UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
1600 E. LAMAR BLVD.
ARLINGTON, TX 76011-4511

August 3, 2016

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

SUBJECT: GRAND GULF NUCLEAR STATION - NRC INTEGRATED INSPECTION REPORT 05000416/2016002

Dear Mr. Fallacara:

On June 30, 2016, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Grand Gulf Nuclear Station. On June 28, 2016, the NRC inspectors discussed the results of this inspection with Mr. G. Hawkins, Acting General Manager Plant Operations, and other members of your staff. Inspectors documented the results of this inspection in the enclosed inspection report.

NRC inspectors documented two findings of very low safety significance (Green) in this report. One of these findings involved a violation of NRC requirements. The NRC is treating the violation as a non-cited violation (NCV) consistent with Section 2.3.2.a of the NRC Enforcement Policy. The other finding did not involve a violation of NRC requirements.

If you contest the violation or significance of the NCV, 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, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC resident inspector at the Grand Gulf Nuclear Station.

If you disagree with a cross-cutting aspect assignment or a finding not associated with a regulatory requirement in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region IV; and the NRC resident inspector at the Grand Gulf Nuclear Station.

In accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) 2.390, “Public Inspections, Exemptions, Requests for Withholding,” a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC’s Public Document Room or from the Publicly Available Records (PARS) component of the NRC's

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Sincerely,

/RA/

Greg Warnick, Branch Chief
Project Branch C
Division of Reactor Projects

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

Enclosure:
Inspection Report 05000416/2016002
w/Attachments:
1: Supplemental Information
2. Request for Information

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Letter to Vin Fallacara from Greg Warnick dated August 3, 2016

SUBJECT: GRAND GULF NUCLEAR STATION - NRC INTEGRATED INSPECTION REPORT 05000416/2016002

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ROPreports.Resource@nrc.gov
ROPassessment.Resource@nrc.gov

U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Docket:	05000416
License:	NPF-29
Report:	05000416/2016002
Licensee:	Entergy Operations, Inc.
Facility:	Grand Gulf Nuclear Station, Unit 1
Location:	7003 Baldhill Road Port Gibson, MS 39150
Dates:	April 1 through June 30, 2016
Inspectors:	M. Young, Senior Resident Inspector N. Day, Resident Inspector G. George, Senior Reactor Inspector P. Hernandez, Health Physicist
Approved By:	Greg Warnick Chief, Project Branch C Division of Reactor Projects

SUMMARY

IR 05000416/2016002; 04/01/2016 - 06/30/2016, Grand Gulf Nuclear Station; operability determinations and functionality assessments, follow-up of events and notices of enforcement discretion

The inspection activities described in this report were performed between April 1 and June 30, 2016, by the resident inspectors at Grand Gulf Nuclear Station and inspectors from the NRC's Region IV office. Two findings of very low safety significance (Green) are documented in this report. One of these findings involved a violation of NRC requirements, and one of these findings did not involve a violation of NRC requirements. The significance of inspection findings is indicated by their color (Green, White, Yellow, or Red), which is determined using Inspection Manual Chapter 0609, "Significance Determination Process." Their cross-cutting aspects are determined using Inspection Manual Chapter 0310, "Aspects within the Cross-Cutting Areas." Violations of NRC requirements are dispositioned in accordance with the NRC Enforcement Policy. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process."

Cornerstone: Initiating Events

- Green. The inspectors reviewed a Green, self-revealed finding of Procedure EN-WM-105, "Planning," Revision 16, for the failure to ensure Work Order 397549 provided detailed instructions for performing troubleshooting on the 'B' phase of the main transformer. Specifically, Work Order 397549 did not contain detailed instructions for performing troubleshooting on the 'B' phase of the main transformer, which resulted in an incorrect current transformer ratio and subsequent reactor scram. The licensee's corrective actions were to incorporate more detailed instructions to the work order, repair the improper wiring, and restore the main transformer prior to transitioning from Mode 3 to Mode 1. Inspectors did not identify a violation of regulatory requirements associated with this finding. This finding was entered into the licensee's corrective action program as Condition Report CR-GGN-1-2016-02950.

  The failure to ensure Work Order 397549 provided detailed instructions for performing troubleshooting on the 'B' phase of the main transformer in accordance with Procedure EN-WM-105 was a performance deficiency. This performance deficiency is more than minor, and therefore a finding, because it is associated with the procedure quality attribute of the Initiating Events Cornerstone and adversely affected the cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, on March 29, 2016, the licensee failed to ensure Work Order 397549 provided detailed instructions for performing troubleshooting on the 'B' phase of the main transformer, which resulted in an incorrect current transformer wiring ratio and subsequent reactor scram. Using Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," and Inspection Manual Chapter 0609, Appendix A, Exhibit 1, "Initiating Events Screening Questions," the inspectors determined that the finding was of very low safety significance (Green) because the finding did result in a reactor trip, but did not result in the loss of mitigating equipment relied upon to transition the plant from the onset of the trip to a stable shutdown condition. This finding has a cross-cutting aspect in the area of human performance associated with field presence, in that, senior managers failed to ensure supervisory and management oversight of work activities, including contractors and supplemental personnel. Specifically, while performing Work Order 397549, the licensee

did not have contractor oversight established, and the contract workers performed troubleshooting without detailed instructions to ensure work was performed properly (Section 4OA3). [H.2]

Cornerstone: Barrier Integrity

- Green. The inspectors identified a Green, non-cited violation of Technical Specification Surveillance Requirement 3.0.1, for the failure to meet Surveillance Requirement 3.6.4.1.1 and declare Limiting Condition for Operation 3.6.4.1 not met. Specifically, the licensee did not maintain the enclosure building hatch penetration in the closed position as required by Surveillance Requirement 3.6.4.1.1, which resulted in secondary containment being inoperable. The licensee restored compliance by closing the hatch following the surveillance, and put corrective actions in place to control the enclosure building hatch penetration in a closed position except for entry and exit for the inspection. This finding was entered into the licensee's corrective action program as Condition Report CR-GGN-1-2016-03707.

  The failure to declare that Limiting Condition for Operation 3.6.4.1 was not met when the enclosure building hatch was maintained in the open position was a performance deficiency. This performance deficiency is more than minor, and therefore a finding, because it is associated with the configuration control attribute of the Barrier Integrity Cornerstone and adversely affected the cornerstone objective to provide reasonable assurance that physical design barriers (secondary containment) protect the public from radionuclide releases caused by accidents or events. Specifically, on April 7, 2016, the licensee did not maintain the enclosure building hatch penetration in the closed position as required by SR 3.6.4.1.1, which resulted in secondary containment being inoperable. Using Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," and Inspection Manual Chapter 0609, Appendix A, Exhibit 3, "Barrier Integrity Screening Questions," the inspectors determined that the finding was of very low safety significance (Green) because the finding only represented a degradation of the radiological barrier function provided for the control room, or auxiliary building, or spent fuel pool, or standby gas treatment (SBGT) system (BWR). This finding has a cross-cutting aspect in the area of human performance associated with documentation, in that, the organization failed to create and maintain complete, accurate and up-to-date documentation. Specifically, Work Order 52671695 for implementing the roof inspection was not complete and accurate with regards to the impact on operability of secondary containment when leaving the enclosure building hatch penetration open during inspection activities (Section 1R15). [H.7]

PLANT STATUS

The Grand Gulf Nuclear Station began the inspection period at 49 percent power.

On June 17, 2016, at 60 percent power, an automatic reactor scram occurred due to oscillating power range monitors providing input to the reactor protection system. The power oscillations were caused by the fast closure of the turbine stop valves B and D. Grand Gulf Nuclear Station stabilized in Mode 3.

On June 18, 2016, operators commenced power ascension, and on June 25, 2016, Grand Gulf Nuclear Station reached 100 percent power.

On June 25, 2016, at 100 percent power, an automatic reactor scram occurred due to a fast closure of turbine control valves B, C, and D. Grand Gulf Nuclear Station stabilized in Mode 3.

On June 27, 2016, operators transitioned the plant to Mode 4 to investigate and correct the cause of the scram.

Grand Gulf Nuclear Station remained in Mode 4 at the end of the inspection period.

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

.1 Summer Readiness for Offsite and Alternate AC Power Systems

a. Inspection Scope

On June 10, 2016, the inspectors completed an inspection of the station's offsite and alternate AC power systems. The inspectors inspected the material condition of these systems, including transformers and other switchyard equipment to verify that plant features and procedures were appropriate for operation and continued availability of offsite and alternate AC power systems. The inspectors walked down the switchyard to observe the material condition of equipment providing offsite power sources.

The inspectors verified that the licensee's procedures included appropriate measures to monitor and maintain availability and reliability of the offsite and alternate AC power systems.

These activities constituted one sample of summer readiness of offsite and alternate AC power systems, as defined in Inspection Procedure 71111.01.

b. Findings

No findings were identified.

.2 Readiness to Cope with External Flooding

a. Inspection Scope

On April 7, 2016, the inspectors completed an inspection of the station's readiness to cope with external flooding. After reviewing the licensee's flooding analysis, the inspectors chose five plant areas that were susceptible to flooding:

- Enclosure building roof
- Auxiliary building roof
- Control building roof
- Diesel generator building roof
- Control building and diesel generator building flood protection of the doors

The inspectors reviewed plant design features and licensee procedures for coping with flooding. The inspectors walked down the selected areas to inspect the design features, including the material condition of seals, drains, and flood barriers. The inspectors evaluated whether credited operator actions could be successfully accomplished.

These activities constituted one sample of readiness to cope with external flooding, as defined in Inspection Procedure 71111.01.

b. Findings

No findings were identified.

1R04 Equipment Alignment (71111.04)

a. Inspection Scope

The inspectors performed partial system walkdowns of the following risk-significant systems:

- May 17, 2016, high pressure core spray system, due to a reactor core isolation cooling system maintenance outage

- May 18, 2016, division III diesel generator, due to a reactor core isolation cooling system maintenance outage

- May 24, 2016, division II diesel generator, due to division I diesel generator maintenance outage

The inspectors reviewed the licensee's procedures and system design information to determine the correct lineup for the systems. They visually verified that critical portions of the systems or trains were correctly aligned for the existing plant configuration.

These activities constituted three partial system walkdown samples, as defined in Inspection Procedure 71111.04.

b. Findings

No findings were identified.

1R05 Fire Protection (71111.05)

a. Inspection Scope

The inspectors evaluated the licensee's fire protection program for operational status and material condition. The inspectors focused their inspection on five plant areas important to safety:

- May 27, 2016, fire area 2, fire zone 1A104, reactor core isolation cooling pump room
- June 2, 2016, fire area 25A, fire zones OC407 and OC409, lower inverter room and electrical space room
- June 2, 2016, fire areas 25A and 25B, fire zones OC706, OC707, OC708 and OC709, corridor, motor-generator (MG) room, instrument MG room, electrical space
- June 3, 2016, fire area 66, fire zones OM101, OM102, and OM103, fire water pump house and storage tanks
- June 22, 2016, fire areas 60, 61, 62 and 63, fire zones 1D301, 1D310, 1D308, and 1D306, diesel generator building breezeway, division I diesel generator room, division II diesel generator room, and division III diesel generator room

For each area, the inspectors evaluated the fire plan against defined hazards and defense-in-depth features in the licensee's fire protection program. The inspectors evaluated control of transient combustibles and ignition sources, fire detection and suppression systems, manual firefighting equipment and capability, passive fire protection features, and compensatory measures for degraded conditions.

These activities constituted five quarterly inspection samples, as defined in Inspection Procedure 71111.05.

b. Findings

No findings were identified.

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

.1 Review of Licensed Operator Requalification

a. Inspection Scope

On June 18, 2016, the inspectors observed simulator training for an operating crew during just-in-time training for a reactor startup following the scram that occurred on

June 17, 2016. The inspectors assessed the performance of the operators and the evaluators' critique of their performance.

These activities constituted completion of one quarterly licensed operator requalification program sample, as defined in Inspection Procedure 71111.11.

b. Findings

No findings were identified.

.2 Review of Licensed Operator Performance

a. Inspection Scope

The inspectors observed the operators' performance of the following activities:

- April 12, 2016, the inspectors observed the performance of on-shift licensed operators in the plant's main control room. At the time of the observations, the plant was in a period of heightened activity and risk due to the recovery of a control rod following a hydraulic control unit control rod issue which inserted a control rod.

- June 17, 2016, the inspectors observed the performance of on-shift licensed operators in the plant's main control room. At the time of the observations, the plant was in a period of heightened activity and risk due to an unplanned plant scram. The inspectors observed the operators transition the plant to Mode 3 and stabilize the conditions.

In addition, the inspectors assessed the operators' adherence to plant procedures, including Procedure EN-OP-115, "Conduct of Operations," Revision 17, and other operations department policies.

These activities constituted completion of two quarterly licensed operator performance samples, as defined in Inspection Procedure 71111.11.

b. Findings

No findings were identified.

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

a. Inspection Scope

The inspectors reviewed four risk assessments performed by the licensee prior to changes in plant configuration and the risk management actions taken by the licensee in response to elevated risk:

- May 10, 2016, containment venting in yellow risk with reactor core isolation cooling out of service but considered available

- May 25, 2016, containment venting in yellow risk with the division I diesel generator out of service for maintenance

- May 31 through June 1, 2016, containment venting in yellow risk with division III diesel generator out of service for a maintenance outage

- June 27, 2016, forced outage 21-03, outage risk assessment, decay heat removal in yellow risk

The inspectors verified that these risk assessments were performed timely and in accordance with the requirements of 10 CFR 50.65 (the Maintenance Rule) and plant procedures. The inspectors reviewed the accuracy and completeness of the licensee's risk assessments and verified that the licensee implemented appropriate risk management actions based on the result of the assessments.

On April 11, 2016, the inspectors observed portions of an emergent work activity when the hydraulic control unit 08-49BM cap screw failure resulted in control rod 08-49 drifting from notch 48 (fully withdrawn) to notch 00 (fully inserted). This emergent work had the potential to cause an initiating event and impact barrier integrity.

The inspectors verified that the licensee appropriately developed and followed a work plan for these activities. The inspectors verified that the licensee took precautions to minimize the impact of the work activities on unaffected structures, systems, and components.

These activities constituted completion of five maintenance risk assessments and emergent work control inspection samples, as defined in Inspection Procedure 71111.13.

b. Findings

No findings were identified.

1R15 Operability Determinations and Functionality Assessments (71111.15)

a. Inspection Scope

The inspectors reviewed seven operability determinations and functionality assessments that the licensee performed for degraded or nonconforming structures, systems, or components:

- April 5, 2016, reactor core isolation cooling fill and vent of instrumentation that caused an isolation of reactor core isolation cooling

- April 6, 2016, operability determination of recirculation loop B flow control valve drift

- April 7, 2016, operability determination of the average power range monitors during startup with 52 local power range monitors in bypass

- April 14, 2016, functionality assessment of the area radiation monitor for the separator storage area

- April 11-20, 2016, operability determination for secondary containment while the enclosure building roof hatch was open during roof inspections

- April 18, 2016, operability determination for hydraulic control units with corroded cap screws

- April 22-26, 2016, functionality assessment of the overspeed trip devices for the main turbine generator

The inspectors reviewed the timeliness and technical adequacy of the licensee's evaluations. Where the licensee determined the degraded structures, systems, or components to be operable and functional, the inspectors verified that the licensee's compensatory measures were appropriate to provide reasonable assurance of operability and functionality. The inspectors verified that the licensee had considered the effect of other degraded conditions on the operability and functionality of the degraded structures, systems, or components.

These activities constituted completion of seven operability and functionality review samples, as defined in Inspection Procedure 71111.15.

b. Findings

Introduction. The inspectors identified a Green, non-cited violation of Technical Specification Surveillance Requirement 3.0.1, for the failure to meet Surveillance Requirement 3.6.4.1.1 and declare that Limiting Condition for Operation (LCO) 3.6.4.1 not met when the enclosure building hatch was being maintained in the open position during roof inspections.

Description. On April 7, 2016, the inspectors were observing the performance of the quarterly enclosure building roof inspection, which utilized Work Order 52671695. During the inspection, the inspectors observed the licensee leave the roof penetration hatch in the open position during the 20 – 30 minute duration of the inspection. When proceeding through the enclosure building hatch, the inspector noticed a placard that stated, "Caution Secondary Containment Penetration. Do Not Open/Remove Without Shift Supervisor's Permission." The inspectors questioned the contractor whether or not they contacted the shift supervisor. The work order being used stated to contact the work control center, which is typically staffed with a licensed senior reactor operator. The inspectors noted that the work order did not include any information in the operational impact section or the instructional steps that discussed the effect on secondary containment when performing the enclosure building roof inspection.

The inspectors reviewed the technical specifications and design basis for the facility, and questioned the licensee with regards to secondary containment operability based on the information below.

- Technical Specification Surveillance Requirement (SR), 3.6.4.1.1 states, "Verify all auxiliary building and enclosure building equipment hatches and blowout panels are closed and sealed," and SR 3.6.4.1.2 states, "Verify one auxiliary building and enclosure building access door in each access opening is closed, except when the access opening is being used for entry and exit."

- Procedure 06-OP-1T48-M-003, "Secondary Containment Integrity Check," Revision 109, is the procedure that was written to perform actions that satisfy

SR 3.6.4.1.1 and SR 3.6.4.1.2. Step 5.2.3 states, in part, "**VERIFY** that all blowout panels **<u>AND</u>** hatches listed on Data Sheet II are Closed **<u>AND</u>** Sealed. Acceptance Criteria: All required equipment hatches and blowout panels Must be Closed **<u>AND</u>** Sealed." Data Sheet II explicitly calls out the hatch to access the enclosure building roof needs to be closed and sealed. Therefore, when this hatch is opened and maintained opened during the inspection without controls established, SR 3.6.4.1.1 is not met, resulting in LCO 3.6.4.1 not being met and secondary containment inoperability.

- Technical specification bases for SR 3.6.4.1.1 and 3.6.4.1.2 state, in part, "Maintaining secondary containment OPERABILITY requires verifying one door in the access opening is closed, except when the access opening is being used for entry and exit or when maintenance is being performed on an access opening."

Subsequently, the licensee determined that during this inspection activity, the enclosure building hatch penetration was not maintained closed in accordance with the surveillance requirements. This resulted in the secondary containment being declared inoperable for the time the penetration hatch was open. The licensee issued a Licensee Event Report 2016-003-00, "Loss of Secondary Containment Safety Function during Routine Roof Inspection," under 10 CFR 50.73(a)(2)(v)(C) for an event or condition that could have prevented the fulfillment of a safety function of structures or systems that are needed to: (C) control the release of radioactive material. In this report, the licensee identified that secondary containment was inoperable a minimum of 30 times in the past five years, and that the duration of each was approximately 20-30 minutes.

<u>Analysis</u>. The failure to declare that LCO 3.6.4.1 was not met when the enclosure building hatch was maintained in the open position was a performance deficiency. This performance deficiency is more than minor, and therefore a finding, because it is associated with the configuration control attribute of the Barrier Integrity Cornerstone and adversely affected the cornerstone objective to provide reasonable assurance that physical design barriers (secondary containment) protect the public from radionuclide releases caused by accidents or events. Specifically, on April 7, 2016, the licensee did not maintain the enclosure building hatch penetration in the closed position as required by SR 3.6.4.1.1, which resulted in secondary containment being inoperable. Using Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," and Inspection Manual Chapter 0609, Appendix A, Exhibit 3, "Barrier Integrity Screening Questions," the inspectors determined that the finding was of very low safety significance (Green) because the finding only represented a degradation of the radiological barrier function provided for the control room, or auxiliary building, or spent fuel pool, or standby gas treatment (SBGT) system (BWR).

This finding has a cross-cutting aspect in the area of human performance associated with documentation, in that, the organization failed to create and maintain complete, accurate and up-to-date documentation. Specifically, Work Order 52671695 for implementing the roof inspection was not complete and accurate with regards to the impact on operability of secondary containment when leaving the enclosure building hatch penetration open during inspection activities. [H.7]

Enforcement. Surveillance Requirement 3.0.1 requires, in part, that "failure to meet a Surveillance… shall be a failure to meet the LCO." Surveillance Requirement 3.6.4.1.1, states, "Verify all auxiliary building and enclosure building equipment hatches and blowout panels are closed and sealed." Limiting Condition for Operation 3.6.4.1 requires, the secondary containment shall be operable in Modes 1, 2 and 3. Contrary to the above, on April 7, 2016, the licensee failed to meet SR 3.6.4.1.1 and therefore, failed to meet LCO 3.6.4.1. Specifically, the licensee did not maintain the enclosure building hatch penetration in the closed position as required by SR 3.6.4.1.1, which resulted in secondary containment being inoperable. The licensee restored compliance by closing the hatch following the inspection, and put corrective actions in place to control the enclosure building hatch penetration in a closed position except for entry and exit for the inspection. Because the finding was of very low safety significance and entered into the licensee's corrective action program as Condition Report CR-GGN-1-2016-03707, this finding is being treated as a non-cited violation consistent with Section 2.3.2.a. of the NRC Enforcement Policy: NCV 05000416/2016002-01, "Failure to Maintain Secondary Containment Operable during Roof Inspections."

1R18 Plant Modifications (71111.18)

.1 Temporary Modifications

a. Inspection Scope

On April 7-8, 2016, the inspectors reviewed a temporary plant modification of the recirculation pump B flow control valve timing during a reactor feedpump trip runback. The inspectors verified that the licensee had installed this temporary modification in accordance with technically adequate design documents. The inspectors verified that this modification did not adversely impact the operability or availability of affected structures, systems, and components. The inspectors reviewed design documentation and plant procedures affected by the modification to verify the licensee maintained configuration control.

These activities constituted completion of one sample of temporary modifications, as defined in Inspection Procedure 71111.18.

b. Findings

No findings were identified.

.2 Permanent Modifications

a. Inspection Scope

On April 18-20, 2016, the inspectors reviewed the like-for-like replacement of spring return Victoreen radiation monitor control room switches with non-spring return switches. This was a permanent plant modification that affected risk-significant structures, systems, and components.

The inspectors reviewed the design and implementation of the modification. The inspectors verified that work activities involved in implementing the modification did not adversely impact operator actions that may be required in response to an emergency or other unplanned event. The inspectors verified that post-modification testing was adequate to establish the functionality of the structures, systems, and components as modified.

These activities constituted completion of one sample of permanent modifications, as defined in Inspection Procedure 71111.18.

b. Findings

No findings were identified.

1R19 Post-Maintenance Testing (71111.19)

a. Inspection Scope

The inspectors reviewed six post-maintenance testing activities that affected risk-significant structures, systems, or components:

- April 14, 2016, reactor protection system train B under-frequency relay removal
- April 22, 2016, replacement of susceptible hydraulic control unit cap screws
- May 2, 2016, division I diesel generator testing following jacket water maintenance
- May 20, 2016, reactor core isolation cooling system turbine overspeed testing following maintenance
- May 25, 2016, division I diesel generator testing following turbocharger leak and air start solenoid valve replacement
- June 1, 2016, division III diesel generator testing following a gasket repair on lube oil system and draining of the standby service water side of the jacket water system to replace a relief valve

The inspectors reviewed licensing- and design-basis documents for the structures, systems, and components and the maintenance and post-maintenance test procedures. The inspectors observed and reviewed the data associated with the performance of the post-maintenance tests to verify that the licensee performed the tests in accordance with approved procedures, satisfied the established acceptance criteria, and restored the operability of the affected structures, systems, or components.

These activities constituted completion of six post-maintenance testing inspection samples, as defined in Inspection Procedure 71111.19.

b. Findings

No findings were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors observed two risk-significant surveillance tests and reviewed test results to verify that these tests adequately demonstrated that the structures, systems, and components were capable of performing their safety functions:

Containment isolation valve surveillance test:

- June 10, 2016, containment purge valve, M41F034, local leak rate test

Other surveillance tests:

- May 21, 2016, reactor core isolation cooling system post-maintenance surveillance run

The inspectors verified that these tests met technical specification requirements, that the licensee performed the tests in accordance with their procedures, and that the results of the test satisfied appropriate acceptance criteria. The inspectors verified that the licensee restored the operability of the affected structures, systems, and components following testing.

These activities constituted completion of two surveillance testing inspection samples, as defined in Inspection Procedure 71111.22.

b. Findings

No findings were identified.

Cornerstone: Emergency Preparedness

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope

The inspectors observed an emergency preparedness drill on April 27, 2016, to verify the adequacy and capability of the licensee's assessment of drill performance. The inspectors reviewed the drill scenario, observed the drill from the simulator, technical support center, operations support center and the emergency offsite facility, and attended the post-drill critique. The inspectors verified that the licensee's emergency classifications, off-site notifications, and protective action recommendations were appropriate and timely. The inspectors verified that any emergency preparedness weaknesses were appropriately identified by the licensee in the post-drill critique and entered into the corrective action program for resolution.

These activities constituted completion of one emergency preparedness drill observation sample, as defined in Inspection Procedure 71114.06.

b. Findings

No findings were identified.

2. RADIATION SAFETY

Cornerstones: Public Radiation Safety and Occupational Radiation Safety

2RS1 Radiological Hazard Assessment and Exposure Controls (71124.01)

a. Inspection Scope

The inspectors evaluated the licensee's performance in assessing the radiological hazards in the workplace associated with licensed activities. The inspectors assessed the licensee's implementation of appropriate radiation monitoring and exposure control measures for both individual and collective exposures. The inspectors walked down various portions of the plant and performed independent radiation dose rate measurements. The inspectors interviewed the radiation protection manager, radiation protection supervisors, and radiation workers. The inspectors reviewed licensee performance in the following areas:

- Radiological hazard assessment, including a review of the plant's isotopic mix and isotopic percent abundance, hard-to-detect radionuclides and potential alpha hazards. The inspectors also reviewed the licensee's evaluations of changes in plant operations and radiological surveys to identify and detect dose rates, neutron hazards, hot particle exposures, severe dose gradients, airborne radioactivity monitoring, and surface contamination levels.

- Instructions to workers, including labeling or marking containers of radioactive material, radiation work permits, actions for electronic dosimeter alarms, and changes to radiological conditions.

- Contamination and radioactive material control including release of potentially contaminated material from the radiologically controlled area, radiological survey performance, radiation instrument sensitivities, material control and release criteria, procedural guidance, and control and accountability of sealed radioactive sources.

- Radiological hazards control and work coverage including field observations of job performance and adequacy of radiological controls. During walkdowns of the facility and job performance observations, the inspectors evaluated ambient radiological conditions, radiological postings, adequacy of radiological controls, radiation protection job coverage, and contamination controls. The inspectors also evaluated the use of electronic dosimeters in high noise areas, dosimetry selection and placement, implementation of effective dose equivalent for external exposures (EDEX), and the application of dosimetry to effectively monitor exposure for work in areas with significant dose rate gradients. The inspectors examined the licensee's controls for highly activated or contaminated materials (non-fuel) stored within spent fuel and other storage pools and evaluated airborne radioactive controls and monitoring.

- High radiation area and very high radiation area controls including posting and physical controls for high radiation areas and very high radiation areas. During plant walkdowns, the inspectors verified the adequacy of posting and physical controls, including for areas of the plan with the potential to become risk-significant high radiation areas.

- Radiation worker performance and radiation protection technician proficiency with respect to radiation protection work requirements. The inspectors determined if workers were aware of the significant radiological conditions in their workplace, radiation work permit controls/limits in place, and were aware of their electronic alarming dosimetry dose and dose rate set points. The inspectors observed radiation protection technician job performance, including the performance of radiation surveys.

- Problem identification and resolution for radiological hazard assessment and exposure controls. The inspectors reviewed audits, self-assessments, and corrective action program documents to verify problems were being identified and properly addressed for resolution.

These activities constituted completion of the seven required samples of radiological hazard assessment and exposure control program, as defined in Inspection Procedure 71124.01.

b. Findings

No findings were identified.

2RS2 Occupational ALARA Planning and Controls (71124.02)

a. Inspection Scope

The inspectors assessed licensee performance with respect to maintaining individual and collective radiation exposures as low as is reasonably achievable (ALARA). The inspectors performed this portion of the attachment during the refueling outage, in order to directly observe the licensee’s ALARA process activities including planning, implementation of radiological work controls, execution of work activities, and ALARA review of work-in-progress. During the inspection the inspectors interviewed licensee personnel, reviewed licensee documents, and evaluated licensee performance in the following areas:

- Implementation of ALARA and radiological work controls. The inspectors observed pre-job briefings, reviewed planned radiological administrative, operational, and engineering controls, and compared the planned controls to field activities.

- Radiation worker and radiation protection technician performance during work activities performed in radiation areas, airborne radioactivity areas, or high radiation areas.

- Problem identification and resolution for ALARA and radiological work controls. The inspectors reviewed audits, self-assessments, and corrective action program documents to verify problems were being identified and properly addressed for resolution.

These activities constituted completion of two of the five required samples of occupational ALARA planning and controls program, as defined in Inspection Procedure 71124.02.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

Cornerstones: Mitigating Systems, Public Radiation Safety, and Occupational Radiation Safety

4OA1 Performance Indicator Verification (71151)

.1 Safety System Functional Failures (MS05)

a. Inspection Scope

For the period of April 1, 2015, through March 31, 2016, the inspectors reviewed licensee event reports, maintenance rule evaluations, and other records that could indicate whether safety system functional failures had occurred. The inspectors used definitions and guidance contained in Nuclear Energy Institute Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 7, and NUREG-1022, "Event Reporting Guidelines: 10 CFR 50.72 and 50.73," Revision 3, to determine the accuracy of the data reported.

These activities constituted verification of the safety system functional failures performance indicator, as defined in Inspection Procedure 71151.

b. Findings

No findings were identified.

.2 Mitigating Systems Performance Index: Emergency AC Power Systems (MS06) and Cooling Water Support Systems (MS10)

a. Inspection Scope

The inspectors reviewed the licensee's mitigating system performance index data for the period of April 1, 2015, through March 31, 2016, to verify the accuracy and completeness of the reported data. The inspectors used definitions and guidance contained in Nuclear Energy Institute Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 7, to determine the accuracy of the reported data.

These activities constituted verification of the mitigating system performance index for emergency ac power systems and cooling water support systems, as defined in Inspection Procedure 71151.

b. Findings

No findings were identified.

.3 Occupational Exposure Control Effectiveness (OR01)

a. Inspection Scope

The inspectors verified that there were no unplanned exposures or losses of radiological control over locked high radiation areas and very high radiation areas during the period of October 1, 2015, to December 31, 2015. The inspectors reviewed a sample of radiologically controlled area exit transactions showing exposures greater than 100 mrem. The inspectors used definitions and guidance contained in Nuclear Energy Institute Document 99-02, “Regulatory Assessment Performance Indicator Guideline,” Revision 7, to determine the accuracy of the reported data.

These activities constituted verification of the occupational exposure control effectiveness performance indicator, as defined in Inspection Procedure 71151.

b. Findings

No findings were identified.

.4 Radiological Effluent Technical Specifications (RETS)/Offsite Dose Calculation Manual (ODCM) Radiological Effluent Occurrences (PR01)

a. Inspection Scope

The inspectors reviewed corrective action program records for liquid or gaseous effluent releases that occurred between October 1, 2015, and December 31, 2015, and were reported to the NRC to verify the performance indicator data. The inspectors used definitions and guidance contained in Nuclear Energy Institute Document 99-02, “Regulatory Assessment Performance Indicator Guideline,” Revision 7, to determine the accuracy of the reported data.

These activities constituted verification of the radiological effluent technical specifications (RETS)/offsite dose calculation manual (ODCM) radiological effluent occurrences performance indicator, as defined in Inspection Procedure 71151.

b. Findings

No findings were identified.

4OA2 Problem Identification and Resolution (71152)

.1 Routine Review

a. Inspection Scope

Throughout the inspection period, the inspectors performed daily reviews of items entered into the licensee’s corrective action program and periodically attended the licensee's condition report screening meetings. The inspectors verified that licensee personnel were identifying problems at an appropriate threshold and entering these problems into the corrective action program for resolution. The inspectors verified that the licensee developed and implemented corrective actions commensurate with the significance of the problems identified. The inspectors also reviewed the licensee's problem identification and resolution activities during the performance of the other inspection activities documented in this report.

b. Findings

No findings were identified.

.2 Semiannual Trend Review

a. Inspection Scope

The inspectors reviewed the licensee's corrective action program, operability determinations, and other documentation to identify trends that might indicate the existence of a more significant safety issue. The inspectors verified that the licensee was taking corrective actions to address identified adverse trends.

These activities constituted completion of one semiannual trend review sample, as defined in Inspection Procedure 71152.

b. Observations and Assessments

The inspectors reviewed Procedure EN-OP-104, "Operability Determinations," Revision 10, and condition reports for the past year. The focus was on immediate operability determinations. During the review, the inspectors identified a weakness in performing accurate immediate operability determinations. The inspectors determined that this was due to the lack of information in the condition report description such that operations personnel could not make an accurate operability determination and the failure to correctly implement the operability determination procedure.

In 2015, the component design basis inspection team and the problem identification and resolution inspection team challenged Grand Gulf Nuclear Station on the quality of operability determinations. Based on the trend review that was performed during this inspection activity, the inspectors identified more examples that were consistent with the observations made by those team inspections. The inspectors identified three examples below:

- Condition Report CR-GGN-1-2016-00758 identified that the standby liquid control system piping was non-functional in accordance with the technical requirements manual due to non-conservative pressure used in the ASME piping pressure test. Operations personnel declared the standby liquid control system operable. The inspectors identified in Procedure EN-OP-104 that this condition placed the standby liquid control system in an operable but degraded/non-conforming condition. The licensee agreed and declared the system operable but degraded/non-conforming until the ASME piping pressure test was completed with the appropriate pressure. The licensee documented this issue in their corrective action program as Condition Report CR-GGN-1-2016-03108.

- Condition Report CR-GGN-1-2016-03495 identified two concerns, one pertaining to potential degradation of diesel generator exhaust piping and one for the position of a secondary containment hatch. Due to the lack of description, operations personnel performed incorrect immediate operability assessments. The licensee documented this issue in their corrective action program as Condition Report CR-GGN-1-2016-03527.

    - Versions 1 and 2 of the operability determination declared the diesel generator system operable. Following inspector's questions, Version 3 of the operability determination was performed and determined that the diesel generator exhaust piping was considered operable but needed an engineering evaluation. Ultimately, Version 4 determined the diesel generators to be operable.

    - Versions 1 and 2 of the operability determination declared the secondary containment operable. Following inspector's questions, Version 3 of the operability determination was performed and determined the secondary containment was considered inoperable. (This is discussed in Section 1R15 of this report).

- Condition Report CR-GGN-1-2016-03376 identified a condition where hydraulic control unit cap screws were corroded. Operations personnel declared the hydraulic control units operable. The inspectors identified that corrosion is listed in the definition of a degraded condition in accordance with Procedure EN-OP-104. The licensee agreed and declared the system operable but degraded/non-conforming until the ASME piping pressure test was complete with the appropriate pressure. The licensee documented this issue in their corrective action program as Condition Report CR-GGN-1-2016-03543.

These three examples are violations of 10 CFR Criterion V, "Instructions, Procedures and Drawings." Using Inspection Manual Chapter 0612, Appendix E, Example 4.f, the inspectors determined these violations to be of minor significance. The licensee performed new operability determinations for all of the examples and ultimately concluded the correct outcome. The failures to comply with 10 CFR Criterion V, "Instructions, Procedures, and Drawings," constituted minor violations that are not subject to enforcement action in accordance with the NRC's Enforcement Policy.

Due to the concerns identified by the inspectors, the licensee implemented several corrective actions that:

- Incorporated discussions during their leadership and alignment meetings explaining the importance of writing timely and accurate condition reports, such that the operations staff can perform accurate operability determinations and presented this message to workers during daily shift briefs;

- Reinforced the message by displaying the slogan, "Don't Delay, Write it Today," on the signage prior to entering the site owner controlled area;

- Incorporated a 3-hour operability training during licensed operator requalification training during last cycle; and

- Implemented a peer check on immediate operability determinations by a licensed senior reactor operator prior to final shift manager approval.

c. Findings

No findings were identified.

.3 Annual Follow-up of Selected Issues

a. Inspection Scope

On March 17, 2016, Grand Gulf Nuclear Station experienced a loss of shutdown cooling while in Refueling Outage 20. The cause of the event was an electrical fault on the Baxter Wilson 115 kV transmission line. This resulted in an undervoltage condition in the Grand Gulf switchyard, which actuated the division II load shedding sequencer. The residual heat removal train B was in service providing shutdown cooling and was shed during the transient. The division II diesel generator started and provided the power necessary for the operators to restore shutdown cooling within approximately three minutes. Spent fuel pool and reactor cavity water temperature remained constant during this event. The apparent cause was determined to be a lack of a pilot scheme protection. This protection would have utilized protective relaying to clear the phase-to-phase fault sooner, thus shortening the duration of undervoltage condition that was observed.

On May 9, 2016, the inspectors completed their review of the apparent cause evaluation documented in Condition Report CR-GGN-1-2016-02513. The inspectors determined that the apparent cause evaluation was done adequately and addressed the need and planning for offsite power supply (i.e. Port Gibson switchyard) protection scheme modifications.

The inspectors assessed the licensee's problem identification threshold, cause analyses, extent of condition reviews and compensatory actions. The inspectors verified that the licensee appropriately prioritized the planned corrective actions and that these actions were to ensure that a pilot scheme protection upgrade is planned and utilized to prevent degraded offsite voltage of offsite power sources.

These activities constituted completion of one annual follow-up sample, as defined in Inspection Procedure 71152.

b. Findings

No findings were identified.

4OA3 Follow-up of Events and Notices of Enforcement Discretion (71153)

1. (Closed) Licensee Event Report (LER) 2015-002-00, Loss of Secondary Containment Differential Pressure during Drawdown Testing

On August 1, 2015, the licensee tested secondary containment using Attachment II of Procedure 06-OP-1T48-R-0002, "Standby Gas Treatment System A," Revision 115, as required per Surveillance Requirements 3.6.4.1.3 and 3.6.4.1.4. Per Surveillance

Requirements 3.6.4.1.3 and 3.6.4.1.4, secondary containment must reach at least 0.311 inch of vacuum water gage within 180 seconds and maintain the vacuum of 0.311 inch for one hour.

During the test, the secondary containment system was not able to maintain the 0.311 inch of water gage during the entire test. Although the data package for the completed surveillance procedure concluded the technical specification acceptance criteria was unacceptable, the licensee failed to declare secondary containment inoperable, as required per Surveillance Requirement 3.0.1, and immediately take actions as required in Technical Specification 3.6.4.1. The licensee was able to restore secondary containment to operable status within the 4-hour action time, and the total amount of secondary containment inoperability time was 1.5 hours.

An NRC identified, Green, non-cited violation was documented in Inspection Report 05000416/2015008 in Section 4OA2.5. This LER is closed.

2. <u>(Closed) Licensee Event Report 2015-003-00, Technical Specification Surveillance on Primary Containment Isolation Valves</u>

On October 14, 2015, at 12:20 p.m., the licensee identified that there were two primary containment isolation valves, 1P11F130 and 1P11F131, in the same flow path that were not local leak rate tested using the post-extended power uprated peak containment pressure. The licensee declared both valves inoperable and closed a valve in the flow path to restore leakage to within limits in the 4-hour action time as required by Technical Specification 3.6.1.3, Condition C. Subsequent leak rate testing of the valves demonstrated that the valves were both operable.

This event was previously reviewed and a licensee-identified, Green, non-cited violation and an NRC-identified, Severity Level IV, non-cited violation were documented in Inspection Report 05000416/2015004 under Sections 1R15 and 4OA7. This LER is closed.

3. <u>(Closed) Licensee Event Report 2016-001-00, Valid Engineered Safety Feature Actuation and Temporary Loss of Shutdown Cooling</u>

This LER was reviewed in this inspection report as an annual follow-up sample in Section 4OA2. This event was also discussed in Inspection Report 05000416/2016001 in Section 4OA3. No findings or violations of NRC requirements were identified. This LER is closed.

4. <u>(Closed) Licensee Event Report 2016-002-00, Automatic Actuation of the Reactor Protection System due to ‘B’ Main Transformer Wiring</u>

a. <u>Inspection Scope</u>

On March 29, 2016, Grand Gulf Nuclear Station was in Mode 1 and raising power to approximately 37 percent, when an automatic reactor scram occurred. The ‘B’ phase main transformer differential relay tripped and caused a generator lock out. The turbine control valves fast closed which caused a turbine trip and subsequent reactor scram. There were no emergency core cooling systems that actuated and no engineered safety features equipment that actuated.

The inspectors independently reviewed data logs, observed procedure usage, and observed control room indications to confirm the initiating cause of the trip. The preliminary cause was identified to be an incorrect wiring ratio for the ‘B’ phase main transformer current transformer. The licensee corrected the condition by resetting the current transformer wiring ratio, and performed post-maintenance testing to confirm the correct ratio. The licensee also inspected the wiring on the ‘A’ and ‘C’ phases of the transformer to confirm there were no extent of condition concerns. The licensee entered this event into their corrective action program as Condition Report CR-GGN-1-2016-02950. This LER is closed.

b. Findings

Introduction. The inspectors reviewed a Green, self-revealed finding of Procedure EN-WM-105, “Planning," Revision 16, for the failure to ensure Work Order (WO) 397549 provided detailed instructions for performing troubleshooting on the ‘B’ phase of the main transformer.

Description. On March 29, 2016, Grand Gulf Nuclear Station was in Mode 1 and raising power to approximately 37 percent and an automatic reactor scram occurred. The ‘B’ phase main transformer differential relay tripped and caused a generator lock out. The turbine control valves fast closed which caused a turbine trip and subsequent reactor scram. There were no emergency core cooling systems that actuated and no engineered safety features equipment that actuated.

During Refueling Outage 20, the licensee was removing a temporary wiring modification in the ‘A’, ‘B’, and ‘C’ phases of the main transformer. Work Order 397549 was used to perform the rewiring of the transformer. Following the work, the contractor performed a bump test to ensure the correct wiring had been performed, and the results were not satisfactory. The contractor initiated a condition report and performed troubleshooting activities. There was no contractor oversight by Entergy during the troubleshooting activities. Additionally, there was no documented evidence of lifted leads or documented troubleshooting.

The licensee determined that the cause of the 'B' phase main transformer differential relay trip was due to an incorrect wiring ratio of the current transformer. The high voltage current transformer wiring was incorrectly landed at the X1/X2 terminals instead of the X1/X3 terminals. This wiring configuration resulted in a current transformer turns ratio of 1000:5 instead of the designed 2200:5. This caused the relay to actuate at a lower setpoint than designed. The licensee corrected the condition by resetting the current transformer wiring ratio and performed post-maintenance testing to confirm the correct ratio. The licensee also inspected the wiring on the ‘A’ and ‘C’ phases of the transformer to confirm there are no extent of condition concerns. The licensee entered this event into their corrective action program as Condition Report CR-GGN-1-2016-02950.

Analysis. The failure to ensure WO 397549 provided detailed instructions for performing troubleshooting on the ‘B’ phase of the main transformer in accordance with Procedure EN-WM-105 was a performance deficiency. This performance deficiency was more than minor, and therefore a finding, because it is associated with the procedure quality attribute of the Initiating Events Cornerstone and adversely affected the cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, on March 29, 2016, the licensee failed to ensure WO 397549 provided detailed instructions for performing troubleshooting activities on the 'B' phase of the main transformer, which resulted in an incorrect current transformer wiring ratio and subsequent reactor scram. Using Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," and Inspection Manual Chapter 0609, Appendix A, Exhibit 1, "Initiating Events Screening Questions," the inspectors determined that the finding was of very low safety significance (Green) because the finding did result in a reactor trip, but did not result in the loss of mitigating equipment relied upon to transition the plant from the onset of the trip to a stable shutdown condition.

This finding has a cross-cutting aspect in the area of human performance associated with field presence, in that, senior managers failed to ensure supervisory and management oversight of work activities, including contractors and supplemental personnel. Specifically, while performing WO 397549, the licensee did not have contractor oversight established and the contract workers performed troubleshooting without detailed instructions to ensure work was performed properly. [H.2]

Enforcement. Procedure EN-WM-105, "Planning," Revision 16, Step 5.5[1], requires, in part, "For all work: the work package must provide enough information for successful completion of the work. Successful completion of the work is defined as follows: Following work, equipment will work as designed, with no deficiencies." Contrary to the above, on March 29, 2016, the licensee failed to have a work package with enough information for successful completion of the work, such that equipment will work as designed, with no deficiencies. Specifically, WO 397549 did not contain detailed instructions for performing troubleshooting on the 'B' phase of the main transformer, resulting in an incorrect current transformer ratio and subsequent reactor scram. The licensee's corrective actions were to incorporate more detailed instructions to the work order, repair the improper wiring, and restore the main transformer prior to transitioning from Mode 3 to Mode 1. Inspectors did not identify a violation of regulatory requirements associated with this finding. This finding has been entered into the licensee's corrective action program as Condition Report CR-GGN-1-2016-02950.
FIN 05000416/2016002-02, "Failure to Provide Detailed Work Instructions Resulted in a Reactor Scram."

5. (Closed) Licensee Event Report 2016-003-00, Loss of Secondary Containment Safety Function during Routine Roof Inspection

This LER was reviewed in this inspection report as an operability determination sample. The event description and an NRC-identified, Green, non-cited violation are documented in Section 1R15 of this report. This LER is closed.

6. Reactor Scram Following Valid Reactor Protection System Actuation Caused by Oscillating Power Range Monitors

a. Inspection Scope

On June 17, 2016, the inspectors responded to the Grand Gulf Nuclear Station control room to observe recovery actions following the reactor scram. At approximately 2:57 a.m., the reactor was at 60 percent power when a valid reactor protection system actuation resulted in a reactor scram. During turbine stop valve testing, only the B stop valve was cycled closed; however, the B and D stop valves closed, which caused a ½ scram condition on division II of the reactor protection system.

Stop valve trip fluid pressure fluctuated approximately 8 psi. This caused the turbine control valves to move, resulting in reactor pressure and power swings. Due to the fluctuating power and pressure, the operators inserted control rods to try and overcome the fluctuations. They were not able to stabilize the reactor, and an automatic scram occurred. There were no emergency core cooling systems that actuated and no engineered safety features equipment that actuated.

The inspectors independently reviewed data logs, toured plant areas and observed control room indications to confirm the initiating cause of the scram, and the appropriate plant response to achieve safe shutdown conditions. The licensee entered this event into their corrective action program as Condition Report CR-GGN-1-2016-04766.

b. Findings

No findings were identified.

7. Reactor Scram Following Valid Reactor Protection System Actuation Caused by Fast Closure of the Turbine Control Valves

a. Inspection Scope

On June 25, 2016, the inspectors responded to the Grand Gulf Nuclear Station control room to observe recovery actions following the reactor scram. At approximately 2:07 p.m., the reactor was at 100 percent power when a valid reactor protection system actuation resulted in a reactor scram. The electrohydraulic control system provided an electronic signal for the fast closure of the turbine control valves. The control valves fast closed which caused the reactor scram. The reactor pressure increased and two safety relief valves opened to restore reactor pressure. There were no emergency core cooling systems that actuated and no engineered safety features equipment that activated.

The inspectors independently reviewed data logs, observed procedure usage, and observed control room indications to confirm the initiating cause of the trip. The preliminary cause was identified to be a control system malfunction in the electrohydraulic control system. The licensee corrected the condition by replacing the two logic cards in the electrohydraulic control system caused the electronic failure. The licensee entered this event into their corrective action program as Condition Report CR-GGN-1-2016-04998.

b. Findings

No findings were identified.

These activities constituted completion of seven event follow-up samples, as defined in Inspection Procedure 71153.

4OA5 Other Activities

.1 <u>(Closed) Unresolved Item (URI) 05000416/2015007-03, Lack of Coordination of Division III HPCS Switchgear 127N Undervoltage Relays</u>

On November 13, 2015, the NRC issued Unresolved Item URI 05000416/2015007-03, "Lack of Coordination of Division III HPCS Switchgear 127N Undervoltage Relays." The inspection team identified multiple potential issues of concern with the coordination of the division III 4.16 kV switchgear bus 127N feeder instantaneous undervoltage relay settings with the coordination of high voltage system protective relays, switchgear overcurrent relays, and loss of voltage relays to allow time for the other relays to perform their required design functions.

From May 9 to May 19, 2016, an NRC inspector reviewed additional design and licensing information to determine whether these issues resulted in a more than minor performance deficiency or a violation of NRC requirements. The inspector reviewed the final safety analysis report, safety evaluation reports, license amendments, electrical distribution drawings, electrical relay coordination studies, design specifications, and relay characteristic curves for the division III high pressure core spray 4.16 kV safety switchgear bus 17AC.

No more than minor findings or violations were identified associated with the issues of concern. The inspector identified two minor violations of NRC requirements.

The inspector identified a performance deficiency against 10 CFR Part 50, Appendix B, Criterion III, "Design Control," which requires that measures are established to assure applicable regulatory requirements and the design basis, are correctly translated into specifications, drawings, procedures, and instructions. Contrary to this requirement, prior to May 18, 2016, the licensee failed to assure applicable regulatory requirements and the design basis were correctly translated into drawings. Specifically, the licensee failed to assure the designed operation of division III 4.16 kV bus undervoltage 127S relays were correctly translated into the relay table of drawing E-1009, "One Line Meter and Relay Diagram, 4.16 kV E.S.F System, Bus 17AC Unit," Revision 9. The relay table states that the 127S relays "TRIP INCOMING BREAKER TO BUS & START DIESEL;" however, these relays start the emergency diesel generator and give permissive to close the generator output breaker, but do not trip the incoming feeder breaker. This violation is minor because it could not be reasonably viewed as a precursor to a significant event, would not lead to a more significant safety concern, did not cause a performance indicator to exceed a threshold, and did not adversely affect any of the cornerstone objectives. The licensee entered this minor violation into the corrective action program as Condition Report CR-GGN-1-2016-04150 to correct the minor drawing discrepancy. This failure to comply with 10 CFR Part 50, Appendix B, Criterion III, "Design Control," constituted a minor violation that is not subject to enforcement action in accordance with the NRC's Enforcement Policy.

The inspector identified a performance deficiency against 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," which requires that measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances, are promptly identified and corrected. Contrary to this requirement, from November 1, 2015, to May 18, 2016, the licensee failed to assure that a condition adverse to quality was promptly identified and corrected. Specifically, the licensee failed to correct a minor drawing deviation (minor violation above) associated with the relay table in Drawing E-1009, "One Line Meter and Relay Diagram, 4.16 kV E.S.F System, Bus 17AC Unit," Revision 9, when an NRC inspector identified and communicated the deviation to the licensee on November 1, 2015. This violation is minor because it could not be reasonably viewed as a precursor to a significant event, would not lead to a more significant safety concern, did not cause a performance indicator to exceed a threshold, and did not adversely affect any of the cornerstone objectives. The licensee entered this minor violation into the corrective action program as Condition Report CR-GGN-1-2016-04162 to correct the failure to correct a condition adverse to quality. This failure to comply with 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," constituted a minor violation that is not subject to enforcement action in accordance with the NRC's Enforcement Policy.

Based on the following discussion and previously discussed minor violations, unresolved item URI 05000416/2015007-03 is closed.

The potential issues of concern and resolution were the following:

1. Issue: The protective relays associated the main generator transformer and its output circuit lack coordination with the division III 4.16 kV switchgear 127N feeder undervoltage relays which could potentially result in a coincident loss of two alternating power supplies, contrary to the requirements of General Design Criterion (GDC) 17. This is because the instantaneous response of the 127N undervoltage relay for the division III 4.16 kV switchgear bus 17AC could respond instantaneously to the momentary voltage dip caused by a fault on the main generator transformer and also cause a loss of the offsite power supply to division III 4.16 kV switchgear AC.

   Resolution: The inspector reviewed the transformer differential relay settings associated with main generator transformer No. 1 to the 500 kV transmission system network, in addition to the transformer differential relay settings for transformers associated with the 500 kV/34.5 kV transmission system network. The inspector reviewed their instantaneous operation to isolate a main transformer fault from the 500 kV grid without causing a loss of power to the transmission network feeding the division III 4.16 kV switchgear 17AC. The inspector compared these transformer differential relay time characteristic curves and settings with those of the 127N division III feeder undervoltage relays. Because of the instantaneous operation of the transformer differential relays, the fault would be isolated at the main transformer and would not communicate to the parts of the distribution system supplying offsite power to the division III 4.16kV switchgear bus.

   The inspector did not identify any more than minor performance deficiencies associated with the coordination of protective relay settings between the main generator transformer and the division III 4.16 kV switchgear bus 17 AC relays.

2. Issue: The protective relays associated with the transmission system bus lack coordination with the division III 4.16 kV switchgear 127N feeder undervoltage relays which could potentially react to the momentary voltage dip caused by a transmission system bus fault, resulting also in instantaneous loss of the offsite power supply to division III.

Resolution: The design of the Grand Gulf Nuclear Station electrical distribution system is such that the division III high pressure core spray 4.16 kV switchgear bus 17AC can be fed by any one of three offsite power transmission networks at a time. Additionally by design, switchgear bus 17AC is not equipped with an automatic transfer to the other offsite power sources if a fault or loss of power occurs to the offsite power source to which the switchgear bus is initially connected. Therefore by design, based on the instantaneous overcurrent and differential relay settings associated with the offsite power transmission networks, offsite power will be lost to switchgear bus 17AC if a fault causes power loss to the offsite power network to which it is connected. This is necessary to start the sequence for starting the high pressure core spray diesel in enough time to meet emergency core cooling accident analyses.

The inspector did not identify any more than minor performance deficiencies associated with the settings of the protective relays and their instantaneous operation to isolate offsite power to the division III 4.16 kV switchgear bus 17AC caused by a transmission system bus fault.

3. Issue: The instantaneous setting 127N feeder undervoltage relays lack coordination with technical specification required division III 4.16 kV switchgear bus undervoltage relays 27S1, -S2, -S3, and -S4 and second-level degraded voltage relays 127-1A, -1B, -2A, and -2B. These second-level relays would not disconnect the offsite power supply to switchgear bus 17AC upon actuation in accordance with NRC Branch Technical Position PSB-1, "Adequacy of Station Electric Distribution System Voltages."

   Resolution: The inspector reviewed one line meter and relay diagrams, the safety evaluation report, final safety analysis report, license amendments, General Electric design specifications, and coordination studies associated with the primary and secondary level undervoltage protection associated with division III 4.16 kV switchgear bus 17AC. No coordination issues were identified with the second-level degraded voltage relays 127-1A, -1B, -2A, and -2B.

   As described in General Electric Licensing Topical Report NEDO-10905, "High Pressure Core Spray System Power Supply," the design and coordination of the first-level undervoltage relays are for the 127N relay to sense feeder undervoltage at 73 percent grid voltage, then disconnect the feeder supply breaker to the division III 4.16 kV switchgear bus 17AC. Concurrently, the 127S relays sense undervoltage on the bus for 2.3 seconds then actuate to start the high pressure core spray emergency diesel generator and give the permissive to close the emergency diesel generator output breaker. The 127S relays do not disconnect offsite power to the switchgear bus 17AC. Since the division III 4.16 kV switchgear bus is not supplied by a load sequencer, the 127N and 127S relays work in tandem to ensure that the high pressure core spray motor will not be damaged when operating below the 73 percent undervoltage condition, while still meeting the emergency core cooling requirements for full high pressure core spray system flow. In addition, the coordination of the 127N and 127S undervoltage relays ensure a reduction in residual voltage of the high pressure core spray motor so that the diesel generator would reliably restart and reaccelerate the high pressure core spray motor following a worst case loss of coolant accident followed by a loss of offsite power.

Grand Gulf Nuclear Station's design of the relay coordination is installed in the configuration as described in NEDO-10905. The instantaneous operation of the 127N was selected based on the high pressure core spray equipment protection and emergency core cooling system safety analysis, which is in agreement with NEDO-10905. The NRC approved the concept and design of the high pressure core spray system power, as documented in NEDO-10905, in a letter to the General Electric Company on December 17, 1976.

The inspector did not identify any more than minor performance deficiencies associated with the electrical coordination between the feeder undervoltage relays and the bus undervoltage relays. Two minor violations associated with the relay table within Drawing E-1009 are documented above.

4. Issue: The instantaneous setting of the 127N feeder undervoltage relays prevents the division III switchgear bus 151B and 151G time overcurrent relays from preserving function and limiting loss of Class 1E equipment function in the event of a switchgear bus 17AC fault. Specifically, the instantaneous setting of the 127N feeder undervoltage relay negates the trip and lockout function of the 151B and 151G feeder overcurrent relays, potentially resulting in closure of the emergency diesel generator output breaker to a faulted bus and damage to the emergency diesel generator.

   Resolution: The inspector reviewed one line and meter diagrams, fault current studies, relay manufacturer documents, and the time characteristic curves of each relay. In addition, the inspector reviewed the manufacturer documents and relay settings associated with the 187D emergency diesel generator differential voltage relay. Switchgear 17AC offsite power feeder circuit breakers are equipped with 151B overcurrent relays that, when actuated, trip and lockout the switchgear feeder supply breakers. The purpose of the lockout function is to prevent attempted re-energization of a faulted switchgear bus to protect equipment tied to the bus. To protect the emergency diesel generator, the emergency diesel generator is equipped with 187D generator differential voltage relay that, when actuated, trips and locks-out the diesel generator output breaker, which subsequently shuts down the emergency diesel generator. The 187D relay is not bypassed during accident conditions.

   The inspector determined that the possibility of the emergency diesel generator loading to a faulted bus because of the instantaneous actuation of the 127N feeder undervoltage relay is plausible. However, the emergency diesel generator is ultimately protected by the instantaneous setting of the 187D generator differential voltage relay. Based on the settings associated with this relay, the emergency diesel generator output breaker would trip and emergency diesel generator shutdown sequence would start.

   The inspector did not identify any more than minor performance deficiencies associated with the electrical coordination between the feeder undervoltage relays and the bus time overcurrent relays.

5. Issue: The licensee failed to perform a coordination study to ensure that coordination between the 150/151M and 150/151T overcurrent relays and the

127N undervoltage relays would isolate faults or overload conditions at the affected equipment downstream of division III 4.16 kV switchgear bus 17AC.

Resolution: The inspector reviewed one line and meter diagrams, fault current studies, relay manufacturer documents, and the time characteristic curves of each relay. The circuit breakers for equipment downstream of division III 4.16 kV switchgear bus 17AC are equipped with 150/151M and 150/151T overcurrent relays that are designed to isolate downstream faults locally. These relays have three unique breaker trip settings. These settings are for time overcurrent trips, instantaneous normal dropout current trips, and instantaneous high dropout instantaneous trips. Based on the calculated fault current levels associated with downstream equipment and its circuit, the normal dropout instantaneous trip of these relays would actuate to trip the associated circuit breaker and isolate a fault at the lowest possible level. Since the overcurrent trip would be instantaneous, the coordination between these relays and 127N relays is maintained.

The inspector did not identify any more than minor performance deficiencies associated with the electrical coordination between the feeder undervoltage relays, 127N, and the equipment overcurrent relays 150/151M and 150/151T.

4OA6 Meetings, Including Exit

Exit Meeting Summary

On April 4, 2016, the inspectors presented the radiation safety inspection results to Mr. K. Mulligan, Site Vice President, and other members of the licensee staff. The licensee acknowledged the issues presented. The licensee confirmed that any proprietary information reviewed by the inspectors had been returned or destroyed.

On June 28, 2016, the inspectors presented the inspection results to Mr. G. Hawkins, Acting General Manager Plant Operations, and other members of the licensee staff. The licensee acknowledged the issues presented. The licensee confirmed that any proprietary information reviewed by the inspectors had been returned or destroyed.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

R. Benson, Superintendent, Radiation Protection
A. Burks, Supervisor, Radiation Protection
B. Busick, Manager, Operations
T. Coles, Engineer 1, Regulatory Assurance
G. Hawkins, (Acting) General Manager Plant Operations
D. James, Senior Technician, Radiation Protection
M. Lanni, Supervisor, Radiation Protection
R. Miller, Manager, Radiation Protection
T. Moncure, Supervisor, Radiation Protection
J. Nadeau, Manager, Regulatory Assurance
P. Stokes, Supervisor, Radiation Protection
S. Sweet, Regulatory Assurance
R. Meister, Regulatory Assurance

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000416/2016002-01	NCV	Failure to Maintain Secondary Containment Operable during Roof Inspections
05000416/2016002-02	FIN	Failure to Provide Detailed Work Instructions Resulted in a Reactor Scram

Closed

05000416/2015007-03	URI	Lack of Coordination of Division III HPCS Switchgear 127N Undervoltage Relays
2015-002-00	LER	Loss of Secondary Containment Differential Pressure during Drawdown Testing
2015-003-00	LER	Technical Specification Surveillance on Primary Containment Isolation Valves
2016-001-00	LER	Valid Engineered Safety Feature Actuation and Temporary Loss of Shutdown Cooling
2016-002-00	LER	Automatic Actuation of the Reactor Protection System due to ‘B’ Main Transformer Wiring
2016-003-00	LER	Loss of Secondary Containment Safety Function during Routine Roof Inspection

LIST OF DOCUMENTS REVIEWED

Section 1R01: Adverse Weather Protection

Procedures

Number	Title	Revision
EN-DC-150	Condition Monitoring of Maintenance Rule Structures	9
EN-TQ-212	Conduct of Training and Qualification	14
EN-PL-159	Summer Reliability Plan	0
ENS-PL-158	Switchyard and Transmission Interface Requirements	3
ENS-DC-201	ENS Transmission Grid Monitoring	6
02-S-1-01-42	Switchyard Control	3
04-S-01-R27-1	500/115 KV System	32
05-1-02-I-4	Loss of AC Power Systems	50

Condition Reports (CR-GGN-1-)

2015-00225	2015-00437	2016-00370	2016-01850	2016-03292
2016-03520	2016-03522	2016-04947		

Work Order

52671695

Miscellaneous

Number	Title	Date
	Exemption of Qualifications for EN-DC-150, Condition Monitoring of Maintenance Rule Structure	April 14, 2016

Section 1R04: Equipment Alignment

Procedures

Number	Title	Revision
04-1-01-E22-1	System Operating Instruction for High Pressure Core Spray System	121
04-1-01-P81-1	System Operating Instruction for High Pressure Core Spray Diesel Generator	76

Miscellaneous

Number	Title	Date
	Operator Risk Profile of RCIC and HPCS unavailability	May 17, 2016
	Operator Risk Profile of RCIC and HPCS Diesel Generator unavailability	May 17, 2016
	Sections 6.3 and 8.3.1 of Updated Final Safety Analysis Report	
	Sections 3.5.1 and 3.8.1 of Grand Gulf Technical Specifications, Bases Document, and Surveillance Requirements	

Section 1R05: Fire Protection

Procedure

Number	Title	Revision
10-S-03-8	Fire Watch Program	13

Drawings

Number	Title	Revision
A-0890A	Wall Penetration Details Control Bldg.	0
M-0859	Blockouts and Penetrations Control Building EL. 189'-0"	19
M-0850-037	Control Building EL. 189'-0" Units 1 & 2	14

Miscellaneous

Number	Title	Revision
	Sections 9A.5.2.3 9A.5.60, 9A.5.61, 9A.5.62, and 9A.5.63 of Updated Final Safety Analysis Report	
	Fire Pre-Plan for zones 1A104, 1D301, 1D302, 1D303 and 1D304	
	Grand Gulf Fire Pre-Plans C-11-1	3
	Grand Gulf Fire Pre-Plans C-11-2	3
	Grand Gulf Fire Pre-Plans C-18	2
	Grand Gulf Fire Pre-Plans FWPH-01	1

Section 1R11: Licensed Operator Requalification Program

Procedures

Number	Title	Revision
02-S-01-27	Operation’s Philosophy	64
03-1-01-1	Cold Shutdown to Generator Carrying Minimum Load	169
03-1-01-3	Plant Shutdown	129
03-1-01-4	Scram Recovery	115

Section 1R13: Maintenance Risk Assessment and Emergent Work Controls

Procedures

Number	Title	Revision
01-S-18-6	Risk Assessment of Maintenance Activities	18
EN-OU-108	Shutdown Safety Management Program (SSMP)	8

Condition Report (CR-GGN-1-)

2016-03238

Section 1R15: Operability Evaluations

Procedures

Number	Title	Revision
EN-OP-104	Operability Determination Process	10
06-OP-1T48-M-0003	Secondary Containment Integrity Check	109
05-1-02-IV-1	Control Rod Drive Malfunction Off Normal Event Procedure	117
17-S-02-40	Bypassing and Unbypassing LPRMs	119

Condition Reports (CR-GGN-1-)

2016-02914	2016-03376	2016-03490	2016-03495	2016-03527
2016-03543	2016-03558	2016-03568	2016-04353	2016-03070
2016-03238	2016-03707	2016-03089	2016-03602	2016-03018

Miscellaneous

Title
Sections 4.5.3, 3.2.4.2, 15.4.1, 15.4.9, 15A.6.5.3 of Updated Final Safety Analysis Report
Grand Gulf Event Number 51827

Section 1R18: Plant Modifications

Procedures

Number	Title	Revision
04-1-01-D21-1	Area Radiation Monitoring System	18
EN-LI-100	Process Applicability Determination	18
EN-DC-115	Engineering Change Process	18
EN-DC-136	Temporary Modifications	002
10-S-01-39	Grand Gulf Equipment Important to Emergency Response	004

Condition Reports (CR-GGN-1-)

2010-00005	2010-00323	2012-08662	2016-00904	2016-03199
2016-03138				

Work Order (WO)

443196

Engineering Changes (EC-)

65409	64093

Section 1R19: Post-Maintenance Testing

Procedures

Number	Title	Revision/ Date
04-1-03-E51-1	RCIC Turbine Mechanical Overspeed Trip	11
04-1-01-P81-1	High Pressure Core Spray Diesel Generator	076
06-OP-1P75-M-0001	Standby Diesel Generator 11Functional Test	May 25, 2016
06-OP-1P81-M-0002	HPCS Diesel Generator 13 Functional Test	June 1, 2016

Condition Reports (CR-GGN-1-)

2016-04353	2016-04365

Work Orders (WO)

00440255	52583877	00387524	52677708

Section 1R22: Surveillance Testing

Procedures

Number	Title	Revision/ Date
06-OP-1E51-Q-0003	RCIC System Quarterly Pump Operability Verification	137
06-OP-1M61-V-0002	Local Leak Rate Test – AIR – 1M41F034	June 10, 2016

Drawing

Number	Title	Revision
M-1100A	Containment Cooling System	029

Condition Report (CR-GGN-1-)

2016-04959

Work Orders (WO)

357845	143880	48874	333135	51672377

Section 2RS1: Radiological Hazard Assessment and Exposure Controls

Procedures

Number	Title	Revision/ Date
01-S-06-5	Reportable Events or Conditions	December 3, 2014
01-S-08-1	Administration of the GGNS Radiation Protection Program	August 30, 2010
08-S-08-2	Exposure and Contamination Control	February 5, 2015
01-S-08-6	Radioactive Source Control	July 20, 2011
08-S-01-10	Qualification and Training of RP Personnel	January 15, 2008
EN-RP-100	Radiation Worker Expectations	9

Procedures

Number	Title	Revision/ Date
EN-RP-101	Access Control for Radiologically Controlled Areas	11
EN-RP-102	Radiological Control	4
EN-RP-103	Radiation Protection Protective Clothing	0
EN-RP-105	Radiation Work Permit	14
EN-RP-106	Radiological Survey Documentation	7
EN-RP-106-01	Radiological Survey Guidelines	2
EN-RP-108	Radiation Protection Posting	16
EN-RP-121	Radioactive Material Control	12
EN-RP-122	Alpha Monitoring	9
EN-RP-123	Radiological Controls for Highly Radioactive Objects	1
EN-RP-141	Job Coverage	5
EN-RP-143	Source Control	11
EN-RP-152	Conduct of Radiation Protection	1
EN-RP-153	Radiation Protection Fundamentals Program	0
EN-RP-201	Dosimetry Administration	4

Condition Reports (CR-GGN-1-)

2015-01384	2015-06461	2015-06512	2015-06792	2015-07437
2016-00040	2016-00563	2016-00965	2016-01153	2016-01294
2016-01296	2016-01695			

Radiation Work Permits

Number	Title	Revision
2016-1508	Undervessel Maintenance	0
2016-1915	Emerging Work for Maintenance, Tours, and Inspections	0

Audits, Self-Assessments, Surveillances

Number	Title	Date
06-HP-S000-SA-0001	Leak Test of Sealed Sources	October 30, 2015

Audits, Self-Assessments, Surveillances

Number	Title	Date
QA-14/15-2015-GGN-1	QA Radwaste/ Radiation Audit	September 11, 2015

Radiological Surveys

Number	Title	Date
GG-1603-0874	Drywell – Subpile and Under Vessel	March 14, 2016
GG-1603-0722	Containment 135' Entire Elevation	March 11, 2016
GG-1308-0431	Containment 173' RWCU Backwash Valve Room	August 26, 2013
GG-1602-1072	Containment 173' RWCU Backwash Valve Room	February 28, 2016
GG-1603-0804	Containment 161' RWCU Backwash Receiving Tank and Pump Room	March 13, 2016
GG-1305-0032	Containment 161' RWCU Backwash Receiving Tank and Pump Room	May 2, 2013
GG-1306-0042	Containment 161' RWCU Backwash Receiving Tank and Pump Room	June 5, 2013
GG-1511-0396	Aux Building 166' Aux Hot Tool Storage Room	November 27, 2015
GG-1603-0470	Routine Daily Surveys	March 7, 2016
GG-1603-0393	Routine Daily Surveys	March 6, 2016
GG-1602-0350	Drywell 100' Entire Elevation	February 19, 2016
GG-1602-0922	Drywell 121' Mezzanine Above TIPs	February 26, 2016
GG-1603-0530	Drywell 114' Entire Elevation	March 8, 2016
GG-1603-0113	Drywell – Subpile and Under Vessel	March 2, 2016
GG-1603-0400	Drywell – Subpile and Under Vessel	March 6, 2016
GG-1602-0692	Drywell – Subpile and Under Vessel	February 24, 2016
GG-1603-0461	Drywell – Subpile and Under Vessel	March 7, 2016

Section 2RS2: Occupational ALARA Planning and Controls

Procedures

Number	Title	Revision
08-S-01-84	Radiological Work Planning	2
08-S-01-112	Radiation Worker Self-Monitoring Training Program	5
EN-RP-104	Personnel Contamination Events	7

Procedures

Number	Title	Revision
EN-RP-110	ALARA Program	13
EN-RP-110-01	ALARA Initiative Deferrals	1
EN-RP-110-03	Collective Radiation Exposure Reduction Guidelines	4
EN-RP-110-04	Radiation Protection Risk Assessment Process	5
EN-RP-110-05	ALARA Planning and Controls	2
EN-RP-110-06	Outage Dose Estimating and Tracking	1
EN-RP-141	Job Coverage Using Remote Monitoring Technology	1

Radiological Work Permits and ALARA Packages

Number	Title
2016-1402	Refuel Floor High Water Activities
2016-1508	Under Vessel Maintenance
2016-1512	Remove and Replace MSRVs
2016-1516	RF20 ISI
2016-1517	Suppression Pool Hangar and Sparger Inspections
2016-1538	1P45F001A/B Replace Check Valve and Supporting Tasks
2016-1604	1G33F406B Replace Valve Stem
2016-1605	1G33F406B Stem Replacement And Support Work
2016-1915	Emergent Work for Maintenance, Tours, and Inspection

Miscellaneous

Number	Title	Date
	OCC Turnover Meeting Agenda	March 12, 2016
20141007027	Scaling Factors	October 22, 2014
20150827004	Scaling Factors	October 22, 2015

Section 1EP6: Drill Evaluation

Procedure

Number	Title	Revision
10-S-01-1	Activation of the Emergency Plan	126

Condition Report (CR-GGN-1-)

2016-03762

Miscellaneous

Number	Title	Revision/ Date
2016/00090	GGN 2016 Second Quarter Yellow Team Drill Report	May 9, 2016
	Grand Gulf Nuclear Station Emergency Plan	11

Section 4OA1: Performance Indicator Verification

Procedure

Number	Title	Revision
EN-LI-114	Performance Indicator Process	7

Miscellaneous

Number	Title	Date
2015-002-00	Loss of Secondary Containment Differential Pressure during Drawdown Testing	October 1, 2016
2015-003-00	Technical Specification Surveillance on Primary Containment Isolation Valves	October 14, 2016
2016-003-00	Loss of Secondary Containment Safety Function during Routine Roof Inspection	April 7, 2016
	MSPI Indicator Margin Remaining in Green, Grand Gulf Unit 1	June 2016
Draft NEI 99-02 MSPI	Methodologies for Computing the Unavailability Index, the Unreliability Index and Determining Performance Index Validity	August 13, 2002
Attachment 9.2	NRC Performance Indicator Technique/Data Sheet, Indicator: Mitigating Systems Performance Indicator Emergency AC Power (EDG)	Second Quarter 2015
Attachment 9.2	NRC Performance Indicator Technique/Data Sheet, Indicator: Mitigating Systems Performance Indicator Emergency AC Power (EDG)	Third Quarter 2015
Attachment 9.2	NRC Performance Indicator Technique/Data Sheet, Indicator: Mitigating Systems Performance Indicator Emergency AC Power (EDG)	Fourth Quarter 2015
Attachment 9.2	NRC Performance Indicator Technique/Data Sheet, Indicator: Mitigating Systems Performance Indicator Emergency AC Power (EDG)	First Quarter 2016

Miscellaneous

Number	Title	Date
	Data Sheet for Determination of System or Component Unavailability, Plant System/Train: P41 A and P41 B	October 2015
	Data Sheet for Determination of System or Component Unavailability, Plant System/Train: P75 System	October 2015
	Data Sheet for Determination of System or Component Unavailability, Plant System/Train: P41 A and P41B	July 2015
	Data Sheet for Determination of System or Component Unavailability, Plant System/Train: P75 System	July 2015
	Data Sheet for Determination of System or Component Unavailability, Plant System/Train: P41 A and P41B	January 2016
	Data Sheet for Determination of System or Component Unavailability, Plant System/Train: P75 System	January 2016
	Data Sheet for Determination of System or Component Unavailability, Plant System/Train: P41 A and P41B	April 2016
	Data Sheet for Determination of System or Component Unavailability, Plant System/Train: P75 System	April 2016
	MSPI Derivation Report, Grand Gulf Unit 1, MSPI Emergency AC Power System Unreliability Index	October 2015
	MSPI Derivation Report, Grand Gulf Unit 1, MSPI Emergency AC Power System Unreliability Index	July 2015
	MSPI Derivation Report, Grand Gulf Unit 1, MSPI Emergency AC Power System Unreliability Index	January 2016
	MSPI Derivation Report, Grand Gulf Unit 1, MSPI Emergency AC Power System Unreliability Index	April 2016
	MSPI Derivation Report, Grand Gulf Unit 1, MSPI Cooling Water System Unreliability Index	October 2015
	MSPI Derivation Report, Grand Gulf Unit 1, MSPI Cooling Water System Unreliability Index	January 2016
	MSPI Derivation Report, Grand Gulf Unit 1, MSPI Cooling Water System Unreliability Index	April 2016
	MSPI Derivation Report, Grand Gulf Unit 1, MSPI Cooling Water System Unreliability Index	July 2014
	MSPI Derivation Report, Grand Gulf Unit 1, MSPI Cooling Water System Unreliability Index	August 2014
	MSPI Derivation Report, Grand Gulf Unit 1, MSPI Cooling Water System Unreliability Index	September 2014

Miscellaneous

Number	Title	Date
	MSPI Derivation Report, Grand Gulf Unit 1, MSPI Cooling Water System Unreliability Index	October 2014

Section 4OA2: Identification and Resolution of Problems

Procedure

Number	Title	Revision
03-1-01-1	Cold Shutdown to Generator Carrying Minimum Load	169

Condition Reports (CR-GGN-1-)

2016-02513	2016-02518	2005-03573	2008-00788	2009-02347
2011-00196	2011-02422	2011-02911	2011-03304	2011-06223
2011-06239	2011-06237	2015-03421	2015-04370	2016-04780
2016-02960				

Section 4OA3: Event Follow-Up

Procedures

Number	Title	Revision/ Date
EN-HU-102	Human Performance Traps & Tools	14
EN-WM-105	Planning	16
EN-WM-118	Causal Evaluation Process	22
07-S-01-205	Conduct of Maintenance Activities	113
01-S-06-26	Post-Trip Analysis	March 29, 2016
01-S-06-26	Post-Trip Analysis	024
05-1-02-I-1	Reactor Scram	125

Condition Reports (CR-GGN-1-)

2016-02950	2016-2686	2016-03456	2016-02973	2016-03402
2016-02972				

Work Order

397549

Section 4OA5: Other Activities

Calculations

Number	Title	Revision
JC-Q1P81-90024	Division III Degraded Bus Voltage Setpoint Validation	4
JC-Q1P81-90027	Division III Loss of Bus Voltage Setpoint Validation	2

Drawings

Number	Title	Revision
E-1188-017	E22 HPCS Power Supply System Transformer CKT, Aux. Compartment	9
E-1188-021	E22 HPCS Power Supply System Breaker 5	12
E-0010	Synchronizing Diagram, ESF Buses 15AA, 16AB, 17AC, Unit 1	11
E-0121-05	R25 Summary of Relay Settings (ESF) 4.16 KV Bus 17 AC and Diesel Gen 13, Unit 1	7
E-1009	One Line Meter and Relay Diagram, 4.16 KV ESF System Bus 17 AC, Unit 1	9
E-0014	One Line Meter & Relay Diagram, Aux. Elect. Dist. Sys. & Site Power Loop Bus 29UD	17
E-0013	One Line Meter & Relay Diagram, Aux. Elect. Dist. Sys. & Bus 19UD	20
J-1261-012	HPCS Diesel Generator Initiation Logic	0
E-1188-018	HPCS Power Supply System Breaker No. 1 Unit 1	11

Vendor Documents

Number	Title	Revision/Date
NEDO-10905	GE Licensing Topical Report: High-Pressure Core Spray System Power Supply Unit	May 1973
NEDO-10905-1	Amendment 1 to GE Licensing Topical Report: High-Pressure Core Spray System Power Supply Unit	October 1974
NEDO-10905-2	Amendment 2 to GE Licensing Topical Report: High-Pressure Core Spray System Power Supply Unit	April 1976
NEDO-10905-3	Amendment 3 to GE Licensing Topical Report: High-Pressure Core Spray System Power Supply Unit	August 1979
22A3742	Specification: Emergency Core Cooling System Network	February 7, 1973

Vendor Documents

Number	Title	Revision/Date
9170600990	Basler Electric, Instruction Manual for Undervoltage, Overvoltage, Under/Overvoltage Relays BE1-27, BE1-59, BE1-27/59	H
GEK-34053G	GE Instructional Manual: Time Overcurrent Relays Types IAC51A, IAC51B, IAC52A, IAC52B	
GEH-1788L	GE Instructional Manual: Time Overcurrent Relays IAC53A, IAC53B, IAC53C, IAC53R, IAC54A, IAC54B	
GEK-86722	GE Instructional Manual: Time Overcurrent Relay Type IAC66K, Forms 51 and Up	
GEK-45405G	GE Instructional Manual: Differential Voltage Relays Types PVD21A, PVD21B, PVD21C, and PVD21D	

Condition Reports (CR-GGN-1-)

1997-0859	2015-04973	2016-04150	2016-04162

The following items are requested for the
Occupational Radiation Safety Inspection
at Grand Gulf
(March 7 - 18, 2016)
Integrated Report 2016001

Inspection areas are listed in the attachments below.

Please provide the requested information on or before **February 9, 2016**.

Please submit this information using the same lettering system as below. For example, all contacts and phone numbers for Inspection Procedure 71124.01 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 the lead inspector, Pete Hernandez at (817) 200-1168 or Pete.Hernandez@nrc.gov.

PAPERWORK REDUCTION ACT STATEMENT

This letter does not contain new or amended information collection requirements subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.). Existing information collection requirements were approved by the Office of Management and Budget, control number 3150-0011.

1. Radiological Hazard Assessment and Exposure Controls (71124.01)

Date of Last Inspection: **November 16, 2015**

A. List of contacts (with official title) 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. RCA Access Controls and Radworker 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 subtiered systems) since date of last inspection

a. Initiated by the radiation protection organization
b. Assigned to the radiation protection organization
c. Identify any CRs that are potentially related to a performance indicator event

NOTE: The lists should indicate the significance level of each issue and the search criteria used. Please provide documents 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

2. Occupational ALARA Planning and Controls (71124.02)
Date of Last Inspection: **November 16, 2016**

A. List of contacts (with official title) 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 subtiered 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 documents which are "searchable."

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