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

November 13, 2015

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

SUBJECT: GRAND GULF NUCLEAR GENERATING STATION, UNIT 1– NRC COMPONENT DESIGN BASES INSPECTION REPORT 05000416/2015007

Dear Mr. Mulligan:

On October 1, 2015, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Grand Gulf Nuclear Station Unit 1. On August 27, 2015, the NRC inspectors discussed the preliminary results of this inspection with you and other members of your staff. On October 1, 2015, the NRC inspectors discussed the final results of this inspection with you and other members of your staff. Inspectors documented the results of this inspection in the enclosed inspection report.

The NRC inspectors documented seven findings of very low safety significance (Green) in this report. All of these findings involved violations of NRC requirements; one of these violations was determined to be Severity Level IV under the traditional enforcement process. Additionally, the NRC inspectors documented three Severity Level IV violations with no associated finding. The NRC is treating these violations as non-cited violations (NCVs) consistent with Section 2.3.2.a of the Enforcement Policy.

If you contest the violations or significance of these NCV's, 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 crosscutting aspect assignment in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region IV; and the NRC resident inspector at the Grand Gulf Nuclear Station.

In accordance with Title 10 of the *Code of Federal Regulations* 2.390, “Public Inspections, Exemptions, Requests for Withholding,” of the NRC's “Rules of Practice,” a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC’s Public Document Room or from the Publicly Available Records (PARS) component of the NRC’s Agencywide Documents Access and Management System (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Thomas R. Farnholtz, Branch Chief
Engineering Branch 1
Division of Reactor Safety

Docket No: 05000416
License No: NPF-29

Enclosure: Inspection Report
05000416/2015007
w/Attachment: Supplemental Information

cc w/encl: Electronic Distribution for Grand Gulf
Nuclear Station

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket:	50-416
License:	NPF-29
Report Nos.:	05000416/2015007
Licensee:	Entergy Operations, Inc.
Facility:	Grand Gulf Nuclear Station, Unit 1
Location:	7003 Baldhill Road Port Gibson, MS 39150
Dates:	July 27, 2015 to October 1, 2015
Team Leader:	G. George, Senior Reactor Inspector, Engineering Branch 1, Region IV
Inspectors:	L. Brandt, Reactor Inspector, Engineering Branch 1, Region IV R. Latta, Senior Reactor Inspector, Engineering Branch 1, Region IV J. McHugh, Senior Reactor Technology Instructor, Reactor Technology Training (BWR) Branch, Technical Training Center J. Watkins, Reactor Inspector, Engineering Branch 2, Region IV
Senior Reactor Analyst:	D. Loveless, Senior Reactor Analyst, Plant Support Branch 2
Accompanying Personnel:	H. Leake, Contractor, Beckman and Associates W. Sherbin, Contractor, Beckman and Associates
Approved By:	Thomas R. Farnholtz, Branch Chief Engineering Branch 1

SUMMARY

IR 05000416/2015007; 07/24/2015 – 10/01/2015; Grand Gulf Nuclear Station; baseline inspection, NRC Inspection Procedure 71111.21, "Component Design Basis Inspection."

The inspection activities described in this report were performed between July, 24, 2015, and October 1, 2015, by four inspectors from the NRC's Region IV office, one instructor from NRC's Technical Training Center, and two contractors. Seven findings of very low safety significance (Green) are documented in this report. All of these findings involved violations of NRC requirements; one of these violations was determined to be Severity Level IV under the traditional enforcement process. Additionally, NRC inspectors documented three Severity Level IV violations with no associated finding. 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 crosscutting aspects are determined using Inspection Manual Chapter 0310, "Aspects Within the Crosscutting Areas." Violations of NRC requirements are dispositioned in accordance with the NRC"s 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: Mitigating Systems

- Green. The team identified two examples of a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," which states, in part, "design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program." Specifically, prior to September 3, 2015, the licensee failed to verify or check the adequacy of: (1) Safety-related motors and control power circuits fed from Division III 480 V ac emergency safety feature bus 17B01, which were not designed or analyzed to operate using higher voltage ranges that are supplied to the safety-related buses; and (2) safety-related equipment connected to the 125 V dc system were not verified for satisfactory operation at elevated equalizing voltage of 140 V dc. In response to this issue, the licensee performed an operability determination which determined that the condition would reduce the life of the equipment but not cause spurious malfunctions. This finding was entered into the licensee's corrective action program as Condition Reports CR-GGN-2015-4413 and CR-GGN-2015-5130.

  The team determined that the licensee's failure to assure that allowable high voltage conditions are within alternating and direct current equipment ratings was a performance deficiency. This performance deficiency was more than minor, and therefore a finding, because it was associated with the equipment performance attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the operation of the equipment outside of its equipment ratings adversely affects the reliability of safety-related equipment. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012, Exhibit 2, "Mitigating Systems Screening Questions," the finding was determined to have very low safety significance (Green) because it was a design or qualification deficiency that did not

represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non-technical specification equipment; and did not screen as potentially risk significant due to seismic, flooding, or severe weather. This finding did not have a crosscutting aspect because the most significant contributor did not reflect current licensee performance. (Section 1R21.2.1.b.1)

- Green. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," which states, in part, "design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program." Specifically, from January 20, 2010, to August 26, 2015, the licensee issued Calculation EC-Q1111-90028, "AC Power Systems," Revision 6, but failed to verify that the calculated fault current levels were within the ratings of the installed Division III circuit breakers. In response to this issue, the licensee performed an operability evaluation to support an operable but degraded/nonconforming condition, recommending an action to perform a detailed fault current study, and reviewing fault current levels at maximum switchyard voltage of 105 percent to verify that they do not create additional concerns. This finding was entered into the licensee's corrective action program as Condition Reports CR-GGN-2015-4607, CR-GGN-2015-4934, and CR-GGN-2015-5112.

  The team determined that failure to ensure that electrical interrupting devices are rated for available fault current levels was a performance deficiency. This performance deficiency was more than minor, and therefore a finding, because it was associated with the equipment performance attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the licensee's failure to verify the design adequacy of the interrupting equipment would operate with a fault resulted in a reasonable doubt with the operability of Division III motor control center 17B01. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012, Exhibit 2, "Mitigating Systems Screening Questions," the finding was determined to have very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non-technical specification equipment; and did not screen as potentially risk significant due to seismic, flooding, or severe weather. This finding had a human performance crosscutting aspect associated with design margins, because the licensee failed to operate and maintain equipment within design margins [H.6]. (Section 1R21.2.2.b.1)

- Green. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," which states, in part, "Activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings." Specifically, on January 22, 2010, the licensee issued calculation EC-Q1111-90028, "AC Power Systems," Revision 6, but failed to meet the procedural requirement that other documents impacted by the change be identified and updated. In response to this issue, the licensee reviewed the affected calculations to

determine if the design bases was met and created a corrective action to update calculations. This finding was entered into the licensee's corrective action program as Condition Reports CR-GGN-2015-4647 and CR-GGN-2015-4859.

The team determined that the licensee's failure to identify and address the impacts of the revised calculation on other documents in accordance with EN-DC-126 was a performance deficiency. This performance deficiency was more than minor, and therefore a finding, because if left uncorrected, the performance deficiency would have the potential to lead to a more significant safety concern. Specifically, routinely failing to revise the obsolete input data in electrical calculations and other design documents was a significant programmatic deficiency which can result in incorrect conclusions regarding the ability of the equipment to meet its design bases. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012, Exhibit 2, "Mitigating Systems Screening Questions," the finding was determined to have very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non-technical specification equipment; and did not screen as potentially risk significant due to seismic, flooding, or severe weather. This finding had a human performance crosscutting aspect associated with procedure adherence, because individuals failed to follow procedures, processes, and work instructions [H.8]. (Section 1R21.2.3.b.1)

- Green. The team identified a Green, non-cited violation of Technical Specification 3.8.1, AC Sources-Operating, LCO 3.8.1, which requires that three diesel generators be operable. Specifically, since July 1985, the licensee failed to perform Surveillance Requirement 3.8.1.9, because surveillance testing performed did not verify that each diesel generator could reject the single largest post-accident load and maintain engine speed within the required criteria. In response to this issue, the licensee performed an immediate operability determination to confirm that test results from full load reject indicated that, if performed correctly, the results of the Surveillance Requirement 3.8.1.9 test would be acceptable. This finding was entered into the licensee's corrective action program as Condition Reports CR-GGN-2015-4611 and CR-GGN-2015-4627.

  The team determined that the failure to perform Technical Specification Surveillance Requirement 3.8.1.9 was a performance deficiency. This performance deficiency was more than minor, and therefore a finding, because it was associated with the procedure quality attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems to respond to initiating events to prevent undesirable consequences. Specifically, the surveillance procedure error resulted in the acceptance of test results that did not satisfy Technical Specification Surveillance Requirement 3.8.1.9; therefore the test did not demonstrate diesel generator operability. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012, Exhibit 2, "Mitigating Systems Screening Questions," the finding was determined to have very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non-technical specification equipment; and did not screen as potentially risk significant due to

seismic, flooding, or severe weather. This finding did not have a crosscutting aspect because the most significant contributor did not reflect current licensee performance. (Section 1R21.2.4.b.1)

- Severity Level IV/Green. The team identified a Green, Severity Level IV non-cited violation of 10 CFR 50.59, "Changes, Tests, and Experiments," paragraph c(2), (1995 version) which requires that a licensee who desires to make a change in the facility described in the final safety analysis report, which involve an unreviewed safety question shall submit an application for amendment of the license pursuant to 10 CFR 50.90. Specifically, on August 31, 1995, the licensee's incorporation of the use of probabilistic methods for evaluation of tornado missiles into the Grand Gulf Final Safety Analysis Report Section 3.5.2.5 involved an unreviewed safety question because it increased the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the safety analysis report. In response to the issue, the licensee prepared a license amendment request to obtain approval to use probabilistic methods for tornado missile evaluations. This finding was entered into the corrective action program as Condition Reports CR-GGN-2015-04615 and CR-GGN-2015-4760.

  The team determined that the failure to obtain a license amendment prior to implementing a proposed change to the tornado missile protection design requirements was a performance deficiency. This performance deficiency was determined to be more than minor, and therefore a finding, because there was a reasonable likelihood the change would require NRC review and approval. This finding was evaluated using traditional enforcement, because the violation may impact the ability for the NRC to perform its regulatory oversight function. In accordance with the NRC Enforcement Policy, the significance determination process was used to inform the significance of the failure to obtain a license amendment prior to implementing a proposed change to the main control room design requirements. Using Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," the inspectors determined the finding involves the total loss of a safety function, identified by the licensee through a probabilistic risk analysis, individual plant examination for external events, or similar analysis, that contributes to external event initiated core damage accident sequences. Therefore, detailed risk evaluation was necessary. The senior reactor analyst reviewed the Grand Gulf Individual Plant Examination for External Events because it was the best available information on missile damage to exposed safety-related equipment. The senior reactor analyst determined that the finding had very low safety significance (Green) because the probability of damage occurring to the exposed safety-related equipment was 7.7E-9/year, which is below the threshold for additional probabilistic risk evaluation. Since the violation was associated with a Green reactor oversight finding, the traditional enforcement violation was determined to be a Severity Level IV violation, consistent with paragraph 6.1.d(2) of the NRC Enforcement Policy. This finding did not have a crosscutting aspect because the most significant contributor did not reflect current licensee performance. (Section 1R21.2.19.b.1)

- Green. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," which states, in part, "design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program." Specifically, prior to August 14, 2015, the licensee failed to

verify that the safety-related alternating current equipment will operate satisfactorily at the extremes of the allowable alternating current frequency ranges as specified in the updated final safety analysis report and technical specifications. This finding was entered into the licensee's corrective action program as Condition Report CR-GGN-2015-4672.

The team determined that the failure to verify safety-related alternating current equipment for operation at the extremes of the allowed frequency range was a performance deficiency. This performance deficiency was more than minor, and therefore a finding, because it was associated with the equipment performance attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of safety systems that respond to initiating events to prevent undesirable consequences. Specifically, lack of verification that the alternating current equipment would function at the extremes of the allowable frequency range can result in incorrect conclusions regarding the ability of the equipment to meet its design bases. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012, Exhibit 2, "Mitigating Systems Screening Questions," the finding was determined to have very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non-technical specification equipment; and did not screen as potentially risk significant due to seismic, flooding, or severe weather. This finding had a problem identification and resolution crosscutting aspect associated with self-assessments, because the organization failed to conduct self-critical and objective assessment of its programs and policies [P.6]. (Section 1R21.3.2.b.1)

- Green. The team identified a Green, non-cited violation of Technical Specification 5.4, "Procedures," 5.4.1, which states, "Written procedures shall be established, implemented, and maintained covering the following activities: (a) The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978." Specifically, prior to August 10, 2015, the licensee failed to follow Procedures 01-S-07-43, "Control of Loose Items, Temporary Electrical Power, and Access to Equipment," GGNS-CS-17 "Standard for Prevention of Potentially Hazardous Seismic II/I Situations due to Loose Items" and EN-MA-118, "Foreign Material Exclusion," when multiple loose items were left in containment since the previous refueling outage. In response to this issue, the licensee immediately removed all loose items in containment that was not permitted by an associated engineering evaluation. This finding was entered into the corrective action program as Condition Report CR-GGN-2015-4568.

  The team determined that failure to implement procedures for prevention of loose items in the containment structure was a performance deficiency. This performance deficiency was more than minor, and therefore a finding, because if left uncorrected, the performance deficiency would have the potential to lead to a more significant safety concern. Specifically, the failure to control materials and temporary equipment was a significant programmatic deficiency which would have the potential to cause unacceptable or degraded conditions if left undetected (MC 0612, App E). In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012, Exhibit 2, "Mitigating Systems Screening Questions," the issue screened as having very low safety significance (Green) because it was a design or

qualification deficiency that did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of nontechnical specification equipment; and did not screen as potentially risk significant due to seismic, flooding, or severe weather. This finding had a human performance crosscutting aspect associated with avoid complacency, in that the licensee failed to recognize and plan for the possibility of latent issues, even while expecting successful outcomes [H.12]. (Section 1R21.4.b.1)

Cornerstone: Miscellaneous

- Severity Level IV. The team identified a Severity Level IV, non-cited violation of 10 CFR Part 50, Appendix B, Criterion XVII, "Quality Assurance Records," which states, in part, that "sufficient records shall be maintained to furnish evidence of activities affecting quality and shall be identifiable and retrievable." Specifically, prior to August 11, 2015, the licensee failed to maintain and retrieve the cable tray overfill analysis for safety-related cable tray 1BATNQ01. In response to the issue, the licensee recreated the cable tray overfill analysis. This violation was entered into the licensee's corrective action program as Condition Report CR-GGN-2015-4602.

  The team determined that the failure to retrieve the safety-related cable tray overfill analysis record in accordance with 10 CFR 50 Appendix B, Criterion XVII was a performance deficiency. Traditional enforcement was applied to this performance deficiency because it involved a violation that impacted the ability of the NRC to perform its regulatory oversight function. Assessing the violation in accordance with the NRC Enforcement Policy, the team determined it to be a Severity Level IV violation because the cable tray overfill analysis was not retrievable. This violation did not have an assigned crosscutting aspect because crosscutting aspects are not assigned to traditional enforcement violations. (Section 1R21.2.3.b.2)

- Severity Level IV. The team identified six examples of a Severity Level IV, non-cited violation of 10 CFR 50.71, "Maintenance of Records, Making of Reports," Section (e) which states, in part, "each person licensed to operate a nuclear power reactor shall update periodically the final safety analysis report originally submitted as part of the application for the license, to assure that the information included in the report contains the latest information developed. This submittal shall contain all the changes necessary to reflect information and analyses submitted to the Commission by the applicant or licensee or prepared by the applicant or licensee pursuant to Commission requirement since the submittal of the original or the last update to the final safety analysis report." Specifically, since July 18, 2012, the licensee failed to ensure the updated final safety analysis report reflected the current plant configuration. In response to these issues, the licensee initiated corrective actions to update the updated final safety analysis report with the correct information. This violation was entered into the licensee's corrective action program as Condition Reports CR-GGN-2015-4381, CR-GGN-2015-4671, CR-GGN-2015-4733, CR-GGN-2015-4753, CR-GGN-2015-4811, and CR-GGN-2015-4867.

  The team determined that the failure to update the final safety analysis report in accordance with 10 CFR 50.71(e) was a performance deficiency. Traditional enforcement was applied to this performance deficiency because it involved a violation that impacted the ability of the

NRC to perform its regulatory oversight function. Assessing the violation in accordance with the NRC Enforcement Policy, the team determined it to be a Severity Level IV violation because the lack of up-to-date information in the final safety analysis report has not resulted in any unacceptable change to the facility or procedures. This violation did not have an assigned crosscutting aspect because crosscutting aspects are not assigned to traditional enforcement violations. (Section 1R21.2.10.b.1)

- Severity Level IV. The team identified a Severity Level IV, non-cited violation of 10 CFR 50.9, "Completeness and Accuracy of Information," Section (a) which requires information provided to the Commission by an applicant for a license or by a licensee or information required by statute or by the Commission's regulations, orders, or license conditions to be maintained by the applicant or the licensee shall be complete and accurate in all material respects. Specifically, since November 1, 1991, the licensee's response did not include verification from the pump suppliers that the minimum flow rates were sufficient to ensure that there will be no pump damage from low flow operation, or a plan to obtain additional test data and/or modify the minimum flow capacity as needed, per Requested Actions 3 and 6 of NRC Bulletin 88-04. In response to this issue, the licensee initiated corrective actions to submit the correct information. This violation was entered into the licensee's corrective action program as Condition Report CR-GGN-2015-4681.

  The team determined that the failure to correct an incomplete and inaccurate response to NRC Bulletin 88-04, Requested Actions 3 and 6 was a performance deficiency. Traditional enforcement was applied to this performance deficiency because it involved a violation that impacted the ability of the NRC to perform its regulatory oversight function. Assessing the violation in accordance with Section 6.9 of the NRC Enforcement Policy, the team determined it to be a Severity Level IV violation because it resulted in no or relatively inappreciable potential safety or security consequences. This violation did not have a crosscutting aspect because crosscutting aspects are not assigned to traditional enforcement violations. (Section 1R21.2.13.b.1)

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

This inspection of component design bases verifies that plant components are maintained within their design basis. Additionally, this inspection provides monitoring of the capability of the selected components and operator actions to perform their design basis functions. As plants age, modifications may alter or disable important design features making the design bases difficult to determine or obsolete. The plant risk assessment model assumes the capability of safety systems and components to perform their intended safety function successfully. This inspectable area verifies aspects of the Initiating Events, Mitigating Systems and Barrier Integrity cornerstones for which there are no indicators to measure performance.

1R21 Component Design Basis Inspection (71111.21)

.1 Overall Scope

To assess the ability of the Grand Gulf Nuclear Station, Unit 1, equipment and operators to perform their required safety functions, the team inspected risk significant components and the licensee's responses to industry operating experience. The team selected risk significant components for review using information contained in the Grand Gulf Nuclear Station, Unit 1, probabilistic risk assessments and the U. S. Nuclear Regulatory Commission's (NRC) standardized plant analysis risk model. In general, the selection process focused on components that had a risk achievement worth factor greater than 1.3 or a risk reduction worth factor greater than 1.005. The items selected included components in both safety-related and nonsafety-related systems including pumps, circuit breakers, heat exchangers, transformers, and valves. The team selected the risk significant operating experience to be inspected based on its collective past experience.

To verify that the selected components would function as required, the team reviewed design basis assumptions, calculations, and procedures. In some instances, the team performed calculations to independently verify the licensee's conclusions. The team also verified that the condition of the components was consistent with the design basis and that the tested capabilities met the required criteria.

The team reviewed maintenance work records, corrective action documents, and industry operating experience records to verify that licensee personnel considered degraded conditions and their impact on the components. For selected components, the team observed operators during simulator scenarios, as well as during simulated actions in the plant.

The team performed a margin assessment and detailed review of the selected risk-significant components to verify that the design basis have been correctly implemented and maintained. This design margin assessment considered original design issues, margin reductions because of modifications, and margin reductions identified as a result

of material condition issues. Equipment reliability issues were also considered in the selection of components for detailed review. These included items such as failed performance test results; significant corrective actions; repeated maintenance; 10 CFR 50.65(a)1 status; operable, but degraded, conditions; NRC resident inspector input of problem equipment; system health reports; industry operating experience; and licensee problem equipment lists. Consideration was also given to the uniqueness and complexity of the design, operating experience, and the available defense in-depth margins.

The inspection procedure requires a review of 15 to 25 total samples that include risk-significant and low design margin components, components that affect the large-early-release-frequency (LERF), and operating experience issues. The sample selection for this inspection was 19 components, 1 component that affects LERF, and 3 operating experience items. The selected inspection and associated operating experience items supported risk significant functions including the following:

a. Electrical power to mitigation systems: The team selected several components in the electrical power distribution systems to verify operability to supply alternating current (ac) and direct current (dc) power to risk significant and safety-related loads in support of safety system operation in response to initiating events such as loss of offsite power, station blackout, and a loss-of-coolant accident with offsite power available. As such the team selected:

    - Emergency Safety Features Transformer 21
    - Division III 4.16 kV Circuit Breaker 152-1705
    - Division I 4.16 kV Switchgear 15AA
    - Division I Emergency Diesel Generator Load Sequencer
    - Division I 125 V dc Battery Charger 1DA4 and 1DA5
    - High Pressure Core Spray Actuation Circuit
    - 480 V Load Center Feeder Breakers 152-1507 and 152-1604
    - Motor Control Center MCC 15B31
    - Reactor Protection System Actuation Circuit
    - Condensate Storage Tank Level Instrumentation
    - Reactor Core Isolation Cooling Actuation Circuit

b. Components that affect LERF: The team reviewed components required to perform functions that mitigate or prevent an unmonitored release of radiation. The team selected the following components:

    - Containment Vent Path

c. Mitigating systems needed to attain safe shutdown: The team reviewed components required to perform the safe shutdown of the plant. As such the team selected:

    - Reactor Core Isolation Cooling Minimum Flow Valve 1E51-F019A
    - Division II Residual Heat Removal Pump
    - Division II Residual Heat Removal Heat Exchanger

- Standby Liquid Control Valves F004A and F004B
- Emergency Safety Features Switchgear Room Coolers
- High Pressure Core Spray Room Cooler T51B001-C
- Division I Emergency Diesel Generator Fuel Storage Tank

.2 Results of Detailed Reviews for Components:

.2.1 Emergency Safety Features Transformer 21

a. Inspection Scope

The team reviewed the updated safety analysis report, design basis document, calculations, the current system health report, selected drawings, maintenance and test procedures, the vendor manual, and condition reports associated with emergency safety features transformer 21. The team also performed walkdowns and conducted interviews with system engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- Voltage calculations and operating procedures to determine whether transformer taps and administrative controls for switchyard voltage would assure the capability and capacity of offsite power during normal and accident conditions.

- Loading calculations to determine whether the capacity of the transformer is adequate to supply worst-case accident loads.

- Component maintenance history to verify the monitoring of potential degradation.

- Corrective action histories to determine whether there had been any adverse operating trends.

- Visual inspection to assess material condition, the presence of hazards, and consistency of installed equipment with design documentation and analyses.

b. Findings

1. Failure to Ensure Safety-Related Alternating Current and Direct Current Equipment Operability and Functionality at Maximum Allowable Voltage Levels

Introduction. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” because the licensee failed to verify or check the adequacy of design of safety-related alternating current and direct current equipment. Specifically, the team identified two examples where the licensee failed to verify the adequacy of safety-related electrical equipment when operating within the maximum allowable voltage ranges.

Description. Example 1: The design of the Grand Gulf Nuclear Station Class 1E alternating current power system is described in Grand Gulf Updated Final Safety

Analysis Report Section 8.3.1.1.2, “Power Sources for Class 1E AC Power System.” Grand Gulf Updated Final Safety Analysis Report Section 8.3.1.1.2.4, “Testability,” states:

> “Voltage tap settings of the intervening transformers have been set to yield optimum voltage levels of the emergency buses for the full load and minimum load conditions expected throughout the anticipated voltage variations of the offsite power source. The adequacy of these voltage tap settings have been verified and these measurements were correlated with predicted analytical results.”

Contrary to the Grand Gulf Updated Final Safety Analysis Report Section 8.3.1.1.2.4, the adequacy of voltage tap settings for Division III 4160 V ac to 480 V ac motor control center transformer, Q1E22S003-C, was not verified to ensure that the setting would yield optimum voltage levels of the emergency buses throughout the anticipated voltage variations of the offsite power source.

According to Calculation EC-Q1111-90028, Attachment 7A.1, the maximum voltage on 480 V ac emergency safety features motor control centers 17B01 and 17B11 could be as high as 530 V ac. The calculated voltages were determined by setting the switchyard voltage at its highest allowable level (105 percent of 500kV), lightly loaded conditions, and a transformer Q1E22S003-C tap setting of -5 percent. The motors downstream of the transformer are rated at 460 V ac and designed in accordance with National Electric Manufacturers Association Standard MG-1, which specifies a maximum operating voltage of +10 percent = 506 V ac. Similarly, the emergency safety feature battery chargers are rated for a maximum voltage of 508 V ac per Vendor Manual 460000347. Therefore, the safety-related equipment downstream of motor control centers 17B01 and 17B11 would be operated outside of their rated voltage. Operation above 506 V ac was not evaluated for effects on the operability or functionality of the equipment. This condition was entered into the corrective action program as Condition Report CR-GGN-2015-4413. The licensee’s immediate corrective actions consisted of performing an operability evaluation to support an operable condition, initiating an action to perform an additional operability evaluation, and initiating an action to develop a long-term solution to the concern.

On April 10, 2008, an overvoltage condition that caused a malfunction of a safety-related component occurred, as discussed in Licensee Event Report 2008-003. During testing of High Pressure Core Spray low flow valve 1E22-F012, the circuit breaker for the valve motor tripped. The cause was attributed to voltage ≥ 521V. To correct the condition, the licensee raised the trip setting of the circuit breaker.

Example 2: Grand Gulf Updated Final Safety Analysis Report Section 8.3.2.1.6.3 “Battery Capacity Considerations,” states:

> “All direct current equipment for Grand Gulf has been specified for operation over the range of 105 V dc to 140 V dc. Components whose qualifications cannot meet this specified range are evaluated on a case by case basis.”

Furthermore,

> "Unless qualified by field testing, all essential components will be qualified by the manufacturer for the calculated limiting voltage supplied to the equipment."

The equipment connected to the safety-related 125 V dc system have a typical voltage rating of 125 V dc, +/- 10 percent. Therefore, the maximum operating voltage for equipment would be 137.5 V dc. However, the equalizing voltages observed at Grand Gulf included voltages at or near 140 V dc, which represents operation above the maximum voltage range of +10 percent. This condition was entered into the corrective action program as Condition Report CR-GGN-2015-5130. The licensee's immediate corrective actions consisted of performing an operability evaluation to support an operable condition, initiating an action to perform an additional operability evaluation, and initiating an action to develop a long-term solution to the concern.

Analysis. The team determined that the licensee's failure to assure that allowable high voltage conditions are within alternating and direct current equipment ratings was a performance deficiency. This finding is more than minor because it is associated with the equipment performance attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the operation of the equipment outside of its equipment ratings adversely affects the reliability of safety-related equipment. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012, Exhibit 2, "Mitigating Systems Screening Questions," the finding was determined to have very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non-technical specification equipment; and did not screen as potentially risk significant due to seismic, flooding, or severe weather. This finding did not have a crosscutting aspect because the most significant contributor did not reflect current licensee performance.

Enforcement. The team identified two examples of a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," which states, in part, "design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program." Contrary to the above, prior to September 3, 2015, the licensee failed to verify the adequacy of the design of safety-related alternating and direct current equipment, by design reviews, calculational methods, or a testing program, to ensure that the equipment would operate satisfactorily when the maximum voltage rating was exceeded. Specifically, the licensee failed to verify or check the adequacy of: (1) Safety-related motors and control power circuits fed from Division III 480 V ac emergency safety features bus 17B01, which were not designed or analyzed to operate using higher voltage ranges that are supplied to the safety-related buses; and (2) safety-related equipment connected to the 125 V dc system were not verified for satisfactory operation at elevated equalizing voltage of 140 V dc. In response to this issue, the licensee performed an operability determination

which determined that the condition would reduce the life of the equipment but not cause spurious malfunctions. This finding was entered into the licensee's corrective action program as Condition Reports CR-GGN-2015-4413 and CR-GGN-2015-5130. Because this finding was of very low safety significance and has been entered into the licensee's corrective action program, this violation is being treated as a non-cited violation, consistent with Section 2.3.2.a of the NRC Enforcement Policy:
NCV 05000416/2015007-01, "Failure to Ensure Safety-Related Alternating Current and Direct Current Equipment Operability and Functionality at Maximum Allowable Voltage Levels."

.2.2 Division III 4.16 kV Circuit Breaker 152-1705

a. Inspection Scope

The team reviewed the updated safety analysis report, design basis document, calculations, the current system health report, selected drawings, maintenance and test procedures, the vendor manual, and condition reports associated with Division III 4.16kV circuit breaker 152-1705. The team also performed walkdowns and conducted interviews with system engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- Calculations for electrical distribution system loading, steady-state and transient voltages, and maximum short-circuit levels.

- Protective device settings and circuit breaker ratings to confirm adequate selective protection and coordination of connected equipment during worst-case short circuit conditions.

- Corrective action histories to determine whether there had been any adverse operating trends.

- Visual inspection to assess material condition, the presence of hazards, and consistency of installed equipment with design documentation and analyses.

b. Findings

1. Failure to Ensure that Electrical Interrupting Devices are Rated for Available Fault Current Levels

Introduction. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," because the licensee failed to verify or check the adequacy of design of fault interrupting devices. Specifically, the licensee failed to provide an analysis that ensures the ratings of interrupting devices, such as circuit breakers, are adequate to interrupt available fault current levels.

Description. Grand Gulf Updated Final Safety Analysis Report Section 8.3.1.1.6.3 states:

> "Fault current available at all voltage levels has been restricted to values within the certified rating of the interrupting devices employed at that level."

Grand Gulf Updated Final Safety Analysis Report Section 8.3.1.1.4.2.5.2 states:

> "Interrupting capacity of switchgear, motor control centers, and distribution panels are compatible with the short circuit current available at the HPCS bus."

The licensee's "Quality Assurance Program Manual," Revision 29, states that the licensee is committed to Regulatory Guide 1.64, "Quality Assurance Requirements for the Design of Nuclear Power Plants," Revision 2, which cites ANSI N45.2.11-1974, "Quality Assurance Requirements for the Design of Nuclear Power Plants." ANSI N45.2.11 states:

> "Measures shall be applied to verify the adequacy of design. Design verification is the process of reviewing, confirming, or substantiating the design by one or more methods to provide assurance that the design meets the specified design inputs…. The results of design verification efforts shall be clearly documented."

The team determined that (1) Calculation EC-Q1111-90028, "AC Electrical Power Systems Calculation," Revision 6, which was issued on January 22, 2010, contains calculated maximum fault current levels that do not bound worst-case levels, and (2) verification that interrupting devices are rated for the available fault current levels has not been documented.

On January 22, 2010, the licensee issued Calculation EC-Q1111-90028, "AC Power Systems," Revision 6, which includes calculated maximum fault current levels at the alternating current distribution system buses. However, the model failed to calculate the worst-case fault current levels, because it did not set the switchyard voltage at its maximum level of 105 percent. The calculation states, in paragraph 2.3, that the calculated fault current values "will be used as inputs to another evaluation or calculation to determine the acceptability." However, the licensee failed to perform another evaluation or calculation to verify that the ratings of the interrupting devices are compatible with the calculated fault current levels.

The inspectors identified an example of interrupting devices that are not rated for the available fault current level. According to Drawing E-1091, "MCC Tabulation, 480 V. MCC 17B01," Revision 23, high pressure core spray motor control center 17B01 contains General Electric model TEC molded case circuit breakers. The licensee provided General Electric Application Guide GET2779F, which documents that TEC breakers are rated to interrupt a maximum of 10,000 amps. This is lower than the available fault current for motor control center 17B01 calculated in Calculation EC-QIIII-90028 of 15,389 amps, as stated on Attachment 7A.5, Page 3. Therefore, for these particular circuit breakers, the design bases are not met.

The licensee acknowledged that the calculated fault current levels are not worst-case and that it has no analysis that verifies that interrupting device ratings are within available fault current levels. The finding was entered into the licensee's corrective action program as Condition Reports CR-GGN-2015-4607, CR-GGN-2015-4934, and CR-GGN-2015-5112. The licensee's immediate corrective actions consisted of performing an operability evaluation to support an operable/degraded/non-conforming condition, recommending an action to perform a detailed fault current study, and reviewing fault current levels at maximum switchyard voltage of 105 percent to verify that they do not create additional concerns.

Analysis. The team determined that failure to ensure that electrical interrupting devices are rated for available fault current levels was a performance deficiency. This performance deficiency was more than minor, and therefore a finding, because it was associated with the equipment performance attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the licensee's failure to verify the design adequacy of the interrupting equipment would operate with a fault resulted in a reasonable doubt with the operability of Division III motor control centers 17B01. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012, Exhibit 2, "Mitigating Systems Screening Questions," the finding was determined to have very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non-technical specification equipment; and did not screen as potentially risk significant due to seismic, flooding, or severe weather. This finding had a human performance crosscutting aspect associated with design margins, because the licensee failed to operate and maintain equipment within design margins [H.6].

Enforcement. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," which states, in part, "design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program." Contrary to the above, from January 20, 2010, to August 26, 2015, the licensee failed to verify or check the adequacy of the design, such as by the use of alternate or simplified calculation methods, or by a suitable testing program. Specifically, the licensee issued Calculation EC-Q1111-90028, "AC Power Systems," Revision 6, but failed to verify that the calculated fault current levels were within the ratings of the installed Division III circuit breakers. In response to this issue, the licensee performed an operability evaluation to support an operable/degraded/non-conforming condition, recommending an action to perform a detailed fault current study, and reviewing fault current levels at maximum switchyard voltage of 105 percent to verify that they do not create additional concerns. This finding was entered into the licensee's corrective action program as Condition Reports CR-GGN-2015-4607, CR-GGN-2015-4934, and CR-GGN-2015-5112. Because this finding was of very low safety significance and has been entered into the licensee's corrective action program, this violation is being treated as a non-cited violation,

consistent with Section 2.3.2.a of the NRC Enforcement Policy:
NCV 05000416/2015007-02, "Failure to Ensure that Electrical Interrupting Devices are Rated for Available Fault Current Levels."

2. (Open) Unresolved Item URI 05000416/2015007-03: Lack of Coordination of Division III HPCS Switchgear 127N Undervoltage Relays

Introduction. The team identified an unresolved item concerned with the coordination of the instantaneous time delay setting of the 127N undervoltage relays with 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.

Description. The following issues were discussed during the inspection; however, the team must review additional information provided by the licensee to determine whether these issues result in a more than minor performance deficiency or a violation of NRC requirements. In accordance with Inspection Manual Chapter 0612, this issue will be characterized as an unresolved item.

The incoming offsite power supply circuit breakers for Division III 4160 V switchgear 17AC are equipped with 127N undervoltage relays. According to Drawing E-1009, "One Line Meter and Relay Diagram, 4.16kV E.S.F. System Bus 17AC," Revision 9, these relays "Trip incoming breaker to bus & start diesel." According to drawing E-0121-005, "Summary of Relay Settings (ESF) 4.16 kV Bus 17AC and Diesel Gen 13," Revision 7, these relays are set with a 0 second time delay. This instantaneous time response potentially results in lack of coordination of the 127N undervoltage relays with high voltage system protective relays, switchgear overcurrent relays, and loss of voltage relays, thus preventing the other relays from performing their credited design functions.

Protective devices that the 127N undervoltage relays are potentially not coordinated with are as follows:

- Protective relays associated the main transformer and its output circuit: Lack of coordination between the 127N undervoltage relays and the protective relays associated with the main transformer and its output circuit can result in coincident loss of two alternating current power supplies, contrary to the requirements of General Design Criterion 17. Grand Gulf Updated Final Safety Analysis Report Section 8.3.1.2.1 states: "The degree of reliability of the power sources required for safe shutdown is considered very high due to independence and ample redundancy; it equals or exceeds all the requirements of Criterion 17." General Design Criterion 17 states: "Provisions shall be included to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit..." Contrary to this General Design Criterion 17 requirement, a fault on the main transformer or its high voltage connection to the transmission system could cause coincident loss of electric power from both the main generator and the offsite power supply to Division III. The protective relaying on the main transformer and its output circuit is designed to initiate tripping of the main

generator and isolation of the area of the fault without causing any cascading failures. However, the 127N undervoltage relay for the Division III 4160 V switchgear would also respond spuriously to the momentary voltage dip caused by the fault and cause loss of the offsite power supply to Division III.

- Transmission system bus protective relays: Grand Gulf Updated Final Safety Analysis Report Section 8.2, "Offsite Power System," Subsection 8.2.2.1, "Availability Considerations" states: "Short circuits on a section of a bus are isolated without interrupting service to any circuit other than that connected to the faulty bus section." Contrary to this requirement, the instantaneous setting of the 127N relays would not coordinate with the transmission bus protective relays and would react to the momentary voltage dip caused by a transmission system bus fault, resulting also in spurious loss of the offsite power supply to Division III.

- Loss of Voltage Relays: In addition to the 127N undervoltage relays, Division III high pressure core spray 4160 V switchgear 17AC is equipped with 127S1, S2, S3, and S4 loss of voltage relays and 127 1A, 1B, 2A, and 2B degraded voltage relays, which also trip the offsite power supply to 17AC upon actuation. NRC Regulatory Issue Summary 2011-12, "Adequacy of Station Electric Distribution System Voltages," Revision 1, describes one of the functions of the degraded voltage relay time delay as follows: "The time delay shall override the effect of expected short duration grid disturbances, preserving availability of the offsite power source(s)." The same principle is relevant to the other undervoltage relays that automatically trip the switchgear offsite power supply. This conclusion is consistent with Grand Gulf Updated Final Safety Analysis Report Section 8.3.1.1.2.3, which states: "Protective devices of Class 1E systems, particularly the ECCS, are set to maintain continuity of power as long as possible short of causing a derangement of the equipment." However, the 127N undervoltage relays do not preserve availability of power as long as possible in the event of harmless transmission grid voltage transients, such as those caused by lightning strikes and normally-cleared faults on transmission lines, because their instantaneous setting miscoordinates with the time delay setting of the Technical Specification credited loss of voltage relays 127S1, S2, S3, and S4. According to Technical Specification Table TR 3.3.8.1-1, the 127S1, S2, S3, and S4 loss of voltage relays have a time delay setting of 2.3 seconds. According to Section 6.17 of calculation JC-Q1P81-90027, "Division III Loss of Bus Voltage Setpoint Validation (T/S 3.3.8.1)," Revision 2, "Spurious segregation from the off-site source is prevented by the time delay function." However, since the non-Technical Specification 127N relays react instantaneously to trip the offsite power source during momentary voltage dips, their 0-second time delay setting invalidates this credited design function of the 2.3-second time delay of the 127S1, S2, S3, and S4 loss of voltage relays. Therefore, spurious segregation from the offsite source is not prevented, and a vulnerability exists for unnecessary loss of offsite power events initiated by, and subsequent to, harmless voltage transients from the transmission system. An actual event of this type occurred on April 2, 2012, as described in Licensee Event Report 2012-003. A lightning strike on a 500 kV transmission circuit resulted in

actuation of the instantaneous 127N relay and unnecessary loss of the offsite power supply to the Division III electrical distribution system.

- Switchgear 17AC offsite power supply circuit breaker overcurrent relays: Grand Gulf Updated Final Safety Analysis Report Section 8.3.1.1.4.2.5.3, "HPCS Class 1E Electrical Equipment Circuit Protection" states: "Emphasis is given in preserving function and limiting loss of Class 1E equipment function in situations of power loss and equipment failure." Contrary to this statement, the instantaneous setting of the 127N undervoltage relays prevents the offsite power supply circuit breaker overcurrent relays from preserving function and limiting loss of Class 1E equipment function in the event of a switchgear bus fault. Switchgear 17AC offsite power supply circuit breakers are equipped with 151B overcurrent relays that, when actuated, trip and lockout the switchgear supply breakers. The purpose of the lockout function is to prevent attempted reenergization of a faulted bus. However, due to the instantaneous response time of the 127N undervoltage relays, the fault would be cleared and the bus deenergized on the undervoltage signal before the overcurrent relays could respond and initiate the bus lockout signal. This would result in automatic starting of the Division III diesel generator, closure of the diesel generator output breaker onto the faulted bus, and the potential for damage to the diesel generator and further damage to the switchgear.

- Switchgear 17AC feeder circuit breaker overcurrent relays: The circuit breakers for feeders downstream of switchgear 17AC are equipped with 150/151M and 150/151T overcurrent relays that are designed to isolate downstream faults locally. Grand Gulf Updated Final Safety Analysis Report Section 8.3.1.1.2.3, "Control Power and Circuit Protection" states: "A complete analysis of the application and coordination of the protective devices on Class 1E distribution has been conducted. This analysis shows that under design operation of these devices, faults, and undervoltages will be detected and corrected at the lowest level of distribution." Referring to the 150/151M and 150/151T overcurrent relays, Grand Gulf Updated Final Safety Analysis Report Section 8.3.1.1.4.2.5.3 states: "Relay settings are coordinated in such a way that interference of service is not communicated to a 'higher' level involving equipment other than that immediately affected by the fault or overload." Contrary to these requirements, the licensee failed to perform a coordination analysis or to ensure that interference of electrical service is limited as described. The voltage dip caused by a fault on a 4160 V circuit downstream of switchgear 17AC would be detected by the 127N relay, which would react instantaneously to trip the 17AC switchgear offsite power supply circuit breaker rather than isolating the fault locally at the downstream circuit breaker. This is contrary to the design criterion that the fault be detected and corrected at the lowest level of distribution and maintain continuity of power to the switchgear.

These issues were entered into the licensee's corrective action program as Condition Report CR-GGN-2015-4973.

.2.3 Division I 4.16 kV Switchgear 15AA

a. Inspection Scope

The team reviewed the updated safety analysis report, design basis documents, the current system health report, selected drawings, maintenance and test procedures, and condition reports associated with Division I 4.16 kV switchgear 15AA. The team also performed walkdowns and conducted interviews with system engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- Calculations for electrical distribution system loading, steady-state and transient voltages, and maximum short-circuit levels.

- Protective device settings and circuit breaker ratings to confirm adequate selective protection and coordination of connected equipment during worst-case short circuit conditions.

- Degraded voltage and loss of voltage relay protection schemes that initiate automatic transfers from the offsite power supply to the diesel generator.

- Sizing of the incoming feeder cable was reviewed to determine its capability under worst case accident conditions.

- Corrective action histories to determine whether there had been any adverse operating trends.

- Visual inspection to assess material condition, the presence of hazards, and consistency of installed equipment with design documentation and analyses.

b. Findings

1. Failure to Identify and Address Impacts of Revised Calculation Output Data

Introduction. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” because the licensee failed to accomplish activities affecting quality in accordance documented procedures. Specifically, the licensee failed to accomplish identifying and addressing the impacts of a revised calculation in accordance with Engineering Calculation Process Procedure EN-DC-126.

Description. Grand Gulf Procedure EN-DC-126, “Engineering Calculation Process,” Revision 5, Section 4, requires that the Responsible Engineer determines other documents that may be affected by the assigned calculation, and initiates appropriate changes or actions tracking items to ensure impacted documents are updated.

On January 22, 2010, the licensee issued Calculation EC-QIIII-90028, “AC Electrical Power Systems Calculation,” Revision 6, which included changes to various calculated alternating current electrical system parameters, such as voltages and loading levels. In accordance with Procedure EN-DC-126, Attachment 9.3, the calculation included a “Calculation Reference Sheet,” which is used to list the documents that are impacted by the calculation change and an engineering change tracking number for the changes to the impacted documents. The “Calculation Reference Sheet” listed only one impacted output document, Calculation JC-Q1R21-90024-1, for which it failed to list an engineering change tracking number. The team identified examples of other impacted documents that were not included on the Calculation Reference Sheet and for which changes or action tracking items were not initiated. This resulted in the impacted documents containing obsolete input data taken from superseded revisions of Calculation EC-QIIII-90028. The following examples are other impacted documents that the team identified:

- Calculation PR0028, “Protective Relay Setting for Bus 15AA Incoming Feeder Breakers,” Revision 1. Uses an obsolete value for bus loading.

- Calculation EC-Q1R20-91038, “Division 1 480/120V AC Class 1E CPT Circuit Voltage Drop Study,” Revision 1. Uses obsolete values for motor control center minimum voltages.

- Calculation EC-Q1R20-91042, “Division III 480/120 VAC Class 1E CPT Circuit Voltage Drop Study,” Revision 0. Uses obsolete values for motor control center minimum voltages.

- Calculation EC-Q1R20-91049, “Division II 480/120V AC Class 1E CPT Circuit Voltage Drop Study,” Revision 1. Uses obsolete values for motor control center minimum voltages.

- Calculation EC-Q1R28-90037, “Division I 120VAC Class 1E Power Panel Voltage Drop Study,” Revision 4. Uses obsolete values for motor control center minimum voltages.

- Calculation EC-Q1R28-90039, “Division II 120VAC Class 1E Power Panel Voltage Drop Study,” Revision 4. Uses obsolete values for motor control center minimum voltages.

- Calculation EC-Q1R28-90041, “Division III 120VAC Class 1E Power Panel Voltage Drop Study,” Revision 0. Uses obsolete values for motor control center minimum voltages.

This finding was entered into the licensee’s corrective action program as Condition Reports CR-GGN-2015-4647 and CR-GGN-2015-4859.

Analysis. The team determined that the licensee's failure to identify and address the impacts of the revised calculation on other documents in accordance with EN-DC-126 was a performance deficiency. This performance deficiency was more than minor, therefore a finding, because if left uncorrected, the performance deficiency would have the potential to lead to a more significant safety concern. Specifically, routinely failing to revise the obsolete input data in electrical calculations and other design documents was a significant programmatic deficiency which can result in incorrect conclusions regarding the ability of the equipment to meet its design bases. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012, Exhibit 2, "Mitigating Systems Screening Questions," the finding was determined to have very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non-technical specification equipment; and did not screen as potentially risk significant due to seismic, flooding, or severe weather. This finding had a human performance crosscutting aspect associated with procedure adherence, because individuals failed to follow procedures, processes, and work instructions [H.8].

Enforcement. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," which states, in part, "Activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings." Contrary to the above, on January 22, 2010, the licensee failed to address impacts of revised calculation results as required by procedure EN-DC-126, "Engineering Calculation Process," Revision 5. Specifically, on January 22, 2010, the licensee issued calculation EC-Q1111-90028, "AC Power Systems," Revision 6, but failed to meet the procedural requirement that other documents impacted by the change be identified and updated. In response to this issued the licensee reviewed the affected calculations to determine if the design bases was met. This finding was entered into the licensee's corrective action program as Condition Reports CR-GGN-2015-4647 and CR-GGN-2015-4859. Because this finding was of very low safety significance and has been entered into the licensee's corrective action program, this violation is being treated as a non-cited violation consistent with the Section 2.3.2a of the NRC Enforcement Policy: NCV 05000416/2015007-04, "Failure to Identify and Address Impacts of Revised Calculation Output Data."

2. Failure to Maintain a Safety-Related Cable Tray Overfill Analysis Record

Introduction. The team identified a Severity Level IV, non-cited violation of 10 CFR 50 Appendix B, Criterion XVII, "Quality Assurance Records," for the licensee's failure to retrieve a safety-related cable tray analysis record. Specifically, the licensee failed to retrieve the cable tray overfill analysis for safety-related cable tray 1BATNQ01.

Description. During the inspection, the team requested documentation of the cable tray overfill analysis for safety-related cable tray 1BATNQ01 and seismic analysis for safety-related cable tray 1AATMH11. The licensee determined that the records were not readily available. The licensee documented this issue in the corrective action program

as Condition Report CR-GGN-2015-4602, to begin searching for the documents. The investigation located the seismic analysis for 1AATMH11, but concluded the cable tray overfill analysis for 1BATNQ01 was not retrievable. After the licensee determined the documentation was not retrievable, the licensee recreated the cable tray overfill analysis.

Analysis. The team determined that the failure to retrieve the safety-related cable tray overfill analysis record in accordance with 10 CFR 50, Appendix B, Criterion XVII was a performance deficiency. Traditional enforcement applied to this performance deficiency because it involved a violation may impact the ability for the NRC to perform its regulatory oversight function. Assessing the performance deficiency in accordance with the NRC Enforcement Policy, the team determined it to be a Severity Level IV violation because the cable tray overfill analysis was not retrievable. This violation did not have an assigned crosscutting aspect because crosscutting aspects are not assigned to traditional enforcement violations.

Enforcement. The team identified a Severity Level IV, non-cited violation of 10 CFR 50 Appendix B, Criterion XVII, "Quality Assurance Records," which states, in part, "sufficient records shall be maintained to furnish evidence of activities affecting quality and shall be identifiable and retrievable." Contrary to the above, prior to August 11, 2015, the licensee did not maintain sufficient records to furnish evidence of activities affecting quality that were retrievable. Specifically, the licensee failed to maintain and retrieve the cable tray overfill analysis for safety-related cable tray 1BATNQ01. In response to the issue, the licensee recreated the cable tray overfill analysis. This violation was entered into the licensee's corrective action program as Condition Report CR-GGN-2015-4602. Because this violation has Severity Level IV significance and was entered into the licensee's corrective action program, this violation is being treated as a non-cited violation consistent with Section 2.3.2.a of the NRC Enforcement Policy:
NCV 05000416/2015007-05, "Failure to Maintain a Safety-Related Cable Tray Overfill Analysis Record."

.2.4 Division I 4.16 kV Load Sequencer

a. Inspection Scope

The team reviewed the updated safety analysis report, technical specifications, design basis documents, the current system health report, selected drawings, operating procedures, and condition reports associated with the Division I 4.16 kV load sequencer. The team also performed walkdowns and conducted interviews with system engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- Load shed and load sequencing schedule drawings to verify that signals are in accordance with design.
- Sequencer logic drawing to verify consistency with design basis.
- Voltage and loading calculations to verify that they correctly model automatic sequencer actuations.

- Setting documents for loss of voltage and degraded voltage bistables to verify consistency with design bases and technical specifications.

- Alarm response procedure to determine whether they adequately address sequencer malfunctions.

- Corrective action histories to determine whether there had been any adverse operating trends.

- Visual inspection to assess material condition, the presence of hazards, and consistency of installed equipment with design documentation and analyses.

b. Findings

1. Failure to Perform Surveillance Requirement 3.8.1.9

Introduction. The team identified a Green, non-cited violation of Technical Specification 3.8.1, AC Sources-Operating, Limiting Condition for Operation 3.8.1, because the licensee failed to maintain three operable diesel generators. Specifically, the licensee failed to perform Surveillance Requirement 3.8.1.9, because surveillance testing performed did not verify that each diesel generator could reject the single largest post-accident load and maintain engine speed within the required criterion.

Description. Technical Specification Surveillance Requirement 3.8.1.9 requires verification that "each DG rejects a load greater than or equal to its associated single largest post-accident load." Contrary to this requirement, Technical Specification Basis Surveillance Requirement 3.8.1.9 and surveillance test procedures for the Divisions I, II, and III diesel generators specify load rejection power values that are lower than the analyzed power demands of the single largest post-accident loads.

Technical Specification Surveillance Requirement 3.8.1.9 states:

> "Verify each DG rejects a load greater than or equal to its associated single largest post accident load and engine speed is maintained less than nominal plus 75% of the difference between nominal speed and the overspeed setpoint of 15% above nominal, whichever is lower."

According to Technical Specification Basis 3.8.1.9, the referenced load for diesel generator 11 is the 1200 kW low pressure core spray pump; for diesel generator 12, the 550 kW residual heat removal pump; and for diesel generator 13 the 2180 kW high pressure core spray pump. These load values are the same as values used as acceptance criteria in Surveillance Procedures 06-OP-1P75-R-0003, "Standby Diesel Generator 11: Functional Test," Revision 124; 06-OP-1P75-R-0004, "Standby Diesel Generator 12: Functional Test," Revision 123; and 06-OP-1P81-R-0001, "HPCS Diesel Generator Functional Test," Revision 123. The team questioned whether the load

values in the surveillance procedures bounded the power demands of the largest loads, since, for Divisions I and II, these procedure values were less than those derived in Calculation MC-Q1P75-90190, “Diesel Fuel Oil Storage Requirements for Division 1 and 2 Diesel Generators,” Revision 4. In response, the licensee recalculated the pump loads. The conclusion of this reanalysis was that the values in Technical Specification Basis Surveillance Requirement 3.8.1.9 and the surveillance procedures were not conservative for all three Divisions. Results of this review were as follows:

- Division I: low pressure core spray pump motor power demand is 1314 kW rather than 1200 kW.

- Division II: residual heat removal pump C motor power demand is 685 kW rather than 550 kW.

- Division III: high pressure core spray pump motor power demand is 2410 kW rather than 2180 kW.

Since the Surveillance Requirement 3.8.1.9 surveillance tests performed to date used load levels that were less than the actual largest post-accident loads, the licensee failed to maintain three operable diesel generators. This finding was entered into the licensee’s corrective action program as Condition Reports CR-GGN-2015-4611 and CR-GGN-2015-4627. The licensee’s immediate corrective actions consisted of performing an operability evaluation to support an operable condition and recommending an action to investigate correcting the technical specification bases.

Analysis. The team determined that the failure to perform Technical Specification Surveillance Requirement 3.8.1.9 was a performance deficiency. This performance deficiency was more than minor, and therefore a finding, because it was associated with the procedure quality attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems to respond to initiating events to prevent undesirable consequences. Specifically, the surveillance procedure error resulted in the acceptance of test results that did not satisfy Technical Specification Surveillance Requirement 3.8.1.9; therefore the test did not demonstrate diesel generator operability. In accordance with Inspection Manual Chapter 0609, Appendix A, “The Significance Determination Process (SDP) for Findings At-Power,” dated June 19, 2012, Exhibit 2, “Mitigating Systems Screening Questions,” the finding was determined to have very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non-technical specification equipment; and did not screen as potentially risk significant due to seismic, flooding, or severe weather. This finding did not have a crosscutting aspect because the most significant contributor did not reflect current licensee performance.

Enforcement. The team identified a Green, non-cited violation of Technical Specification 3.8.1, AC Sources-Operating, Limiting Condition for Operation 3.8.1, which requires that three diesel generators be operable. Contrary to the above, since July 1985, the licensee failed to maintain three operable diesel generators. Specifically, the licensee failed to perform Surveillance Requirement 3.8.1.9, because surveillance testing performed did not verify that each diesel generator could reject the single largest post-accident load and maintain engine speed within the required criteria. In response to this issue, the licensee performed an immediate operability determination to confirm that test results from full load reject indicated that, if performed correctly, the results of the Surveillance Requirement 3.8.1.9 test would be acceptable. This finding was entered into the licensee's corrective action program as Condition Reports CR-GGN-2015-4611 and CR-GGN-2015-4627. Because this finding was of very low safety significance and has been entered into the licensee's corrective action program, this violation is being treated as a non-cited violation consistent with the Section 2.3.2.a of the NRC Enforcement Policy: NCV 05000416/2015007-06, "Failure to Perform Surveillance Requirement 3.8.1.9."

.2.5 Division I Safety-related 125 Vdc Battery Chargers 1DA4 and 1DA5

a. Inspection Scope

The team reviewed the updated safety analysis report, system description, design basis documents, the current system health report, selected drawings and calculations, maintenance and test procedures, and condition reports associated with Division I 125 V dc battery chargers 1DA4 and 1DA5. The team also performed walkdowns and conducted interviews with system engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- Component maintenance history and corrective action program reports to verify the monitoring of potential degradation.
- Calculations for electrical distribution, system load flow/voltage drop to verify that bus capacity and voltages remained within minimum acceptable limits.
- Sizing calculations to verify input assumptions, design loading, and environmental parameters are appropriate and that the battery cell is sized to perform the battery design basis function.
- Procedures for preventive maintenance, inspection, and testing to compare maintenance practices against industry and vendor guidance.
- Battery charger testing methodology was conducted to verify the battery chargers are being tested to ensure that design requirements are being met.
- Battery and battery charger vendor manuals, maintenance activities performed on the batteries and battery chargers.

- Modifications made to the battery chargers.

- Electrolytic capacitor replacement program.

- The material condition of the battery chargers to ensure the battery charger design criteria and maintenance requirements are met.

b. Findings

One example of a violation was identified and documented in Section 1R21.2.1.b.1.

.2.6 High Pressure Core Spray Actuation Circuit

a. Inspection Scope

The team reviewed the updated safety analysis report, system description, design basis documents, the current system health report, selected drawings and calculations, maintenance and test procedures, and condition reports associated with the high pressure core spray actuation system including actuation for the support systems. The team also performed walkdowns and conducted interviews with system engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- Component maintenance history and corrective action program reports to verify the monitoring of potential degradation.

- Sizing calculations, setpoint calculations for protective relaying, design specifications, installation drawings, modifications and upgrades made to the system.

- Procedures for preventive maintenance, procedures for calibrations, inspection, and testing to compare maintenance practices against industry and vendor guidance.

b. Findings

No findings were identified.

.2.7 480 V Load Center Feeder Breakers 152-1507 and 152-1604

a. Inspection Scope

The team reviewed the updated safety analysis report, system description, design basis documents, the current system health report, selected drawings and calculations, maintenance and test procedures, and condition reports associated with the 480 V load center feeder breakers 152-1507 and 152-1604. The team also performed walkdowns

and conducted interviews with system engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- Maintenance and testing activities performed on the breakers in accordance with industry standards.
- Mechanical condition of breaker operating mechanisms.
- Vendor manuals for installation and maintenance.
- Maintenance and testing activities performed on the breakers in accordance with industry standards.

b. Findings

No findings were identified.

.2.8 Motor Control Center MCC 15B31

a. Inspection Scope

The team reviewed the updated safety analysis report, system description, design basis documents, the current system health report, selected drawings and calculations, maintenance and test procedures, and condition reports associated with motor control center MCC 15B31. The team also performed walkdowns and conducted interviews with system engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- Maintenance and testing activities performed on the breakers in accordance with industry standards.
- Mechanical condition of breaker operating mechanisms.
- Design specifications.
- Installation drawings.
- Vendor manuals for installation, maintenance, and testing of the motor control center and the associated installed molded case circuit breakers.
- The material condition of the motor control center to ensure the motor control center design criteria and maintenance requirements are met.

b. Findings

No findings were identified.

.2.9 Reactor Protection System Relays

a. Inspection Scope

The team reviewed the updated safety analysis report, system description, design basis documents, the current system health report, selected drawings and calculations, maintenance and test procedures, and condition reports associated with the reactor protection system relays. The team also performed walkdowns and conducted interviews with system engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- Component maintenance history and corrective action program reports to verify the monitoring of potential degradation.
- Operating experience documents created in the past 10 years to verify that similar experiences applicable to Grand Gulf have been evaluated appropriately.
- Corrective action documents and root cause evaluations for reactor scrams in the past 3 years to verify that the evaluations and identified causes were appropriate.
- Maintenance and testing activities performed on the breakers in accordance with industry and vendor standards, especially those associated with the scram discharge volume.
- Reactor protection system power distribution drawings and logic diagrams to verify drawings represented plant configuration noted during the walkdowns.

b. Findings

No findings were identified.

.2.10 Condensate Storage Tank Level Instrumentation

a. Inspection Scope

The team reviewed the updated safety analysis report, system description, design basis documents, the current system health report, selected drawings and calculations, maintenance and test procedures, and condition reports associated with the condensate storage tank level instrumentation. The team also performed walkdowns and conducted interviews with system engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- Component maintenance history and corrective action program reports to verify the monitoring of potential degradation.

- Operating experience documents created in the past 10 years to verify that similar experiences applicable to Grand Gulf have been evaluated appropriately.

- Procedures for preventive maintenance, inspection, and testing to compare maintenance practices against industry and vendor guidance.

- Calculations and analyses associated with the condensate storage tank water capacity required for station blackout to verify the condensate storage tank had the necessary volume.

- Calculations, procedures, and analyses associated with both the automatic and manual suction swap over of high pressure core spray and reactor core isolation cooling from the condensate storage tank to the suppression pool to verify concerns associated with air entrapment and vortexing were appropriately incorporated.

b. Findings

1. Failure to Update the Final Safety Analysis Report

Introduction. The team identified six examples of a Severity Level IV, non-cited violation of 10 CFR 50.71, "Maintenance of Records, Making of Reports," for the licensee's failure to update the final safety analysis report. Specifically, the team identified six examples where the licensee failed to ensure the final safety analysis report reflected the current plant configuration.

Description. During the course of this inspection, the team identified six examples where the licensee failed to meet the requirements of 10 CFR 50.71(e) to update the final safety analysis report. The updated safety analysis report sections contained information and data that did not reflect the current plant configuration. The examples are:

- Table 6.2-5, "Summary of Short-Term Accident Results for Containment Response to Recirculation Line and Steam Line Breaks," was not updated to reflect analyzed peak post-accident drywell pressure following the extended power uprate. Post power uprate analysis states that peak pressure is 27 psig. The table states peak pressure is 19.4 psig.

- Appendix 8A, "Loss of all Alternating Current Power (Station Blackout)" Section 8A.3, "Condensate Inventory for Decay Heat Removal," states the condensate storage tank is designed to ensure that more than the 115,278 gallons of water required to cope with a four hour station blackout event is maintained. However, the GGNS-NE-10-00034, "Extended Power Uprate Station Blackout Report," Revision 1, states the correct condensate storage tank volume under station blackout conditions is 136,014 gallons.

- Chapter 8.2, “Offsite Power System,” Section 8.2.1.1, “Transmission System,” states the minimum voltage of the 500 kV grid is 496 kV. Section 8.2.4, “Operating Limits” states 500 kV system – minimum 496 kV. However, the Amended and Restated Nuclear Plant Operating Agreement for Grand Gulf Nuclear Station dated December 18, 2013, states that the Grand Gulf 500 kV bus voltage minimum limit is 491 kV.

- Chapter 8.3, “Onsite Power Systems,” Table 8.3-3, “HPCS Division 3 Loads” states that the total loss of coolant accident diesel generator load is 2,960.7 kW. However, Calculation EC-Q1111-90028, “AC Power Systems,” Attachment 7R.4, page 165, reports a load for bus 17 AC of 3216 kW. This is just one example of numerous such discrepancies in Tables 8.3-1, 8.3-2, and 8.3-3.

- Chapter 8.3, “Onsite Power Systems” Table 8.3-4, “Automatic and Manual Loading and Unloading of Engineered Safety Features Bus,” states that the minimum operating requirement for Class 1E battery chargers is energization at 0 seconds during Forced Shutdown and Loss-of-Coolant Accident events. However, Table 8.3-9, “Load Shedding and Sequencing System LSS Table 2, Division 2 – Unit 1” states that the battery chargers are automatically actuated at 10 seconds for a bus undervoltage signal, 20 seconds for a loss of offsite power signal, and not at all for a loss-of-coolant accident signals.

- Section 9.2.6 “Condensate and Refueling Water Storage and Transfer System” Subsection 9.2.6.3 “Safety Evaluation” states that for station blackout only 115,278 gallons of water is required to cope with a four hour station blackout. However, the Extended Power Uprate Station Blackout Report, GGNS-NE-10-00034 Revision 1, states the correct condensate storage tank volume under station blackout conditions is 136,014 gallons.

The licensee documented this issue in the corrective action program as Condition Reports CR-GGN-2015-4381, CR-GGN-2015-4671, CR-GGN-2015-4733, CR-GGN-2015-4753, CR-GGN-2015-4811, and CR-GGN-2015-4867.

Analysis. The team determined that the failure to update the final safety analysis report in accordance with 10 CFR 50.71(e) was a performance deficiency. Traditional enforcement is applied to this performance deficiency because it involved a violation may impact the ability for the NRC to perform its regulatory oversight function. Assessing the performance deficiency in accordance with the NRC Enforcement Policy, the team determined it to be a Severity Level IV violation because the lack of up-to-date information in the final safety analysis report has not resulted in any unacceptable change to the facility or procedures. This violation did not have an assigned crosscutting aspect because crosscutting aspects are not assigned to traditional enforcement violations.

Enforcement. The team identified six examples of a Severity Level IV, non-cited violation of 10 CFR 50.71, “Maintenance of Records, Making of Reports,” Section (e) which states, in part, “each person licensed to operate a nuclear power reactor shall

update periodically the final safety analysis report originally submitted as part of the application for the license, to assure that the information included in the report contains the latest information developed. This submittal shall contain all the changes necessary to reflect information and analyses submitted to the Commission by the applicant or licensee or prepared by the applicant or licensee pursuant to Commission requirement since the submittal of the original or the last update to the final safety analysis report." Contrary to the above, since July 18, 2012, the licensee did not update the final safety analysis report to assure that the information included in the report contains the latest information developed. Specifically, the licensee failed to ensure the final safety analysis report reflected the current plant configuration. In response to these issue, the licensee created a corrective action to update the final safety analysis report. The violation was entered into the licensee's corrective action program as Condition Reports CR-GGN-2015-4381, CR-GGN-2015-4671, CR-GGN-2015-4733, CR-GGN-2015-4753, CR-GGN-2015-4811, and CR-GGN-2015-4867. Because this violation is Severity Level IV significance and entered into the corrective action program, this violation is being treated as a non-cited violation, consistent with Section 2.3.2.a of the NRC Enforcement Policy: NCV 05000416/2015007-07, "Failure to Update the Final Safety Analysis Report."

.2.11 Reactor Core Isolation Cooling Actuation Circuit

a. Inspection Scope

The team reviewed the updated safety analysis report, design basis documents, the current system health report, selected drawings, maintenance and test procedures, and condition reports associated with the reactor core isolation cooling actuation circuit. The team also performed walkdowns and conducted interviews with system engineering personnel to ensure the capability of this attribute to perform its desired design basis function. Specifically, the team reviewed:

- Automatic reactor core isolation cooling turbine trip setpoint settings for high exhaust back pressure, low pump suction pressure , high room temperature, and turbine overspeed to ensure setpoint bases are in accordance with design requirements .

- Periodic testing of the reactor core isolation cooling turbine initiation logic to ensure the reactor core isolation cooling pump is available to perform its design function when required.

- Periodic testing of the reactor core isolation cooling steam turbine to ensure speed controller operates as required.

- Logic diagrams of the reactor core isolation cooling system initiation to ensure the reactor core isolation cooling pump starts as required by Technical Specification logic settings.

b. Findings

No findings were identified.

.2.12 Containment Vent Path

a. Inspection Scope

The team reviewed the updated safety analysis report, system description documentation, selected drawings, installation and test procedures, and condition reports associated with the 20-inch containment vent path including the associated containment isolation valves. The team also performed walkdowns and conducted interviews with system engineering personnel to ensure the capability of these components to perform their desired design basis function. Specifically, the team reviewed:

- Component installation documentation and corrective action program reports to verify the configuration and functional testing of the components including valve differential pressure parameters.

- Procedures for inspection, testing, and operator actions, to insure performance, appropriate configuration control and adherence to industry and vendor guidance.

- The team also evaluated the functional testing methodology, and pressure values associated with acceptance criteria and whether the values appropriately supported design criteria.

b. Findings

No findings were identified.

.2.13 Reactor Core Isolation Cooling Minimum Flow Valve 1E51F019A

a. Inspection Scope

The team reviewed the updated safety analysis report, system description, design basis documents, the current system health report, selected drawings and calculations, maintenance and test procedures, and condition reports associated with the reactor core isolation cooling minimum flow valve 1E51F019A. The team also performed walkdowns and conducted interviews with system engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- Component maintenance history and corrective action program reports to verify the monitoring of potential degradation.

- Operating experience documents created in the past 10 years to verify that similar experiences applicable to Grand Gulf have been evaluated appropriately.

- Maintenance rule functional failure evaluations to identify any potential underlying common causes.

- Procedures for preventive maintenance, inspection, and testing to compare maintenance practices against industry and vendor guidance.

- Grand Gulf's response to NRC Bulletin 88-04, "Potential Safety-Related Pump Loss."

b. Findings

1. Incomplete and Inaccurate Response to NRC Bulletin 88-04

Introduction. The team identified a Severity Level IV, non-cited violation of 10 CFR 50.9, "Completeness and Accuracy of Information," for the licensee's failure to respond with a complete and accurate response to NRC Bulletin 88-04, "Potential Safety-Related Pump Loss." Specifically, the licensee's response did not include either verification from the pump suppliers that the minimum flow rates were sufficient to ensure that there will be no pump damage from low flow operation, or a plan to obtain additional test data and/or modify the minimum flow capacity as needed, per Requested Actions 3 and 6.

Description. NRC Bulletin 88-04, dated May 5, 1988, identified in part, a concern regarding the adequacy of minimum flow capacity for centrifugal pumps. The bulletin stated many licensees had accounted for thermal effects in establishing minimum flow capacities but had failed to consider flow instability effects. Including the flow instability effects could necessitate an increase in minimum flow settings to protect the pump. Requested Action 3 instructed licensees to evaluate the adequacy of the minimum flow bypass lines for safety-related centrifugal pumps with respect to damage resulting from operation and testing in the minimum flow mode. In addition, the evaluation should include verification from the pump suppliers that current minimum flow rates, or any proposed modifications, were sufficient to ensure there would be no pump damage from low flow operation. If the pump supplier did not verify the adequacy of the current minimum flow capacity, the licensees were instructed to provide a plan to obtain additional test data and/or modify the minimum flow line as necessary. Verification of the adequacy of current minimum flow capacity by the pump manufacturer was reiterated in Requested Action 6.

The team reviewed the licensee's response to NRC Bulletin 88-04 which were described in letters AECM-88/0136, dated July 8, 1988 and AECM-88/0158, dated August 9, 1988, and supporting engineering reports SERI-88-0016, dated July 11, 1988, and SERI-88-0018, dated August 2, 1988. Letter AECM-88/0158 identified that procedures for the residual heat removal pumps would be changed to maintain flow above 4000 gallons per minute in fuel pool cooling assist to alleviate minimum flow concerns. The letter and supporting engineering report SERI-88-0018 stated that minimum flow for the

standby service water pumps was verified to be adequate by the pump vendor. For the remaining pressure core spray, high pressure core spray, and reactor core isolation cooling pumps, the letter stated that the pumps could be run intermittently for up 30 minutes but not more than 2 cumulative hours in a 24 hour period, however this was a generic recommendation from the BWR Owner's Group letter BWROG-8836, "Response to NRC Bulletin 88-04," dated July 29, 1988. None of the documents included information or verification from the pump suppliers regarding the adequacy of the minimum flow capacities of the low pressure core spray, high pressure core spray, and reactor core isolation cooling pumps, as requested by NRC Bulletin 88-04. At the time of the response, the licensee stated that no additional action was necessary.

Upon further inspection, the team identified the reactor core isolation cooling pump had undergone a modification in 1995 to increase the minimum flow capacity from 90 gallons per minute to 95 gallons per minute. The licensee had received a letter in 1991, GEXI91-01686, from Sulzer Bingham Pumps Inc. that referenced NRC Bulletin 88-04 and stated the manufacturer's recommendations for minimum flow rates for the reactor core isolation cooling pump had been modified to accommodate for flow instabilities, impeller recirculation, and vibration. This information was communicated to the licensee's General Plant Manager in 1992 via GIN-92-02553 and incorporated into the plant via Change Notice 95-0061 and Calculation MC-Q1E51-95027. Although the information provided the correct action, the licensee did not correct the erroneous information previously submitted to the NRC in the response to NRC Bulletin 88-04. This performance deficiency was entered into the licensee's corrective action program as Condition Report CR-GGN-2015-4681.

Analysis. The team determined that the failure to correct an incomplete and inaccurate response to NRC Bulletin 88-04, Requested Actions 3 and 6 was a performance deficiency. Traditional enforcement applied to this performance deficiency because it involved a violation may impact the ability for the NRC to perform its regulatory oversight function. Assessing the performance deficiency in accordance with Section 6.9 of the NRC Enforcement Policy, the team determined it to be a Severity Level IV violation because it resulted in no or relatively inappreciable potential safety or security consequences. This violation did not have a crosscutting aspect because crosscutting aspects are not assigned to traditional enforcement violations.

Enforcement. The team identified a Severity Level IV, non-cited violation of 10 CFR 50.9, "Completeness and Accuracy of Information," Section (a) which requires information provided to the Commission by an applicant for a license or by a licensee or information required by statute or by the Commission's regulations, orders, or license conditions to be maintained by the applicant or the licensee shall be complete and accurate in all material respects. Contrary to the above, since November 1, 1991, the licensee provided information to the Commission that was not complete and accurate in all material respects. Specifically, the licensee's response did not include verification from the pump suppliers that the minimum flow rates were sufficient to ensure that there will be no pump damage from low flow operation, or a plan to obtain additional test data and/or modify the minimum flow capacity as needed, per Requested Actions 3 and 6. In response to this issue, the licensee initiated corrective actions to submit the correct information. This violation was entered into the licensee's corrective action program as

Condition Report CR 2015-4681. Because this violation is Severity Level IV significance and entered into the corrective action program, this violation is being treated as a non-cited violation, consistent with Section 2.3.2.a of the NRC Enforcement Policy: NCV 05000416/2015007-08, "Incomplete and Inaccurate Response to NRC Bulletin 88-04."

.2.14 Division II Residual Heat Removal Pump

a. Inspection Scope

The team reviewed the updated safety analysis report, system description, design basis documents, the current system health report, selected drawings and calculations, maintenance and test procedures, and condition reports associated with the Division II residual heat removal pump. The team also performed system walkdowns and conducted interviews with system engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- Component maintenance history and corrective action program reports to verify the monitoring of potential degradation.
- Corrective Action documents issued in the last 5 years to verify that repeat failures, and potential chronic issues, will not prevent the residual heat removal pumps from performing their safety function.
- Procedures for preventive maintenance, inspection, and testing to compare maintenance practices against industry and vendor guidance.
- Schematic diagrams to confirm the pump operation conformed to the design requirements.
- Voltage drop calculations to determine whether the motor had adequate voltage for starting and running under degraded voltage conditions.
- Cable sizing calculations to determine whether the motor circuit cabling had adequate ampacity.
- The maximum power demand of the pump was reviewed to verify it was properly reflected in alternating current distribution system and diesel generator loading analyses.

b. Findings

No findings were identified.

.2.15 Division II Residual Heat Removal Heat Exchanger

a. Inspection Scope

The team reviewed the updated safety analysis report, system description, design basis documents, the current system health report, selected drawings and calculations, maintenance and test procedures, and condition reports associated with the Division II residual heat removal heat exchanger. The team also performed system walkdowns and conducted interviews with system engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- Component maintenance history and corrective action program reports to verify the monitoring of potential degradation.

- Heat exchanger documentation associated with inspection results to ensure that the heat exchanger inspections adequately addressed structural integrity and cleanliness of their tubes.

- Procedures for preventive maintenance, inspection, and testing to compare maintenance practices against industry and vendor guidance.

b. Findings

No findings were identified.

.2.16 Standby Liquid Control Tank, Pump, and Valves F004A and F004B

a. Inspection Scope

The team reviewed the updated safety analysis report, system description, the current system health report, selected drawings, maintenance and test procedures, and condition reports associated with the standby liquid control tank, pump, and valves F004A and F004B. The team also performed walkdowns and conducted interviews with system engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- Component maintenance history and corrective action program reports to verify the monitoring of potential degradation.

- Procedures for preventive maintenance, inspection, and testing to compare maintenance practices against industry and vendor guidance.

- System design basis documents and system modifications to provide sufficient shutdown margin associated with the extended power uprate.

b. Findings

No findings were identified.

.2.17 Emergency Safety Feature Switchgear Room Coolers 1T46-B001A,B through 1T46-B005A,B

a. Inspection Scope

The team reviewed the updated safety analysis report, design basis documents, the current system health report, selected drawings, maintenance and test procedures, and condition reports associated with the emergency safety feature switchgear room coolers. The team also performed walkdowns and conducted interviews with system engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- Heat load calculations used to size the room coolers.
- Periodic thermal performance monitoring to ensure the room coolers were capable of removing design bases heat loads.
- Preventive Maintenance activities to ensure the room coolers were maintained according to manufacturer's recommendations.

b. Findings

No findings were identified.

2.18 High Pressure Core Spray Pump Room Cooler T51B001-C

a. Inspection Scope

The team reviewed the updated safety analysis report, design basis documents, the current system health report, selected drawings, maintenance and test procedures, and condition reports associated with the high pressure core spray pump room cooler. The team also performed walkdowns and conducted interviews with system engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- Heat load calculations used to size the room cooler.
- Periodic thermal performance monitoring to ensure the room cooler was capable of removing design bases heat loads.
- Preventive maintenance activities to ensure the room cooler was maintained according to manufacturer's recommendations.

b. Findings

No findings were identified.

.2.19 Division I Emergency Diesel Generator Fuel Storage Tank

a. Inspection Scope

The team reviewed the updated safety analysis report, design basis documents, the current system health report, selected drawings, maintenance and test procedures, and condition reports associated with the Division I emergency diesel generator fuel storage tank. The team also performed walkdowns and conducted interviews with system engineering personnel to ensure the capability of this attribute to perform its desired design basis function. Specifically, the team reviewed:

- Fuel oil consumption calculations to ensure adequate fuel oil supply in tank to meet design and licensing requirements.
- Tornado missile protection for exposed fuel oil storage and fuel oil day tank vents.
- Fuel oil sampling activities to ensure the diesel fuel is delivered according to site specifications.

b. Findings

1. Failure to Obtain a License Amendment for Use of Probabilistic Methods to Evaluate Tornado Missile Hazards

Introduction. The team identified a Severity Level IV, Green non-cited violation of 10 CFR 50.59, “Changes, Tests, and Experiments,” paragraph c(2) for the licensee’s failure to obtain a license amendment prior to changing the facility by incorporating the use or probabilistic methods to evaluate tornado missile generation.

Description. On August, 31 1995, the licensee issued Updated Final Safety Analysis Report Change Notice 4268. The change to the updated final safety analysis report was to accurately reflect the as-built condition of the plant and to clearly state the basis for accepting unshielded components and openings which are vulnerable to tornado missile hazards. Specifically, the descriptions in Grand Gulf Updated Final Safety Analysis Report Section 3.5 and Table 3.5-8 did not accurately reflect that some safety-related components and building openings were partially exposed. These conditions were identified during the licensee’s plant inspections to develop the Grand Gulf Individual Plant Examination for External Events.

To determine vulnerability of the exposed safety-related components to tornado missile damage, the licensee issued Calculation CC-Q1111-94004, “Probabilistic Evaluation of Tornado Missile Strike for IPEEE Study,” Revision 0, to demonstrate that the cumulative annual probability of a tornado missile strike and its effects on vulnerable safety-related

targets is 7.7E-9/year. Since the annual probability of tornado missile damage was smaller than 1E-7/year, as cited in NUREG 75/087, the change to the final safety analysis report did not deviate from any existing regulatory requirements. Therefore, the licensee determined that the accurate description and incorporation of the analysis into the updated final safety analysis report was not an unreviewed safety question. Or, implementation of the change would not result in an increase in the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the final safety analysis report.

The incorporation of the analysis was included in the December 1995 revision of the Grand Gulf Updated Final Safety Analysis Report. Section 3.5.2.5, "Missile Barriers for Outdoor Equipment" was changed from,

> "Protection is afforded to safety-related components located outdoors as listed in Table 3.5-8."

To,

> "The protection against potential tornado missile damage which is afforded to partially exposed building openings and safety-related components located outdoors as listed in Table 3.5-8. The acceptability of these potential tornado missile targets is based upon a cumulative annual probability of tornado missile strike (and consequently a cumulative probability of target damage) smaller than 1×10^{-7}. The use of 1×10^{-7} as an acceptable probability is in accordance with the criteria set forth in NUREG 75/087 section 3.5.1.4. Further, it is in accordance with the review procedures in NUREG 0800, section 3.5.1.4, and the acceptable probability described in Regulatory Guide 1.117 (See Appendix 3.A page 3A/1.117-1). Finally, the use of this value is consistent with industry practice as stated in EPRI Report NP-2005."

In addition, the licensee added page 3A/1.117-1 to the final safety analysis report, which reads that Section D of Regulatory Guide 1.117, "Tornado Design Classification," dated April 1978, indicates that implementation of this guide is not applicable to Grand Gulf based upon the docket date of the Grand Gulf construction permit. It also reads that Grand Gulf complies with the requirements of the regulatory guide to the extent discussing in Grand Gulf Updated Final Safety Analysis Report Section 3.5.2.5.

The team determined that the incorporation into the final safety analysis report the use of probabilistic methods to evaluate to vulnerability of tornado missile strikes on exposed equipment involved an unreviewed safety question, under the 10 CFR 50.59 rule in effect in 1995. The team determined that the change was an unreviewed safety question because it increased the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the safety analysis report. The change was an increase in probability of occurrence of malfunction of equipment important to safety because, prior to the change, the licensee's final safety analysis report described all outdoor equipment was protected from the design basis tornado threat. Therefore, the probability of malfunction to the outdoor equipment was zero. The incorporation of the probabilistic method of evaluation accepts an increase in the

probability of a malfunction because the probability associated with the malfunction increased from zero to 7.7E-9/year.

In accordance with the NRC Enforcement Manual, the team reviewed this issue under the current 10 CFR 50.59 regulation to determine if this issue would require NRC review and approval. The team determined that this change would require prior approval because the change results in a departure from a method of evaluation described in the final safety analysis report used in establishing the design basis or in the safety analysis.

Analysis. The team determined that the failure to obtain a license amendment prior to implementing a proposed change to the tornado missile protection design requirements was a performance deficiency. This performance deficiency was determined to be more than minor, and therefore a finding, because there was a reasonable likelihood the change would require NRC review and approval. The finding was evaluated using traditional enforcement, because the violation may impact the ability for the NRC to perform its regulatory oversight function. In accordance with the NRC Enforcement Policy, the significance determination process was used to inform the significance of the failure to obtain a license amendment prior to implementing a proposed change to the main control room design requirements. Using Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," the inspectors determined the finding involves the total loss of a safety function, identified by the licensee through a probabilistic risk analysis, individual plant examination for external events, or similar analysis, that contributes to external event initiated core damage accident sequences. Therefore, detailed risk evaluation was necessary. The senior reactor analyst reviewed the Grand Gulf Individual Plant Examination for External Events because it was the best available information on missile damage to exposed safety-related equipment. The senior reactor analyst determined that the finding had very low safety significance (Green) because the probability of damage occurring to the exposed safety-related equipment was 7.7E-9/year, which is below the threshold for additional probabilistic risk evaluation. Since the violation was associated with a Green reactor oversight finding, the traditional enforcement violation was determined to be a Severity Level IV violation, consistent with paragraph 6.1.d(2) of the NRC Enforcement Policy. This finding did not have a crosscutting aspect because the most significant contributor did not reflect current licensee performance.

Enforcement. The team identified a Green, Severity Level IV non-cited violation of 10 CFR 50.59, "Changes, Tests, and Experiments," paragraph c(2), 1995 version, which requires that a licensee who desires to make a change in the facility described in the final safety analysis report, which involve an unreviewed safety question shall submit an application for amendment of the license pursuant to 10 CFR 50.90. Contrary to the above, on August 31, 1995, the licensee's failed to obtain a license amendment prior to making a change to the Grand Gulf final safety analysis report, as updated, that involved an unreviewed safety question. Specifically, the licensee's incorporation of the use of probabilistic methods for evaluation of tornado missiles into the Grand Gulf Final Safety Analysis Report Section 3.5.2.5 involved an unreviewed safety question because it increased the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the safety analysis report. In response to the issue, the licensee prepared a license amendment request to obtain approval of the use of probabilistic

methods. This finding was entered into the corrective action program as Condition Reports CR-GGN-2015-4615 and CR-GGN-2015-4760. Because this violation is Severity Level IV significance and entered into the corrective action program the violation is being treated as a non-cited violation, consistent with Section 2.3.2.a of the NRC Enforcement Policy: NCV 05000416/2015007-09, "Failure to Obtain a License Amendment for Use of Probabilistic Methods to Evaluate Tornado Missile Hazards."

.3 Results of Reviews for Operating Experience

.3.1 Inspection of NRC Information Notice 1994-71, "Degradation of Scram Solenoid Pilot Valves"

a. Inspection Scope

The team reviewed the licensee's evaluation of Information Notice 94-71, "Degradation of Scram Solenoid Pilot Valves", to verify that a program was in place to address replacement of the viton seals associated with the scram pilot solenoid valves. The review included a sample of five completed preventative maintenance items, along with a review of the required 15-year replacement frequency in the plants scheduling system. The team verified that the licensee's review adequately addressed the issues in the information notice.

b. Findings

No findings were identified.

.3.2 Inspection of NRC Generic Letter 2006-002, "Grid Reliability and the Impact on Plant Risk and the Operability of Offsite Power

a. Inspection Scope

The team reviewed the licensee's evaluation of Inspection of Generic Letter 2006-002, "Grid Reliability and the Impact on Plant Risk and the Operability of Offsite Power," to verify that interfaces between the nuclear power plant and the transmission system operator are in place and adequate to maintain the operability of offsite power.

b. Findings

1. Failure to Ensure Equipment Operability and Functionality of Allowable Alternating Current Frequency Range

Introduction. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," because the licensee failed to verify or check the adequacy of design of the alternating current electrical equipment. Specifically, the licensee failed to verify that the safety-related alternating current equipment will operate satisfactorily at the extremes of the allowable alternating current frequency ranges as specified in the updated final safety analysis report and technical specifications.

Description. Grand Gulf Updated Final Safety Analysis Report, Section 8.2.4, states that the electrical distribution system is designed to operate within a frequency range of 58.5 Hz to 61.8 Hz. The Nuclear Plant Operating Agreement between the licensee and the transmission system operator, directs the transmission system operator to maintain the same range: "Grand Gulf Frequency Limits = 58.5 Hz to 61.8 Hz." Technical Specification Surveillance Requirement 3.8.1.2 specifies that the diesel generators shall be verified to achieve a steady-state frequency of 58.8 to 61.2 Hz.

In the, "Quality Assurance Program Manual," Revision 29, the licensee is committed to Regulatory Guide 1.64, "Quality Assurance Requirements for the Design of Nuclear Power Plants," Revision 2, which cites ANSI N45.2.11-1974, "Quality Assurance Requirements for the Design of Nuclear Power Plants." ANSI N45.2.11 states,

> "Measures shall be applied to verify the adequacy of design. Design verification is the process of reviewing, confirming, or substantiating the design by one or more methods to provide assurance that the design meets the specified design inputs…. The results of design verification efforts shall be clearly documented."

The team identified that verification that safety-related alternating current equipment will operate properly at the extremes of the allowable frequency range has not been documented. Implied in the current analyses was the non-conservative assumption that the power supply is providing 60 Hz. These analyses include diesel generator loading, diesel generator fuel consumption, pump developed head and flow, and motor operated valve torque.

Off-nominal frequency operation affects the performance of alternating current equipment in various ways. For example, for induction motors, operation at low frequency causes increased torque and decreased operating speed. Operation at high frequency causes decreased torque, increased starting time, increased operating speed, and increased power demand. For pumps, operation at lower speed results in decreased flow and pressure, and operation at higher speed results in increased flow and pressure. The licensee's calculations do not consider these effects.

In May 2015, the licensee conducted a "Pre-Component Design Basis Inspection" focused self-assessment in accordance with their Procedure EN-LI-104, "Self-Assessment and Benchmark Process," Revision 11, as documented in report LO-GLO-2015-00100. One of the requirements of the self-assessment was to review the issues identified in NRC Information Notice 2008-02, "Findings Identified During Component Design Bases Inspections." The Information Notice discusses "instances where the emergency diesel generators (EDGs) loading calculations failed to account for the increased electrical load resulting from EDG operation at the maximum frequency allowed by technical specifications," and "failure to account for EDG underfrequency in pump test acceptance criteria." However, the licensee failed to address these issues in the self-assessment.

Analysis. The team determined that the failure to verify safety-related alternating current equipment for operation at the extremes of the allowed frequency range in accordance

with ANSI N45.2.11 was a performance deficiency. This performance deficiency was more than minor, and therefore a finding, because it was associated with the equipment performance attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of safety systems that respond to initiating events to prevent undesirable consequences. Specifically, lack of verification that the alternating current equipment would function at the extremes of the allowable frequency range can result in incorrect conclusions regarding the ability of the equipment to meet its design bases. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012, Exhibit 2, "Mitigating Systems Screening Questions," the finding was determined to have very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non-technical specification equipment; and did not screen as potentially risk significant due to seismic, flooding, or severe weather. This finding had a problem identification and resolution crosscutting aspect associated with self-assessments, because the organization failed to conduct self-critical and objective assessment of its programs and policies [P.6].

Enforcement. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," which states, in part, "design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program." Contrary to this above, prior to August 14, 2015, the licensee failed to verify or check the adequacy of design of the alternating current electrical equipment. Specifically, the licensee failed to verify that the safety-related alternating current equipment will operate satisfactorily at the extremes of the allowable alternating current frequency ranges as specified in the updated final safety analysis report and technical specifications. In response to this issue, the licensee is updating calculations to reflect the allowable frequency range. This finding was entered into the licensee's corrective action program as Condition Report CR-GGN-2015-4672. Because this finding was of very low safety significance and has been entered into the licensee's corrective action program, this violation is being treated as a non-cited violation, consistent with Section 2.3.2.a of the NRC Enforcement Policy: NCV 05000416/2015007-10, "Failure to Ensure Equipment Operability and Functionality of Allowable Alternating Current Frequency Range."

3.3 Inspection of NRC Information Notice 2012-06, "Ineffective Use of Vendor Technical Information"

a. Inspection Scope

The team reviewed the licensee's evaluation of Information Notice 2012-06, "Ineffective Use of Vendor Technical Information," to verify that a program was in place to address the receipt and evaluation of vendor information to ensure that adequate attention was applied to the importance of vendor-supplied information. Specifically, the team reviewed the licensee's use of the service life information associated with Agastat relays.

The team verified that the licensee's review adequately addressed the issues in the information notice.

b. Findings

No findings were identified.

.4 Results of Reviews for Operator Actions

a. Inspection Scope

The team selected risk-significant components and operator actions for review using information contained in the licensee's probabilistic risk assessment. This included components and operator actions that had a risk achievement worth factor greater than two or Birnbaum value greater than 1E-6.

For the review of operator actions, the team observed operators during simulator scenarios associated with the selected components as well as observing simulated actions in the plant.

The selected operator actions were

- Completion of the bypass of reactor core isolation cooling isolations within 30 minutes.
- Opening of the control room panel doors within 30 minutes.
- Operator response to the failure of the Division I emergency diesel generator load sequencer.
- Operator response to lowering condensate storage tank level due to reactor core isolation cooling 1E51-F019 valve failures.
- Fire water injection into the feedwater system to maintain adequate core cooling.

b. Findings

1. Failure to Implement Equipment Control Procedures for Loose Items in Containment

Introduction. The team identified a Green, non-cited violation of Technical Specification 5.4, "Procedures," for failure to implement procedures covering equipment control, as recommended in Regulatory Guide 1.33, Revision 2, Appendix A. Specifically, the licensee failed to follow equipment control procedures when multiple loose items were left in containment since the previous refueling outage.

Description. The team performed a walkdown in the containment building and identified multiple loose items were found in the containment structure that had not been

evaluated or secured. Examples of these items include ladders, tool boxes, carts, plastic coverings and oil pads. These items had the potential to either clog or damage emergency core cooling system strainers or to directly impact safety-related components. Following the walkdown, the team identified that the licensee failed to follow the following procedures:

- Standard GGNS-CS-17 "Standard for Prevention of Potentially Hazardous Seismic II/I Situations due to Loose Items," which requires items which weigh more than 10 pounds not be left unattended in, on, or elevated above safety-related components or equipment without an engineering evaluation.

- Procedure 01-S-07-43, "Control of Loose Items, Temporary Electrical Power, and Access to Equipment," Revision 6, Step 6.2, which requires "Loose items should not be left unattended in Containment unless one of the following exists: a) the item is tightly restrained. b) The item has been approved by Design Engineering."

- Procedure EN-MA-118, "Foreign Material Exclusion," Step 5.5.16(b) requires tools and materials to be made failsafe, secured to a tether or in an enclosure device.

In response to this issue, the utility generated nine separate condition reports to identify, evaluate and, where necessary, remove items from containment. These were Condition Reports CR-GGN-2015-4346, CR-GGN-2015-4350, CR-GGN-2015-4352, CR-GGN-2015-4353, CR-GGN-2015-4402, CR-GGN-2015-4403, CR-GGN-2015-4499, CR-GGN-2015-4498, and CR-GGN-2015-4510. These conditions were combined into one condition report to assess any negative trends as Condition Report CR-GGN-2015-4568.

Analysis. The team determined that failure to implement procedures for prevention of loose items in the containment structure is a performance deficiency. This performance deficiency was more than minor, and therefore a finding, because if left uncorrected, the performance deficiency would have the potential to lead to a more significant safety concern. Specifically, the failure to control materials and temporary equipment was a significant programmatic deficiency which would have the potential to cause unacceptable or degraded conditions if left undetected (MC 0612, App E). In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012, Exhibit 2, "Mitigating Systems Screening Questions," the issue screened as having very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of nontechnical specification equipment; and did not screen as potentially risk significant due to seismic, flooding, or severe weather. This finding had a human performance crosscutting aspect associated with avoid complacency, in that the licensee failed to recognize and plan for the possibility of latent issues, even while expecting successful outcomes [H.12].

Enforcement. The team identified a Green, non-cited violation of Technical Specification 5.4, "Procedures," 5.4.1, which states, "Written procedures shall be established, implemented, and maintained covering the following activities: (a) The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978." Contrary to this requirement, prior to August 10, 2015, the licensee failed to implement procedures covering equipment control, as recommended in Regulatory Guide 1.33, Revision 2, Appendix A. Specifically, the licensee failed to follow Procedures 01-S-07-43, "Control of Loose Items, Temporary Electrical Power, and Access to Equipment," GGNS-CS-17 "Standard for Prevention of Potentially Hazardous Seismic II/I Situations due to Loose Items" and EN-MA-118, "Foreign Material Exclusion," when multiple loose items were left in containment since the previous refueling outage. In response to this issue, the licensee immediately removed all loose items in containment that was not permitted by an associated engineering evaluation. This finding was entered into the corrective action program as Condition Report CR-GGN-2015-4568. Because this finding was of very low safety significance and has been entered into the licensee's corrective action program, this violation is being treated as a non-cited violation, consistent with Section 2.3.2.a of the NRC Enforcement Policy: NCV 05000416/2015007-11, "Failure to Implement Equipment Control Procedures for Loose Items in Containment."

4. OTHER ACTIVITIES

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

4OA2 Problem Identification and Resolution (71152)

Component Design Basis Review

The team reviewed condition reports associated with the selected components, operator actions, and operating experience notifications. Any related findings are documented in prior sections of this report. However, the team noted the licensee's engineering staff experienced challenges with the implementation of the corrective action program. These challenges include failures to recognize degraded or nonconforming conditions, and failures to adequately describe degraded or nonconforming conditions when identified. The team noted that these challenges affected the station's ability to identify problems at a low threshold and to promptly correct conditions adverse to quality.

4OA6 Meetings, Including Exit

Exit Meeting Summary

On August 27, 2015, the NRC inspectors discussed the preliminary results of this inspection with K. Mulligan, Site Vice President, and other members of the licensee's staff. On October 1, 2015, the NRC inspectors discussed the final results of this inspection with K. Mulligan, Site Vice President, and other members of the licensee's 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

C. Boschetti, Manager, Nuclear Improvement Oversight
K. Boudreaux, Manager, System Engineering
R. Busick, Manager, Operations
T. Coles, Engineer, Regulatory Assurance
T. Coutu, Director, Regulatory and Performance Improvement
V. Fallacara, General Plant Manager, Operations
B. Grant, Manager, Production
M. Greenough, Design Engineer, Design Engineering
J. Hallenbeck, Manager, Design and Program Engineering
G. Hawkins, Senior Manager, Site Projects-Maintenance
J. Hendrick, Engineer, System Engineering
L. Hendrick, Engineer, Design Engineering
C. Landry, Engineer, Design Engineering
C. Landry, Engineer, System Engineering
N. Matthew, Engineer, System Engineering
A. McMahan, Design Engineer, Design Engineering
E. Meaders, Manager, Training
R. Meister, Senior Specialist, Regulatory Assurance
R. Meyer, Assistant Operations Manager, Operations
M. Milly, Senior Manager, Maintenance
K. Mulligan, Site Vice President
J. Nadeau, Manager, Regulatory Assurance
G. Ormon, Engineer, Design Engineering
G. Phillips, Supervisor, Design Engineering
B. Rowland, Electrician, Maintenance
P. Salgado, Manager, Performance Improvement
A. Sayre, Engineer, System Engineering
R. Scarbrough, Engineer, Regulatory Assurance
T. Wallace, Design Engineer, Design Engineering
M. Whigham, Senior Reactor Operator, Operations
D. Wiles, Director, Engineering
Q. Winston, Engineer, System Engineering

NRC Personnel

N. Day, Resident Inspector
A. Wang, Project Manager, DORL
M. Young, Senior Resident Inspector

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000416/2015007-03	URI	Lack of Coordination of Division III HPCS Switchgear 127N Undervoltage Relays (Section 1R21.2.2)

Opened and Closed

05000416/2015007-01	NCV	Failure to Ensure Safety-Related Alternating Current and Direct Current Equipment Operability and Functionality at Maximum Allowable Voltage Levels (Section 1R21.2.1.b.1)
05000416/2015007-02	NCV	Failure to Ensure that Electrical Interrupting Devices are Rated for Available Fault Current Levels (Section 1R21.2.2.b.1)
05000416/2015007-04	NCV	Failure to Identify and Address Impacts of Revised Calculation Output Data (Section 1R21.2.3.b.1)
05000416/2015007-05	NCV	Failure to Maintain a Safety-Related Cable Tray Overfill Analysis Record (Section 1R21.2.3.b.2)
05000416/2015007-06	NCV	Failure to Perform Surveillance Requirement 3.8.1.9 (Section 1R21.2.4.b.1)
05000416/2015007-07	NCV	Failure to Update the Final Safety Analysis Report (Section 1R21.2.10.b.1)
05000416/2015007-08	NCV	Incomplete and Inaccurate Response to NRC Bulletin 88-04 (Section 1R21.2.13.b.1)
05000416/2015007-09	NCV	Failure to Obtain a License Amendment for Use of Probabilistic Methods to Evaluate Tornado Missile Hazards (Section 1R21.2.19.b.2)
05000416/2015007-10	NCV	Failure to Ensure Equipment Operability and Functionality of Allowable Alternating Current Frequency Range (Section 1R21.3.2.b.1)
05000416/2015007-11	NCV	Failure to Implement Equipment Control Procedures for Loose Items in Containment (Section 1R21.4.b.1)

LIST OF DOCUMENTS REVIEWED

Calculations

NUMBER	TITLE	REVISION
1M41F034/F035	Open/Close Stroke Analysis Margin Analysis	0
38579	HPCS and RCIC System Performance with Regards to CST and Suppression Pool Suction for Level Transmitters E22N054C&G and E51N035A&E	3
9645.01	Condensate & Refueling Water Storage Tank FDN	40
E0013Q	The Effect on MCC Load Feeders of Exceeding 30% Fill of Cable Trays	2
E0015Q	Cable Ampacity Calculations (Power)	11
E0025Q	Cable Sizing Calculations for DC Feeder Cables	3
E0041Q	Control Circuit Voltage Drop Study	0
E0045Q	Protective Devices for Penetration Feed Thru and Pigtail Cables (Power, Control, and Instrumentation)	1
EC 21848	Station Blackout	0
EC 42967	Containment Shielding evaluation	0
EC 44184	Removal of Single Point Vulnerability: Level Switch 1P11N014 Abandoned in Place	0
ECN 95-0061	M-1083A: P & I Diagram, RCIC System	24
EC-Q1111-90016	Voltage Drop Study for AC Motor Operated Valve	14
EC-Q1E12-04001	Evaluation of RHR Pump Motor Acceleration Time	0
EC-Q1E22-93009	Voltage Study for HPCS Injection Valve - Q1E22F004	0
EC-Q1L21-90032	Sizing of 125 VDC Division I Battery and Chargers	2
EC-Q1L21-90033	Division I 125V DC Class 1E Voltage Drop Study	3
EC-Q1L21-90046	Division II 125V DC Class 1E Voltage Drop Study	2
EC-Q1L21-90047	Sizing of 125 VDC Division II Battery and Chargers	0
EC-Q1L21-90047	Sizing of 125 VDC Division II Battery and Chargers	1

Calculations

NUMBER	TITLE	REVISION
EC-Q1L21-90047	Sizing of 125 VDC Division II Battery and Chargers	2
EC-Q1R20-91030	Division I 480/120V AC Class 1E CPT Circuit Coordination Study	1
EC-Q1R20-91031	Division II 480/120V AC Class 1E CPT Circuit Coordination Study	0
EC-Q1R20-91038	Division I 480/120V AC Class 1E CPT Circuit Voltage Drop Study	1
EC-Q1R20-91042	Division III 480/120 VAC Class 1E CPT Circuit Voltage Drop Study	0
EC-Q1R20-91049	Division II 480/120V AC Class 1E CPT Circuit Voltage Drop Study	0
EC-Q1R28-90037	Division I 120VAC Class 1E Power Panel Voltage Drop Study	4
EC-Q1R28-90039	Division II 120VAC Class 1E Power Panel Voltage Drop Study	4
EC-Q1R28-90041	Division III 120VAC Class 1E Power Panel Voltage Drop	0
EC-QIIII-90028	AC Power Systems	6
E-DCP82/5020-1	Transient Loading on Diesel Generator During Load Sequencing	A
JC-Q1E22-N654-1	Instrument Loop Uncertainty and Setpoint Determination for Loops 1E22-N654C&G HPCS Pump Suction Transfer on Low CST Level (TS 3.3.5.1)	4
JC-Q1E51-N635-1	Instrument Loop Uncertainty and Setpoint Determination for Loops 1E51-N635A&E RCIC Pump Suction Transfer on Low CST Level (TS 3.3.5.2)	2
JC-Q1P81-90024	Division III Degraded Bus Voltage Setpoint Validation	4
JC-Q1P81-90027	Division III Loss of Bus Voltage Setpoint Validation	2
JC-Q1R21-90024-1	Division 1 & 2 Degraded Voltage Setpoint Validation	2
M3.9.102	SGTS Infiltration Due to Pipe Breaks	5
M6.7.013	Condensate Storage Tank Reserve Capacity	2
MC-Q1111-08005	Calculation of Vortexing of ECCS Pumps	1

Calculations

NUMBER	TITLE	REVISION
MC-Q1111-93035	Calculation of Degraded Voltage Actuator Capability Torque, Using Motor Torque Derated for Temperature Effect, For Select Generic Letter 89-10 Motor Operated Gate and Globe Valves With AC Motor Actuators	14
MC-Q1E21-04019	LPCS Flow Calculation with Minimum Flow Line Open	0
MC-Q1E22-00010	HPCS and RCIC System Performance with Regards to CST and Suppression Pool Suction for Level Transmitters E22N054C&G and E51N035A&E	3
MC-Q1E22-92002	HPCS Pump Minimum Flow Line Orifice	0
MC-Q1E22-92019	Time at which HPCS Min-Flow Valve will Receive a Close Signal due to High Flow During HPCS Start	0
MC-Q1E51-95027	Orifice Q1E51D005 & Q1E51D012 Sizing Calculation	1
MC-Q1P75-09190	Diesel Fuel Oil Storage Requirement for Division 1 and 2 Diesel Generator	2
MC-Q1P75-90190	Diesel Fuel Oil Storage Requirements for Division 1 and 2 Diesel Generators	4
MC-Q1Z77-92001	Safeguard Switchgear and Battery Room Heating and Cooling Requirements	3
PC-Q1M41-02051	Calculation of Maximum Differential Pressure for AOV 1M41F036 for GGNS AOV Program	1
PC-Q1M41-02221	Calculation of Maximum Differential Pressure for AOV 1M41F034 for GGNS AOV Program	1
PC-Q1M41-02228	Calculation of Maximum Differential Pressure for AOV 1M41F037 for GGNS AOV Program	1
PC-Q1M41-02233	Calculation of Maximum Differential Pressure for AOV 1M41F035 for GGNS AOV Program	1
PC-Q1M41-07014	Operating Thrust/Torque, Actuator Output Capability & Available Actuator Margin for AOV 1M41F034/F035	0
PR0028	Protective Relay Setting for Bus 15AA Incoming Feeder Breakers	1
XC-Q 1111-99001	Emergency Procedure / Severe Accident Calculation	9
XC-Q1P53-05011	Radiological Impact of Secondary Containment Bypass Leakage	2

Procedures

NUMBER	TITLE	REVISION
01-S-06-4	Access and Conduct in the Control Room	13
01-S-06-41	Verification and Validation of EOP and SAG procedures	6
02-S-01-2	Operations Section Procedure: Control and use of Operations Section Directives	54
02-S-01-31	Operations Section Procedure Control Room Rounds Non-Safety-related	37
02-S-01-40	EP Technical bases	6
02-S-01-42	Operations Section Procedure, Switchyard Control	2
02-S-01-43	Operations Section Procedure: Transient Mitigation Strategy	1
02-S-01-9	Operations Section Procedure: Key Control	26
04-1-01-E12-1	System Operating Instruction RHR	144
04-1-01-E21-1	System Operating Instruction LPCS	40
04-1-01-E22-1	System Operating Instruction HPCS	119
04-1-01-E22-1	System Operating Instruction RCIC	133
04-1-01-L11-1	System Operating Instruction Plant DC Systems	124
04-1-01-P11-1	System Operating Instruction: Condensate Storage and Transfer (CS&T) System	44
04-1-01-P11-2	System Operating Instruction: Refueling Water Storage and Transfer System	62
04-1-01-P64-1	System Operating Instruction Fire Protection Water	63
04-1-01-P72-1	Drywell Chilled Water System	41
04-1-01-P75-1	System Operating Instruction Standby Diesel Generator	102
04-1-01-R21-1	System Operating Instruction, Load Shedding and Sequencing System	105
04-1-01-R21-15	System Operating Instruction, ESF Bus 15AA	22
04-1-01-R21-15	System Operating Instruction ESF Bus 15AA	22
04-1-01-R21-17	System Operating Instruction, ESF Bus 17AC	10

Procedures

NUMBER	TITLE	REVISION
04-1-01-Z77-1	Safeguard Switchgear and Battery Room Ventilation System	24
04-1-02-1H13-P601-16A-C4	CST LVL LO	151
04-1-02-1H13-P864-1A-H1	Div 1 LSS Sys Fail	29
04-1-02-1H13-P870-5A-D4	CST LVL HI/LO	152
04-S-01-R23-1	System Operating Instruction, 34.5 KV Switchgear and Transformers	45
04-S-01-R27-1	System Operating Instruction, 500/115 kV System	32
04-S-04-2	General Operating Instruction Operation of Electrical Circuit Breakers Safety-related	57
05-1-02-I-4	ONEP Loss of AC Power	46
05-1-02-II-1	Shutdown From the Remote Shutdown Panel	47
05-1-02-III-5	ONEP Automatic Isolation	49
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06-IC-1B21-R-0002	Reactor Vessel Low/High Water Level Calibration	107
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06-IC-1C11-Q-2001	CRD SCRAM Discharge Volume High Water Level (RPS) Functional Test	103
06-IC-1C11-R-2001	SCRAM Discharge Volume High Water Level (RPS) Calibration	105
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07-S-12-150	General Electric AM 4.16 kV Breaker Overhaul Instructions	2
07-S-12-40	General Cleaning and Inspection of Rotating Electrical Equipment	3
07-S-12-61	Inspection of GE Magna Blast Circuit Breakers	6
07-S-13-61	General Maintenance Instruction Power Supply/Inverter Conditioning/Capacitor Reforming Safety-related	4
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BWROG-TP-09-001	Containment Walkdown Procedure for Potential Strainer Debris	0
CGNS-CS-15	Civil Standard for Temporary Rigging	2

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EN-DC-126	Engineering Calculation Process	5
EN-DC-204	Maintenance Rule Scope and Basis	3
EN-DC-316	Heat Exchanger Performance and Condition Monitoring (Attachment 9.1)	April 11, 2012
EN-LI-113-01	Updated Final Safety Analysis Report Change Process	1
EN-MA-133	Control of Scaffolding	12
EN-MA-134	Offline Motor Electrical Testing	5
EN-OP-115	Conduct of Operations	15
EN-OP-115-08	Annunciator Response	4
EN-OP-200	Plant Transient Response Rules	3
ENS-DC-201	ENS Transmission Grid Monitoring	6
ENS-PL-158	Switchyard and Transmission Interface Requirements	3
EN-WM-105	Remove End Cover, Mechanically Clean Tubes and Reinstall End Cover	September 28, 2008
GGNS-CS15	Civil Standard for Temporary Rigging	2
GGNS-CS17	Standard For Criteria for Prevention of Potentially Hazardous Seismic II/I Situations Due to Loose Items	8
GGNS-NE-10-00018	GGNS EPU Transient	1
OPG-040	Operations Standards and Expectations	12
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J-1268-004	RPS Trip Settings	2
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J-1268-006	RPS System Trip	0
J-1268-007	Backup Scram valve Logic	1
J-1268-008	Recirculation Pump Trip System Logic	1
J-1268-009	Scram Discharge Volume Vent and Drain Valve Logic	0
J-1268-011	RPS Misc. Annunciators	1
J-1268-012	RPS Misc. Annunciators	0
J-1268-013	RPS Miscellaneous Annunciators	0
J-1268-014	RPS Misc. Annunciators	0
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M-KD0128	Hypochlorite tank	A
M-KE0128	Area Piping Water Treatment El 133	A
MO128	Water Treatment Building Plan El 133	8
MO129	Water Treatment Building Plan El 133	4

Design Basis Document

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CGNS-98-0059	Evaluation of Limits and margins associated with ECCS Suction Strainer blockage	17
CGNS-EE-11-00001	ECCS Auto Initiation Att 11 & Att 1	0
GGNS-MS-52	Grand Gulf Nuclear Station Nuclear Plant Engineering Mechanical Standard for HELB Impact Review	0
GGNS-NE-10-00018	GGNS EPU Transient Analysis	1
GGNS-NE-10-00034	GGNS EPU Station Blackout	1
SDC-01	1-125 Volt DC Class 1E Distribution System Divisions I and II	1
SDC-09	4.16 kV ESF Division I and II Distribution System (R11 & R21)	0
SDC-10	ESF Division III Power Distribution System	0
SDC-15	Electrical Penetration Assembly Protection	0
SDC-16	Load Shedding & Sequencing System	0
SDC-E22	High pressure Core Spray (E22)	4
SDC-E51	Design Engineering Criteria Grand Gulf Nuclear Station Reactor Core Isolation Cooling	3
SDC-P41	Standby Service Water (P41)	5
SDC-P81	HPCS Diesel Generator System	1
SDC-Z77	Safeguard Switchgear and Battery Rooms Ventilation System	2

Condition Reports (CR-GGN-…)

2005-00080	2005-02986	2005-03749	2008-00139	2008-04914
2008-04914	2010-00572	2010-00679	2010-06033	2010-07591
2010-07713	2010-08141	2011-00309	2011-01295	2011-01861
2011-01894	2011-03868	2011-03868	2011-05045	2011-06333
2011-06784	2011-07274	2011-08010	2011-08720	2011-08733

Condition Reports (CR-GGN-…)

2011-08951	2011-09005	2011-09033	2011-09046	2011-09071
2011-09095	2011-09154	2011-09264	2011-09340	2012-00028
2012-00202	2012-01302	2012-01391	2012-02210	2012-02265
2012-02825	2012-04887	2012-05498	2012-06396	2012-06761
2012-08885	2012-09873	2012-12426	2012-12511	2012-13283
2012-13290	2013-00319	2013-00853	2013-01222	2013-02592
2013-02653	2013-02873	2013-02975	2013-03719	2013-03826
2013-04943	2013-05215	2013-05611	2013-06235	2013-06500
2013-06692	2013-07108	2013-07108	2013-07119	2013-07392
2013-07465	2013-07883	2014-00070	2014-00873	2014-01072
2014-02573	2014-02824	2014-03131	2014-03147	2014-04602
2014-04911	2014-04911	2014-05029	2014-05534	2014-06344
2014-06966	2014-07299	2014-07794	2014-08262	2015-00485
2015-00647	2015-00648	2015-00801	2015-01297	2015-01412
2015-01709	2015-02341	2015-02346	2015-02617	2015-03190
2015-03411	2015-03648	2015-03648	2015-03945	2015-03980
2015-03999	2015-04013	2015-04023	2015-04910	
LR-LAR-2008-00034		CR-HQN-2008-0591		

Condition Reports Generated During the Inspection (CR-GGN-…)

2015-04259	2015-04275	2015-04276	2015-04291	2015-04333
2015-04346	2015-04347	2015-04348	2015-04349	2015-04350
2015-04352	2015-04353	2015-04358	2015-04360	2015-04364
2015-04381	2015-04402	2015-04403	2015-04404	2015-04405
2015-04413	2015-04498	2015-04499	2015-04510	2015-04525
2015-04568	2015-04600	2015-04602	2015-04607	2015-04609
2015-04610	2015-04611	2015-04612	2015-04615	2015-04627
2015-04647	2015-04652	2015-04671	2015-04672	2015-04681
2015-04682	2015-04733	2015-04740	2015-04753	2015-04760

Condition Reports Generated During the Inspection (CR-GGN-…)

2015-04777	2015-04780	2015-04859	2015-04860	2015-04867
2015-04885	2015-04901	2015-04904	2015-04906	2015-04908
2015-04910	2015-04911	2015-04912	2015-04934	2015-04943
2015-04946	2015-04947	2015-04955	2015-04958	2015-04969
2015-04970	2015-04973	2015-05003	2015-05112	2015-05130

Work Orders

00082250	00084193	00095293	00099358	00194704
00219979	00273593	00302800	00333400	00338345
00350277	00362574	00368342	00397522	00400364
00400501	00402652	00409616	00410278	00410279
00412721	00418496	00418611	04159561	04160586
04160587	04162222	04395986	04395987	04395988
04701425	04701426	04701427	04701428	04705876
04705877	04705878	04706511	04717367	50287039
50315977	51030719	51512609	52023582	52277952
52277954	52323529	52323530	52323539	52323821
52345327	52345327	52347965	52371923	52381722
52395428	52396380	52398216	52429661	52430969
52431368	52444385	52456108	52472241	52476954
52478229	52481029	52485429	52497029	52501160
52502857	52519412	52550077	52550078	52565156
52565914	52576337	52576340	52579042	52590200
52590896	52590896	52592295	52592295	52593833
52593834	52597144	52597375	52598656	52599203
52600485	52600487	52600654	52600654	52600655
52601178	52602309	52602314	52603824	52605474
52605475	52607076	52608111	52608115	52608116
52608293	52613918	52614598	52614601	52615341

Work Orders

52615342	52615343	52615344	52615476	52618689
52618690	52618693	52619000	52621103	52622455
52622813	52622977	52623998	52623999	52626109
52630898	52630898			

Miscellaneous

NUMBER	TITLE	REVISION/DATE
	Entergy Quality Assurance Program Manual	29
	Amended and Restated Nuclear Plant Operating Agreement for Grand Gulf Nuclear Station	1
	Entergy to NRC Letter, "Proposed Amendment to the Operating License" (ML8605270247)	May 19, 1986
	NRC to Entergy letter, "Electrical Distribution System Functional Inspection at Grand Gulf Unit 1; Report Number 50-416/90-24" (ML9103050442)	February 19, 1991
	Entergy to NRC letter, "Response to Generic Letter 2006-02, Grid Reliability and the Impact on Plant Risk and the Operability of Offsite Power (ML060950257)	April 3, 2006
	Entergy to NRC letter, "Response to Request for Additional Information for Generic Letter 2006-02, Grid Reliability and the Impact on Plant Risk and the Operability of Offsite Power (ML070320374)	January 31, 2007
	NRC to Grand Gulf letter, "Response to Generic Letter 2006-02, 'Grid Reliability and the Impact on Plant Risk and the Operability of Offsite Power" (ML071080148)	April 30, 2007
	Entergy to NRC Letter, "LER 2007-001-00, Failure to Comply with Technical Specification 3.3.8.1 - Function 1.b - Loss of Voltage Time Delay" (ML071570207)	June 5, 2007
	Entergy to NRC Letter, "LER 2008-003-00" (ML081570400)	June 5, 2008
	Entergy to NRC Letter, "Licensee Event Report 2012-003-00 ESF Actuation Due to Division III Bus	May 29, 2012

Miscellaneous

NUMBER	TITLE	REVISION/DATE
	Undervoltage following a Lightning Strike" (ML12150A183)	
	CR Status Report for PNP-2008-2095 and HQN-2008-591	August 6, 2008
	System Health Report L11 - ESF 125V BATTERY	Q1-2015
	System Health Report R20 – 480 VAC Distribution	Q1-2015
	C&D Charter Power Systems Model LCR Capacity Factor Data	June 1, 1997
	Division I, II, and III Battery Room Temperature Trends from the Weeks 8/14/15 and 12/12/14	August 20, 2015
	System Health P41- Standby Service Water	Q1-2015
	System Health P81- Emergency Diesel Generator	Q1-2015
	System Health E22- High Pressure Core Spray	Q1-2015
04-1-01-E12-1	Residual Heat Removal System Operating Instruction	144
22A3759AE	Containment and NSSS Interface	1
460000158	General Electric Instruction Manual for High Pressure Core Spray Motor Control Center Vendor Manual	September 25, 1997
460000159	Instructions for LPCS, RHR and HPCS Motors	December 1, 1997
460000163	GE Metal Clad Switchgear	January 31, 2001
460000245	Instruction Manual for Load Shedding and Sequencing Panel	June 15, 2004
460000247	C & D Batteries, Chargers, and Racks	April 19, 2010
460000466	Manual for Installation, Maintenance, Handling and Storage of 480V Motor Control Centers Vendor Manual	January 12, 1998

Miscellaneous

NUMBER	TITLE	REVISION/DATE
460000468	Indoor Metal Clad Switchgear	January 31, 2001
741-S-1400	RHR Pump Performance Test Data, Byron Jackson	June 7, 1977
741-S-1401	RHR Pump Curve, T-36555-1, Byron Jackson	May 13, 1977
741-S-1402	RHR Pump Curve, T-36562, Byron Jackson	May 12, 1977
A-22716	Goulds Pumps, Inc. LPCS System: Q1E21-C002-A Characteristic Curve Centrifugal Pump and Test Data	February 25, 1977
A-22717	Goulds Pumps, Inc. HPCS System: Q1E22-C003-C Characteristic Curve Centrifugal Pump and Test Data	February 25, 1977
A-23270	Goulds Pumps, Inc. RHR System: Q1E12-C003-B-B Characteristic Curve Centrifugal Pump and Test Data	July 1, 1977
A-24149	Goulds Pumps, Inc. RHR System: Q1E12-C003-A-A Characteristic Curve Centrifugal Pump and Test Data	January 24, 1978
A-24262	Goulds Pumps, Inc. RHR System: Q1E12-C002-C-B Characteristic Curve Centrifugal Pump and Test Data	March 1, 1978
AECM 86-0049	Letter from Grand Gulf Nuclear Station Unit 1 to NRC regarding Condensate Storage Tank Transfer Level Setpoint	February 15, 1986
AECM 88-0136	Letter from Grand Gulf Nuclear Station Unit 1 to NRC regarding NRC Bulletin 88-04, Potential Safety-Related Pump Loss	July 8, 1988
AECM 88-0158	Letter from Grand Gulf Nuclear Station Unit 1 to NRC regarding NRC Bulletin 88-04, Potential Safety-Related Pump Loss	August 9, 1988
B262A7898	Wire, Electrical (Insulated)	May 5, 1975
DCP 93/0050	Add Cell to Division I (1A3) and Division II (1B3)	1
DRN 05-1181	Document Revision Notice, SDC-01, 1-125 Volt DC Class 1E Distribution System Division 1 and 2	1
DRN 06-278	Document Revision Notice, SDC-01, 1-125 Volt DC Class 1E Distribution System Division 1 and 2	1

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NUMBER	TITLE	REVISION/DATE
E-009.1	Technical Specification for 350 MVA 4160 Volt Metal-Clad Switchgear	9
E-074.0	Technical Specification for Engineered Safety Features Transformer	6
E-075.1	Technical Specification for Engineered Safety Features Transformer No. 12	7
EC-40251	Design Change to Replace DTE-797 Oil in the Three RHR Train Motors With DTE-732	0
G-EE-2003-005	Evaluation of Ampacity De-rating for 3M Interam Enclosed Cables in the Auxiliary Building	0
GEXI-91-01686	RCIC Pumps – Minimum Flow Requirements Letter	November 1, 1991
GGNS-97-0043	Engineering Report for Allowable Crack Widths for Concrete and CMU Walls	0
GGNS-NE-10-08	GGNS EPU High Pressure Core Spray System	0
GGNS-NE-10-34	GGNS EPU Station Blackout	1
GGNS-NE-10-75	GGNS EPU Containment System Response	2
GGNS-NE-12-00022-000	0000-0125024820R0 Task Report T0407 Small Break Analysis	
GIN-92-02553	RCIC Pump Minimum Operating Flow Letter	May 23. 1992
MNCR 92-028	Material Non-Conformance Report	
MNCR 93-00077	Tap setting change for HPCS MCC transformer	January 3, 1994
MRFF Eval CR-GGN-1999-0167	A RCIC Division 2 isolation occurred during testing	February 7, 1999
MRFF Eval CR-GGN-2012-05384	During calibration of inverter 1E51K603 found output at 62.5 vac but should be 120 vac	April 11, 2012

Miscellaneous

NUMBER	TITLE	REVISION/DATE
MRFF Eval CR-GGN-2012-08403	RCIC Inverter 1E51K603 failed due to high voltage trip	June 19, 2012
OE-NOE-2005-00467	TR3-20 Analysis of Planned Unavailability of BWR HPCI/RCIC Systems	August 12, 2005
OE-NOE-2006-00066	TR3-20 Analysis of Planned Unavailability of BWR HPCI/RCIC Systems Status Report	0
OE-NOE-2008-00186	Non-conservative results associated with scale model testing of HPCI and RCIC pump suction to determine the onset of air entrainment	June 27, 2008
OE-NOE-2008-00200	Calculating submergence level for CST	August 6, 2008
OE-NOE-2014-00031	NRC Question Concerning Reactor Core Isolation Cooling Test Method for System Startup	August 21, 2014
PRGGN-2015-00527 CA-001	Procedure Change to SOI 04-1-01-E22-1	August 25, 2015
Q1E12PT01	Residual Heat Removal Preoperational Test	2
RHR/E12	Residual Health Removal System Health Report	First Quarter 2015
SERI 89-002	GGNS Engineering Report Input for EPG Support	3
SERI-88-0016	System Energy Resources, Inc. Grand Gulf Nuclear Station Engineering Report for Identifying Pumps Affected by NRC Bulletin 88-04	August 17, 1988
SERI-88-0018	Engineering Report for Pump Minimum Flow Adequacy per NRC Bulletin 88-04	0
SLC/C41	Standby Liquid Control System Health Report	First Quarter 2015
TR3-20	Analysis of Planned Unavailability of BWR HPCI/RCIC Systems July 1998 – September 2002	March 2003
VM 460000149	Goulds Pumps Installation, Operation and Maintenance Instructions: Model 3196, Model 3196 MT, Model 3196 XLT	June 24, 2003

In accordance with Title 10 of the *Code of Federal Regulations* 2.390, "Public Inspections, Exemptions, Requests for Withholding," of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC's Public Document Room or from the Publicly Available Records (PARS) component of the NRC's Agencywide Documents Access and Management System (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Thomas R. Farnholtz, Branch Chief
Engineering Branch 1
Division of Reactor Safety

Docket No: 05000416
License No: NPF-29

Enclosure: Inspection Report
05000416/2015007
w/Attachment: Supplemental Information

cc w/encl: Electronic Distribution for Grand Gulf
Nuclear Station

ADAMS ACCESSION NUMBER:

■ SUNSI Review By: GAG	ADAMS ■ Yes ☐ No	■ Non-Sensitive ☐ Sensitive		■ Publicly Available ☐ Non-Publicly Available			Keyword: NRC-002
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SIGNATURE	/RA/	/RA/	/RA/	/RA/	/RA/	/RA/	/RA/	/RA/
DATE	11/3/15	11/9/15	11/2/15	10/29/15	11/9/15	11/9/15	11/9/15	11/13/15

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Letter to Kevin Mulligan from Thomas Farnholtz, dated November 13, 2015

SUBJECT: GRAND GULF NUCLEAR GENERATING STATION, UNIT 1– NRC COMPONENT DESIGN BASES INSPECTION REPORT 05000416/2015007

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