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

October 9, 2012

Michael Perito
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 – NRC COMPONENT DESIGN BASIS INSPECTION REPORT 05000416/2012008

Dear Mr. Perito:

On September 10, 2012, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Grand Gulf Nuclear Station. The enclosed inspection report documents the inspection results which were discussed on September 10, 2012, with J. Browning, General Manager Plant Operations, and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

Six NRC identified findings were identified during this inspection. All six of the findings were determined to have very low safety significance (Green). One of the findings was determined to be a Severity Level IV violation. All of the findings were determined to involve violations of NRC requirements. The NRC is treating these violations as non-cited violations (NCV's) consistent with Section 2.3.2 of the Enforcement Policy.

If you contest these non-cited violations, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001; with copies to the Regional Administrator, Region IV; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Grand Gulf Nuclear Station. The information you provide will be considered in accordance with Inspection Manual Chapter 0305. In addition, if you disagree with the characterization of the crosscutting aspect assigned to any finding 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 Grand Gulf Nuclear Station.

In accordance with 10 CFR 2.390 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 Public Document Room or from the Publicly Available Records (PARS) component of NRC's Agencywide Document Access and Management System (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

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

Dockets No: 50-416
License No: NPF-29

Enclosure: Inspection Report 05000416/2012008
w/ Attachment: Supplemental Information

cc w/ encl:
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U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket:	50-416
License:	NPF-29
Report No.:	2012008
Licensee:	Entergy Operations Inc
Facility:	Grand Gulf Nuclear Station
Location:	7003 Bald Hill Road Port Gibson, MS 39150
Dates:	June 25, 2012, to September 10, 2012
Team Leader:	Gerond A. George, Senior Reactor Inspector, Engineering Branch 1
Inspectors:	James Drake, Senior Reactor Inspector Isaac Anchondo, Reactor Inspector Jonathan Braisted, Ph. D., Reactor Inspector Nnaerika Okonkwo, Reactor Inspector
Accompanying Personnel:	Craig Baron, Beckman and Associates James Leivo, Beckman and Associates
Approved By:	Thomas R. Farnholtz, Branch Chief Engineering Branch 1

SUMMARY OF FINDINGS

IR 05000416/2012008; 06/25/2012 – 09/10/2012; Grand Gulf Nuclear Generating Station; baseline inspection, NRC Inspection Procedure 71111.21, “Component Design Basis Inspection.”

The report covers an announced inspection by a team of five regional inspectors and two contractors. Six findings were identified. Five of the findings were of very low safety significance (Green) and one finding was assigned a Severity Level IV. The final significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, “Significance Determination Process.” Findings for which the significance determination process does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, “Reactor Oversight Process,” Revision 4, dated December 2006.

A. NRC-Identified Findings

Cornerstone: Mitigating Systems

- Green. The team identified a Green non-cited violation of 10 CFR 50, Appendix B, Criterion XI, “Test Control,” which states, in part, “A test program shall be established to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents.” Specifically, prior to July 27, 2012, the licensee’s preventive maintenance Procedures 07-S-12-41, 07-S-12-42, and 07-S-12-61 failed to assure that the 4160 Vac circuit breakers would perform satisfactorily in service when the licensee performed maintenance prior to completing “as-found” tests to verify past operability of the circuit breakers. This finding has been entered into licensee’s corrective action program as Condition Reports CR-GGN- 2012-09035 and CR- GGN-2012-9103.

  The team determined that failure to establish a test program which ensures that test and maintenance procedures associated with safety-related 4160 Vac circuit breakers would perform satisfactorily in service was a performance deficiency. This finding was more than minor because, if left uncorrected, it would lead to a more significant safety concern. Specifically, the failure to perform “as-found” tests prior to performing maintenance in preventive maintenance procedures was a significant programmatic deficiency which could cause unacceptable conditions to go undetected. Using the Inspection Manual Chapter 0609, Appendix A, “The Significance Determination Process for Findings At-Power,” 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 safety function. This finding had a crosscutting aspect in the area of human performance, resources component, because the licensee failed to ensure that test and maintenance procedures were complete, accurate, and up-to-date to assure nuclear safety. [H.2(c)] (1R21.2.1)

- Green. The team identified a Green non-cited violation of 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," which states, in part, "A test program shall be established to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents." Specifically, prior to July 27, 2012, the licensee failed to establish a test program for 125 Vdc safety-related molded case circuit breakers incorporating the requirements of IEEE 308, to ensure the breakers would not degrade and would perform satisfactorily in service. The finding was entered into the licensee's corrective action program as Condition Reports CR-GGN-2012-09030 and CR-GGN-2012-09175.

  The team determined that the failure to establish a testing program incorporating the requirements of IEEE 308 was a performance deficiency. The finding was more than minor, because if left uncorrected, it would lead to a more significant safety concern. Specifically, the failure to establish a testing program was a significant programmatic deficiency that would lead to missed opportunities to detect potential common cause failures from degradation of performance in more than one redundant safety division. Using the Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process for Findings At-Power," 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 safety function. This finding had a crosscutting aspect in the area of problem identification and resolution, corrective action program component; because the licensee failed to thoroughly evaluate problems such that resolutions address cause and extent of condition. Specifically, the licensee failed to thoroughly evaluate the extent of condition associated with previously identified NRC violation involving the failure to test 480 Vac molded case circuit breakers identified during the 2009 component design basis inspection. [P.1(c)] (1R21.2.2)

- Severity Level IV. The team identified a Severity Level IV non-cited violation of 10 CFR 50.59, "Changes, Tests and Experiments" which states, in part, that "a licensee shall obtain a license amendment pursuant to Section 50.90 prior to implementing a proposed change, test, or experiment if this activity would; result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the final safety analysis report (as updated)." Specifically, on August 18, 1987, the licensee implemented a change to the updated safety analysis report which limited credible passive failures in the standby service water system to pump and valve seal leakage without obtaining a license amendment. This finding was entered into the licensee's corrective action program as Condition Report CR-GGN-2012-09267.

  The team determined that the licensee's failure to receive prior NRC approval for changes in licensed activities regarding single passive failure criteria for the standby service water system was a performance deficiency. The performance deficiency was evaluated using traditional enforcement because the finding had the ability to impact the regulatory process. The performance deficiency was more than minor because there was a reasonable likelihood that the change would require NRC review and approval prior to implementation. In accordance with the NRC Enforcement Manual, risk insights from the Inspection Manual Chapter 0609, "Significance Determination Process," are used in determining the significance of 10 CFR 50.59 violations. Using the Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process for Findings At-Power," the team determined that the finding represented a loss of system safety function in that the standby service water system could not meet its 30-day mission time to provide decay heat removal. Therefore, a Detailed Risk Evaluation was necessary. In accordance with Manual Chapter 0609, Appendix A, Section 6, "Detailed Risk Evaluation," the senior reactor analyst evaluated the risk of the degraded condition that resulted from the finding. According to the *Risk Assessment of Operational Events Handbook, Volume 1 – Internal Events*, Section 4.1, "Mission Time Modeling," in most events, 24 hours is sufficient time to bring numerous resources to bear on core cooling. In some events, the choice is conservative and the analysis results are overestimates. Additionally, the analyst determined that Section 4.2 on increasing mission time was not applicable to the subject finding because the decrease in standby service water system water inventory would be obvious and there would be days to respond with makeup sources. Therefore, the analyst determined that the finding was of very low safety significance (Green) because, although the standby service water system could not provide 30 days of decay heat removal without operator action to provide makeup water to the system, it would have been able to complete its 24-hour risk significant mission time. Since the finding had very low safety significance, the finding was determined to be Severity Level IV, in accordance with the NRC Enforcement Policy. The finding does not have a crosscutting aspect because the most significant contributor to the finding does not reflect current licensee performance. (1R21.2.3)

- <u>Green</u>. The team identified a Green non-cited violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," which states, in part, that "Measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformance are promptly identified and corrected." Specifically, on July 12, 2012, the NRC informed the licensee of a violation of 10 CFR 50.59 requirements, but the licensee failed to promptly identify this as an adverse condition and enter this condition into their corrective action program until July 19, 2012. The finding was entered into the licensee's corrective action program as CR-GGN-2012-10075.

  The team determined that the licensee's failure to promptly enter the NRC violation as condition adverse to quality into the corrective action program was a performance deficiency. This finding was more than minor because it adversely affected the design control attribute of the Mitigating Systems Cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the licensee failed to promptly document a violation of 10 CFR 50.59, which delayed an operability evaluation that ultimately determined that compensatory measures were required to ensure that the standby service water system could perform its specified safety function for its entire mission time. Using the Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process for Findings At-Power," the team determined that the finding represented a loss of system safety function in that the standby service water system could not meet its 30-day mission time to provide decay heat removal. Therefore, a Detailed Risk Evaluation was necessary. In accordance with Manual

Chapter 0609, Appendix A, Section 6, “Detailed Risk Evaluation,” the senior reactor analyst evaluated the risk of the degraded condition that resulted from the finding. According to the *Risk Assessment of Operational Events Handbook, Volume 1 – Internal Events*, Section 4.1, “Mission Time Modeling,” in most events, 24 hours is sufficient time to bring numerous resources to bear on core cooling. In some events, the choice is conservative and the analysis results are overestimates. Additionally, the analyst determined that Section 4.2 on increasing mission time was not applicable to the subject finding because the decrease in standby service water system water inventory would be obvious and there would be days to respond with makeup sources. Therefore, the analyst determined that the finding was of very low safety significance (Green) because, although the standby service water system could not provide 30 days of decay heat removal without operator action to provide makeup water to the system, it would have been able to complete its 24-hour risk significant mission time. This finding had a crosscutting aspect in the area of problem identification and resolution, corrective action program component, because the licensee failed to ensure that issues potentially impacting nuclear safety are promptly identified, fully evaluated, and that actions are taken to address safety issues, in a timely manner, commensurate with their safety significance. Specifically, the licensee did not implement a corrective action program with a low threshold for identifying issues completely, accurately, and in a timely manner commensurate with their safety significance. [P.1(a)] (1R21.2.3)

- <u>Green</u>. 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, that “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, from July 19, 2012, to July 29, 2012, the licensee failed correctly evaluate the operability of the standby service water system with a degraded or nonconforming condition and failed to document a sound basis for a reasonable expectation of operability of the standby service water system as required by Procedure EN-OP-104, “Operability Determination Process.” The finding was entered into the licensee’s corrective action program as Condition Report CR-GGN-2012-09356.

  The team determined that the failure to implement the requirements of the operability determination process procedure was a performance deficiency. The finding was more than minor because it adversely affected the equipment performance attribute of the Mitigating Systems Cornerstone to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the standby service water system was incapable of performing its specified safety function for the entire 30-day mission time without compensatory measures. Using the Inspection Manual Chapter 0609, Appendix A, “The Significance Determination Process for Findings At-Power,” the team determined that the finding represented a loss of system safety function in that the standby service water system could not meet its 30-day mission time to provide decay heat removal. Therefore, a Detailed Risk Evaluation was necessary. In accordance with Manual Chapter 0609, Appendix A, Section 6, “Detailed Risk Evaluation,” the senior reactor analyst evaluated the risk of the degraded condition that resulted from the finding. According to the *Risk Assessment of Operational Events Handbook, Volume 1 – Internal Events*, Section 4.1, "Mission Time Modeling," in most events, 24 hours is sufficient time to bring numerous resources to bear on core cooling. In some events, the choice is conservative and the analysis results are overestimates. Additionally, the analyst determined that Section 4.2 on increasing mission time was not applicable to the subject finding because the decrease in standby service water system water inventory would be obvious and there would be days to respond with makeup sources. Therefore, the analyst determined that the finding was of very low safety significance (Green) because the standby service water system could would have been able to complete its 24-hour risk significant mission time although it could not provide 30 days of decay heat removal without operator action to provide makeup water to the system. This finding had a crosscutting aspect in the area of human performance, decision making component, because the licensee did not make decisions that demonstrated that nuclear safety was an overriding priority. Specifically, the licensee did not make safety significant decisions using a systematic process to ensure safety is maintained. [H.1(a)] (1R21.2.3)

- Green. The team identified a Green non-cited violation of 10 CFR 50, Appendix B, Criterion XI, "Test Control," which states, in part, "A test program shall be established to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design document." Specifically, prior to July 27, 2012, the licensee's safety-related 4160 Vac circuit breaker preventive maintenance Procedures 07-S-12-41, 07-S-12-42, and 07-S-12-61 failed to incorporate inspection and test requirements for minimum voltage tests, reduced voltage tests, and inspection of auxiliary switch relay contacts as established in the licensee's circuit breaker maintenance program. This condition was entered into the licensee's corrective action program as Condition Reports CR-GGN 2012-08885 and CR-GGN-2012-09111.

  The team determined that the failure to incorporate required tests and inspections into preventive maintenance procedures for safety-related 4160 Vac circuit breakers was a performance deficiency. This finding was more than minor because, if left uncorrected, it would lead to a more significant safety concern. Specifically, the failure to incorporate the testing, cleaning, and inspection requirements into preventive maintenance procedures were a significant programmatic deficiency which could cause unacceptable conditions to go undetected. Using the Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process for Findings At-Power," 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 safety function. This finding had a crosscutting aspect in the area of problem identification and resolution, operating experience component, because the licensee failed to use operating experience information, including vendor recommendations and internally generated lessons learned, to support plant safety. Specifically, the licensee did not implement and institutionalize operating experience through changes to processes, procedures, equipment, and training programs. [P.2(b)] (1R21.2.4)

REPORT DETAILS

1. REACTOR SAFETY

Inspection of component design basis verifies the initial design and subsequent modifications and provides monitoring of the capability of the selected components and operator actions to perform their design basis functions. As plants age, their design basis may be difficult to determine and important design features may be altered or disabled during modifications. 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)

To assess the ability of the Grand Gulf Nuclear Station 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 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.

.1 Inspection Scope

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 the review of operator actions, 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, containment related components, and operating experience issues. The sample selection for this inspection was 16 components, 5 operating experience items, and 4 event based activities associated with the components. 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:

    - Division III Emergency Diesel Generator Output Circuit Breaker 152-1701
    - Division III 125 Vdc Battery and Safety Bus
    - Division III Emergency Diesel Generator 13
    - Division III 4160 Vac Engineered Safety Feature Switchgear Bus 17 AC
    - Division III 480 Vac Load Center 17B01
    - Engineered Safety Feature Transformer 11
    - Power Range Neutron Monitoring System

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

    - Division I Standby Service Water System Pump
    - High Pressure Core Spray Pump 1E22-C001
    - High Pressure Core Spray Valves 1E22-F001 and 1E22-F015
    - High Pressure Core Spray Valve 1E22-F012
    - Division I Low Pressure Core Spray Pump 1E21-C001
    - Division I Residual Heat Removal Pump
    - Emergency Diesel Generator 13 Ventilation
    - Emergency Pump Room Fan Cooler T51-B001-C
    - Division I Residual Heat Removal Heat Exchanger

.2 Results of Detailed Reviews for Components

.2.1 Division III Emergency Diesel Generator Output Circuit Breaker 152-1701

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 division III 4160 Vac emergency diesel generator output breaker 152-1701. 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:

- Schematics and control wiring diagrams of record for the breaker.
- Preventive maintenance procedures for the breaker.
- Vendor manual and specifications for the breaker.
- Load calculations of record and supporting documentation.
- Calculations of record for protection settings and alarms.
- Completion of last preventive maintenance work orders.
- Breaker control power circuit and ancillary supporting component and equipment.

During the inspection, the licensee was conducting an apparent cause evaluation on the recent failure of circuit breaker 152-1701 documented under Condition Reports CR-GGN-2012-07922 and CR-GGN-2012-07935. Upon completion of the apparent cause evaluation, the NRC will review this failure and document the review in NRC Inspection Report 05000416/2012005.

b. Findings

1. Preconditioning of 4160 Vac Circuit Breaker for As-Found Tests

Introduction. The team identified a Green non-cited violation of 10 CFR 50, Appendix B, Criterion XI, “Test Control,” involving the licensee’s failure to establish a test program which demonstrates that components will perform satisfactorily in service. Specifically, the licensee failed to record “as-found” test values prior to performing maintenance for 4160 Vac circuit breakers.

Description. The team reviewed six-year preventive maintenance procedures for 4160 Vac circuit breakers. During the review, the team identified that Procedure 07-S-12-41,“Inspection and Testing of ITE 5 KV Circuit Breakers,” Procedure 07-S-12-42, “Inspection and Testing of Westinghouse DHP 4.16KV Circuit Breakers,” and Procedure 07-S-12-61, “Inspection of GE Magna Blast Circuit Breakers,” directed maintenance personnel to clean, adjust, and manipulate the physical condition of 4160 Vac circuit breaker contacts, insulators, and other critical circuit breaker components before performing an “as-found” test to determine if the circuit breakers would have performed their intended design function.

For example, Procedure 07-S-12-61, “Inspection of GE Magna Blast Circuit Breakers,” Section 7.1, “Breaker Cleaning and Inspection,” directs maintenance personnel to clean and inspect the circuit breaker. In particular, Step 7.1.8, states, “Remove the interrupter and box barriers. Inspect the movable arcing contacts, stationary arcing contacts, movable primary contacts, and stationary primary contacts. If contacts are burned and pitted, file smooth with a contact file.” Step 7.1.8 is completed before any “as-found” tests are performed to verify the operability of the critical components of the circuit breaker, such as main contact resistance, main contact gap, and insulation resistance.

The team reviewed the data sheet resulting from the December 16, 2011, tests and preventative maintenance performed on 4160 Vac circuit breaker 152-1701 using Procedure 07-S-12-61. Those results show that maintenance personnel documented the same results for “as-found” and “as-left” for multiple tested parameters; therefore, the team determined that the procedure could mask existing conditions such as unacceptable contact resistance, setpoint drift, and mechanical binding. Additionally, the procedure resulted in the inability to verify past operability of circuit breaker 152-1701.

<u>Analysis</u>. The team determined that failure to establish a test program which ensures that test and maintenance procedures associated with safety-related 4160 Vac circuit breakers would perform satisfactorily in service was a performance deficiency. This finding was more than minor because, if left uncorrected, it would lead to a more significant safety concern. Specifically, the failure to perform “as-found” tests prior to performing maintenance in preventive maintenance procedures was a significant programmatic deficiency which could cause unacceptable conditions to go undetected. Using the Inspection Manual Chapter 0609, Appendix A, “The Significance Determination Process for Findings At-Power,” 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 safety function. This finding had a crosscutting aspect in the area of human performance, resources component, because the licensee failed to ensure that test and maintenance procedures were complete, accurate, and up-to-date to assure nuclear safety. [H.2(c)]

<u>Enforcement</u>. The team identified a Green non-cited violation of 10 CFR 50, Appendix B, Criterion XI, “Test Control,” which states, in part, “A test program shall be established to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents.” Contrary to the above, the licensee failed to establish a test program that assured that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service was identified and performed in accordance with written test procedures which incorporated the requirements and acceptance limits contained in applicable design documents. Specifically, prior to July 27, 2012, the licensee’s preventive maintenance Procedures 07-S-12-41, 07-S-12-42, and 07-S-12-61 failed to assure that the 4160 Vac circuit breakers would perform satisfactorily in service when the licensee performed maintenance prior to completing “as-found” tests to verify past operability of the circuit breakers. This finding was entered into the licensee’s corrective action program as Condition Reports CR-GGN- 2012-09035 and CR- GGN-2012-9103. Because this

finding is 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 NRC Enforcement Policy: NCV 05000416/2012008-01, "Preconditioning of 4160 Vac Circuit Breakers for As-Found Tests."

.2.2 Division III 125 Vdc Battery and Safety Bus 11DC

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 division III 125 Vdc battery and associated safety bus. The team also performed walkdowns and conducted interviews with engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- Calculations that established the basis for battery loading and sizing.
- Voltage drop calculations, short circuit calculations, and coordination studies.
- Results of the recent surveillance tests and maintenance activities to determine inclusion of vendor recommendations and industry standards.
- Separation criteria, configuration, and installation to confirm separation of safety-related and nonsafety-related loads.
- Visible material condition and configuration of the components.
- Calculations and vendor documents addressing required heat removal performance requirements during design and maximum ambient temperature conditions.
- Recent temperature data recorded in the division III switchgear and battery rooms.
- Evaluation of the potential impact of elevated temperatures on safety-related equipment located within the division III switchgear and battery rooms under accident conditions.

b. Findings

1. Failure to Establish a Testing Program for Safety-Related 125 Vdc Circuit Breakers

Introduction. The team identified a Green non-cited violation of 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," involving the licensee's failure to establish a test program which incorporates test requirements and acceptance limits contained in applicable design documents. Specifically, the licensee failed to establish a periodic test program for safety-related 125 Vdc molded case circuit breakers which incorporated the requirements of IEEE Standard 308, "Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations."

Description. Grand Gulf Nuclear Station Updated Safety Analysis Report, Section 8.3, "Onsite Power Systems," states, in part, that all Class 1E power systems conform to IEEE Standard 308. IEEE Standard 308 requires, in part, that testing shall be performed at scheduled intervals to: 1) Detect within practical limits the deterioration of the equipment toward an unacceptable condition, and 2) Demonstrate that standby equipment and other components that are not exercised during normal operation of the station are operable.

For a sample of division III 125 Vdc molded case circuit breakers associated with 125 Vdc distribution center 11DC, the team requested the preventive maintenance procedures for maintaining and periodically testing the circuit breakers. Additionally, the team requested the associated records from the last maintenance and testing performed for these breakers.

The sample selected by the team included the following division III breakers:

- 72-11C01 (bus supply breaker from battery - GE Type TFK)
- 72-11C03 (bus supply breaker from charger 1C3 - GE Type TFK)
- 72-11C11 (DG 13 field flash - GE Type TEB)
- 72-11C12 (4160 V switchgear bus 17AC control power - GE Type TEB)
- 72-11C13 (DG 13 engine control power - GE Type TEB)
- 72-11C14 (DG 13 protective relaying - GE Type TEB)

In response to the team's request, the licensee stated that testing of 125 Vdc molded case circuit breakers was not included in their preventive maintenance program. To address the team's concern, the licensee initiated CR-GGN-2012-09030 for the division III 125 Vdc molded case circuit breakers. The licensee subsequently initiated CR-GGN-2012-09175 to extend their evaluation to division I and II 125 Vdc molded case circuit breakers that support the division I and II engineered safety features and safe shutdown functions.

<u>Analysis</u>. The team determined that the failure to establish a testing program incorporating the requirements of IEEE 308 was a performance deficiency. The finding was more than minor, because if left uncorrected, it would lead to a more significant safety concern. Specifically, the failure to establish a testing program was a significant programmatic deficiency that would lead to missed opportunities to detect potential common cause failures from degradation of performance in more than one redundant safety division. Using the Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process for Findings At-Power," 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 safety function. This finding had a crosscutting aspect in the area of problem identification and resolution, corrective action program component; because the licensee failed to thoroughly evaluate problems such that resolutions address cause and extent of condition. Specifically, the licensee failed to thoroughly evaluate the extent of condition associated with a previously identified NRC violation involving the failure to test 480 Vac molded case circuit breakers identified during the 2009 component design basis inspection. [P.1(c)]

<u>Enforcement</u>. The team identified a Green non-cited violation of 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," which states, in part, "A test program shall be established to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is identified and performed in

accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents." Contrary to the above, the licensee failed to establish a test program that assured that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service was identified and performed in accordance with written test procedures which incorporated the requirements and acceptance limits contained in applicable design documents. Specifically, prior to July 27, 2012, the licensee failed to establish a test program for 125 Vdc safety-related molded case circuit breakers incorporating the requirements of IEEE 308, to ensure the breakers would not degrade and would perform satisfactorily in service. The finding was entered into the licensee's corrective action program as Condition Reports CR-GGN-2012-09030 and CR-GGN-2012-09175. Because this finding is 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 NRC Enforcement Policy: NCV 05000416/2012008-02, "Failure to Establish a Testing Program for Safety-Related 125 Vdc Circuit Breakers."

.2.3 Division I Standby Service Water System Pump

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 standby service water pump. The team also performed walkdowns and conducted interviews with engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- Work orders and corrective action program documents.
- System design criteria.
- Piping and instrumentation diagrams and structural drawings.
- Technical specifications.
- Standby service water loop A valve and pump operability test.
- Standby service water system operating and alarm response instructions.
- Plant operations manual for chemical additions to plant systems.
- Ultimate heat sink inventory calculations and assumptions.

b. Findings

1. Failure to Obtain NRC Approval for a Change to Credible Passive Failures in the Standby Service Water System

Introduction. The team identified a Severity Level IV non-cited violation of 10 CFR 50.59, "Changes, Tests, and Experiments," involving the licensee's failure to obtain a license amendment, pursuant to 10 CFR 50.90, prior to implementing a change to the standby service water system passive failure analysis. Specifically, the licensee changed the final safety analysis report (as updated) to limit credible post-accident, non-electrical passive failures in the standby service water system to pump or valve seal leakage without submitting or obtaining a license amendment.

Description. On July 12, 2012, while reviewing the Grand Gulf Nuclear Station Updated Final Safety Analysis Report, Chapter 9.2, “Water Systems,” the team identified a footnote that states: “Credible non-electrical passive failures post-accident are limited to pump or valve seal leakage. A piping failure concurrent with the accident is not considered credible as noted in subsection 9.2.1.6, References 2 and 3.” The team requested the document that approved this change. The licensee produced Change Notice 3758.

In Change Notice 3758, the licensee performed a 10 CFR 50.59 safety evaluation. In this evaluation, the licensee answered that there were no unreviewed safety questions; therefore, the licensee was not required to submit the change to the NRC for approval, and subsequently, the licensee implemented the change. This change modified the original final safety analysis report to include the footnote referenced above in addition to several mark-ups in Table 9.2-1, “Standby Service Water System Passive Failure Analysis,” which removed pipe ruptures, heat exchanger tube ruptures, or pipe fitting ruptures as credible passive failures.

Title 10 CFR 50.59, “Changes, Tests, and Experiments,” was revised and became effective on March 13, 2001. The NRC issued a Regulatory Issue Summary 2001-03, dated January 23, 2001, that stated, in part, that licensees may implement the revised rule at a time later than March 13, 2001. In a letter dated March 5, 2001, Entergy Operations, Inc. informed the NRC that Grand Gulf Nuclear Station would implement the revised rule on July 2, 2001, and those evaluations begun before July 2, 2001 would be processed and completed in accordance with the old rule. Since the licensee approved the 10 CFR 50.59 safety evaluation on August 18, 1987, the evaluation was performed under the requirements of the old rule.

However, the team determined that the licensee answered one of the questions in the safety evaluation incorrectly. Specifically, in Part III – “Unreviewed Safety Question” of the safety evaluation, the licensee responded “No” to Question 3, which states, “Increase the probability of a malfunction of equipment important to safety previously evaluated in the FSAR.” The team determined that the answer to the question was “Yes” because the change significantly relaxed the licensee’s licensing basis for credible passive failures in the accident analysis.

Because the licensee performed the safety evaluation under the old 10 CFR 50.59 regulations, the team also reviewed the change as it applies to the revised regulations. The team determined that the change resulted in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety. According to NEI 96-07, “Guidelines for 10 CFR 50.59 Implementation,” Revision 1, departures from the design, fabrication, construction, testing and performance standards as outlined in the General Design Criteria are not compatible with a "no more than minimal increase" standard. Specifically, the change was a departure from 10 CFR Part 50, Appendix A, Criterion 44, “Cooling Water,” which requires that the safety function of the standby service water system can be accomplished, assuming a single failure. Therefore, the team determined that prior NRC review and approval was required under the old and revised rule.

On July 19, 2012, the licensee entered this concern into their corrective action program as Condition Report CR-GGN-2012-09267.

Analysis. The team determined that the licensee's failure to receive prior NRC approval for changes in licensed activities regarding single passive failure criteria for the standby service water system was a performance deficiency. The performance deficiency was evaluated using traditional enforcement because the finding had the ability to impact the regulatory process. The performance deficiency was more than minor because there was a reasonable likelihood that the change would require NRC review and approval prior to implementation. In accordance with the NRC Enforcement Manual, risk insights from the Inspection Manual Chapter 0609, "Significance Determination Process," are used in determining the significance of 10 CFR 50.59 violations. Using the Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process for Findings At-Power," the team determined that the finding represented a loss of system safety function in that the standby service water system could not meet its 30-day mission time to provide decay heat removal. Therefore, a Detailed Risk Evaluation was necessary. In accordance with Manual Chapter 0609, Appendix A, Section 6, "Detailed Risk Evaluation," the senior reactor analyst evaluated the risk of the degraded condition that resulted from the finding. According to the *Risk Assessment of Operational Events Handbook, Volume 1 – Internal Events*, Section 4.1, "Mission Time Modeling," in most events, 24 hours is sufficient time to bring numerous resources to bear on core cooling. In some events, the choice is conservative and the analysis results are overestimates. Additionally, the analyst determined that Section 4.2 on increasing mission time was not applicable to the subject finding because the decrease in standby service water system water inventory would be obvious and there would be days to respond with makeup sources. Therefore, the analyst determined that the finding was of very low safety significance (Green) because, although the standby service water system could not provide 30 days of decay heat removal without operator action to provide makeup water to the system, it would have been able to complete its 24-hour risk significant mission time. Since the finding had very low safety significance, the finding was determined to be Severity Level IV, in accordance with the NRC Enforcement Policy. The finding does not have a crosscutting aspect because the most significant contributor to the finding does not reflect current licensee performance.

Enforcement. The team identified a Severity Level IV non-cited violation of 10 CFR 50.59, "Changes, Tests and Experiments" which states, in part, that "a licensee shall obtain a license amendment pursuant to Section 50.90 prior to implementing a proposed change, test, or experiment if this activity would result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the final safety analysis report (as updated)." Contrary to the above, the licensee failed to obtain a license amendment pursuant to Section 50.90 prior to implementing a proposed change that resulted in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety. Specifically, on August 18, 1987, the licensee implemented a change to the updated safety analysis report which limited credible passive failures in the standby service water system to pump and valve seal leakage without obtaining a license amendment. This finding was entered into the licensee's corrective action program as Condition Report CR-GGN-2012-09267. Because this finding was determined to be 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 NRC Enforcement Policy: NCV 05000416/2012008-03, "Failure to Obtain NRC Approval for a Change to Credible Passive Failures in the Standby Service Water System."

2. Failure to Promptly Enter an NRC Violation Regarding the Standby Service Water System into the Corrective Action Program

Introduction. The team identified a Green non-cited violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," involving the licensee's failure to promptly enter an NRC violation regarding the standby service water system into the corrective action program.

Description. On July 12, 2012, the inspection team identified a violation of 10 CFR 50.59, "Changes, Tests, and Experiments," regarding a change to credible post-accident, non-electrical passive failures in the standby service water system. Specifically, the change limited credible passive failures to pump and valve seal leakage. Pipe, pipe fitting, and heat exchanger tube ruptures were no longer deemed credible. At this time, the team informed the licensee of the violation and questioned the licensee whether or not the standby service water system and ultimate heat sink remained operable, given the single failure of a pipe, pipe fitting, or heat exchanger tube rupture.

The licensee then reviewed the change to determine whether or not they agreed with the 10 CFR 50.59 violation. On July 19, 2012, the licensee entered this condition into their corrective action program as Condition Report CR-GGN-2012-09267. Procedure EN-LI-102, "Corrective Action Process," provides examples of adverse conditions requiring initiation of a condition report in Attachment 9.2. Attachment 9.2 lists regulatory issues, potential or actual NRC violations, as adverse conditions. In addition, EN-LI-102 states that the condition is expected to be promptly documented in a condition report. Because the 10 CFR 50.59 violation constituted a regulatory issue, the team determined that the licensee was required to enter the condition promptly into their corrective action program on July 12, 2012.

Subsequently, on July 29, 2012, the licensee performed an operability evaluation regarding the single passive failure aspect of the violation and concluded that the standby service water system was unable to perform its specified safety function for its entire mission time without compensatory measures. In effect, the seven-day delay in documenting the condition delayed evaluation of the standby service water system's ability to withstand single failures and, ultimately, implementation of compensatory measures necessary for the standby service water system to perform its specified safety function.

Analysis. The team determined that the licensee's failure to promptly enter the NRC violation as a condition adverse to quality into the corrective action program was a performance deficiency. This finding was more than minor because it adversely affected the design control attribute of the Mitigating Systems Cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the licensee failed to promptly document a violation of 10 CFR 50.59, which delayed an operability evaluation that ultimately determined that compensatory measures were required to ensure that the standby service water system could perform its specified safety function for its entire mission time. Using the Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process for Findings At-Power," the team determined that the finding represented a loss of system safety function in that the standby service water system could not meet its 30-day mission time to provide decay heat removal. Therefore, a Detailed Risk Evaluation was necessary. In accordance with Manual Chapter 0609, Appendix A, Section 6, "Detailed Risk Evaluation," the senior reactor analyst evaluated the risk of the degraded condition that resulted from the finding. According to the *Risk Assessment of Operational Events Handbook, Volume 1 – Internal Events*, Section 4.1, "Mission Time Modeling," in most events, 24 hours is sufficient time to bring numerous resources to bear on core cooling. In some events, the choice is conservative and the analysis results are overestimates. Additionally, the analyst determined that Section 4.2 on increasing mission time was not applicable to the subject finding because the decrease in standby service water system water inventory would be obvious and there would be days to respond with makeup sources. Therefore, the analyst determined that the finding was of very low safety significance (Green) because, although the standby service water system could not provide 30 days of decay heat removal without operator action to provide makeup water to the system, it would have been able to complete its 24-hour risk significant mission time. This finding had a crosscutting aspect in the area of problem identification and resolution, corrective action program component, because the licensee failed to ensure that issues potentially impacting nuclear safety are promptly identified, fully evaluated, and that actions are taken to address safety issues, in a timely manner, commensurate with their safety significance. Specifically, the licensee did not implement a corrective action program with a low threshold for identifying issues completely, accurately, and in a timely manner commensurate with their safety significance. [P.1(a)]

<u>Enforcement</u>. The team identified a Green non-cited violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," which states, in part, that "Measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformance are promptly identified and corrected." Contrary to the above, the licensee failed to promptly identify a condition adverse to quality. Specifically, on July 12, 2012, the NRC informed the licensee of a violation of 10 CFR 50.59 requirements, but the licensee failed to promptly identify this as an adverse condition and enter this condition into their corrective action program until July 19, 2012. The finding was entered into the licensee's corrective action program as Condition Report CR-GGN-2012-10075. Because this finding is 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 NRC Enforcement Policy:
NCV 05000416/2012008-04, "Failure to Promptly Enter an NRC Violation Regarding the Standby Service Water System into the Corrective Action Program."

3. Failure to Follow Operability Determination Process Procedure

Introduction. The team identified a Green non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings" involving the licensee's failure to implement requirements of Procedure EN-OP-104, "Operability Determination Process".

Description. On July 12, 2012, the inspection team identified a violation of 10-CFR 50.59, "Changes, Tests, and Experiments," regarding a change to credible, post-accident, non-electrical single passive failures in the standby service water system. Specifically, the change limited credible single passive failures to pump and valve seal leakage. The Updated Final Safety Analysis Report no longer included pipe, pipe fitting, and heat exchanger tube ruptures as credible failures. On July 12, 2012, the team informed the licensee of the violation and inquired as to whether or not the standby service water system and ultimate heat sink remained operable, given the single passive failure of a pipe, pipe fitting, or heat exchanger tube rupture.

On July 19, 2012, the licensee agreed with the team's determination regarding the 10-CFR 50.59 violation and entered the condition into their corrective action program as Condition Report CR-GGN-2012-09267. Consequently, the licensee performed an immediate operability determination, using Procedure EN-OP-104, "Operability Determination Process," concluding that the standby service water system and ultimate heat sink were OPERABLE based on the justification that "No Degraded or Nonconforming Conditions exist per EN-OP-104, Revision 6, Attachment 9.1, Table 1." In addition, the "described condition does not render standby service water system inoperable, due to the low probability of a passive failure."

The team reviewed the initial operability determination and disagreed with the licensee's conclusion of OPERABLE. First, because the condition questioned the ability of a technical specification required system to meet the single failure criterion, the standby service water system was potentially in noncompliance with the requirements of 10 CFR Part 50, Appendix A, Criterion 44, "Cooling Water." For this condition, Attachment 9.1, Table 1, allows the following permissible classifications: OPERABLE-DEGRADED or NONCONFORMING, OPERABLE-OPERABILITY EVALAUATION, INOPERABLE, or INOPERABLE-OPERABILITY EVALUATION. Second, EN-OP-104 states that "it is not acceptable to use Probabilistic Risk Assessment for making operability determinations". The team requested that the licensee rescreen the immediate operability determination.

On July 24, 2012, the licensee screened the condition as OPERABLE-OPERABILITY EVALUATION "based on engineering input." OPERABLE-OPERABILITY EVALUATION is a condition where a technical specification structure, system, or component has a reasonable expectation of performing its specified safety function; however, a more thorough technical analysis is necessary to support the initial conclusion. Engineering input is technical information that can be used by the shift manager for operability determinations. Engineering judgment is a determination based on engineering principles, objective evidence, or available data that provide a reasonable expectation that the structure, system, or component will perform its normal and design function until a detailed analysis can be performed. Furthermore, the supporting basis for the reasonable expectation of operability should provide a high degree that the structure, system, or component remains operable.

The team reviewed the second operability determination. The team disagreed with the licensee's conclusion that the standby service water system remained in an OPERABLE-OPERABILITY EVALUATION classification because reasonable expectation of operability was not established. First, the licensee used engineering judgment to assume a reasonable expectation of operability. According to EN-OP-104, "if Engineering Judgment is used, a sound basis must be documented." A sound basis for reasonable expectation of operability was never documented. Second, the team calculated that the cooling water inventory margin in the standby service water system was less than 50 gallons per minute averaged over the mission time yet the leak detection capability of the standby service water system was 1,200 gallons per minute. Therefore, any undetected leak above 50 and below 1,200 gallons per minute would render the standby service water system incapable of performing its specified safety function.

On July 29, 2012, the licensee completed the final operability determination. The evaluation concluded that the standby service water system could not meet its specified safety function for a 30-day mission time without compensatory measures. Therefore, the licensee implemented compensatory measures to maintain operability and changed the permissible classification to OPERABLE-COMPENSATORY MEASURE, which is a condition where a technical specification structure, system, or component is operable but a degraded or nonconforming condition exists that requires compensatory measures. The licensee entered this condition into their corrective action program as Condition Report CR-GGN-2012-09356.

<u>Analysis</u>. The team determined that the failure to implement the requirements of the operability determination process procedure was a performance deficiency. The finding was more than minor because it adversely affected the equipment performance attribute of the Mitigating Systems Cornerstone to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the standby service water system was incapable of performing its specified safety function for the entire 30-day mission time without compensatory measures. Using the Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process for Findings At-Power," the team determined that the finding represented a loss of system safety function in that the standby service water system could not meet its 30-day mission time to provide decay heat removal. Therefore, a Detailed Risk Evaluation was necessary. In accordance with Manual Chapter 0609, Appendix A, Section 6, "Detailed Risk Evaluation," the senior reactor analyst evaluated the risk of the degraded condition that resulted from the finding. According to the *Risk Assessment of Operational Events Handbook, Volume 1 – Internal Events*, Section 4.1, "Mission Time Modeling," in most events, 24 hours is sufficient time to bring numerous resources to bear on core cooling. In some events, the choice is conservative and the analysis results are overestimates. Additionally, the analyst determined that Section 4.2 on increasing mission time was not applicable to the subject finding because the

decrease in standby service water system water inventory would be obvious and there would be days to respond with makeup sources. Therefore, the analyst determined that the finding was of very low safety significance (Green) because the standby service water system could have been able to complete its 24-hour risk significant mission time although it could not provide 30 days of decay heat removal without operator action to provide makeup water to the system. This finding had a crosscutting aspect in the area of human performance, decision making component, because the licensee did not make decisions that demonstrated that nuclear safety was an overriding priority. Specifically, the licensee did not make safety significant decisions using a systematic process to ensure safety is maintained. [H.1(a)]

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, that "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, the licensee failed to accomplish activities affecting quality in accordance with prescribed procedures. Specifically, from July 19, 2012, to July 29, 2012, the licensee failed to correctly evaluate the operability of the standby service water system with a degraded or nonconforming condition and failed to document a sound basis for a reasonable expectation of operability of the standby service water system as required by Procedure EN-OP-104, "Operability Determination Process." The finding was entered into the licensee's corrective action program as Condition Report CR-GGN-2012-09356. Because this finding is 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 NRC Enforcement Policy: NCV 05000416/2012008-05, "Failure to Follow Operability Determination Process Procedure."

.2.4 Division III 4160 Vac Engineered Safety Feature Switchgear Bus 17AC

a. Inspection Scope

The team reviewed the updated safety analysis report, design basis documents, the current system health report, calculations, maintenance and test procedures, and condition reports associated with the division III 4160 Vac engineered safety feature switchgear bus 17AC. The team also performed walkdowns and conducted interviews with engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- Maintenance history to verify the monitoring and correction of potential degradation.
- Calculations for electrical distribution system load flow/voltage drop, short-circuit, and electrical protection and coordination.
- Protective device settings and circuit breaker ratings to confirm adequate selective protection and coordination of connected equipment during worst-case short circuit conditions.

- Circuit breaker preventive maintenance, inspection, and testing procedures to confirm inclusion of relative industry operating experience and vendor recommendations.
- Results of completed preventive maintenance on 4160 Vac switchgear and breakers.
- Degraded voltage and loss of voltage relay protection scheme and circuit breaker control logics that initiate automatic bus transfers.
- NRC Information Notice 1993-091, "Misadjustment Between General Electric 4.16-KV Circuit Breakers and Their Associated Cubicles ," dated December 3, 1993

b. Findings

1. Failure to Incorporate Test and Inspection Requirements for 4160 Vac Circuit Breakers into Preventive Maintenance Procedures

Introduction. The team identified a Green non-cited violation of 10 CFR 50, Appendix B, Criterion XI, "Test Control," involving the licensee's failure to establish a test program which incorporated test requirements and acceptance limits contained in applicable design documents. Specifically, the licensee failed to incorporate minimum control voltage drop-out tests, reduced voltage tests, and inspection of auxiliary contacts in safety-related 4160 Vac circuit breaker preventive maintenance procedures.

Description. The licensee's 4160 Vac circuit breaker maintenance and testing program was established using Preventive Maintenance Basis Template, "EN-Switchgear-Medium Voltage – 1 KV to 7KV," Revision 3. This 4160 Vac preventive maintenance basis template establishes the cleaning, inspection, and testing program which incorporates requirements from vendor documents and Electrical Power Research Institute guideline TR-112814, "Reduced Voltage Testing of Low and Medium Voltage Breakers." The "Breaker – Detailed Inspection, Cleaning, and Testing" task lists the types of inspection and tests that should be incorporated into the 4160 Vac circuit breaker testing procedures. Listed in this section are requirements for minimum control voltage tests, reduced voltage tests, and measuring resistance and cleaning of relay contacts.

The team reviewed preventive maintenance procedures for the 4160 Vac circuit breakers used in the engineered safety feature electrical buses. During this review, the team identified that Procedure 07-S-12-41,"Inspection and Testing of ITE 5 KV Circuit Breakers," Procedure 07-S-12-42, "Inspection and Testing of Westinghouse DHP 4.16KV Circuit Breakers," and Procedure 07-S-12-61, "Inspection of GE Magna-Blast Circuit Breakers," did not incorporate testing or inspection of the minimum voltage drop-out settings, reduced voltage settings, and inspection and resistance measurement of auxiliary switch contact relays.

The team determined that the preventive maintenance Procedures 07-S-12-41, 07-S-12-42, and 07-S-12-61 did not incorporate required tests or inspections that would provide assurance that all testing required to demonstrate that structures, systems and components will perform satisfactorily in service.

Analysis. The team determined that the failure to incorporate required tests and inspections into preventive maintenance procedures for safety-related 4160 Vac circuit breakers was a performance deficiency. This finding was more than minor because, if left uncorrected, it would lead to a more significant safety concern. Specifically, the failure to incorporate the testing, cleaning, and inspection requirements into preventive maintenance procedures were a significant programmatic deficiency which could cause unacceptable conditions to go undetected. Using the Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process for Findings At-Power," 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 safety function. This finding had a crosscutting aspect in the area of problem identification and resolution, operating experience component, because the licensee failed to use operating experience information, including vendor recommendations and internally generated lessons learned, to support plant safety. Specifically, the licensee did not implement and institutionalize operating experience through changes to processes, procedures, equipment, and training programs. [P.2(b)]

Enforcement. The team identified a Green non-cited violation of 10 CFR 50, Appendix B, Criterion XI, "Test Control," which states, in part, "A test program shall be established to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design document." Contrary to the above, the licensee failed to establish a test program that assured that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service was identified and performed in accordance with written test procedures which incorporated the requirements and acceptance limits contained in applicable design document. Specifically, prior to July 27, 2012, the licensee's safety-related 4160 Vac circuit breaker preventive maintenance Procedures 07-S-12-41, 07-S-12-42, and 07-S-12-61 failed to incorporate inspection and test requirements for minimum voltage tests, reduced voltage tests, and inspection of auxiliary switch relay contacts as established in the licensee's circuit breaker maintenance program. This condition was entered into the licensee's corrective action program as Condition Reports CR-GGN 2012-08885 and CR-GGN-2012-09111. Because this finding is 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 NRC Enforcement Policy: NCV 05000416/2012008-06, "Failure to Incorporate Test and Inspection Requirements for 4160 Vac Circuit Breakers into Preventive Maintenance Procedures."

.2.5 Division III Emergency Diesel Generator 13

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 division III emergency diesel generator 13. The team also performed walkdowns and conducted interviews with engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team selectively reviewed:

- Component maintenance history and corrective action history to confirm the licensee was appropriately monitoring potential degradation.
- Calculations for the diesel generator loading, voltage, and frequency conditions, including load flow and voltage regulation.
- Control logic and circuits for the starting and loading of the diesel generator.
- The range of ambient temperature conditions and their basis for the diesel generator and electrical auxiliaries.
- The visible material condition and configuration of the components.
- The off-normal emergency procedure for back-feed of power from the Division III diesel generator to either the Division I or Division II 4160 V bus.

b. Findings

No findings of significance were identified.

.2.6 Division III 480 Vac Load Center 17B01

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 division III 480 Vac Load Center 17B01. 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:

- Vendor installation and maintenance manuals.
- Electrical distribution system load flow/voltage drop, short circuit, and electrical protection and coordination calculations.
- Protective device settings and circuit breaker ratings to confirm operation during worst-case short circuit conditions.
- Circuit breaker preventive maintenance inspection and testing procedures to determine adequacy relative to industry and vendor recommendations.

b. Findings

No findings of significance were identified.

.2.7 Engineered Safety Features Transformer 11

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 engineered safety features transformer 11. 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 tap settings, nameplate data, and protective relay settings, and loading requirements.
- Recently completed transformer preventive maintenance.
- Steady state loading calculation and protection.
- Metering and relay diagram and instrumentation.
- Relay protection, relay coordination, and short circuit calculations.
- Test performance records and the result of dissolved oil gas and Doble test analysis.

b. Findings

No findings of significance were identified.

.2.8 Power Range Neutron Monitoring System

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 power range neutron monitoring system. 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 selectively reviewed:

- The safety evaluation report to confirm that the installation of the system conformed to the safety evaluation report acceptance criteria.
- The engineering change package and implementing work orders.
- Fiber optic cable installation.
- Features provided for electromagnetic compatibility, physical separation, and independence.
- Precautions for electrostatic discharge and software configuration control.

- Site acceptance tests and condition reports initiated during site installation.

b. Findings

No findings of significance were identified.

.2.9 High Pressure Core Spray Pump 1E22-C001

a. Inspection Scope

The team reviewed the updated safety analysis report, the current system health report, selected drawings, maintenance and test procedures, and condition reports associated with the high pressure core spray pump 1E22-C001. The team also performed walkdowns and conducted interviews with engineering personnel to ensure the capability of this component perform its desired design basis function. Specifically, the team reviewed:

- General Electric design specification data sheets defining the system design requirements.
- Pump calculation addressing the available net positive suction head during system suction from the suppression pool and condensate storage tank at design temperature limits.
- Quarterly functional test procedures and test results used to monitor potential high pressure core spray pump degradation.
- Calculation addressing the environmental parameters limits of the high pressure core spray pump room.

b. Findings

No findings of significance were identified.

.2.10 High Pressure Core Spray Suppression Pool and Condensate Storage Tank Suction Valves 1E22-F001 and 1E22-F015

a. Inspection Scope

The team reviewed the updated safety analysis report, the current system health report, selected drawings, maintenance and test procedures, and condition reports associated with the high pressure core spray suppression pool and condensate storage tank suction valves 1E22-F001 and 1E22-F015. The team also performed walkdowns and conducted interviews with engineering personnel to ensure the capability of this component perform its desired design basis function. Specifically, the team reviewed:

- General Electric design specification data sheets defining the system design requirements.
- Quarterly valve test procedure and surveillance results as part of the inservice testing program.

- Accident analysis calculation of a postulated loss of offsite power with a concurrent lost of coolant accident resulting in a catastrophic failure of the condensate storage tank.
- Calculation addressing the environmental parameters limits of the high pressure core spray pump room.

b. Findings

No findings of significance were identified.

.2.11 High Pressure Core Spray Minimum Flow Valve 1E22-F012

a. Inspection Scope

The team reviewed the updated safety analysis report, the current system health report, selected drawings, maintenance and test procedures, and condition reports associated with the high pressure core spray minimum flow valve 1E22-F012. The team also performed walkdowns and conducted interviews with engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- General Electric design specification data sheets defining the system design requirements.
- Quarterly valve test procedure and surveillance results as part of the inservice testing program.
- Licensee's response to NRC Bulletin 88-04, "Potential Safety-Related Pump Loss."
- Engineering design change package addressing the increase in minimum rate of flow.
- Logic and wiring diagrams for minimum flow valve 1E22-E012.
- Vendor installation and maintenance manuals.

b. Findings

No findings of significance were identified.

.2.12 Low Pressure Core Spray Pump 1E21-C001

a. Inspection Scope

The team reviewed the updated safety analysis report, the current system health report, selected drawings, maintenance and test procedures, and condition reports associated with the low pressure core spray pump 1E21-C001. The team also performed walkdowns and conducted interviews with engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- Safety-related calculations addressing required low pressure core spray pump performance requirements during design basis accidents.
- Calculations addressing the uncertainties of the instruments used to verify pump performance during required technical specification surveillances.
- Surveillance procedures and test results used to monitor potential low pressure core spray pump degradation.
- Safety-related calculations and surveillance tests addressing the performance of associated low pressure core spray injection valve.
- Safety-related calculation determining the maximum differential pressure across the associated low pressure core spray injection valve.

b. Findings

No findings of significance were identified.

.2.13 Residual Heat Removal Pump 1E12-C002A

a. Inspection Scope

The team reviewed the updated safety analysis report, the current system health report, selected drawings, maintenance and test procedures, and condition reports associated with the residual heat removal pump 1E12-C002A. The team also performed walkdowns and conducted interviews with engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- Corrective action program documents and system health reports.
- System design criteria.
- Piping and instrumentation diagrams.
- Residual heat removal system operating instructions.
- Residual heat removal subsystem A quarterly functional tests.
- Technical specifications and bases document.
- Grand Gulf Nuclear Station response to NRC Information Notice 1987-10, "Potential for Water Hammer during Restart of Residual Heat Removal Pumps."

b. Findings

No findings of significance were identified.

.2.14 Emergency Diesel Generator 13 Ventilation

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 division III emergency diesel generator ventilation system. The team also performed walkdowns and conducted interviews with engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- Safety-related calculations addressing required heat removal performance requirements during design ambient temperature conditions.
- Safety-related calculations addressing required heat removal performance requirements during postulated maximum ambient temperature conditions.
- Temperature data recorded during extended emergency diesel generator operation.
- Ventilation fan flow data recorded after fan blade adjustments.

b. Findings

No findings of significance were identified.

.2.15 Emergency Pump Room Fan Cooler T51-B001-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 emergency pump room fan cooler T51B001-C. The team also performed walkdowns and conducted interviews with system and design engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- Corrective action program documents.
- Piping and instrumentation diagrams.
- System design criteria and health reports.
- Vendor documentation.

b. Findings

No findings of significance were identified.

.2.16 Division I Residual Heat Removal Heat Exchanger

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 residual heat removal heat exchanger. 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:

- Work orders and corrective action program documents.

- System design criteria and system health reports.
- Corrective action program reports to verify the monitoring and correction of potential degradation, operability evaluations and Root/Apparent Cause evaluations
- Piping and instrumentation diagrams.
- Residual heat removal heat exchanger plugged tube map.

b. Findings

No findings of significance were identified.

.3 Results of Reviews for Operating Experience:

.3.1 Inspection of NRC Information Notice 1987-10, "Potential for Water Hammer during Restart of Residual Heat Removal Pumps"

a. Inspection Scope:

The team reviewed the licensee's evaluation of Information Notice 1987-10 "Potential for Water Hammer during Restart of Residual Heat Removal Pumps" to verify that the review adequately addressed the industry operating experience. The team verified that the licensee's evaluation adequately addressed the issues in the Information Notice. The team verified that the licensee implemented changes to the system operating instructions based on recommendations given in the evaluations.

b. Findings:

No findings of significance were identified.

.3.2 Inspection of NRC Inspection of Information Notice 1993-091 "Misadjustment between General Electric 4.16 kV Circuit Breaker and their Associated Cubicles"

a. Inspection Scope

The team reviewed the licensee's evaluation of Information Notice 1993-091, "Misadjustment between General Electric 4.16-KV Circuit Breaker and their Associated Cubicles," to verify that the review adequately addressed the industry operating experience. The team verified that the licensee's evaluation adequately addressed the issues in the information notice. The team verified that the licensee assured that Procedure 07-S-12-61, "Inspection of GE Magna Blast Circuit Breakers," Revision 6 prevented the concerns addressed in the Information Notice.

b. Findings

No findings of significance were identified.

.3.3 Inspection of NRC Information Notice 2005-30, "Safe Shutdown Potentially Challenged by Unanalyzed Internal Flooding Events and Inadequate Design"

a. Inspection Scope

The team reviewed the licensee's evaluation of Information Notice 2005-30, "Safe Shutdown Potentially Challenged by Unanalyzed Internal Flooding Events and Inadequate Design," to verify that the review adequately addressed the industry operating experiences discussed in the information notice. The team reviewed the licensee's existing evaluation and performed independent reviews of plant areas to verify adequate protection from postulated internal flooding events.

b. Findings

1. Potential Internal Flooding Caused by Circulation Water System Failure

Introduction. The inspectors identified an unresolved item related to the licensee's evaluation of internal flooding events resulting from the postulated failure of circulating water system components in the turbine building. Specifically, the licensee's design basis flooding analyses were based on comparing the volume of the circulating water system to the volumes of the affected buildings and did not consider the effect of closed doors between the flood source in the Unit 1 turbine building, the canceled Unit 2 turbine building, and the radwaste building.

Description. The inspectors reviewed Calculation M6.3.051, "Circulating Water System-Calculate Revised Plant Flooding Elevations Due to Aux Cooling Tower," Revision B, to verify that the postulated failure of circulating water system components in the turbine building would not affect safety-related equipment required for achieving safe shutdown. This calculation assumes that the entire inventory of the circulating water system, 13.4 million gallons, is released into the Unit 1 turbine building due to a circulating water system failure and determines the resulting flood elevations. The calculation does not consider postulated flood flow rates; it is a steady state calculation based on the total circulating water system inventory being contained within the plant buildings. The calculation includes an assumption that the Unit 2 turbine building volume would be available to accommodate floodwater because "the passage/corridor between the Unit 1 and Unit 2 turbine buildings is not watertight." In addition, the maximum flood elevation is calculated based on the volume of the radwaste building being available to accommodate floodwater. The sliding door between the Unit 1 turbine building and the radwaste building is not addressed in the calculation. Based on these assumptions, the calculation determines that the bounding flood elevation is 104.0 feet, and that the flood will not reach safety-related equipment located in the control building at elevation 111 feet. The calculation also determines that the bounding flood elevation would reach 111.4 feet in the control building if the volume of the Unit 2 turbine building were not considered. These calculated flood elevations do not include the additional volume contributed by 23,200 gallon per minute makeup from the plant service water system to the circulating water system. The calculation concludes that operator action to stop the makeup flow within 70 minutes is acceptable due to the margin available in the calculation.

The inspectors questioned the assumptions of this calculation; especially the assumption that buildings connected by passageways that are "not watertight" would flood coincidently with each other. The inspectors asked if the expected leak rate between the Unit 1 turbine building, the Unit 2 turbine building, and the radwaste building through large sliding doors would be sufficient to limit the maximum flood elevation in the control building which is connected to the Unit 1 turbine building with a conventional door.

During the inspection, the licensee performed Calculation M6.3.051-001, "Circulating Water Systems – Calculate Revised Unit 1 Turbine Building and Unit 1 Control Building Flooding Elevations," Revision 0. This calculation was performed to address the inspectors' questions documented in Condition Report CR-GGN-2012-9424. This calculation was a transient analysis of the flood level considering the closed sliding doors between the Unit 1 turbine building and the Unit 2 turbine building and the Unit 1 turbine building and radwaste building. The calculation considered the gaps around the closed doors, and included the contribution of the makeup flow from the plant service water system to the circulating water system.

However, Calculation M6.3.051-001, revision 0 was based on a limited flowrate from an expansion boot failure in the circulating water system. The calculation used the methodology of NRC Branch Technical Position MEB 3-1 to predict the maximum flow from a failed circulating water system expansion joint. Applying the MEB 3-1 methodology to the 10-foot diameter expansion joint results in a postulated crack of 5-feet long and 1-inch wide. This crack results in a calculated flowrate of approximately 15,500 gpm. Based on this limited flowrate, the calculation determined that the maximum flood elevation would be approximately 104 feet.

The inspectors question the applicability of NRC Branch Technical Position MEB 3-1 to nonsafety-related expansion joints and asked the licensee to determine the maximum flood flowrate that would not exceed a flood elevation of 111 feet. In response to these questions, the licensee performed an informal analysis and determined that a flowrate of approximately 75,000 gpm or greater would result in exceeding a flood elevation in the Unit 1 turbine building, potentially communicating with the control building. The licensee also stated that they considered the application of the MEB 3-1 methodology to the expansion joints to be consistent with their licensing basis (UFSAR Section 3.6a.2.1) and that a "gross failure" of the expansion joint is highly unlikely since the expansion joint in reinforced with steel belts and leakage would be through a local defect. They also stated that the metal shield covering the expansion joints would serve to limit flow from the expansion joint failure, but did not provide the expected flowrate from a large failure of an expansion joint within the metal shield.

The inspectors performed a review of licensing basis documentation related to flooding resulting from failures of circulating water components and did not identify any specific value for the maximum flood flowrate or the maximum postulated failure size in an expansion joint. Grand Gulf Nuclear Station Update Safety Analysis Report, Section 10.4.5.3, describes the potential of the entire volume of the circulating water system flooding the Unit 1 turbine building, discusses a potential "gross failure" in the circulating water system, and describes the maximum circulating water system flowrate but does not specifically address the maximum postulated flood flowrate from a circulating water system failure.

The inspectors determined that design basis calculation M6.3.051, Revision B did not adequately verify that the postulated failure of circulating water system components in the turbine building would not affect safety-related equipment required for achieving safe shutdown. This steady state calculation did not consider the effects of closed doors on the maximum flood level in the control building. Calculation M6.3.051-001, Revision 0 was a transient analysis that did address the effects of the closed doors. However, this calculation was based on calculating a limited flood flowrate by applying the methodology of NRC Branch Technical Position MEB 3-1 to non safety-related circulating water system expansion joints. The inspectors were not able to determine if this methodology was consistent with the licensing basis during the period of the inspection. Resolution of this issue will require determining the maximum flowrate resulting from the postulated failure of a circulating water system component in the turbine building and verifying that the resulting flood elevation will not affect safety-related equipment required for achieving safe shutdown.

The inspectors have discussed this design and licensing basis issue with NRC staff in the Office of Nuclear Reactor Regulation. Due to complexity of establishing the appropriate design and licensing bases for this issue, this item is considered unresolved pending further NRC review to determine if a finding exists. This will be tracked as URI 05000416/2012008-07, "Internal Flooding Caused by Circulation Water System Failure."

.3.4 Inspection of Information Notice 2007-34 "Operating Experience Regarding Electric Circuit Breakers"

a. Inspection Scope

The team reviewed the licensee's evaluation of NRC Information Notice 2007-34, "Operating Experience Regarding Electrical Circuit Breakers," to verify that the review adequately addressed the industry operating experiences discussed in the information notice. The team verified that the licensee's evaluation adequately addressed the operating experience and inadequate maintenance practices identified in the Information Notice. The licensee initiated TEAR 2007-0667 to address the inadequate maintenance practices addressed in the Information Notice. The team verified that the licensee's corrective actions were adequate to prevent inadequate preventive maintenance from occurring.

b. Findings

No findings of significance were identified.

.3.5 Inspection of NRC Information Notice 2012-01 “Seismic Considerations - Principally Issues Involving Tanks”

a. Inspection Scope

The team reviewed the licensee’s response to Information Notice 2012-01, “Seismic Considerations – Principally Issues Involving Tanks,” to verify that the review adequately addressed the industry operating experience. The team verified that the licensee’s review, documented in Condition Reports CR-GGN-2012-03716 and CR-GGN-2011-07337, adequately addressed the issues in the Information Notice. The team verified that the licensee evaluated the human performance errors identified in the Information Notice, and had procedural steps in place that would prevent those errors from occurring.

b. Findings

No findings of significance were identified.

.4 Results of Reviews for Operator Actions

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.

a. Inspection Scope

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:

- Plant stabilization during station blackout conditions (Scenario)
- Alternate Power supply operation of safety relief valves (Job Performance Measure)
- Control room evacuation due to toxic gas (Scenario)
- Plant stabilization at remote shutdown panel (Job Performance Measure)

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems

The team reviewed actions requests associated with the selected components, operator actions and operating experience notifications. In addition, this report contains the following issue that has problem identification cross-cutting aspects.

4OA6 Meetings, Including Exit

On July 26, 2012, the team leader presented the preliminary inspection results to Mr. M. Perito, Vice President, and other members of the licensee’s staff. On September 10, 2012, the team leader conducted a telephonic final exit meeting with J. Browning, General Plant Manager Operations, and other members of the licensee's staff. The licensee acknowledged the findings during each meeting. While some proprietary information was reviewed during this inspection, no proprietary information was included in this report.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel

M. Bacon, Superintendent
J. Browning, General Manager, Plant Operations
D. Chipley, Electrical Design Engineering
J. Edwards, Site Representative, South Mississippi Electric
J. Giles, Manager, Training
J. Hixson, Electrical Design Engineering
D. Hollis, Electrical Design Engineering
K. Howard, Manager, Projects
M. Humphries, Programs Engineer, Circuit Breakers, Relays, and Motors
D. Jones, Manager, Design Engineering
C. Loyd, Supervisor, System Engineering
J. Miller, Manager, Operations
J. Nadeau, Manager, Corrective Actions & Assessment
M. Novogoratz, System Engineer, PRNMS
C. Perino, Manager, Licensing
M. Perito, Site Vice President, Operations
G. Phillips, Supervisor, Design Engineering, Instrumentation & Control
A. Pittman, PRA Engineer, Fuels and Analysis
M. Richey, Director, Nuclear Safety Assurance
M. Runion, Manager, Maintenance
A. Sayre, System Engineer, 125 VDC system
R. Scarbrough, Licensing Specialist, Licensing
J. Seiter, Senior Licensing Specialist, Licensing
R. Sumners, System Engineer, Diesel Generator
T. Tankersly, Manager, Quality Assurance
T. Thurmon, Supervisor, Design Engineering, Mechanical
R. Turcotte, Superintendent, Security
D. Wiles, Director, Engineering
C. Williams, Supervisor, Design Engineering, Electrical

NRC personnel

D. Loveless, Senior Reactor Analyst
S. M. Wong, Senior Reactor Analyst
B. Rice, Resident Inspector
R. Smith, Senior Resident Inspector

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000416/2012008-01	NCV	Preconditioning of 4160 Vac Circuit Breakers for As-Found Tests (1R21.2.1)
05000416/2012008-02	NCV	Failure to Establish a Testing Program for Safety-Related 125 Vdc Circuit Breakers (1R21.2.2)
05000416/2012008-03	NCV	Failure to Obtain NRC Approval for a Change to Credible Passive Failures in the Standby Service Water System (1R21.2.3)
05000416/2012008-04	NCV	Failure to Promptly Enter an NRC Violation Regarding the Standby Service Water System into the Corrective Action Program (1R21.2.3)
05000416/2012008-05	NCV	Failure to Follow Operability Determination Process Procedure (1R21.2.3)
05000416/2012008-06	NCV	Failure to Incorporate Test and Inspection Requirements for 4160 Vac Circuit Breakers into Preventive Maintenance Procedures (1R21.2.4)

Opened

05000416/2012008-07	URI	Potential Internal Flooding Caused by Circulation Water System Failure (1R21.3.3)

LIST OF DOCUMENTS REVIEWED

Calculations

NUMBER	TITLE	REVISION
36	Bechtel Calculation: Containment and Auxiliary Building Electrical Heat Output, Sheet 65	0
C-C956.0	SSW Basin Perforated Plate for Basin Sump	0
EC-01E51-92004	Selection and Sizing of thermal overload Relays for 480 Volt Class 1E Motor Operated Valves 01E51F064-A and 01E51F063-B	0
EC-Q1111-90028	AC Electrical Power System Calculation	6
EC-Q1111-90028	AC Electrical Power System Calculation	6

Calculations

NUMBER	TITLE	REVISION
EC-Q1111-92002	Evaluation of Safety-Related Electrical Equipment in Various Rooms with Elevated Post-LOCA Temperatures	2
EC-Q1111-93001	Control Building Electrical Heat Load Calculation	5
EC-Q1111-93001	Calculation Sheet 133	1
EC-Q1L21-90018	125 VDC Division III Battery Short Circuit Evaluation	3
EC-Q1L21-90020	Sizing of 125 VDC Battery C and Associated Battery Charger	1
EC-Q1L21-90023	Division III 125 VDC Class 1E Voltage Drop Study	2
EC-Q1L21-91018	Division III 125 VDC Class 1E Coordination Study	1
EC-Q1L21-95003	Evaluation of Division I, II, III Direct Current Bus Ground Detection Circuit	0
EC-Q1R20-91042	Div. III 480/120 VAC class IE CPT circuit Voltage Drop study	0
GGNS-89-0028	Grand Gulf Nuclear Station Engineering Report on Functionality under High Ambient Conditions of Auxiliary Building ESF Switchgear Room Equipment Important to Safety	2
GGNS-NE-11-00005	Engineering Report: GGNS EPU Ultimate Heat Sink Calculation Assumptions	0
JC-Q1E21-N651-2	Instrument Loop Uncertainty and Setpoint Determination for System E21 Loop N651	1
JC-Q1E21-N652-1	LPCS Pump Discharge Pressure – High Tech Spec Setpoint	0
JC-Q1P81-90027	Division III Loss of Voltage Setpoint Validation (T/S 3.3.8.1)	1
M-3.8.035	HPCS DG Room Heating and Ventilation	1
M-3.8.035 – Supplement 2	HPCS DG Room Heating and Ventilation	0
M-3.8.036	Standby Diesel Generator Room Heating and Ventilation	A
M-3.8.036 – Supplement 1	Standby Diesel Generator Room Heating and Ventilation System	0

Calculations

NUMBER	TITLE	REVISION
M6.3.043	Circulation Water System – Calculate Water Volume of Circulation Water System	C
M6.3.051	Circulation Water Systems – Calculate Revised Flooding Elevations Due to the Aux Cooling Tower	B
M6.3.051-001	Circulation Water Systems – Calculate Revised Unit 1 Turbine Building and Unit 1 Control Building Flooding Elevations	0
MC-Q1111-84016	ECCS Pump Surveillance Criteria	4
MC-Q1111-91132	Minimum Stem Thrust Required For Motor Operated Gate and Globe Valves	16
MC-Q1E21-93042	Maximum Expected Differential Pressure for Valves in Low Pressure Core Spray System	0
MC-Q1E22-010	HPCS and RCIC System Performance with Regard to CST and Suppression Pool Suction for Level Transmitters E22N054C&G and E51N035A&E	2
MC-Q1E22-010	HPCS and RCIC System Performance with Regard to CST and Suppression Pool Suction for Level Transmitters E22N054C&G and E51N035A&E	3
MC-Q1E22-91124	NPHS Calculation – HPCS Pump (Q1E22C001)	1
MC-Q1P41-03016	Standby Service Water Maximum Allowable Post-LOCA System Leakage	0
MC-Q1P41-11001	GGNS Standby Service Water Ultimate Heat Sink Thirty Day Performance at EPU	0
MC-Q1Z77-92001	Safeguards Switchgear & Battery Room Cooling & Heating Requirement	3
MC-Q1Z77-92001 – Supplement 1	Safeguards Switchgear & Battery Room Cooling & Heating Requirement	0
PR26	Relay Setting, ESF Secondary Breaker	0

Procedures

NUMBER	TITLE	REVISION/DATE
02-S-01-31	Operations Section Procedure – Control Room Rounds	29
02-S-01-35	Operations Section Procedure – Outside Rounds	67

Procedures

NUMBER	TITLE	REVISION/DATE
04-1-01-E12-1	System Operating Instruction – Residual Heat Removal System	141
04-1-01-P41-1	System Operating Instruction - Standby Service Water System	136
04-1-01-R21-1	System Operating Instruction - Load Shedding And Sequencing System	105
04-1-01-R21-17	System Operating Instruction ESF Bus 17AC	10
04-1-02-1H13-P870-1A-E1	Alarm Response Instruction – SSW Loop A Leak Hi	119
04-1-02-1H13-P870-5A-F2	Alarm Response Instruction – SSW Loop C Leak Hi	100
04-1-02-1H13-P870-7A-E1	Alarm Response Instruction – SSW Loop B Leak Hi	119
04-S-02-SH13-P807	Alarm Response Instruction Panel Sh13-P807	30
04-S-02-SH13-P807-030-4A-E6	Alarm Response Instructions for ESF XFRM 11 Trouble	22
04-S-02-Sh13-P808	Alarm Response Instruction Panel Sh13-P808	12
05-1-02-II-1	Off-Normal Event Procedure, Inadequate Decay Heat Removal	35
05-1-02-I-4	Off-Normal Event Procedure, Loss of AC Power Safety-Related	42
05-1-02-II-1	Off-Normal Event Procedure, Shutdown From The Remote Shutdown Panel	38
05-S-01-EP-2	Emergency Procedure, RPV Control	43
05-S-01-EP-3	Emergency Procedure, Containment Control	28
06-0P-1 P42-Q-0001	Surveillance Procedure, CCW Fuel Pool Heat Exchanger Valve Test	107
06-EL-1L11-Q-0001	Surveillance Procedure, 125 Volt Battery Bank All Cell Check	105
06-EL-1L11-R-0001	Surveillance Procedure, 125 Volt Battery Bank Physical Condition Check	102

Procedures

NUMBER	TITLE	REVISION/DATE
06-EL-1L11-W-0001	Surveillance Procedure, 125 Volt Battery Bank Pilot Cell Check	104
06-EL-1L21-O-0001	Surveillance Procedure, 125 Volt Battery Bank Performance Discharge Test	105
06-EL-IR21-M-00001	4.16KV Degraded Voltage Functional Test & Calibration Division 1 Bus 15 AA	104
06-IC-1E51-R-0002	Surveillance Procedure, Condensate Storage Tank (RCIC), Low Level Calibration	105
06-ME-1P41-R-0001	Safety Position Verification of P41 Check Valves	115
06-OP-122E-Q-002	Surveillance Procedure HPCS Quarterly Valve Test	109
06-OP-1E12-Q-0023	Surveillance Procedure – LPCI/RHR Subsystem A Quarterly Functional Test	124
06-OP-1E21-C-0004	LPCS Cold Shutdown Valve Test	108
06-OP-1E21-M-0001	LPCS Monthly Functional Test	105
06-OP-1E22-Q-005	Surveillance Procedure HPCS Quarterly Functional Test	120
06-OP-1P41-Q-0004	Surveillance Procedure – Standby Service Water Loop A Valve and Pump Operability Test	120
06-OP-1P81-M-0002	Surveillance Procedure, HPCS Diesel Generator 13 Functional Test	125
06-OP-1P81-M-0002	HPCS Diesel Generator 13 Functional Test	125
06-OP-1P81-R-0001	Surveillance Procedure, HPCS Diesel Generator 18-Month Functional Test	121
06-OP-1R21-R-0001	Surveillance Procedure ESF Division 3 Power Supply Functional Test	101
06-OPIP41-Q-0006-01	HPCS Service Water Pump & Valve Operation Test	1
07-1-22-R20-16BB2	Preventive Maintenance Instruction Load Center 16BB2 Relay Functional Test	1

Procedures

NUMBER	TITLE	REVISION/DATE
07-S-12-10	Calibration Check of GE AC Under-voltage Relays	7
07-S-12-120	Inspection and Cleaning of 4160 Volts and 6900 Volt Switchgear	4
07-S-12-13	General Maintenance Instruction Calibration Checks of GE Type HAA Auxiliary Relays	1
07-S-12-145	General Maintenance Instruction ITE 5HK350 4.16 KV Breaker Overhaul Instructions	1
07-S-12-149	General Maintenance Instruction Westinghouse 4.16 KV Breaker Overhaul Instructions	1
07-S-12-150	General Maintenance Instruction General Electric AM 4.16 KV Breaker Overhaul Instructions	0
07-S-12-29	General maintenance Instruction Calibration Checks of GE Type 1JF51A Over-Frequency Relays	3
07-S-12-41	Inspection and Testing of Westinghouse DHP 4.16K.V. Circuit Breakers	2
07-S-12-42	Inspection and Testing of Westinghouse ITE 5KV Power Circuit Breakers	5
07-S-12-61	Inspection of GE Magna Blast Circuit Breakers	6
08-S-03-14	Plant Operations Manual – Chemical Additions to Plant Systems	25
08-S-04-120	Plant Operations Manual – Chemistry Evolutions at Standby Service Water	12
10-S-01-1	Emergency Plan Procedure – Activation Of The Emergency Plan	121
EN-LI-102	Corrective Action Process	19
EN-LI-108	Event Notification and Reporting	5
EN-MA-118	Foreign Material Exclusion	9
EN-OP-104	Operability Determination Process	6
EN-OP-111	Operational Decision-Making Issue (ODMI) Process	9
GEK-83280	Operation and Maintenance Instructions – Residual Heat Removal System Heat Exchangers (E12-B001)	May 1981

Procedures

NUMBER	TITLE	REVISION/DATE
SEP-GGNS-IST-1	GGNS Inservice Testing Bases Document	0
SEP-GGNS-IST-2	GGNS Inservice Testing Plan	0

Drawings

NUMBER	TITLE	REVISION
3-115 419	Solenoid Valve Diagram	0
762E445	High Pressure Core Spray	7
B57746-4	Powell Electric Manufacturing Co. Drawing – HPCS MCC, Grand Gulf Units 1 & 2	2
C-1736D	Units 1 & 2 – SSW Cooling Tower Basin Miscellaneous Steel Plans, Selections, and Details	3
C-1760A	Unit 1 Standby Service Water Supply & Return Lines Plan & Profile	2
E-1719	Raceway Plan Diesel Generator Building, El. 133’ 0” Area 12 Unit 1	8
E-0001	Main One-Line Diagram	48
E-0013	One Line Meter & Relay Diagram, Aux. Electric. Dist. Sys. & Bus 19UD	20
E-0020	One Line Meter & Relay Diagram, 125V DC Buses 11DG & 21DG	20
E-0023	One Line Diagram 125V DC 34.5 Switch Yard PNLs IPR02 & 2PR02 Unit 1 & 2	1
E-0660	Site Raceway Plans, Unit 1&2	31
E-0662	Enlarged Site Raceway Plans, Unit 1&2	17
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Design Basis Documents

NUMBER	TITLE	REVISION
GE 22A3131AC	High Pressure Core Spray System Design Spec Data Sheet	11
GE22A3131	High Pressure Core Spray System Design Specifications	5
SDC-07	480 Volt Load Center (R20) and Transformer System (R20)	0
SDC-10	ESF Division III Power Distribution System (R11 and R21)	0
SDC-16	Load Shedding and Sequencing System	0
SDC-C51	System Design Criteria, Neutron Monitoring System	0
SDC-E12	Residual Heat Removal System	3
SDC-E21	Low Pressure Core Spray	1
SDC-E22	High Pressure Core Spray System (E22)	3
SDC-M23	Containment and Drywell Personnel Airlock	0
SDC-P41	Standby Service Water System	3
SDC-P75	Standby Diesel Generator System (P75)	1
SDC-P81	HPCS Diesel Generator System	1
SDC-T46	ESP Electrical Switchgear Rooms Cooling System	0
SDC-T46	ESF Electrical Switchgear Rooms Cooling System	0
SDC-T51	Emergency Pump Room Ventilation System	0
SDC-X77	Diesel Generator Building Ventilation System	0
SDC-Z77	Safeguard Switchgear and Battery Rooms Ventilation System	1

Engineering Changes

NUMBER	TITLE	REVISION/DATE
EC 2437	EVALUATION TO STORE TWO SRV CARTS IN AUXILIARY BUILDING,	0
EC 34994	Pressure Relief Valves 1P41F299A and 1P41F299B	0

Engineering Changes

NUMBER	TITLE	REVISION/DATE
EC 38959	Internal Flooding	0
EC 21999	Engineering Change: upgrade of analog neutron monitoring system with digital power range neutron monitoring system	0
ECT 21999-01	Engineering Change Test: Power Range Neutron Monitoring System Modification Test [performed]	January 23, 2012
ECT 21999-01, Attachment B6	Engineering Change Test: Power Range Neutron Monitoring System Modification Test, Attachment B6, 2 out of 4 Voter Logic Functional Test [performed]	April 23, 2012

Condition Reports (CR-GGN-…)

1999-00386	1999-00433	1999-00481	2000-01349	2000-01418
2001-00066	2003-01580	2008-01201	2008-01201	2008-01804
2008-02829	2008-04990	2008-06400	2009-00868	2009-01035
2009-01100	2009-01452	2009-02013	2009-02049	2009-02100
2009-02364	2009-02585	2009-02787	2009-02848	2009-03458
2009-03459	2009-03462	2009-04408	2009-04892	2009-05527
2009-05678	2009-06689	2010-00162	2010-00572	2010-00641
2010-00679	2010-00684	2010-01333	2010-01338	2010-01344
2010-01366	2010-01381	2010-01381	2010-01909	2010-02841
2010-03371	2010-03650	2010-04802	2010-05546	2010-06042
2010-06064	2010-06351	2010-06785	2010-07351	2011-00070
2011-00070	2011-00734	2011-00758	2011-00765	2011-00771
2011-00930	2011-01773	2011-01879	2011-01879	2011-01901
2011-01901	2011-01927	2011-02008	2011-03471	2011-03820
2011-04879	2011-05604	2011-07101	2011-07337	2011-07337
2011-08224	2011-08445	2011-08591	2011-08623	2011-08642
2011-08720	2011-08728	2011-08733	2011-08990	2011-08990

Condition Reports (CR-GGN-…)

2011-09081	2011-09095	2011-09161	2011-09170	2011-09310
2012-00303	2012-00471	2012-00653	2012-00682	2012-00731
2012-03716	2012-03763	2012-05501	2012-05666	2012-06264
2012-06469	2012-06738	2012-07829	2012-07922	2012-07935
2012-08224	2012-08225	2012-08258	2012-08349	2012-08351
2012-08406	2012-08610	2012-08638	2012-08742	2012-09267
LO-NOE-2007-00398				

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

2012-08670	2012-08672	2012-08673	2012-08674	2012-08675
2012-08680	2012-08681	2012-08682	2012-08683	2012-08704
2012-08708	2012-08709	2012-08720	2012-08758	2012-08870
2012-08885	2012-08935	2012-09030	2012-09035	2012-09103
2012-09111	2012-09112	2012-09172	2012-09175	2012-09194
2012-09207	2012-09267	2012-09330	2012-09356	2012-09380
2012-09419	2012-09421	2012-09424	2012-10075	2012-10076
CR-HQN-2012-00680				

Work Orders

136453	156406	179329	179330	230701
236781	243472	248578	258426	273496
274805	277014	277910	278617	283659
287609 01	289867	295355	296928	299001
299044	299990	300143	300719	304361
317205	317205	5220062	5229229	50988455
51025613	51046998	51084808	51670809	51697430
52023431	52028498	52038284	52189709	52205957

Work Orders

52218509	52222813	52227318	52242082	52243286
52250248	52266778	52274073	52284416 01	52284541
52289442	52289442 01	52291688	52304041	52311817
52311818	52311819	52311820	52316055	52334072
52342884	52350295	52352745	52361897	52369928
52381737	52387998	52387999	52397683	52399364
52399868	52399869			

Miscellaneous

NUMBER	TITLE	REVISION/DATE
	Open Loop Strategic Plan	2
	OAS Document Review Summary Sheet: IEN 87-010 - Potential for Water Hammer during Restart of RHR Pumps	January 30, 1990
	Memo To: C.W. Angle; From: C.R. Hutchinson; Subject: IEN 87-10 (Unit 1); Ref: NPE-OAS 88/303, 88/309; PMI: 89/1093	August 9, 1989
	Memo To: C.W. Angle, Manager Operations Analysis Section; From: C.R. Hutchinson, GGNS General Manager; Subject: NPE-OAS, SER 55-83; Ref: (1) IEN 87-10, (2) SER 55-83, (3) NPE-OAS 88/303, (4) NPE-OAS 90/011; PMI: 90/02514	July 17, 1990
	Plugged Tube Map – RHR “A” Heat Exchanger	May, 2012
	Issuance of Amendment No. 5 to Facility Operating License NPF-29 Grand Gulf Nuclear Station, Unit No. 1	October 12, 1985
	NRC Letter, “Requests for Additional Information for the Review of the Grand Gulf Nuclear Station License Renewal Application (TAC No. ME7493)”	April 26, 2012
	Response to Request for Additional Information (RAI) dated April 26, 2012, Grand Gulf Nuclear Station, Unit 1, Docket No. 50-416, License No. NPF-29	May 25, 2012

Miscellaneous

NUMBER	TITLE	REVISION/DATE
	NRC Letter, “Issuance Of Amendment No. 69 To Facility Operating License No. NPF-29 - Grand Gulf Nuclear Station, Unit 1, Regarding General Requirements in Section 3.0 and 4.0 of the Technical Specifications (TAC No. 69184)”	August 14, 1990
	Process Applicability Determination – EC-25649 – SSW UHS Siphon Line Extension Mod	0
	Nuclear Licensing FSAR Change Notice No. 3758	August 18, 1987
	GG CDBI 2012 Time-critical Op Actions.xlsx	
	E21 HPCS/Drywell System Walkdown	April 19, 2012
	Letter, D. N. Grace (BWROG) to NRC, dated June 29, 1988, Response to NRC Bulletin 88-04, “Potential Safety-Related Pump Loss”	August 9, 1988
	Grand Gulf PSA, Event R21-FO-HE-XTIE-A, Failure to Cross-Tie Division III Diesel to Division I or II DG	July 12, 2012
	System Health Report, Division III Emergency Diesel Generator	Q1 2012
	System Health Report, Division III Emergency Diesel Generator	Q3 2011
	System Health Report, L11 Division III ESF 125 Vdc Battery	Q1 2012
	System Health Report, L11 Division III ESF 125 Vdc Battery	Q3 2011
04-1-01-M23-1	System Operating Instruction Drywell/Containment Airlock System	10
06-OP-1000-D0001	Switchgear Battery Room Temperature Data Sheet	June 26, 2012
21A3598	General Electric Transformer Specification, Sheet 4 (4.6 – Environmental Conditions)	0
21A9236	General Electric Specification, Engine-Generator for High Pressure Core Spray System	5
21A9236AN	General Electric Specification, Engine-Generator for High Pressure Core Spray System	2
460000156	Technical Manual For Vertical HPCS Pump	April 6, 1994

Miscellaneous

NUMBER	TITLE	REVISION/DATE
460000161 (IB-376)	Elma Cast Coil Power transformers Installation, Maintenance, Operating and Storage Instructions	May 25, 1977
460000503	Instruction Manual for Joy Nuclear Non-Containment Axivane Fan & Nuclear Centrifugal Fans	300
D-376	C&D Charter Power Systems Discharge Characteristic Curve D-376	2
DCP 91/107	HPCS Min-Flow Increase	0
E21	Low Pressure Core Spray - System Health Report	Q1-2012
EN-LI-119	Apparent Cause Evaluation for CR-GGN-2008-6400	7
EPRI 1000014	Circuit Breaker Maintenance Programmatic Consideration	December 2000
EPRI TR-112783	Circuit Breaker Timing and Travel Analysis	May 1999
EPRI TR-112814	Reduced Control Voltage Testing of Low and medium Voltage Circuit Breakers	July 1999
ER No. 96/0044	NPE Concurrence is required for MNCR 0049-96 “Accept-As-Is” Disposition for check valve seat leakage	0
ER-GG-1999-217	Replace & Respan Transmitters 1E22N054C&G and 1E51N035A&E, Respan and Change Set Points	0
ESF 21 EMI 10-08-11	Doble Test for ESF Transformer ESF21	July 11, 2012
GGNS-92-0002	Evaluation of Safety-Related Electrical Equipment in Various Rooms with Elevated Post-LOCA Temperatures	2
GGNS-94-0054	IPEEE Summary Report	1
GGNS-97-0011	Auxiliary Building Doors and Line Break Analysis	0
GGNS-E-100.0	Environmental Parameter for GGNS	5
GIN 2008-055	E-mail Subject: EC4638 – Resolution of Spurious CST Suction Swamps	February 11, 2008
GNRO-2010/00071	Supplemental Information, License Amendment Request, Extended Power Uprate, Grand Gulf Nuclear Station, Unit 1, Docket No. 50-416, License No. NPF-29	November 18, 2010

Miscellaneous

NUMBER	TITLE	REVISION/DATE
GNRO-2010-00010	Responses to NRC Requests for Additional Information Pertaining to License Amendment Request for Power Range Neutron Monitoring System (TAC No. ME2531)	February 8, 2010
GNRO-2010-00070	Response to NRC Request for Additional Information Pertaining to License Amendment Request for Power Range Neutron Monitoring System (TAC No. ME2531)	December 14, 2010
GSMS-RO-EP050	STATION BLACKOUT w/DIV III D/G INOP	76
GSMS-RO-ON051	Toxic Gas in the Control Room / Shutdown From the Remote Shutdown Panel / Bus 15AA Lockout	0
Job No. 9645-001	Interoffice Memorandum: SSW Basin Platform	December 22, 1980
LBDCR-2010-027	LBDCR for upgrade of analog neutron monitoring system with digital power range neutron monitoring system [includes SER]	April 19, 2012
LO-NOE-2005-00376	IN 2005-30 Evaluation	November 23, 2005
MNCR 0049-96	Material Nonconformance Report (MNCR) 0049-96	March 25, 1996
N/A	IST Surveillance Test Data for the E21 System Pump	2009-2012
NEDO 10905	Licensing Topical Report: High Pressure Core Spray System Power Supply Unit, Section 2.6	May 1973
PMOS	PM Basis Template – EN-Switchgear-Medium Voltage – 1KV to 7KV	3
QA-8-2011-GGNS-1	Quality Assurance Audit Report	0
Receipt 31497	QC Receipt Inspection Data, GGNS Purchase Order 10329464, C&D Technologies Inc.	December 27, 2011
SEP-GGNS-IST-1	GGNS Inservice Testing Bases Document	1
SERI-88-0018	Engineering Report for Pump Minimum Flow Adequacy per NRC Bulletin 88-04	0
Supplement 20	GGNS AC Electric Power System	April 13, 2012
Supplement 29	GGNS High Pressure Core Spray	April 13, 2012

Miscellaneous

NUMBER	TITLE	REVISION/DATE
VM 460000158	Vendor Instruction Manual for General Electric Company HPCS MCC	September 25, 1997
VM 460000161	Elma Engineering 1B-376, Elma Cast Coil Power Transformers	A
VM 460000163	Vendor Instruction Manual for General Electric Company SWGR	January 31, 2001
VM 460000456	Instructions 25000/28000KVA, 3 phase 60 Hertz, 55/65 degree C Outdoor Power Transformer	301
VM 460000469	ITE Indoor Secondary Unit Substation	June 16, 2005
X77	Diesel Generator Building Vent - System Health Report	Q1-2012