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

October 29, 2015

Mr. Eric W. Olson, Site Vice President
Entergy Operations, Inc.
River Bend Station
5485 U.S. Highway 61N
St. Francisville, LA 70775

SUBJECT: RIVER BEND STATION – NRC EVALUATIONS OF CHANGES, TESTS, AND EXPERIMENTS AND PERMANENT PLANT MODIFICATIONS BASELINE INSPECTION REPORT 05000458/2015007

Dear Mr. Olson:

On October 8, 2015, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your River Bend Station. On October 8, 2015, the NRC inspectors discussed the results of this inspection with you and other members of your staff. Inspectors documented the results of this inspection in the enclosed inspection report.

The NRC inspectors documented two findings of very low safety significance (Green) in this report. Both of these findings involved violations of NRC requirements, and one was determined to also be a Severity Level IV violation under the traditional enforcement process. The NRC is treating these violations as non-cited violations (NCVs) consistent with Section 2.3.2.a. of the Enforcement Policy.

Further, inspectors documented one licensee-identified violation, with two examples, which is determined to be of very low safety significance. The NRC is treating this violation as a non-cited violation (NCV) consistent with Section 2.3.2.a of the Enforcement Policy.

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

If you disagree with a cross-cutting aspect assignment, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region IV; and the NRC resident inspector at River Bend Station.

In accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) 2.390, “Public Inspections, Exemptions, Requests for Withholding,” a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC’s Public Document Room or from the Publicly Available Records (PARS) component of the NRC

Agencywide Documents Access and Management System (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

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

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

Enclosure:
Inspection Report 05000458/2015-007
w/Attachment: Supplemental Information

cc w/encl: Electronic Distribution

U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Docket:	05000458
License:	NPF-47
Report:	05000458/2015007
Licensee:	Entergy Operations, Inc.
Facility:	River Bend Station
Location:	5485 U.S. Highway 61N St. Francisville, LA 70775
Dates:	September 21 through October 8, 2015
Inspectors:	M. Williams, Reactor Inspector J. Watkins, Reactor Inspector L. Brandt, Reactor Inspector
Approved By:	Thomas R. Farnholtz Branch Chief Division of Reactor Safety

SUMMARY

IR 05000458/2015007; 09/21/2015 – 10/8/2015; River Bend Station; Evaluations of Changes, Tests, and Experiments and Permanent Plant Modifications.

This report covers a two-week announced baseline inspection on evaluations of changes, tests, and experiments and permanent plant modifications. The inspection was conducted by Region IV based engineering inspectors. Two findings of very low safety significance (Green) are documented in this report. Both of these findings involved violations of NRC requirements, and one was also determined to be a Severity Level IV violation under the traditional enforcement process. Additionally, a licensee-identified violation, with two examples, of very low safety significance is documented in this report. The significance of most findings is indicated by their color (i.e., greater than Green, or Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Cross-cutting aspects were determined using IMC 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 February 4, 2015. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 5, dated February 2014.

A. NRC-Identified Findings and Self-Revealed Findings

Cornerstone: Mitigating Systems

- Green. The team identified a Green, non-cited violation of Technical Specification 5.4.1, which states in part, "procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, shall be established, implemented, and maintained for combating emergencies, including tornados." Specifically, prior to September 22, 2015, the licensee failed to establish adequate procedures to ensure loose debris (drift eliminators/grating that had come loose from the cooling towers) was secured. In response to this issue, the licensee inspected the area and prepared a work order to remove all loose drift eliminators. This finding was entered into the licensee's corrective action program as Condition Report CR-RBS-2015-06891.

  The team determined that the failure to maintain adequate procedures to ensure compliance with technical specifications and Regulatory Guide 1.33 was a performance deficiency. This finding was more than minor because it was associated with the protection against external factors attribute of the Mitigating Systems Cornerstone, and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (severe weather). Specifically, the licensee failed to establish adequate procedures to ensure protection of the switchyard against external factors such as the loose drift eliminators on the cooling tower as a potential missile hazard during high wind events. 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 team determined that the finding was determined to have very low safety significance (Green) since the systems, structures, and components maintained their operability and functionality. The finding was determined to have a

cross-cutting aspect in the area of problem identification and resolution, identification, because the licensee failed to implement a corrective action program with a low threshold for identifying issues. Individuals failed to identify issues completely, accurately, and in a timely manner in accordance with the program (P.1). (Section 1R17.1.b.1)

- Severity Level IV/Green. The team identified a Severity Level IV, Green, non-cited violation of 10 CFR 50.59, "Changes, Tests, and Experiments," Section (c)(2) which states, in part, "A licensee shall obtain a license amendment pursuant to Section 50.90 prior to implementing a proposed change, test, or experiment if the change, test, or experiment would result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the final safety analysis report (as updated)." Specifically, prior to October 8, 2015, the licensee failed to correctly evaluate that a spurious reactor core isolation cooling actuation injecting into the feedwater line resulted in a more than minimal increase in the frequency of occurrence of the loss of feedwater heating accident previously evaluated in the updated final safety analysis report. In response to this issue, the licensee initiated a condition report to document completion of a new evaluation under current regulatory guidelines. This finding was entered into the licensee's corrective action progam as Condition Report CR-RBS-2015-7259.

  The team determined that the failure to perform an adequate evaluation of a design change was a performance deficiency. This finding was also evaluated using traditional enforcement because it had the potential to impact the NRC's ability to perform its regulatory function. This finding was more than minor because it was associated with the design control attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences, and there was a reasonable likelihood that the change would have required NRC review and approval prior to implementation. Specifically, the licensee failed to correctly evaluate that a spurious reactor core isolation cooling actuation injecting into the feedwater line resulted in a more than minimal increase in the frequency of occurrence of the loss of feedwater heating accident previously evaluated in the updated final safety analysis report. 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 where the mitigating structure, system, or component maintained its operability or functionality. Since the violation is associated with a Green reactor oversight process violation, the traditional enforcement violation was determined to be a Severity Level IV violation, consistent with the example in paragraph 6.1.d(2) of the NRC Enforcement Policy. There is no cross-cutting aspect assigned to this performance deficiency because the performance deficiency is not indicative of current performance and also because cross-cutting aspects are not assigned to traditional enforcement violations. (Section 1R17.1.b.2)

B. Licensee-Identified Violations

One violation with two examples of very low safety significance that were identified by the licensee have been reviewed by the inspectors. Corrective actions taken or planned

by the licensee have been entered into the licensee's corrective action program. This violation and associated corrective action tracking numbers are listed in Section 4OA7 of this report.

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, and Emergency Preparedness

1R17 Evaluations of Changes, Tests, and Experiments and Permanent Plant Modifications (71111.17T)

.1 Evaluations of Changes, Tests, and Experiments

a. Inspection Scope

The inspectors reviewed eight evaluations performed pursuant to Title 10, Code of Federal Regulations (CFR), Part 50, Section 59, to determine whether the evaluations were adequate and that prior NRC approval was obtained as appropriate. The inspectors also reviewed 19 screenings, where licensee personnel had determined that a 10 CFR 50.59 evaluation was not necessary. The inspectors reviewed these documents to determine if:

- the changes, tests, and experiments performed were evaluated in accordance with 10 CFR 50.59 and that sufficient documentation existed to confirm that a license amendment was not required;
- the safety issue requiring the change, tests and experiment was resolved
- the licensee conclusions for evaluations of changes, tests, and experiments were correct and consistent with 10 CFR 50.59; and
- the design and licensing basis documentation was updated to reflect the change.

The inspectors used, in part, Nuclear Energy Institute (NEI) 96-07, “Guidelines for 10 CFR 50.59 Implementation,” Revision 1, to determine acceptability of the completed evaluations and screenings. The NEI document was endorsed by the NRC in Regulatory Guide 1.187, “Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments,” dated November 2000. The list of evaluations, screenings and/or applicability determinations reviewed by the inspectors is included as an attachment to this report.

This inspection constituted 8 samples of evaluations and 19 samples of screenings and/or applicability determinations as defined in IP 71111.17-04.

b. Findings

.1 Failure to Establish Adequate Procedures for Severe Weather Operations

Introduction. The team identified a Green, non-cited violation of Technical Specification 5.4.1, for the licensee's failure to maintain adequate procedures for severe weather operations. Specifically, prior to September 22, 2015, the licensee failed to establish adequate procedures to be used during severe weather preparations to ensure loose debris (drift eliminators/grating that had come loose from the cooling towers) was secured.

Description. The team reviewed Procedure AOP-0029, "Severe Weather Operation", for instructions before, during, and after hurricanes, tornadoes, and severe thunderstorms. Under the hurricane checklist, Attachment 1, step 1.15 directs the station to remove all loose material in the protected area and within line of sight of the Reactor Building to safe areas, while step 1.31 directs plant personnel to perform a walkdown of the Owner Controlled Area and identify items that could generate missiles. Attachment 2, step 1.14 instructs the licensee to initiate a condition report if unsecured jobsite material in the protected area and within line of sight of the Reactor Building or Transformers are noted.

During a walkdown of the cooling tower area, the inspectors observed multiple drift eliminators around the uppermost ring of all four cooling towers that had become detached from the structure and were lying loose on the edges. With the switchyard adjacent to the cooling towers, and within the line of sight, the inspector questioned if the loose drift eliminators could become potential missile projectiles in a high wind event and threaten electrical components in the switchyard.

Analysis. The team determined that the failure to maintain adequate procedures to ensure compliance with technical specifications and Regulatory Guide 1.33 was a performance deficiency. This finding was more than minor because it was associated with the protection against external factors attribute of the Mitigating Systems Cornerstone, and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (severe weather). Specifically, the licensee failed to establish adequate procedures to ensure protection of the switchyard against external factors such as the loose drift eliminators on the cooling tower as a potential missile hazard during high wind events. 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 team determined that the finding was determined to have very low safety significance (Green) since the systems, structures, and components maintained their operability and functionality. The finding was determined to have a cross-cutting aspect in the area of problem identification and resolution, identification, because the licensee failed to implement a corrective action program with a low threshold for identifying issues. Individuals failed to identify issues completely, accurately, and in a timely manner in accordance with the program (P.1).

Enforcement. The team identified a Green, non-cited violation of Technical Specification 4.5.1, which states, in part, "procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, shall be established, implemented, and maintained"

for combating emergencies, including tornados. Contrary to the above, prior to September 22, 2015, the licensee failed to establish adequate procedures for combating emergencies. Specifically, the licensee failed to establish adequate procedures to ensure loose debris (drift eliminators/grating that had come loose from the cooling towers) was secured. In response to this issue, the licensee inspected the area and prepared a work order to remove all loose drift eliminators. This finding was entered into the licensee's corrective action program as Condition Report CR-RBS-2015-06891. Because this finding is of very low safety significance and has been entered into the licensee's corrective action program, this violation is being treated as a non-cited violation consistent with Section 2.3.2.a of the NRC Enforcement Policy:
NCV 5000458/2015007-001, "Failure to Establish Adequate Procedures for Severe Weather Operations."

.2 Failure to Obtain Prior NRC Approval for a Change in Reactor Core Isolation Cooling Injection Point

Introduction. The team identified a Severity Level IV, Green, non-cited violation of 10 CFR 50.59, "Changes, Tests, and Experiments," for the licensee's failure to perform an adequate evaluation of a design change. Specifically, the licensee's modification that changed the reactor core isolation cooling injection point from the head spray nozzle to the feedwater line resulted in a more than minimal increase in the frequency of occurrence of an accident previously evaluated in the updated final safety analysis report.

Description. The licensee is permitted to make changes to the facility as described in the Updated Final Safety Analysis Report without prior NRC approval, provided that these changes do not result in a more than minimal increase in the frequency of occurrence of an accident previously evaluated in the final safety analysis report. Regulatory Guide 1.187, "Guidance for Implementation of 10 CFR 50.59, 'Changes, Tests, and Experiments'," states that the methods described in Nuclear Energy Institute NEI 96-07, "Guidelines for 10 CFR 50.59 Evaluations," Revision 1, are acceptable to the NRC staff for complying with the provisions of 10 CFR 50.59. NEI 96 07 Section 4.3.1 states that licensees can make changes to the facility if the change does not more than minimally increase the frequency of occurrence of an accident. If the increase of the pre-change accident or transient frequency exceeds 10 percent, or the resultant frequency of occurrence exceeds 1E-6 or applicable plant-specific threshold, then the change is considered to involve more than a minimal increase in the frequency of occurrence of an accident, and prior NRC approval is required. The team concluded that the modification, rerouting the reactor core isolation cooling injection point from the head spray nozzle to the feedwater line, resulted in a change that more than minimally increased the frequency of occurrence of the loss of feedwater heating accident.

The head spray function for the reactor core isolation cooling system was removed in Refueling Outage RF-8 via maintenance request MR 96-0069 when the licensee moved the reactor core isolation cooling system injection point from the head spray nozzle to the feedwater line. License Amendment Request LAR 98-01 dated March 3, 1998, was initiated to update the affected licensing basis documents at the site, but no License Amendment Request was submitted to the NRC. Section 3.9.5.1.1.12B was erroneously omitted in LAR 98-01 so the licensee initiated licensing basis document

change request LBDCR 03.09B-024, dated June 10, 2015, to revise Updated Safety Analysis Report Section 3.9.5.1.1.12B to remove references to the head spray function. The team reviewed LBDCR 03.09B-024.

The loss of feedwater heating event is an accident evaluated in the Updated Final Safety Analysis Report that is classified as an incident of moderate frequency. A spurious reactor core isolation cooling actuation event is also evaluated in the Updated Final Safety Analysis Report as a separate accident. The NRC Standard Review Plan classifies the inadvertent operation of the reactor core isolation cooling system as an incident of moderate frequency. NEI 96-07 Revision 1 defines incidents of moderate frequency as any one incident expected per plant during a calendar year. The licensee's Abnormal Operating Procedure AOP-007 classifies any event that causes a decrease in feedwater temperature of 3 percent as a loss of feedwater heating event. In their 50.59 evaluation for the modification to reroute the reactor core isolation cooling injection point, the licensee evaluated a spurious injection of the reactor core isolation cooling system into the feedwater system with the condensate storage tank temperature at 40°F. Based on this evaluation, the licensee concluded that a spurious reactor core isolation cooling injection would be classified as a loss of feedwater heating event.

The licensee did not assign the proper frequency of a spurious reactor core isolation cooling actuation when evaluating the new frequency of a loss of feedwater heating event. The licensee concluded that since a spurious reactor core isolation cooling injection with the reactor at power had never occurred at the site, the probability of an occurrence was sufficiently low that it did not result in an increase in the probability of a loss of feedwater event. This assumption led the licensee to conclude that the effects associated with the modification were inconsequential and, therefore, did not constitute an Unreviewed Safety Question. Since the change was not considered an Unreviewed Safety Question, the licensee completed the modification without submitting a License Amendment Request to the NRC. Had the licensee used the proper frequency (once per year) for the inadvertent operation of the reactor core isolation cooling, this would have led them to conclude that the effects associated with the modification were not inconsequential and, therefore, did constitute an Unreviewed Safety Question. At the time of that evaluation, any Unreviewed Safety Question required review by the NRC via a License Amendment Request prior to implementing a change to the facility.

Currently, NEI 96-07 Section 4.3.1 states that licensees can make changes to the facility if the change does not more than minimally increase the frequency of occurrence of an accident. If the increase of the pre-change accident or transient frequency exceeds 10 percent, or the resultant frequency of occurrence exceeds 1E-6 or applicable plant-specific threshold, then the change is considered to involve more than a minimal increase in the frequency of occurrence of an accident, and prior NRC approval is required. The licensee still classifies a spurious initiation of the reactor core isolation cooling system as a loss of feedwater heating event per AOP-007, and both events are still defined as incidents of moderate frequency (once per year). Adding the spurious initiation of the reactor core isolation cooling system frequency (once per year) to the loss of feedwater heating event frequency (once per year) essentially doubles the original frequency of the loss of feedwater heating event. Therefore, this reroute of the reactor core isolation cooling system from the head spray nozzle to the feedwater line would meet the criteria to require NRC approval because it more than minimally increased the frequency of occurrence of the loss of feedwater heating accident. The

licensee documented this issue in the corrective action program as Condition Report CR-RBS-2015-7259.

Analysis. The team determined that the failure to perform an adequate evaluation of a design change was a performance deficiency. This finding was also evaluated using traditional enforcement because it had the potential to impact the NRC's ability to perform its regulatory function. This finding was more than minor because it was associated with the design control attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences, and there was a reasonable likelihood that the change would have required NRC review and approval prior to implementation. Specifically, the licensee failed to correctly evaluate that a spurious reactor core isolation cooling actuation injecting into the feedwater line resulted in a more than minimal increase in the frequency of occurrence of the loss of feedwater heating accident previously evaluated in the updated final safety analysis report. 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 where the mitigating structure, system, or component maintained its operability or functionality. Since the violation is associated with a Green reactor oversight process violation, the traditional enforcement violation was determined to be a Severity Level IV violation, consistent with the example in paragraph 6.1.d(2) of the NRC Enforcement Policy. There is no cross-cutting aspect assigned to this performance deficiency because the performance deficiency is not indicative of current performance and also because cross-cutting aspects are not assigned to traditional enforcement violations.

Enforcement. The team identified a Severity Level IV, Green, non-cited violation of 10 CFR 50.59, "Changes, Tests, and Experiments," Section (c)(2) which states, in part, "A licensee shall obtain a license amendment pursuant to Section 50.90 prior to implementing a proposed change, test, or experiment if the change, test, or experiment would result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the final safety analysis report (as updated)." Contrary to the above, prior to October 8, 2015, the licensee failed to obtain a license amendment pursuant to 50.90 prior to implementing a change, test, or experiment that resulted in a more than minimal increase in the frequency of occurrence of an accident previously evaluated in the final safety analysis report (as updated). Specifically, the licensee failed to correctly evaluate that a spurious reactor core isolation cooling actuation injecting into the feedwater line resulted in a more than minimal increase in the frequency of occurrence of the loss of feedwater heating accident previously evaluated in the updated final safety analysis report. In response to this issue, the licensee initiated a condition report to document completion of a new evaluation under current regulatory guidelines. This finding was entered into the licensee's corrective action progam as Condition Report CR-RBS-2015-7259. Because this violation was entered into the corrective action program and the violation was not repetitive or willful, this Severity Level IV violation is being treated as a non-cited violation (NCV), consistent with Section 2.3.2.a of the Enforcement Policy: NCV 05000458/2015007-02, "Failure to Obtain Prior NRC Approval for a Change in Reactor Core Isolation Cooling Injection Point."

.2 Permanent Plant Modifications

a. Inspection Scope

The inspectors reviewed 11 permanent plant modifications that had been installed in the plant during the last three years. 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. The list of modifications and other documents reviewed by the inspectors is included as an Attachment to this report

This inspection constituted 11 permanent plant modification samples as defined in IP 71111.17 04.

.2.1 Remove Ball Check from Inlet and Outlet Reactor Feedwater Pumps Sight Glasses Valves

The inspectors reviewed Engineering Change package EC-20398, implemented to replace the feedwater pumps' inlet and outlet sight glass valves FWL-V3002A(B)(C) and FWL-V3003A(B)(C) to remove the ball checks. The valves are required to provide isolation to the corresponding gear increaser lube oil level indicator without impeding the ability to obtain an accurate oil level indication from the sight glass. The existing valves were giving inaccurate oil level readings which the licensee attributed to the ball checks sticking and impeding flow to the sight glass and sight glass vents. This engineering change involved replacement of the existing valves with the same type of valve from the manufacturer, without the ball check.

The inspectors reviewed the design package, discussed the change with the project engineer, and visually inspected the new equipment. The inspectors did not identify any concerns with the design change package.

.2.2 Instrument Air System Dryer Replacement

The inspectors reviewed Engineering Change package EC-37952, implemented to replace the instrument air system heatless air dryers IAS-DRY2(3) with heated air dryers. The dryers are designed to remove any remaining moisture in the compressed

air that escaped the dryer prefilters and provide clean, dry, oil free air at a dew point of -40 degrees Fahrenheit. The existing dryers had a repressurization flow rate as high as 875 standard cubic feet per minute in 15 seconds, which exceeded the design capacity of 600 standard cubic feet per minute for a single instrument air system compressor and resulted in reducing the instrument air system design operating margin to 8 percent. This engineering change involved replacement of the existing heatless air dryers with externally heated air dryers with blower purge air flow. This engineering change also modified safety-related Electrical Transient and Analysis Program (ETAP) calculations since the new heated dryers have 460 VAC power requirements.

The inspectors reviewed the design package, discussed the change with the project engineer, and visually inspected the new equipment. The inspectors did not identify any concerns with the design change package.

.2.3 Diesel Generator Margin Recovery Governor Setpoint Change

The inspectors reviewed Engineering Change package EC-38515, implemented to reduce the Division I and II (EGS-SC90A and EGS-SC90B) diesel generator governor frequency setpoints from 60.0 Hz to 59.7 Hz. The governor frequency setpoint is required to ensure that the diesel generators reach and load at the proper frequency after starting.

During the 2008 Component Design Basis Inspection by the NRC, it was found that the diesel generator electrical load calculations did not account for either the maximum allowable frequency or voltage in the technical specification or for the maximum expected load conditions. The Division I and II diesel generators have a technical specification allowable maximum frequency of 61.2 Hz which results in a 6.12 percent increase in loading on the diesel generator when operating at the upper limit versus operation at the 60.0 Hz frequency setpoint. This increase in load is because the loading on the generator is related to the cube of the difference between the maximum frequency (61.2 Hz) and the frequency setpoint (60.0 Hz). Engineering Change 40578 had decreased the maximum allowable frequency from 61.2 Hz to 60.2 Hz, therefore decreasing the maximum accident loading to be considered on the diesel generators. The existing frequency setpoint of 60.0 Hz did not leave appropriate operating margin below the maximum technical specification maximum frequency. Therefore, this engineering change involved adjusting the diesel generator governor frequency setpoint from 60.0 Hz to 59.7 Hz in order to provide appropriate operating margin above the technical specification minimum frequency (58.8 Hz) and below the proposed technical specification maximum frequency (60.2 Hz).

The inspectors reviewed the design package, discussed the change with the project engineer, and visually inspected the new equipment. The inspectors did not identify any concerns with the design change package.

.2.4 Condenser Refrigerant Pressure Setpoint Change for SWP-PVY32A/B/C/D

The inspectors reviewed Engineering Change package EC-41350, implemented to adjust the crack pressure setpoint for outlet control valves SWP-PVY32A/B/C/D as recommended in Condition Report CR-RBS-2012-0501. The valves are provided to automatically maintain refrigerant pressure independent of fluctuations of the chiller

loading and service water supply temperature. The existing crack pressure setpoint was indicated to be 34.9 psig, but the Setpoint Data Sheet did not accurately reflect the operation of these valves at that setting. This engineering change involved adjusting the crack pressure setpoint to 28.5 ± 3.5 psig. This engineering change also included an evaluation to verify the condenser pressure operating range for the control valves was consistent with the new crack pressure setpoint.

The inspectors reviewed the design package, discussed the change with the project engineer, and visually inspected the new equipment. The inspectors did not identify any concerns with the design change package.

.2.5 Replace ITE 62 K Relay with ABB 62T Relay in Division I and Division II Diesel Generator Panels

The inspectors reviewed Engineering Change package EC-25957, implemented to replace the ITE 62K relays (EGS-PNL2A-62G and EGS-PNL2B-62G) used in differential tripping circuits for the Division I Diesel Generator, EGS-EG1A, and Division II Diesel Generator, EGS-EG1B, that are obsolete. The EGS-PNL2A-62G and EGS-PNL2B-62G timing relays are located in the differential trip circuits of their respective emergency diesel generators. The purpose of a differential relay is to provide short circuit protection for the devices located in its area of protection. The area of protection is between the current transformers that feed a milliamp (mA) signal into the relay that is proportional to the current flowing through the cable at the current transformer . A trip signal is initiated by the differential relay when the magnitude of current in one current transformer does not match the magnitude of current in the other current transformer because a fault has occurred. The new 62T relays are qualified for the same environmental conditions and installed in the same location as the original 62K relays. The new 62T relays are a draw out type similar to the obsolete 62K relays so no modification of EGS-PNL2A and EGS-PNL2B is required. Testing of the new relays will be performed in a similar manner.

The inspectors reviewed the vendor manuals, wiring diagrams and schematics for the new relays and walked down the installation of the new relays to verify the installation met the requirements of the modification and was in accordance with the design. The inspectors did not identify any concerns with the design change package.

.2.6 New Model Level Switch for LSV-LS28A Located on LSCV-C3A

The inspectors reviewed Engineering Change package EC-35100, implemented to change an obsolete level switch. The purpose of LSV-C3A is to provide sealing air to the main steam penetration valve leakage control system. The main control room alarm H13-P808/81A/H04, Div. I penetration valve leakage control system Air Compressor A Trip or Trouble annunciates each time the penetration valve leakage control system compressor LSV-C3A is started. When operations personnel are dispatched to investigate the LSV-C3A separator tank low level alarm, they report that separator tank level is in its normal operating band. Troubleshooting via WO-306064-01 confirmed that the alarm was being received approximately ¼ inch below the normal level. The actual alarm should be received at 4 5/16 inches decreasing. The existing level switch is a GEMS LS-800 with seven contacts or reed switches (three of the contacts are unused) but is no longer available. The replacement level switch is a GEMS LS-800 with four contacts having a slightly different configuration. Another difference between the new

and existing level switch is the relocation of the “low water level reset” vendor contact IFS-3. Actuation occurs such that it will be 9 5/16 inch above the separator tank bottom from the original location of 10 5/16 above tank bottom. The replacement level switch has the same ratings as the original switch. Layout and arrangement requirements of level switch LSV-LS28A are not changed. ASME and fire area boundaries, divisional separation, security considerations, piping configurations, and plant walk-down requirements are not affected or changed. Testing of the new level switch will be performed in a similar manner.

The inspectors reviewed the vendor manuals, wiring diagrams and schematics for the new level switch and walked down the installation of the new switch to verify the installation met the requirements of the modification and was in accordance with the design. The inspectors did not identify any concerns with the design change package.

.2.7 Breaker Modification EJS-SWG1B ACB044 (HVK-CHL1D)

The inspectors reviewed Engineering Change package EC-54464, implemented to modify existing Masterpact circuit breakers associated with Class 1E Switchgear EJS-SWG1B ACB044 (HVK-CHL1D) to eliminate a standing close signal currently present on the close coils of breakers affected. The standing close signal has been determined to be detrimental to breaker closing reliability. This breaker requires use of pre-existing breaker auxiliary contacts (OF1) inherent to the breaker to interrupt the close signal to the breaker coil following closure. The addition of the auxiliary contact in series with the close circuit has no impact on breaker operation or timing and only provides a means to remove the standing close signal. In addition, the circuit requires a new safety related time-delay relay to be installed within its respective motor control center/switchgear cubicle that prevents breaker re-closure within a certain period following the opening of the breaker to allow the chiller start signal (1CR or 1K12) to clear. Condition Report CR-RBS-2014-6284 documents ten safety-related Masterpact breakers could fail to close due to a standing close signal in the breaker control logic. Earlier condition reports CR-RBS-2014-3731 and CR-RBS-2014-3779 document the failures of two air handling units to start due to the failure of their supply breakers to close. Nonconformance Report NCR-573 provided by Nuclear Logistics Incorporated determined that the root cause of the failure was due to the pressure from the anti-pump latch pushing on the close coil plunger causing the rear of the lever to rock up in the back and intermittently catch on the top frame of the mechanism. The result is that any Masterpact breaker with a standing close signal could intermittently fail to close. The addition of the time delay relay will not change the start and stop times of the affected load and is used to delay start signal until the compressor start signal is cleared after opening the breaker. The relay contact closes after a time delay when the breaker is open and opens immediately when the breaker is closed. The auxiliary contact and timer inhibits the close signal after the breaker closes and the relay inhibits access of the chilled water flow signal to the close coil for a five second delay after the breaker opens. The system re-start time delay will bound the time delay added by this change.

The inspectors reviewed the wiring diagrams and schematics for the new auxiliary contact and timer for the affected equipment and verified the post modification for both auto start and manual start testing methods and verified the installation met the requirements of the modification and was in accordance with the design. The inspectors did not identify any concerns with the design change package.

.2.8 Replace Degraded Anchor Bolts

The inspectors reviewed Engineering Change package EC-25335, implemented to replace degraded anchor bolts on at the chemical feed system acid storage tank 1B. The modification included cutting exiting anchor bolts flush with the top of the existing tank foundation, and drilling and installing new embedded expansion anchors in a new adjacent location. The design required a commercial change of ½ in diameter bolts moved no closer than 18 inches to the center of the tank, and ensuring no reinforcing steel in the foundation was cut during the installation of the new bolts.

The inspectors reviewed the design package, discussed the change with the project engineer, and walked down the tank installation with new anchor bolts. The inspectors did not identify any concerns with the design change package.

.2.9 Replace Intercooler Support Bracket on Division I Standby Diesel Generator

The inspectors reviewed Engineering Change package EC-31779, implemented to install three new fabricated intercooler support brackets on the Division I diesel generator engine in order to reduce potential cold spring loads. This engineering change also modified the weld configuration connecting various fabricated pieces of the brackets, and the bolt material and torque value for the bolts used to connect the brackets to the engine block, in order to facilitate the work in this area with limited accessibility.

The inspectors reviewed the design package, discussed the change with the project engineer, and performed a walkdown of the diesel generator in order to visually inspect the new welded pieces. The inspectors did not identify any concerns with the design change package.

.2.10 Independent Spent Fuel Storage Installation (ISFSI) Concrete Erosion Protection Repair

The inspectors reviewed Engineering Change package EC-32683, implemented to repair eroded slopes surrounding the independent spent fuel storage installation pad. During the life of the plant, the slopes in this area have experienced erosion due to groundwater under the original shotcrete. The design change includes the installation of drains to help minimize groundwater seepage, and the slopes will be reinforced by concrete canvas, a material that is installed in rolls while pliable, then hardens with the application of water. This product has proven to be more effective with slope stabilization with a longer lifespan without degradation.

The inspectors reviewed the modification package, discussed it with the project engineer, and walked down the eroded slopes and areas receiving the new stabilization product. The inspectors did not identify any concerns with the design change package.

.2.11 Sally Port Gate Replacement

The inspectors reviewed Engineering Change package EC-53068, implemented to replace the existing sally port cantilever crash gate. The gate is a vehicle barrier system located outside the building yard to provide and maintain physical separation between the protected area and the areas outside the fence. The existing gate experienced wear

and degradation from years of service, including damage from accidental impact with vehicles, causing it to only open halfway and often become stuck. The new gate is a rolling gate design, requiring some modification to the concrete foundation for wheels, and protection of nearby plant equipment.

The inspectors reviewed the design package, discussed the change with the project engineer, and visually inspected the new sally port gate and concrete wheel base. The inspectors did not identify any concerns with the design change package.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

4OA2 Problem Identification and Resolution

.1 Review of Corrective Action Program Documents

The inspectors reviewed corrective action program documents that identified or were related to 10 CFR 50.59 program and permanent plant modifications. The inspectors reviewed these documents to evaluate the effectiveness of corrective actions related to permanent plant modifications and evaluations of changes, tests, and experiments. In addition, corrective action documents written on issues identified during the inspection were reviewed to verify adequate problem identification and incorporation of the problems into the corrective action system. The list of specific corrective action documents that were sampled and reviewed by the inspectors are listed in the attachment to this report.

4OA6 Meetings

Exit Meeting Summary

On October 8, 2015, the inspectors presented the initial inspection results to Mr. T. Brumfield, Director of Regulatory and Performance Improvement, and other members of the licensee's staff. The licensee acknowledged the results as presented. While some proprietary information was reviewed during this inspection, no proprietary information was included in this report.

4OA7 Licensee-Identified Violation

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

The licensee failed to meet the requirements of 10 CFR 50.71(e) to update the final safety analysis report. The final safety analysis report sections contained prior information and data that did not reflect the current plant configuration. The examples are:

- Chapter 3.9B, “Mechanical Systems and Components (GE Scope Supply)” Section 3.9.5.1.1.12B, “Vent and Head Spray Nozzle” described the head spray function of the reactor core isolation cooling system. However, the head spray function of the reactor core isolation cooling system had been removed from the final safety analysis report in 1998 via License Amendment Request 98-01 in which this section was erroneously omitted.

- Sections 2.4.2.3.1 and 12.3.2.4.5 of the updated safety analysis report describe the roofing systems on site to be sloped with built-up roofing material for construction. In 1996, a condition report was written documenting the discrepancy between design documents which indicate built-up roofing and insulation, and the actual condition of the roofs on the Auxiliary Building, Standby Cooling Tower Pump house, Radwaste Building, and Diesel Generator Building, which have neither of these materials.

Title 10 CFR 50.71, “Maintenance of Records, Making of Reports,” Section (e) which states, in part, “Each person licensed to operate a nuclear power reactor shall update periodically the final safety analysis report originally submitted as part of the application for the license, to assure that the information included in the report contains the latest information developed. This submittal shall contain all the changes necessary to reflect information and analyses submitted to the Commission by the applicant or licensee or prepared by the applicant or licensee pursuant to Commission requirement since the submittal of the original or the last update to the final safety analysis report.” Contrary to the above, the licensee did not update the final safety analysis report to assure that the information included in the report contains the latest information developed. Specifically, the licensee failed to ensure the final safety analysis report reflected the current plant configuration, as evidenced by the above examples.

The NRC’s significance determination process considers the safety significance of findings by evaluating their potential safety consequences. The traditional enforcement process separately considers the significance of willful violations, violations that impact the regulatory process, and violations that result in actual safety consequences. Traditional enforcement applied to this performance deficiency because it involved a violation that impacted the regulatory process. Assessing the violation in accordance with the NRC Enforcement Policy, the team determined it to be a Severity Level IV violation because the lack of up-to-date information in the final safety analysis report has not resulted in any unacceptable change to the facility or procedures (NRC Enforcement Policy example 6.1.d.4). This issue was entered into the licensee’s corrective action program as Condition Report CR-RBS-2015-7259.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

M. E. Olson, Site Vice President, Operations
D. Baker, Training Instructor
C. Blackledge, EFIN Supervisor
K. Borneman, EFIN Engineer
B. Burmeister, Licensing
J. Carter, Design Engineer
J. Clark, Manager of Regulatory Assurance
E. Clevenger, Design Engineer
A. Coates, Design Engineer
F. Corley, Manager of Programs and Design Engineering
R. Crawford, System Engineering Supervisor
E. Deweese, Acting Design Supervisor
R. Doer, Systems Engineer
J. Dunkelberg, Contractor
K. Fancher, System Engineer
R. Finish, Design Engineer
A. Frederickson, System Engineer
T. Gates, Manager of Operations Support
G. Hendl, EFIN Engineer
K. Jelks, System Engineer
A. Johnson, Fire Marshall
K. Klamert, System Engineer
J. Langberg, Contractor
M. Litherland, Design Engineer
P. Matzke, Design Supervisor
G. Mermigas, Design Engineer
S. Miller, Design Engineer
M. Price, System Engineer
R. Schellinger, Operator
H. Smith, EFIN Engineer
G. Svestka, Design Engineer
S. Tiwari, Design Engineer
A. Wilson, Operator

NRC Personnel

J. Sowa, Senior Resident Inspector
N. Hernandez, Acting Senior Resident Inspector
B. Parks, Acting Resident Inspector

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000458-2015-007-01	NCV	(Section 1R17)
05000458-2015-007-02	NCV	(Section 1R17)

LIST OF DOCUMENTS REVIEWED

The following is a list of documents reviewed during the inspection. Inclusion on this list does not imply that the NRC inspectors reviewed the documents in their entirety, but rather, that selected sections of portions of the documents were evaluated as part of the overall inspection effort. Inclusion of a document on this list does not imply NRC acceptance of the document or any part of it, unless this is stated in the body of the inspection report.

Engineering Change Package

EC - 45118	EC- 44959	EC- 31779
EC –32683	EC- 51521	EC - 37847
EC –55796	EC- 01121	EC - 25335
EC –25957	EC- 32107	EC - 35100
EC –37843	EC- 37952	EC - 40180
EC –41350	EC- 44962	EC - 44963
EC –44964	EC- 54464	EC - 47374
EC - 38941	EC - 31485	EC – 50517
EC – 01121	EC - 41542	EC – 01036
EC – 5445	EC - 31715	EC - 37403
EC - 37404		

10 CFR 50.59 Evaluations

Number	Description or Title	Revision
2012-02	Chiller Control Digital Upgrade	0
2015-01	Cycle 19 COLR (LAR-2015-01)	
2012-01	EC-37843 Add time delay to RCIC/RHR steam flow transmitters	11
1995-08	Safety Evaluation to Support Design Document Modification for Adding Low Level Radwaste Building	0
1997-27	Safety Evaluation to Support Design Document Modification	1
1998-21	RCIC Injection Modification	1
2013-01	EC – 14988 Addition of Constant Level Oilers	0

Miscellaneous

Number	Description or Title	Revision/Date
LCN 12.04-002	USAR Change for Adding Radioactive Storage Areas	August 23, 2013
LCN 03.09B-024	USAR Change for Head Spray Function of RHR/RCIC	June 10, 2015
51-9207360	Engineering Information Record for Entergy Fleet Fukushima Program Flood Hazard Reevaluation Report for RBS	January 30, 2014
VTD-N432-0118	Cutler Hammer Definite Purpose Contactors	0
VTD-F081-0114	Fenwal Series 15000,16000, 17000, and 18000 Thermoswitch Temperature Controllers	0
1.ILLSV.027	Loop Calibration Report for LSV-LS28A	9
	Transamerica Delaval GEMS LS-800 Multi Station Liquid Level Switches	September 2011
RC-5046-A	Equipment Performance Specifications Certification for ABB 62T Time Delay Relay	December 2010
RC-5146-A	Class 1E Electrical Equipment Qualification for ABB 62T Time Delay Relay	December 2010
RC-5546	ABB 62T Seismic Qualification Report	July 15, 1988
41-529S	ABB Descriptive Bulletin for Circuit Shield Type 62T Time Delay Relay	September 1995
IB 7.7.1.7-6	ABB Circuit Shield Type 62T Solid-State Timing Relay Instructions	0
3221.512-327-003A	Turbonetics Inc. Compressor Instruction Manual for SAC AD-73	301
FDI/MCNP	General Electric MCNP for MPL H13-P618, P621 RCIC Break Detection	0
	Gould/ITE Vendor Manual Unitized Combination Starter	0
LBDCR#7.3-209	RBS Masterpact Rewire to Eliminate Standing Close	0
LBDCR#8.01-026	FLEX Electrical AC Bus Transfer Switch	1
LTR-LAM-13-31, Rev. 1	Evaluation of a Continuous Nitrogen Supply to Waterford 3 SIT 2B during LOCA	April 18, 2013
51-9207360-000	Entergy Fleet Fukushima Program Flood Hazard Reevaluation Report for River Bend Station	January 30, 2014
07.06-051	LBDCR for LSV-LS28A	February 28, 2012

Corrective Action Program Documents

CR-RBS-2015-05308	CR-RBS-2015-00806	CR-RBS-2015-01320	CR-RBS-2013-05492
CR-RBS-2015-06936	CR-RBS-2015-00040	CR-RBS-2015-06891	CR-RBS-2015-06880
CR-RBS-2015-06881	CR-RBS-2015-06899	CR-RBS-2015-06901	CR-RBS-2015-06902
CR-RBS-2015-05948	CR-RBS-2015-05508	CR-RBS-2015-02683	CR-RBS-2015-03226
CR-RBS-2013-05492	CR-RBS-2015-07205	CR-RBS-2015-06953	CR-RBS-2015-07092
CR-RBS-2015-05038	CR-RBS-2015-01320	CR-RBS-2015-00806	CR-RBS-2015-07157
CR-RBS-2006-04689	CR-RBS-2006-04460	CR-RBS-2012-06015	CR-RBS-2014-00633
CR-RBS-2010-02632	CR-RBS-2015-06947	CR-RBS-2015-07163	

Calculations

Number	Description or Title	Revision
G13.18.13.2*086	SSW Maximum Basin Temperature Calculation	1
G13.18.3.6*016	Degraded Voltage Calculation for Class IE Buses and 480V Motor Operated Valves	2

Procedures

Number	Description or Title	Revision
EN-LI-100	Process Applicability Determination	9
EN-LI-100	Process Applicability Determination	16
EN-LI-101	10 CFR 50.59 Evaluations	12
EN-LI-112	Engineering Change Request Process	8
EN-DC-115	Engineering Change Process	17
EN-DC-117	Post Modification Testing and Special Instructions	7
EN-DC-136	Temporary Modifications	11
EN-DC-141	Design Inputs	15
EN-DC-213	Engineering Quality Review	6
STP-302-0601	DIV I Off Site AC Sources Transfer Test	13
STP-303-1601	120 and 480V AC Breaker Overload Functional Test	30
STP-303-1601	120 and 480V AC Breaker Overload Functional Test	31
AOP-0029	Severe Weather Operation	36

Procedures

Number	Description or Title	Revision
EN-FAP-EP-010	Severe Weather Response	2

Drawings

Number	Description or Title	Revision
105D5116AA SH2	Functional Control Diagram Leak Detection System	6
105D5116AA SH2	Functional Control Diagram Leak Detection System	13
762E293AA Sh1	IED Leak Detection System	2
762E293AA SH4	IED Leak Detection System	2
762E297AA Sht. 1	FCD Reactor Core Isolation Cooling system	6
0216.210.085-009	Electronic Control Diagram for Nuclear Plant Duty control Building Centrifugal Liquid Chiller	301
0221.512-327-027	Equipment No. LSV-C3A & B Schematic Process System	J
0221.512-327-027	Equipment No. LSV-C3A & B Schematic Process System	301
0221.512-327-032	Schematic Electrical Control	302
EE-003KY	Wiring Diagram leakage Panels LSV-PNL55A & LSV-PNL55B	12
ESK-07LSV04	Elementary Diagram 120 VAC Control CKT Leakage Cont. Air Compressor LSV-C3A, LSV-PNL55A	12
TLD-LSV-027	Test Loop Diagram Separator Tank Multi-Level LSV-LS28A	0
LSK-27-29B	Logic Diagram Penetration Valve Leakage Control System	12
TLD-HVK-026	Test Loop Diagram HVK-CHL1B Compressor Oil heater Temperature HVK-TS67B	1
EE-003E	Wiring Diagram HVK-CHL1B & 1D Control Building	5
828E539AA Sht.4	Elementary Diagram Reactor Core Isolation Cooling System	28
828E539AA Sht.4	Elementary Diagram Reactor Core Isolation Cooling System	31
828E539AA Sht. 5	Elementary Diagram Reactor Core Isolation Cooling System	31
828E539AA Sht. 10	Elementary Diagram Reactor Core Isolation Cooling	7

Drawings

Number	Description or Title	Revision
	System	
828E539AA Sht. 10	Elementary Diagram Reactor Core Isolation Cooling System	22
828E539AA Sht. 10	Elementary Diagram Reactor Core Isolation Cooling System	24
828E539AA Sht. 13	Elementary Diagram Reactor Core Isolation Cooling System	27
828E539AA Sht. 13	Elementary Diagram Reactor Core Isolation Cooling System	29
828E539AA Sht. 15	Elementary Diagram Reactor Core Isolation Cooling System	29
1-1CS-001	Auxiliary Building Elevation 95'-9" Line 1-RHS-008-036-2	7
1-ICS-001-CD-B	Reactor Building Elevation 95'-9" Line 1-ICS-008-001-1	15
1-ICS-003-CD-A	Auxiliary Building Elevation 114'-0" Line 1-ICS-008-003-1	9
1-ICS-004	Reactor Building Elevation 114'-0" Line 1-ICS-008-004-2	3
1-1CS-013	Auxiliary Building Elevation 70'-6" Line 1-ICS-004-13-2	7
1-RHS-036-CD-A	Auxiliary Building Elevation 114'-0" Line 1-RHS-008-036-2	15
1-RHS-036-CD-B	Auxiliary Building Elevation 114'-0" Line 1-RHS-008-036-2	7
PCD-RHS-036-CD-B	Auxiliary Building Elevation 114'-0" Line 1-RHS-008-036-2	7
PID-27-06A	Engineering P and I Diagram system 209 Reactor Core Isolation Cooling	45
ESK-5ENS01	Elementary Diagram – 4.16kV Switchgear Standby Bus 1A Normal Supply ACB	18
USAR Figure 8.1-4	Fancy Point Substation 230kV Bays and Peripheral Loads	22
0242.5363-265-083	Miscellaneous Details EJS-LDC1B	301
0242.5363-265-090	Wiring Diagram Unit #4 EJS-LD1CB	302
0242.5363-265-091	Wiring Diagram Unit #4 EJS-LD1CB	301
0242.5363-265-092	Wiring Diagram Unit #5 EJS-LD1CB	301

Drawings

Number	Description or Title	Revision
0242.5363-265-093	Wiring Diagram Unit #5 EJS-LD1CB	300
ESK-02X	480V Switchgear Details	0
ESK-06HVK04 Sht. 1	Elementary Diagram480V Switchgear Control Building Chilled Water Compressor HVK-CH1D	20
ESK-06HVK04 Sht. 2	Elementary Diagram480V Switchgear Control Building Chilled Water Compressor HVK-CH1D	3
ESK-06HVC19 Sht. 1	Elementary Diagram480V Switchgear Control Building Chilled Air Handling Unit ACU2B	14
LSK-22-12U	Logic Diagram Control Building Chilled Water	0

Work Orders

52499054 00372861 108105-03

Agencywide Documents Access and Management System (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

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

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DATE	10/29/15	10/26/15	10/23/15	10/29/15

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Letter to Eric W. Olson from Thomas R. Farnholtz, dated October 29, 2015

SUBJECT: RIVER BEND STATION – NRC EVALUATIONS OF CHANGES, TESTS, AND EXPERIMENTS AND PERMANENT PLANT MODIFICATIONS BASELINE INSPECTION REPORT 05000458/2015007

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