when the individual failed to comply with established radiological barriers and protective measures, and entered a high radiation area. Specifically, an individual leaned over a high radiation area barricade rope, thereby entering the high radiation area. The individual's radiation work permit did not permit entry into a high radiation area.			

Failure to Follow Procedure Requirements Resulting in Unplanned Dose			
Cornerstone	Significance	Cross-cutting Aspect	Report Section
Occupational Radiation Safety	Green NCV 05000416/2018002-05 Closed	[H.12] – Human Performance, Avoid Complacency	71124.02 – Occupational ALARA Planning and Controls
A self-revealed, Green non-cited violation of Technical Specification 5.4.1 was identified when an individual alarmed a personnel contamination monitor upon exit from the radiologically controlled area. Specifically, the licensee failed to follow procedures and establish a decontamination plan or procedure, conduct a specific pre-job brief addressing appropriate contamination risk, and receive approval by radiation protection supervision prior to conducting decontamination activities on the reactor pressure vessel O-rings.			

Improper Evaluation and Resolution of Intermediate Range Monitor Noise Leads to Manual Reactor Shutdown			
Cornerstone	Significance	Cross-cutting Aspect	Report Section
Mitigating Systems	Green NCV 05000416/2018002-06 Closed	[P.2] – Problem Identification and Resolution, Evaluation	71152 – Problem Identification and Resolution
A self-revealed, Green non-cited violation of 10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Action,” was identified for the licensee’s failure to identify and correct a condition adverse to quality. Specifically, the licensee failed to implement appropriate corrective actions related to intermediate range monitor (IRM) nuclear instrument (NI) electronic noise spiking. The failure to implement adequate corrective actions over the course of at least 5 years resulted in a plant shutdown due to a declaration of multiple IRM channels to be inoperable while in Mode 2.			

Loss of Shutdown Cooling			
Cornerstone	Significance	Cross-cutting Aspect	Report Section
Initiating Events	Green NCV 05000416/2018002-07 Closed	[H.12] – Human Performance, Avoid Complacency	71153 – Followup of Events and Notices of Enforcement Discretion
A self-revealed, Green non-cited violation of Technical Specification 5.4, “Procedures,” for the licensee’s failure to follow written procedures was identified when the residual heat removal (RHR) system automatically isolated due to an inadvertent emergency core cooling system (ECCS) actuation. While the plant was shut down with the RHR system in decay heat removal mode, maintenance personnel inadvertently opened an incorrect valve during a transmitter calibration activity, which caused a false low reactor pressure vessel (RPV) water level signal, an ECCS actuation, and a loss of decay heat removal for approximately 31 minutes.			

Performance of Surveillance Testing Following Maintenance on Containment Airlock			
Cornerstone	Significance	Cross-cutting Aspect	Report Section
Barrier Integrity	Green NCV 05000416/2018002-08 Closed	[H.5] – Human Performance, Work Management	71153 – Followup of Events and Notices of Enforcement Discretion
The inspectors identified a Green non-cited violation of 10 CFR Part 50, Appendix B, Criterion XI, “Test Control,” for the licensee’s failure to perform surveillance testing of containment airlock seals under appropriate conditions. The licensee failed to appropriately control the sequence of maintenance and testing activities to ensure that surveillance testing was not performed subsequent to maintenance which could affect the validity of surveillance test results.			

Additional Tracking Items

Type	Issue number	Title	Report Section	Status
LER	05000416/2018001-00	Reactor Manual Shutdown due to Turbine Pressure Control Valve Position Changes	71153	Closed
LER	05000416/2018002-00	Both 208 Containment Air Lock Doors Simultaneously Inoperable	71153	Closed

PLANT STATUS

Unit 1 began this inspection period at rated thermal power. On March 6, 2018, operations personnel began reducing power to prepare for Refueling Outage 21. At 12:01 a.m. on March 7, 2018, the unit was shutdown to commence Refueling Outage 21 and remained shutdown for the remainder of the inspection period.

INSPECTION SCOPES

Inspections were conducted using the appropriate portions of the inspection procedures (IPs) in effect at the beginning of the inspection unless otherwise noted. Currently approved IPs with their attached revision histories are located on the public website at http://www.nrc.gov/reading-rm/doc-collections/insp-manual/inspection-procedure/index.html. Samples were declared complete when the IP requirements most appropriate to the inspection activity were met consistent with Inspection Manual Chapter (IMC) 2515, "Light-Water Reactor Inspection Program - Operations Phase." The inspectors performed plant status activities described in IMC 2515, Appendix D, "Plant Status," and conducted routine reviews using IP 71152, "Problem Identification and Resolution." The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel to assess licensee performance and compliance with Commission rules and regulations, license conditions, site procedures, and standards.

REACTOR SAFETY

71111.01—Adverse Weather Protection

External Flooding (1 Sample)

The inspectors evaluated readiness to cope with external flooding on April 5, 2018.

71111.04—Equipment Alignment

Partial Walkdown (1 Sample)

The inspectors evaluated system configurations during a partial walkdown of the following system:

(1) Alternate decay heat removal system with residual heat removal B system out of service for maintenance on April 24, 2018

Complete Walkdown (1 Sample)

The inspectors evaluated system configurations during a complete walkdown of the Division 1 standby diesel generator with the Division 2 standby diesel generator out of service for maintenance on April 24, 2018.

71111.05Q—Fire Protection Quarterly

Quarterly Inspection (4 Samples)

The inspectors evaluated fire protection program implementation in the following selected areas:

(1) Division 1 emergency diesel generator room, 133 foot elevation, on May 21, 2018
(2) 119 foot auxiliary building, containment access point on June 1, 2018
(3) 166 foot turbine building, 1Y97 relay house, on June 12, 2018
(4) 185 foot control building, Unit 2 upper cable spreading room on June 22, 2018

71111.08—Inservice Inspection Activities (1 Sample)

The inspectors evaluated boiling water reactor nondestructive testing by reviewing the following examinations from April 23, 2018, to April 27, 2018:

The inspectors directly observed the following nondestructive examinations:

(1) Visual 3

  a) Scram Discharge, Pipe Support Q1C11ASP9

The inspector reviewed records for the following nondestructive examinations:

(1) Visual 2

  a) Residual Heat Removal (RHR) B Discharge Line, 105 feet RHR Penetration 73

(2) Visual 3

  a) Low Pressure Core Spray (LPCS) Pump Support Q1E21C001-S2
  b) LPCS, Support w/Thermal Movement Q1E21G002H01
  c) Reactor Core Isolation Cooling, Multi-Directional Restraint Q1E51G004C04
  d) Standby Liquid Control, Support w/Thermal Movement Q1C41G136C02
  e) Combustible Gas Control, Multi-Directional Restraint Q1E61G001R07
  f) Nuclear Boiler, Support w/Thermal Movement Q1B21G026H02
  g) Nuclear Boiler, Multi-Directional Restraint Q1B21G030C01
  h) Standby Service Water, Multi-Directional Restraint Q1P41G010C02
  i) Reactor Water Cleanup, Multi-Directional Restraint Q1G33G002H16
  j) Floor and Equipment Drain, Multi-Directional Restraint Q1P45G813C03
  k) Reactor Water Cleanup, Multi-Directional Restraint Q1G33G001H01
  l) Reactor Water Cleanup, Support w/Thermal Movement Q1G33G012H01
  m) Reactor Water Cleanup, Support w/Thermal Movement Q1E12G050H02

(3) Ultrasonic

  a) LPCS Pump Bolting E21C001
  b) High Pressure Core Spray, Pipe to Elbow Circ. Weld 1E22G003W25
  c) Reactor Core Injection Cooling, Pipe and Valve Circ. Weld 1E51G004W9

(4) Radiographic

a) RHR Discharge Header Safety Relief Valve, Blind Fitting to Pipe Butt Weld, Weld W901

b) RHR Discharge Header Safety Relief Valve, Elbow to Blind Fitting Butt Weld, Weld W902

71111.11—Licensed Operator Requalification Program and Licensed Operator Performance

Operator Requalification (1 Sample)

The inspectors observed and evaluated the simulated loss of condenser vacuum with an anticipated transient without a scram on June 19, 2018.

Operator Performance (1 Sample)

The inspectors observed and evaluated plant shutdown for Refueling Outage 21 on April 7, 2018.

71111.13—Maintenance Risk Assessments and Emergent Work Control (3 Samples)

The inspectors evaluated the risk assessments for the following planned and emergent work activities:

(1) Operational readiness assessment team and Shutdown Operations Protection Plan on April 6, 2018

(2) Updated risk assessment during loss of plant service water and alternate decay heat removal unavailable on April 29, 2018

(3) Updated risk assessment following fire in invertor 1Y97 on June 13, 2018

71111.15—Operability Determinations and Functionality Assessments (5 Samples)

The inspectors evaluated the following operability determinations and functionality assessments:

(1) Surface cracks identified on primary containment, CR-GGN-2018-03767, on April 17, 2018

(2) Standby gas, CR-GGN-2018-03118, on April 19, 2018

(3) High pressure core spray Level 8 on May 10, 2018

(4) High particulate counts in the high pressure core spray diesel starting air system, CR-GGN-2018-05814, on May 19, 2018

(5) Containment airlock door seal testing application on May 10, 2018

71111.19—Post Maintenance Testing (6 Samples)

The inspectors evaluated the following post maintenance tests:

(1) 119 foot elevation airlock door testing following replacement of the 14A clevis valve on August 10, 2016

(2) Division 2 emergency diesel generator fuel oil storage tank cleaning and inspection on April 23, 2018

(3) Source range monitor B troubleshooting and repair on May 17, 2018

(4) Containment spray isolation valve E12F028B on June 27, 2018

(5) Engineered safety feature transformer 11 on June 29, 2018

(6) Nuclear utilities service (NUS) temperature switch replacement on June 29, 2018.

71111.20—Refueling and Other Outage Activities (1 Sample)

The inspectors evaluated Refueling Outage 21 activities from April 7, 2018, to June 30, 2018.

71111.22—Surveillance Testing

The inspectors evaluated the following surveillance tests:

Routine (4 Samples)

(1) Secondary containment test using A319A and standby gas treatment system A on April 4, 2018

(2) Standby liquid control injection test on April 23, 2018

(3) Division 1, Level 1 pressure transmitter calibration on May 1, 2018

(4) Division 1 loss of offsite power/loss of coolant accident testing on May 28, 2018

In-service (1 Sample)

(1) Residual heat removal C retest due to vibrations on April 2, 2018

Containment Isolation Valve (1 Sample)

(1) Reactor core isolation cooling steam supply isolation valve E51F063 local leak rate test on April 18, 2018

RADIATION SAFETY

71124.01—Radiological Hazard Assessment and Exposure Controls

Radiological Hazard Assessment (1 Sample)

The inspectors evaluated radiological hazards assessments and controls.

Instructions to Workers (1 Sample)

The inspectors evaluated worker instructions.

Contamination and Radioactive Material Control (1 Sample)

The inspectors evaluated contamination and radioactive material controls.

Radiological Hazards Control and Work Coverage (1 Sample)

The inspectors evaluated radiological hazards control and work coverage.

High Radiation Area and Very High Radiation Area Controls (1 Sample)

The inspectors evaluated risk-significant high radiation area and very high radiation area controls.

Radiation Worker Performance and Radiation Protection Technician Proficiency (1 Sample)

The inspectors evaluated radiation worker performance and radiation protection technician proficiency.

71124.02—Occupational As Low As Reasonably Achievable (ALARA) Planning and Controls

Implementation of ALARA and Radiological Work Controls (1 Sample)

The inspectors reviewed ALARA practices and radiological work controls by reviewing the following activities:

(1) RWP 20181402, “Refuel Floor High Water Activities”
(2) RWP 20181800, “Turbine Building Work”
(3) RWP 20181952, “Radiography and ARM Calibration in Drywell and Containment”

Radiation Worker Performance (1 Sample)

The inspectors evaluated radiation worker and radiation protection technician performance.

OTHER ACTIVITIES – BASELINE

71151—Performance Indicator Verification

The inspectors verified licensee performance indicators submittals listed below:

(1) MS05: Safety System Functional Failures (SSFFs) (April 1, 2017 – March 31, 2018) (1 Sample)

(2) OR01: Occupational Exposure Control Effectiveness (January 1, 2017 – March 31, 2018) (1 Sample)

(3) PR01: Radiological Effluent Technical Specifications/Offsite Dose Calculation Manual Radiological Effluent Occurrences (RETS/ODCM) Radiological Effluent Occurrences (January 1, 2017 – March 31, 2018) (1 Sample)

71152—Problem Identification and Resolution

Semiannual Trend Review (1 Sample)

The inspectors reviewed the licensee's corrective action program for trends that might be indicative of a more significant safety issue.

Annual Follow-up of Selected Issues (2 Samples)

The inspectors reviewed the licensee's implementation of its corrective action program related to the following issues:

(1) CR-GGN-2017-11733, noise related intermediate-range monitor (IRM) nuclear instrument (NI) spiking leads to the inoperability of multiple IRM channels and a plant shutdown

(2) CR-GGN-2018-3204, ball bearings and cable bull nut in spent fuel pool

71153—Follow-up of Events and Notices of Enforcement Discretion

Events (3 Samples)

The inspectors evaluated:

(1) Plant and operator response due to Division 1 emergency core cooling system actuation on May 1, 2018

(2) Plant and operator response due to 15AA bus de-energization on May 12, 2018

(3) Operator response to a fire in inverter 1Y97 on June 12, 2018

Licensee Event Reports (2 Samples)

The inspectors evaluated the following licensee event reports which can be accessed at https://lersearch.inl.gov/LERSearchCriteria.aspx:

(1) Licensee Event Report (LER) 05000416/2018001-00, Reactor Manual Shutdown due to Turbine Pressure Control Valve Position Changes (ADAMS Accession No. ML18088B198), on April 25, 2018

(2) LER 05000416/2018002-00, Both 208 Containment Air Lock Doors Simultaneously Inoperable (ADAMS Accession No. ML18094A172), on May 10, 2018

INSPECTION RESULTS

Observation	71152 – Problem Identification and Resolution

The inspectors reviewed the licensee's corrective action program for potential adverse trends in the categorization of issues in the corrective action program and the timeliness of operability determinations that might be indicative of a more significant safety issue. In light of the most recent problem identification and resolution team inspection completed at Grand Gulf Nuclear Station (reference NRC Inspection Report 05000416/2017011, dated February 12, 2018, ADAMS Accession No. ML18043B137), which found numerous examples of the licensee's failure to categorize and evaluate conditions in accordance with the requirements of Entergy Procedure EN-LI-102, "Corrective Action Program," Revision 33, as well as observations during routine condition report reviews, the inspectors evaluated a sample of condition reports generated over the course of the past year to determine if observations during routine condition report reviews were singular events or trends.

The inspectors reviewed eight condition reports written on safety-related components that the licensee had closed from the corrective action program to departmental trending programs. Of the eight condition reports reviewed, the inspectors identified two conditions that should have been classified as adverse, and therefore should not have been closed to a trending process. Both condition reports were reopened, and Condition Report CR-GGN-2018-04585 was generated documenting the incorrect classification of the two conditions.

Additionally, the inspectors reviewed 10 condition reports written in 2018 that required immediate operability determinations and in which the shift operators requested that the engineering department assist in a prompt operability determination. Entergy Procedure EN-OP-104, "Operability Determination Process," Revision 15, provides guidance that prompt determinations should be completed in a time commensurate with the safety significance of the system, structure, or component, subject to the evaluation and that prompt determinations can often be done within 24 hours. In cases that require more time to gather information, the procedure allows the licensee to evaluate the risk importance of the additional information to decide whether to prolong the prompt determination. One measure of safety significance that is recommended to be taken into account is the technical specification allowed outage time for the equipment if it were inoperable. Of the 10 condition reports reviewed, the inspectors noted that none had the initial due dates of the engineering operability evaluations set within 24 hours of the immediate operability determinations.

Additionally, many of the prompt determination due dates were extended beyond the original due dates and beyond the technical specification allowed outage times for the subject equipment. The inspectors determined that this did not violate any regulations or procedural requirements, but was contrary to the procedural guidance provided, and did raise the risk of inadvertently violating technical specifications if engineering determined an immediate operability determination was incorrect and the equipment was inoperable. None of the conditions reviewed resulted in a determination that the subject equipment was inoperable.

Failure to Institute Effective Corrective Action to Preclude Repetition.

Cornerstone	Significance	Cross-cutting Aspect	Report Section
Mitigating Systems	Green NCV 05000416/2018002-01 Closed	[P.2] – Problem Identification and Resolution, Evaluation	71111.01 – Adverse Weather Protection

An NRC-identified, Green non-cited violation of 10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Action,” was identified when the licensee failed to institute effective corrective actions to preclude repetition of a significant condition adverse to quality. Specifically, the licensee left a secondary containment personnel hatch in an open configuration for approximately 30 minutes while performing a roof inspection, which rendered secondary containment inoperable. This issue had also previously occurred in 2016, but corrective actions to prevent it from occurring again were ineffective.

<u>Description</u>: On April 5, 2018, workers performed a periodic inspection of the enclosure building roof. In order to access the roof, a hatch credited for secondary containment integrity must be opened. Technical Specification 3.6.4.1 and Basis allows this hatch to be opened only for normal ingress and egress purposes and did not include any provisions to allow the hatch to remain open under administrative controls.

During this roof inspection, workers opened the hatch to access the roof but left it open under administrative controls (e.g. being able to close the hatch based on communications/orders from the main control room) until the last worker exited the roof at the conclusion of the inspection approximately thirty minutes later. The workers believed they could keep the hatch open under administrative controls because of a communication error.

A similar issue occurred on April 7, 2016, which is documented in Licensee Event Report 05000416/2016-003-00 (ADAMS Accession No. ML16158A106). The licensee determined after a root cause evaluation was performed that the model work order instructions lacked administrative controls, and developed corrective actions to prevent repetition (CAPRs), which were in place for the April 5, 2018, planned enclosure building roof inspection. Additionally, the licensee performed two causal evaluations for the April 7, 2016, issue as documented in Condition Reports CR-GGN-2017-10866 (root cause) and CR-GGN-2016-03707 (apparent cause).

Upon reviewing the configuration of the hatch after the roof inspection concluded, the inspectors noted that the hatch remained in a configuration prohibited by Technical Specification 3.6.4.1 because it was open for a significantly longer period than was required for normal ingress and egress. Immediately following the roof inspection, the NRC inspectors brought the issue to the attention to the main control room.

Corrective Action(s): Licensee corrective actions included, in part, securing the hatch before leaving the area, stationing operations personnel with constant communications with the control room to ensure the hatch is in the appropriate configuration during roof inspection activities, briefing the issue to all licensed operators and revising model work orders to ensure specific actions to control the operational impact of personnel gaining access to the enclosure building roof. Corrective Action Reference(s): The licensee entered the open roof hatch issue into the corrective action program as Condition Report CR-GGN-2018-03185.
Performance Assessment: Performance Deficiency: The licensee's failure to appropriately control an activity that affected the operability of secondary containment was a performance deficiency. Specifically, a secondary containment personnel hatch was placed in an open configuration for approximately 30 minutes while performing a roof inspection. Screening: The inspectors determined the performance deficiency was more than minor because it was associated with the configuration control attribute of the Barrier Integrity Cornerstone and adversely affected the objective to provide reasonable assurance that physical design barriers (fuel cladding, reactor coolant system, and containment) protect the public from radionuclide releases caused by accidents or events. Specifically, the licensee's failure to maintain configuration control of secondary containment while performing a planned roof inspection resulted in undue inoperability of secondary containment. Significance: The inspectors screened the significance of the finding using Inspection Manual Chapter 0609, Appendix A, Exhibit 2, Section A. Since the finding only represented a degradation of the radiological barrier function provided for the auxiliary building, the finding was determined to be of very low safety significance (Green). Cross-cutting Aspect: This cause of this finding is related to the cross-cutting area of problem identification and resolution – evaluation because Entergy personnel did not thoroughly evaluate issues to ensure that resolutions address causes and extent of conditions commensurate with their safety significance. Specifically, the corrective action program failed to fully evaluate the issue in two previous cause evaluations, and therefore failed to develop specific, measurable, achievable, realistic, and timely corrective actions.
Enforcement: Violation: As required, in part, by 10 CFR Part 50, Appendix B, Criterion XVI, in the case of significant conditions adverse to quality, the measures shall assure that the cause of the condition is determined and corrective action taken to preclude repetition. Contrary to the above, prior to April 5, 2018, the licensee failed to establish measures to assure that corrective action was taken to preclude repetition of a significant condition adverse to quality. Because of the ineffective corrective actions to prevent repetition of the condition, secondary containment was rendered inoperable when a hatch was left in the open configuration while performing a roof inspection. Disposition: This violation is being treated as a non-cited violation (NCV), consistent with Section 2.3.2 of the Enforcement Policy.

<table>
<tr><td colspan="4">Failure to Follow ASME Requirements for Maintaining Inservice Inspection (ISI) Cycles and Perform ASME Required Inservice Inspections within the Scheduled ISI Cycle</td></tr>
<tr><td>Cornerstone</td><td>Significance/Severity</td><td>Cross-cutting Aspect</td><td>Report Section</td></tr>
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NCV 05000416/2018002-02
Closed</td><td>[H.5] – Human Performance, Work Management</td><td>71111.08 – Inservice Inspection Activities</td></tr>
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<tr><td colspan="4"><u>Description</u>: On July 1, 1985, the licensee began operation of the Grand Gulf Nuclear Station (GGNS) and commenced the first 120-month inservice inspection cycle in accordance with the requirements of 10 CFR 50.55(a)(g)(4)(ii). In a letter dated April 14, 1994, Entergy Operations, Inc. requested authorization to extend the 120-month intervals for Arkansas Nuclear One, Unit 1; Grand Gulf Nuclear Station, Unit 1; and Waterford Steam Electric Station, Unit 3, to avoid an update of their inservice inspection and inservice testing (ISI/IST) programs until after the NRC staff had completed changes to 10 CFR 50.55(a)(g)(4)(ii) through the rulemaking process. On August 1, 1994, the NRC granted the requested extensions. The end date of the first 120-month ISI cycle for GGNS was extended from July 1, 1995, to January 1, 1997. This added 18 months to the 120-month ISI/IST cycle. The NRC stipulated in the authorization letter that the licensee must shorten a subsequent 120-month interval to bring the ISI cycle back into compliance with the requirements of ASME Code Section XI, Paragraph IWA-2430(c).

ASME Code Section XI, Paragraph IWA-2430(c)(1), “Inspection Intervals,” states that each (120-month ISI) inspection interval may be extended by as much as 1 year (12 months). Adjustments shall not cause successive intervals to be altered by more than 1 year from the original pattern of intervals.

Subsequent to this, the licensee extended its second interval from January 1, 2007, to May 1, 2008, without authorization from the NRC, contrary to the requirements of ASME XI, Paragraph IWA-2430(c)(1). In total, the licensee had extended the interval end dates by 28 months from the original pattern of intervals.

In February 2018, the licensee identified 15 items that had not been inspected during the third ISI cycle, which ended on November 30, 2017. The licensee inappropriately extended the third cycle until May 3, 2018; however, since the intervals were already outside the allowable limits of the Code, the licensee needed to request authorization from the NRC in advance.</td></tr>
</table>

The failure to complete all the prescribed inservice inspections within the third ISI cycle in accordance with 10 CFR 50.55(a)(g)(4)(ii) was a performance deficiency. Corrective Actions: The licensee completed 14 of the 15 ISI examinations by May 31, 2018, and all 14 met their associated acceptance criteria. At the conclusion of this inspection, the licensee was in the process of scheduling the remaining examination to be completed prior to completion of the plant outage. The licensee is investigating the cause of the failure to maintain the station's ISI cycle within ASME Code, Section XI, Paragraph IWA-2430, requirements. Both issues were documented in the licensee's corrective action program. Corrective Action Reference: Condition Report CR-GGN-2018-04661
<u>Performance Assessment</u>: Performance Deficiency: The licensee's failure to maintain its 120-month ISI cycles, and complete inservice inspections within those inspection cycles, in accordance with 10 CFR 50.55(a)(g)(4)(ii) requirements, was a performance deficiency. Screening: The inspector determined the performance deficiency was more than minor because it was associated with the equipment reliability attribute of the Mitigating Systems Cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the failure to maintain the ISI cycles and perform the examinations within those inspection cycles may allow system equipment/component flaws to go undetected and result in mitigating system equipment failures, placing into question the reliability and availability of those components necessary to effect plant safe shutdown. Significance: The inspector assessed the significance of the finding using Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012. The inspector determined the finding screened as having very low significance (Green) because: it was not a design deficiency; did not represent a loss of system and/or function; did not represent an actual loss of function of at least a single train for longer than its technical specification allowed outage time; and did not result in the loss of a high safety-significant nontechnical specification train. Cross-cutting Aspect: The cause of this finding is related to the cross-cutting area of human performance - work management, due to the licensee's failure to implement a process of planning, controlling, and executing work activities such that nuclear safety is the overriding priority.

Enforcement:

Violation: As required by 10 CFR 50.55(a)(g)(4)(ii), components classified as ASME Code, Class 1, Class 2, and Class 3 must meet the requirements set forth in Section XI of the applicable editions of the ASME Boiler and Pressure Vessel Code, and Addenda. As required by 10 CFR 50.55(a)(g)(4)(ii), inservice examination of components must be conducted during successive 120-month inspection intervals and requires compliance with the requirements of the latest edition and addenda of the Code applicable to the specific interval. ASME Code, Section XI (of prior and current applicable editions of the Code), Paragraph IWA-2430(c)(1), stipulates that each (120-month) inspection interval may be extended by as much as 1 year, but adjustments shall not cause successive intervals to be altered by more than 1 year from the original pattern of interval.

Contrary to the above, on November 30, 2017, the licensee made inspection interval adjustments that caused successive intervals to be altered by more than 1 year from the original pattern of interval. Specifically, the licensee inappropriately extended the Grand Gulf Nuclear Station's third ISI cycle from November 30, 2017, to May 30, 2018, without obtaining approval from the NRC, which resulted in a failure to perform 15 inservice inspections during the third ISI cycle.

Disposition: This violation is being treated as a non-cited violation, consistent with Section 2.3.2 of the Enforcement Policy.

Failure to Adequately Test NUS Temperature Switch			
Cornerstone	Significance	Cross-cutting Aspect	Report Section
Mitigating Systems	Green NCV 05000416/2018002-03 Closed	None	71111.19 – Post Maintenance Testing

A self-revealed, Green non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," was identified when the reactor core isolation cooling (RCIC) system automatically isolated due to an inadvertent high temperature input from the leakage detection system. Specifically, the licensee failed to fully test a modification that installed a new type of temperature switches, and the system inappropriately isolated the RCIC system when a loss and subsequent restoration of power occurred.

Description: On December 12, 2017, while the station was operating at power, the Division 1 electrical safety bus was aligned to the engineered safety feature (ESF) transformer 11. Due to an underground cable failure and fault, the ESF 11 transformer experienced an electrical fault and lockout, and the Division 1 electrical safety bus lost voltage. The Division 1 emergency diesel generator was automatically started as designed due to the load shedding and sequencer sensing the power loss, and restored power to the Division 1 electrical safety bus. Upon power restoration to the Division 1 electrical safety bus, the RCIC system automatically isolated, and was declared inoperable due to the inadvertent isolation.

After troubleshooting, the licensee determined that, during the restoration of power, the NUS type temperature switches associated with the leak detection systems, specifically the 1E31N608A and 1E31N610A switches, provided a high temperature signal which completed the isolation logic to isolate the RCIC system. The 1E31N608A and 1E31N610A switches provide temperature input for the residual heat removal Subsystem A equipment area. During the restoration of power to the Division 1 electrical safety bus, the NUS switches reenergized and performed a self-check function that provided a high temperature signal for 1.58 seconds, which was longer than the leak detection system's time counting logic (1 second) to screen out spurious alarms.

In 2009, due to obsolete parts, Grand Gulf Nuclear Station evaluated the NUS style temperature switches for replacement of the Riley style temperature switches in the leak detection system. However, Grand Gulf Nuclear Station failed to evaluate or test the switches during or following this modification to determine whether the operational characteristics of the replacement switches were fully compatible to support the system's functionality, specifically during a loss of power and restoration of power scenario for the associated electrical safety bus. Therefore, Grand Gulf Nuclear Station failed to adequately evaluate the design modification that included installation of replacement switches into their leakage detection system under Work Orders 181384 and 181385 on June 30, 2009.

Corrective Actions: Licensee corrective actions included modification of the leakage detection system counting logic from 1 second to 10 seconds, to screen out spurious alarms. Grand Gulf Nuclear Station also evaluated all other NUS switches to determine whether any other plant systems may be adversely impacted during a loss of power or restoration of power scenario. The licensee also held an engineering department stand down to discuss the importance of post modification testing to determine if a system is appropriate to the design prior to system restoration.

Corrective Action References: The licensee entered this issue into the corrective action program as Condition Reports CR-GGN-2017-12314 and CR-GGN-2018-05105.

Performance Assessment:

Performance Deficiency: The failure to appropriately evaluate or test the impact of replacement NUS switches prior to placing them in service was a performance deficiency.

Screening: The inspectors determined the performance deficiency was more than minor because it was associated with the design control attribute of the Mitigating Systems Cornerstone and adversely affected the objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, failing to effectively evaluate or test the leak detection system NUS temperature switches caused an inappropriate RCIC isolation on December 12, 2017.

Significance: The inspectors screened the significance of the finding using Inspection Manual Chapter 0609, Appendix A, Exhibit 2, Section A. Because the finding represented a loss of the RCIC system, the finding required a detailed risk evaluation.

The senior reactor analyst evaluated the risk of the subject performance deficiency in accordance with Appendix A, Section 6.0, "Detailed Risk Evaluation." The analyst noted that the performance deficiency only affected the RCIC system function when the Division I vital bus is deenergized and then reenergized. Therefore, the dominant risk contributors were from a loss of offsite power. Additionally, the analyst determined that long-term station blackout scenarios would not be directly impacted because RCIC would start and run without isolation.

Using the site-specific SPAR model, Version 8.2, the analyst quantified the frequency of all loss of offsite power scenarios that either: (1) did not result in a station blackout; or (2) resulted in a station blackout that was recovered in 1 hour or less. The core damage frequency for these scenarios was 1.0 x 10^{-6}/yr. Because the isolation signal provided by the performance deficiency was very short lived, recovery of RCIC was possible and provided for in the emergency operating procedures. The analyst used a screening value of 0.1 to determine the impact of this recovery. The resulting core damage frequency was 1.0 x 10^{-7}/yr. The dominant core damage scenarios included: 1. Loss of offsite power Failure of all three emergency diesel generators Failure to recover an emergency diesel in 30 minutes Failure to recover offsite power in 30 minutes Failure to recover RCIC isolation 2. Loss of offsite power Failure of operators to control SRVs or depressurize Failure of the Division III Diesel Generator Failure to recover an emergency diesel in 1 hour Failure to recover offsite power in 1 hour Failure to recover RCIC isolation The analyst noted that some of the cutsets in the evaluated scenarios included hard failures of the RCIC system. These failures would not have been affected by an inadvertent isolation of the steam to the RCIC pump because the system was already failed. Therefore, qualitatively, the core damage frequency of the scenarios of concern would be less than 1 x 10^{-7}/yr. Given that the incremental conditional core damage frequency can be no higher than the core damage frequency of the associated sequences (less than 1 x 10^{-7}/yr), this finding is of very low safety significance (Green). Cross-cutting Aspect: Since the cause of the performance deficiency occurred in 2009, it was not determined to be indicative of current licensee performance; therefore no cross-cutting aspect was assigned.
<u>Enforcement</u>: Violation: As required, in part, by 10 CFR Part 50, Appendix B, Criterion III, design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program. Contrary to the above, on June 30, 2009, the licensee's design control measures failed to provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program. Specifically, the licensee failed to adequately evaluate or test the NUS temperature switches installed by Engineering Change 13834, under Work Orders 181384 and 181385. Disposition: This violation is being treated as a non-cited violation (NCV), consistent with Section 2.3.2 of the Enforcement Policy.

High Radiation Area Boundary Violation			
Cornerstone	Significance	Cross-cutting Aspect	Report Section
Occupational Radiation Safety	Green NCV 05000416/2018002-04 Closed	[H.8] – Human Performance, Procedure Adherence	71124.01 – Radiological Hazard Assessment

A self-revealed, Green non-cited violation of Technical Specification 5.7.1 was identified when an individual received a dose rate alarm when the individual failed to comply with established radiological barriers and protective measures and entered a high radiation area. Specifically, an individual leaned over a high radiation area barricade rope, thereby entering the high radiation area. The individual's radiation work permit (RWP) did not permit entry into a high radiation area.

Description: On March 23, 2018, an individual was authorized by a RWP to access a radiation area and had self-briefed on the radiological conditions. The individual went to the 166 foot elevation of the turbine building to take pictures of pump parts located inside a posted high radiation area. While taking pictures, the individual leaned over the high radiation area boundary rope, thus entering the high radiation area, and received a dose rate alarm on his electronic dosimeter. The RWP task for this individual did not permit access to the high radiation area. Upon receiving the dose rate alarm, the individual exited the turbine building and reported the dose rate alarm to radiation protection personnel.

Corrective Action: The immediate actions by the licensee included surveying the area, excluding the individual from the radiologically controlled area, and conducting an evaluation of the electronic dosimeter. The licensee followed up with a human performance evaluation and an operational experience communication to site personnel.

Corrective Action Reference: CR-GGN-2018-02718

Performance Assessment:

Performance Deficiency: Licensee personnel did not comply with established radiological barriers and protective measures and improperly entered a high radiation area.

Screening: The inspectors determined that the performance deficiency was more than minor because it was associated with the program and process attribute of the Occupational Radiation Safety Cornerstone and adversely affected the cornerstone objective to ensure adequate protection of worker health and safety from exposure to radiation from radioactive material during routine civilian nuclear reactor operation. Specifically, the failure to follow requirements involving radiological controls had the potential to increase the individual's dose.

Significance: The inspectors assessed the significance of the finding using NRC Inspection Manual Chapter 0609, Appendix C, "Occupational Radiation Safety Significant Determination Process." The inspectors determined the finding to be of very low safety significance (Green) because: (1) it was not associated with as low as is reasonably achievable (ALARA) planning or work controls; (2) there was no overexposure; (3) there was no substantial potential for an overexposure; and (4) the ability to assess dose was not compromised.

Cross-Cutting Aspect: H.8 – Procedure Adherence: Individuals follow processes, procedures, and work instructions. Specifically, the individual failed to comply with established procedures for controlling access to high radiation areas.

Enforcement:

Violation: Technical Specification 5.7.1 requires, in part, that each high radiation area, in which the intensity of radiation is > 100 mrem/hr but < 1000 mrem/hr, shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a radiation work permit.

Contrary to the above, on March 23, 2018, entrance to a high radiation area was not controlled by requiring issuance of a radiation work permit. Specifically, an individual leaned over a high radiation area boundary, thereby entering the high radiation area, on a radiation work permit that only authorized radiation area (< 100 mrem/hr) access.

Disposition: This violation is being treated as a non-cited violation (NCV), consistent with Section 2.3.2 of the Enforcement Policy.

Failure to Follow Procedure Requirements Resulting in Unplanned Dose			
Cornerstone	Significance	Cross-cutting Aspect	Report Section
Occupational Radiation Safety	Green NCV 05000416/2018002-05 Closed	[H.12] – Human Performance, Avoid Complacency	71124.02 – Occupational ALARA Planning and Controls

A self-revealed, Green non-cited violation of Technical Specification 5.4.1 was identified when an individual alarmed a personnel contamination monitor upon exit from the radiologically controlled area. Specifically, the licensee failed to follow procedures to establish a decontamination plan or procedure, conduct a specific pre-job brief addressing appropriate contamination risk, and receive approval by radiation protection supervision prior to conducting decontamination activities on the reactor pressure vessel (RPV) O-rings.

Description: On April 20, 2018, a decontamination worker was asked by a senior radiation protection (RP) technician to conduct decontamination activities on the RPV O-rings. The decontamination worker was signed onto radiation work permit (RWP) 20181400, "RP/Decon Support for Refuel Floor," Task 4, for other activities. An evaluation was performed that determined only a face shield was needed for the work the decontamination worker was originally tasked to perform. The maximum level of contamination was documented as 150,000 dpm/100 cm^2 (Survey GGN-1804-1555). Prior to entering the work area, the senior RP technician performed a high radiation area briefing with only "spot checking" of area dose rates for the tasks to be performed; the worker had not been specifically briefed for the additional work he was asked to conduct.

Licensee Procedure EN-RP-401, "Decontamination Program," Revision 6, requires that precautions be taken to prevent personnel contaminations and that a decontamination plan or procedure is developed.

Procedure EN-RP-401, step 5.2[11], states, "Take precautions necessary to prevent personnel contaminations." The licensee's review of the event determined that necessary precautions would have included a specific pre-job brief with RP supervision approval; this brief was not conducted. This specific brief was required to appropriately address the contamination risk associated (i.e., on the head flange) with the decontamination job.

Procedure EN-RP-401, step 5.6[5], states, “Prior to major decontamination projects, i.e., refuel canal/refuel cavity decontamination, spent fuel pool transfer canal, moisture separator pit, sumps and any large components or per RP supervision, a decontamination plan or procedure will be developed to capture decontamination activities.” Subsequent to the event the licensee determined that no decontamination plan or procedure was developed because RP supervision was not aware of the requested activity or involved in its assignment. Further, the O-rings were not normally decontaminated prior to removal (from the RPV).

The inspectors concluded that the lack of appropriate contamination controls and decontamination plan for the RPV O-rings resulted in a personnel contamination event, an intake of radioactive material, and an assigned internal dose of 13.5 millirem committed effective dose equivalent to the individual. Further, because the methods used were not adequate or effective for decontaminating the RPV O-rings, three unsuccessful attempts were made, resulting in accrual of additional external dose by the decontamination worker. The inspectors concluded that coordinating with RP supervision to develop a plan or procedure commensurate with the contamination risk, as required by procedure, would have ensured appropriate contamination controls and decontamination methods were being used.

Corrective Action: The immediate actions by the licensee included decontaminating the individual, conducting a whole body count, restricting the individual’s access to the radiologically controlled area, and performing an internal dose assessment. The senior RP technician and the individual were coached on procedural requirements and radiation worker practices.

Corrective Action Reference: CR-GGN-2018-04288 and CR-GGN-2018-04298

Performance Assessment:

Performance Deficiency: Licensee personnel failed to follow procedural requirements associated with contamination controls.

Screening: The inspectors determined that the performance deficiency was more than minor because it was associated with the program and process attribute of the Occupational Radiation Safety Cornerstone and adversely affected the cornerstone objective to ensure adequate protection of worker health and safety from exposure to radiation from radioactive material during routine civilian nuclear reactor operation. Specifically, a failure to follow procedural requirements associated with contamination controls can result in avoidable intakes of radioactive material or skin contamination.

Significance: The inspectors assessed the significance of the finding using NRC Inspection Manual Chapter 0609, Appendix C, “Occupational Radiation Safety Significance Determination Process.” The inspectors determined the finding to be of very low safety significance (Green) because: (1) it was not associated with as low as is reasonably achievable (ALARA) planning or work controls; (2) there was no overexposure; (3) there was no substantial potential for an overexposure; and (4) the ability to assess dose was not compromised.

Cross-Cutting Aspect: H.12 – Avoid Complacency: Individuals recognize and plan for the possibility of mistakes, latent issues, and inherent risk, even while expecting successful

outcomes. This includes implementing error reduction tools. Specifically, the licensee failed to develop a decontamination plan and conduct a specific pre-job brief that included contamination controls appropriate for the contamination risk, instead assuming the task could be successfully planned and performed while the activity was in progress.
Enforcement: Violation: Technical Specification 5.4.1 requires, in part, that written procedures shall be established, implemented, and maintained covering the procedures recommended in Regulatory Guide 1.33, Appendix A, Revision 2, dated February 1978. Section 7(e) of Appendix A requires radiation protection procedures. Licensee Procedure EN-RP-401, "Decontamination Program," requires that precautions be taken to prevent personnel contaminations and that a decontamination plan or procedure be developed for major decontamination projects. Contrary to the above, on April 20, 2018, the licensee failed to take precautions to prevent personnel contaminations and failed to develop a decontamination plan or procedure for a major decontamination project, as specified by Procedure EN-RP-401. Specifically, licensee personnel attempted decontamination of the RPV O-ring without meeting the requirements of Procedure EN-RP-401, resulting in a personnel contamination event and unplanned internal and external dose to the individual. Disposition: This violation is being treated as a non-cited violation (NCV), consistent with Section 2.3.2 of the Enforcement Policy.

Improper Evaluation and Resolution of Intermediate Range Monitor Noise Leads to Manual Reactor Shutdown			
Cornerstone	Significance	Cross-cutting Aspect	Report Section
Mitigating Systems	Green NCV 05000416/2018002-06 Closed	[P.2] – Problem Identification and Resolution, Evaluation	71152 – Problem Identification and Resolution
A self-revealed, Green non-cited violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," was identified for the failure of the licensee to identify and correct a condition adverse to quality. Specifically, the licensee failed to implement appropriate corrective actions related to intermediate range monitor (IRM) nuclear instrument (NI) electronic noise spiking. The failure to implement adequate corrective actions over the course of at least 5 years resulted in a plant shutdown due to declaration of multiple IRM channels inoperable while in Mode 2.			
Description: On November 25, 2017, operators received upscale alarms on IRMs A, C and D. Additionally, neither a half scram nor a full reactor scram was received despite IRM high and IRM trip lights being lit on the associated IRM chassis cabinets. Operators determined that the spiking IRM NIs and associated alarms were consistent with previous IRM NI noise-related spiking associated with source range monitor (SRM) withdrawal. Operators verified that the plant parameters were stable and that an actual power excursion reflective of an IRM upscale condition did not exist in the plant, but were unable to identify why a half or full reactor scram did not occur. The operators declared the three affected IRM NI channels inoperable, and the station subsequently made the decision to shut down the plant to diagnose and correct the cause of the three inoperable IRM NI channels.			

The inspectors reviewed the event and historic data associated with IRM spiking and the failure to receive associated half scram or full reactor scram signals. During the review, the inspectors identified a history of IRM spiking associated with electronic noise over a period of more than 5 years. In addition, the inspectors identified multiple condition reports (CR-GGN-2017-000214, 00976, 01052 and 02804) in which instances of IRM spiking without an expected corresponding half or full scram signal occurred. Each of the condition reports reviewed by the inspectors were closed without any substantive corrective actions taken to correct the underlying cause of the noise-related IRM spikes, nor were the reasons for half or full scram signals only being received in some of the instances but not others evaluated.

The inspectors reviewed licensee procedure EN-LI-102, "Corrective Action Program." EN-LI-102, Revision 29, defines, in part, a condition adverse to quality as "abnormal plant conditions or indications that cannot be readily explained or long-term, unexplained plant conditions," "unplanned actuations of RPS, ESF, or Emergency Power Systems," and "unplanned entry or failure to enter a LCO."

Corrective Actions: Following the event on November 25, 2017, and subsequent plant shutdown, the licensee initiated Condition Report CR-GGN-2017-11733. The inspectors reviewed the condition report and identified that the licensee developed corrective actions consisting of an operational decision making issue (ODMI) specifying actions following the failure to receive a scram signal during an IRM spike condition and changed procedures for the methods used for driving in SRMs to reduce noise being introduced into the system. Additionally, corrective actions to schedule the replacement of SRM drive motor connectors, scheduling of the installation of snubber circuits on contractors to reduce noise, and the conduct of a vibration analysis of under vessel cables to identify sources of electronic noise were also assigned. Entergy also performed an evaluation, with vendor support, to determine the cause of reactor protection system half or full scram signals not being received in some of the instances of IRM noise spiking.

Corrective Action Reference(s): Condition Report CR-GGN-2017-11733

Performance Assessment:

Performance Deficiency: The licensee's failure to identify and correct a condition adverse to quality associated with IRM spiking was a performance deficiency.

Screening: The inspectors determined the performance deficiency was more than minor because it was associated with the equipment performance attribute of the Mitigating Systems Cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, inadequate corrective actions associated with noise on IRM channels resulted in the inoperability of three channels of IRM NIs and an unplanned plant shutdown

Significance: The inspectors assessed the significance of the finding using Exhibit 2 of Insepction Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings-At-Power," dated June 19, 2012. The inspectors determined that the finding was of very low safety significance (Green) because it did not result in an actual loss of safety function of at least a single train for greater than its technical specification allowed outage time.

Cross-cutting Aspect: The cause of this finding is related to the cross-cutting area of Problem Identification and Resolution – Evaluation because Entergy personnel did not thoroughly evaluate issues to ensure that resolutions address causes and extent of conditions commensurate with their safety significance. Specifically, Entergy personnel failed to appropriately evaluate and resolve an adverse condition impacting safety-related IRMs.

Enforcement:

Violation: As required by 10 CFR Part 50, Appendix B, Criterion XVI, measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective materials and equipment, and nonconformances are promptly identified and corrected.

Contrary to the above, on November 25, 2017, the licensee failed to establish measures to assure that a condition adverse to quality was promptly identified and corrected. Specifically, Entergy failed to appropriately evaluate and correct multiple instances of IRM NI channel spiking caused by electronic noise that led to the inoperability of three channels of IRM NIs, an unexplained condition in which a half or full RPS scram was not received, and the eventual decision to conduct a plant shutdown for troubleshooting.

Disposition: This violation is being treated as a non-cited violation, consistent with Section 2.3.2 of the Enforcement Policy.

Loss of Shutdown Cooling

Cornerstone	Significance	Cross-cutting Aspect	Report Section
Initiating Events	Green NCV 05000416/2018002-07 Closed	[H.12] – Human Performance, Avoid Complacency	71153 – Followup of Events and Notices of Enforcement Discretion

A self-revealed, Green non-cited violation of Technical Specification 5.4, “Procedures,” for the licensee’s failure to follow written procedures was identified when the residual heat removal (RHR) system automatically isolated due to an inadvertent emergency core cooling system (ECCS) actuation. While the plant was shut down with the RHR system in decay heat removal mode, maintenance personnel inadvertently opened an incorrect valve during a transmitter calibration activity, which caused a false low reactor pressure vessel (RPV) water level signal, an ECCS actuation, and a loss of decay heat removal for approximately 31 minutes.

Description: On May 1, 2018, while the plant was shut down with RPV water level at the high water level, an invalid Division I RPV Level 1 signal was received. This signal caused an actuation of the Division I load shed and sequencing system that shed and subsequently reenergized loads supplied from the Division I 4160-volt safety bus.

RHR system A was operating in shutdown cooling mode at the time, with its suction aligned to the spent fuel pool and the discharge aligned to the upper containment pool. Both pools were connected via the transfer canal, with all gates open and the RPV head removed. Power to RHR pump A was lost during the bus shed and was resequenced upon reenergization of the bus. The RHR injection valve to the RPV opened for low pressure coolant injection mode, and the RHR system A began discharging into the RPV as well as the upper pool, with the suction source still aligned to the spent fuel pool. The RHR heat exchanger bypass valve opened upon the ECCS actuation, which resulted in a loss of the shutdown cooling function.

Operations personnel verified that no actual RPV level transient occurred and restored the shutdown cooling function within approximately 31 minutes of the inadvertent ECCS actuation. Reactor coolant system temperature increased approximately 5 degrees Fahrenheit (F) as a result of the loss of shutdown cooling. The calculated time to reach 200 degrees F at the time of the event was approximately 7 hours.

At the time of the event, maintenance personnel were performing Procedure 06-IC-1B21-R-2005, "Reactor Vessel Water Level (Levels 1 and 2) Calibration," Revision 108, to calibrate RPV level transmitter 1B21N081A under Work Order 52708988. This transmitter shared a common reference sensing line with other transmitters, including the Reactor Vessel Water Level 1 transmitters, 1B21N091A and 1B21N091E. With the low pressure side of the transmitter vented and depressurized, maintenance personnel incorrectly opened the low pressure isolation valve instead of the equalization valve. Step 5.5.11 of Procedure 06-IC-1B21-R-2005, "Reactor Vessel Water Level (Levels 1 and 2) Calibration," Revision 108, required maintenance personnel to open the equalizing valve F03 on the transmitter rack. Instead, maintenance personnel inappropriately opened the low pressure isolation valve F02. As a result, the pressure in the shared common reference sensing line suddenly decreased, which was sensed by level transmitters 1B21N091A and E and resulted in ECCS RPV level instruments sensing a Level 1 condition and providing the associated ECCS actuation signal.

Corrective Actions: Licensee corrective actions included identification of all instrumentation and control surveillances that have the potential to trip the plant or result in a safety system actuation, and requiring concurrent verification of all associated component manipulations by maintenance personnel.

Corrective Action Reference: The licensee entered the issue into the corrective action program as Condition Report CR-GGN-2018-07661.

Performance Assessment:

Performance Deficiency: The failure to perform maintenance in accordance with written procedures was a performance deficiency.

Screening: The performance deficiency was more than minor, and therefore a finding, because it was associated with the equipment performance attribute of the Initiating Events Cornerstone and affected the associated cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown operations. Specifically, the failure to follow maintenance procedure instructions resulted in an inadvertent ECCS initiation event that included a loss of shutdown cooling function. This upset plant stability by causing a flow transient in the residual heat removal system and challenged the critical safety function of decay heat removal.

Significance: The inspectors screened the significance of the finding using Inspection Manual Chapter 0609, Appendix G, Attachment 1, "Shutdown Operations Significance Process Phase 1 Initial Screening and Characterization of Findings," dated May 9, 2014. The inspectors determined that the finding did not require a quantitative risk assessment because the event occurred when the refuel canal/cavity was flooded. Therefore, the finding had very low safety significance (Green). Cross-cutting Aspect: The finding had a cross-cutting aspect in the area of human performance associated with avoiding complacency because maintenance personnel failed to implement appropriate error reduction tools. Specifically, the transmitter valves being manipulated during the calibration activity were not labeled and were located beside each other on the instrument panel. Personnel performing the activity reached down, without looking, and opened the wrong valve without first using appropriate error reduction tools to ensure that they were about to manipulate the correct valve.
<u>Enforcement</u>: Violation: Technical Specification 5.4.1.a requires, in part, that written procedures shall be established, implemented, and maintained covering the applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2. Section 9.a of Appendix A to Regulatory Guide 1.33, Revision 2, requires, "Maintenance that can affect the performance of safety-related equipment should be properly preplanned and performed in accordance with written procedures, documented instructions, or drawings appropriate to the circumstances." Step 5.5.11 of licensee Procedure 06-IC-1821-R-2005, "Reactor Water Vessel Water Level (Levels 1 and 2) Calibration," Revision 108, required, "On transmitter manifold, open equalizing valve F03." Contrary to the above, on May 1, 2018, the licensee failed to open equalizing valve F03 on the transmitter manifold. Specifically, while calibrating safety-related transmitter 1B21N081A under Work Order 52708988, maintenance personnel opened the low pressure isolation valve (F02) instead of the equalizing valve, which caused a loss of shutdown cooling for approximately 31 minutes. Disposition: This violation is being treated as a non-cited violation, consistent with Section 2.3.2 of the Enforcement Policy.

Performance of Surveillance Testing Following Maintenance on Containment Airlock			
Cornerstone	Significance	Cross-cutting Aspect	Report Section
Barrier Integrity	Green NCV 05000416/2018002-08 Closed	[H.5] – Human Performance, Work Management	71153 – Followup of Events and Notices of Enforcement Discretion
The inspectors identified a Green non-cited violation of 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," for the licensee's failure to perform surveillance testing of containment airlock seals under appropriate conditions. The licensee failed to appropriately control the sequence of maintenance and testing activities to ensure that surveillance testing was not performed subsequent to maintenance which could affect the validity of surveillance test results.			

Description: On April 15, 2015, and August 11, 2016, the licensee performed post-maintenance tests on the 119 foot elevation containment inflatable inner door seal following maintenance. These tests were also intended to be credited as satisfying the technical specification surveillance requirement to verify the ability of the seal to provide adequate leak tightness for containment integrity. As discussed in Licensee Event Report 05000416/2018-002-00 (ADAMS Accession No. ML18094A172) and NRC Inspection Report 05000416/2018001 (ADAMS Accession No. ML18134A007), the inspectors identified that the test data failed to meet the surveillance acceptance criteria on both of these occasions; however, the seals were inappropriately declared operable and returned to service.

In January 2017, the licensee performed Work Order 453422 to troubleshoot the 119 foot elevation containment airlock inner door. The licensee removed inflatable seals from the door in order to access components being repaired. Upon conclusion of the repair work, the seals were reinstalled, and the licensee performed Procedure 06-ME-1M23-R-0001, “Personnel Airlock Door Seal Air System Leak Test,” Revision 112, as a post-maintenance test for removing and reinstalling the inflatable door seals on January 10, 2017. The inspectors found that no technical specification surveillance requirement data was recorded for the inflatable seal system before removal of the seals or in the as-found condition. Similarly, the inspectors identified that no as-found test was performed during the August 2016 maintenance discussed above. Consequently, the inner door inflatable seal system had been left in an inoperable status after the technical specification surveillance requirement test was performed on April 15, 2015, and August 11, 2016.

The inspectors determined that as a result of not performing as-found testing of the inflatable seals in both 2016 and 2017, the inoperability of the seals from April 15, 2015, until January 10, 2017, remained undiscovered until an extent of condition review was conducted following the failure of the technical specification surveillance requirement test of the 208 foot elevation airlock in February 2018 (discussed in Licensee Event Report 05000416/2018-002-00 and NRC Inspection Report 05000416/2018001).

The inspectors reviewed NRC Information Notice 97-16, “Preconditioning of Plant Structures, Systems, and Components Before ASME Code Inservice Testing or Technical Specification Surveillance Testing,” and determined that the act of removing and reinstalling the inflatable seals was relevant to the Information Notice guidance because this practice bypassed or masked the as-found condition of the inflatable seal system.

The inspectors reviewed licensee Procedures 06-ME-1-M23-R-0001, “Personnel Airlock Door Seal Air System Leak Test,” Revision 116, and EN-WM-105, “Planning,” Revision 22. The inspectors noted that these procedures did not include provisions to evaluate whether an unacceptable sequence of maintenance and testing activities existed while planning or executing surveillance testing in conjunction with maintenance that has the possibility of affecting the performance of safety-related equipment.

Corrective Action(s): The licensee entered this issue into their corrective action program as Condition Report CR-GGN-2018-07661 to evaluate the maintenance and surveillance practices associated with the airlock inflatable seals and to develop appropriate corrective actions.

Corrective Action Reference: CR-GGN-2018-07661

<u>Performance Assessment</u>: Performance Deficiency: The failure to ensure that technical specification surveillance testing of the airlock door seals was performed under suitable conditions was a performance deficiency. Screening: This finding was more than minor because it was associated with the procedure quality attribute of the Barrier Integrity (containment) Cornerstone and adversely affected the cornerstone objective to provide reasonable assurance that physical design barriers (containment) protect the public from radionuclide releases caused by accidents or events. Additionally, if left uncorrected, the performance deficiency could lead to a more significant safety concern. Specifically, the inadequate testing practices of the inflatable seals contributed to the licensee not identifying the inoperable condition of the seals during the surveillance interval that followed maintenance on April 15, 2015, and August 11, 2016. Significance: The inspectors assessed the significance of the finding using Inspection Manual Chapter (IMC) 0609.04, “Initial Characterization of Findings,” and IMC 0609, Appendix A, Exhibit 3, “Barrier Integrity Screening Questions.” The inspectors determined that this finding did not represent an actual open pathway in the physical integrity of the reactor containment. Therefore, the inspectors determined the finding to be of very low safety significance (Green). Cross-cutting Aspect: This finding has a cross-cutting aspect of human performance associated with work management because the licensee did not implement a work planning process that identified the need for coordination with different groups and job activities.
<u>Enforcement</u>: Violation: As required by 10 CFR Part 50, Appendix B, Criterion XI, “Test Control,” testing is to be performed under suitable environmental conditions. Suitable environmental conditions include those that are representative of the expected standby configuration and the condition in which the equipment would be when required to perform its safety function. Contrary to the above, on August 8, 2016, and January 10, 2017, the licensee failed to assure that testing was performed under suitable environmental conditions. Specifically, the licensee performed surveillance testing of the safety-related inflatable door seals for the 119 foot elevation containment air lock inner door following maintenance that affected the condition of the equipment. Licensee test procedures did not include consideration of the performance of the surveillance testing relative to the performance of activities that could impact the validity of the surveillance testing. Disposition: This violation is being treated as a non-cited violation, consistent with Section 2.3.2 of the Enforcement Policy.

EXIT MEETINGS AND DEBRIEFS

The inspectors verified no proprietary information was retained or documented in this report.

On April 27, 2018, the inspectors presented the occupational radiation safety inspection results to Mr. E. Larson, Site Vice President, and other members of the licensee staff while onsite. On May 29, 2018, the inspectors conducted a telephonic re-exit and presented the final disposition of an open item and the occupational radiation safety inspection results to Mr. E. Larson, Site Vice President, and other members of the licensee staff.

On May 2, 2018, the inspectors presented the inservice inspection results to Mr. D. Byars, Manager, System Engineering, and other members of the licensee staff.

On July 24, 2018, the inspectors presented the quarterly resident inspector inspection results to Mr. B. Franssen, General Manager of Plant Operations, and other members of the licensee staff.

DOCUMENTS REVIEWED

71111.08—Inservice Inspection Activities

Drawings

Number	Title
HP-11-01	High Pressure Core Spray, Pipe to Elbow Circ. Weld / Outside Bio Shield at 240º AZ.
LP-11-02	Low-Pressure Coolant Injection System, LPCS Support With Thermal Movement / 150' EL at 120º AZ.
RI-08-01	RICI Pipe to Valve Circ. Weld / Steam Tunnel 142' EL

Miscellaneous Documents

Number	Title	Date
GNRI94-00184	NRC Letter to GGNS – Interim Extension of 120-Month Interval for Inservice Inspection and Inservice Testing (ISI/IST) Programs for GGNS (TAC No. M89274)	August 1, 1994
GNRI96-00244	NRC Letter to GGNS – Evaluation of Entergy Operations, Inc., Request for Authorization to Update Inservice Inspection Programs to the 1992 and Portions of the 1993 ASME Boiler and Pressure Vessel Code, Section XI for Arkansas Nuclear One, Units 1 and 2, Grand Gulf Nuclear Station, River Bend Station, and Waterford Steam Electric Station, Unit 3, (TAC Nos. M94472, M94471, M94454, and M94488)	December 12, 1996
LO-GLO-2017-50	GGNS – RF21 Inservice Inspection (ISI) Pre-NRC (71111.08) Assessment	January 15, 2018

Procedures

Number	Title	Revision
CEP-NDE-0255	Radiographic Examination – ASME, ANSI, AWS, API, AWWA Welds and Components	9
CEP-NDE-0400	Ultrasonic Examination	7
CEP-NDE-0404	Manual Ultrasonic Examination of Ferritic Piping Welds (ASME XI)	
CEP-NDE-0423	Manual Ultrasonic Examination of Austenitic Piping Welds (ASME XI)	8
CEP-NDE-0731	Magnetic Particle Examination (MT) for ASME Section XI	6
CEP-NDE-0902	VT-2 Examination	8
CEP-NDE-0903	VT-3 Examination	6

Procedures

Number	Title	Revision
CEP-NDE-0965	Visual Welding Inspection ASME, ANSI B31.1	5
SEP-ISI-GGN-001	Program Section for ASME Section XI, Division 1 GGNS Inservice Inspection Program	5

Condition Reports (CRs)

CR-GGN-2016-01707	CR-GGN-2016-01802	CR-GGN-2017-06356	CR-GGN-2017-07866
CR-GGN-2018-01616	CR-GGN-2018-04303	CR-GGN-2018-04561	

Work Orders

391224	391254	391470	471203-04

71124.01—Radiological Hazard Assessment and Exposure Controls

Air Sample Surveys

Number	Title	Date
AS-GGN-2018-04309	Aux 119 –RWCU B Pump Breach	January 23, 2018
AS-GGN-2018-04797	DW 100 – Drywell U/V Pulling PIP	April 18, 2018
AS-GGN-2018-04800	DW 121 – 6 Inch RHR Pipe-Decon	April 18, 2018
AS-GGN-2018-04804	DW 100 – U/V LPRM Pull	April 18, 2018
AS-GGN-2018-04809	DW 121 – Suppression Pool Cut Out of RHR Line	April 18, 2018
AS-GGN-2018-04856	CTMT 208 – Work Under RPV Head	April 20, 2018
AS-GGN-2018-04989	TB 166 – 1N11F0260 Stop/Control Valve	April 26, 2018

Audits and Self-Assessments Number

Number	Title	Date
LO-GLO-2017-00051	Pre-NRC Radiological Hazard Assessment and Exposure Controls (71124.01)	November 15, 2017

Audits and Self-Assessments Number

Number	Title	Date
LO-GLO-2017-0053	Radiation Protection Pre-NRC Occupational Exposure Control Effectiveness (71151-OR01)	November 15, 2017

Miscellaneous Documents

Number	Title	Date
	GGNS Plant Awareness Note - Radiography	April 23, 2018
	Inventory of Radioactive Sources	2018
	Non-Nuclear Inventory Form	July 11, 2017
2018-02718	Operating Experience Fleet Bulletin: Unplanned Dose Rate Alarm	March 23, 2018
2018-1800	TEDE-ALARA Evaluation	April 23, 2018
2018-1800	Respiratory Protection Permit	April 23, 2018
GIN-2018-00003	NSTS Annual Inventory Reconciliation Report	January 9, 2018
WO 52759183	Semi Annual Leak Test of Sealed Sources	November 8, 2017

Procedures

Number	Title	Revision
EN-LI-114	Regulatory Performance Indicator Process	11
EN-RP-101	Access Control for Radiologically Controlled Areas	14
EN-RP-102	Radiological Control	6
EN-RP-108	Radiation Protection Posting	20
EN-RP-121	Radioactive Material Control	14
EN-RP-131	Air Sampling	15
EN-RP-142	Failed Fuel Response	2
EN-RP-143	Source Control	13
EN-RP-150	Radiography and X-Ray Testing	14
EN-RP-151	Radiological Diving	3
EN-RP-202	Personnel Monitoring	13
EN-RP-204	Special Monitoring Requirements	11
EN-RP-303-1	Automated Contamination Monitor Performance Testing	1

Procedures

Number	Title	Revision
EN-RP-311	Electronic Alarming Dosimeters	2
EN-RP-314	Passive Monitor Sensitivity Testing	0

Radiation Surveys

Number	Title	Date
GGN-1603-0206	166 TB Center Section: Pre and Post-Decon of Stop and Control Valve	March 3, 2016
GGN-1802-0322	139 TB Condensate Demineralizers	February 26, 2018
GGN-1803-0273	166 TB North Section	March 23, 2018
GGN-1803-0275	166 TB North Section	March 23, 2018
GGN-1804-0402	147 Drywell	April 9, 2018
GGN-1804-1007	147 Drywell	April 16, 2018
GGN-1804-1474	113 TB Seal Steam Generator Room	April 20, 2018
GGN-1804-1848	113 TB Seal Steam Generator Room	April 23, 2018
GGN-1804-1904	93 AB RHR B Pump Room	April 21, 2018
GGN-1804-2134	166 TB Center Section	April 25, 2018

Radiation Work Permits

Number	Title	Revision
20181402	Refuel Floor High Water Activities	0
20181403	RX Vessel Disassembly and Re-Assembly	2
20181407	Aux 208’ Spent Fuel Area Fuel Rechanneling and Inspection	1
20181508	Under Vessel Maintenance	1
20181512	Remove and Replace Main Stream Relief Valves	0
20181525	1B21F016 Valve Replacement and Support Work	0
20181534	1B33F023A Seal Weld	0
20181800	Turbine Building Work	5
20181952	Radiography and ARM Calibration in Drywell and Containment	0

Condition Reports (CRs)

CR-GGN-2017-04215	CR-GGN-2017-04698	CR-GGN-2017-06915	CR-GGN-2017-06916
CR-GGN-2017-08967	CR-GGN-2017-08994	CR-GGN-2017-11051	CR-GGN-2018-00355
CR-GGN-2018-00837	CR-GGN-2018-01852	CR-GGN-2018-02141	CR-GGN-2018-02281
CR-GGN-2018-02718	CR-GGN-2018-03419	CR-GGN-2018-03445	CR-GGN-2018-03627

71124.02—Occupational ALARA Planning and Controls

ALARA Planning, In-Progress Review and Post-Job Review

Number	Title	Revision
20171083	RHR A System Recovery	01

Audits and Self-Assessments Number

Number	Title	Date
LO-GLO-2017-0052	Pre-NRC Inspection: Occupational ALARA Planning and Controls Assessment (71124.02)	November 19, 2017
QA-14-15-2017-GGNS-1	Combined Radiation and Radwaste Quality Assurance Audit Report	October 23, 2017

Miscellaneous Documents

Number	Title	Date
	GGNS 5-Year Exposure Reduction Plan 2018-2022	November 19, 2017
	GGNS Refueling Outage 20 Station Outage ALARA Report	2017
	RF-21 Daily RP Outage Report	April 23 - 26, 2018
CR-GGN-2018-04288	Human Performance Evaluation: Dose Alarm/PCE and associated uptake	May 2, 2018

Procedures

Number	Title	Revision
EN-DC-341	Cobalt Reduction	06
EN-HU-102	Human Performance Traps and Tools	15
EN-RP-104	Personnel Contamination Events	10
EN-RP-105	Radiological Work Permits	18

Procedures

Number	Title	Revision
EN-RP-110	ALARA Program	14
EN-RP-110-03	Collective Radiation Exposure Reduction Guidelines	04
EN-RP-110-04	Radiation Protection Risk Assessment Process	07
EN-RP-110-06	Outage Dose Estimating and Tracking	01
EN-RP-202	Personnel Monitoring	13
EN-RP-203	Dose Assessment	10
EN-RP-208	Whole Body Counting/In-Vitro Bioassay	07
EN-RP-401	Decontamination Program	06

Radiation Surveys

Number	Title	Date
GGN-1804-0719	208 CTMT HFTS Dive Survey	April 13, 2018
GGN-1804-0818	208 CTMT Underneath RPV Head	April 14, 2018
GGN-1804-1260	147 Drywell Entire Elevation	April 18, 2018
GGN-1804-1336	114 Drywell Entire Elevation	April 18, 2018
GGN-1804-1555	RPV Head	April 20, 2018
GGN-1804-1723	208 CTMT Containment Pool Area	April 22, 2018
GGN-1804-1997	161 Drywell Entire Elevation	April 24, 2018

Radiation Work Permits

Number	Title	Revision
20181006	Radwaste Shipping and Processing	00
20181054	Locked High Radiation Area Entries for Plant/System Investigations, Valve Manipulations, Tagouts, and Misc. Activities	00
20181057	Refuel Floor Pre/Post-Outage Activities and Dry Fuel Preps	01
20181060	Pre-RFO-21 Activities	01
20181400	RP/Decon Support for Refuel Floor	01
20181402	Refuel Floor High Water Activities	00
20181403	Rx Vessel Disassembly and Re-Assembly	01
20181800	Turbine Building Work	02

Radiation Work Permits

Number	Title	Revision
20181952	Radiography and ARM Calibration in Drywell and Containment	00

Condition Reports (CRs)

CR-GGN-2017-02422	CR-GGN-2017-03779	CR-GGN-2017-08146	CR-GGN-2017-08918
CR-GGN-2017-09972	CR-GGN-2017-10851	CR-GGN-2017-10949	CR-GGN-2017-11497
CR-GGN-2018-00144	CR-GGN-2018-02755	CR-GGN-2018-02838	CR-GGN-2018-03068
CR-GGN-2018-03517	CR-GGN-2018-04288	CR-GGN-2018-04298	CR-HQN-2017-00794

71152—Problem Identification and Resolution

Procedures

Number	Title	Revision/Date
01-S-02-9	Procedure Change Process	6
01-S-06-26	Post-Trip Analysis	24
04-1-02-1H13-P680-7A-A9	Alarm Response Instruction - IRM Upscale Alarm	100
04-1-02-1H13-P680-7A-A9	Alarm Response Instruction – RPS CH A IRM UPSC Trip Inop	100
05-S-01-EP-2	RPV Control	46
Cause Analysis	Inoperability of Multiple IRMs Results in Plant Shutdown	November 25, 2017
EC 75015	Neutron Monitoring System (IRM) Trip Inputs to RPS CR-GGN-2017-11733	0
EN-LI-102	Corrective Action Program	29
EN-OP-115	Conduct of Operations	24
EN-OP-200	Plant Transient Response Rule	4
ODMI	Nuclear Instrumentation Monitoring During Startup	November 28, 2017

Condition Reports (CRs)

CR-GGN-2016-05557	CR-GGN-2017-00214	CR-GGN-2017-00976	CR-GGN-2017-01052
CR-GGN-2017-03804	CR-GGN-2017-07765	CR-GGN-2017-11733	CR-GGN-2017-11753
CR-GGN-2018-03277	CR-GGN-2018-05030		

The following items are requested for the
Occupational Radiation Safety Inspection
Grand Gulf Nuclear Station
April 23 thru 27, 2018
Integrated Report 2018002

Inspection areas are listed in the attachments below.

Please provide the requested information on or before **April 2, 2018**.

Please submit this information using the same lettering system as below. For example, all contacts and phone numbers for Inspection Procedure 71124.0**1** should be in a file/folder titled "**1**- A," applicable organization charts in file/folder "**1**- B," etc.

If information is placed on *ims.certrec.com*, please ensure the inspection exit date entered is at least 30 days later than the onsite inspection dates, so the inspectors will have access to the information while writing the report.

In addition to the corrective action document lists provided for each inspection procedure listed below, please provide updated lists of corrective action documents at the entrance meeting. The dates for these lists should range from the end dates of the original lists to the day of the entrance meeting.

If more than one inspection procedure is to be conducted and the information requests appear to be redundant, there is no need to provide duplicate copies. Enter a note explaining in which file the information can be found.

If you have any questions or comments, please contact Natasha Greene at (817) 200-1154 or natasha.greene@nrc.gov.

PAPERWORK REDUCTION ACT STATEMENT

1. Radiological Hazard Assessment and Exposure Controls (71124.01) and Performance Indicator Verification (71151)

Date of Last Inspection: **February 13, 2017**

A. List of contacts and telephone numbers for the Radiation Protection Organization Staff and Technicians

B. Applicable organization charts

C. Audits, self-assessments, and LERs written since date of last inspection, related to this inspection area

D. Procedure indexes for the radiation protection procedures

E. Please provide specific procedures related to the following areas noted below. Additional Specific Procedures may be requested by number after the inspector reviews the procedure indexes.

1. Radiation Protection Program Description
2. Radiation Protection Conduct of Operations
3. Personnel Dosimetry Program
4. Posting of Radiological Areas
5. High Radiation Area Controls
6. Radiologically Controlled Area Access Controls and Radiation Worker Instructions
7. Conduct of Radiological Surveys
8. Radioactive Source Inventory and Control
9. Declared Pregnant Worker Program

F. List of corrective action documents (including corporate and sub-tiered systems) since date of last inspection

a. Initiated by the radiation protection organization
b. Assigned to the radiation protection organization

NOTE: The lists should indicate the significance level of each issue and the search criteria used. Please provide in document formats which are “searchable” so that the inspector can perform word searches.

If not covered above, a summary of corrective action documents since date of last inspection involving unmonitored releases, unplanned releases, or releases in which any dose limit or administrative dose limit was exceeded (for Public Radiation Safety Performance Indicator verification in accordance with IP 71151)

G. List of radiologically significant work activities scheduled to be conducted during the inspection period (If the inspection is scheduled during an outage, please also include a list of work activities greater than 1 rem, scheduled during the outage with the dose estimate for the work activity.)

H. List of active radiation work permits

I. Radioactive source inventory list

   a. All radioactive sources that are required to be leak tested

   b. All radioactive sources that meet the 10 CFR Part 20, Appendix E, Category 2 and above threshold. Please indicate the radioisotope, initial and current activity (w/assay date), and storage location for each applicable source.

J. The last two leak test results for the radioactive sources inventoried and required to be leak tested. If applicable, specifically provide a list of all radioactive source(s) that have failed its leak test within the last two years

K. A current listing of any non-fuel items stored within your pools, and if available, their appropriate dose rates (Contact / @ 30cm)

L. Computer printout of radiological controlled area entries greater than 100 millirem since the previous inspection to the current inspection entrance date. The printout should include the date of entry, some form of worker identification, the radiation work permit used by the worker, dose accrued by the worker, and the electronic dosimeter dose alarm set-point used during the entry (for Occupational Radiation Safety Performance Indicator verification in accordance with IP 71151).

2. Occupational ALARA Planning and Controls (71124.02)

Date of Last Inspection: **November 7, 2016**

A. List of contacts and telephone numbers for ALARA program personnel

B. Applicable organization charts

C. Copies of audits, self-assessments, and LERs, written since date of last inspection, focusing on ALARA

D. Procedure index for ALARA Program

E. Please provide specific procedures related to the following areas noted below. Additional Specific Procedures may be requested by number after the inspector reviews the procedure indexes.
   1. ALARA Program
   2. ALARA Committee
   3. Radiation Work Permit Preparation

F. A summary list of corrective action documents (including corporate and sub-tiered systems) written since date of last inspection, related to the ALARA program. In addition to ALARA, the summary should also address Radiation Work Permit violations, Electronic Dosimeter Alarms, and RWP Dose Estimates

   NOTE: The lists should indicate the significance level of each issue and the search criteria used. Please provide in document formats which are "searchable" so that the inspector can perform word searches.

G. List of work activities greater than 1 rem, since date of last inspection, Include original dose estimate and actual dose.

H. Site dose totals and 3-year rolling averages for the past 3 years (based on dose of record)

I. Outline of source term reduction strategy

J. If available, provide a copy of the ALARA outage report for the most recently completed outages for each unit

K. Please provide your most recent Annual ALARA Report.

GRAND GULF NUCLEAR STATION – NRC INTEGRATED INSPECTION REPORT 05000416/2018002 – AUGUST 2, 2018

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