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

August 14, 2017

Mr. William F. Maguire, Vice President
Entergy Operations, Inc.
River Bend Station
5485 US Highway 61N
St. Francisville, LA 70775

SUBJECT: RIVER BEND STATION – NRC DESIGN BASES ASSURANCE INSPECTION REPORT 05000458/2017007

Dear Mr. Maguire.:

On June 30, 2017, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your River Bend Station and discussed the results of this inspection with Mr. S. Vazquez, Director, Engineering, and other members of your staff. The results of this inspection are documented in the enclosed report.

NRC inspectors documented two findings of very low safety significance (Green) in this report. Both of these findings involved violations of NRC requirements. The NRC is treating these violations as non-cited violations (NCVs) consistent with Section 2.3.2.a of the Enforcement Policy.

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

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

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

Sincerely,

/RA/

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

Docket No. 50-458
License No. NPF-47

Enclosure:
Inspection Report 05000458/2017007
w/Attachment: Supplemental Information

cc w/ enclosure: Electronic Distribution

RIVER BEND STATION – NRC DESIGN BASES ASSURANCE INSPECTION
REPORT 05000458/2017007 – August 14, 2017

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U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Docket:	05000458
License:	NPF-47
Report Nos.:	05000458/2017007
Licensee:	Entergy Operations, Inc.
Facility:	River Bend Station
Location:	River Bend Station 5485 US Highway 61N St. Francisville, LA 70775
Dates:	June 12 through June 30, 2017
Team Leader:	W. Sifre, Senior Reactor Inspector, Engineering Branch 1
Inspectors:	R. Latta, Senior Reactor Inspector, Engineering Branch 1 W. Cullum, Reactor Inspector, Engineering Branch 1 N. Okonkwo, Reactor Inspector, Engineering Branch 2 R. Smith, Nuclear Systems Engineer, Response Coordination Branch
Accompanying Personnel:	C. Baron, Contractor, Beckman and Associates S. Gardener, Contractor, Beckman and Associates
Approved By:	Thomas R. Farnholtz Branch Chief, Engineering Branch 1 Division of Reactor Safety

SUMMARY

IR 05000458/2017007; 06/12/2017 – 06/30/2017; River Bend Station; Baseline Inspection, NRC Inspection Procedure 71111.21M, "Design Basis Assurance Inspection."

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

Cornerstone: Mitigating Systems

- Green. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," which states, in part, "Activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings." Specifically, on October 28, 2014, the licensee failed to perform extent of condition on other safety-related 4.16KV Magne Blast circuit breakers due to the failure of 4.16KV Magne Blast circuit breaker ACB03 on bus E22-S003, with damaged brush and misaligned brush holder of the circuit breaker charging motor, in accordance with Procedure EN-OP-104, "Operability Determination Process." Failure to perform this evaluation could adversely impact safety-related circuit breakers. In response to this issue the licensee reviewed their breaker performance records to assure that no additional failures had occurred and revised the procedure to assure that extent of condition is addressed. This finding was entered into the licensee's corrective action program as Condition Report CR-RBS-2017-05078.

  The team determined the failure to evaluate the impact of a damaged brush and misaligned brush holder of the charging motor of a safety-related 4.16KV Magne Blast breaker was a performance deficiency. The performance deficiency was more than minor, and therefore a finding, because it related to the equipment performance attribute of the Mitigating Systems Cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, failure to perform an extent of condition on other safety-related 4.16KV Magne Blast circuit breakers due to the failure of E22-S003 safety-related circuit breaker ACB03, 4.16KV Magne Blast breaker with damaged brush and misaligned brush holder could adversely affect the ability of these breakers to perform their safety functions. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated July 19, 2012, the finding screened as having very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non-technical specification equipment; and did not screen as potentially risk-significant due to seismic, flooding, or severe weather. This finding had a cross-cutting aspect in the area of human performance associated with conservative bias because the licensee failed to ensure that individuals used decision-making practices that emphasized prudent choices [H.14]. (Section 1R21.2.4.b)

- Green. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Action,” which states, in part, 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. Specifically, between June 15, 2017, and June 28, 2017, the licensee failed to address the operability of a terminal block installed within an unsealed junction box. In response to this issue the licensee performed an operability determination to ensure that the terminal block would perform its design function in this condition. This finding was entered into the licensee’s corrective action program as Condition Report CR-RBS-2017-05084.

  The team determined that the failure to perform an adequate operability determination was a performance deficiency. The finding was determined to be more than minor because it was associated with the Mitigating Systems Cornerstone attribute of equipment performance and affected the cornerstone objective of ensuring the availability, reliability, and capability of mitigating systems to respond to initiating events to prevent undesirable consequences. Specifically, the failure to ensure operability of valve E51-AOVF054 and its associated circuits would impact the operability of the reactor core isolation cooling system. In accordance with Inspection Manual Chapter 0609, Appendix A, “The Significance Determination Process (SDP) for Findings At-Power,” dated June 19, 2012, Exhibit 2, “Mitigating Systems Screening Questions,” the issue screened as having very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of nontechnical specification equipment; and did not screen as potentially risk-significant due to seismic, flooding, or severe weather. This finding had a cross-cutting aspect in the area of problem identification and resolution associated with resolution because the licensee failed to take effective corrective actions to address issues in a timely manner commensurate with their safety significance. Specifically, the licensee failed to perform an adequate operability determination for an identified condition [P.3]. (Section 1R21.3.4.b)

REPORT DETAILS

1. REACTOR SAFETY

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

1R21 Component Design Bases Inspection (71111.21M)

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

.1 Inspection Scope for Components Selected

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

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

The team performed a margin assessment and detailed review of the selected risk-significant components to verify that the design bases have been correctly implemented and maintained. This design margin assessment considered original design issues, margin reductions because of modifications, and margin reductions identified as a result of material condition issues. Equipment reliability issues were also considered in the selection of components for detailed review. These included items such as failed performance test results; significant corrective actions; repeated maintenance; 10 CFR 50.65(a)1 status; operable, but degraded conditions; NRC resident inspector input of problem equipment; system health reports; industry operating experience; and licensee problem equipment lists. Consideration was also given to the uniqueness and complexity of the design, operating experience, and the available defense in-depth margins.

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

The inspection procedure requires a review of 4 to 6 components based on risk significance and 4 to 6 modifications to mitigation structures, systems, and components. One of the inspection samples selected shall be associated with containment-related structures, systems, and components which are considered for large early release frequency (LERF) implications. The samples selected for this inspection were 14 components (1 containment-related component), 4 modifications, and 4 operating experience items.

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

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

  - 125 VDC Division II Battery, 1ENB-BAT01B and Battery Charger, 1ENB-CHGR1B

  - 125 VDC Bus, 1ENB-SWG01B

  - Voltage Regulator for Emergency Diesel Generator, EGS-EG1A

  - Electrical Bus 4.16KV Switchgear, E22-S004, for High Pressure Core Spray

  - High Pressure Core Spray Pump E22-PC001 Motor

- Initiating events minimization:
    - High Pressure Core Spray Pump E22PC001
    - High Pressure Core Spray Injection Valve E22MOVF004

- Decay heat removal:
    - Residual Heat Removal Heat Exchanger A, E12-EB001A and E12-EB001C
    - Residual Heat Removal Heat Exchanger A Bypass Valve, E12-MOVF048A
    - Residual Heat Removal A Suction Strainer
    - Residual Heat Removal Pump A
    - Residual Heat removal Minimum Flow Valve 46A
    - Residual Heat Removal Pump A Discharge Check Valve

- Containment integrity following design basis accident:
    - Residual Heat Removal Motor Operated Valves E12-F069 and F094

.2 <u>Results of Detailed Reviews of Components</u>

.2.1 <u>125 VDC Division II Battery, 1ENB-BAT01B and Battery Charger, 1ENB-CHGR1B</u>

a. <u>Inspection Scope</u>

The team reviewed the updated safety analysis report, system description, design basis documents, the current system health report, selected drawings and calculations, maintenance and test procedures, and condition reports associated with the 125 VDC battery, 1ENB-BAT01B, and Battery Charger, 1ENB-CHGR1B. The team also performed walkdowns and conducted interviews with system and design engineering personnel to ensure the capability of these components to perform their desired design basis function. Specifically, the team reviewed:

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

- Operations surveillance history to verify monitoring of battery room conditions as assumed in battery capacity calculations.

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

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

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

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

b. Findings

No findings were identified.

.2.2 125 VDC Bus, 1ENB-SWG01B

a. Inspection Scope

The team reviewed the updated safety analysis report, system description, the current system health report, selected drawings, maintenance procedures, test procedures, and condition reports associated with the 125 VDC bus, SWG01B. The team also performed walkdowns and conducted interviews with system engineering personnel to ensure capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- Component maintenance history and corrective action program reports to verify the monitoring of potential degradations.
- Procedures for preventative maintenance, inspection, and testing to compare maintenance practices against industry standards and vendor guidance.
- Calculations for electrical distribution, system load flow/voltage drop, short circuit, and electrical protection to verify that bus capacity and voltages remained within minimum acceptable limits.

b. Findings

No findings were identified.

.2.3 Voltage Regulator for Emergency Diesel Generator, EGS-EG1A

a. Inspection Scope

The team reviewed the updated safety analysis report, system description, the current system health report, selected drawings, maintenance procedures, test procedures, and condition reports associated with the voltage regulator for emergency diesel generator, EGS-EG1A. The team also performed walkdowns and conducted interviews with system engineering personnel to ensure capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- Component maintenance history and corrective action program reports to verify the monitoring of potential degradations.
- Procedures for preventative maintenance, inspection, and testing to compare maintenance practices against industry.

b. Findings

No findings were identified.

.2.4 Electrical Bus 4.16KV Switchgear, E22-S004, for High Pressure Core Spray

a. Inspection Scope

The team reviewed the updated safety analysis report, system description, design basis documents, the current system health report, selected drawings and calculations, maintenance and test procedures, and condition reports associated with Electrical Bus E22-S004. The team also performed walkdowns and conducted interviews with system engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

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

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

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

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

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

b. Findings

Failure to Evaluate the Extent of Condition of a Degraded 4.16KV Magne Blast Safety-Related Circuit Breaker

Introduction: The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” involving the failure to perform extent of condition to evaluate the impact of a damaged brush and misaligned brush holder on the charging motor of a safety-related 4.16KV circuit breaker, on other Safety-related circuit breakers of similar make and model.

Description: Feeder circuit breaker E22-ACB3 is a 1200 A, 4.16KV Magne Blast breaker. This Magne Blast circuit breaker which feeds power to the high pressure core spray transformer, E22-S002, is located in 4.16KV standby switchgear, E22-S004.

While implementing work order WO 396046-01 to perform preventive maintenance on Magne Blast 4.16KV circuit breaker E22-ACB03, the licensee observed that the circuit breaker charging motor had a damaged brush and misaligned brush holder. The licensee declared the circuit breaker inoperable and subsequently initiated Condition Report CR-RBS-2014-05472. The team reviewed the corrective actions in Condition Report CR-RBS-2014-05472, associated with the failure of E22-ACB03, and observed that on October 28, 2014, the licensee declared the structure, system, and component inoperable and as a corrective action, promptly repaired the damaged brushes and properly aligned the misaligned brush, in WO 396046-03, but failed to review the impact of the condition on other safety-related breakers of the same make and model.

The licensee failed to implement extent of condition in accordance with Procedure EN-OP-104, "Operability Determination Process," Section 5.2.6, which requires the licensee to "Determine the extent of condition for all similar SSCs;" specifically, to perform investigation as needed to determine if damaged brush and misaligned brush holder of a safety-related 4.16kV breaker charging motor condition exist in the other SSCs.

Contrary to that provision, the licensee tracked the condition under 1-TS-14-DIV 3 Dist. System-902, repaired the condition under Work Order WO-396046-03, but failed to perform an evaluation to ensure that the failure did not impact the ability of other safety-related 4.16KV Magne Blast circuit breakers to perform their intended safety function.

<u>Analysis</u>. The team determined the failure to evaluate the impact of a damaged brush and misaligned brush holder of the charging motor of a safety-related 4.16KV Magne Blast breaker was a performance deficiency. The performance deficiency was more-than-minor, and therefore a finding, because it related to the equipment performance attribute of the Mitigating Systems Cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, failure to perform an extent of condition on other safety-related 4.16KV Magne Blast circuit breakers due to the failure of E22-S003 safety-related circuit breaker ACB03, 4.16KV Magne Blast breaker with damaged brush and misaligned brush holder could adversely affect the ability of these breakers to perform their safety functions. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012, the finding screened as having very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non- technical specification equipment; and did not screen as potentially risk-significant due to seismic, flooding, or severe weather. This finding had a cross-cutting aspect in the area of human performance associated with conservative bias because the licensee failed to ensure that individuals used decision making-practices that emphasized prudent choices [H.14].

<u>Enforcement</u>. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," which states, in part, "Activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings." Contrary to the above, on October 28, 2014, the licensee failed to perform activities in accordance with instructions, procedures and drawings. Specifically, the licensee failed to assess the

extent of condition on other safety-related 4.16KV Magne Blast circuit breakers due to the failure of 4.16KV Magne Blast circuit breaker ACB03 on bus E22-S003, with damaged brush and misaligned brush holder of the circuit breaker charging motor, in accordance with Procedure EN-OP-104, "Operability Determination Process." Failure to perform this evaluation could adversely impact safety-related circuit breakers. In response to this issue the licensee reviewed their breaker performance records to assure that no additional failures had occurred and revised the procedure to assure that extent of condition is addressed. This finding was entered into the licensee's corrective action program as Condition Report CR-RBS- 2017-05078. Because this finding was of very low safety significance and has been entered into the licensee's corrective action program, this violation is being treated as a non-cited violation, consistent with Section 2.3.2.a of the NRC Enforcement Policy: NCV 05000458/2017007-01, "Failure to Evaluate the Extent of Condition for a Degraded 4.16KV Magne Blast Safety-Related Circuit Breaker."

.2.5 High Pressure Core Spray Pump E22-PC001 Motor:

a. Inspection Scope

The team reviewed the updated safety analysis report, system description, design basis documents, the current system health report, selected drawings and calculations, maintenance and test procedures, and condition reports associated with high pressure core spray pump (E22-PC001) motor. The team also performed walkdowns and conducted interviews with system engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

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

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

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

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

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

b. Findings

No findings were identified.

.2.6 High Pressure Core Spray Pump E22PC001

a. Inspection Scope

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

- Surveillance test procedures and frequency
- Calculations for determining injection flow and timing requirements
- Inservice test data for pump performance
- Vendor recommendations for pump maintenance and performance

b. Findings

No findings were identified.

.2.7 High Pressure Core Spray Injection Valve E22MOVF004

a. Inspection Scope

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

- Inservice test data for valve stroke times
- Vendor recommendations for maintenance
- Surveillance requirements for high pressure core spray injection
- Plans to address operating experience for anchor/darling valve failures

b. Findings

No findings were identified.

.2.8 Residual Heat Removal Heat Exchanger A, E12-EB001A and E12-EB001C

a. Inspection Scope

The team reviewed the updated safety analysis report, system description, design basis documents, the current system health report, selected drawings and calculations, maintenance and test procedures, and condition reports associated with the “A” residual heat removal heat exchanger, E12-EB001A and E12-EB001C. The team also performed walkdowns and conducted interviews with system and design engineering personnel to ensure the capability of these components to perform their desired design basis function. Specifically, the team reviewed:

- Component maintenance history and corrective action reports to verify potential degradation of the heat exchangers.
- Thermal performance calculations for the heat exchangers under both test and postulated accident conditions.
- Operating procedures related to putting the heat exchangers in service in the event of a postulated loss of cooling accident during shutdown cooling operations.
- Recent thermal performance testing results and the analysis of the test results to verify adequate thermal performance under accident conditions.
- Analysis of test instrument uncertainty to verify the validity of heat exchanger thermal performance testing results.

b. Findings

No findings were identified.

.2.9 Residual Heat Removal Heat Exchanger A Bypass Valve, E12-MOVF048A

a. Inspection Scope

The team reviewed the updated safety analysis report, system description, design basis documents, the current system health report, selected drawings and calculations, maintenance and test procedures, and condition reports associated with the “A” Residual Heat Removal Heat Exchanger Bypass Valve, E12-MOVF048A. The team also performed walkdowns and conducted interviews with system and design engineering personnel to ensure the capability of this valve to perform its desired design basis function. Specifically, the team reviewed:

- Component maintenance history and corrective action reports to verify potential degradation of the valve.
- Valve capability calculations for the valve to verify its capacity to open and close under the most limiting operating conditions.
- Design and testing of the control circuit for the automatic and manual operation of the valve to verify the valve will operate under postulated accident conditions.
- Design and testing of the motor-operated valve thermal overload protection to verify that the valve will be protected under test conditions while ensuring operation under postulated accident conditions.

b. Findings

No findings were identified.

.2.10 Residual Heat Removal A Suction Strainer

a. Inspection Scope

The team reviewed the updated safety analysis report, system description, design basis documents, the current system health report, selected drawings, inspection procedures, and condition reports associated with the “A” Residual Heat Removal Suction Strainer. The team also conducted interviews with system and design engineering personnel to ensure the capability of the strainer to perform its desired design basis function. Specifically, the team reviewed:

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

- Completed inspection procedure of the strainer to verify its material condition and the scope of the inspections.

- Photographs of the strainers from the most recent inspection to verify its material condition.

b. Findings

No findings were identified.

.2.11 Residual Heat Removal Pump A:

a. Inspection Scope

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

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

- Calculations for required net positive suction head, system hydraulic analyses to assure the pump will provide the required flow and pressure under the most limiting design basis conditions.

- Inservice test procedures, full flow and periodic test results, and test trends to assure the pump remains operationally ready and can fulfill the most limiting design basis requirements.

b. Findings

No findings were identified.

.2.12 Residual Heat Removal Minimum Flow Valve 46A

a. Inspection Scope

The inspectors reviewed the updated safety analysis report, design basis documents, the current system health report, selected drawings, maintenance and test procedures, and condition reports associated with the residual heat removal minimum flow valve, 46A. The inspectors also performed a system walkdown and conducted interviews with system engineering and design personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the inspectors reviewed:

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

- Generic Letter 89-10 calculations, margins associated with the valve qualification, periodic verification analyses in accordance with Generic Letter 96-05, to verify that the valve meets design basis capability requirements and complies with MOV program requirements.

- Inservice test procedures and test trends to assure the valve remains operationally ready.

- Modifications to the component during the life of the plant to assure that the component remains fully operable and meets all programmatic and regulatory requirements.

b. Findings

No findings were identified.

.2.13 Residual Heat Removal Pump A Discharge Check Valve

a. Inspection Scope

The inspectors reviewed the updated safety analysis report, design basis documents, the current system health report, selected drawings, maintenance and test procedures, and condition reports associated with the residual heat removal pump A discharge check valve. The inspectors also performed a system walkdown and conducted interviews with system engineering and design personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the inspectors reviewed:

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

- System hydraulic testing to assure the check valve will provide the required flow and pressure under the most limiting design basis conditions.

- Inservice test procedures and periodic test results, and test trends to assure the check valve remains operationally ready and can fulfill the most limiting design basis requirements.

.2.14 Residual Heat Removal Motor Operated Valves E12-F069 and F094

a. Inspection Scope

The team reviewed the updated safety analysis report, system description, design basis documents, the current system health report, selected drawings and calculations, maintenance and test procedures, and condition reports associated with Unit 1 service water system pump motor. The team also performed walkdowns and conducted interviews with system engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

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

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

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

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

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

b. Findings

No findings were identified.

.3 Results of Detailed Reviews of Permanent Plant Modifications

a. Inspection Scope

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

- The supporting design and licensing basis documentation was updated.

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

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

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

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

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

.3.1 Engineering Change 41543: Install Diode to Prevent Sneak Circuit at Isolator E21A-AT7-0 Pin F

The inspectors reviewed Engineering Change 41543, implemented to prevent a sneak circuit at isolator 21A-AT7-0 Pin F. General Electric Hitachi issued 10 CFR Part 21 Safety Information Communication GE-21-SC-12- 05 - High Level Optical Isolator (HLOI) "Sneak Power Path" Analysis on May 14, 2012, after being notified that during maintenance at a domestic BWR 6 a performance issue was identified relative to operation of a 204B6188AAG001/G002 HLOI and related system circuitry.

General Electric Hitachi determined that the as-designed circuit configuration of a HLOI output pin in some applications may cause a separate standing voltage at the output pin from an independently fused source during specific plant conditions. General Electric Hitachi refers to this as a WIRED-OR condition due to Sneak Power Path (SPP). The WIRED-OR configuration typically works as designed, but under specific conditions (HLOI 125 VDC fuse open and voltage available at the output pin via an independently fused source) General Electric Hitachi has determined that the standing voltage present at the output pin can leak back through the output transistor pair of the affected HLOI circuit. If this occurs, the remaining nine output isolator circuits, if used, may continue to operate normally, even though the power supply fuse to the output isolator is open. This may affect the reliability of the affected components.

EC 41543 allowed installation of a safety-related diode at pin F of E21A-AT07-O to address the vulnerability and that reactor core isolation cooling /residual heat removal division 1 to division 2 isolator out of file or loss of power status light and the reactor core isolation cooling system annunciator may not operate properly during a loss of division 1 power. The installation of these diodes prevents the presence of a standing voltage at the reactor core isolation cooling /residual heat removal instrument and enables the circuits to function as designed. The inspectors did not identify any concerns with the design change package.

.3.2 Engineering Change 34435: Snubber to E-bar Parent Engineering Change

The inspectors reviewed Modification Engineering Change 34435 to reduce the number of snubbers in the plant and replace them with E-bars. The change was driven by the high frequency of maintenance and failure rates of snubbers. The licensee replaced 31 snubbers in the plant with E-bars. Current pipe stress calculations for the affected supports were modified to reflect the new analyses with E-bars. This modification was implemented by 12 child engineering changes and affected piping in numerous plant systems. Inspectors reviewed snubber replacements affecting the residual heat removal and the service water systems. The inspectors did not identify any concerns with the design change package.

.3.3 Engineering Change 64890: Install Backup Fire Water Connections

The inspectors reviewed Engineering Change 64890, implemented to develop the technical and quality requirements for modifying the fire protection system water supply connections on tank, FPW-TK1A to provide a backup method of supplying fire water. Specifically, this change provided engineering instructions for installation of a new discharge manifold off of the east end of the fire pump house that connects with the 12-inch fire protection underground main line, PW-012-001. The discharge manifold received pump discharge and routed it to the fire water ring header. The manifold included connections for accepting discharge from a backup pump with its suction line connected to tank FWP-TK1A and had a 10-inch underground isolation valve. This modification also included the installation of two blind flanges on each end to allow additional configuration options if needed. The inspectors performed a walkdown of the new system and reviewed a sample of the affected equipment and the post modification testing results to verify the installation satisfied the design requirements. The inspectors did not identify any concerns with the design change package.

.3.4 Engineering Change 44960: Upper Containment Pools to Reactor Core Isolation Cooling Cross Tie Piping

The inspectors reviewed Engineering Change 44960, implemented to provide an additional makeup water source to the reactor core isolation cooling pump from the Upper Containment Pools to be utilized during a Beyond Design Basis External Event, in response to the March 2011 accident at the Fukushima Daiichi Nuclear Power Plant in Japan. This additional makeup water source was implemented by installing a permanent flexible mitigation capability (FLEX) crosstie from the upper containment pools to the reactor core isolation cooling pump suction line. The new crosstie is isolated by a locked closed manual gate valve. This system crosstie is limited for use during a Beyond Design Basis External Event, and the overall flexible mitigation capability strategy, and is not intended for normal plant operation, or design basis event. The inspectors reviewed a sample of the design basis information and the implementing procedures associated with this modification. The inspectors did not identify any concerns with the design change package.

b. Findings

Failure to Perform an Adequate Operability Determination for a Condition Identified During an NRC Walkdown

Introduction. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Action,” for the licensee’s failure to perform an adequate operability determination for a condition identified during an NRC walkdown. Specifically, the walkdown identified a junction box with missing cover screws located in the reactor core isolation cooling room. The cover had been secured with duct tape.

Condition Report CR-RBS-2017-04737 included the inadequate operability determination.

Description. The NRC inspector, accompanied by the licensee's system engineer, conducted a walkdown of the reactor core isolation cooling room on June 15, 2017. This activity was related to the NRC's evaluation of Engineering Change 37843. The inspector observed that junction box I*JB8281 was missing two cover screws and that duct tape had been applied to ensure that the cover remained fully closed. There was no indication that this condition had been previously identified. The licensee initiated Condition Report CR-RBS-2017-04737 to document this condition on June 15, 2017.

Condition Report CR-RBS-2017-04737 stated that the junction box was associated with valve E51-AOVF054, reactor core isolation cooling drain pots drain. The condition report included an operability determination that stated, in part, that E51-AOVF054 supports maintaining the reactor core isolation cooling system in a normal standby lineup and that the reactor core isolation cooling turbine and system is required to remain operable to support Technical Specification 3.5.3. The Operability Determination referred to Engineering Change 17980, "Engineering Evaluation to Provide Alternate Repair Methodologies for Degraded Flexible Conduit Jackets," dated October 12, 2009, and Condition Report CR-RBS-2011-6063, dated August 12, 2011, as the basis for determining operability.

The operability determination stated, "Damaged conduit has no effect on equipment function from an electrical or structural standpoint. These further indicate that no credit is taken for the moisture barrier function the conduit provides. The purpose of the conduit is to provide mechanical protection which is unaffected with the cover in place. While the conduit provides protection against moisture intrusion, no credit is taken for this property as the cables themselves are qualified for harsh environmental conditions." The operability determination concluded that the condition was "operable" and that no further functionality review was needed.

The inspection team reviewed the bases for the operability determination, EC-17980 and Condition Report CR-RBS-2011-6063, and determined that these documents only addressed the impact of damaged conduit on electrical cable. They did not address the potential impact of moisture on other electrical components. The team questioned if other electrical components were installed in the junction box and if they had also been evaluated for operability.

In response to the team's concerns, the licensee initiated Condition Report CR-RBS-2017-05084 on June 28, 2017. This condition report stated that junction box I*JB8281 contained a terminal block and that the operability determination associated with Condition Report CR-RBS-2017-04737 was not correct. Condition Report CR-RBS-2017-05084 included an operability determination that addressed the terminal block that was installed in the subject junction box. This operability determination referred to the environmental testing that had been performed for this terminal block design and concluded that it would remain operable in an "unsealed junction box." Therefore, the operability determination concluded that the condition was "operable."

Analysis. The team determined that the failure to perform an adequate operability determination was a performance deficiency. The finding was determined to be more than minor because it was associated with the Mitigating Systems Cornerstone attribute of equipment performance and affected the cornerstone objective of ensuring the availability, reliability, and capability of mitigating systems to respond to initiating events to prevent undesirable consequences. Specifically, the failure to ensure operability of valve E51-AOVF054 and its associated circuits would impact the operability of the reactor core isolation cooling system. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012, Exhibit 2, "Mitigating Systems Screening Questions," the issue screened as having very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non-technical specification equipment; and did not screen as potentially risk-significant due to seismic, flooding, or severe weather. This finding had a cross-cutting aspect in the area of problem identification and resolution associated with resolution because the licensee failed to takes effective corrective actions to address issues in a timely manner commensurate with their safety significance. Specifically, the licensee failed to perform an adequate operability determination for an identified condition [P.3].

Enforcement. The team identified a Green non-cited violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," which states, in part, measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected. Contrary to the above, between June 15, 2017, and June 28, 2017, the licensee failed to assure that a condition adverse to quality was promptly identified and corrected. Specifically, the licensee failed to address the operability of a terminal block installed within an unsealed junction box. In response to this issue the licensee performed an operability determination to ensure that the terminal block would perform its design function in this condition. This finding was entered into the licensee's corrective action program as Condition Report CR-RBS-2017-05084. Because this finding was of very low safety significance and has been entered into the licensee's corrective action program, this violation is being treated as a non-cited violation consistent with Section 2.3.2.a of the NRC Enforcement Policy: NCV 05000458/2017007-02, "Failure to Perform an Adequate Operability Determination for a Condition Identified During an NRC Walkdown."

.4 Results of Detailed Reviews of Operating Experience

.4.1 Inspection of NRC Information Notice 2013-05, "Battery Expected Life and its Potential Impact on Surveillance Requirements"

The team reviewed the licensee's evaluation of Information Notice 2013-05, "Battery Expected Life and its Potential Impact on Surveillance Requirements" to verify that the licensee's Technical Specifications and battery sizing for age is consistent with Institute of Electrical and Electronics Engineers (IEEE) 450, "Maintenance, Testing and Replacement of Large Lead Storage Batteries for Generating Stations" and IEEE 485, "Sizing Large Lead Storage Batteries for Generating Stations" as cited in the Information Notice. The inspectors did not identify any concerns with how the licensee addressed this operating experience.

.4.2 Inspection of NRC Information Notice 87-10, "Potential for Water Hammer During Restart of Residual Heat Removal Pumps"

The team reviewed the licensee's evaluation of NRC Information Notice 87-10, "Potential for Water Hammer During Restart of Residual Heat Removal Pumps" to verify the licensee performed an applicability review and took corrective actions, if appropriate, to address the concerns described in the information notice. This information notice was issued to address the potential of a water hammer in the residual heat removal system during a design basis loss of coolant accident coincident with a loss of offsite power (LOOP). Supplement 1 of the information notice address the increased use of residual heat removal pumps in suppression pool cooling mode due to leaking safety relief valves. The inspectors did not identify any concerns with how the licensee addressed this operating experience.

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

The team reviewed the licensee's evaluation of NRC Information Notice 2009-09, "Improper flow controller settings renders injection systems inoperable and surveillance did not identify," to verify that the licensee has the correct flow controller settings for reactor containment isolation cooling and high pressure core spray injection systems. The inspectors did not identify any concerns with how the licensee addressed this operating experience.

.4.4 Inspection of NRC Information Notice 2006-21, "Operating Experience Regarding Entrainment of Air into Emergency Core Cooling and Containment Spray Systems"

The team reviewed the licensee's evaluation of NRC Information Notice 2006-21, "Operating experience regarding entrainment of air into emergency core cooling and containment spray systems," to verify that the licensee has adequate protection from air entrainment in high pressure core spray during the most limiting injection scenario. The inspectors did not identify any concerns with how the licensee addressed this operating experience.

.5 Results of Reviews for Operator Actions

a. Inspection Scope

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

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

- Initiation of high pressure core spray in manual during a total loss of feedwater and failure of the reactor core isolation cooling system to operate

The inspectors observed a simulator job performance measure where one operator initiates high pressure core spray per Attachment 4 of Procedure OSP-0053, “Emergency and Transient Response Support Procedure,” Revision 25. This activity was observed being performed by two separate operators. Both operators manually started up high pressure core spray immediately and restored reactor water level to the directed level band. This action is necessary to ensure reactor water level does not lower below top of active fuel, which would result in core damage. This activity was satisfactorily performed within the required time.

- <u>Initiation of low pressure emergency core cooling system(s), with the plant in Mode 3 (hot shutdown) and division 1 residual heat removal system is in shutdown cooling with a loss of cooling accident</u>

  The inspectors observed a crew align low pressure emergency core cooling systems that failed to automatically start following a large break loss of cooling accident and a loss of division 1 of residual heat removal, per Section 4.2 of Procedure SOP-0032, “Low Pressure Core Spray,” Revision 24, and Section 4.2 of Procedure SOP-0031, “Residual Heat Removal,” Revision 337, from the simulator main control room. The activity was observed on two separate crews. The first crew immediately recognized the loss of cooling accident condition and the failure for emergency core cooling systems to start automatically. They took manual action to restore reactor water level using low pressure core spray, and manually started low pressure coolant injection pumps B and C and high pressure core spray pump. The second crew also immediately recognized the loss of cooling accident condition and the failure for emergency core cooling systems to start automatically. They took manual action to restore reactor water level using low pressure core spray pump and manually started low pressure coolant Injection pumps B and C, and high pressure core spray pump. However, the second crew was slow to establish a reactor water level of +75 inches required by procedure to establish natural circulation in the core with a both reactor recirculation pumps not operating. Additionally, both crews did not actively pursue leak isolation. The licensee recognized these weakness and was going to perform training analysis to determine appropriate training to remediate these weaknesses observed. This action is necessary to ensure reactor water level does not lower below top of active fuel, which would result in core damage. This activity was completed satisfactorily.

- <u>Recognize and direct the field operators to align fire water to inject into the reactor pressure vessel (RPV) with a station blackout condition</u>

  The inspectors observed a crew operating in the simulator control room and operators in the plant during a station blackout and with a loss of reactor core isolation cooling. The crew had to recognize that they had a total loss of ac power and that reactor core isolation cooling was not operating and immediately ordered field operators to align the fire water system per Attachment 2 of Procedure AOP 50, “Station Blackout,” Revision 57, from the simulator control room and in-the-plant locations. The activity was observed on two separate crews. The first crew dispatched field operators within approximately 7 minutes and obtained injection in the reactor pressure vessel within approximately 39 minutes. The second crew dispatched field operators within approximately 17 minutes and obtained injection in the reactor pressure vessel within approximately 55 minutes. This action is necessary to prevent core damage. This activity was satisfactorily performed by the first crew within the required time of 49 minutes. The second crew exceeded the required time to complete this task by 6 minutes. The licensee entered this failure to perform a time critical action in required time into their corrective action program as Condition Report CR-RBS-2017-05148. The licensee performed remedial training for the second crew followed by re-evaluation prior to them returning to shift in the plant. The licensee operator training group will be working with operations management to evaluate these operator deficiencies and determine actions necessary to ensure all operations personnel are aware of these deficiencies and the importance of correct prioritization of actions during a station blackout. Also, the operations department will be reviewing the station's blackout abnormal procedure to make enhancements and improvements. The licensee entered these enhancements and improvements into their corrective action program under Condition Report CR-RBS-2017-05144.

The inspectors asked the licensee's engineering staff how the time critical action of 49 minutes was determined. The licensee engineering staff determined from a review of their engineering calculation used to determine the time was in error based on assumptions that had changed since it was developed. The new estimated time prior to formal calculation development is approximately 42 minutes. This issue has been entered into the licensee's corrective actions program as Condition Report CR-RBS-2107-05133. The licensee, as an interim action, put out an operator notification informing all operators of this change of the time critical action time for fire water injection, per Attachment 2 of Procedure AOP 50, "Station Blackout," from 49 minutes to 42 minutes. This estimated time will be used until the formal calculation of actual time can be completed in September of 2017 and then the actual time will be revised into the procedure.

- <u>Initiate actions to reduce heat up in the main control room during a station blackout by shedding DC loads</u>

  The inspectors observed main control room time critical job performance measure of an operator simulating shedding DC loads during a station blackout per Attachment 3 of Procedure AOP 50, "Station Blackout," Revision 57. This activity was observed being performed by two separate operators. The first operator simulated performing this task in approximately eight minutes and the second operator simulated performing this task in approximately nine minutes. This action is required to be performed within 30 minutes following a station blackout. This is to ensure heat up of the main control room does not exceed 120°F prior to the first four hours of the station blackout. This activity was satisfactorily performed within the required time.

- <u>Initiate actions to reduce heat up in the main control room during a station blackout by opening control room instrument panel doors</u>

  The inspectors observed main control room actions of an operator simulating opening control room instrument panel doors during a station blackout per Attachment 4 of Procedure AOP 50, "Station Blackout," Revision 57. This activity was observed being performed by three separate operators. The first operator simulated performing this task in approximately 23 minutes, the second operator performed this task in approximately 16.5 minutes, and the third operator performed this task in 10.25 minutes. Due to the first operator performing the task at a slower than expected pace, an additional operator was observed. Due to this task performance by the second and third operators in expected time, it was determined by the inspectors that first operator's time to perform the task was an exception rather than the norm. The licensee entered this issue into their corrective action program as Condition Report CR-RBS-2017-05144. The licensee intends to modify the procedure to improve the timeliness of task performance. This action is required to be performed within 30 minutes following a station blackout. This is to ensure heat up of the main control room does not exceed 120°F prior to the first four hours of the station blackout. This activity was completed satisfactorily.

- <u>Align the station blackout diesel generator to the 125 VDC backup switchgear via the backup battery charger</u>

  The inspectors observed an operator in the plant during a station blackout simulating aligning the station blackout diesel generator to the 125 VDC backup switchgear via the backup battery charger per Sections 4.1 and 5.1 of Procedure SOP 54, "Contingency Equipment Operations," Revision 323, and Section 5.7 of Procedure SOP 49, "125 VDC System (SYS #305)," Revision 37. This activity was observed being performed by two separate operators. The first operator simulated performing this task in approximately 1 hour and 27 minutes and the second operator performed this task in approximately 1 hour and 58 minutes. This action is necessary if ac power restoration to the site will exceed four hours from the start of the station blackout. This alignment will allow the recharging of one of the station's safety-related batteries to ensure continued operation of various equipment, such as reactor core isolation cooling critical to preventing core damage. This activity was completed satisfactorily. Furthermore, the licensee entered issues noted with procedures used to perform this task into their corrective action program as Condition Reports CR-RBS-2107-05145 and CR-RBS-2017-05146.

b. <u>Findings</u>

No findings were identified.

4OA6 Meetings, Including Exit

On June 30, 2017, the inspectors presented the inspection results to Mr. S. Vasquez, Director of Engineering, and other members of the licensee staff. The licensee acknowledged the issues presented. The licensee confirmed that any proprietary information reviewed by the inspectors had been returned or destroyed.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

D. Bourgoin, Engineer
M. Chase, Director, Regulatory and Performance Improvement
A. Coates, Senior Engineer, Regulatory Assurance
R. Conner, Manager, Nuclear Oversight
K. Crissman, Senior Manager, Production
E. Deweese, Supervisor, Engineering
M. Feltner, Assistant Manager, Operations
J. Henderson, Manager, Systems and Components Engineering
K. Huffstatler, Senior Licensing Specialist
B. Kelley, Talisman International
P. Lucky, Manager, Performance Improvement
J. O'Conner, Senior Manager, Maintenance
J. Rogers, Supervisor, Engineering
D. Sandlin, Manager, Design and Program Engineering
K. Stupak, Manager, Training
S. Vasquez, Director, Engineering

NRC Personnel

R. Deese, Senior Risk Analyst
J. Sowa, Senior Resident Inspector
B. Parks, Resident Inspector

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000458/2017007-01	NCV	Failure to Evaluate the Extent of Condition for a Degraded 4.16KV Magne Blast Safety-Related Circuit Breaker (Section 1R21.2.4)
05000458/2017007-02	NCV	Failure to Perform an Adequate Operability Determination for a Condition Identified During an NRC Walkdown (Section 1R21.3.4.b)

LIST OF DOCUMENTS REVIEWED

Calculations

Number	Title	Revision
G13.18.2.3*162	GL 89-10 Design Basis Review for E12-MOVF048A/B	5
G13.18.2.3*316	RBS NRC GL 96-05 MOV Periodic Static Test Frequency	7
PX-888	LPCI Pump Discharge Line Water Hammer Analysis with Trapped Air for Mode B	1
G13.18.2.7*05	LPCI Injection Piping Evaluation with Trapped Air Due to Incorrect High Point Venting	0
G13.18.6.1-E51*015	Instrument Loop Uncertainty/Setpoint Determination RCIC Steam Line Flow, Isolation Timers E51A-K65, E51A-K85	0
G13.18.14.1-038	Determination of Acceptable Time Delay for RCIC/RHR High Steam Flow Isolation	0
ES-162	Mass and Energy Release Due to the High Energy Line Breaks in Auxiliary Building and in Steam Tunnel	1
ES-167	Pressure and Temperature Transients in Auxiliary Building and Steam Tunnel Due to the High Energy Line Breaks	3
PX-873	LPCI Pump Discharge Lines Water Hammer Analysis with Trapped Air	0
G13.18.14.1-037	Residual Heat Removal System Heat Exchanger Performance Guidance	0
G13.18.2.7*108	Auxiliary Building GOTHIC Model for High Energy Line Break	1
0247.521-207-018	Nozzle Type Relief Valve	300
G13.18.12.4-033	Minimum Temperature During Station Blackout: Battery Rooms A, B, and C	0
G13.18.3.6* 021	DC System Analysis, Methodology & Scenario Development	1
E-143	Standby Battery “ENB*BAT01A” Duty Cycle, Current Profile and Size Verification	11
G13.18.14.1-038	Determination of Acceptable Time Delay for RCIC/RHR High Steam Flow Isolation	0
G13.18.3.6*05	Coordination Study of Appendix R and Class 1E Low Voltage Protective devices	2
G13.18.3.6*08	Class 1E 125 VDC Systems Overvoltage Study	1
PRA-RB-01-002S14	RBS PRA Level 1 Success Criteria	0

Calculations

Number	Title	Revision
G13.18.2.6*034	Determine Number of SRV Actuations from LSV Air Receiver	2
G13.18.6.3-012	Drift Study for Model NGV13B Undervoltage Relays	0
G13.18.6.2-ENS*004	Loop Uncertainty Determination for Div. III Loss of Voltage Relay GE Model NGV Under-Voltage Relay	2
PN-268	RHR System Pumps TDH And NPSH Except LPCI (Mode A-2) Operation	5
G13.18.2.3*166	G.L. 89-10 Design Basis Review for E-12- MOV F064 A/B	4
F064A-ST-006	MOV Test Report E12	0

Drawings

Number	Title	Revision
PID-27-07A	System 204 – Residual Heat Removal – LPCI	38
PID-27-07B	System 204 – Residual Heat Removal – LPCI	42
GE-828E534AA, Sheet 008	Elementary Diagram - Residual Heat Removal System	28
ESK-06RHS13	Elementary Diagram - Residual Heat Removal System	9
PID-27-04A	System 203 High Pressure Core Spray System	26
PID-27-07C	System 204 Residual Heat Removal LPCI	29
PID-09-11B	System 130 Service Water Cooling	8
PID-09-11A	System 130 Service Water Cooling	13
PID-09-10H	System 118 Service Water Normal	28
PID-09-10G	SWP Corrosion Coupon And Monitoring Rack System 118	3
PID-09-10F	System 118 Service Water Normal	33
PID-09-10E	System 256 Service Water Standby	24
PID-09-10D	System 118 Service Water Normal	36
PID-09-10C	System 118 Service Water Normal	26

Drawings

Number	Title	Revision
PID-09-10B	System 118 Service Water Normal	47
PID-09-10A	System 118 Service Water Normal	33
TLD-ICS-037, Sht.1	Test Loop Diagram RCIC Turbine Steam Supply Differential Pressure	0
TLD-ICS-037, Sht. 2	Test Loop Diagram RCIC Turbine Steam Supply Differential Pressure	0
0244.54-14-000-009	Schematic 30 kVA Isolimiter	0
0244.54-14-000-004	Schematic 20kVA Inverter	0
0244.54-14-000-008	Schematic 20kVA Rectifier	0
0244.54-14-000-007	Schematic 20kVA Static Switch	A
0244.700-041-148	System Schematic Exciter Regulation	B
0244.700-041-141	Interconnection Diagram Standby Diesel Generator	301
0244.700-041-122	Schematic Standby Diesel Generator System Excitation Panel	301
EE-001ZH	125 VDC One Line Diagram Standby Bus B	23
EE-001ZJ	125 VDC One Line Diagram Normal & Standby Backup Charger System	20
EE-001SA	480V One Line Diagram, 1E22-S002, Control Building	12
EE-001M	4.16KV One Line Diagram Standby Bus E22-S004	9
ESK-05SWP06, Sh. 1	Elementary Diagram – 4.16KV SWGR, Standby Service Water Pump 2C	27
EE-001L	4.16KV One line Diagram, Standby ENS-SWGR 1B	18
944E114, Sh. 1	High Pressure Core Spray Diesel Generator Protection Relay VB Panel H22-P028	6
944E114, Sh. 2	High Pressure Core Spray Diesel Generator Protection Relay VB Panel H22-P028	3
EE-001AC	Start Up Electrical Distribution Chart	54
ESK-05NNS06, Sh.1	Elementary Diagram 4.16KV Switchgear Bus 1B Preferred Supply ACB	20

Drawings

Number	Title	Revision
EE-001TG	480V One Line Diagram 1EHS*MCC2F Auxiliary Building	14
GE-828E537AA, Sh. 2	Elementary Diagram, High Pressure Core Spray Power Supply System	33
GE-828E537AA, Sh. 9A	Elementary Diagram, High Pressure Core Spray Power Supply System	26
GE-828E537AA, Sh. 8	Elementary Diagram, High Pressure Core Spray Power Supply System	28
GE-828E537AA, Sh. 9	Elementary Diagram, Residual Heat Removal System	28
GE-914E551, Sh. 1	Connection Diagram, Reactor Core Cooling BB Panel H13-P601	16
GE-914E551, Sh. 18	Connection Diagram, Reactor Core Cooling BB Panel H13-O601 A18	14
GE-914E551, Sh. 19	Connection Diagram, Reactor Core Cooling BB Panel H13-P601 A18	14
PID -27-07A	Residual Heat Removal - LPCI, System 204	38

Procedures

Number	Title	Revision/ Date
SPX-HX-RBS-003	Performance Monitoring Program for RHR Heat Exchangers E12-EB001A and E12-EB001C	2
STP-309-0601	Division I ECCS Test	51
STP-204-6301	Division I LPCI (RHR) Pump and Valve Operability Test	27
STP-200-0605	Remote Shutdown System Control Circuit Operability Test (Switches S1, S6, S7, S8, S9, And S12)	308
T1623	Clean Inspect Strainer	April 5, 2017
T620	Major Inspection of Actuator	October 7, 2014
STP-204-6601	Division I RHR Position Indication Verification Test	302
EN-DC-136	Temporary Modifications	13
ARP-RMS-DSPL230	Alarm Response for RHR HX Service Water High Radiation	9
ADM-0037	Equipment Identification and labeling	17

Procedures

Number	Title	Revision/ Date
STP-305-1701	1ENB*BAT01B Performance Discharge Test	25
STP-305-1607	1ENB*BAT01B Service Discharge Test	20
AOP-0050	Station Blackout Procedure	57
CMP-EM-000-1006	4.16KV ABB 5HK Breakers Clean, Inspect, Lubricate and Test	1
T429	ABB 5HK Clean/Inspect	October 30, 2012
CMP-EM-302-1001	ABB 4.16KV Switchgear- Clean and Inspect	0
EN-LI-102	Corrective Action Program	24
SOP-0049	125 VDC System	
OSP-0022	Operations Administrative Guidelines	103
OSP-0053	Emergency and Transient Response Support Procedure	25
SOP-0031	Residual Heat Removal (SYS #204)	337
AOP-0065	Extended Loss of AC Power (ELAP)	1
SOP-0054	Contingency Equipment Operations	323
EOP-0001	Emergency Operating Procedure - RPV CONTROL	27
RBS-FSG-003	Alternate Reactor Vessel Cooling	1
AOP-0003	Automatic Isolations	37
EOP-0004	Emergency Operating Procedure – Contingencies	15
EOP-0002	Emergency Operating Procedure - Primary Containment Control	16
EOP-0003	Emergency Operating Procedure - Secondary Containment and Radioactive Release Control	17
AOP-0001	Reactor Scram	36
AOP-0004	Loss of Offsite Power	55
SOP-0035	Reactor Core Isolation Cooling System (SYS #209)	53

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Number	Title	Revision/ Date
SOP-0030	High Pressure Core Spray (SYS. #203)	31
OSP-0066	Extensive Damage Mitigation Procedure	28
RBS-FSG-003	Alternate Reactor Vessel Cooling	1
SOP-0032	Low Pressure Core Spray (SYS #205)	24
EN-SA-G-001	Identification and Documentation of Time Critical Actions	1
STP-302-1601	ENS-SWG1B Loss of Voltage Channel Calibration and Logic System Functional Test	21
STP-302-1604	High Pressure Core Spray Loss of Voltage Channel Calibration and Logic System Functional Test	22
EN-DC-324	Preventive Maintenance Program	17
ARP-863-74	P863-74 Alarm Response	25
STP-302-0603	Division III Off Site AC Sources Transfer Test	4
EN-LI-121	Trending and Performance Review Process	22
EN-DC-310	Predictive Maintenance Program	8
STP-204-6301	Division I RHR/LPCI Pump and Valve Operability Test	27
STP-204-6303	Division I RHR Quarterly Valve Operability Test	21
STP-204-6601	Division I RHR Position Indication Verification Test	302

Design Change Packages

Number	Title	Revision
EC-34435	Snubber To E-Bar Parent EC	0
EC-37843	Add Time Delay to E31-PDTN084A and B To Prevent RCIC Steam Line Isolation from Non-Line Break Pressure Transients	0
EC-34473	Snubber to E-bar RHS-3149	0
EC-34472	Snubber to E-bar RHS-3148	0
EC-34462	Snubber to E-bar RHS-2367	0

Design Change Packages

Number	Title	Revision
EC-34461	Snubber to E-bar RHS-2327	0
EC-34460	Snubber to E-bar RHS-2312	0
EC-34454	Snubber to E-bar RHS-2037	0
EC-41542	Install Diodes to Prevent Sneak Circuit at Isolator E12-AT08-O Pin B & F in Panel H13-P601	0
EC-41543	Install Diodes to Prevent Sneak Circuit at Isolator E21-AT07-O Pin F	0
EC-41528	Revise Optical Isolator Schematics and Wiring Diagram to Eliminate Sneak Power Path”" REF. CR-RBS-2012-03632 CA 02, 10CFR PART21 SAFETY INFORMATION GE SC 12-05 AND CR-RBS-2012-03632 CA 06	0
EC - 44960	Upper Containment Pools to Reactor Core Isolation Cooling Cross Tie Piping	0
EC - 64890	Install Backup Fire Water Connections	0

Vendor Documents

Number	Title	Revision Date
23A5462	RHR Heat Exchanger Calculated Performance	1
VTD-A391-0100	Anchor/Darling Maintenance Manual for Flexible Wedge Type Gate Valves	0
VTD-A391-0105	Anchor/Darling Instruction Manual for Carbon Steel Motor Operated Gate and Globe Valves	0
VTD-B580-0117	Byron Jackson Pump Division Vertical High Pressure Core Spray Pump	0
G185-0100	GNB Industrial Power Installation & Operating Instructions for Classic Flooded Lead-Acid Batteries	3
VTD-P319-0100	Instruction Manual for Three Phase Thyristor Controlled Model 3SD-130-300	2
VTD-B455-0100	ABB Maintenance Instruction for Medium Voltage Power Circuit Breakers Type 5HK	December 21, 1994
VTD-B455-0102	ABB HK Breaker Electrical Operating Sequence DC Closing	November 13, 1997
VTD-P076-0104	Manual for Automatic Voltage Regulator	0

Vendor Documents

Number	Title	Revision Date
VTD E355-0002	Exide Light guard Installation and Operating Instruction (PUB # 9140050206)	0
VTD-S322-0100	Southern Transformer Guide for Installation and Maintenance of Dry Type Transformers	0

Design Basis Documents

Number	Title	Revision
SDC-204	Residual Heat Removal System	4
SDC-205	Low Pressure Core Spray System	3
SDC-305	Safety-Related 125 VDC System Design Criteria	2
SDC-309	Standby Diesel Generator Design Criteria	3
SDC-209	Reactor Core Isolation Cooling System	5
SDC-302	4.16KV Electrical Distribution System Design Criteria System Number 203-302	1

Other

Number	Title	Revision Date
N/A	RHR Strainer Inspection Photos	March 16, 2015
G9.33.3	IN 87-10, SUPPL 1: Potential for Water Hammer During Restart of Residual Heat Removal Pumps	June 19, 1997
EEAR 87-R0073	Response to IE Information Notice 87-10	0
M/C 95-013	Evaluation of Gate Valves Susceptible to Thermal Binding and Bonnet Pressurization	January 31, 1996
EC-17980	Engineering Evaluation to Provide Alternate Repair Methodologies for Degraded Conduit Jacket	0
EC-56521	Gag Closed RHR Division II Heat Exchanger Cooling Water Side Thermal Relief Valve	0
N/A	High Pressure Core Spray Diesel Generator Division III, Diesel Generator Building Ventilation System Design Criteria	3
N/A	High Pressure Core Spray System Design Criteria	5
1.ILCSH.019	Loop calibration Report for high pressure core spray pump discharge	2

Other

Number	Title	Revision Date
6244.521-078-005C	Battery Qualification Report	2
6244.523-072-001A	Qualification of Class 1E Battery Chargers	3
LAR 2009-05	24-Month Operating Cycles	0
LAR 2003-23	Removal of MODE Restrictions for Surveillance Testing of the Division III Battery	October 21, 2003
A-16973	NRC Commitment- RG1.32, RG1.29 and IEEE-450	August 10, 2009
6244.514-000-008A	Inverter Final Test Report-ENB-INV01A1	300
6244.521-078-005C	Battery Qualification Report	August 19, 1996
6244.523-072-001A	Qualification of Class 1E Battery Chargers	June 16, 1978
244.523	Specification for Standby Static Battery Chargers	1
6244.700-041-030E	Qualification Report of Generators and Accessories for Standby Diesel Generator System	May 10, 1985
IEEE 946-1992	Recommended Practice for the Design of DC Auxiliary Power Systems for Generating Stations	1992
IEEE 485-1983	Recommended Practice for Sizing Large Lead Storage Batteries for Generating Stations and Substations	1983
IEEE 450-1975	Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations	1975
EN-2012-001	EC-37843 10CFR50.59 Evaluation	1
RFPT-HLO-0541	AOP-0050 Station Blackout	3
RSMSOpS.0676	Loss of All RPV Injection ATWS Requiring RPV Flooding Using High Pressure Core Spray to Restore RPV Inventory	0
RLP-STM-0203-LO	High Pressure Core Spray	0
RGAT-NLOR-1611-PW	Plant Walk Down Cycle 16-11	0
RLP-STM-0305-LO	DC Distribution	0
RGAT-NLOR-1701-305	DC Distribution Plant Walk Down	0

Other

Number	Title	Revision Date
RGAT-NLOR-16-9-0250	Fire System	0
R5M5.OPS.0434	Station Blackout with High Pressure Core Spray Diesel Generator	7
RJPM-OPS-700-06	Shedding DC Loads During Station Blackout	1
RSMS-OPS-0434	Station Blackout with High Pressure Core Spray Diesel Generator	7
RSMS-OPS-0420	Seismic Event, Station Blackout and Restoration	14
5616 D	Transformer Arrangement 225kVA 3ph 60hz 4.16KV +-21/2 95% 277/480V	D
CF8506120022	Seismic Qualification Reevaluation Class 1E Equipment-High Pressure Core Spray Transformer E22-S003	2
VPF-3831-014-2	High Pressure Core Spray Transformer Storage & Install Instruction	0
21A9302	Purchase Specification for Electrical Transformer, High Pressure Core Spray System	K5
6242.533-265-003B	IEEE 323 Qualification plan for 1500kVA Standby Transformers	0
VTM-W120-0004	Engineering Data for MG-6, J10, J12, J13 Relays	November 29, 1999

Condition Reports

CR-HQN-2009-00738	CR-RBS-2014-03684	CR-RBS-2016-02060
CR-HQN-2017-00655	CR-RBS-2014-05472	CR-RBS-2016-04706
CR-RBS-2002-01722	CR-RBS-2014-06163	CR-RBS-2016-07338
CR-RBS-2003-02437	CR-RBS-2014-06189	CR-RBS-2016-08136
CR-RBS-2006-04478	CR-RBS-2015-00727	CR-RBS-2017-00669
CR-RBS-2009-02975	CR-RBS-2015-04891	CR-RBS-2017-01182
CR-RBS-2009-03519	CR-RBS-2015-05115	CR-RBS-2017-02025
CR-RBS-2011-06063	CR-RBS-2015-06372	CR-RBS-2017-02109
CR-RBS-2011-07713	CR-RBS-2015-07040	CR-RBS-2017-02341
CR-RBS-2012-04607	CR-RBS-2016-00742	

Condition Reports Generated During this Inspection

CR-RBS-2017-04672	CR-RBS-2017-04726	CR-RBS-2017-05084	CR-RBS-2017-05148
CR-RBS-2017-04673	CR-RBS-2017-04737	CR-RBS-2017-05144	CR-RBS-2017-05153
CR-RBS-2017-04677	CR-RBS-2017-05050	CR-RBS-2017-05145	CR-RBS-2017-05157
CR-RBS-2017-04700	CR-RBS-2017-05078	CR-RBS-2017-05146	CR-RBS-2017-05166
CR-RBS-2017-04720			

Work Orders

00109894	00457244	52349264	52565264	52641956
00198637	50373789	52413106	52581972	52648317
00234831	51008326	52450952	52594292	52648323
00234832	51008327	52476641	52599326	52666433
00343515	51037462	52500398	52609848	52682857
00388743	51523930	52500757	52622390	52697633
00394231	52249845	52508802	52632012	52701399
00396046	52322970	52513628	52633872	52716218
00418680	52330687	52525609	52637578	52730928
00427025	52331760	52546754	52638089	52731811
00429228	52345594	52562760	52639249	52751286
00441801				

System Health Reports

Number	Title	Revision/Date
RBS 204	Residual Heat Removal - LPCI	Q1-2017
RBS 309	Standby Emergency Diesel Generators System Health Report	Q1-2017
RBS 305	125 VDC Electric Distribution and Battery Charger System Health Report	Q4-2016