UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
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ARLINGTON, TEXAS 76011-4125

October 27, 2008

Michael Perito
Vice President, Operations
Entergy Operations, Inc.
River Bend Station
5485 US Highway 61N
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SUBJECT: RIVER BEND STATION - NRC COMPONENT DESIGN BASES INSPECTION REPORT 05000458/2008006

Dear Mr. Perito:

On August 26, 2008, the US Nuclear Regulatory Commission (NRC) completed an inspection at your River Bend Station. The enclosed report documents the inspection findings, which were discussed on August 26, 2008, with you and other members of your staff.

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

This report documents six NRC-identified findings of very low safety significance (Green). All of these findings were determined to involve violations of NRC requirements. Some of the violations had multiple examples. However, because of the very low safety significance of the violations and because they were entered into your corrective action program, the NRC is treating these violations as non-cited violations (NCV) consistent with Section VI.A.1 of the NRC Enforcement Policy. If you contest any NCV in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the US Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region IV, 612 East Lamar Blvd., Suite 400, Arlington, Texas 76011-4125; the Director, Office of Enforcement, US Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the River Bend Station Nuclear Power Plant.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records component of NRC's document system. ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Russell L. Bywater, Chief
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Division of Reactor Safety

Docket: 50-458
License: NPF-47

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Inspection Report 05000458/2008006

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SUNSI Review Completed: ___Y__ ADAMS: √ Yes □ No Initials: _JFD__
√ Publicly Available □ Non-Publicly Available □ Sensitive √ Non-Sensitive

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SRI: EB1	SRI: EB1	SOE: EB1	RI: EB1	RI: EB1	C: PBC	C: EB1
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U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket:	50-458
License:	NPF-47
Report:	05000458/2008006
Licensee:	Entergy Operations, Inc.
Facility:	River Bend Station
Location:	5485 U.S. Highway 61 St. Francisville, LA
Dates:	April 28 through August 26, 2008
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Approved By:	Russell L. Bywater, Chief Engineering Branch 1 Division of Reactor Safety

SUMMARY OF FINDINGS

IR 05000458/2008006; 04/28/2008 - 08/26/2008; River Bend Station; Baseline Inspection, NRC Inspection Procedure 71111.21, Component Design Bases Inspection

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

A. NRC-Identified and Self-Revealing Violations

Cornerstone: Mitigating Systems

- Green. The team identified a finding of very low safety significance involving a noncited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," with eight examples.

  - Example 1: Non-conservative inputs and assumptions used without adequate technical justification to evaluate the minimum terminal voltage and actuator output torque for safety-related motor operated valves. After identification, the licensee entered the issue into the corrective action program as Condition Report CR-RBS-2008-03339.

  - Example 2: Failure to perform a conservative analysis to ensure that Technical Specification Setpoints were adequate. After identification, the licensee entered the issue into the corrective action program as Condition Report CR-RBS-2008-03911.

  - Example 3: Non-conservative inputs and methodologies used in calculating control circuit voltages to safety-related 480V motor operated valves motor-operated valve and motors that would be required to operate for mitigation of design bases events. After identification, the licensee entered the issue into the corrective action program as Condition Report CR-RBS-2008-03858.

  - Example 4: Failure to evaluate E12-MOV-F042A, residual heat removal injection valve, and E12-MOV-F064A, residual heat removal minimum flow valve, to verify adequate voltage would be available to operate the associated 120VAC control circuit devices. After identification, the licensee entered the issue into the corrective action program as Condition Report CR-RBS-2008-03641

  - Example 5: Inadequate design basis documentation for hydrogen concentration control in the Division I and II Battery Rooms in the control building. After identification, the licensee entered the issue into the corrective action program as Condition Reports CR-RBS-2008-02566 and CR-RBS-2008-03403.

- Example 6: Failure to ensure design basis information for safety related 125VDC batteries was controlled and correctly translated into procedures and instructions. After identification, the licensee entered the issue into the corrective action program as Condition Report CR-RBS-2008-03659.

- Example 7: Failure to maintain adequate design basis calculations for ultimate heat sink loading. After identification, the licensee entered the issue into the corrective action program as Condition Report CR-RBS-2008-3712.

- Example 8: Failure to account for the technical specification allowed emergency diesel generator frequency variation in the diesel loading calculation. After identification, the licensee entered the issue into the corrective action program as Condition Report CR-RBS-2008-03556.

The examples associated with this finding were more than minor per Manual Chapter 612, Appendix E, Appendix E, “Examples of Minor Issues,” Example 3j, in that each example resulted in a condition where there was reasonable doubt on the operability of a system or component. The finding was associated with the design control attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Traditional enforcement does not apply because the issue did not have any actual safety consequences or potential for impacting the NRC’s regulatory function, and was not the result of any willful violation of NRC requirements. In accordance with Inspection Manual Chapter 0609, Attachment 4, “Phase 1 – Initial Screening and Characterization of Findings,” a Phase 1 screening was performed and determined each example was of very low safety significance (Green) because it was a design or qualification deficiency confirmed not to result in a loss of operability or functionality. (Section 1R21.b.1)

- <u>Green.</u> The team identified a finding of very low safety significance involving a noncited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” in that, design control measures for verifying the adequacy of design were not implemented. Specifically, the licensee did not recalculate suppression pool peak temperature response when a more severe single failure condition was identified. In response, the licensee entered this issue in the corrective action program as Condition Report CR-RBS-2008-03661 and determined that suppression pool peak temperature response was acceptable.

  The finding was more than minor because it was associated with the design control attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Traditional enforcement does not apply because the issue did not have any actual safety consequences or potential for impacting the

NRC's regulatory function, and was not the result of any willful violation of NRC requirements. In accordance with Inspection Manual Chapter 0609, Attachment 4, "Phase 1 – Initial Screening and Characterization of Findings," a Phase 1 screening was performed and determined the finding was of very low safety significance (Green) because it was a design or qualification deficiency confirmed not to result in a loss of operability or functionality of the suppression pool. The finding had a cross-cutting aspect in the area of problem identification and resolution because the licensee initiated a corrective action program action to re-evaluate long-term suppression pool peak temperature performance but closed the action without its completion. (P.1 (d)) (Section 1R21.b.2)

- Green. The team identified a finding of very low safety significance involving a noncited violation of 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," with three examples. Specifically, the team identified that the licensee failed to develop and implement adequate testing programs for 4-kV circuit breakers, Class 1E molded-case circuit breakers, and the emergency diesel generators that met design or vendor requirements and recommendations. In response, the licensee entered these examples in the corrective action program as Condition Reports CR-RBS-2008-04379, CR-RBS-2008-3634, CR-RBS-2008-3676 and CR-RBS-2008-3701 and determined there was no loss of safety function for the affected components.

  The examples associated with this finding were more than minor because they were associated with the equipment control attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Traditional enforcement does not apply because the issue did not have any actual safety consequences or potential for impacting the NRC's regulatory function, and was not the result of any willful violation of NRC requirements. In accordance with Inspection Manual Chapter 0609, Attachment 4, "Phase 1 – Initial Screening and Characterization of Findings," a Phase 1 screening was performed and determined each example was of very low safety significance (Green) because it did not represent a loss of system safety function, did not represent an actual loss of safety function of a single train for greater than its Technical Specification allowed outage time, did not represent an actual loss of one or more risk-significant non-Technical Specification trains of equipment for greater than 24 hours, and did not screen as potentially risk-significant due to seismic, flooding, or severe weather. (Section 1R21.b.3)

- Green. The team identified a finding of very low safety significance involving a noncited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for five examples of failure to follow the requirements of ADM-0073 "Temporary Installation Guidelines" during the installation of modifications to the plant. Specifically, four modifications were installed in the plant that did not meet the criteria of a temporary installation and one was not removed when no longer needed, as required by the procedure. After identification, the licensee entered the issue into the corrective action program as CR-RBS-2008-3410.

Although the team considered each of the above examples minor in significance, the team determined that this finding, which was associated with design control attribute of the Mitigating Systems cornerstone, was more than minor per Manual Chapter 612, Appendix E, "Examples of Minor Issues," Example 4a. The finding involved multiple examples of failure to follow licensee procedural requirements and if left uncorrected it could result in design modifications to the plant that were not properly evaluated, controlled, documented and installed. Traditional enforcement does not apply because the issue did not have any actual safety consequences or potential for impacting the NRC's regulatory function, and was not the result of any willful violation of NRC requirements. In accordance with Inspection Manual Chapter 0609, Attachment 4, "Phase 1 – Initial Screening and Characterization of Findings," a Phase 1 screening was performed and determined the finding was of very low safety significance (Green) because the condition did not represent a loss of system safety function, did not represent an actual loss of safety function of a single train for greater than its Technical Specification allowed outage time, did not represent an actual loss of one or more risk-significant non-Technical Specification trains of equipment for greater than 24 hours, and did not screen as potentially risk-significant due to seismic, flooding, or severe weather. The finding had a crosscutting aspect associated with resources in the human performance area because the licensee failed to ensure that personnel, equipment, procedures, and other resources were available and adequate to assure nuclear safety. Specifically, those necessary for maintaining long term plant safety by maintenance of design margins, minimization of long-standing equipment issues, minimizing preventative maintenance deferrals, and ensuring maintenance and engineering backlogs which were low enough to support safety. [H.2 (a)] (Section 1R21.b.4)

- Green. The team identified a finding of very low safety significance involving a noncited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for the failure to follow procedures to evaluate conditions adverse to quality for impacts on the operability of safety-related equipment. Specifically, the licensee did not assess the impact on operability of previous steam leaks and motor-stall events on the corrosion of magnesium-rotors in safety-related motor-operated valves. The licensee entered this issue into the corrective action program as Condition Reports CR-RBS-2008-3713 and CR-RBS-2008-3766.

  The finding was more than minor because it is associated with the equipment performance attribute of the Mitigating Systems cornerstone objective of ensuring the availability, reliability, and capability of safety-related motor-operated valves to respond to initiating events to prevent undesirable consequences. Traditional enforcement does not apply because the issue did not have any actual safety consequences or potential for impacting the NRC's regulatory function, and was not the result of any willful violation of NRC requirements. In accordance with Inspection Manual Chapter 0609, Attachment 4, "Phase 1 – Initial Screening and Characterization of Findings," a Phase 1 screening was performed and determined the finding was of very low safety significance (Green) because the condition did not represent a loss of system safety function, did not represent an actual loss of safety function of a single train for greater than its Technical Specification allowed outage time, did not represent an actual loss of one or more risk-significant non-Technical

Specification trains of equipment for greater than 24 hours, and did not screen as potentially risk-significant due to seismic, flooding, or severe weather. The cause of the finding had crosscutting aspects associated with the corrective action program in the problem identification and resolution area because the licensee did not thoroughly evaluate the problems with magnesium-rotor corrosion including the extent of the condition and operability impact. [P.1(c)] (Section 4OA2.b.1)

Cornerstone: Barrier Integrity

- Green. The team identified a finding of very low safety significance involving a noncited violation of 10 CFR Part 50 Appendix B, Criterion XVI, "Corrective Action," for failure to promptly identify magnesium-rotor motor-operated valve degradation. Specifically, the licensee did not identify magnesium-rotor degradation in May 2007 after failure of Valve B21-MOV-FO65A, "Reactor Inlet Heater 'A' Outboard Motor Operated Isolation Valve," until after failure of Valve B21-MOV-FO98C, "Main Steam Shutoff Valve," in September 2007. The licensee entered this issue into the corrective action program as Condition Reports CR-RBS-2008-3713 and CR-RBS-2008-3766.

  This finding was more than minor because Valve B21-MOV-FO98C was associated with the Barrier Integrity Cornerstone and affected the cornerstone objective of providing reasonable assurance that the physical design barriers (fuel cladding, reactor coolant system, and containment) protect the public from radionuclide releases caused by accidents or events. Traditional enforcement does not apply because the issue did not have any actual safety consequences or potential for impacting the NRC's regulatory function, and was not the result of any willful violation of NRC requirements. Inspection Manual chapter 0609 Appendix H, "Containment Integrity Significance Determination Process," Table 4.1, indicated that the Main Steam Shutoff Valves do not impact large early release frequency. Based on the results of the Appendix H analysis, the finding was determined to have very low safety significance. This finding had cross-cutting aspects associated with decision-making in the human performance area in that the licensee did not use conservative assumptions in decision-making regarding the likelihood of magnesium-rotor degradation in motor-operated valves. [H.1 (b)] (Section 4OA2.b.2)

B. Licensee-Identified Violations

None

REPORT DETAILS

1. REACTOR SAFETY

Inspection of component design bases verifies the initial design and subsequent modifications and provides monitoring of the capability of the selected components and operator actions to perform their design bases functions. As plants age, their design bases may be difficult to determine and an important design feature may be altered or disabled during a modification. The plant risk assessment model assumes the capability of safety systems and components to perform their intended safety function successfully. This inspectable area verifies aspects of the Initiating Events, Mitigating Systems and Barrier Integrity cornerstones for which there are no indicators to measure performance.

1R21 Component Design Bases Inspection (71111.21)

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

a. Inspection Scope

To verify that the selected components would function as required, the team reviewed design bases assumptions, calculations, and procedures. In some instances, the team performed independent calculations to verify the appropriateness of the licensee engineers' analysis methods. 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 information 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 associated with the selected components, as well as, observing 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 modification, or margin reductions identified because 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 inspectors' input of problem equipment; system health reports; industry operating experience; and licensee problem equipment lists. Consideration was also given to the uniqueness and complexity of the design, operating experience, and the available defense-in-depth margins.

The inspection procedure requires a review of 20-30 risk-significant and low design margin components, 3-5 relatively high-risk operator actions, and 4-6 operating experience issues. The sample selection for this inspection was 23 components, 4 operator actions, and 6 operating experience items.

The components selected for review were:

- High-Pressure Core Spray Pump Min Flow Valve, HPCS-E22-MOV-FO12
- Residual Heat Removal Pump Min Flow Valve, RHR-E12-MOV-FO64A
- Reactor Core Isolation Cooling Steam Inlet Valve, RCIC-E51-MOV-FO45
- Low-Pressure Core Injection Train A Outboard Isolation Valve, E12-MOV-F042A
- Containment Cooling Fan A, HVR-UC1A
- Residual Heat Removal Pump B
- Full Flow Test Return and Suppression Pool Cooling Valve, RHR-E12-MOV-F024A
- Condensate Storage Tank Level Instrumentation
- High-Pressure Core Spray Emergency Diesel Generator, E22-EDS001
- Emergency Diesel Generator Division II Air and Fuel Consumption Calculations
- Standby Service Water Inlet to Cooling Tower, SWP-MOV-FO55A
- Transformer RTX-XSR-1C
- High-Pressure Core Spray system thermal relief valve, E22-RV-F035
- 4160 Volt Switchgear, ENS-SWG-1A
- Emergency Diesel Generator Division I EDG-1A and Feeder Breakers
- 480 Volt Switchgear, EJS-SWG-2A and EJS-X2A
- Vital Station Battery Div II, BAT01B
- Vital Battery Charger, ENB CHGR 1B
- Vital Uninterruptible Power Supply, Div II
- Division I 125 Volt dc Switchgear, 2ENB*SWG2A
- Station Blackout Emergency Diesel Generator
- 480 Volt Motor Control Center, EHS-MCC2E

- Service Water Outlet from Residual Heat Removal Heat Exchanger, RHR-E12-MOV-FO68A

The selected operator actions were:

- Hook-up Station Blackout Emergency Diesel Generator
- Start Standby Cooling Tower Fans
- Standby Liquid Control System Not Available
- Failure to Inhibit Automatic Depressurization System

The operating experience issues were:

- Generic Letter 89-10, “Safety-Related Motor-operated Valve Testing and Surveillance”
- Bulletin 88-04, “Potential Safety-Related Pump Loss”
- Information Notice 2006-26, “Failure of Magnesium Rotors in Motor-Operated Valve Actuators”
- Information Notice 2006-31, “Inadequate Fault Interrupting Rating of Breakers”
- Suppression Pool Post-Accident Temperature Response
- K1 Relays for Emergency Diesel Generators

The team reviewed the Generic Letter 89-10 Design Basis calculations, which included design features of the motors, actuators and valves. The control logic was reviewed and coupled to the potential spectrum of design basis events requiring valve operation. Maximum opening and closing differential pressures, including the required opening and closing stroke times and attendant water hammer effects were reviewed. The team also reviewed in-service testing data, valve operation test and evaluation system testing data, surveillance test procedures, corrective action entries, and significant procedures that require operation of these valves. Investigation of inadvertent starts of the high-pressure core spray system led the inspectors to review the operating history of E22RVF035, the high-pressure core spray system thermal relief valve. Several valves were identified as having a magnesium rotor motor, and inspectors examined whether corrective actions regarding inspection and/or repair of magnesium rotor motors applied.

The team reviewed the design basis documentation, the Updated Final Safety Analysis Report, Technical Specifications, system drawings and Piping and Instrumentation drawings, for the Division II Standby Diesel Generator and the Division III High Pressure Core Spray Diesel Generator. Specific reviews of the fuel oil storage tank capacity calculations, day tank capacity calculation, fuel oil chemical composition, and diesel generator building design calculation for heat loading were performed. Further, a walkdown of the Division II and III Diesel Generators coupled with discussions with system and design engineers was also undertaken to assess the overall material condition of the diesel generator.

The team inspected the 4kV switchgear, including EJS-X2A transformer and Motor Control Center EHS-MCC-2E to verify they would operate during design basis events. The inspectors reviewed selected calculations for electrical distribution system load flow/voltage drop, degraded voltage protection, short-circuit, and electrical protection and coordination. This review was conducted to assess the adequacy and appropriateness of design assumptions, and to verify that bus capacity was not exceeded and bus voltages remained above minimum acceptable values under design basis conditions. Additionally, the switchgear's protective device settings and breaker ratings were reviewed to ensure that selective coordination was adequate for protection of connected equipment during worst-case, short-circuit conditions. The inspectors evaluated selected portions of the licensee response to NRC Generic Letter (GL) 2006-02, "Grid Reliability and the Impact on Plant Risk and the Operability of Offsite Power," dated February 1, 2006. The station's interface and coordination with the transmission system operator for plant voltage requirements and notification set points were reviewed. Additionally, bus-operating procedures were reviewed to determine if adequate guidance was given to the operators to ensure design basis assumptions were maintained. The inspectors reviewed the degraded and loss of voltage relays modification to assess whether the setpoints were in accordance with design calculations and the associated calibration procedures were consistent with calculation assumptions. To determine if breakers were maintained in accordance with industry and vendor recommendations, the inspectors reviewed the preventive maintenance inspection and testing procedures. The inspectors reviewed breaker opening and closure logic to verify the appropriate functionality was implemented. The 125Vdc voltage calculations were reviewed to determine if adequate voltage would be available for the breaker open/close coils and spring charging motors. Finally, the inspectors performed a walkdown of portions of the safety-related 4160Vac switchgear and interviewed system and design engineers to assess the installation configuration, material condition, and potential vulnerability to hazards.

b. Findings

1. Noncited Violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control"

The team identified a finding of very low safety significance (Green) involving a noncited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," with eight examples.

Example 1: Motor Operated Valves Calculation Used Non-Conservative Inputs and Methodologies

Introduction. The team identified an example of the Green NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control." The licensee used non-conservative inputs or methodologies in calculating terminal voltages to safety-related motor operated valve motors that would be required to operate for mitigation of design bases events. Specifically, the licensee used non-conservative steady-state motor control center voltages and 50 percent of rated locked rotor current without adequate technical justification to evaluate the minimum terminal voltage and actuator output torque for safety-related motor operated valves. The non-conservative assumptions resulted in over-estimating the minimum voltage available for motor-operated valves and other loads on the safety-related 480 VAC buses.

Description. Calculation E-225, “Voltage Calculation of Category 1 480V Motor-Operated Valves,” determines the motor terminal voltage and minimum actuator output torque for each safety-related motor-operated valve in the Generic Letter 89-10 program. The calculation uses as input, minimum steady-state motor control center voltage results from Calculation G13.18.3.6*016, which was also identified as using non-conservative inputs during this assessment. The calculated motor-operated valve terminal voltages and minimum actuator output torque were direct design input into the applicable mechanical motor-operated valve thrust and torque calculations. The licensee, in Calculation E-225, non-conservatively used the combination of steady-state motor control center voltages from Calculation G13.18.3.6*016 and a reduced motor-operated valve locked rotor current of 50 percent of rated locked rotor current without adequate technical justification to calculate the minimum terminal voltage and minimum actuator output torque for the safety-related motor-operated valves that were required to change state during a design basis event. At the initiation of a design basis event, many safety-related 480V motors and motor-operated valves receive an accident signal to actuate, thus “block starting” at the beginning of the event. That effect along with the starting and sequencing of large 4.16-kV safety-related motors can cause the bus voltages to drop below the degraded voltage relay setting and take several seconds to recover to the steady-state values specified in Calculation G13.18.3.6*016 and used in Calculation E-225. Combined with the use of 50 percent of rated locked rotor current for the starting motor-operated valves, that would predict a significantly higher terminal voltage and actuator output torque than would actually exist. Additionally, during the period of the accident-loading transient, the motor-operator valves could be in a stall condition and the licensee failed to analyze the effect on the motor-operated valve torque capability during this period or the affect of the delay in actuating the motor-operated valves on the safety analysis of the plant.

The licensee’s approach was contrary to the commitments in Updated Final Safety Analysis Report page 8.3-57, which states, “To ensure proper motor-operated valve function under design conditions, an analysis is performed to determine the torque or thrust that must be delivered to the valve stem by the motor operator under degraded voltage conditions (the calculated minimum starting voltage available).” Additionally, NRC guidance given in Question 36 to GL 89-10 Supplement 1, states that, “100 percent locked rotor current be used to determine the voltage drop from the motor control center to the motor-operated valve terminals unless there is data to support use of a lower value for each valve.” Contrary to this guidance, the licensee used 50 percent of locked rotor current without adequate technical justification or supporting data to justify its use.

By the end of the inspection, the licensee was able to demonstrate operability of the systems and no adverse affect on the plant safety analyses.

Analysis. The team determined that the failure to evaluate adequately the minimum terminal voltage and actuator output torque for safety-related motor-operated valves was a performance deficiency. The finding was more than minor because it was similar to NRC Inspection Manual Chapter 0612, Appendix E, “Examples of Minor Issues,” Example 3j, in that the engineering calculation errors resulted in a condition where there was reasonable doubt on the operability of a system or component. The finding was associated with the design control attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Traditional enforcement does not apply because the issue did not have any actual safety consequences or potential for impacting the NRC's regulatory function, and was not the result of any willful violation of NRC requirements. In accordance with Inspection Manual Chapter 0609, Attachment 4, "Phase 1 – Initial Screening and Characterization of Findings," a Phase 1 screening was performed and determined the finding was of very low safety significance (Green) because it was a design or qualification deficiency confirmed not to result in a loss of operability or functionality.

Enforcement. 10 CFR Part 50, Appendix B, Criterion III, "Design Control," requires, in part, that design control measures provide for verifying or checking the adequacy of design and design changes are required to be subjected to design control measures commensurate with those applied to the original design. Contrary to the above, from October 2000 until June 2008, the licensee did not provide design control measures to evaluate adequately the minimum terminal voltage and actuator output torque for safety-related motor-operated valves. Because this violation was of very low safety significance (Green) and has been entered into the corrective action program (CR-RBS-2008-03339), this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000458/2008006-01, Eight Examples of a Failure to Meet 10 CFR Part 50, Appendix B, Criterion III, "Design Control." This was the first of eight examples.

Example 2: Degraded Voltage Calculation used Non-Conservative Inputs

Introduction. The team identified a second example of the Green NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control." The licensee's design control measures failed to verify the adequacy of design modification ER-RB-2001-0360-000 by failing to perform a conservative analysis to ensure that Technical Specification Setpoints are adequate. After identification, the licensee entered the issue into the plant's corrective action program as CR-RBS-2008-03911 and started revising the affected calculation to determine the impact of the non-conservatisms.

Description. Calculation G13.18.3.6*016, "Degraded Voltage Calculations for Safety-Related Buses," provides the design basis for the Degraded Voltage relay setpoints in Technical Specifications Table 3.3.8.1-4 and Technical Requirements Manual, Table 3.3.8.1-1 that were changed as a result of design modification ER-RB-2001-0360-000. Additionally, Calculation G13.18.3.6*016 is a design input to Calculation E-225, "Voltage Calculation of Category 1 Motor-Operated Valves," which determined the minimum torque output for motor-operated valves under worst anticipated conditions. Calculation G13.18.3.6*016 non-conservatively assumed the following: (1) the diesel generator loading calculation was the basis for the safety-related loading with offsite power available, (2) motor-operated valves as continuously running at a 50 percent diversity instead of starting at the beginning of the event, and (3) it utilized sequence times for high-pressure core spray, low-pressure core spray, and residual heat removal from system and operator training manuals instead of appropriate design inputs and did not evaluate for Technical Specification allowable tolerances. The non-conservative assumptions would result in the predicted voltages at the safety-related buses and connected loads being higher than would be actually available. This could result in safety-related motors and motor-operated valves, which were required to change state for the design basis event, not having adequate terminal and/or control circuit voltages, and sufficient output torque to operate. This could result in actuation and timeout of the degraded voltage relays, resulting in offsite power being disconnected and the safety loads transferred to the emergency diesel generators needlessly, thus challenging the emergency diesel generators and safety-related distribution system.

By the end of the inspection, no instances were identified where using the Technical Specification degraded voltage allowable setpoint values would have resulted in inoperable equipment.

Analysis. The team determined that the failure to evaluate adequately design modifications and the resulting Technical Specification changes was a performance deficiency. The finding was more than minor because it was similar to NRC Inspection Manual Chapter 0612, Appendix E, "Examples of Minor Issues," Example 3j, in that the engineering calculation errors resulted in a condition where there was reasonable doubt on the operability of a system or component. The finding was associated with the design control attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Traditional enforcement does not apply because the issue did not have any actual safety consequences or potential for impacting the NRC's regulatory function, and was not the result of any willful violation of NRC requirements. In accordance with Inspection Manual Chapter 0609, Attachment 4, "Phase 1 – Initial Screening and Characterization of Findings," a Phase 1 screening was performed and determined the finding was of very low safety significance (Green) because it was a design or qualification deficiency confirmed not to result in a loss of operability or functionality.

Enforcement. 10 CFR Part 50, Appendix B, Criterion III, "Design Control," requires that measures be established to assure that applicable regulatory requirements and the design basis for structures, systems and components are correctly translated into specifications, drawings, procedures and instructions. Contrary to the above, prior to June 6, 2008, the licensee used incorrect and non-conservative assumptions in calculations performed to ensure that electrical equipment would remain operable under degraded voltage conditions. Because this violation was of very low safety significance (Green) and has been entered into the corrective action program (CR-RBS-2008-03911), this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000458/2008006-01, Eight Examples of a Failure to Meet 10 CFR Part 50, Appendix B, Criterion III, "Design Control." This was the second of eight examples.

Example 3: Motor Control Center Control Circuit Voltage Drop Calculation - Use of Non-Conservative Inputs

Introduction. The team identified a third example of the Green NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control." The licensee used non-conservative inputs or methodologies in calculating control circuit voltages to safety-related motor-operated valves and motors that were required to operate for mitigation of design bases events.

Description. The licensee used non-conservative inputs or methodologies in calculating control circuit voltages to safety-related 480V motor operated valves motor-operated valve and motors that would be required to operate for mitigation of design bases events. Calculation E210, "Cable Loop Length Calc for Voltage Drop-AC Circuits," used steady-state post-event motor control center voltages to evaluate the control circuit voltage drop for the control circuits for 480V motor-operated valves and motors that were required to change state for a design basis event. The use of steady-state voltages instead of transient voltages would predict higher control circuit voltages than would actually exist during the actuation for a design basis event. As a result, the control circuit contactor might not have adequate voltage to energize until after the upstream 4.16-kV and 480V switchgear starting loads have accelerated. Voltage recovery may take up to 5.7 seconds in which safety-related loads may not start due to inadequate control circuit voltages. Upon identification, the licensee evaluated this potential time delay and the impact on the plant safety analyses and the impact on residual heat removal, low-pressure safety injection, high-pressure core spray, and low-pressure core spray pumps in a dead head condition for 5.7 seconds, and concluded that no adverse impact would result. Additionally, the licensee used a bounding motor control center voltage of 422V for all the evaluations, which was based on calculation E-132, "Voltage Profile," which has not been updated for cumulative effect since 1986. The licensee evaluated the use of 422V from E-132, and concluded that it was conservative based on more recent Calculation G13.18.3.6*016, "Degraded Voltage Calculation for Class 1E Buses."

Analysis. The team determined that failure to evaluate the control circuits during the period that components receive an actuation signal to change state was a performance deficiency. The finding was more than minor because it was similar to NRC Inspection Manual Chapter 0612, Appendix E, “Examples of Minor Issues,” Example 3j, in that the engineering calculation errors resulted in a condition where there was reasonable doubt on the operability of a system or component. The finding was associated with the design control attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Traditional enforcement does not apply because the issue did not have any actual safety consequences or potential for impacting the NRC’s regulatory function, and was not the result of any willful violation of NRC requirements. In accordance with Inspection Manual Chapter 0609, Attachment 4, “Phase 1 – Initial Screening and Characterization of Findings,” a Phase 1 screening was performed and determined the finding was of very low safety significance (Green) because it was a design or qualification deficiency confirmed not to result in a loss of operability or functionality.

Enforcement. 10 CFR Part 50, Appendix B, Criterion III, "Design Control," requires, in part, that design control measures provide for verifying or checking the adequacy of design and design changes are required to be subjected to design control measures commensurate with the original design. Contrary to the above, prior to June 6, 2008, the licensee did not adequately verify that the control circuits for safety-related motor-operated valves and motors required to operate for a design basis event would have sufficient voltages to operate when required. Because this violation was of very low safety significance (Green) and has been entered into the corrective action program (CR-RBS-2008-03858), this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000458/2008006-01, Eight Examples of a Failure to Meet 10 CFR Part 50, Appendix B, Criterion III, “Design Control.” This was the third of eight examples.

Example 4: Motor Control Center Control Circuit Voltage Drop Calculation Did Not Evaluate E12-MOV-F042A and E12-MOV-F064A

Introduction. The team identified a fourth example of the Green NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control." In Calculation E210, "Cable Loop Length Criteria for Voltage Drop – AC Circuits," the licensee failed to evaluate E12-MOV-F042A, residual heat removal injection valve, and E12-MOV-F064A, residual heat removal minimum flow valve, to verify adequate voltage would be available to operate the associated 120VAC control circuit devices. These valves would be required to operate for mitigation of design bases events. After identification, the licensee entered the issue into the plant's corrective action program as CR-RBS-2008-03641, performed an engineering evaluation for the two motor-operated valves, and concluded that sufficient voltage would be available to operate the valves.

Description. Calculation E-210 determines if sufficient control circuit voltage is available to operate 120Vac control circuit devices associated with motor control center connected loads by calculating the maximum allowable control circuit cable length for each type control circuit. Each connected motor control center load is evaluated to determine if the associated circuit length is less than the maximum allowed by Calculation E-210. Motor operated valves, E12-MOV-F042A, residual heat removal injection valve, and E12-MOV-F064A, residual heat removal minimum flow valve, which would be required to operate for mitigation of design bases events, were not evaluated based on the criteria in Calculation E-210. Lack of analysis of the two affected motor-operated valve control circuits indicated a reasonable doubt on the operability of the two motor-operated valves and their ability to perform the required safety function during a design basis event. After the issue was identified, the licensee performed an engineering evaluation of the two circuits and determined that sufficient voltage would be available to operate the motor-operated valves when required.

Analysis. The team determined that failure to evaluate the control circuits for the two motor-operated valves was a performance deficiency. The finding was more than minor because it was similar to NRC Inspection Manual Chapter 0612, Appendix E, "Examples of Minor Issues," Example 3j, in that the engineering calculation error (failure to evaluate) resulted in a condition where there was reasonable doubt on the operability of a system or component. The finding was associated with the design control attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Traditional enforcement does not apply because the issue did not have any actual safety consequences or potential for impacting the NRC's regulatory function, and was not the result of any willful violation of NRC requirements. In accordance with Inspection Manual Chapter 0609, Attachment 4, "Phase 1 – Initial Screening and Characterization of Findings," a Phase 1 screening was performed and determined the finding was of very low safety significance (Green) because it was a design or qualification deficiency confirmed not to result in a loss of operability or functionality.

Enforcement. 10 CFR Part 50, Appendix B, Criterion III, "Design Control," requires, in part, that design control measures provide for verifying or checking the adequacy of design. Design changes are required to be subjected to design control measures commensurate with those applied to the original design. Contrary to the above, prior to June 6, 2008, the licensee did not adequately verify the control circuits for safety related motor-operated valves required to operate for a design basis event would have sufficient voltage to operate. Because this violation was of very low safety significance (Green) and has been entered into the corrective action program (CR-RBS-2008-03641), this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000458/2008006-01, Eight Examples of a Failure to Meet 10 CFR Part 50, Appendix B, Criterion III, “Design Control.” This was the fourth of eight examples.

Example 5: Inadequate Design Basis Documentation for Safety-Related Control Building HVAC System

Introduction. The team identified a fifth example of the Green NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control." No formal safety-related calculation was available that bounded the heating, ventilation, and air conditioning system airflow requirements for hydrogen concentration control in the Division I and II Battery Rooms in the control building. References contained in the System Design Criteria documents for the control building HVAC system either did not provide the calculation methodology used to determine the required airflow or could not be located for review. The system design criteria are considered a design basis document.

Description. During a tour of the Division II station battery room, the team questioned the capacity of the control building HVAC system to remove hydrogen. The team requested the calculation showing the system was capable of maintaining the hydrogen concentration below design limits. The licensee provided the team System Design Criteria document, "Control Bldg. HVAC System, Control Bldg. Chilled Water System, Ventilation Chilled Water System Design Criteria System Numbers 402, 410 & 410." This document specified, in part, that the safety-related function of the control building HVAC system is to provide cooling, heating, ventilation, and pressurization for the battery rooms during normal, shutdown, loss of offsite power, and design basis accident conditions. The system design criteria also stated the exhaust system maintains the hydrogen concentration levels in the battery room below the explosive limits; however, no analysis was performed in the system design criteria. Further discussion with the licensee indicated this analysis was contained in a corrective action document, CR-RBS-1996-1227 and had not been translated into a formal design calculation. CR-RBS-1996-1227 referenced vendor technical document VTD-T956-0005, "Globe Union – Stationary Battery," as the source of data for hydrogen gassing and battery room ventilation requirements. This reference number did not correspond to the current licensee vendor document identification methodology, and could not be located. The system design criteria referred to PID-22-09A-C, "HVAC – Control Building," a piping and instrumentation drawing, as the reference document for the control building HVAC system airflow requirements. This piping and instrument drawing did not provide the calculation methodology used to support the system design criteria assertions. Procedure EN-DC-126, the licensee document for engineering calculations, stated: “If credit is taken for a numerical value in a design basis document, or as an operational parameter, the basis for that value must be documented. If there is a need to maintain and retrieve the basis for this value, a calculation should be generated.”

Analysis. The failure to maintain design documentation for a safety-related system is a performance deficiency. The finding was more than minor because it was similar to NRC Inspection Manual Chapter 0612, Appendix E, “Examples of Minor Issues,” Example 3j, in that the failure to maintain the design documentation resulted in a condition where there was reasonable doubt on the operability of a system or component. By the end of the inspection, the licensee was able to provide sufficient information to the team to demonstrate that the HVAC system was sufficient to maintain the hydrogen concentration in the battery rooms below design limits. The finding was associated with the design control attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Traditional enforcement does not apply because the issue did not have any actual safety consequences or potential for impacting the NRC's regulatory function, and was not the result of any willful violation of NRC requirements. In accordance with Inspection Manual Chapter 0609, Attachment 4, "Phase 1 – Initial Screening and Characterization of Findings," a Phase 1 screening was performed and determined the finding was of very low safety significance (Green) because it was a design or qualification deficiency confirmed not to result in a loss of operability or functionality.

Enforcement. 10 CFR Part 50, Appendix B, Criterion III, "Design Control," states, in part, "that measures shall be established to assure that the design basis as defined in § 50.2 and as specified in the license application, for those structures, systems, and components to which this appendix applies are correctly translated into specifications, drawings, procedures, and instructions." Contrary to the above, prior to June 6, 2008, the licensee failed to ensure that parts of the design bases for the control building HVAC system were correctly translated into design basis documents. No formal bounding calculation for the Division I and II battery room ventilation systems' ability to control hydrogen concentration was available, and adequate references in System Design Criteria documents supporting this analysis were not maintained. Because this violation was of very low safety significance (Green) and has been entered into the corrective action program (CR-RBS-2008-02566 and CR-RBS-2008-03403), this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000458/2008006-01, Eight Examples of a Failure to Meet 10 CFR Part 50, Appendix B, Criterion III, "Design Control." This was the fifth of eight examples.

Example 6: Failure to Ensure Design Basis Information for Safety Related 125VDC Batteries was Controlled and Correctly Translated into Procedures and Instructions

Introduction. The team identified a sixth example of the Green NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control." The licensee failed to ensure that the intercell resistance limits specified in the maintenance procedure and administrative limits for safety related 125Vdc batteries, were correctly incorporated from vendor specified design requirements.

Description. During a review of the River Bend Station direct current electrical system component design basis, a corrective action document, CR-RBS-2007-00202 was presented for review, which discussed the licensee review of operating experience OE23813. This operating experience discusses inconsistencies between industry standards for station batteries, and Technical Specification limits on battery connection resistances. After review of CR-RBS-2007-00202, a request was made for access to the technical documentation from which maintenance and surveillance testing was performed on the Division I and II station batteries. River Bend Station replaced their Division I and II batteries in 2001; both divisional batteries were manufactured by GNB Technologies and were similar in construction. A vendor manual, Z99-003384, "Stationary Battery Installation and Operating Instruction," Revision 8/93, was provided as part of this request. The team noted that Section 19.1, "Connection Resistance," was deleted by lineout, dated April 24, 1994. Section 19.1 required an annual measurement of connection resistances, and any connection having a resistance value of more than 20 percent above its benchmark value to be corrected by retorquing or cleaning. The licensee informed the team that the section was not considered applicable to River Bend Station. However, no technical justification was provided as a basis for this decision. Another vendor manual, Z99-003384, "Stationary Battery Installation and Operating Instruction," Revision 2/88, was provided to the team for review. This revision also had Section 19.1 deleted, with deletion date January 3, 1990. This manual version had different requirements for connection resistance allowances in Section 19.1, than the 1993 version. After further discussion with licensing staff and requests for clarification on which version of the battery vendor manual was in use, a third copy of the vendor manual, identified by the licensee vendor technical document tracking system as VTD-G185-0100, Rev. 0, was provided. This manual, identified by the vendor as Manual Z99-003384, was dated 10/97. This version did not delete Section 19.1. Further discussions to determine which vendor manual version was in use and why Section 19.1 was not applicable to River Bend Station were inconclusive.

The licensee later determined that Manual Z99-003384, "Stationary Battery Installation and Operating Instruction," Revision 2/88, was the appropriate manual and the other two were supplements to the original document. A review of Engineering Request ER-RB-2001-0484-000, "ENB-BAT01A and ENB-BAT01B Supplemental Vendor Information," dated August 28, 2001, indicated that the vendor provided additional vendor information that is applicable to batteries ENB-BAT01A and ENB-BAT01B and that Manual VTD-G185-0100, Rev. 0, contained supplemental information to be input into another calculation (Calculation 3244.522-075-003). Manual VTD-G185-0100, Rev. 0, contained section 19.1, which required the intercell resistances be taken at installation and used as a benchmark. The manual also provided a special procedure that would allow benchmark data to be obtained at a later date if this was not done at installation. Intercell resistances are required to be checked annually and any connections that exceed the benchmark average by 20 percent or 5 micro ohms, whichever is greater, are an indication of a degrading connection that should be corrected. Increased resistance of the connection could be caused by corrosion or a relaxation in hardware torque value. The manual stated in part, "Maintaining electrical integrity of connections is important as poor connections will result in reduced battery output and in extreme cases may cause melted cell posts, circuit interruptions or battery fires." The licensee has currently identified that the technical specification limits of 150 micro ohms is non-conservative and has set administrative limits of 45 micro ohms based on design calculation limits for acceptable voltage drop. The Technical Specification issue is being addressed separately with the Office of Nuclear Reactor Regulation.

Analysis. The failure to ensure that the design bases for safety-related components were correctly translated into procedures or instructions or to provide technical justifications for deviation from vendor recommendations is a performance deficiency. The finding was associated with the design control attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Traditional enforcement does not apply because the issue did not have any actual safety consequences or potential for impacting the NRC's regulatory function, and was not the result of any willful violation of NRC requirements. In accordance with Inspection Manual Chapter 0609, Attachment 4, "Phase 1 – Initial Screening and Characterization of Findings," a Phase 1 screening was performed and determined the finding was of very low safety significance (Green) because it was a design or qualification deficiency confirmed not to result in a loss of operability or functionality.

Enforcement. 10 CFR Part 50, Appendix B, Criterion III, "Design Control," requires in part, that measures be established to assure that the design basis for those structures, systems, and components to which this appendix applies are correctly translated into specifications, drawings, procedures, and instructions. These measures shall include provisions to assure that appropriate quality standards are specified and included in design documents and that deviations from such standards are controlled. Contrary to this requirement, prior to June 6, 2008, the licensee failed to ensure that the design basis for safety-related components were correctly translated into procedures or instructions, and no technical justifications were provided supporting decisions to deviate from vendor recommendations. Because this violation was of very low safety significance (Green) and has been entered into the corrective action program (CR RBS-2008-03659), this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000458/2008006-01, Eight Examples of a Failure to Meet 10 CFR Part 50, Appendix B, Criterion III, “Design Control.” This was the sixth of eight examples.

Example 7: Failure to Maintain Adequate Design Basis Calculations for Ultimate Heat Sink Loading

Introduction. The team identified a seventh example of the Green NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control." The licensee failed to properly quantify the total heat loads required to be dissipated by the Ultimate Heat Sink during a design basis event.

Description. Calculations PM-194, "Standby Cooling Tower Performance and Evaporation Losses Without Drywell Unit Coolers," and G13.18.14.0*088, "Temperature and Inventory Effects of Maximum Safeguards Operation on the Ultimate Heat Sink (Standby Cooling Towers)'" evaluate the ultimate heat sink response to peak and integrated heat loads for temperature and inventory considerations. The decay heat and emergency core cooling system pump loads were obtained from Calculation G13.18.14.0*190, Rev.1, "Post-Accident Heat Load Development for Power Uprate Service Water Evaluations." The Auxiliary Loads, Fuel Pool Coolers, Fuel Pool and Low-Pressure Core Spray pumps were also included as sources of heat to be dissipated by the ultimate heat sink. In the calculation, the section addressing the pump heat loads, only accounted for a portion of the total energy from the pumps that would be transferred to containment.

With one exception, the heat loads used in G13.18.14.0*190 were obtained from General Electric Calculation 7222.250-000-009A, "105 Percent Power Uprate Evaluation Report for GE Task No. 13.0 Containment Analysis.” To facilitate the following discussion, the energy supplied to the pump will be defined as brake horsepower, which can be broken down into the pump inefficiency and the pump efficiency. InBhp, the pump inefficiency term, is that portion of the pump energy supplied as heat to the fluid during the pumping process. EfBhp, the pump efficiency term, is the remaining portion of the energy supplied to the pump (note that Bhp = InBhp + EfBhp).

The General Electric containment analysis conservatively assumed that 100% of the Bhp was transferred to containment. When the licensee revised Calculation

G13.18.14.0*190 (Calculation G13.18.14.0*190, Revision 1), the assumption was made that only the InBhp term was necessary to be dissipated by the ultimate heat sink. The licensee could not provide adequate technical justification for this change in Calculation G13.18.14.0*190. Although the assumption in the original calculation that 100 percent of the pump Bhp should be required to be supplied to the ultimate heat sink was overly conservative, a technical basis was necessary for the assumption that only the InBhp needed evaluation as a heat load delivered to the ultimate heat sink. Since the energy supplied by the pump as flow velocity and pressure increase is converted to heat as the fluid travels through the system and only an undetermined fraction of it is dissipated from the system outside of containment and the ultimate heat sink, the neglect of this potential additional source of heat, requiring ultimate heat sink dissipation, was a non-conservative assumption, which was used without adequate technical justification.

Analysis. Failure to adequately incorporate expected heat loads into calculations that establish the required ultimate heat sink inventory was a performance deficiency. The finding was more than minor because it was similar to NRC Inspection Manual Chapter 0612, Appendix E, “Examples of Minor Issues,” Example 3j, in that the engineering calculation errors resulted in a condition where there was reasonable doubt on the operability of a system or component. The finding was associated with the design control attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Traditional enforcement does not apply because the issue did not have any actual safety consequences or potential for impacting the NRC’s regulatory function, and was not the result of any willful violation of NRC requirements. In accordance with Inspection Manual Chapter 0609, Attachment 4, “Phase 1 – Initial Screening and Characterization of Findings,” a Phase 1 screening was performed and determined the finding was of very low safety significance (Green) because it was a design or qualification deficiency confirmed not to result in a loss of operability or functionality.

Enforcement. 10 CFR Part 50, Appendix B, Criterion III, "Design Control," requires in part, that measures 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, prior to June 6, 2008, the licensee failed to incorporate expected heat loads into design basis calculations for ultimate heat sink loading. Specifically, the licensee’s analysis failed to account for all heat loads to which the ultimate heat sink may be subjected during a design basis event. Because this violation was of very low safety significance (Green) and has been entered into the corrective action program (CR-RBS-2008-3712), this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000458/2008006-01, Eight Examples of a Failure to Meet 10 CFR Part 50, Appendix B, Criterion III, “Design Control.” This was the seventh of eight examples.

Example 8: Diesel Generator Frequency Variation Not Considered in Loading Calculations

Introduction. The team identified an eighth example of the Green NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control." The licensee failed to take into account the effect of emergency diesel generator frequency variation in the diesel loading calculation E-192, “Standby Diesel Generator Loading Calculation.”

Description. Calculation E-192 determined diesel loading based on maximum loads during a large break loss of coolant accident. The loading was based on nominal 60-Hertz operation of pumps and fans, and did not account for the $\pm$ 2 percent variation allowed by River Bend Station Technical Specification 3.8.1. Since the emergency diesel generator accident loading was comprised primarily of centrifugal loads, the inspectors determined this should have been considered in loading calculations because the power demanded by centrifugal pumps and fans increases by the cube of the ratio of the speeds. In response to the inspector's question, the licensee provided a preliminary engineering evaluation that showed that diesel loading would increase by approximately six percent above nominal. Calculation E-192, Updated Final Safety Analysis Report Table 8.3-2, and Figure 8.3-14a (b) showed that the highest automatically sequenced accident load was approximately 3017 kW and occurred on the Division 1 emergency diesel generator between 10 minutes and 2 hours after the initiation of the design basis event. Consequently, when the maximum allowed frequency variation was included; the loading was shown to be approximately 3198 kW, which exceeded the River Bend Station imposed emergency diesel generator limit of 3130 kW maximum. The licensee's engineering evaluation identified that load HVR-FN11A, Annulus Mixing System Fan, had been disabled per ER02-0223, but was still included in Calculation E-192 as conservatism. Removing the 101.8 kW load reduced the calculated loading to within the maximum rating of 3130 kW. This issue was entered into the licensee's corrective action program as CR-RBS-2008-03556. By the end of the inspection, the licensee had performed an evaluation that demonstrated operability and functionality of the emergency diesel generators.

Analysis. The inspectors determined that failure to properly account for the effect of frequency variation on diesel generator loading was a performance deficiency. The finding was more than minor because it was similar to NRC Inspection Manual Chapter 0612, Appendix E, "Examples of Minor Issues," Example 3j, in that the failure to account for frequency variations had more than a minimal effect on the outcome of the calculation. Failure to account for technical specification allowable frequency variations reduced the available margin such that conservatisms had to be removed in order to show that the diesel generator remained within the River Bend Station imposed emergency diesel generator limit of 3130 kW maximum. The finding was associated with the design control attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Traditional enforcement does not apply because the issue did not have any actual safety consequences or potential for impacting the NRC's regulatory function, and was not the result of any willful violation of NRC requirements. In accordance with Inspection Manual Chapter 0609, Attachment 4, "Phase 1 – Initial Screening and Characterization of Findings," a Phase 1 screening was performed and determined the finding was of very low safety significance (Green) because it was a design or qualification deficiency confirmed not to result in a loss of operability or functionality.

Enforcement. 10 CFR Part 50, Appendix B, Criterion III, "Design Control," requires, in part, that 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, prior to June 6, 2008, the licensee did not adequately translate design basis information into the diesel generator loading calculation. Calculation E-192 did not properly account for the TS allowable diesel generator $\pm$2 percent frequency variations. The licensee failed to consider how the frequency variation could affect the design and licensing basis of the diesel engines. Because this violation was of very low safety significance (Green) and has been entered into the corrective action program (CR-RBS-2008-03556), this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000458/2008006-01, Eight Examples of a Failure to Meet 10 CFR Part 50, Appendix B, Criterion III, “Design Control.” This was the eighth of eight examples.

2. Failure to Recalculate Suppression Pool Peak Temperature Response

Introduction. The team identified a finding of very low safety significance (Green) involving a noncited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” in that, design control measures for verifying the adequacy of design were not implemented. Specifically, the licensee did not recalculate suppression pool peak temperature response when a more severe single failure condition was identified.

Description. During a review of operating experience information regarding assessment of suppression pool post-accident temperature response, the team identified that the licensee had failed to recalculate suppression pool peak temperature response as identified by General Electric Energy, Nuclear, 10 CFR Part 21 Safety Information Communication SC-06-01 (January 19, 2006). The existing calculation assumed that the worst-case single failure condition for long-term suppression pool temperature response to the design basis accident (loss-of-coolant accident) was a loss of off-site power with failure of a diesel generator. This would result in minimum emergency core cooling system flow and minimum heat removal capability due to loss of a complete emergency core cooling system division. Safety Information Communication SC-06-01 informed the licensee that the worst single failure condition for some facilities for determination of the peak suppression pool temperature was loss of cooling capability of one residual heat removal heat exchanger, such that all emergency core cooling system pumps continued to operate and transfer pump heat to the suppression pool. The licensee initiated Condition Report CR-RBS-2006-00234 to address the concern in Safety Information Communication SC-06-01. The corrective action to revise the River Bend Station containment analysis to resolve the issue identified in Safety Information Communication SC-06-01 was transferred to Condition Report CR-RBS-2002-01734 on February 21, 2006. This action was in conjunction with planned revision of the containment analysis required as part of a planned power uprate. The corrective action item had a completion date of July 26, 2006. Prior to corrective action completion, the power uprate was placed on indefinite hold. The recalculation of suppression pool peak temperature response for the worst single failure condition was not completed.

By the end of the inspection, the licensee was able to demonstrate an acceptable long-term suppression pool peak temperature response.

Analysis. The team determined that the failure to recalculate long-term suppression pool peak temperature response in response to Safety Information Communication SC-06-01 was a performance deficiency. The finding was more than minor because it was associated with the design control attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Traditional enforcement does not apply because the issue did not have any actual safety consequences or potential for impacting the NRC’s regulatory function, and was not the result of any willful violation of NRC requirements. In accordance with Inspection

Manual Chapter 0609, Attachment 4, "Phase 1 – Initial Screening and Characterization of Findings," a Phase 1 screening was performed and determined the finding was of very low safety significance (Green) because it was a design or qualification deficiency confirmed not to result in a loss of operability or functionality of the suppression pool. The finding had a cross-cutting aspect in the area of problem identification and resolution because the licensee initiated a corrective action program action to re-evaluate long-term suppression pool peak temperature performance but closed the action without its completion (P.1 (d)).

Enforcement. 10 CFR Part 50, Appendix B, Criterion III, "Design Control," requires, in part, that design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculation methods, or by the performance of a suitable testing program. Contrary to the above, between January 2006 and June 2008, the licensee failed to perform a design review based on new information supplied by Safety Information Communication SC-06-01, which indicated the need to recalculate long-term suppression pool peak temperature performance. Because this violation was of very low safety significance (Green) and was entered into the licensee's corrective action program (CR-RBS-2008-03661), this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000458/2008006-02, Failure to Recalculate Suppression Pool Peak Temperature Response.

3. Inadequate Testing Programs for 4-kV Circuit Breakers, Class 1E Molded-Case Circuit Breakers, and the Emergency Diesel Generators

The team identified a finding of very low significance (Green) involving a noncited violation of 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," with three examples. Specifically, the team identified that the licensee failed to develop and implement adequate testing programs for 4-kV circuit breakers, Class 1E molded-case circuit breakers, and the emergency diesel generators.

Example 1: Failure to Establish an Adequate Test Program for Safety-Related 4-kV Circuit Breakers

Introduction. The team identified an example of the Green NCV of 10 CFR Part 50, Appendix B, Criterion XI, "Test Control." The licensee's maintenance and post-maintenance testing procedures for safety-related 4-kV circuit breakers did not have technical justification for deviation from vendor-recommended maintenance periodicity nor vendor-recommended testing using minimum-expected control voltage levels.

Description. During a review of documents related to 4-kV circuit breaker 125VDC control circuits, the team determined that vendor recommendations for circuit breaker servicing had not been followed. The licensee used Asea Brown Boveri 5HK type circuit breakers in safety- and non-safety related 4-kV electrical systems. Vendor Bulletin MS 3.2.1.9-1B, identified by licensee number VTD-B455-0122, "Asea Brown Boveri Maintenance and Surveillance for I-T-E Medium-Voltage Switchgear Equipment Type HK," provided guidance on the maintenance and surveillance of HK-type switchgear and circuit breakers, including those in service at River Bend Station. The vendor bulletin provided a list of tests that should be accomplished at each service interval, including a breaker operation check. The vendor bulletin also stated the breaker operation check should be performed after breaker servicing and that it be performed at the minimum-expected control voltage level. The minimum control voltage requirement at the device terminals was 71VDC, determined analytically from Calculations E-143, Rev. 9, and E-144, Rev. 5, "Standby Battery ENB-BAT01A (B) Duty Cycle, Current Profile and Size Verification."

In a letter dated October 21, 1985, from BBC Brown Boveri, Inc. Switchgear Products Division to Stone and Webster Engineering Corporation, 5HK circuit breaker control component reliability and operability were discussed. Item 1 of the letter stated, in part, that the 125VDC nominal Close coils will function reliably at a minimum voltage of 70VDC over the design life of the coil. Item 6 of the letter stated, in part, that the Trip coil is certified to have the same characteristics as the Close coil (i.e. functions at 70VDC). Item 7 of the letter stated, in part, that the above presuppose adequate maintenance of the circuit breaker mechanisms.

In discussions with River Bend Station staff, and review of corrective action documents, tracking spreadsheets, and work orders, the team determined that the licensee had not met the vendor recommendations for timely 5HK circuit breaker overhauls and had not performed breaker operation checks at the minimum expected control voltage level as is recommended by the vendor. As a result, the licensee could not provide assurance that the safety-related breakers would function reliably at a minimum voltage of 70VDC.

The team reviewed an example where this was a problem in review of vendor data associated with the overhaul of safety-related 5HK circuit breaker (Serial # 51128A—4---03544), dated February 22, 2008. The as-found breaker failed to trip at the minimum control voltage, and the sluggish operation was due to gummed grease.

With respect to periodicity of breaker overhaul, the team learned that several 5HK safety-related breakers have not had overhauls since 1990. The industry recommendation for overhaul of Asea Brown Boveri 5HK breakers was to not exceed 12 years between overhauls, based on Electric Power Research Institute information that the average age for failures due to dirt/contaminated lubricant is approximately ten years, and most failures from failed parts occur between ten and twelve years. Asea Brown Boveri recommended overhauling these breakers at the ten-year period and the vendor's breaker overhaul procedures included reduced control voltage testing of the control circuitry.

By the end of the inspection, the team concluded the licensee's evaluation of this issue, documented in the corrective action program, provided a reasonable basis for operability.

<u>Analysis</u>. The failure to include vendor recommended operability testing of safety-related circuit breaker control circuits at the voltages postulated to exist at the device terminals during design basis events, or to provide justification for not performing the testing, was a performance deficiency. The finding was more than minor because it is associated with the equipment performance attribute of the Mitigating Systems cornerstone objective of ensuring the availability, reliability, and capability of the safety-related circuit breakers to respond to initiating events to prevent undesirable consequences. Traditional enforcement does not apply because the issue did not have any actual safety consequences or potential for impacting the NRC's regulatory function, and was not the result of any willful violation of NRC requirements. In accordance with Inspection Manual Chapter 0609, Attachment 4, "Phase 1 – Initial Screening and

Characterization of Findings," a Phase 1 screening was performed and determined the finding was of very low safety significance (Green) because the condition did not represent a loss of system safety function, did not represent an actual loss of safety function of a single train for greater than its Technical Specification allowed outage time, did not represent an actual loss of one or more risk-significant non- Technical Specification trains of equipment for greater than 24 hours, and did not screen as potentially risk-significant due to seismic, flooding, or severe weather.

Enforcement. 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," requires in part, that a test program be established to assure that all testing required to demonstrate that components will perform satisfactorily in service is identified and performed in accordance with procedures that incorporate the requirements and acceptance limits contained in applicable design documents. Contrary to these requirements, prior to June 6, 2008, licensee preventative and post-maintenance procedures for safety-related 4-kV circuit breakers did not include vendor recommended testing for performing operability tests of breaker control circuits at the minimum expected control voltage levels postulated to exist at the device terminals during design basis events or performance of maintenance at the vendor-recommended periodicity. Because this violation was of very low safety significance (Green) and was been entered into the corrective action program (CR-RBS-2008-04378 and CR-RBS-2008-04379), this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000458/2008006-03, Three Examples of a Failure to Meet 10 CFR Part 50, Appendix B, Criterion XI, "Test Control." This was the first of three examples.

Example 2: Lack of Molded-Case Circuit Breaker Periodic Testing and Preventive Maintenance:

Introduction. The team identified a second example of the Green NCV of 10 CFR Part 50, Appendix B, Criterion XI, "Test Control." The licensee failed to implement a test program to assure that all installed safety-related molded-case circuit breakers would perform satisfactorily in service and failed to ensure that the molded-case circuit breaker preventive maintenance program remained current with industry and NRC operating experience to ensure that the installed safety-related and important-to-safety molded-case circuit breakers did not degrade and would perform satisfactorily in service.

Description. During review of the licensee's testing program for Class 1E molded-case circuit breakers, the team determined the program did not ensure the reliability of the installed breakers because the program did not include test methods or failure assessment that would accurately and conclusively demonstrate molded-case circuit breakers continued to be operable. With the exception of those molded-case circuit breakers associated with containment penetration circuits (Technical Requirements Manual Sections TR3.8.11 and TR 3.8.12); molded-case circuit breakers were not under a periodic test and preventive maintenance program that assessed age-related degradation of electrical components in the breakers.

The team noted that considerable industry experience was available regarding molded-case circuit breaker problems, including NRC Information Notice 93-64, "Periodic Testing and Preventive Maintenance of Molded Case Circuit Breakers," which identified generic concerns with aging molded-case circuit breakers. In particular, Information Notice 93-64 stated that molded-case circuit breaker preventive maintenance practices (such as manual exercising) could mitigate the effects of aging and help ensure continued molded-case circuit breaker reliability. However, manual exercising alone was not found effective in detecting or assessing age-related degradation. Detecting or assessing degradation could only be accomplished through appropriate periodic testing and monitoring. Certain standard molded-case circuit breaker tests (such as individual pole resistance, 300-percent thermal overload, and instantaneous magnetic trip tests) performed periodically were found effective along with the additional techniques of infrared temperature measurement and vibration testing. River Bend Station Updated Final Safety Analysis Report, Section 8.3.1.1.5.1 stated:

> "Maintenance and testing of auxiliary electrical power system equipment are conducted to ensure that all components are operational within their design limits. Maintenance and testing are performed periodically throughout station life in accordance with normal station operating procedures to: 1. Detect the deterioration of the components of the system toward an unacceptable condition and to take corrective action as required to bring the components to an acceptable condition. 2. Demonstrate the capability of the components that will normally be de-energized to perform properly when energized."

The testing items identified above were consistent with IEEE 308-1974, "IEEE Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations," (Section 7.4.1), which is endorsed by NRC Regulatory Guide 1.32, "Criteria for Safety-Related Electric Power Systems for Nuclear Power Plants," Revision 2. The Updated Final Safety Analysis Report identified that the licensee was committed to these documents. Other industry standards, such as NEMA AB-4, "Guidelines for Inspection and Preventive Maintenance of Molded-Case Circuit Breakers," also provided the recommended industry good practices to ensure molded-case circuit breaker reliability.

By the end of the inspection, the team determined the licensee's evaluation of the issue in the corrective action program provided reasonable assurance of operability and functionality of molded-case circuit breakers.

<u>Analysis</u>. The team determined that the licensee's failure to implement an adequate testing program to detect and assess degradation of safety-related molded-case circuit breakers was a performance deficiency. Failure of the breaker to operate properly could lead to a loss of power to safety-related components or lead to a potential for compromising other equipment on a single fault that the molded-case circuit breaker was designed to isolate.

The finding was more than minor because it is associated with the equipment performance attribute of the Mitigating Systems cornerstone objective of ensuring the availability, reliability, and capability of the safety-related molded-cased circuit breakers to respond to initiating events to prevent undesirable consequences. Traditional enforcement does not apply because the issue did not have any actual safety consequences or potential for impacting the NRC's regulatory function, and was not the result of any willful violation of NRC requirements. In accordance with Inspection Manual Chapter 0609, Attachment 4, "Phase 1 – Initial Screening and Characterization of Findings," a Phase 1 screening was performed and determined the finding was of very low safety significance (Green) because the condition did not represent a loss of system safety function, did not represent an actual loss of safety function of a single train for greater than its TS allowed outage time, did not represent an actual loss of one or more risk-significant non-TS trains of equipment for greater than 24 hours, and did not screen as potentially risk-significant due to seismic, flooding, or severe weather.

Enforcement. 10 CFR Part 50, Appendix B, Criterion XI, “Test Control,” states in part, that a test program shall be established to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents. Contrary to the above, prior to June 6, 2008, the licensee failed to assure that installed safety related and important-to-safety molded-case circuit breakers were in a periodic testing and preventive maintenance program that was capable of detecting age related deterioration of the components in the molded-case circuit breakers toward an unacceptable condition so they could be assessed and evaluated to ensure satisfactory in service performance. Because this violation was of very low safety significance (Green) and has been entered into the corrective action program (CR-RBS-2008-3634), this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000458/2008006-03, Three Examples of a Failure to Meet 10 CFR Part 50, Appendix B, Criterion XI, “Test Control.” This was the second of three examples.

Example 3: Inadequate Emergency Diesel Generator Surveillance Testing

Introduction. The team identified a third example of the Green NCV of 10 CFR Part 50, Appendix B, Criterion XI, "Test Control." The licensee failed to incorporate the requirements and acceptance limits contained in applicable design documents into the emergency diesel generator test procedures.

Description. The Updated Final Safety Analysis Report identified that the licensee was committed to NRC Regulatory Guide 1.9, “Selection, Design, Qualification, and Testing of Emergency Diesel Generator Units Used as Class 1E Onsite Power Systems at Nuclear Power Plants,” Revision 2. During review of Surveillance Test Procedure STP-309-0601, “Div I ECCS Test,” the team determined the licensee failed to incorporate adequate acceptance limits identified in Regulatory Guidance 1.9 in the test procedure. Regulatory Guide 1.9, Revision 2, Section 1.4 stated in part, “The diesel generator should be designed such that the frequency will not decrease, at any time during the loading sequence, to less than 95 percent of nominal and the voltage will not decrease to less than 75 percent of nominal." Section 2.3.2.3 stated in part, "The capability of the overall emergency diesel generator design should be demonstrated during every refueling outage (or at a frequency of not more than every 24 months)."

The team determined that STP-309-0601 only verified that the steady state voltage and frequency of the emergency busses was maintained at greater than 3,740 Volts and less than 4,580 Volts and greater than 58.8 Hertz and less than 61.2 Hertz, respectively, during the test. Additionally, the procedure did not require maintenance and testing equipment certified recording instruments to collect the data, but rather utilized plant Emergency Response Information System data, which is used for trending and historical information. Although the Emergency Response Information System data points are periodically calibrated with approved maintenance and testing equipment, due to the 1-second resolution of the Emergency Response Information System data and round-off errors, it was not possible to determine the lowest voltage and frequency transient with the test data available. Based on review of the data collected during performance of

STP-309-0601, the team could not verify that the response of the emergency diesel generator exciter/voltage regulator and governor control system was capable of accelerating the loads and remaining within the design requirements.

By the end of the inspection, the team determined the licensee's evaluation of the issue in the corrective action program provided reasonable assurance of operability and functionality of the emergency diesel generators.

Analysis. The team determined that the licensee's failure to properly translate the requirements and acceptance limits contained in applicable design documents for the emergency diesel generators into the testing program was a performance deficiency. The finding was more than minor because it is associated with the equipment performance attribute of the Mitigating Systems cornerstone objective of ensuring the availability, reliability, and capability of the emergency diesel generators to respond to initiating events to prevent undesirable consequences. Traditional enforcement does not apply because the issue did not have any actual safety consequences or potential for impacting the NRC's regulatory function, and was not the result of any willful violation of NRC requirements. In accordance with Inspection Manual Chapter 0609, Attachment 4, "Phase 1 – Initial Screening and Characterization of Findings," a Phase 1 screening was performed and determined the finding was of very low safety significance (Green) because the condition did not represent a loss of system safety function, did not represent an actual loss of safety function of a single train for greater than its TS allowed outage time, did not represent an actual loss of one or more risk-significant non-TS trains of equipment for greater than 24 hours, and did not screen as potentially risk-significant due to seismic, flooding, or severe weather.

Enforcement. 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," required, in part, that a test program shall be established to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate the requirements and acceptable limits contained in applicable design documents. Contrary to the above, prior to June 6, 2008, the licensee failed to require verification that the voltage and frequency response of the emergency diesel generator during performance of Surveillance Test Procedure STP-309-0601. Because this violation was of very low safety significance (Green) and has been entered into the corrective action program (CR-RBS-2008-3676 and CR-RBS-2008-3701), this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000458/2008006-03, Three Examples of a Failure to Meet 10 CFR Part 50, Appendix B, Criterion XI, "Test Control." This was the third of three examples.

4. Examples of Failure to Follow Procedure ADM-0073, "Temporary Installation Guidelines"

Introduction. The team identified a finding of very low safety significance (Green) involving a noncited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for five examples of failure to follow procedural requirements with respect to temporary installations.

Description. The team identified five examples where the licensee failed to follow Procedure ADM-0073 "Temporary Installation Guidelines." Specifically, from October

2006 until June 2008, the licensee failed to ensure that modifications made to the plant were installed and removed in accordance with the appropriate procedures.

Each of the following modifications were installed in the plant as a temporary installation, but did not meet the criteria of a temporary installation consistent with the requirements of ADM-0073:

- Alternate Health Physics Control Station
- Gaitronics Communication System Modification
- Radiation Protection ALARA Alert Signs
- Remote Acquisition and Display System Modification

Procedure ADM-0073 also required that temporary installations be removed when they were no longer needed. A temporary HEPA filter system was installed in containment during a previous refueling outage and not removed when it was no longer required.

Analysis. The team determined that the repetitive failure to properly implement procedure ADM-0073 was a performance deficiency. Although the team considered each of the above examples minor in significance, the team determined that this finding, which was associated with design control attribute of the Mitigating Systems cornerstone, was more than minor per Manual Chapter 612, Appendix E, "Examples of Minor Issues," Example 4a. The finding involved multiple examples of failure to follow licensee procedural requirements and if left uncorrected it could result in design modifications to the plant that were not properly evaluated, controlled, documented and installed. Traditional enforcement does not apply because the issue did not have any actual safety consequences or potential for impacting the NRC's regulatory function, and was not the result of any willful violation of NRC requirements. In accordance with Inspection Manual Chapter 0609, Attachment 4, "Phase 1 – Initial Screening and Characterization of Findings," a Phase 1 screening was performed and determined the finding was of very low safety significance (Green) because the condition did not represent a loss of system safety function, did not represent an actual loss of safety function of a single train for greater than its Technical Specification allowed outage time, did not represent an actual loss of one or more risk-significant non-Technical Specification trains of equipment for greater than 24 hours, and did not screen as potentially risk-significant due to seismic, flooding, or severe weather.

The cause of this finding has a crosscutting aspect associated with resources in the human performance area because the licensee failed to ensure that personnel, equipment, procedures, and other resources were available and adequate to assure nuclear safety. Specifically, those necessary for maintaining long term plant safety by maintenance of design margins, minimization of long-standing equipment issues, minimizing preventative maintenance deferrals, and ensuring maintenance and engineering backlogs which were low enough to support safety (H.2 (a)).

Enforcement. 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. Instructions, procedures, or drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Contrary to this requirement, from October 2006 until June 2008, the licensee failed to adequately evaluate changes and installations in the plant to ensure compliance with ADM-0073 for installation and removal. Because this violation is of very low safety significance (Green) and it was entered into the corrective action program (CR-RBS-2008-3410), this finding is being treated as an NCV, consistent with section VI.A.1 of the NRC Enforcement Policy: NCV 05000458/2008006-04, Inadequate Implementation of Temporary Installation Procedure.

4. OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems (71111.21)

a. Inspection Scope

The team reviewed a sample of problems that the licensee had identified previously and entered into the corrective action program. The team reviewed these issues to verify an appropriate threshold for identifying issues and to evaluate the effectiveness of corrective actions. In addition, condition reports written on issues identified during the inspection were reviewed to verify adequate problem identification and incorporation of the problem into the corrective action system. The specific documents that were sampled and reviewed by the team are listed in the attachment.

b. Findings

1. Failure to Follow Operability Determination Procedure for Magnesium Rotor Corrosion

Introduction. The team identified a finding of very low safety significance (Green) involving a noncited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for the failure of the licensee to follow procedures to evaluate conditions adverse to quality for impacts on the operability of safety-related equipment.

Description. The team reviewed the licensee’s operating experience assessment and actions to address NRC Information Notice 2006-26, “Failure of Magnesium Rotors in Motor-Operated Valve Actuators.” The Information Notice discussed failures of motor-operated valve actuators attributed to oxidation and corrosion of the magnesium motor rotor fan blades and shorting ring. Three main failure mechanisms were identified: galvanic corrosion, general corrosion, and thermally-induced stress. Conditions that could cause these failure mechanisms included: motor overheating events (typically due to locked-rotor conditions), high-humidity, or high-temperature operating environments.

The licensee’s 2006 evaluation of the industry operating experience, documented in Condition Report CR-RBS-2006-02882, confirmed that River Bend Station had installed motor-operated valves with magnesium rotors and that they were susceptible to the described corrosion and oxidation. The existing motor-operated valve program did not include an inspection for the problem. However, the licensee concluded all motor-operated valves were operable because there had been no in-service testing stroke failures or signature tests that indicated any degradation of these valves. The licensee initiated work orders to inspect the actuator motors of 41 “high-critical” and/or Generic Letter 89-10 Program valves known to have magnesium rotors in Refueling Outage RF-

14 (January 2008). In November 2006, the licensee reduced the scope of the inspection to a total of eight valves, with deferral of the remainder to Refueling Outage RF-15 (2009) based on no failures of magnesium-rotor motor-operated valves at River Bend Station.

Two motor-operated valves with magnesium rotors failed in 2007 during separate forced outages. The team reviewed documentation associated with the May 25, 2007, failure of Valve B21-MOV-FO65A, "Reactor Inlet Heater 'A' Outboard Motor-Operated Isolation Valve," (Condition Reports CR-RBS-2007-02175 and CR-RBS-2007-04387) and the September 29, 2007, failure of Valve B21-MOV-FO98C, "Main Steam Shutoff Valve," (Condition Report CR-RBS-2007-04305). The licensee stroked safety-related and high-critical motor-operated valves with magnesium rotors in the steam tunnel and drywell following the September 29, 2007, failure, and no additional valves failed.

Both of the valves that failed were located in the steam tunnel, a plant area with a high-temperature operating environment. Steam leaks had also occasionally occurred in the steam tunnel, resulting in a higher-humidity operating environment when steam leaks were present. After the failure of Valve B21-MOV-FO98C, the licensee sent its motor to an offsite vendor for failure analysis. The licensee also revised its magnesium rotor inspection plan in Condition Report CR-RBS-2006-02882 to include all motor-operated valves with active safety function located in the steam tunnel or drywell during the Refueling Outage RF-14 inspections. This increased the number of planned inspections to 12.

The results of the vendor's motor failure analysis confirmed the motor from Valve B21-MOV-FO98C had failed from magnesium rotor corrosion. The vendor's report recommended that magnesium rotor motors not removed from service be inspected visually with a boroscope to examine for degradation of the rotor shorting ring. Additionally, the vendor's report stated:

> "…operation of the motor may not show any signs of degradation prior to failure, it is possible that the motor will provide indication(s) of imminent failure. Such indications might include the motor taking longer to cycle the valve… the current being somewhat higher than previously … and /or the motor having trouble forward or reverse seating the valve. The operational signs may be harder or even impossible to detect…"

The licensee documented its review of the vendor's report in its apparent cause evaluation for Condition Report CR-RBS-2007-04305. The licensee's evaluation stated, in part:

> "…three things can be concluded: (1) Stroking valves does not guarantee the motor is not degraded from corrosion, (2) Even when motors are severely corroded, it does not mean they will not produce the required torque to operate the valve, and (3) No testing other than visual inspection will provide 100-percent assurance the motor is not degraded by corrosion."

The team questioned the licensee’s application of its operability determination process in response to the failures of these motor-operated valves with respect to other valves in the plant with magnesium rotors. Procedure EN-OP-104, “Operability Determinations,” Revision 3, contained the following information:

> "Determining Operability and Plant Safety is a Continual Decision-Making Process. Operability is verified by day-to-day operation, plant tours, and observations from the Control Room, surveillances, test programs, and other similar activities. Identified deficiencies in the design basis or safety analysis or operational problems identified trigger the operability determination process by which the specific deficiency and overall capability of the component or system are examined. The process, in one form or another, is ongoing and continuous.”
>
> "Since evidence may change as more information is obtained, the weight of evidence is dynamic. As the weight of evidence changes, the overall conclusion may require update. For this reason, the use of engineering judgment may require follow-up analyses, tests or inspections to confirm the validity of the conclusions reached."
>
> “A visual examination of the nonconforming/degraded equipment should be made. Any notable comparisons with similar conforming/qualified equipment should be made.”
>
> “Consideration should be made regarding the potential that the conditions assessed in the Operability Evaluation could degrade. If the possibility of degradation exists, monitoring parameters and trigger points should be developed and actions issued to trend potentially degrading conditions. This should be done in the operability evaluation or as an ODMI in accordance with EN-OP-111.”

The team noted that none of the licensee’s operability determinations regarding motor-operated valve magnesium rotor issues addressed the potential impact on operability of motor-operated valves that had been subjected previously to steam leaks or had experienced locked-rotor conditions. When questioned by the team, the licensee identified that a review of records for these events had not been performed. In response to the team’s questions, the licensee performed a review and identified that the following safety-related valves (in addition to Valve B21-MOV-FO98C) had experienced motor-stall events or a steam environment in the past:

> Valve G33-MOV-FO04
> Valve E51-MOV-FO64
> Valve B21-MOV-FO98A
> Valve B21-MOV-FO98B
> Valve B21-MOV-FO98D
> Valve FWS-MOV-7A

Based on the industry information provided in the NRC Information Notice and the licensee’s own apparent cause assessment results following failure of Valve B21-MOV-FO98C described above, the team concluded that licensee performed an inadequate evaluation of the operability of magnesium-rotor motor-operated valves. The licensee did not address the impact on operability for valves that had been subjected to previous steam leaks and/or motor-stall events.

During Refueling Outage RF-14, all of the motors in the above motor-operated valves were replaced. Although most of the motors showed signs of magnesium rotor degradation, each was found in an operable condition.

Analysis. The team determined that the failure to perform an adequate operability determination and update it as new evidence and information became available as required by Procedure EN-OP-104, "Operability Determinations," Revision 3, was a performance deficiency. The finding was more than minor because it is associated with the equipment performance attribute of the Mitigating Systems cornerstone objective of ensuring the availability, reliability, and capability of safety-related motor-operated valves to respond to initiating events to prevent undesirable consequences. Traditional enforcement does not apply because the issue did not have any actual safety consequences or potential for impacting the NRC's regulatory function, and was not the result of any willful violation of NRC requirements. In accordance with Inspection Manual Chapter 0609, Attachment 4, "Phase 1 – Initial Screening and Characterization of Findings," a Phase 1 screening was performed and determined the finding was of very low safety significance (Green) because the condition did not represent a loss of system safety function, did not represent an actual loss of safety function of a single train for greater than its Technical Specification allowed outage time, did not represent an actual loss of one or more risk-significant non-Technical Specification trains of equipment for greater than 24 hours, and did not screen as potentially risk-significant due to seismic, flooding, or severe weather. The cause of the finding had crosscutting aspects associated with the corrective action program in the problem identification and resolution area because the licensee did not thoroughly evaluate the problems with magnesium-rotor corrosion including the extent of the condition and operability impact (P.1(c)).

Enforcement. 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," states in part that, "Activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. The assessment of operability of safety-related equipment needed to mitigate accidents was an activity affecting quality and was implemented by Procedure EN-OP-104, "Operability Determinations," Revision 3. Contrary to this requirement, from May 2007 until June 2008, River Bend Station failed to adequately assess the operability of safety-related equipment as required by Procedure EN-OP-104. Specifically, the licensee failed to assess the impact on operability of previous steam leaks and motor-stall events on the corrosion of magnesium-rotors in safety-related motor-operated valves. Because this violation was of very low safety significance (Green) and has been entered into the corrective action program (CR-RBS-2008-3713 and CR-RBS-2008-3766), this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000458/2008006-05, Inadequate Implementation of Operability Determination Procedure.

2. Inadequate/Untimely Corrective Action for Failure of Magnesium-Rotor Motor-Operated Valves

Introduction. The team identified a finding of very low safety significance (Green) involving a noncited violation of 10 CFR Part 50 Appendix B, Criterion XVI, "Corrective Action," for failure to promptly identify magnesium-rotor motor-operated valve degradation after failure of Valve B21-MOV-FO65A, “Reactor Inlet Heater ‘A’ Outboard Motor Operated Isolation Valve” until after failure of Valve B21-MOV-FO98C, “Main Steam Shutoff Valve.”

Description. As described in Section 4OA2.1 of this report, the team reviewed the licensee’s operating experience assessment and actions with respect to magnesium-rotor motor-operated valves documented in Condition Report CR-RBS-2006-02882. In its July 2006 assessment, the licensee documented that there were no identified adverse trends or adverse conditions associated with magnesium-rotor motor-operated valves at River Bend Station. Therefore, the licensee concluded that an acceptable plan to address the industry operating issue was to inspect all of the magnesium-rotor motor-operated valves that were in the Generic Letter 89-10 program and/or if the valves were “high critical” during the January 2008 refueling outage (Refueling Outage RF-14). In November 2006, the licensee reduced the scope of the inspection to a total of eight valves, with deferral of the remainder to Refueling Outage RF-15 (2009), based on no failures of magnesium-rotor motor-operated valves at River Bend Station.

During a forced outage on May 25, 2007, Valve B21-MOV-FO65A, failed during operation. Although it did not have an active safety function, the team determined that the valve was safety-related. The licensee’s troubleshooting indicated that the valve’s motor was grounded and the open starter contact in the power-supply breaker was welded closed. The licensee’s Condition Report CR-RBS-2007-02175, documenting failure of the valve motor, stated, "there is a potential generic issue affecting other safety related valves due to magnesium rotor degradation which is being addressed per CR-RBS-2006-02882 for safety-related motor-operated valve motors.” The licensee repaired Valve B21-MOV-FO65A but did not perform any inspections of other magnesium-rotor motor operated valves for degradation.

During a subsequent forced outage on September 29, 2007, Valve B21-MOV-FO98C also failed during operation. The valve was part of the Main Steam Positive Leakage Control System, required by Technical Specification 3.6.1.9. Its motor failed in the same manner (grounded) as the previous failure. The licensee documented failure of this valve in Condition Report CR-RBS-2007-04305. The licensee repaired the valve, conducted a stroke test of the remaining safety-related or high-critical magnesium-rotor motor-operated valves that were located in a harsh environment, and performed a walkdown in the vicinity of the valves to check for steam leaks.

The licensee’s apparent cause evaluation in Condition Report CR-RBS-2007-04305 included a failure analysis performed by an offsite vendor. The failure of the motor was confirmed to have been caused by magnesium-rotor corrosion. The evaluation also included the following information with respect to the previous failure of Valve B21-MOV-FO65A:

> “… there was a desire on Engineering’s part to understand the failure mechanism. However, given the fact that B21-MOVFO65A performs no active

safety function and is strictly used as a maintenance valve it was evaluated as below the Condition Reporting Process threshold and closed to a low priority work order. (WO-00113396). From the time of the B21-MOV-FO65A failure and initial effort to perform a causal inspection, approximately 4 1/2 months transpired. The most recent failure of B21-MOV-FO98C once again prompted a challenge to the OCC to allocate resource and take a follow up look at the cause of the failure on B21-MOV-FO65A."

The team concluded that the degradation of motor-operated valves at River Bend Station was a nonconforming condition that was not promptly identified. The failure of Valve B21-MOV-FO65A to stroke on demand on May 24, 2007, was a symptom of a greater nonconforming condition not promptly identified.

Inspection of the motor from Valve B21-MOV-FO65A was not performed until after a second magnesium-rotor motor-operated valve (Valve B21-MOV-FO98C) failed to stroke upon demand on September 29, 2007. The team concluded the licensee failed to take adequate actions in May 2007, given indications of a common-cause mode of failure, to verify the condition of safety-related and high-critical magnesium-rotor motor-operated valves and prevent the subsequent failure of B21-MOV-FO98C in September 2007.

Analysis. The team determined that the failure to promptly identify magnesium-rotor motor-operated valve degradation and take adequate corrective actions following failure of Valve B21-MOV-FO65A to assure that safety-related and high-critical motor-operated valves were not degraded was a performance deficiency which resulted in the later discovery of an additional failed valve (Valve B21-MOVFO98C). This finding was more than minor because Valve B21-MOV-FO98C was associated with the Barrier Integrity Cornerstone and affected the cornerstone objective of providing reasonable assurance that the physical design barriers (fuel cladding, reactor coolant system, and containment) protect the public from radionuclide releases caused by accidents or events. Traditional enforcement does not apply because the issue did not have any actual safety consequences or potential for impacting the NRC's regulatory function, and was not the result of any willful violation of NRC requirements. Inspection Manual chapter 0609 Appendix H, "Containment Integrity Significance Determination Process," Table 4.1, indicated that the Main Steam Shutoff Valves do not impact large early release frequency. Based on the results of the Appendix H analysis, the finding was determined to have very low safety significance. This finding had cross-cutting aspects associated with decision-making in the human performance area in that the licensee did not use conservative assumptions in decision-making regarding the likelihood of magnesium-rotor degradation in motor-operated valves (H.1 (b)).

Enforcement. 10 CFR Part 50, Appendix B, Criterion XVI, requires that measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected. In the case of significant conditions adverse to quality, measures shall be taken to assure that the cause of the condition is determined and corrective action taken to preclude repetition. Contrary to the above, magnesium-rotor motor-operated valve degradation was not promptly identified as a nonconforming condition, responsible for failure of Valve B21-MOV-FO65A on May 24, 2007, until after a second magnesium-rotor motor-operated valve was discovered failed on September 29, 2007. Because this violation was of very low safety significance (Green) and has been entered into the corrective action program

(CR-RBS-2008-3713 and CR-RBS-2008-3766), this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000458/2008006-06, Inadequate/Untimely Corrective Action for Failure of Magnesium-Rotor Motor-Operated Valves.

4OA6 Meetings, including Exit

.1 Exit Meeting Summary

On June 6, 2008, the team presented a preliminary debriefing of the inspection results following the onsite portion of the inspection. On August 26, 2008, the team leader presented the inspection results to Mr. M. Perito, Vice President, Operations and other members of his staff who acknowledged the findings. No proprietary information was provided or examined during this inspection.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel

K. Borneman, System Engineer – 4kV/480V Distribution
R. Buell, Auditor, Quality Assurance
M. Chase, Training
F. Corley, Electrical Design Supervisor
C. Forpahl, Manager, Programs and Components Engineering
R. Heath, Superintendent, Chemistry
R. Hebert, Manager, Materials, Purchasing, and Contracts
K. Higginbotham, Assistant Operations Manager
B. Houston, Manager, Radiation Protection
K. Huffstatler, Senior Licensing Specialist
K. Jelks, System Engineer – Transformers and Switchyard
K. Klamert, System Engineer - EDG
D. Lorfing, Manager, Licensing
B. Mashburn, Manager, Design Engineering
B. Matherne, Manager, Planning, Scheduling, and Outage
O. Miller, Manager, Operations
B. Morris, motor-operated valve Engineer
E. Olson, General Manager, Plant Operations
M. Perito, Vice President, Operations
J. Roberts, Director, Nuclear Safety Assurance
C. Stout, Manager, Maintenance
T. Tankersley, Manager, Quality Assurance
D. Wiles, Director, Engineering
D. Williamson, Engineer, Licensing

NRC personnel

Troy Pruett, Deputy Director, Division of Reactor Safety
Russell Bywater, Chief, Engineering Branch 1
Grant Larkin, Senior Resident Inspector
Chuck Norton, Resident Inspector

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

None

Opened and Closed

05000458/2008006-01	NCV	Eight Examples of a Failure to Meet 10 CFR Part 50, Appendix B, “Design Control” (1R21.b.1)
05000458/2008006-02	NCV	Failure to Recalculate Suppression Pool Peak Temperature Response (1R21.b.2)
05000458/2008006-03	NCV	Inadequate Testing Programs for 4-kV Circuit Breakers, Class 1E Molded Case Circuit Breakers, and the Emergency Diesel Generators (1R21.b.3)
05000458/2008006-04	NCV	Inadequate Implementation of Temporary Installation Procedure (1R21.b.4)
05000458/2008006-05	NCV	Inadequate Implementation of Operability Determination Procedure (4OA2.1)
05000458/2008006-06	NCV	Inadequate/Untimely Corrective Action for Failure of Magnesium-Rotor Motor-Operated Valves (4OA2.2)

Closed

None

Discussed

None

LIST OF DOCUMENTS REVIEWED

Calculations

NUMBER	TITLE	REVISION
DC-304	MOV Thrust/Torque Setpoint Calculations	0
E-131	Station Service Short Circuit Analysis	1
E-132	Voltage Profile	3
E-143	Standby Battery ENB-BAT01A Duty Cycle, Current Profile and Size Verification	9
E-144	Standby Battery ENB-BAT01B Duty Cycle, Current Profile and Size Verification	5
E-164-4	Procedure for selecting trip coils, and motor overload heaters	4
E-167	5kV Power Cable Sizing	1
E-176	Standby LC, MCC, & 120V Panel Short Circuit Calc	2
E-192	Standby Diesel Generator Loading Calculation	5
E-200	Over current Devices Set Points	1
E-201	Protective Relaying Set Points	2
E-209	Cable Loop Length Criteria for Voltage Drop – D.C. Circuits	1
E-209, Addendum G	Cable Loop Length Criteria for Voltage Drop – DC Circuits	1G
E-210	Cable Loop Length Criteria for Voltage Drop	1
E-218	Ampacity Verification of Cables Within Raceways Wrapped with App R Protection	1
E-219	480VAC MCC Load Tabulation	1
E-222	Load Tabulation for 480V Load Centers	0

Calculations

NUMBER	TITLE	REVISION
E225	Voltage Calculation of Cat 1 480V MOVS	5
G13.18.2.1*081-0	Evaluation of Hydrogen Accumulation and Ventilation Requirements for Control Building Division III Replacement	0
G13.18.2.1*083	Diesel Generator Building Design Basis Calculation - Summer Condition	0
G13.18.2.3*155	GL 89-10 Design Basis Review for E12-MOVFO24 A	5
G13.18.2.3*160	GL 89-10 Design Basis Review for E12-MOVFO42 A/B	3
G13.18.2.3*166	GL 89-10 Design Basis Review for E12-MOVFO64 A/B	3
G13.18.2.3*167	G.L. 89-10 Design Basis Review for E12-MOVF068	2
G13.18.2.3*167	G.L. 89-10 Design Basis Review for E12-MOVF068 A	2A
G13.18.2.3*167	G.L. 89-10 Design Basis Review for E12-MOVF068 A	2C
G13.18.2.3*185	GL 89-10 Design Basis Review for E22-MOVFO12	5
G13.18.2.3*202	GL 89-10 Design Basis Review for E51-MOVFO45	4
G13.18.2.3*293	GL 89-10 Design Basis Review for SWP-MOVFO55 A/B	1
G13.18.2.3*325	Grid Voltage Operability Evaluation	0
G13.18.2.3*325	Grid Voltage Operability Evaluation Addendum A	0A
G13.18.2.3*325	Grid Voltage Operability Evaluation Addendum C	0C
G13.18.2.4*25	Line Size Adequacy of 2” Overflow line of Day Tank for Standby Diesel Generator	0

Calculations

NUMBER	TITLE	REVISION
G13.18.2.6*39	Lube Oil Consumption For Division I, II, and III DG	0
G13.18.2.6*068-0	Division I, II &III Diesel Generator Lube Oil Sump Dipstick Markings for Technical Specification Compliance	NA
G13.18.3.1*001	Sustained & Degraded Voltage Relay Setpoints	3
G13.18.3.1*001	Sustained & Degraded Voltage Relay Setpoints for ENS-SWG01A/01B	3 Addendum
G13.18.3.6*5	Coordination Study of Appendix R and Class 1E Low Voltage Protection Devices	1
G13.18.3.6*016	Degraded Voltage Calculation for Class 1E Buses	0 Addendum
G13.18.3.6*018	PowerStation Data Base Input Study	1
G13.18.3.6*021	DC System Analysis, Methodology & Scenario Development	0
G13.18.6.1-RHR*01	Setpoint Calc for RHR Time Delay Relay	0
G13.18.6.2-ENS*005	Loop Uncertainty Determination for Degraded Voltage Relays	0
G13.18.10.1-014	Standby Diesel Generator Fuel Oil Storage tank Capacity	0
PRA-RB-01-005	River Bend PSA Summary Report	0
2005-11023-1-R01.2	Riverbend Plant Study	07/29/05

Condition Reports

CR-RBS-1999-00930	CR-RBS-2000-01764	CR-RBS-2003-03531	CR-RBS-2004-02236
CR-RBS-2004-01679	CR-RBS-2004-00428	CR-RBS-2005-01238	CR-RBS-2005-02515
CR-RBS-2005-01238	CR-RBS-2005-02515	LO-RLO-2006-00125	CR-RBS-2006-00170
CR-RBS-2006-00350	CR-RBS-2006-00424	CR-RBS-2006-00283	CR-RBS-2006-00305
CR-RBS-2006-00424	CR-RBS-2006-00506	CR-RBS-2006-00933	CR-RBS-2006-00633
CR-RBS-2006-02815	CR-RBS-2006-02882	CR-RBS-2006-03262	CR-RBS-2006-03776
CR-RBS-2006-03860	CR-RBS-2006-04478	CR-RBS-2006-04479	LO-NOE-2007-00013
CR-RBS-2007-00359	CR-RBS-2007-00448	CR-RBS-2007-01287	CR-RBS-2007-02466
CR-RBS-2007-04490	CR-RBS-2007-05223	CR-RBS-2007-05394	CR-RBS-2008-00130
CR-RBS-2008-01017	CR-RBS-2008-01287	CR-RBS-2008-02076	CR-RBS-2008-02091
CR-RBS-2008-03226	CR-RBS-2008-02284	CR-RBS-2008-03206	CR-RBS-2008-03262
CR-RBS-2008-03269	CR-RBS-2008-03270	CR-RBS-2008-03272	CR-RBS-2008-03274
CR-RBS-2008-03275	CR-RBS-2008-03276	CR-RBS-2008-03277	CR-RBS-2008-03278
CR-RBS-2008-03279	CR-RBS-2008-03339	CR-RBS-2008-03449	CR-RBS-2008-03556
CR-RBS-2008-03558	CR-RBS-2008-03574	CR-RBS-2008-03634	CR-RBS-2008-03638
CR-RBS-2008-03641	CR-RBS-2008-03654	CR-RBS-2008-03699	

Drawings

NUMBER	TITLE	REVISION
BE-230A	4kV Bus 1ENS-SWG1A Relay Settings	8
BE-230F	Standby DG Protective Relay Settings	8
BE-260B	CB Trip Device Settings – 480V Bus 1EJS*SWG2A	7
BE-260G	Trip Coil Settings – Safety Related MOV's	2
BE-270B	Circuit Breaker Trip Device Settings 125V DC Bus ENB-SWG01B	NA
0242.562-082-092	Front View and BOM 1ENS-MCC2E	1
EE-001AA	480V Standby Bus 1EJS*LDC 1A/2A	16
EE-001AC	Startup Electrical Distribution Chart	39
EE-001K	4160V Standby Bus ENS-SWG1A	19

NUMBER	TITLE	REVISION
EE-001S A	480V One Line Diagram 1E22-500 – Control Building	11
EE-001 TB	480V One Line Diagram EHS-MCC2C &2D – Auxiliary Building	11
EE-001TC	480V One-Line Diagram ENS-MCC2E	10
EE-001 YA	480V One Line Diagram EHS-MCC16 A – Standby Cooling Tower No. 1	12
EE-001ZC	One-line Diagram STBY Bus A & B Low Voltage Distribution System	14
EE-001 ZD	125 VDC One Line Diagram ENB-MCC1 – Auxiliary Building	6
EE-001ZH	125VDC One-line Diagram Standby Bus B 1ENB*SWG01B, 1ENB-PNL02B, 03B	22
EE-001ZJ	125VDC One-line Diagram Normal & Standby Backup Charger Sys	17
ESK-2J	Instruction Drawing 4160V Switchgear Details	6
ESK-5ENS01	Elementary Diag – 4.16kV Swgr Stby Bus 1A Norm Sply ACB	18
ESK-5ENS06	Elementary Diagram 4.16kV Switchgear Standby Bus 1A Gen ACB	20
ESK-8EGS01	Elementary Diagram Standby Diesel Gen. 1EGS*EG1A Prot. & Mtr.	9
ESK-08EGS09	STBY BUS ENS*SWG1A Undervoltage Protection	13
ESK-08EGS11	Elementary Diagram STBY GEN 1EGS*EG1A Excitation	6
ESK-08EGS13	STBY BUS ENS*SWG1A Undervoltage Protection	11
ESK-08EGS15	STBY BUS ENS*SWG1A UV Protect & Load Seq	9
ESK-11EGA01	Elem Diag 125VDC Control Stby Dsl1A Rear Start Ckt	20
ESK-11ENB02	Elementary Diagram 125VDC Standby SWGR Battery Systems	11
ESK-11ENB07	Elem Diag~125VDC Standby SWGBR Battery Systems	4
ESK-11ENB08	Elementary Diagram 125VDC Division III Battery System	8
KA-0221.434-000-017	Process Diagram, Residual Heat Removal System	0
KC-0244.700-041-113	Interconnection Diagram Engine Generator EGS-PNL3A	0

NUMBER	TITLE	REVISION
KC-0244.700-041-114	Engine Generator Interconnection	0
PID-09-10E	Service Water – Standby	20
PID-09-10F	Service Water – Normal	29
PID-09-10D	Service Water – Normal	33
PID-27-04A	HPCS System	26
PID-27-07A	Residual Heat Removal – LPCI	36
PID-27-07B	Residual Heat Removal – LPCI	40
PID-27-07C	Residual Heat Removal – LPCI	25
0244.700-041-083	Control Panel Schematic for EGS-PNL3A Standby Diesel Generator EGS-EG1A	G

Modifications

NUMBER	TITLE	REVISION
ER-RB-2001-0360-000	Replace the existing ITE 27H relays with ABB Model 27N for the Div I/II Degraded Voltage Relays	06/28/01

Procedures

NUMBER	TITLE	REVISION
AOP-0004	Loss of Off-Site Power	31
AOP-0009	Loss of Normal Service Water	15
AOP-0014	Loss of 125VDC	20
AOP-0016	Loss of Standby Service Water	015
AOP-0020	Alternate Method of Decay Heat Removal	2
AOP-0050	Station Blackout	025
AOP-0051	Loss of Decay Heat Removal	304
AOP-0053	Initiation of Standby Service Water with Normal Service Water Running	10
AOP-0059	ECCS Suction Strainer Blockage	5

NUMBER	TITLE	REVISION
AOP-0064	Degraded Grid	0
DC-312	Motor Operated Valve Test Data Review Standard	1
E12-MOVFO24A-ST-005	Votes MOV Test Report	11/30/99
E12-MOVFO42A-ST-005	Votes MOV Test Report	03/31/03
E12-MOVFO64A-ST-005	Votes MOV Test Report	05/05/06
E12-MOVFO68A-ST-005	Votes MOV Test Report	03/26/03
E22-MOVFO12-ST-005	Votes MOV Test Report	11/01/04
E51-MOVFO45-ST-004	Votes MOV Test Report	10/05/01
EDP-AA-20	Engineering Calculations	17
EDP-ME-25	Design Basis Review for Motor Operated Valves	3
EDP-ME-26	Stem Thrust/Torque Evaluation for Motor Operated Valves	6
EN-DC-126	Engineering Calculation Process	1
ENS-DC-199	Offsite Power Supply Design Requirements	2
ENS-DC-201	ENS Transmission Grid Monitoring	2
EN-LI-100	Process Applicability Determination	6
EN-LI-101	10 CFR 50.59 Review Program	4
EN-MP-112	Shelf Life Program	2
EN-OP-104	Operability Determination	
EN-OP-115	Conduct of Operations	005
EOP-1A	RPV Control, ATWS	021
EOP-2	Primary Containment Control	014
EOP-003	Emergency Operating Procedure – Secondary Containment and Radioactive Release Control	13
EOP-4	RPV Flooding	013
EOP-4A	RPV Flooding, ATWS	013

NUMBER	TITLE	REVISION
EOP-5 enclosure 15	Alternate SLC Injection	301
OSP-0063	Grid Monitoring	1
PM T429	ABB 5HK Clean/Inspect	7/10/2007
PM T431	Post Maintenance Testing 5HK Breaker	1/25/2007
PM T1989	DC Circuit Breaker Major Clean/Inspect	2/28/2008
PM T10258	Post Maintenance Test ABB K800 Breaker	1/16/2008
RSMS-OPS-428	Grid Instability / Drywell Steam Leak	000
SAP-1	RPV and Primary Containment Control	005
SAP-2	Containment and Radioactivity Release Control	003
SDC-309	Standby Diesel Generator Division I and II	3
SOP-42	Standby Service Water	028
SOP-48	120 Vac	311
SOP-49	125 Vdc	026
SOP-54	Station Blackout Diesel Generator	301
STP-203-1605	E22-S001CGR Load Test	18
STP-203-6305	HPCS Quarterly Pump and Valve Operability Test	19
STP-203-6805	HPCS Cold Shutdown Valve Operability Test	10
STP-204-6301	Div I LPCI (RHR) Quarterly Pump and Valve Operability Test	21
STP-204-6303	Div I RHR Quarterly Valve Operability Test	16
STP-204-6801	Div I ECCS Cold Shutdown Valve Operability Test	11
STP-209-6310	RCIC Quarterly Pump and Valve Operability Test	28
STP-256-6301 (2)	Div I Standby Service Water Quarterly Valve Operability Test	13
STP-256-6310 (2)	RCIC Quarterly Pump and Valve Operability Test	28
STP-302-0102	Power Distribution System Operability Check	16
STP-303-1601	120/480VAC Breaker Overload Functional Test	24
STP-303-1609	Div 1 Over current Protective Device and Breaker Test	13
STP-303-1700	120/480VAC Breaker Inspection	16

NUMBER	TITLE	REVISION
STP-305-1607	ENB-BAT01B Service Discharge Test	18
STP-305-1701	ENB-BAT01B Performance Discharge Test	24
STP-305-1101	ENB-BAT01B Weekly Surveillance	21
STP-305-1301	ENB-BAT01B Quarterly Surveillance	20
STP-305-1601	ENB-BAT01B Inspection	11
STP-305-1604	ENB-CHGR1B Load Test	301
STP-309-0203	Division 3 Diesel Generator Operability Test	Rev 26A
STP-309-0203	Division 3 Diesel Generator Operability Test	06/14/06
STP-309-0207	Division II Diesel Generator 184 Day Operability Test	Rev 00
STP-309-0602	Division II ECCS Test	Rev 26
STP-309-0602	Division II ECCS Test	Rev 23
STP-309-0603	Division III 18 Month ECCS Test	Rev 24
STP-309-0612	Division II Diesel Generator 24 Hour Run	Rev 17
SWP-MOV55A-ST-004	Votes MOV Test Report	05/05/06
T302	Clean and Inspect MCC	01/29/07
3242562-082-016A	Motor Overload Heater Selection Procedure	05/07/96

Work Orders

NUMBER	TITLE	DATE
50972768	RTX-XSR1C – Inspect	05/03/06
50972950	ENS-SWG1A Clean and Inspect	05/04/06
51042898	STP-309-0601 Div I EECS	03/03/08
50374112	ENS SWG1A – Protective Relay Test	02/16/05
51565454	Thermography of RTX-XSR1C	05/27/08
MAI 363504	PM – 480V MCC 2E	01/09/03
MWR 35634	Transformer 1EJS*X2A – Voltage Tap	04/06/86
51041842	STP-302-1602 ENS-SWG1A DV Channel Calibration	02/17/08

NUMBER	TITLE	DATE
51042891 01	STP-309-0602: Division II ECCS Test	1/6/2008
00116756 01	ENS-SWG1B ACB25 Refurbish Breaker	1/3/2008
51650708 01	ENS-SWG1B ACB27 Refurbish Breaker	4/8/2008
00116803 01	ENB-SWG1B ACB584 Refurbish Breaker	8/24/2007
51034741	PERFORM A STATIC SIGNATURE TEST ON E22-MOVF012	07/11/06
51043605	E22-MOVF012 - CLEAN, INSPECT, LUBRICATE, VALVE OPERATOR	10/19/06
00111610	E22-MOVF012 - CLEAN, INSPECT, LUBRICATE, VALVE OPERATOR	01/29/08
51015595	MINOR INSPECTION OF E22-MOVF012.	08/06/07
51041893	STP-204-6801: DIV I ECCS COLD SHUTDOWN VALVE OPERABILITY TEST	11/02/06
51204759	E12-MOVF042A - CLEAN, INSPECT, INSULATION TEST, LUBRICATE	05/07/08
00124689	E12-MOVF042A - REPACK AND PERFORM A STATIC SIGNATURE TEST E1	12/14/07
51039520	MINOR, INSPECT, E12-MOVF042A	04/09/08
51054946	STP-204-6801: DIV I ECCS COLD SHUTDOWN VALVE OPERABILITY TEST	02/21/08
00124534	STP-204-6801: DIV I ECCS COLD SHUTDOWN VALVE OPERABILITY TEST	02/21/08
00036749	PERFORM A STATIC SIGNATURE TEST ON E51-MOVF045	05/07/08
51043999	E51-MOVF045 MINOR INSPECTION	04/29/08
50995669	E12-MOVF024B - CLEAN, INSPECT, INSULATION TEST, AND LUBRICATE	01/31/07
50993919	E12-MOVF064A - PERFORM A STATIC SIGNATURE TEST	05/09/06
51008880	E12-MOVF064A - MINOR INSPECTION OF E12-MOVF064A.	02/14/07
00081712	E12-MOVF068A - PERFORM A STATIC SIGNATURE TEST ON E12-MOVF06	09/12/06

NUMBER	TITLE	DATE
00116885	E12-MOVF068A - PERFORM A STATIC SIGNATURE TEST ON E12-MOVF06	02/07/08
00125837	SWP-MOV55A - PERFORM A STATIC SIGNATURE TEST ON SWP-MOV55A.	02/09/08
00095288	THERMAL LAG ON CONDUIT HAS DETERIORATED	02/15/07

Engineering Requests

NUMBER	TITLE	REVISION/ DATE
98-0007	Resolution of SDC-203/305 and SDC-305/306 open items	12/22/1997
99-0144	Division I and II Diesel Generator K1 Relay Reset Coil Pressure Switch Setpoint Change	10/28/1999
RB-2004-0131-000	Replacement of AK / AKR Breakers w/ New Maintenance Free Masterpact Breakers	3/10/2004
EC-4861	Emergency Diesel Generator EGS-EG1B turbocharger discharge combustion air piping cracking	0
EC-275	Emergency Diesel Generator EGS-EG1A and EGS-EG1B Air Starting System, elimination of water trap	0

Miscellaneous Documents

NUMBER	TITLE	DATE/ REVISION
300	System Health Report – 230 kV Electric Distribution	0
302	System Health Report - 4.16 kV Electric Distribution	0
303	System Health Report - 480 VAC Electric Distribution	0
309	System Health Report – Standby EDG Div I, II, & III	0
311	System Health Report – Main & Station Transformers	0

NUMBER	TITLE	DATE/ REVISION
CEP-IST-1	In-service Testing Bases Document, Entergy Nuclear South, Central Engineering Programs	301
CEP-IST-1	RBS Appendix to In-service Testing Bases Document, Entergy Nuclear South, Central Engineering Programs	4
DRN 06-39	Battery ENB-BAT01B Duty Cycle, Current Profile and Size Verification	7/03/2006
DRN 06-178	Battery ENB-BAT01B Duty Cycle, Current Profile and Size Verification	4/30/2007
MR93-0009	Back-up Power to Support Safety-Related Control Power during SBO	2/8/1993
MAI 327931	Replace the ENB-BAT01B Battery Cells (60 each)	10/01/2001
MAI 331569	Perform Initial Torque/Testing of the new ENB-BAT01B battery	10/05/2001
Purchase Order 10184541	Refurbishment of 5HK250 1200A Circuit Breaker	2/2008
Purchase Order 10163309	Refurbishment of K800 125VDC Circuit Breaker	10/12/2007
Purchase Order 10150771, Revision 6	Overhaul of 5HK250 1200A Circuit Breaker	6//2007
SDC-402&410, Revision 2	Control Bldg. HVAC System, Control Bldg Chilled Water System, Ventilation Chilled Water System Design Criteria System Numbers 402, 410 &410	5/05/2003
	Letter dated October 21, 1985 from Stone & Webster Engineering to Mr. D. P. Barry – RE: Purchase Order 242.521-102, River Bend NPGS 5HK Circuit Breaker Control Components	10/21/1985