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

December 1, 2017

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

SUBJECT: GRAND GULF NUCLEAR STATION – NRC DESIGN BASES ASSURANCE INSPECTION REPORT 05000416/2017007

Dear Mr. Larson:

On October 18, 2017, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Grand Gulf Nuclear Station (GGNS) and discussed the results of this inspection with you and other members of your staff. The results of this inspection are documented in the enclosed report.

Nuclear Regulatory Commission inspectors documented five findings of very low safety significance (Green) and one of Severity Level IV in this report. These findings involved violations of NRC requirements. The NRC is treating these violations as non-cited violations (NCVs) consistent with Section 2.3.2.a of the Enforcement Policy.

Further, inspectors documented two licensee-identified violations, which were determined to be of very low safety significance (Green) in this report. The NRC is treating these violations as NCVs consistent with Section 2.3.2.a of the Enforcement Policy.

Further, the inspectors documented one Severity Level IV traditional enforcement violation associated with impeding the regulatory process. Inspection Procedure 92723, “Follow up Inspection for Three or More Severity Level IV Traditional Enforcement Violations in the Same Area in a 12-Month Period,” will be performed for three of the violations as described in the 05000416/2017002 report (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17220A152). The additional Severity Level IV traditional enforcement violation documented in this report will be included in the population of violations for which the NRC plans to conduct an additional Inspection Procedure 92723 inspection to assess your evaluation of these additional violations and review the adequacy of associated corrective actions.

If you contest the violations or significance of these NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region IV; the Director, Office of Enforcement; and the NRC resident inspector at the GGNS.

If you disagree with a cross-cutting aspect assignment in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region IV; and the NRC resident inspector at the GGNS.

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

Sincerely,

/RA/

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

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

Enclosure:
Inspection Report 05000416/2017007
w/Attachment: Supplemental Information

cc w/ enclosure: Electronic Distribution

U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Docket:	05000416
License:	NPF-29
Report Nos.:	05000416/2017007
Licensee:	Entergy Operations, Inc.
Facility:	Grand Gulf Nuclear Station
Location:	P.O. Box 756 Port Gibson, MS 39150
Dates:	September 11 through September 29, 2017
Exit Date:	October 18, 2017
Team Leader:	W. Sifre, Senior Reactor Inspector, Engineering Branch 1
Inspectors:	C. Stott, Reactor Inspector, Engineering Branch 1 N. Okonkwo, Reactor Inspector, Engineering Branch 2 N. Day, Resident Inspector, Grand Gulf Nuclear Station S. Hedger, Emergency Preparedness Inspector, Plant Support Branch 2
Accompanying Personnel:	C. Baron, Contractor, Beckman and Associates S. Gardener, Contractor, Beckman and Associates
Approved By:	Thomas R. Farnholtz Chief, Engineering Branch 1 Division of Reactor Safety

Enclosure

SUMMARY

IR 05000416/2017007; 09/11/2017 – 09/29/2017; Grand Gulf Nuclear Station; Baseline Inspection, NRC Inspection Procedure 71111.21M, “Design Basis Assurance Inspection”

The report covers an announced inspection by a team of four regional inspectors, two contractors, and one inspector in training. Six findings were identified. Five of the findings were of very low safety significance and one of the findings was Severity Level IV. The final significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter 0609, “Significance Determination Process.” Cross-cutting aspects were determined using Inspection Manual Chapter 0310, “Aspects Within the Cross-Cutting Areas.” Findings for which the Significance Determination Process does not apply may be Green or be assigned a severity level after Nuclear Regulatory Commission (NRC) management review. All violations of NRC requirements are dispositioned in accordance with the NRC’s Enforcement Policy, dated July 9, 2013. The NRC’s program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, “Reactor Oversight Process,” Revision 6, dated July 2016.

Cornerstone: Mitigating Systems

- Green. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” which states, in part, “Activities affecting quality shall be prescribed by documented instructions, procedures, or drawings of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings.” Specifically, prior to September 27, 2017, the licensee failed to initiate an Engineering Evaluation of the long term scaffolds constructed within the 2 inch seismic acceptance criteria for separation from the Safety-Related Diesel Air Start System as required by Procedure EN-MA-133, “Scaffolding Control.” Failure to perform this evaluation could adversely impact the ability of the air start system to start the Standby Diesel Generator in a Design Basis Earthquake. In response to this issue the licensee performed an engineering evaluation for the scaffold. The finding was entered into the licensee’s corrective action program as Condition Report CR-GGN-2017-09748.

  The team determined that the failure to perform an engineering evaluation for current scaffolding to assure it met seismic design requirements was a performance deficiency. The performance deficiency was more than minor, and therefore a finding, because it was associated with the equipment performance 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, failure to perform the seismic evaluation within the 2 inch seismic acceptance criteria for separation could adversely affect the Air Start system’s ability to start the Standby Diesel Generators in the event of a Design Basis Earthquake. In accordance with Inspection Manual Chapter 0609, Appendix A, “The Significance Determination Process for Findings At Power,” dated July 19, 2012, the finding screened as having very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non-technical specification equipment; and did not screen as

potentially risk-significant due to seismic, flooding, or severe weather. This finding had a cross-cutting aspect in the area of human performance associated with design margins. Margins are carefully guarded and changed only through a systematic and rigorous process. [H.6]. (Section 1R21.2.2.b)

- Green. The team identified a Green, self-revealed, non-cited violation of 10 CFR 50 Appendix B, Criterion XVI, “Corrective Actions,” which states, in part, “Conditions adverse to quality are promptly identified and corrected.” On June 22, 2016, the station identified a condition adverse to quality affecting the standby diesel generators, but did not promptly correct the issue until September 22, 2017. Specifically, the actions described in Standing Order 17-0011 were not appropriate for restoring full capability during a design basis tornado event, which could affect the capability of the Division I and II standby diesel generators. In response to this issue the licensee revised the standing order to have the operator press the diesel generator manual start button while the diesel is running to eliminate the associated non-safety trips. The finding was entered into the corrective action program as Condition Report CR-GGN-2017-09751.

  The team determined that the failure to promptly correct a condition adverse to quality, regarding diesel generator capability during a design basis tornado is a performance deficiency. The performance deficiency was more than minor, and therefore a finding, because it was associated with the design control attribute of the Mitigating Systems Cornerstone. Specifically, the failure to correct an identified condition adverse to quality resulted in a prolonged design challenge to the Division I and II standby diesel generator capability, which adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. In accordance with Inspection Manual Chapter 0609, Appendix A, “Exhibit 2”, dated June 19, 2012, the team determined that the finding required a detailed risk evaluation, per Exhibit 4 screening question number 1, for external event mitigation systems. According to the tornado analysis database prepared by the Office of Reactor Research, the frequency of an F-2 tornado or stronger at Grand Gulf Nuclear Station (GGNS) is 4.71E-4/year. The design basis tornado at GGNS has a maximum wind velocity of 360 mph which correlates to a strong F-5 tornado. The design basis tornado generates a differential pressure of 3 psi. The pressure of concern is the 1.5 psi that could affect operation of the Division I and II diesel generators. Given that pressure is proportional to the square of the velocity, the wind speed affecting the diesel generators would be approximately 250 mph. 250 mph is in the range of an F-4 tornado. Using generic distributions of the frequency of varying tornado strengths, the analyst estimated that the frequency of an F-4 tornado or stronger at GGNS is 3.93E-6/year. Using the site-specific SPAR model, the analyst quantified the conditional core damage probability for a tornado-induced loss of offsite power with the failure of both Division I and II diesel generators. The conditional core damage probability was 6.35E-2. Therefore, the incremental conditional core damage probability of the performance deficiency, using the bounding assumption that all F-4 or stronger tornados striking the site would fail both diesel generators, was 2.50E-7. Qualitatively, given this bounding assumption, and the potential to recover the diesels after failure, the analyst determined that the ΔCDF was less than 1E-7. This results in a finding of very low safety significance (Green). This finding had a cross-cutting aspect in the area of human performance associated with procedure adherence because the licensee failed to follow the operability evaluation process to properly determine operability [H.8]. (Section 1R21.2.2.b)

- Green. The team identified a Green, non-cited violation of Technical Specification (TS) 5.4.1, which states, in part, "Written procedures shall be established, implemented, and maintained covering the following activities," referenced in Regulatory Guide (RG) 1.33, Revision 2, dated February 1978, Appendix A.9, "Procedures for Performing Maintenance," which requires that maintenance that can affect the performance of safety-related equipment should be properly pre-planned and performed in accordance with written procedures, documented instructions, or drawings appropriate to the circumstance." Specifically, prior to September 29, 2017, the licensee did not have a procedure to implement maintenance as recommended by the vendor in Vendor Document VM460000161, "ELMA Cast Coil Power Transformers Installation, Maintenance, Operating, and Storage Instructions." In response to this issue, the licensee performed testing to ensure that the transformer will perform its design function and is developing an improved maintenance procedure. The finding was entered into the corrective action program as Condition Report CR-GGN-2017-09390.

  The team determined that the failure to implement vendor recommended preventive maintenance is a performance deficiency. The finding was more than minor because it was associated with the equipment performance 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 failure to have procedures that implement vendor recommended maintenance resulted in a question regarding the functionality of the transformer at elevated temperature. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated July 19, 2012, the finding screened as having very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non-technical specification equipment; and did not screen as potentially risk-significant due to seismic, flooding, or severe weather. This finding had a cross-cutting aspect in the area of human performance associated with conservative bias because the licensee failed to ensure that maintenance implement a process of planning, controlling, and executing work activities such that nuclear safety is the overriding priority [H.5]. (Section 1R21.2.3.b)

- Severity Level IV. The team identified three examples of a Severity Level IV, non-cited violation of 10 CFR 50.71, "Maintenance of Records, Making of Reports," Section (e), which states, in part, "Each person licensed to operate a nuclear power reactor shall update periodically the final safety analysis report originally submitted as part of the application for the license, to assure that the information included in the report contains the latest information developed." This submittal shall contain all the changes necessary to reflect information and analyses submitted to the Commission by the applicant or licensee, or prepared by the applicant or licensee pursuant to Commission requirement since the submittal of the original or the last update to the final safety analysis report." Specifically, prior to September 29, 2017, the licensee failed to ensure the final safety analysis report reflected the current plant configuration. In response to this issue, the licensee created a corrective action to update the final safety analysis report. The finding was entered into the licensee's corrective action program as Condition Reports CR-GGN-2017-09154, CR-HQN-2017-01356, and CR-GGN-2017-09747.

The team determined that the failure to update the final safety analysis report in accordance with 10 CFR 50.71(e) was a performance deficiency. Following the Reactor Oversight Process (ROP's) significance determination process, the team determined this violation was associated with a minor performance deficiency. The ROP's significance determination process does not specifically consider the regulatory process impact in its assessment of licensee performance. Therefore, it is necessary to also address this violation which impedes the NRC's ability to regulate using traditional enforcement to adequately deter non-compliance. Assessing the performance deficiency in accordance with the NRC Enforcement Policy, the team determined it to be a Severity Level IV violation because the lack of up-to-date information in the final safety analysis report has not resulted in any unacceptable change to the facility or procedures. This finding did not have an assigned cross-cutting aspect because cross-cutting aspects are not assigned to traditional enforcement violations. (Section 1R21.2.6.b)

- Green. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," which states, in part, "Measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions." Specifically, since September 25, 2013, the licensee failed to include a design basis standby service water system (SSWS) piping crack in the appropriate design calculation and procedure. In response to this issue the licensee performed an operability determination to ensure that the ultimate heat sink basins would still have sufficient capacity to meet the 30-day mission time. This finding was entered into the licensee's corrective action program as Condition Report CR-GGN-2017-10192.

  The team determined that the failure to update a design calculation and a procedure to address a postulated standby service water passive failure was a performance deficiency. The finding was more than minor because it was associated with the Mitigating Systems Cornerstone attribute of equipment performance and affected the cornerstone objective of ensuring the availability, reliability, and capability of mitigating systems to respond to initiating events to prevent undesirable consequences. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012, the issue screened as having very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non-technical specification equipment; and did not screen as potentially risk-significant due to seismic, flooding, or severe weather. The team determined that this finding did not have a cross-cutting aspect because the most significant contributor did not reflect current licensee performance. (Section 1R21.3.5.b)

- Green. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," which states, in part, "Measures shall be established to assure that applicable regulatory requirements and the design basis…for those structures, systems, and components to which this appendix applies are correctly translated into specifications, drawings, procedures, and instructions." Specifically, prior to October 17, 2017, the licensee failed to provide adequate procedures or training to licensed operators to ensure the main steam isolation valve-leakage control system and feedwater leakage control system are manually started consistent with the licensee's design basis assumptions. In response to this issue the licensee has provided specific guidance and training to the operators. This finding was entered into the licensee's corrective action program as Condition Report CR-GGN-2017-09112.

The team determined that the failure to ensure adequate design control measures are translated into procedures and training is a performance deficiency. The performance deficiency is more than minor, and therefore a finding, because it was associated with the barrier performance attribute of the Barrier Integrity Cornerstone and adversely affected the cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Specifically, the plant was operated at power for an extended period of time without adequate procedures and training for licensed operators to ensure that the system would be placed in service in a manner that ensured radiological leakage across main steam isolation valves and through feedwater piping is addressed during a postulated accident. In accordance with Manual Chapter 0609, "Significance Determination Process," Attachment 4 (effective date October 7, 2016); and the corresponding Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," Exhibit 3, "Barrier Integrity Screening Questions" (issue date June 19, 2012); the issue was evaluated using Appendix H, "Containment Integrity Significance Determination Process" (issue date May 6, 2004). Because the opportunities to ensure the design control measures were correctly captured in procedures and instructions for the main steam isolation valve-leakage control system and feedwater leakage control system were in 2001 and 1987, respectively; and the licensee instituted a time-critical operator action program within the last year to prevent such issues from occurring, the issue was determined to have very low safety significance (Green). The performance deficiency was not indicative of current performance. Therefore, no cross-cutting aspect is being assigned. (Section 1R21.4.5.b)

Licensee-Identified Violations

Violations of very low safety significance (Green) that were identified by the licensee have been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program. These violations and associated corrective action tracking numbers are listed in Section 4OA7 of this report.

REPORT DETAILS

1. REACTOR SAFETY

Inspection of component design bases and modifications made to structures, systems, and components verifies that plant components are maintained within their design basis. Additionally, this inspection provides monitoring of the capability of the selected components and operator actions to perform their design bases functions. The inspection also monitors the implementation of modifications to structures, systems, and components. Modifications to one system may also affect the design bases and functioning of interfacing systems as well as introduce the potential for common cause failures. As plants age, modifications may alter or disable important design features making the design bases difficult to determine or obsolete. The plant risk assessment model assumes the capability of safety systems and components to perform their intended safety function successfully. This inspectable area verifies aspects of the Initiating Events, Mitigating Systems, and Barrier Integrity Cornerstones for which there are no indicators to measure performance.

1R21 Design Basis Assessment (71111.21M)

The inspection team selected risk-significant components, industry operating experience issues, modifications, and operator actions for review using information contained in the licensee's probabilistic risk assessment. In general, this included components, industry operating experience issues, modifications, and operator actions that had a risk achievement worth factor greater than two or a Birnbaum value greater than 1E-6.

.1 Inspection Scope for Components Selected

To verify that the selected components and modifications 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 bases 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 bases 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 team selected permanent plant modifications including permanent plant changes, design changes, set point changes, procedure changes, equivalency evaluations, suitability analyses, calculations, and commercial grade dedications to verify that design bases, licensing bases, and performance capability of components have not been degraded through modifications. The team determined whether post-modification testing established operability. The team verified that supporting design basis documentation, such as calculations, design specifications, vendor manuals, the updated final safety analysis report, technical specification and bases, and plant specific safety evaluation reports, were updated consistent with the design change. The team verified that other design basis features, such as structural, fire protection, flooding, environmental qualification, and potential emergency core cooling system strainer blockage mitigation, which could be affected by the modification, were not adversely impacted. The team verified that procedures and training plans, such as abnormal operating procedures, alarm response procedures, and licensed operator training manuals, affected by the modifications were updated.

The inspection procedure requires a review of four to six components based on risk significance and four to six modifications to mitigation structures, systems, and components. One of the inspection samples selected shall be associated with containment-related structures, systems, and components which are considered for large early release frequency (LERF) implications. The samples selected for this inspection were nine components, five modifications, and five operating experience items.

The selected inspection items supported risk-significant functions as follows:

- Electrical power to mitigation systems: The team selected several components in the offsite and on-site 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 accident, station blackout, and a loss-of-coolant accident with offsite power available. As such, the team selected:

    - 125VDC Battery, L21-1B3

    - Standby Diesel Generator, P75-Division II

    - 4160 VAC Electrical Bus 17AC

    - Engineered Safety Features Transformer 12

    - Feeder Breaker to Engineered Safety Features Transformer No. 12 - R21-152-1704

    - Automatic Depressurization System Logic System

    - Standby Service Water System Pump, 1P41C001A, Motor

- Initiating events minimization:

    - Condensate Storage Tank, 1-P11-A002

- Decay heat removal (LERF):

    - Alternate Decay Heat Removal Heat Exchangers

.2 Results of Detailed Reviews of Components

.2.1 125VDC Battery, L21-1B3

a. Inspection Scope

The team reviewed the updated safety analysis report, system description, design basis documents, the current system health report, selected drawings and calculations, maintenance and test procedures, and condition reports associated with the 125VDC battery, L21-1B3. The team also performed walkdowns and conducted interviews with system and design engineering personnel to ensure the capability of these components to perform its desired design basis function. Specifically, the team reviewed:

- Maintenance history and corrective action program reports to verify the monitoring of potential degradations by comparing the last 5-year performance tests and service tests.

- Calculations for electrical distribution, system load flow/voltage drop to verify that battery capacity and voltages remained within minimum acceptable limits.

- Sizing calculations to verify input assumptions, design loading, and environmental parameters are appropriate, and that the battery cell and battery charger are sized to perform the design basis function.

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

- Operations Procedure for Station Blackout to verify assumptions and loads identified in battery sizing calculation.

b. Findings

No findings were identified.

.2.2 Standby Diesel Generator, P75-Division II

a. Inspection Scope

The team reviewed the updated safety analysis report, system description, the current system health report, selected drawings, maintenance procedures, test procedures, and condition reports associated with the Standby Diesel Generator, P75-Division II. The team also performed walkdowns and conducted interviews with system engineering personnel to ensure capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- Maintenance history and corrective action program reports to verify the monitoring of potential degradations.
- Procedures for preventative maintenance, inspection, and testing to compare maintenance practices against industry and vendor guidance.
- Calculations for electrical distribution, system load flow/voltage drop, short circuit, and electrical protection to verify that bus capacity and voltages remained within minimum acceptable limits.
- Maintenance documentation, focusing on electrical generator end-of-life.
- Diesel Generator loading calculations.

b. Findings

Failure to initiate Engineering Seismic Evaluation for the Construction of Long Term Scaffolding in the Division I and Division II Diesel Rooms

Introduction. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," involving the failure to initiate an Engineering Seismic Evaluation for the construction of Long Term Scaffolding within the allowable seismic acceptance criteria for proximity to Safety-Related Equipment.

Description. Long term scaffolds in the Division I and Division II Diesel Rooms were used for frequent maintenance on the Air Dry system for Diesel air start. The scaffolds were outside the seismic acceptance criteria for scaffold (within 2 inches of Safety-Related equipment). In accordance with Procedure EN-MA-133, "Control of Scaffolding," Revision 17, this condition required initiating an Engineering Evaluation Request (attachment 9.5). Contrary to this, evaluations from previous work orders for Diesel Room scaffold erections, dated March 17, 2003, and March 15, 2013, were credited without a current Engineering Evaluation for the construction. A walkdown of the scaffold revealed deviations from the configuration approved in the evaluations/design (base plates were not placed on a level surface). The scaffold, if not built to meet seismic standards, may adversely affect the response of the emergency generators to an initiating event.

As required by procedure, the scaffolding was tagged with the current work order and date, but Maintenance Department inspection tags were dated earlier than the current work order, indicating the structure was in place when the current work order was tagged. Tracing the actual date of construction was not forthcoming. No actual date of construction for the long term scaffold was in the scaffolding log. Maintenance orders requiring construction of the scaffolding were left blank if the scaffolding was still in place from previous work orders.

Analysis. The team determined that the failure to perform an engineering evaluation for current scaffolding to assure it met seismic design requirements is a performance deficiency. The performance deficiency was more than minor, and therefore a finding, because it related to the equipment performance 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, failure to perform the seismic evaluation for scaffold erected within the 2 inch seismic acceptance criteria for separation could adversely affect the Air Start system's ability to start the Standby Diesel Generators in the event of a Design Basis Earthquake. In accordance with Inspection Manual Chapter 0609, Chapter 609, Appendix A, "The Significance Determination Process (SDP) for Findings At Power," dated July 19, 2012, the finding screened as having very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non-technical specification equipment; and did not screen as potentially risk-significant due to seismic, flooding, or severe weather. This finding had a cross-cutting aspect in the area of human performance associated with Design Margins. Margins are carefully guarded and changed only through a systematic and rigorous process [H.6].

Enforcement. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," which states, in part, "Activities affecting quality shall be prescribed by documented instructions, procedures, or drawings of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings." Contrary to the above, prior to September 27, 2017, the licensee failed to perform activities in accordance with instructions, procedures, and drawings. Specifically, the licensee failed to initiate an Engineering Evaluation of the long term scaffolds constructed within the 2 inch seismic acceptance criteria for separation from the Safety-Related Diesel Air Start System as required by Procedure EN-MA-133, "Scaffolding Control." Failure to perform this evaluation could adversely impact the ability of the air start system to start the Standby Diesel Generator in a Design Basis Earthquake. In response to this issue the licensee has performed an engineering evaluation for the scaffold. This finding was entered into the licensee's corrective action program as Condition Report CR-GGN-2017-09748. Because this finding was of very low safety significance and has been entered into the licensee's corrective action program, this violation is being treated as a non-cited violation, consistent with Section 2.3.2.a of the NRC Enforcement Policy: NCV-05000416/2017007-01, "Failure to Initiate Engineering Seismic Evaluation for the Construction of Long Term Scaffolding in the Division I and Division II Diesel Rooms."

Failure to Correct Standby Diesel Generator Trip

Introduction. The team identified a Green, self-revealed, non-cited violation of 10 CFR 50 Appendix B, Criterion XVI, "For the failure to ensure a condition adverse to quality was promptly identified and corrected." On June 22, 2016, GGNS identified the atmospheric differential pressures associated with a design basis tornado could trip the Division I and II standby diesel generators. This issue was not appropriately corrected until September 28, 2017, due to an ineffective recovery action.

Description. On June 22, 2016, GGNS, using an effective operating experience program, identified that a design basis tornado includes a differential pressure of 3.0 psig, whereas an active diesel generator trip on high crankcase pressure actuates at 1.5 psig. This issue was documented in Condition Report CR-GGN-2016-04919. The condition described in this condition report was declared Operable-Compensatory Measures, based on required operator actions described in Standing Order 17-0011.

However, Standing Order 17-0011 directed site personnel to station at the standby diesel generators (during Tornado Warning conditions) to restart the diesel generator using the emergency start push button in the event that the diesel had tripped due to the tornado and atmospheric conditions initiating a spurious (and inappropriate) high crankcase pressure trip. The operator was directed to use the emergency start push button to restart the diesel. When this is done the high crankcase pressure trip will be disabled and the diesel will not take another spurious trip.

The team reviewed the standing order and design basis of the diesel. The team concluded that Standing Order 17-0011 did not provide an appropriate response to ensure that the diesel would not trip when needed. Grand Gulf Nuclear Station entered this issue into the corrective action program using Condition Report CR-GGN-2017-09759. For corrective actions, GGNS revised their 05-1-02-VI-2, off-normal event procedure for Hurricanes, Tornados, and Severe Weather to include depressing the emergency start push button with the standby diesel generators running (before the diesel tripped on high crankcase pressure) to prevent the diesel logic from spuriously tripping the diesel generator.

Analysis. The team determined that the failure to promptly correct a condition adverse to quality, regarding diesel generator capability during a design basis tornado is a performance deficiency. The finding is more than minor since it is associated with the design control attribute of the Mitigating Systems Cornerstone. Specifically, the failure to correct an identified condition adverse to quality resulted in a prolonged design challenge to the Division 1 and 2 standby diesel generator capability, which adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. In accordance with Inspection Manual Chapter 0609, Attachment 4, "Initial Characterization of Findings," dated June 19, 2012, and Manual Chapter 0609, Appendix A, "Exhibit 2", dated June 19, 2012, the team determined that the finding required a detailed risk evaluation, per Exhibit 4 screening question number 1, for external event mitigation systems. According to the tornado analysis database prepared by the Office of Reactor Research, the frequency of an F-2 tornado or stronger at GGNS is 4.71E-4/year. The design basis tornado at GGNS has a maximum wind velocity of

360 mph which correlates to a strong F-5 tornado. The design basis tornado generates a differential pressure of 3 psi. The pressure of concern is the 1.5 psi that could affect operation of the Division I and II diesel generators. Given that pressure is proportional to the square of the velocity, the wind speed affecting the diesel generators would be approximately 250 mph. 250 mph is in the range of an F-4 tornado. Using generic distributions of the frequency of varying tornado strengths, the analyst estimated that the frequency of an F-4 tornado or stronger at GGNS is 3.93E-6/year. Using the site-specific SPAR model, the analyst quantified the conditional core damage probability for a tornado-induced LOOP with the failure of both Division I and II diesel generators. The conditional core damage probability was 6.35E-2. Therefore, the incremental conditional core damage probability of the performance deficiency, using the bounding assumption that all F-4 or stronger tornados striking the site would fail both diesel generators, was 2.50E-7. Qualitatively, given this bounding assumption, and the potential to recover the diesels after failure, the analyst determined that the ΔCDF was less than 1E-7. This results in a finding of very low safety significance (Green). The team determined this finding had a cross-cutting aspect in the area of human performance associated with procedure adherence because the licensee failed to follow the operability evaluation process when performing an operability evaluation [H.8].

Enforcement. The team identified a Green, self-revealed, non-cited violation of 10 CFR 50 Appendix B, Criterion XVI, which states, in part, "Conditions adverse to quality are promptly identified and corrected." Contrary to the above, between June 22, 2016, and September 22, 2017, the licensee failed to promptly correct a condition adverse to quality. Specifically, the actions described in Standing Order 17-0011 were not appropriate for restoring full capability during a design basis tornado event, which could affect the capability of the Division I and II standby diesel generators. Because the finding was of very low safety significance and has been entered into the corrective action program as Condition Report CR-GGN-2017-09751. This violation is being treated as a non-cited violation, consistent with Section 2.3.2.a of the NRC Enforcement Policy. NCV-05000416/2017007-02, "Failure to Correct Standby Diesel Generator Trip."

.2.3 4160 VAC Electrical Bus 17AC

a. Inspection Scope

The team reviewed the updated safety analysis report, system description, design basis documents, the current system health report, selected drawings and calculations, maintenance and test procedures, and condition reports associated with 4160 VAC Electrical Bus 17AC. 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:

- System health reports, component maintenance history, and corrective action program reports to verify the monitoring and correction of potential degradation.

- Calculations for electrical distribution, system load flow/voltage drop, short-circuit, and electrical protection to verify that bus capacity and voltages remained within the minimum acceptable limits.

- The protective device settings and feeder circuit breaker ratings to ensure adequate selective protection coordination of connected equipment during worst-case short circuit conditions.

- Procedures for preventive maintenance, inspection, and testing to compare maintenance practices against industry and vendor guidance; including the cable aging management program.

- Results of completed preventative maintenance on switchgear and breakers, including breaker tracking.

b. Findings

Failure to Establish a Preventive Maintenance Procedure for Safety-Related Equipment

Introduction. The team identified a Green, non-cited violation of Technical Specification (TS) 5.4.1, "For the failure to establish a procedure to implement maintenance for High Pressure Core Spray (HPCS) Transformer 1E22S003," as recommended by the vendor in Vendor Document VM460000161, "ELMA Cast Coil Power Transformers Installation, Maintenance, Operating and Storage Instructions."

Description. On September 12, 2017, the inspectors performed a walkdown of 4160 VAC Electrical Bus 17AC and noted that the HPCS Transformer 1E22S003 was very warm to touch, considering the transformer is lightly loaded in non-accident condition. The inspector noted that there was no temperature gauge to record the temperature of the transformer and inquired if the preventive maintenance of the transform included performance of thermography to record the temperature of the transformer. The licensee had no record of thermography infrared performance on the transformer.

Calculation EC-Q1111-93001 stated that the full load losses for the transformer was 7.6KW and the no-load loss was 2.2KW, however, the calculation did not document the temperature of the transformer core winding at full load. The licensee stated that they perform inspection and cleaning of the transformer in accordance with Procedure 01-S-02-9, "General Cleaning and Inspection of Non-Rotating Electrical Equipment." In review of the procedure the inspector found that it did not list any specific guidelines for inspecting and cleaning the HPCS transformer. In addition, this procedure referenced vendor manuals for ITE metal clad switchgears, Klocker-Moeller 480-Volt Motor control centers and Delta Switchboards. It failed to list the vendor document of the transformer as source for vendor recommended inspection guidelines and acceptance criteria. The inspector reviewed work orders 00136453 and 00361101, which implemented the Procedure 01-S-02-9, but could not identify any tests or data that indicated that the transformer was inspected, cleaned, and maintained as recommended by the vendor in Vendor Manual 460000161. The vendor indicated that the transformer was designed for operation over a temperature range of 40 to 120 degrees F (4.44 to 48.89 degrees C) and rated for a temperature rise of 115 degrees C. The resulting maximum operating temperature based on the rated temperature range and the rated temperature rise was approximately 163.89 C or 327 degrees F. The inspector noted that the licensee had not performed any thermography to record or trend the temperature of the transformer.

Transformer 1E22S003 is a cast coil transformer per Vendor Document VM460000161. The vendor recommends that, "careful consideration should be given to indication of overheating." The inspector determined that preventive maintenance as recommended by the vendor had not been performed and the Procedure 01-S-02-09 did not provide adequate guidance for preventive maintenance in accordance with the vendor specifications. The licensee documented this issue in their corrective action program as Condition Report CR-GGN-2017-09390 to perform infrared thermography testing and perform an operability determination to ensure that the transformer is not overheated and therefore operable.

Analysis. The team determined that the failure to implement vendor recommended preventive maintenance is a performance deficiency. The performance deficiency was more-than-minor, and therefore a finding, because it was related to the equipment performance 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, failure to perform vendor recommended maintenance on safety-related equipment, the HPCS transformer could adversely affect the ability of the transformer to perform its safety function. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated July 19, 2012, the finding screened as having very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non- technical specification equipment; and did not screen as potentially risk-significant due to seismic, flooding, or severe weather. This finding had a cross-cutting aspect in the area of human performance associated with conservative bias because the licensee failed to ensure that maintenance implement a process of planning, controlling, and executing work activities such that nuclear safety is the overriding priority [H.5].

Enforcement. The team identified a Green, non-cited violation of Technical Specification (TS) 5.4.1, which requires, in part, "Written procedures shall be established, implemented, and maintained covering the following activities," referenced in Regulatory Guide (RG) 1.33, Revision 2, dated February 1978, Appendix A.9, "Procedures for Performing Maintenance," which requires that, "Maintenance that can affect the performance of a safety-related equipment should be properly pre-planned and performed in accordance with written procedures, documented instructions, or drawings appropriate to the circumstance." Contrary to the above, prior to September 29, 2017, the licensee failed to properly pre-plan and perform maintenance on safety-related equipment. Specifically, the licensee did not have a procedure to implement maintenance as recommended by Vendor Document VM460000161, "ELMA Cast Coil Power Transformers Installation, Maintenance, Operating and Storage Instructions," for HPCS Transformer 1E22S003. In response to this issue, the licensee performed testing to ensure that the transformer will perform its design function and is developing an improved maintenance procedure. The licensee entered this performance deficiency into their corrective action program as Condition Report CR-GGN-2017-09390. Because this finding was of very low safety significance and has been entered into the licensee's corrective action program, this violation is being treated as a non-cited violation, consistent with Section 2.3.2.a of the NRC Enforcement Policy: NCV-05000416/2017007-03, "Failure to Establish a Preventive Maintenance Procedure for Safety-Related Equipment."

.2.4 Engineered Safety Features Transformer 12

a. Inspection Scope

The team reviewed the updated safety analysis report, system description, design basis documents, the current system health report, selected drawings and calculations, maintenance and test procedures, and condition reports associated with Engineered Safety Features Transformer 12. 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:

- System health reports, component maintenance history, and corrective action program reports to verify the monitoring and correction of potential degradation.

- Calculations for transformer loading to ensure capacity, transformer cabling, and protection to verify that transformer nameplate capacity and voltages remained within minimum acceptable limits.

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

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

- Results of completed preventative maintenance on the transformer to ensure data acceptability and within tolerance.

b. Findings

No findings were identified.

.2.5 Feeder Breaker to Engineered Safety Features Transformer No. 12 - R21-152-1704

a. Inspection Scope

The team reviewed the updated safety analysis report, system description, design basis documents, the current system health report, selected drawings and calculations, maintenance and test procedures, and condition reports associated with Engineered Safety Features Transformer No. 12, Breaker R21-152-1704. The team also performed walkdowns and conducted interviews with system engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

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

- Calculations for electrical distribution, system load flow/voltage drop, short-circuit, and electrical protection to verify that bus capacity and voltages remained within minimum acceptable limits.

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

- Procedures for preventive maintenance, inspection, and testing to compare maintenance practices against industry and vendor guidance; including the cable aging management program.

b. Findings

No findings were identified.

.2.6 Standby Service Water System Pump, 1P41C001A, Motor

a. Inspection Scope

The team reviewed the updated safety analysis report, system description, design basis documents, the current system health report, selected drawings and calculations, maintenance and test procedures, and condition reports associated with SSWS Pump (1P41C001A) motor. 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:

- Service water system health report, logic and schematic diagrams of pump motor, and design bases functional description to understand and evaluate system functional design and routing of the pump motor power cable.

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

- Corrective action documents issued in the past 5 years to verify that repeat failures and potential chronic issues will not prevent the pump and associated components from performing their safety function.

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

- Motor sizing calculations to verify input assumptions, motor power cable routing to ensure clearance, and design loading to ensure adequate design for pumping capacity. The team put special emphasis on pump motor testing methodology, the values assigned to acceptance criteria, and whether the values supported design parameters and assumptions.

b. Findings

Failure to Update the Final Safety Analysis Report

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

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

- Since March 4, 2001, the licensee failed to update information in Section 6.7.1.1.1.f.1 that states, “The main steam isolation valve-leakage control system was designed to be actuated within 20 minutes of a postulated design-basis loss of coolant accident.”

- Since September 29, 2011, the licensee failed to update various pages of the Updated Final Safety Analysis Report with information concerning the lengthening of the siphon piping between the two standby service water basins due to a power uprate evaluation. This information was located in the following: Sections 2.3.1.2.7, 9.2.1, and 9.2.5; Tables 2.3-18, 19, and 20; Tables 9.2-4, 5a, 5b, 6a, 6b, 15a, 15b, 16, 16a, 17, and 17a; and Figures 9.2-001, 6a, 6b, 7a, 7b, 8a, 8b, 35, 37a, 37b, 40, 42, and 43.

- Since September 25, 2013, the licensee failed to update footnotes in Sections 9.2.1.1.1 and 9.2.1.3 concerning the allowable passive failures which may be postulated in the SSWS concurrent with an accident scenario.

The licensee documented this issue in the corrective action program as Condition Reports CR-GGN-2017-09154, CR-HQN-2017-01356, and CR-GGN-2017-09747.

Analysis. The team determined that the failure to update the final safety analysis report in accordance with 10 CFR 50.71(e) was a performance deficiency. Following the ROP’s significance determination process, the team determined this violation was associated with a minor performance deficiency. The ROP’s significance determination process does not specifically consider the regulatory process impact in its assessment of licensee performance. Therefore, it is necessary to also address this violation, which impedes the NRC’s ability to regulate, using traditional enforcement to adequately deter non-compliance. Assessing the performance deficiency in accordance with the NRC Enforcement Policy, the team determined it to be a Severity Level IV violation because the lack of up-to-date information in the final safety analysis report has not resulted in any unacceptable change to the facility or procedures. This violation did not have an assigned cross-cutting aspect because cross-cutting aspects are not assigned to traditional enforcement violations.

Enforcement. The team identified three examples of a Severity Level IV, non-cited violation of 10 CFR 50.71, “Maintenance of Records, Making of Reports,” Section (e), which states, in part, “Each person licensed to operate a nuclear power reactor shall update periodically the final safety analysis report originally submitted as part of the application for the license, to assure that the information included in the report contains the latest information developed.” This submittal shall contain all the changes necessary to reflect information and analyses submitted to the Commission by the applicant or licensee, or prepared by the applicant or licensee pursuant to Commission requirement since the submittal of the original or the last update to the final safety analysis report.” Contrary to the above, since March 4, 2001, the licensee did not update the final safety analysis report to assure that the information included in the report contains the latest information developed. Specifically, the licensee failed to ensure the final safety analysis report reflected the current plant configuration. In response to these issue, the licensee created a corrective action to update the final safety analysis report. The violation was entered into the licensee’s corrective action program as Condition Reports CR-GGN-2017-09154, CR-HQN-2017-01356, and CR-GGN-2017-09747. Because this violation is Severity Level IV significance and entered into the corrective action program, this violation is being treated as a non-cited violation, consistent with Section 2.3.2.a of the NRC Enforcement Policy: NCV-05000416/2017007-04, “Failure to Update the Final Safety Analysis Report.”

.2.7 Automatic Depressurization System Logic System

a. Inspection Scope

The team reviewed the updated safety analysis report, system description, the current system health report, selected drawings, maintenance procedures, test procedures, and condition reports associated with the automatic depressurization logic system, with an emphasis on time delay relay B21-K5A. The team also performed walkdowns and conducted interviews with system engineering personnel to ensure capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- Component maintenance history and corrective action program reports to verify the monitoring of potential degradations.
- Procedures for preventative maintenance, inspection, and testing to compare maintenance practices against industry standards and vendor guidance.
- Work orders to implement technical specification surveillance requirements and correct identified problems.
- Corrective action program causal evaluations for time delay relay drift.
- Calculations and methodology to determine failure analysis to support loss of safety function determinations.
- Licensee event reports.

c. Findings

No findings were identified.

.2.8 Condensate Storage Tank, 1-P11-A002

a. Inspection Scope

The team reviewed the updated safety analysis report, design basis documents, selected drawings and calculations, maintenance and test procedures, and condition reports associated with the Condensate Storage Tank, 1-P11-A002, and associated instruments. The team also performed walkdowns and conducted interviews with system and design engineering personnel to ensure the capability of this tank to perform its desired design basis function. Specifically, the team reviewed:

- Maintenance history and corrective action reports to verify potential degradation of the tank and instruments.

- Pump net positive suction head and submergence calculations to verify the adequacy of the condensate storage tank level setpoints under the most limiting operating conditions.

- The capacity of the condensate storage tank to verify it's capability to mitigate a postulated station blackout event.

b. Findings

No findings were identified.

.2.9 Alternate Decay Heat Removal Heat Exchangers

a. Inspection Scope

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

- Maintenance history and corrective action program reports to verify the monitoring of potential degradations.

- Procedures for preventative maintenance, inspection, and testing to compare maintenance practices against industry standards and vendor guidance, and commitments to Generic Letter 89-13.

- Calculations for decay heat removal capability, over a range of heat loads and plant service (cooling water) temperatures.

- Maintenance work orders, planned and completed.

- Operating logs with an emphasis on the August 31, 2017, demonstration.

- Seismic qualification of residual heat removal/alternate decay heat removal piping.

- Shutdown operations protection plan, with an emphasis on providing operations bounding parameters of alternate decay heat removal evaluated capability.

b. Findings

No findings were identified.

.3 Results of Detailed Reviews of Permanent Plant Modifications

a. Inspection Scope

The inspectors reviewed five permanent plant modifications that had been installed in the plant during the last three years. This review included in-plant walkdowns for portions of the accessible systems. The modifications were selected based upon risk significance, safety significance, and complexity. The inspectors reviewed the modifications selected to determine if:

- The supporting design and licensing basis documentation was updated.

- The changes were in accordance with the specified design requirements.

- The procedures and training plans affected by the modification have been adequately updated.

- The test documentation as required by the applicable test programs has been updated.

- Post-modification testing adequately verified system operability and/or functionality.

The inspectors also used applicable industry standards to evaluate acceptability of the modifications.

.3.1 Engineering Change 47972, Adjust Orifices in Standby Service Water C to Increase Flow to High Pressure Core Spray Room Cooler

The inspectors reviewed Engineering Change 47972, which was implemented to provide a rebalance of flow rates in the HPCS service water system. This modification changed the flow rates by resizing and removing restricting orifices that were permanently installed in the system, to ensure appropriate cooling water flow to the HPCS diesel generator heat exchangers (two due to tandem style diesel generator) and the HPCS pump room cooler. The inspectors reviewed the calculations in the engineering change, to determine that the flow rates were consistent with required system flow for heat removal capability. The inspectors interviewed GGNS System Engineers to determine the need for the resizing and removal of restricting orifices, and system degradation mechanisms which may require additional corrective actions. The inspectors did not identify any concerns with the design change package.

.3.2 Engineering Change 02202, Replace Time Delay Relays on the Diesel Air Start System and Modify the Low Air Pressure Shutdown Interlock

The inspectors reviewed Engineering Change 02202, which was implemented to replace time delay relays on the diesel air start system and modify the low air pressure shutdown interlock. Specifically, the air start B circuits are configured such that automatic shutdowns that are required to be bypassed during a loss of coolant accident are re-activated (on train B circuits only) if starting air pressure falls below 120 psig. As a result, if starting air pressure falls below 120 psig, and the associated train A starting circuit fails, the diesel generator would have all automatic shutdown permissives active even during a loss of coolant accident run. The modification relocated the interlock outside the loss of coolant accident logic. Additionally, the TD2A and TD2B time delay relays currently installed in the diesel starting circuits were replaced with better performing relays. The inspectors reviewed the qualification of the relays and the drawings affected by this modification. The inspectors did not identify any concerns with the design change package.

.3.3 Engineering Change 071386: Remove Diesel Diodes per 10 CFR Part 21 Recommendation (Division I and Division II) Condition Reports CR-GGN-2017-3643 AND CR-GGN-2017-3964

The inspectors reviewed Engineering Change 071386, which was implemented to remove diodes associated with standby diesel generators. The change applies to both the Division I and Division II P75 (IP75E001A and IP75E001B) slow speed start circuits by removing diodes identified as defective by Engine Systems Incorporated (ESI) Report 10 CFR 21-0116, Revision 0. Failure of the installed diodes could result in loss of the control power to one standby diesel generator start circuit, train A and to the governor control circuits. To eliminate these potential failures, ESI recommended removal of the diodes. This design change affected the slow start feature used for maintenance and surveillances, and did not affect the required fast start function. The inspectors did not identify any concerns with the design change.

.3.4 Engineering Change 2113: Replace 1E12F025C Residual Heat Removal C Pump Pressure Relief Valve

The team reviewed Engineering Change 2113, which was implemented to change the pressure relief valve 1E12F025C for Residual Heat Removal Pump C to a different design. Specifically, the licensee replaced the previous discontinued Lonergan LCT-11 model relief valve with an Anderson Greenwood / Crosby series 900 OMNI-TRIM style 9511122B relief valve. The Lonergan LCT-11 relief valve is no longer produced and it has a very limited availability of replacement parts. The licensee replaced this style relief valve with an Anderson Greenwood / Crosby series 900 OMNI-TRIM style 9511122B relief valve that is currently being produced by the manufacturer. The change was driven by the need to rebuild the older style Lonergan LCT-11 relief valve. The team reviewed the seismic calculations performed for this engineering change since the replacement pressure relief valve was heavier than the previous valve. The team did not identify any concerns with the design change package.

.3.5 Engineering Change 25649: Grand Gulf Nuclear Station, Standby Service Water Ultimate Heat Sink Siphon Line Extension

The inspectors reviewed Engineering Change 25649, which was implemented to extend the siphon line between the ultimate heat sink basins. This change increased the volume of water available to each train of the SSWS in the event of a single failure of the opposite train. The team reviewed the change and associated calculations to verify the modified design satisfied the ultimate heat sink design requirements. The team evaluated the capability of the ultimate heat sink to withstand any postulated active or passive single failure while ensuring sufficient capacity to complete its 30-day mission.

b. Findings

Failure to Update a Calculation and Procedure to Address Standby Service Water Passive Failure

Introduction. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, Design Control," for the failure to ensure that design basis requirements were correctly translated into procedures and instructions. Specifically, design calculations and plant procedures were not updated to include the design basis methodology for postulating single passive failures in the SSWS following a loss of coolant accident.

Description. The team reviewed Licensing Amendment Request GNRO-2012/00102, "Request to Revise the Standby Service Water Passive Failure Methodology," dated September 14, 2012, and License Amendment 196, dated September 25, 2013, to verify that a previous NRC violation (NCV 05000416/2012008-03) had been appropriately resolved. The team noted that the SSWS passive failure methodology described in License Amendment 196 was more conservative than the methodology requested in the license amendment request. Specifically, License Amendment 196 included consideration of a crack in the SSWS from 30 minutes to 24 hours post-loss of coolant accident. The additional ultimate heat sink inventory loss associated with this postulated pipe crack had not been included in design basis calculation MC-Q1P41-11001, "Grand Gulf Nuclear Station Standby Service Water Ultimate Heat Sink Thirty Day Performance at EPU," or in off-normal event Procedure 05-1-02-III-12, "Standby Service Water Basin Level Control."

In response to the team's concerns, the licensee initiated Condition Report CR-GGN-2017-10192 on October 10, 2017. This condition report stated that the licensee failed to adequately incorporate the Significant Event Report information associated with the SSWS critical crack leakages into the appropriate design calculation and procedure. The licensee performed additional analyses and an operability determination, to determine that the ultimate heat sink would still have sufficient capacity to meet its 30-day mission time. However, the design calculation and plant procedures do require revision to reflect the reduction in margin associated with the correct design basis passive failure.

Analysis. The team determined that the failure to update a design calculation and a procedure to address a postulated SSWS passive failure was a performance deficiency. The finding was determined to be more than minor because it was associated with the

Mitigating Systems Cornerstone attribute of equipment performance and affected the cornerstone objective of ensuring the availability, reliability, and capability of mitigating systems to respond to initiating events to prevent undesirable consequences. Specifically, the failure to update a design calculation and a procedure to address a postulated SSWS passive failure could result in a delay identifying and isolating SSWS leakage after a loss of coolant accident. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012, Exhibit 2, "Mitigating Systems Screening Questions," the issue screened as having very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non-technical specification equipment; and did not screen as potentially risk-significant due to seismic, flooding, or severe weather. The team determined that this finding did not have a cross-cutting aspect because the most significant contributor did not reflect current licensee performance.

Enforcement. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," which states, in part, "Measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions." Contrary to the above, between September 25, 2013, and October 10, 2017, the licensee failed to translate applicable regulatory requirements and the design basis into specifications, drawings, procedures, and instructions. Specifically, the licensee failed to include a design basis SSWS piping crack in the appropriate design calculation and procedure. In response to this issue the licensee performed an operability determination to ensure that the ultimate heat sink basins would still have sufficient capacity to meet the 30-day mission time. This finding was entered into the licensee's corrective action program as Condition Report CR-GGN-2017-10192. Because this finding was of very low safety significance and has been entered into the licensee's corrective action program, this violation is being treated as a non-cited violation consistent with Section 2.3.2.a of the NRC Enforcement Policy: NCV-05000416/2017007-05, "Failure to Update a Calculation and Procedure to Address Standby Service Water Passive Failure."

.4 Results of Detailed Reviews of Operating Experience

.4.1 Inspection of NRC Information Notice 2009-09, "Improper Flow Controller Settings Renders Injection Systems Inoperable and Surveillance Did Not Identify"

The team reviewed the licensee's evaluation of NRC Information Notice 2009-09, "Improper flow controller settings renders injection systems inoperable and surveillance did not identify," to verify that the licensee has the correct flow controller settings for reactor containment isolation cooling and HPCS injection systems. The team also reviewed the performance of the Reactor Core Isolation Cooling system during a plant trip. The inspectors did not identify any concerns with how the licensee addressed this operating experience.

.4.2 Inspection of NRC Part 21 Event Report 50371, "Notice of Deviation Regarding K-Line Circuit Breaker Secondary Close Latch"

The team reviewed the licensee's evaluation of NRC Event Report 50371, to verify that the licensee evaluated the secondary close latch for Asea Brown Bovary part number 716610K01. The inspectors reviewed the GGNS supply chain search for the suspect components. The result of the search determined that GGNS did not have any of the suspect components and did not require any corrective actions. The inspectors did not identify any concerns with how the licensee addressed this operating experience.

.4.3 Inspections of NRC Information Notice 2006-17, "Recent Operating Experience of Service Water Systems Due to External Conditions"

The team reviewed the licensee's evaluation of Information Notice 2016-17, "Recent Operating Experience of Service Water Systems Due to External Conditions," to verify that the licensee has adequate protection from blockages that can occur in service water systems and impact the system's operability. The information notice specifies examples of blocking agents as silt, sand, small rocks, grass, weeds, frazil ice, and small aquatic fauna, such as fish. The inspectors did not identify any concerns with how the licensee addressed this operating experience.

.4.4 Inspection of NRC Information Notice 2017-005, "Potential Binding of Schneider Electric/Square-D Masterpact NT AND NW 480-VAC Circuit Breaker Anti-Pump Feature"

The team reviewed the licensee's evaluation of Information Notice 2017-005, "Potential Binding of Schneider Electric/Square-D Masterpact NT AND NW 480-VAC Circuit Breaker Anti-Pump Feature," to verify that the licensee's evaluation of the impact as cited in the Information Notice. The licensee documented the issue in their corrective action review program as OE-NOE-2017-00321 and they are reviewing the impact on their system. The inspectors did not identify any concerns with how the licensee addressed this operating experience.

.5 Results of Reviews for Operator Actions

a. Inspection Scope

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

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

- Restoration of power to affected buses after taking actions to address a reactor scram following lockout fault on the balance of plant 12A transformer

  The inspectors observed two separate crews in the simulator take actions responding to reactor scram resulting from feedwater and condensate sources being lost with the balance of plant 12A transformer lockout. After addressing these actions, the crews were evaluated restoring power to the balance of plant buses 11HD and 14AE. To address the power loss, the crews used actions delineated in off-normal event Procedure 05-1-02-I-4, “Loss of AC Power,” Revision 51; as well as Alarm Response Instruction 04-S-02-SH13-P807, “Alarm Response Instruction, Panel No. SH13-P807,” Revision 32. The licensee’s probabilistic risk assessment assumes that the time taken for licensed operators to restore power to 4.16kV or 6.9 kV buses with various power losses will take 10 minutes (Event R21-FO-HEBOPTRM). This action is necessary to ensure that adequate cooling water sources are available to avert core damage. The two crews performed the task in 2 and 8 minutes, respectively. This activity was satisfactorily performed within the required time. Text in the Alarm Response Instruction procedure was identified for revision. The licensee entered this into the corrective action program as Condition Reports CR-GGN-2017-09734 and CR-GGN-2017-09758.

- <u>Recognize and direct the field operators to address inadequate cooling in the standby service water train A pump house during a design basis loss of cooling accident</u>

  The inspectors observed two separate crews in the simulator responding to a design basis loss of coolant accident event combined with the ventilation dampers for standby service water train A pump house failing closed. During a design basis loss of coolant accident, it is also assumed that offsite power is lost, and a single failure occurs. The single failure in the design basis was the failure of emergency diesel generator 12 to start, losing power to safety bus 16AB. The licensee’s probabilistic risk assessment evaluates this design basis accident combined with the failure of standby service water pump house ventilation (Event Y47-FO-HEMOD-U). Action is necessary to establish alternate cooling to the pump house, if the room’s temperature alarm is valid, in order to avert a standby service water pump high temperature failure. The probabilistic risk assessment assumes that it takes 10 minutes to identify the standby service water A pump house high temperature alarm is alarming in the control room, and 10 minutes to respond and initiate alternate cooling. Areas for improvement with licensed operator performance were observed for both crews by the inspectors and licensee staff. The observations were entered in the licensee’s corrective action program as Condition Reports CR-GGN-2017-09746 and CR-GGN-2017-09755.

- <u>Diagnose and take actions to address inadequate cooling in the standby service water train A pump house</u>

  The inspectors observed a job performance measure of a non-licensed operator performing diagnosis and response actions associated with Alarm Response Instruction 04-1-02-1H13-P870, “Alarm Response Instruction, Panel No. 1H13-P870,” Revision 154. Related to the previous described sample, the operator implemented diagnosis and response actions to address inadequate cooling in the standby service water train A pump house associated with control room alarm 1A-G2 on Panel P870. This activity was observed being performed by two separate operators. The operators took actions to determine the cause of the alarm. Once the operators determined the cause of the alarm (ventilation dampers failed closed), action was simulated to establish alternate cooling to the pump house. This was accomplished by propping two building doors open. Areas for improvement with non-licensed operator performance and procedure quality were observed by the inspectors and licensee staff. The observations were entered in the licensee's corrective action program as Condition Reports CR-GGN-2017-09746 and CR-GGN-2017-09755.

- Manually start the main steam isolation valve and feedwater leakage control systems after the initiation of a design basis loss of coolant accident

  The inspectors observed two separate crews respond to a design basis loss of coolant accident in the simulator. Following the onset of a loss of coolant accident, both the main steam isolation valve-leakage control system and feedwater leakage control system are to be manually started within 20 minutes. Operators in the simulator were to discuss and take actions allowed in the simulator using each system's standard operating procedure. The manual start of these safety-related systems is necessary to ensure that assumptions about the projected dose received by control room staff and members of the public at the exclusion area boundary stays within the limits stated in 10 CFR 50, Appendix A, General Design Criterion 19 and 10 CFR 100, respectively.

b. Findings

Failure to Ensure Adequate Design Control Measures Are in Place Associated with Leakage Control Systems

Introduction. The team identified two examples of a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for failure to ensure design control measures assumed in the design basis were correctly translated into procedures and training. Specifically, the licensee failed to provide direction to licensed operators to ensure that the main steam isolation valve-leakage control system and feedwater leakage control system would be started consistent with the licensee's design basis assumptions.

Description.

Example 1: During a design basis loss of coolant accident event, the licensee assumes that an operator takes action to manually start the main steam isolation valve-leakage control system within 20 minutes of the event's onset. Section 3.1.1 of the Safety Evaluation Report for License Amendment 145 (March 14, 2001), states, "The licensee assumed that the MSIV-LCS is manually actuated within 20 minutes of the postulated LOCA." As part of the planning process for the operator actions portion of the inspection, the inspectors inquired with licensee staff about what procedures and training tools were available to ensure that time critical operator actions could be met onsite. Based on this inquiry, licensee staff found the supporting documentation. When the available procedures and tools were found, the licensee indicated that the content of the main steam isolation valve-leakage control system standard operating

Procedure 04-1-01-E32-1 SU, Revision 28, did not have adequate information to ensure that the associated time critical operator action would be performed as assumed in the licensing basis. Further review by the inspectors revealed the following:

- The licensee's emergency operating procedures, which are the priority use procedures during major events, make no reference to starting the main steam isolation valve-leakage control system during loss of coolant accident conditions.

- In the "System Design Bases," Section of Lesson Plan GLP-OPS-E3200, Revision 7, "MSIV Leakage Control System – E32," it indicates that the system, "shall be manually actuated…no sooner than 20 minutes after a postulated design-basis LOCA." This statement, based on text in Section 6.7 of a Safety Evaluation Report, dated September 1981 (NUREG-0831) had been superseded by the design assumptions in License Amendment 145.

- There is one control room Job Performance Measure associated with the main steam isolation valve-leakage control system (GJPM-OPS-E3201, Revision 2). However, it doesn't evaluate the capability of licensed operators to manually start the system within 20 minutes of the start of a loss of coolant accident.

- Observation of two licensed operator crews responding to a design basis loss of coolant accident event in the simulator revealed that neither crew discussed, or attempted, manual startup of the system.

Example 2: During a design basis loss of coolant accident event, the licensee assumes that an operator takes action to manually start the feedwater leakage control system within 20 minutes of the event's onset. This is stated in Updated Final Safety Analysis Report Section 7.3.2...6.2.a, Requirement 4.1 currently, and has been so stated since Revision 0 (January 1, 1987). Similar to Example 1, after inquiries had been made by the inspectors about the procedures and training materials used to implement the feedwater leakage control system startup, the licensee indicated that the content of the feedwater leakage control system standard operating Procedure 04-1-01-E38-1, Revision 101, did not have adequate information to ensure that the associated time critical operator action would be performed as assumed in the licensing basis. Further review by the inspectors revealed the following:

- The licensee's emergency operating procedures, which are the priority use procedures during major events, make no reference to starting the feedwater leakage control system during loss of coolant accident conditions.

- Lesson Plan GLP-OPS-E3800, Revision 6, "Feedwater Leakage Control System – E38," provides no training content regarding the 20 minute startup design basis assumption.

- There are no job performance measures currently developed to evaluate operator performance using the feedwater leakage control system.

- Observation of two licensed operator crews responding to a design basis loss of coolant accident event in the simulator revealed that neither crew made efforts to, or discussed manual startup, of the feedwater leakage control system.

Therefore, the combination of inadequate procedural direction and licensed operator training content, confirmed by licensed operator observations, failed to ensure that the time critical assumptions associated with the licensee's licensing basis for the main steam line isolation valve-leakage control system and feedwater leakage control system were met.

In response to these issues, the licensee initiated actions via Standby Order Number 17-0021 to establish interim emergency operator procedure directions for the licensed operators to ensure that credited safety-related equipment will be manually started as assumed if required. Site procedures are being revised to ensure permanent corrective action is taken. The licensee analyzed the loss of coolant accident event scenario to determine if the failure to start these two systems would result in radiological doses in excess of those assumed in 10 CFR 50, Appendix A, General Design Criterion 19 and 10 CFR 100. Based on communications with the licensee on October 17, 2017, the radiological dose thresholds for both regulations could have been exceeded during the subject event, which has a very low probability of occurrence. This finding was entered into the licensee's corrective action program as Condition Reports CR-GGN-2017-03750 and CR-GGN-2017-09762.

Analysis. The team determined that the failure to ensure adequate design control measures are translated into procedures and training is a performance deficiency. The performance deficiency is more than minor, and therefore a finding, because it was associated with the barrier performance attribute of the Barrier Integrity Cornerstone and adversely affected the cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Specifically, the plant was operated at power for an extended period of time without adequate procedures and training for licensed operators to ensure that the system would be placed in service in a manner that ensures radiological leakage across main steam isolation valves and through feedwater piping is addressed during a postulated accident. Traditional enforcement does not apply because the issue did not have any actual safety consequences or potential of impacting the NRC's regulatory function, and was not the result of any willful violation of NRC requirements. Using Manual Chapter 0609, "Significance Determination Process," Attachment 4, Tables 1, 2, and 3 worksheets (effective date October 7, 2016); and the corresponding Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," Exhibit 3, "Barrier Integrity Screening Questions," (issue date June 19, 2012); the issue, based on the answer to Question B.1, was evaluated using Appendix H, "Containment Integrity Significance Determination Process," (issue date May 6, 2004).

The inspectors noted that the plant specific attributes of GGNS's leakage control systems did not allow for screening the finding using Appendix H, so a detailed risk evaluation by a senior reactor analyst was performed. The analyst performed a bounding analysis where the leakage control systems were assumed to be significant contributors to large early release frequency. The analyst used PRAB-02-01, "Assessment of Boiling Water Reactor Main Steam Line Release Consequences," dated October 2002 to aid in identifying pertinent sequences of importance for both systems.

The most significant screened sequences were those where power was unavailable to the leakage control systems, thus an increase in large early release frequency from not operating the systems would not be present (e.g., station blackouts). Analysis of the pertinent sequences yielded an estimate for the increase in large early release frequency of 4.4E-8/year. Therefore, the finding was of very low safety significance (Green). Loss of coolant accidents were the dominant sequences. Core damage frequency is not affected by the finding. The analyst obtained this estimate using Revision 8.50 of the GGNS SPAR model run on SAPHIRE, Version 8.1.5.

Because the opportunities to ensure the design control measures were correctly captured in procedures and instructions for the main steam isolation valve-leakage control system and feedwater leakage control system were in 2001 and 1987, respectively; and the licensee instituted a time critical operator action program within the last year to prevent such issues from occurring, the performance deficiency is not indicative of present performance. Therefore, no cross-cutting aspect is being assigned.

Enforcement. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," which states, in part, "Measures shall be established to assure that applicable regulatory requirements and the design basis…for those structures, systems, and components to which this appendix applies are correctly translated into specifications, drawings, procedures, and instructions." Contrary to the above, from March 14, 2001 (Example 1), and January 1, 1987 (Example 2), to present, the licensee failed to ensure that design control measures were established that assured the design basis was correctly translated into procedures and instructions. Specifically, the licensee failed to provide adequate procedures or training to licensed operators to ensure the main steam isolation valve-leakage control system and feedwater leakage control system are manually started consistent with the licensee's design basis assumptions. The licensee entered this performance deficiency into their corrective action program as Condition Report CR-GGN-2017-09112. In response to this issue the licensee has provided specific guidance and training to the operators. Because this finding was of very low safety significance and has been entered into the licensee's corrective action program, this violation is being treated as a non-cited violation, consistent with Section 2.3.2.a of the NRC Enforcement Policy: NCV-05000416/2017007-06, "Failure to Ensure Adequate Design Control Measures Are in Place Associated with Leakage Control Systems."

4. OTHER ACTIVITIES

4OA2 Problem Identification and Resolution

The team reviewed condition reports associated with the selected components, operator actions, and operating experience notifications. Any related findings are documented in prior sections of the report.

4OA6 Meetings, Including Exit

Exit Meeting Summary

On September 29, 2017, the inspectors presented the preliminary inspection results to Mr. E. Larson, Site Vice President, and other members of the licensee staff. The licensee acknowledged the issues presented. The licensee confirmed that any proprietary information reviewed by the inspectors had been returned or destroyed.

On October 18, 2017, the inspectors presented the final inspection results, via telephone, to Mr. E. Larson, Site Vice President, and other members of the licensee staff. The licensee acknowledged the issues presented. The licensee confirmed that any proprietary information reviewed by the inspectors had been returned or destroyed.

4OA7 Licensee-Identified Violations

- The following violations of very low safety significance (Green) were identified by the licensee and are violations of NRC requirements which meet the criteria of the NRC Enforcement Policy for being dispositioned as non-cited violations.

  Technical Specification 5.4.1(a) requires written procedures to be established, implemented, and maintained as recommended by Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Section 4.e recommends, in part, instructions for startup of shutdown cooling and reactor vessel head spray system be prepared. Contrary to the above, from about 2004 until September 1, 2017, the 04-1-01-E12-2 instruction failed to provide instruction for placing the alternate decay heat removal system in service. Specifically, Step 4.9.2a.7(d) instructs an operator to, “Manually control component cooling water temperature by throttling P44-F010A(B)(C), PSW inlet to CCW HXs.” However, the purpose of that step is to throttle plant service water flow through the alternate decay heat removal system and component cooling water system to ensure both systems have plant service water flow, which is not accomplished by the instruction step. The licensee identified this procedural violation before the system was credited for availability during an inservice demonstration on September 1, 2017, and entered it in the corrective action program as Condition Report CR-GGN-2017-08643. The violation is of very low safety significance (Green) because, although the procedure did delay placing the system in service due to the procedure error, the system was capable of performing it’s design function, consistent with Inspection Manual Chapter 0609, Appendix G, Attachment 1, Exhibit 3 screening.

- 10 CFR 50 Appendix B, Criterion III, requires in part, “That measures shall be established to assure that the design bases are correctly translated into specifications, drawings procedures, and instructions.” Contrary to the above, from original plant construction until June 22, 2016, GGNS failed to ensure the design basis tornado and differential pressures associated with it, would not cause a spurious trip of the Division I and II standby diesel generators. Specifically, a design basis tornado, includes a differential pressure of 3.0 psig, whereas an active diesel generator trip on high crankcase pressure actuates at 1.5 psig. The licensee identified this issue using an effective operating experience program and entered it in the corrective action program as Condition Report CR-GGN-2016-04919. The violation is of very low safety significance (Green), for the same reason as NCV-05000416/2017007-05, discussed in Section 1R21.4.5 of this report.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

E. Larson - Site Vice President
G. Hawkins - Director, Regulatory and Performance Improvement
P. Williams - Director, Engineering
D. Ellis - Acting Manager, Regulatory Assurance
R. Meister - Senior Specialist, Regulatory Assurance
J. Hallenbeck - Manager, Design and Program Engineering
L. Hendrick - Supervisor, Engineering
F. Hopkins - Temporary Manager, Design Engineering
T. Wallace - Engineer, Design Engineering
T. Robinson - Engineer, Design Engineering
T. Wallace - Engineer, Design Engineering

NRC Personnel

M. Young, Senior Resident Inspector

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

05000416/2017007-01	NCV	Failure to Initiate Engineering Seismic Evaluation for the Construction of Long Term Scaffolding in the Div I and Div II Diesel Rooms (Section 1R21.2.2)
05000416/2017007-02	NCV	Failure to Correct Standby Diesel Generator Trip (Section 1R21.2.2)
05000416/2017007-03	NCV	Failure to Establish a Preventive Maintenance Procedure for Safety-Related Equipment (Section 1R21.2.3)
05000416/2017007-04	NCV	Failure to Update the Final Safety Analysis Report (Section 1R21.2.6)
05000416/2017007-05	NCV	Failure to Update a Calculation and Procedure to Address Standby Service Water Passive Failure (Section 1R21.3.5)
05000416/2017007-06	NCV	Failure to Ensure Adequate Design Control Measures Are in Place Associated with Leakage Control Systems (Section 1R21.4.5)

LIST OF DOCUMENTS REVIEWED

Calculations

Number	Title	Revision/Date
XC-Q1Y41-92007	SSW Pump House Room Temperatures for a LOCA and for SSW Pump House Cooling Inoperable	0
XC-Q1111-92010	LOCA Dose Analysis	5, 6
XC-Q1111-98017	LOCA Dose Analysis with Revised Source Terms	5
3.6.25	Control Room Doses and Inleakage Rates (N.S. Calc. No. 7.6.5-42-9645, Rev. 0)	1, 2
XC-Q1B21-94006	Feedwater Evaporation during DBA LOP/LOCA	0
MC-Q1P41-11001	GGNS Standby Service Water Ultimate Heat Sink Thirty Day Performance at EPU	0, EC69026
MC-Q1P41-08014	SSW Loop 3 Restricting Orifice Calculation	3
MC-Q1P41-90138	SSW Loop C Restricting Orifice Calculation	0
MC-Q1P41-97020	Determination of Minimum Allowable SSW Flows to Safety-Related Heat Exchangers	12
MC-Q1E22-00010	HPCS and RCIC System Performance with Regards to CST and Suppression Pool Level Transmitters	3
EC-Q1R20-91042	Div III 480/120VAC Class 1E CPT Circuit Voltage Drop Study	0
EC-Q1111-93001	Control Building Electrical Heat Load Calculation	3
E-DCP82/5020-1	Transient Diesel Loading	July 26, 1985
MC-Q1P75-15001	Div I and II Maximum Load Reject Pump Load	0
EC-Q1L21-90047	Sizing of 125VDC Div II Battery & Chargers	2
EC-Q1L21-90046	Div II 125VDC Class IE Voltage Drop Study	2
Q1111-90028	AC Electrical Power System Calculation	6
MC-Q1P41-11001	GGNS Standby Service Water Ultimate Heat Sink Thirty Day Performance at EPU	0
2.2.59-Q	SSW System – SFD Calculation	A

Drawings

Number	Title	Revision
M-1112	Updated Final Safety Analysis Report Figure Number – 6.7-005, P&I Diagram, Feedwater Leakage Control System	9
C143.0-N1P11A002-1.3-002	Condensate Storage Tank	2
M-1065	Condensate & Refueling Water Storage	50
E-0001	Main One Line Diagram	52
E-1009	One Line Meter and Relay Diagram 4.16KV ESF System Bus 17AC, Unit1	11
E-1008	One Line Meter and Relay Diagram 4.16KV E.S.F System Bus 15AA & 16AB, Unit 1	22
E-1091	MCC Tabulation 480 V. MCC 17B01, Control Building	24
A-0113	Control BLDG – Switchgear RMs, Fl. Plan at El. 111’ 0	16
E-1188-016	Schematic Diagram E22, HPCS Power Supply System, Breaker No. 4	10
E-1188-017	Schematic Diagram E22, HPCS Power Supply System, Transformer CKT, Aux. Compartment	11
E-1188-018	Schematic Diagram E22, HPCS Power Supply System, Breaker No. 1, Unit 1	12
E-1188-019	Schematic Diagram E22, HPCS Power Supply System, Breaker No. 2, Unit 1	11
E-1188-020	Schematic Diagram E22, HPCS Power Supply System, Breaker No. 3	11
E-1188-021	Schematic Diagram E22, HPCS Power Supply System, Breaker No. 5	12
E-1188-022	Schematic Diagram E22, HPCS Power Supply System, Breaker No. 6	10
E-0121-17	R25 Summary of Relay Settings (ESF) 4.16KV Bus 17AC and Diesel Gen 13, Unit 1	0
J-1261-012	HPCS Diesel Generator System Initiation Logic	0
J-1261-015	HPCS Diesel Generator System Power Supply Miscellaneous Annunciators	0

Drawings

Number	Title	Revision
J-1261-016	HPCS Diesel Generator System Power Supply Miscellaneous Annunciators	1
J-1261-017	HPCS Diesel Generator System Power Supply Miscellaneous Annunciators	0
J-1248-001	Logic Diagram HPCS Initiation Logic	2
J-1248-002	Logic Diagram HPCS Pump	2
E-1110-012	Schematic Diagram P75 Stand-by Diesel Generator Sys. Div. I train “A” Start & Stop Circuit	21
E-1110-028	Schematic Diagram P75 Stand-by Diesel Generator Sys. Diesel Governor Setting Control	9
E-0110-02	4.16KV BOP System, Incoming Breaker 152-1902, Units 1 & 2	7
M-1061A	SSWS	68
M-1061B	SSWS	52
M-1061C	SSWS	38
M-1061D	SSWS	40
C-1734	Units 1 & 2 SSW Cooling Tower Basin Reinforced Concrete Wall Sections & Details	13
C-1733	Units 1 & 2 SSW Cooling Tower Basin Reinforced Concrete Wall Sections & Details	1
C-1736D	Units 1 & 2 SSW Cooling Tower Basin Miscellaneous Steel Plans Sections & Details	3
M-018.0-Q1P75E001A	Delaval Control Panel Schematic	16
E-1111-013	Schematic DG Sys Div II train B Start Circuit	18
E-1043	Logic Diagram ESF Div II Diesel	10
E-1120-004	R21 Load Shedding & Sequencing Sys Div 2 Part 2	15
E-1120-003	R21 Load Shedding & Sequencing Sys Div 2 Part 1	15
E-1109-024	Schematic 4.16KV ESF Diesel Gen Breaker 152-1608	15

Drawings

Number	Title	Revision
GFIG-OPS-E3800	Feedwater Leakage Control System (E38) Figures	0
GFIG-OPS-E3200	MSIV Leakage Control System (E32) Figures	0
GFIG-OPS-P4100	Standby Service Water (SSW) System – Figures	0
GFIG-OPS-Y4700	SSW Pump House Ventilation System – Y47 Figures	2

Procedures

Number	Title	Revision/Date
02-S-01-44	Time Critical Actions	0
05-1-02-I-4	Loss of AC Power	50,51
04-1-01-Y47-1	Standby Service Water Ventilation System	102
04-1-02-1H13-P870	Alarm Response Instruction, Panel No. 1H13-P870	154
04-S-02-SH13-P807	Alarm Response Instruction, Panel No. SH13-P807	32
05-1-02-I-1	Reactor Scram	130
05-1-02-III-3	Reduction in Recirculation System Flow Rate	115
04-1-01-E38-1	Feedwater Leakage Control System	101
04-1-01-R21-11	BOP Bus 11HD	35
04-1-01-R21-12	BOP Bus 12HE	40
04-1-01-R21-3	BOP Bus 13AD	32
04-1-01-R21-14 SU	BOP Bus 14AE	26
EN-LI-102	Corrective Action Program	30
01-S-02-1	Description and Use of the GGNS Operations Manual	34
01-S-02-9	Procedure Change Process	6

Procedures

Number	Title	Revision/Date
01-S-06-37	Revision and Control of Emergency Procedures and Severe Accident Procedures	6
EN-AD-106	Site Procedure Writer's Manual	0
EN-HU-106	Procedure and Work Instruction Use and Adherence	4
EN-OP-200	Plant Transient Response Rules	3
05-1-02-III-12	Standby Service Water Basin Level Control	0
01-S-18-6	Risk Assessment of Maintenance Activities	18
EN-WM-104	On Line Risk Assessment	15
GGNS-94-0052	GGNS Engineering Report for Evaluation of Containment Leak Paths	0
04-1-01-E32-1 SU	Main Steam Isolation Valve-Leakage Control System	28
GGNS-CS-05	Civil Standard for Erection of Scaffolding in Seismic Category I Buildings	4
EN-MA-133	Control of Scaffolding	17
04-1-01-R21-1	System Operating Instruction-Load Shedding and Sequencing System	106
EN-OP-104	Operability Determination Process	12
06-EL-1L11-R-0003	125-Volt Battery Bank Service Discharge Test Div II, 1B3 Battery	107
06-EL-1L21-O-0001	Battery 1B3 Performance Discharge Test	108
EN-DC-115	Engineering Change Process	21
06-OP-1P75-R-0004	SDG12, Functional Test Div. 2 LOP/LOCA	125
EN-DC-132	Control of Engineering Documents	7
EN-LI-100	Process Applicability Determination	20
07-S-12-40	General Cleaning and Inspection of Rotating Electrical Equipment	3

Procedures

Number	Title	Revision/Date
06-OP-1P41-Q-0004	Standby Service Water Loop A Valve and Pump Operability Test	125
GGNS-CS-17	Standard for Criteria for Prevention of Potentially Hazardous Seismic II/I Situations Due to Loose Items (Non-Safety Related)	10
04-1-01-P41-1	SSWS	143
04-1-01-R21-17 SU	System Operating Instruction ESF Bus 17AC	10
06-OP-1P41-Q-0004	Standby Service Water Loop A Valve and Pump Operability Test	125
07-S-12-61	Inspection of GE Magne Blast Circuit Breakers	7
EN-OP-104	Operability Determination Process	11
EN-MP-115	Material Issues and Returns	5
06-EL-1P81-R-001	Surveillance Procedure ESF Div. 3 Bus Under voltage and Time delay Relay Calibration Safety-Related	103
07-S-12-39	General Cleaning and Inspection of Non-Rotating Electrical Equipment	14
01-S-06-26	Post-Trip Analysis SCRAM No. 143	April 4, 2017
05-1-02-III-12	Standby Service Water Basin Level Control	0
EN-SA-G-001	Identification and Documentation of Time Critical Actions	0
05-1-02-VI-2	Hurricanes, Tornados, and Severe Weather	135

Design Change Packages

Number	Title	Revision/Date
EC-47972	Increasing the Orifice Size for the P41D013 to Increase SSW Flow to the HPCS Room Cooler and HPCS Diesel Generator Jacket Water Coolers	0
EC-25649	GGNS EPU SSW UHS Siphon Line Extension	0
EC-073401	SSW A Pump Re-Baseline after Changing Flow Instrumentation from 1P41N018A and 1P41N016A	0
EC-073289	Operability Input for CR-GGN-2017-07237, SWW “A” Pump Motor Degraded PI Ration	0

Design Change Packages

Number	Title	Revision/Date
EC-071386	Remove Diesel Diodes per 10 CFR Part 21 Recommendation (Div 1 and Div 2) CR-GGN-2017-3643	0
EC-02065	Div 1 Diesel Generator Governor	0
EC-02048	Div 2 Diesel Generator Governor	0
EC-02113	Replace E12F025C – RHR C Pump PSV	0
EC-43357	Temporary Scaffolding Diesel Rooms	March 15, 2013
EC-17595	Transient Loading on Diesel Generators During Load	July 19, 2010
EC-02202	Replacement of Div 2 Diesel Generator Start Circuit Time Delay Relays TD2A & TD2B	1
EC-74267	SSW System and UHS Evaluation	0

Vendor Documents

Number	Title	Revision/Date
460004160	Standby Service Water (SSW) Vertical Pumps Installation, Operation, Maintenance, and Storage Manual	0
460000163	Vendor Manual for Switchgear E22S004	January 31, 2001
460004314	Sun Pumps, Inc. - Operation and Installation Manual SDSP Series Solar Electric Pump & PCA-30-MI Series Pump Controller	October 14, 2010
460002250	P45 Man-Hole Sump Pumps NSP45CO45A	September 12, 1995
460000450	Div II EDG Instruction Manual	1
460000453	Div II EDG Publications Manual Vol. 3 Book 2	301

Design Basis Documents

Number	Title	Revision
SDC-X77	Diesel Generator Building Ventilation System	2
SDC-P75	Standby Diesel Generator System	1
SDC-P41	SSWS (P41)	5

Design Basis Documents

Number	Title	Revision
SDC-P75	Standby Diesel Generator System	1
SDC-01	125-Volt DC Class 1E Distribution System, Div I & II	1
SDC-E38	Feedwater Leakage Control System (E38)	1
SDC-E32	Main Steam Isolation Valve-Leakage Control System (E32)	3

Other

Number	Title	Revision/Date
GGNS-NE-16-00004	Time Critical Operator Actions for GGNS	0
PRA-GG-01-001S03	GGNS Human Reliability Analysis/Rule Recovery Work Package	0
XC-Q1N21-940010	Dose from Feedwater System Leakage During a DBA LOP/LOCA	0
LBDCR 2016-037	UFSAR Changes – Sections 6.7.2, 15.8.1	April 12, 2016
STNA-037	Panel H13-P871 & P872 ESF Logic VB is Not in the Simulator	July 30, 2013
GGNS-NE-10-00004	GGNS EPU Anticipated Transient Without Scram	1
GGNS-NE-12-00025	GGNS MELLLA+ Anticipated Transient Without Scram	0
GNRI-2015/00114	GGNS, Unit 1 – Issuance of Amendment Regarding Maximum Extended Load Line Limit Analysis Plus (TAC No. MF2798)	August 31, 2015
Y47-FO-HEMOD-U	GGNS PSA – Failure to Install Alternate Means of Cooling to SSW Pump House	March 5, 2009
R21-FO-HEBOTRM	GGNS PSA – Failure to Align Alternate Power to 4.16kV or 6.9 kV Buses	March 5, 2009
M-264-92	Temperatures in SSW Pump Room Analysis	August 7, 1992
GNRO-2015/00094	Maximum Extended Load Limit Analysis Plus (MELLLA+) Time Critical Operator Action Training Results, GGNS, Unit 1; Docket No. 50-416, License No. NPF-29	March 2, 2016

Other

Number	Title	Revision/Date
95-0019-R02	Engineering Report GGNS-94-0039, Rev. 1, “Evaluation of Changes to the GGNS Feedwater Isolation Valve-Leak Testing Methodology”	July 11, 1997
GLP-OPS-E3800	Feedwater Leakage Control System – E38	5,6
GLP-OPS-P4100	SSWS	8
GLP-OPS-Y4700	SSW Pump House Ventilation System – Y47	9
AECM-76/27	GGNS Units 1 & 2, Docket Nos. 50-416/417, File 0272/6205/B727, Ref: Letter, Dr. Walter R. Butler, to Mr. N. L. Stampley, of June 19, 1975	June 7, 1976
MAEC-76/48	GGNS Main Steam Line Isolation Valve-Leakage Control System	October 6, 1976
NUREG-0926-Rev. 1	Technical Specifications, GGNS, Unit 1, Docket No. 50-416, Appendix “A” to License No. NPF-13	August 1984
95-0019-R02	Engineering Report GGNS-94-0039, Rev. 1, “Evaluation of Changes to the GGNS Feedwater Isolation Valve-Leak Testing Methodology”	June 11, 1997
96-01935	Bechtel Telecon 0750/0751/L-334.0	August 1, 1996
NEDC-32988-A	Technical Justification to Support Risk-Informed Modification to Selected Required Action End States for BWR Plants	2
NUREG-0831	Safety Evaluation Report Related to the Operation of GGNS, Units 1 and 2, Docket Nos. 50-416 and 50-417	September 1981
GNRI-2001/00032	GGNS, Unit 1 – Issuance of Amendment RE: Full-Scope Implementation of an Alternate Accident Source Term (TAC No. MA8065)	March 14, 2001
GGNS-94-0039	Evaluation of Changes to GGNS Feedwater Isolation Valve-Leak Testing Methodology	3
Standing Order No. 1709921	MSIV Leakage Control System and FW Leakage Control System Time Critical Actions	September 29, 2017
Scaffolding Request 13-5834	Scaffolding Evaluation Div I Dsl Gen Room	March 17, 2003
IEEE 344	Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations	1971, 1975

Other

Number	Title	Revision/Date
IEEE 387	Criteria for Diesel-Generator Units Applied as Standby Power Supplies for Nuclear Power Generating Stations	1977, 1984
M018.0	Standby Diesel Generators Design Specification	May 20, 1978
P-75	System Health Report Div 1 & 2 Standby Diesel Generator	Q2-2017
L-11	System Health Report ESF 125V Battery	Q1-2017
QR-07711303-1	Qualification Testing Agastat 2112D4YF Time Delay Relay	0
GIN 97-01830	Loose Item Evaluation for Corrosion Rack and Deposit Monitor in SSW Valve Rooms A & B	September 23, 1997
	2016 GGNS SSW Inspection	January 26, 2017
2011-023	Licensing Basis Document Change Request	September 29, 2011
PMRQ 50020825-06/50020826-04	Inspect & Vacuum SSW Basin	July 9, 2016
9645-E-017-0	Technical Specification for Load Center Unit Substations for Mississippi Power and Light Company GGNS, Units 1 & 2	14
IB-11.1.7-2	Unit Substation Transformers Instructions, Installation-Maintenance, Ventilated Dry-Transformer Type VU-9	A
IEEE Std. C57.94	IEEE Recommended Practice for Installation, Application, Operation, and Maintenance of Dry-Type Distribution Transformer	December 5, 2015
SEP-THERM-GGN-001	GGNS Infrared Thermography Program Section	1
Amendment 196	Revise the Standby Service Water Passive Failure Methodology in the UFSAR	September 25, 2013
LDC 2001-157	UFSAR Consistency Review for the SSW System	November 27, 2001
OE-NOE-2009-00251	NRC Information Notice 2009-09 Response	September 29, 2009
GNRO-2012/00102	Request to Revise the Standby Service Water Passive Failure Methodology	September 14, 2012

Condition Reports

CR-GGN-2015-07209	CR-GGN-2017-00548	CR-GGN-2017-09746
CR-GGN-2017-09747	CR-GGN-2017-09750	CR-GGN-2017-09755
CR-GGN-2017-09758	CR-GGN-2017-09762	CR-GGN-2006-00680
CR-GGN-2006-01377	OE-NOE-2006-00303	CR-GGN-2016-08329
CR-GGN-2016-08295	CR-GGN-2017-01559	CR-GGN-2017-03052
CR-GGN-2017-02952	CR-GGN-2016-06414	CR-GGN-2016-02027
CR-GGN-2012-06208	CR-GGN-2012-06181	CR-GGN-2015-05057
CR-GGN-2017-09269	CR-GGN-2016-04781	CR-GGN-2016-09055
CR-GGN-2017-07408	CR-GGN-2017-07237	CR-GGN-2017-06021
CR-GGN-2017-04056	CR-GGN-2016-05329	CR-GGN-2016-08398
CR-GGN-2017-03643	CR-GGN-2017-03964	CR-GGN-2016-09050
CR-GGN-2016-08325	CR-GGN-2017-06021	CR-GGN-2017-04649
CR-GGN-2017-07850	CR-GGN-2017-03529	CR-GGN-2012-12003
CR-GGN-2012-09267		

Condition Reports Generated During this Inspection

CR-GGN-2017-09662	CR-GGN-2017-09064	CR-GGN-2017-09154
CR-GGN-2017-09671	CR-GGN-2017-09195	CR-GGN-2017-09692
CR-GGN-2017-09658	CR-GGN-2017-09452	CR-GGN-2017-09763
CR-GGN-2017-09728	CR-GGN-2017-10192	CR-GGN-2017-10192
CR-GGN-2017-09149	CR-GGN-2017-09155	CR-GGN-2017-09390
CR-GGN-2017-09273	CR-GGN-2017-09694	CR-GGN-2017-09148
CR-GGN-2017-01356	CR-GGN-2017-09747	CR-GGN-2017-09645
CR-GGN-2017-09557	CR-GGN-2017-09011	CR-GGN-2017-09045
CR-GGN-2017-09112	CR-GGN-2017-09734	

Work Orders

52406242	00308901	52566579	52594761	52761204
52743923	52730530	52489450	52568492	00419008
00194704	00422140	00462338	00302267	00450578
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