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

June 25, 2019

EA-18-122

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

SUBJECT: RIVER BEND STATION – REVISED NRC BASELINE INSPECTION REPORT 05000458/2018012

Dear Mr. Vercelli:

On July 16, 2018, the U.S. Nuclear Regulatory Commission (NRC) completed a baseline inspection at your River Bend Station, Unit 1. On May 31 and July 16, 2018, the NRC inspection team discussed the results of this inspection with Mr. Bill Maguire and other members of your staff. The results of this inspection were originally documented in a report dated July 18, 2018 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18194A413). On August 23, 2018, you provided a response (ADAMS Accession ML18235A636) in which you contested two of the violations issued in the original report. On May 14, 2019, the NRC issued a response (ADAMS Accession ML19134A313) in which we concluded that one of the two contested violations were valid, and one of the violations should be withdrawn. Accordingly, the enclosed inspection report is being re-issued to reflect the removal of non-cited violation (NCV) 05000458/2018012-07.

The original inspection report documented five findings of very low safety significance (Green), four which involved violations of NRC requirements. Two additional violations were determined to be Severity Level IV under the traditional enforcement process. In accordance with the result of the contested violation review referenced above, one of these two Severity Level IV violations is being withdrawn. The revised inspection report is enclosed, which includes five Green findings, four which involved violations of NRC requirements, as well as one additional Severity Level IV violation. The NRC is treating these violations as non-cited violations (NCVs) consistent with Section 2.3.2.a of the NRC 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 by CYoung Acting For/

Jason W. Kozal, Chief
Project Branch C
Division of Reactor Projects

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

Enclosure:
Inspection Report 05000458/2018012
w/ Attachment: Documents Reviewed

U.S. NUCLEAR REGULATORY COMMISSION
Inspection Report

Docket Number:	05000458
License Number:	NPF-47
Report Number:	05000458/2018012
Enterprise Identifier:	I-2018-012-0015
Licensee:	Entergy Operations, Inc.
Facility:	River Bend Station
Location:	Saint Francisville, Louisiana
Inspection Dates:	February 1, 2018 to July 16, 2018.
Inspectors:	J. Sowa, Senior Resident Inspector J. Drake, Senior Reactor Inspector C. Young, Senior Project Engineer M. O'Banion, Resident Inspector (Acting) B. Parks, Resident Inspector
Approved By:	J. Kozal, Chief, Branch C Division of Reactor Projects

SUMMARY

The U.S. Nuclear Regulatory Commission (NRC) continued monitoring the licensee's performance by conducting a baseline inspection at River Bend Station in accordance with the Reactor Oversight Process. The Reactor Oversight Process is the NRC's program for overseeing the safe operation of commercial nuclear power reactors. Refer to https://www.nrc.gov/reactors/operating/oversight.html for more information. Findings and violations being considered in the NRC's assessment are summarized in the tables below.

List of Findings and Violations

Failure to Identify and Correct a Broken Feedwater Chemistry Probe			
Cornerstone	Significance	Cross-cutting Aspect	Report Section
Barrier Integrity	Green NCV 05000458/2018012-02 Closed	None	71152 – Problem Identification and Resolution
Two examples of a self-revealed non-cited violation (NCV) of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," were identified for the licensee's failure to identify that a broken chemistry probe in the feedwater system had the potential to cause an adverse impact on plant safety, and promptly implement appropriate measures to address that condition.			

Failure to Provide Adequate Procedures for Post-Scram Recovery			
Cornerstone	Significance	Cross-cutting Aspect	Report Section
Mitigating Systems	Green NCV 05000458/2018012-06 Closed	None	71111.18 – Plant Modifications
The inspectors reviewed a self-revealed, non-cited violation of Technical Specification 5.4.1.a for the licensee's failure to establish, implement and maintain a procedure required by Regulatory Guide 1.33, Revision 2, Appendix A, dated February 1978. Specifically, Procedure OSP-0053, "Emergency and Transient Response Support Procedure," Revision 22, which is required by Regulatory Guide 1.33, inappropriately directed operations personnel to establish feedwater flow to the reactor pressure vessel using the main feedwater regulating valve as part of the post-scram actions. This resulted in the main feedwater regulating valves being operated outside their design limits. This resulted in catastrophic failure of the main feedwater regulating valve variseals and subsequent damage to multiple fuel assemblies.			

Failure to Develop an Adequate Operational Decision-Making Issue for Compensatory Measures Related to a Degraded Condition of the Feedwater System Sparger Nozzles			
Cornerstone	Significance	Cross-cutting Aspect	Report Section
Mitigating Systems	Green NCV 05000458/2018012-05 Closed	[H.3] – Human Performance, Change Management	71111.15 – Operability Determinations and Functionality Assessment
The inspectors identified a Green non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” for the failure to develop an adequate Operational Decision-Making Issue (ODMI) document per Procedure EN-OP-111, “Operational Decision-Making Issue Process.” Specifically, the licensee failed to develop an ODMI that provided adequate guidance to the operators for safely operating the plant with degraded feedwater sparger nozzles.			

Failure to Establish Procedural Guidance for Determining Core Flow During Unanticipated Single Loop Operations			
Cornerstone	Significance	Cross-cutting Aspect	Report Section
Initiating Events	Green NCV 05000458/2018012-03 Closed	[P.3] – Problem Identification and Resolution, Resolution	71153 – Follow-up of Events and Notices of Enforcement Discretion
The inspectors reviewed a self-revealed, non-cited violation of 10 CFR Part 50 Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” for the licensee’s failure to establish appropriate instructions in the abnormal operating procedure for thermal hydraulic instabilities. Specifically, the procedural step for determining core flow when in single loop operations at low power did not provide appropriate instructions to operators. As a result, station personnel could not conclusively determine core flow and inserted a manual reactor scram.			

Failure to Conduct Adequate Transient Snap Shot Assessment Following Recirculation Pump Trip			
Cornerstone	Significance	Cross-cutting Aspect	Report Section
Initiating Events	Green FIN 05000458/2018012-01 Closed	None	71152 – Problem Identification and Resolution
The inspectors identified a finding for the licensee’s failure to adequately validate simulator response during a transient snap shot assessment following an unexpected trip of reactor recirculation pump A on December 19, 2012.			

Failure to Submit a Licensee Event Report for a Manual Scram			
Cornerstone	Significance	Cross-cutting Aspect	Report Section
None	SL-IV NCV 05000458/2018012-04 Closed	None	71153 – Follow-up of Events and Notices of Enforcement Discretion
The inspectors identified a Severity Level IV non-cited violation of 10 CFR 50.73, “Licensee Event Report System,” for the licensee’s failure to submit a required licensee event report (LER). Specifically, on February 1, 2018, after an unexpected trip of the recirculation pump B, the licensee initiated a manual scram of the reactor that was not part of a preplanned sequence and failed to submit an LER within 60 days.			

INSPECTION SCOPES

Inspections were conducted using the appropriate portions of the inspection procedures (IPs) in effect at the beginning of the inspection unless otherwise noted. Currently approved IPs with their attached revision histories are located on the public website at http://www.nrc.gov/reading-rm/doc-collections/insp-manual/inspection-procedure/index.html. Samples were declared complete when the IP requirements most appropriate to the inspection activity were met consistent with Inspection Manual Chapter (IMC) 2515, "Light-Water Reactor Inspection Program - Operations Phase." The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel to assess licensee performance and compliance with Commission rules and regulations, license conditions, site procedures, and standards.

REACTOR SAFETY

71111.15—Operability Determinations and Functionality Assessments (1 Sample)

The inspectors evaluated the following operability determinations and functionality assessments:

(1) Review of Operational Decision-Making Issue (ODMI) associated with damaged feedwater sparger on February 8, 2018

71111.18—Plant Modifications (2 Samples)

The inspectors evaluated the following temporary or permanent modifications:

(1) OSP-0053, "Emergency And Transient Response Support Procedure," following decision to control reactor vessel level with main feedwater regulating valves during post-scram operations

(2) Review of plant operation following modification to feedwater sparger nozzles 7 and 8

OTHER ACTIVITIES – BASELINE

71152—Problem Identification and Resolution

Annual Follow-up of Selected Issues (3 Samples)

The inspectors reviewed the licensee's implementation of its corrective action program related to the following issues:

(1) Review of 1) simulator modelling of core parameters during a recirculation pump trip at low power and 2) licensed operator training associated with single loop operations at low power

(2) Actions to address a broken isokinetic chemistry sampling probe in the feedwater system

(3) Actions to address fuel failures caused by debris material in the reactor vessel

71153—Follow-up of Events and Notices of Enforcement Discretion

Personnel Performance (1 Sample)

(1) The inspectors evaluated operator response to the unexpected trip of the reactor recirculation pump B on February 1, 2018.

INSPECTION RESULTS

Failure to Identify and Correct a Broken Feedwater System Chemistry Probe			
Cornerstone	Significance	Cross-cutting Aspect	Report Section
Barrier Integrity	Green NCV 05000458/2018012-02 Closed	None	71152 – Problem Identification and Resolution
Two examples of a self-revealed Green finding and associated NCV of 10 CFR Part 50, Appendix B, Criterion XVI, were identified for the licensee's failure to identify that a broken chemistry probe in the feedwater system had the potential to cause an adverse impact on plant safety, and promptly implement appropriate measures to address that condition.			

Description:

In 1999, the licensee initiated Condition Report CR-RBS-1999-1011 to document that an isokinetic chemistry sample probe was found to be missing from its installed location in the feedwater system, having broken off in the system. Following unsuccessful attempts to locate and remove the missing probe, the licensee performed evaluation ER-99-0539 to evaluate the potential impact of the missing probe on the continued operation and function of feedwater system components. This evaluation concluded that the missing probe remaining in the system would not present any hazard to any feedwater system components and would have no adverse effect on continued operation. This conclusion was based, in part, on a calculation showing that feedwater flow would not have enough energy to levitate the probe past a 20-foot vertical riser portion of the system, and therefore would not have the potential to enter a feedwater sparger in the reactor vessel downstream of the vertical riser. Another calculation showed that the impact energy of the loose probe on any feedwater components would be negligible.

In March 2004, the NRC issued Information Notice (IN) 2004-06, "Loss of Feedwater Isokinetic Sampling Probes at Dresden Units 2 and 3" (ADAMS Accession No. ML040711214). The IN discussed that broken probes had been discovered at five other stations from 1990 to 2001, and further described the conditions discovered at Dresden Nuclear Power Station (Dresden), Units 2 and 3. In 2003, three holes in a feedwater sparger at Dresden Unit 2 were discovered, along with the missing feedwater probe in the sparger, which had apparently caused the damage. Two probes were discovered to be in a feedwater sparger in Dresden Unit 3, with no damage to the sparger having occurred yet. These conditions demonstrated that not only could the probes be transported to the feedwater spargers in the reactor vessel, but that they could potentially damage the spargers. The licensee's evaluation of this operating experience concluded that, since the broken probe at River Bend had been replaced with a probe of a design not susceptible to the same failure, no further action was needed. The licensee failed to address the potential impacts of the adverse condition of the broken probe that remained loose in the feedwater system.

In 2011, the licensee documented an evaluation of a similar condition that had been discovered at Brunswick Steam Electric Plant, Unit 2, where a feedwater sample probe was discovered inside a feedwater sparger. The licensee's evaluation of this operating experience concluded that the current design (i.e. the probe that replaced the previous broken probe) was not susceptible to this kind of failure. The licensee again failed to address the impact of the previous broken probe that remained in the system, given that its potential to be transported into a feedwater sparger in the reactor vessel had been shown.

In January 2018, the licensee discovered damage in the form of two holes in feedwater sparger nozzles in the reactor vessel, with the broken probe protruding from one of the holes in the direction of the other. The broken probe remaining in the feedwater system resulted in potential adverse impacts on the reactor vessel wall due to impingement of feedwater flow through the holes in the damaged sparger, as well as potential adverse impacts on the integrity of fuel cladding due to the introduction of foreign material (pieces of the feedwater sparger and chemistry probe) in the reactor vessel.

Corrective Actions: The broken probe was removed from the system. The licensee performed evaluations to identify plant operational limitations to ensure that adverse impacts to reactor pressure vessel wall integrity from additional holes in a feedwater sparger are minimized. The licensee also issued an action to perform a review of historical loose parts evaluations to add to tracking mechanisms and ensure adequacy of previous evaluations.

Corrective Action Reference: CR-RBS-2018-0294, CR-RBS-2018-0613, and CR-RBS-2017-2828.

Performance Assessment:

Performance Deficiency: The licensee's failure on two occasions to identify a broken chemistry probe in the feedwater system had the potential to cause an adverse impact on plant safety and to promptly implement appropriate measures to address that condition was a performance deficiency.

Screening: The inspectors determined the performance deficiency was more than minor because it was associated with the Cladding Performance, as well as the RCS Equipment and Barrier Performance, attributes of the Barrier Integrity Cornerstone, and adversely impacted the cornerstone objective to provide reasonable assurance that physical design barriers (fuel cladding, reactor coolant system, and containment) protect the public from radionuclide releases caused by accidents or events. Specifically, the unaddressed condition of the broken probe remaining in the feedwater system resulted in damage to the feedwater sparger, which resulted in thermal stresses to the reactor pressure vessel due to feedwater impingement on the inner reactor pressure vessel wall, as well as the introduction of foreign material inside the reactor vessel having the potential to result in damaged fuel. The licensee performed an evaluation to determine what plant operational limitations were necessary to ensure that additional thermal stresses on the reactor pressure vessel inner wall remained below a threshold that would challenge the structural integrity of the vessel.

Significance: In accordance with Inspection Manual Chapter 0609, Appendix A, Section 5.0, RCS boundary issues other than pressurized thermal shock are evaluated under the Initiating Events Cornerstone. Using Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process for Findings At-Power," Exhibit 1, "Initiating Events Screening Questions," the finding was screened, as a potential loss of coolant accident (LOCA) initiator, as having very low safety significance (Green) because, after a reasonable assessment of degradation, the finding could not result in exceeding the RCS leak rate for a small LOCA and could not have likely affected other systems used to mitigate a LOCA.

Cross-cutting Aspect: A cross-cutting aspect of P.5, Operating Experience, was determined to be applicable to the performance deficiencies; however, no cross-cutting aspect was assigned since the performance deficiencies occurred in 2004 and 2011 and are not indicative of current licensee performance.

Enforcement:

Violation: Title 10 CFR Part 50, Appendix B, Criterion XVI, requires, in part, that measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected. Contrary to the above, from June 2004 to January 2018, the licensee failed to establish measures to assure that a condition adverse to quality was promptly identified and corrected. Specifically, the licensee failed to identify and correct a condition involving a broken sampling probe inside the feedwater system. The uncorrected condition resulted in damage to a feedwater sparger, with the potential to impact the available margin for integrity of the reactor vessel.

Disposition: This violation is being treated as a non-cited violation, consistent with Section 2.3.2.a of the Enforcement Policy.

Failure to Provide Adequate Procedures for Post-Scram Recovery			
Cornerstone	Significance	Cross-cutting Aspect	Report Section
Mitigating Systems	Green NCV 05000458/2018012-06 Closed	None	71111.18 – Plant Modifications

The inspectors reviewed a self-revealing, non-cited violation of Technical Specification 5.4.1.a for the licensee's failure to establish, implement and maintain a procedure required by Regulatory Guide 1.33, Revision 2, Appendix A, dated February 1978. Specifically, Procedure OSP-0053, "Emergency and Transient Response Support Procedure," Revision 22, which is required by Regulatory Guide 1.33, inappropriately directed operations personnel to establish feedwater flow to the reactor pressure vessel using the main feedwater regulating valve (MFRV) as part of the post-scram actions. This resulted in the MFRVs being operated outside their design limits. This resulted in catastrophic failure of the MFRV variseals and subsequent damage to multiple fuel assemblies.

Description:

In January 2015, the licensee revised Procedure OSP-0053, "Emergency And Transient Response Support Procedure," to use one of the three MFRVs to control reactor water level following a scram event, and not use C33-LVF002, Start-Up FRV, which is designed to be used for this application. This resulted in proceduralizing the use of a MFRV in circumstances below the minimum controllable flow for the MFRV of 209,000 lbs/hr that the Main FRV Copes Vulcan sizing datasheet provides as a minimum controllable flow condition. As a result of this change to the procedure to use a MFRV, the valves cycled numerous times in the process of controlling level at low flow post-scram when feedwater flow demand was below the MFRV minimum controllable flow volume. This repeated cycling of the valve led to excessive open/close cycling of the MFRVs and caused the catastrophic failure of the variseals.

As a result, foreign material parts of the variseal were introduced into the core. It is suspected that this material resulted in six nuclear fuel cladding failures caused by debris fretting.

Corrective Actions: The licensee revised Procedure OSP-0053, “Emergency and Transient Response Support Procedure,” to control reactor vessel level post scram using a startup feedwater regulating valve and modified the design of the MFRV variseal.

Corrective Action Reference: CR-RBS-2016-00893

Performance Assessment:

Performance Deficiency: The failure to establish adequate procedural guidance for operation of the main feedwater system was a performance deficiency.

Screening: The performance deficiency was more than minor, and therefore a finding, because it was associated with the procedure quality 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. Specifically, Procedure OSP-0053, “Emergency and Transient Response Support Procedure,” Revision 22, inappropriately directed operations personnel to establish feedwater flow to the reactor pressure vessel using the MFRV as part of the post-scram actions. This resulted in the MFRVs being operated outside their design limits.

Significance: The inspectors screened the finding in accordance with Inspection Manual Chapter 0609, Appendix A, “The Significance Determination Process for (SDP) for Findings At-Power.” Using Inspection Manual Chapter 0609, Appendix A, Exhibit 2, “Mitigating Systems Screening Questions,” the inspectors determined this finding was of very low safety significance (Green) because the finding: (1) was not a deficiency affecting the design or qualification of a mitigating structure, system, or component, and did not result in a loss of operability or functionality; (2) did not represent a loss of system and/or function; (3) did not represent an actual loss of function of at least a single train for longer than its technical specification allowed outage time, or two separate safety systems out-of-service for longer than their technical specification allowed outage time; and (4) did not represent an actual loss of function of one or more nontechnical specification trains of equipment designated as high safety-significant in accordance with the licensee’s maintenance rule program.

Cross-cutting Aspect: No cross-cutting aspect was assigned since the performance deficiency occurred in January 2015 and is not indicative of current licensee performance.

Enforcement:

Violation: Technical Specification 5.4.1.a requires in part, that written procedures shall be established, implemented, and maintained covering the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, dated February 1978. Regulatory Guide 1.33, Appendix A, Section 6.u., identifies procedures for responding to a reactor trip as required procedures. Procedure OSP-0053, Attachment 16, “Post Scram Feedwater/Condensate Manipulations Below 5% Reactor Power,” was a procedure established by the licensee for responding to a reactor trip.

Contrary to the above, from January 30, 2015, until April 13, 2017, the licensee failed to maintain adequate written procedures for responding to a reactor trip. Specifically, Procedure OSP-0053 inappropriately directed operations personnel to establish feedwater

flow to the reactor pressure vessel using the MFRV as part of the post-scram actions. The MFRV operator characteristics are not designed to operate at the low flow conditions immediately following a reactor scram from high power. As a result, the MFRV variseals degraded and resulted in damage to multiple fuel assemblies. Subsequent to the event, the licensee changed the procedure, directing operations personnel to utilize one of the startup feedwater regulating valves. Disposition: This violation is being treated as a non-cited violation consistent with Section 2.3.2.a of the NRC Enforcement Policy.

Failure to Develop an Adequate Operational Decision-Making Issue for Compensatory Measures Related to a Degraded Condition of the Feedwater System Sparger Nozzles			
Cornerstone	Significance	Cross-cutting Aspect	Report Section
Mitigating Systems	Green NCV 05000458/2018012-05 Closed	[H.3] – Human Performance, Change Management	71111.15 – Operability Determinations and Functionality Assessments
The inspectors identified a Green non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for the failure to develop an adequate operational decision-making issue (ODMI) document per Procedure EN-OP-111, "Operational Decision-Making Issue Process." Specifically, the licensee failed to develop an ODMI that provided adequate guidance to the operators for safely operating the plant with degraded feedwater sparger nozzles.			
<u>Description</u>: During a reactor startup on February 1, 2018, reactor recirculation pump B unexpectedly tripped during an attempted upshift to fast speed. As a result, the plant was operating with recirculation pump A in fast speed and recirculation pump B not running. Prior to this startup, during an outage that was being conducted to replace failed fuel assemblies, damage to feedwater sparger nozzles was identified. Example 1: The evaluation of the damaged feedwater sparger nozzles 7 and 8 on sparger N4C identified that the damaged sections of the feedwater sparger nozzles had the potential to adversely affect the vessel cladding by allowing relatively colder water to directly flow into the relatively hotter vessel wall, thus inducing thermal fatigue. All components of the reactor coolant system (RCS) are designed to withstand effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, power transients, and reactor trips. Limits are established for pressure and temperature changes during RCS heatup and cooldown, such that plant systems remain within the design assumptions and the stress limits for cyclic operation. Limits on RCS pressure, temperature, heatup rate, and cooldown rate define allowable operating regions and operating cycles to prevent nonductile failure of system components. Because operation with the sparger nozzle damage was outside the limits originally analyzed, the licensee requested General Electric-Hitachi (GEH) to provide an operability analysis of the degraded condition. GEH Report 004N6557, Revision 0, dated January 26, 2018, "Operability Assessment of the River Bend Station Feedwater Sparger Assembly in the January 2018 As-found Condition," stated, in part, "this evaluation does not account for Final			

Feedwater Temperature Reduction (FFWTR), Feedwater Heater Out-of-Service (FWH OOS) conditions, nor Single Loop Operation (SLO) operating conditions." Based on this analysis, the licensee's engineering department concluded that the recommended classification of this condition was OPERABLE-COMP MEAS (operable with compensatory measures), with the degraded/nonconforming condition being the holes in the feedwater sparger nozzles. Based on the results of this analysis, one of the operational restrictions/limitations stipulated in the licensee's ODMI was that, "RBS will not operate in Single Loop Operation (SLO)."

The ODMI developed by the licensee included two trigger points:

"Trigger Point 1:

An unexpected operational state below approximately 85 percent power in which the vessel wall-to-feedwater delta-T stabilizes at less than or equal to 154 degrees Fahrenheit (F), as detected by periodic monitoring during normal operations, OR due to a transient as defined above.

Trigger Point 2:

An unexpected operational state in which the vessel wall-to-feedwater delta-T stabilizes at greater than 154 degrees F, as detected by periodic monitoring during normal operations, OR due to a transient as defined above."

The ODMI failed to provide adequate guidance to the operators if they found themselves in any of the conditions that GEH had listed as not being evaluated for continued operation with the degraded condition. When reactor recirculation pump B failed to shift to fast speed at 9:46 a.m., the operators logged entry into Procedure GOP-004, "Single Loop Operations." The plant was in single loop operating conditions and remained there until 10:57 a.m. when the Mode switch was placed in shutdown. The ODMI failed to provide adequate guidance on the actions required if the plant entered any of the conditions that were not evaluated for the degraded sparger condition. In addition, the "Just In Time Training" given to the operators prior to taking the watch to commence power operations with the degraded condition did not address these issues either. As a result, rather than take prompt actions to place the plant in a known safe condition upon entry into single loop operations, the control room supervisor requested that GEH be contacted to determine if it was acceptable to remain in single loop operations.

Example 2: The evaluation of the damaged feedwater sparger nozzles 7 and 8 on sparger N4C identified that the damaged sections of the feedwater sparger nozzles had the potential to adversely affect the "B" narrow range level instrument. The damage on feedwater sparger N4C created unexpected feedwater flow paths in the reactor vessel during plant operation that had the potential to adversely affect the "B" variable leg reactor water level instruments. There were two potential impacts of this condition on indicated level from narrow range level instruments that tap off from the B variable leg. Flow from the holes in the feedwater sparger nozzle elbows could flow across the variable leg nozzle opening at AZ 200 degrees (B Leg), lowering the pressure on the variable leg side of the differential pressure measurements, or the flow from the sparger nozzle damage could directly impact the B variable leg, increasing the pressure on the variable leg side of the differential pressure measurements.

The narrow range RPV level instrumentation supports two reactor water level trips: low level (Level 3) and high level (Level 8). During a transient or accident event where the RPV water level is changing, the trip signal from the "B" narrow range instrument could be affected.

Based on the GE report, during a transient or accident event where the RPV water level is increasing, the high level (Level 8) trips (RPS trip and Feedwater Pump trip) in the affected channel may occur later than the trips in the unaffected channels. This may delay the overall Level 8 trips. For the Level 8 RPS trip, the margin between the calculated nominal trip setpoint and the technical specification allowable value is 0.77 inches. For the Level 3 RPS trip, the margin between the calculated nominal trip setpoint and the technical specification allowable value is 0.67 inches. An operability determination of the narrow range level instruments was performed under CR-RBS-2018-00633 CA-01.

The ODMI developed by the licensee included two trigger points:

Trigger Point 1:
Action: Refer to applicable SRs as specified by STP-000-0001, Att. 9.2
    Step 30 in STP-000-0001 not within 4 inches
    Step 71 in STP-000-0001 not within 6 inches
Notify the Duty Manager and the Ops Duty Manager

Trigger Point 2:
The magnitude of the B channel deviation is ≥ 1.5 inches in either direction from the average of the A, C and D channel average + 1.1 inches.
Notify the Duty Manager and the Engineering Duty Manager.

The ODMI implemented by the licensee allowed level indication deviation in the affected channel for the B21-LTN080 instruments to be monitored to ensure it remained within the allowable margin to ensure the technical specification trip limit is not exceeded. It stated in part that, "If the deviation exceeds a change of 1.5 inches from historical deviation of 1.1 inches above the average of the A, C, and D channels in either an increasing or decreasing direction, then condition will be evaluated by engineering. The monitored trigger point of +1.5 inches will provide adequate margin for both the Level 3 and Level 8 trips." However, if a 1.5-inch bias in the low direction would have been reached, two Technical Specification (TS) Allowable Values could have been exceeded (by 0.5 inches for TS Table 3.3.5.2-1, Function 2, "Reactor Core Isolation Cooling System Instrumentation," and by 0.49 inches for TS Table 3.3.5.2-1, Function 5, "Reactor Protection System Instrumentation"). The 1.5-inch bias in the low direction would have rendered the instrument inoperable based on 10 CFR 50.36(c)(2)(i), which states, "Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility." Since the limiting conditions for operations (LCOs) include Allowable Values (e.g., LCO 3.3.5.2 includes Table 3.3.5.2-1 which has Allowable Values for Functions 2 and 5), the Allowable Values are understood to be "the lowest functional capability or performance levels of equipment required for safe operation of the facility."

The licensee's technical specifications provide the following guidance: Surveillance Requirement 3.0.1, "Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO."

1.1 Definitions: “A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds within the necessary range and accuracy to known values of the parameter that the channel monitors…”

In addition, the TS Bases state, “SR 3.0.1 through SR 3.0.4 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated. The OPERABILITY of the RPS (Reactor Protection System) is dependent on the OPERABILITY of the individual instrumentation channel Functions specified in Table 3.3.1.1-1. Each Function must have a required number of OPERABLE channels [2 per RPS trip system for the vessel level function] per RPS trip system, with their setpoints within the specified Allowable Value, where appropriate. The actual setpoint is calibrated consistent with applicable setpoint methodology assumptions. Each channel must also respond within its assumed response time. Allowable Values are specified for each RPS Function specified in the Table. Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the actual setpoints do not exceed the Allowable Value between successive channel calibrations. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value.”

Process effects impact the establishment of the appropriate Nominal Trip Setpoint, which is determined by addressing all instrument channel uncertainties (including biases) and offsetting them from the Analytical Limit. The currently licensed Allowable Values are fixed within the technical specification tables. Nominal Trip Setpoints are established on the basis of a calculation that identifies all known uncertainties between the Analytical Limit and the Nominal Trip Setpoint. If a new, unaccounted-for process effect bias in the nonconservative direction is discovered, this effect needs to be reflected in the calculation of a new Nominal Trip Setpoint and a corresponding new Allowable Value. However, in this case, the licensee did not elect to pursue a license amendment or other process to change its currently licensed Allowable Value, nor did it ask for a temporary enforcement discretion. Therefore, with the new (unaccounted for) postulated process effect present, this has the effect of making the existing Nominal Trip Setpoint (calibrated value) offset in the nonconservative direction by the amount of the new postulated process effect (i.e., up to 1.5 inches), which reduces the margin between the “actual trip setpoint” and the existing licensed Allowable Value.

Therefore, to meet the River Bend technical specification requirement that a channel be considered “inoperable if its actual trip setpoint is not within its required Allowable Value” without changing the currently licensed Allowable Value, only approximately a 1/2-inch of the 1.5 inches of new postulated process effect can be accommodated between the existing calibrated setpoint and the (existing) licensed Allowable Value. Thus, the direction to notify engineering “only if the Rx vessel level indication bias had reached a value of 1.5 inches in either direction” was inadequate direction for the operating staff in order to ensure that the instruments remained operable.

Corrective Actions: The licensee corrected the condition by revising the ODMI to include adequate operator guidance and trigger points.

Corrective Action Reference: CR-RBS-2018-03148

<u>Performance Assessment</u>: Performance Deficiency: The failure to establish ODMI guidance per Procedure EN-OP-111 to address the compensatory measures implemented to maintain operability of the plant with degraded feedwater sparger nozzles was a performance deficiency. Screening: For Example 1, the performance deficiency was more than minor, and therefore a finding, because it was associated with the equipment reliability 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. Specifically, the licensee failed to provide adequate guidance to the operators for actions required if the plant inadvertently entered any of the unanalyzed conditions for continued operation with the degraded sparger. For Example 2, the performance deficiency was more than minor, and therefore a finding, because if left uncorrected it would have the potential to lead to a more significant safety concern. Specifically, the use of less conservative calculated values than the Allowable Values stated in the facility TS as a basis for establishing a threshold for operability of TS equipment could result in the inappropriate evaluation of actual degraded conditions that impact the ability of components to perform their required safety functions. Significance: The inspectors screened the finding in accordance with Inspection Manual Chapter 0609, Appendix A, “The Significance Determination Process for (SDP) for Findings At-Power.” Using Inspection Manual Chapter 0609, Appendix A, Exhibit 1, “Initiating Events Screening Questions,” the inspectors determined this finding was of very low safety significance (Green) because for Example 1, the finding would not result in exceeding the RCS leak rate for a small LOCA and could not have likely affected other systems used to mitigate a LOCA. For Example 2, it was not a design/qualification deficiency, did not represent a loss of system safety function, did not result in a loss of function of a single train for greater than its TS-allowable outage time, did not result in a loss of function of nonsafety-related risk-significant equipment and was not risk significant due to external events. In addition, no actual deviation of the “B” narrow range level instrument was observed during plant startup on February 9, 2018. Cross-cutting Aspect: This finding had a cross-cutting aspect of human performance, change management H.3: Leaders use a systematic process for evaluating and implementing change so that nuclear safety remains the overriding priority. Specifically, the licensee did not use a systematic process to develop and verify the adequacy of the ODMIs associated with the compensatory measures implemented for the degraded sparger.
<u>Enforcement</u>: Violation: Title 10 CFR Part 50, Appendix B, Criterion V, requires in part that, “activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances.” Licensee Procedure EN-OP-111, “Operational Decision-Making Issue (ODMI) Process,” Revision 16, an Appendix B quality-related procedure, provides instructions for developing guidance for plant operation with compensatory measures in place to maintain plant system operable with degraded conditions. Procedure EN-OP-111, step 5.2.4, states that Operational Decision-Making Considerations should ensure that a course of action is selected based upon a critical consideration of risks and potential consequences, as well as a thorough understanding of alternate solutions. The final decision should be a deliberate act, providing clear direction, trigger points, contingencies, and abort criteria. The Action Plans should provide clear

guidance in each ODMI which delineate actions to be taken when conditions escalate unexpectedly, conditions are outside the scope of the ODMI analysis, or actions are not able to be implemented. Actions that contain recommendations to "consider or evaluate" in response to triggers should be avoided. When such actions are used, a definite period to finish the evaluation or consideration should be provided.

Contrary to the above, prior to February 1, 2018, the licensee failed to ensure that the ODMIs provided a course of action based upon a critical consideration of risks and potential consequences, as well as a thorough understanding of alternate solutions; and that the final decision was a deliberate act providing clear direction, trigger points, contingencies, and abort criteria. Specifically, the licensee failed to develop adequate guidance for the operators to maintain safe operation of the plant with compensatory measures in place for degraded feedwater sparger nozzles. The action plans failed to provide clear guidance in each ODMI to delineate actions to be taken when conditions escalate unexpectedly; instead, the actions specified directed the operators to consult with offsite contractors regarding the acceptability of allowing the plant to remain in operation with given conditions.

Disposition: This violation is being treated as a non-cited violation, consistent with Section 2.3.2.a of the NRC Enforcement Policy.

Failure to Establish Procedural Guidance for Determining Core Flow During Unanticipated Single Loop Operations			
Cornerstone	Significance	Cross-cutting Aspect	Report Section
Initiating Events	Green NCV 05000458/2018012-03 Closed	[P.3] – Problem Identification and Resolution, Resolution	71153 – Follow-up of Events and Notices of Enforcement Discretion
The inspectors reviewed a self-revealed, non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures and Drawings,” for the licensee’s failure to establish appropriate instructions in the abnormal operating procedure for thermal hydraulic instabilities. Specifically, the procedural step for determining core flow when in single loop operations at low power did not provide appropriate instructions to operators. As a result, station personnel could not conclusively determine core flow and inserted a manual reactor scram.			

<u>Description</u>:

On February 1, 2018, with the unit in Mode 1 at approximately 27 percent power, reactor recirculation pump B unexpectedly tripped during an upshift in the speed of the pump. As a result, the reactor was in a single loop configuration with the recirculation pump A running in fast speed and the recirculation pump B not running. Operators entered Abnormal Operating Procedure AOP-0024, “Thermal Hydraulic Instability Controls,” Revision 30, as a result of the unplanned entry into single loop operations. Step 5.8 of this procedure directed operators to determine core flow and enter the General Operating Procedure GOP-004, for single loop operations. Step 5.8 also instructed operators to determine core flow using process computer point B33NA01V when in a configuration with one recirculation pump in fast speed and one recirculation pump off. Control room operators observed the value of this data point as 13.9 Mlbm/hr. The operators concluded that this value was not valid since the indicated flow was much lower than expected with one recirculation pump running in fast speed. The operators then observed a value of 27.3 Mlbm/hr core flow using the ERIS data point for B33NA01V. This value appeared to be a valid number based on the single loop operation power/flow map contained in AOP-0024, Attachment 2. Normal data points are displayed in ERIS with a white text, but control room operators observed the ERIS data point displayed in a magenta color. Additionally, the word “suspect” appeared adjacent to the data point for core flow. Control room operators contacted information technology personnel and attempted to understand the magenta color and “suspect” information associated with the core flow data point. Concurrently, operators attempted to validate core flow using alternate means but were unsuccessful as the alternate indications did not provide accurate core flow readings at low reactor power when in a single loop configuration. After approximately one hour spent seeking to understand the unfamiliar indication associated with B33NA01V, control room operators conducted a brief and made the decision to shut down the unit due to the uncertainties associated with the core flow data point. Following plant shutdown and subsequent troubleshooting and investigation, licensee personnel concluded that the magenta text and “suspect” note associated with ERIS B33NA01V was an expected system response. Below approximately 40 percent core flow, the plant process computer shifts the calculation method from the primary means of calculating core flow using the sum of jet pump flows to an alternate process that uses core plate differential pressure. As a result of shifting to the alternate calculation of core flow, data point ERIS B33NA01V was programmed to turn magenta in color and display “suspect” to alert operators that the method of calculating core flow had changed.

The inspectors reviewed Condition Report CR-RBS-2012-07759. This condition report was generated by operations department personnel on December 19, 2012 and identified that ERIS point B33NA01V indicated “suspect” and was not available for use. The condition report also stated that this data point was needed for determining core flow when the plant configuration consisted of one recirculation pump running in fast speed and another recirculation pump was off. The inspectors confirmed that this condition report was generated during a single loop plant configuration that was the result of an unanticipated reactor recirculation pump A trip on December 19, 2012. The condition report corrective actions explained the reason for the “suspect” reading of ERIS point B33NA01V. No corrective actions were generated to address AOP-0024, which directs licensed operators to validate core flow in single loop operations. Additionally, no corrective actions were generated to validate plant simulator response to unanticipated single loop operations.

Corrective Actions: After this information was disseminated to licensed operators, the licensee implemented procedural changes to AOP-0024 that provided amplifying information regarding B33NA01V validated core flow. Specifically, the licensee revised the procedure on February 7, 2018, to 1) direct operators to determine core flow using ERIS data point B33NA01V during single loop operations when core flow is below 40 percent and 2) provide clear guidance regarding expected system response of the process computer data points during abnormal flow configurations.

Corrective Action Reference: CR-RBS-2018-00776

Performance Assessment:

Performance Deficiency: The failure to establish appropriate guidance to determine core flow during single loop operations in quality-related abnormal operating procedure AOP-0024, “Thermal Hydraulic Instability Controls,” Revision 30, was a performance deficiency.

Screening: The performance deficiency was more than minor, and therefore a finding, because it was associated with the procedure quality attribute of the Initiating Events Cornerstone and adversely affected the cornerstone objective to limit the likelihood of events that upset plant stability. Specifically, the failure to understand core flow data indicated by plant process computer point B33NA01V and ERIS data point B33NA01V resulted in confusion and the ultimate decision to insert a manual reactor scram. Significance: The inspectors screened the finding in accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process for (SDP) for Findings At-Power." Using Inspection Manual Chapter 0609, Appendix A, Exhibit 1, "Initiating Events Screening Questions," the inspectors determined this finding is of very low safety significance (Green) because the finding did not cause a reactor trip and the loss of mitigation equipment relied upon to transition the plant from the onset of the trip to a stable shutdown condition. Cross-cutting Aspect: This finding has a cross-cutting aspect in the area of problem identification and resolution, resolution, because the licensee failed to take effective corrective actions to address issues in a timely manner commensurate with their safety significance. Specifically, the station failed to implement procedure changes to AOP-0024 after discovering similar confusing indications associated with B33NA01V on December 19, 2012.
<u>Enforcement</u>: Violation: Title 10 CFR Part 50, Appendix B, Criterion V, requires in part that, "activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances." Contrary to the above, prior to February 7, 2018, the licensee failed to provide a procedure of a type appropriate to the circumstances for an activity affecting quality. Specifically, AOP-0024, "Thermal Hydraulic Stability Controls," a quality-related procedure, was not appropriate to the circumstances. AOP-0024 did not provide accurate and adequate instruction to operators to determine core flow during single loop operations. The licensee restored compliance by revising AOP-0024 to include accurate and adequate guidance to determine core flow during unanticipated single loop operations. Disposition: This violation is being treated as a non-cited violation consistent with Section 2.3.2.a of the NRC Enforcement Policy.

Failure to Conduct Adequate Transient Snap Shot Assessment Following Recirculation Pump Trip			
Cornerstone	Significance	Cross-cutting Aspect	Report Section
Initiating Events	Green FIN 05000458/2018012-01 Closed	None	71152 – Problem Identification and Resolution
The inspectors identified a Green finding for the licensee's failure to adequately validate simulator response during a transient snap shot assessment following an unexpected trip of reactor recirculation pump A on December 19, 2012.			

Description:

On December 19, 2012, with the plant operating at 100 percent power, reactor recirculation pump A unexpectedly tripped off. As a result, the plant configuration consisted of one recirculation pump running in fast speed and the other recirculation pump secured. During this single loop configuration, station personnel identified that emergency response information system (ERIS) point B33NA01V indicated “suspect” and was not available for use. The station documented this condition in Condition Report CR-RBS-2012-07759.

On February 1, 2018, with the unit in Mode 1 at approximately 27 percent power, reactor recirculation pump B unexpectedly tripped during an upshift in the speed of the pump. As a result, the reactor was in a single loop configuration with the recirculation pump A running in fast speed and the recirculation pump B not running. Operators entered abnormal operating procedure AOP-0024, “Thermal Hydraulic Instability Controls,” Revision 30, as a result of the unplanned entry into single loop operations. Step 5.8 of this procedure directed operators to determine core flow and enter general operating procedure GOP-004, “Single Loop Operations.” Step 5.8 also instructed operators to determine core flow using process computer point B33NA01V (which can be observed in both ERIS and the plant process computer) when in a configuration with one recirculation pump in fast speed and one recirculation pump off. Control room operators observed the value of this data point as 13.9 million pounds mass per hour (Mlbm/hr) of flow through the reactor core. The operators concluded that this value was not valid since the indicated flow was much lower than expected with one recirculation pump running in fast speed. The operators then observed a value of 27.3Mlbm/hr core flow using the ERIS data point for B33NA01V. This value appeared to be a valid number based on the single loop operation power/flow map contained in AOP-0024, Attachment 2. Normal data points on ERIS are displayed with a white text, but control room operators observed the ERIS data point displayed in a magenta color. Additionally, the word “suspect” appeared adjacent to the data point for core flow. Control room operators contacted information technology personnel and attempted to understand the magenta color and “suspect” information associated with the core flow data point. Concurrently, operators attempted to validate core flow using alternate means but were unsuccessful, as the alternate indications did not provide accurate core flow readings at low reactor power when in a single loop configuration. After approximately one hour spent seeking to understand the unfamiliar indication associated with B33NA01V, control room operators conducted a brief and made the decision to shut down the unit due to the uncertainties associated with the core flow data point. Following plant shutdown and subsequent troubleshooting and investigation, licensee personnel concluded that the magenta text and “suspect” note associated with ERIS B33NA01V was an expected system response. Below approximately 40 percent core flow, the plant process computer shifts the calculation method from the primary means of calculating core flow using the sum of jet pump flows to an alternate process that uses core plate differential pressure. As a result of shifting to the alternate calculation of core flow, data point ERIS B33NA01V was programmed to turn magenta in color and display “suspect” to alert operators that the method of calculating core flow had changed. After this information was disseminated to licensed operators, the licensee implemented procedural changes to AOP-0024 that provided amplifying information regarding B33NA01V validated core flow. Specifically, the licensee revised the procedure on February 7, 2018, to provide clear guidance regarding expected system response of the process computer data points during abnormal flow configurations.

The inspectors compared the actual plant response to the simulator response for the trip of a recirculation pump while at low power. The actual conditions in the main control room during the event on February 1, 2018, resulted in ERIS point B33NA01V indicating the correct flow (27.3Mlbm/hr), but the data point turned magenta in color and displayed the warning label "suspect." This was later determined by information technology personnel to be the correct response and data display, and was the result of the core flow calculation methodology swapping from the primary method (jet pump flow) to the alternate method (core plate differential pressure).

In the simulator, the inspectors determined that ERIS point B33NA01V provided erratic indications of core flow following a simulated trip of the recirculation pump B from an initial condition of approximately 25 percent. The indicated flow varied, and ultimately stabilized at approximately 10Mlbm/hr, which is less than half of the expected indication. Additionally, B33NA01V did not change to a magenta color, and it did not display the word "suspect." The inspectors determined that ERIS B33NA01V was programmed to calculate core flow using the sum of jet pump flows at all power levels. As a result, the displayed value was inaccurate below 40 percent core flow, and the data point was not programmed to turn magenta or indicate "suspect" since no swap to a backup means of calculation below 40 percent core flow was modelled.

The inspectors reviewed procedure EN-OP-117, "Operations Assessments," Version 4, Section 5.4, which states that "transient snap-shot assessments are performed whenever a plant transient occurs." A plant transient is defined in section 5.4[2] as including "any turbine generator power change in excess of 10 percent of rated power in less than one minute other than a momentary spike due to a grid disturbance or a manually initiated runback." The inspectors concluded that the recirculation pump A trip on December 19, 2012, met the definition of a transient. EN-OP-117, Attachment 9.2, "Transient Snap Shot Assessment Documentation Form," Objective 7, discusses the training preparation aspect of the assessment. Specifically, the transient snap-shot assessment is performed in order to validate that the simulator accurately represented the plant characteristics of the transient. The licensee provided a Post-Event Simulator Test report that was run on February 14, 2013. The report concluded that the simulator response matched the parameters observed in the plant. The inspectors determined that although the snap-shot assessment was performed, station personnel did not validate that ERIS B33NA01V (validated core flow) provided operators with the same indications seen by operators in the plant during a recirculation pump trip.

The inspectors determined that no condition report or simulator deficiency report was generated to document the discrepancy between the plant and the simulator for displaying ERIS B33NA01V. The simulator ERIS B33NA01V core flow indication did not display the correct value for core flow and also did not indicate "suspect" or turn magenta. The inspectors reviewed training documentation to determine why this discrepancy was not observed during continuing simulator training scenarios. The inspectors concluded that this discrepancy was not documented because the station did not conduct training on abnormal single loop operations during low power operations. The inspectors reviewed industry standards and guidelines for simulator training and determined that the station is required to periodically conduct training on abnormal events that occur during low power operations.

Corrective Actions: The station documented the core flow indication simulator deficiency in a deficiency report and generated actions to incorporate the discrepancy into future licensed operator training sessions.

Corrective Action Reference: CR-RBS-2018-03145

Performance Assessment: Performance Deficiency: The licensee's failure to validate core flow in the simulator during a transient snap shot assessment following the trip of the reactor recirculation pump A on December 19, 2012, was a performance deficiency. Screening: The performance deficiency was more than minor, and therefore a finding, because it was associated with the human performance attribute of the Initiating Events Cornerstone and adversely affected the cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, the failure to validate simulator fidelity following a plant transient prevented the licensee from identifying simulator model discrepancies when determining core flow during low power, single loop operations. Significance: The inspectors screened the finding in accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process for Findings At-Power." The finding was determined to be of very low safety significance (Green) because the finding did not contribute to both the likelihood of a reactor trip and the likelihood that mitigating equipment would not be available. Cross-cutting Aspect: No cross cutting aspect was assigned because the performance deficiency is not indicative of current licensee performance.
Enforcement: Inspectors did not identify a violation of regulatory requirements associated with this finding.

Failure to Submit a Licensee Event Report for a Manual Scram			
Cornerstone	Significance	Cross-cutting Aspect	Report Section
None	SLIV NCV 05000458/2018012-04 Closed	None	71153 – Follow-up of Events and Notices of Enforcement Discretion
The inspectors identified a Severity Level IV non-cited violation of 10 CFR 50.73, "Licensee Event Report System," for the licensee's failure to submit a required licensee event report (LER). Specifically, on February 1, 2018, after an unexpected trip of the recirculation pump B, the licensee initiated a manual scram of the reactor that was not part of a preplanned sequence and failed to submit an LER within 60 days.			
Description: At approximately 9:46 a.m. on February 1, 2018, with the unit operating at approximately 27 percent power, the recirculation pump B unexpectedly tripped during an attempted transfer from slow to fast speed. The licensee promptly entered AOP-0024, "Thermal Hydraulic Instability," and GOP-0004, "Single Loop Operation." Note 5.8 of AOP-0024 and Precaution 3.6 of GOP-0004 instruct the licensee to use process computer point B33NA01V to determine core flow while in single loop operation. The plant process computer (PPC) and emergency response information system (ERIS) readouts showed conflicting indications for this computer point, with the PPC showing approximately 13,900 Mlbm/hr of flow and ERIS showing approximately 26,000 Mlbm/hr of flow.			

Step 5.1 of AOP-0024 instructs the licensee to determine where on the power-to-flow map the plant is operating. If the plant is operating in the restricted region, the procedure states to exit that region by lowering power or raising flow. If the plant is operating in the exclusion region, the procedure states to verify that a scram has occurred. The indicated PPC value for core flow put the plant in an unanalyzed region of the power-to-flow map, with less flow than the minimum amount of flow that defines any region, including the exclusion region. The indicated ERIS value put the plant in the restricted region, just above the boundary that delineates the restricted region from the monitoring region.

The licensee initially believed the ERIS value to be the correct value; however, this value was accompanied by a magenta "suspect" note on the ERIS screen, which caused the licensee to question its validity. In an effort to determine the true value of core flow, the licensee performed a manual calculation using other known inputs. The licensee performed this calculation incorrectly and wrongly corroborated the PPC value as the correct value. Given the inability to establish that the plant was operating in any allowed region of the power-to-flow map, the licensee made the decision to manually actuate the reactor protection system (RPS) by taking the reactor mode switch to shutdown.

During the investigation after the scram, the licensee determined that the ERIS value was, in fact, a valid indication of core flow at the time of the event. Operators had not been adequately trained on the meaning of the magenta "suspect" indication, and were therefore unable to determine the implications of the indications on the validity of the data point.

Pursuant to the requirements of 10 CFR 50.72(b)(3)(iv), the licensee reported the scram event to the NRC at 1:23 p.m. as an event that resulted in an actuation of the RPS. On March 23, 2018, the licensee retracted the report claiming the actuation was part of a pre-planned sequence during testing or reactor operation. The inspectors concluded that this retraction was inappropriate and that the event was reportable for the reasons provided below.

The inspectors reviewed NUREG-1022, "Event Report Guidelines 10 CFR 50.72 and 50.73," revision 3, which provides the following guidance: "Actuations that need not be reported are those initiated for reasons other than to mitigate the consequences of an event (e.g., at the discretion of the licensee as part of a preplanned procedure)." In the case of the February 1, 2018, River Bend scram event, the inspectors determined that the manual RPS actuation was initiated in order to mitigate the consequences (i.e., uncertainty as to the condition of the plant with respect to core flow and power-to-flow considerations) of an event (i.e., the unexpected loss of a reactor recirculation pump).

NUREG-1022 also provides an example of a reportable manual scram that was event driven and not part of a preplanned sequence during testing or reactor operation:

> "At a BWR, both recirculation pumps tripped as a result of a breaker problem. This placed the plant in a condition in which BWRs are typically scrammed to avoid potential power/flow oscillations. At this plant, for this condition, a written off-normal procedure required the plant operations staff to scram the reactor. The plant staff performed a reactor scram, which was uncomplicated. This event is reportable as a manual RPS actuation. Even though the reactor scram was in response to an existing written procedure, this event does not involve a preplanned sequence because the loss of recirculation pumps and the resultant off-normal procedure entry were event driven, not preplanned. Both an ENS notification and an LER are required. In this

case, the licensee initially retracted the ENS notification, believing that the event was not reportable. After staff review and further discussion, it was agreed that the event is reportable for the reasons discussed above."

As with the scram in the above example, the scram that occurred at River Bend Station was not part of a preplanned sequence during testing or reactor operation, but was instead an event driven response to a series of unplanned and unexpected adverse occurrences in the plant. These occurrences included: a trip of the recirculation pump B, entry into an abnormal operating procedure for thermal hydraulic instability, an inability to determine core flow and location on the power-to-flow map in accordance with that procedure, a realization that the PPC indication of core flow put the plant outside of any allowed operating region of the power-to-flow map, an incorrect manual calculation that wrongly corroborated the accuracy of the PPC indication, and the presence of a poorly understood "suspect" indication that appeared to undermine the validity of the ERIS flow indication. These adverse occurrences generated uncertainty as to the status of reactor safety. The subsequent decision to perform a manual reactor scram was consistent with general instruction provided in EN-OP-115, "Conduct of Operations," which states: "do not hesitate to reduce power or perform an immediate reactor shutdown when reactor safety is uncertain." As with the scram in the above example, the February 1, 2018, River Bend scram also involved entry into an off-normal procedure due to an unexpected plant equipment malfunction that resulted in the potential for the plant to be in an undesired condition with respect to power-to-flow considerations.

The senior resident inspector was present in the control room during the events and was able to confirm that the shutdown was event driven rather than preplanned. At 10:55 a.m., the control room briefed that because PPC and ERIS showed conflicting indications of core flow with ERIS indicating "suspect," the mode switch was going to be placed in shutdown. At 10:57 a.m., roughly two minutes after the brief was completed, the reactor operator scrammed the reactor, and the following station log entry was made: "MCR [main control room] announces placing plant in shut down due to inability to regulate recirculation flow." If the reactor shutdown had been preplanned, it would not have proceeded at this accelerated pace. Rather, the licensee would have worked through the relevant steps of the applicable shutdown procedure, GOP-0004, "Single Loop Operation," scramming the reactor only after those steps had been completed and signed for. Upon review of the copy of GOP-0004 that was in use by the operators on February 1, 2018, the inspectors noted that no steps of Attachment 3, "Shutdown from Single Loop Operation," were marked as completed, and the attachment was not signed off as being initiated or completed. The deviation from normal practice was appropriate because the scram was not being initiated as part of a preplanned sequence. It was instead being initiated in response to the unanticipated emergence of a safety concern.

Corrective Actions: The licensee documented the violation in the corrective action program and generated corrective actions to review reportability requirements.

Corrective Action Reference(s): CR-RBS-2018-03953

Performance Assessment:

Performance Deficiency: The failure to submit a required licensee event report was a performance deficiency.

Screening: The performance deficiency was evaluated in accordance with the reactor oversight process and was determined to be minor because it could not be reasonably viewed as a precursor to a significant event, would not have the potential to lead to a more significant safety concern, does not relate to a performance indicator that would have caused the performance indicator to exceed a threshold, and did not adversely affect a cornerstone objective. The performance deficiency was evaluated in accordance with the traditional enforcement process because it impacted the ability of the NRC to perform its regulatory oversight function. Significance: Using example 6.9.d.9 from the NRC Enforcement Policy, the violation was determined to be a Severity Level IV violation. Cross-cutting Aspect: Because the violation was dispositioned using the traditional enforcement process, no cross-cutting aspect was assigned.
Enforcement: Violation: 10 CFR 50.73(a)(1) requires, in part, that the licensee shall submit a Licensee Event Report (LER) for any event of the type described in this paragraph within 60 days after the discovery of the event. 10 CFR 50.73(a)(2)(iv)(A) requires, in part, that the licensee shall report any event or condition that resulted in manual actuation of the reactor protection system (RPS) except when the actuation resulted from and was part of a pre-planned sequence during testing or reactor operation. Contrary to the above, on April 2, 2018, the licensee failed to submit an LER within 60 days after the discovery of an event or condition that resulted in manual actuation of the RPS that did not result from and that was not a part of a pre-planned sequence during testing or reactor operation. Specifically, the licensee failed to submit an LER within 60 days of a manual reactor scram that occurred on February 1, 2018. Disposition: Because this SLIV violation was neither repetitive nor willful, and because it was entered into the licensee's corrective action program as Condition Report CR-RBS-2018-03953, it is being treated as a non-cited violation consistent with Section 2.3.2.a of the NRC Enforcement Policy.

EXIT MEETINGS AND DEBRIEFS

The inspectors verified no proprietary information was retained or documented in this report.

On May 31, 2018, and on July 16, 2018, the inspectors presented the inspection results to Mr. W. Maguire, Site Vice President, and other members of the licensee staff.

DOCUMENTS REVIEWED

71111.15—Operability Determinations and Functionality Assessments

Procedures

Number	Title	Revision
EN-OE-100	Operating Experience Program	12 & 13
STP-051-4206	(RPS Bypassed) RPS/RHR Reactor Vessel Level-Low, Level 3, High, Level 8, Channel Calibration and Logic System Functional Test (B21-N680B, B21-N683B, B21-N080B)	305
STP-051-4227	ECCS/RCIC Actuation Ads Trip System “B” Reactor Vessel Water Level Low, Level 3/High, Level 8 Channel Calibration, and Logic System Functional Test (B21-N095B, B21-N695B, B21-N693B)	20
STP-501-4202	FWS/MAIN Turbine Trip System – Reactor Vessel Water Level – High Level 8, Channel Calibration and LSFT (C33-N004B, C33-K624B, C33-R606B, C33-K650-3)	15
G13.18.6.1.B21	Reactor Vessel Water Level – Low, Level 3 Trip Function	3
G13.18.6.1.B21*003	Reactor Vessel Water Level – Low, Level 3 Trip Function	3
G13.18.6.1.B21*010	Reactor Vessel Water Level – Low, Level 8 Narrow Range	0, 1, 2, & 3
MCP-IC-501-4202	FWS/FEED Pump Trip System (MSO) – Reactor Vessel Water Level – High Level 8, Loop Calibration (C33-LTN006B, C33-ESN606B)	0

71111.18—Plant Modifications

Condition Reports (CR-RBS-)

CR-RBS-2014-05194	CR-RBS-2014-06685	CR-RBS-2014-06691	CR-RBS-2015-03253
CR-RBS-2015-03983	CR-RBS-2015-04065	CR-RBS-2015-04117	CR-RBS-2015-08476
CR-RBS-2015-08515	CR-RBS-2016-00791	CR-RBS-2016-00893	CR-RBS-2016-00893
CR-RBS-2016-04351	CR-RBS-2016-04353	CR-RBS-2017-02828	OE-NOE-2004-00008
OE-NOE-2004-00084			

Engineering Changes

Number	Title	Revision
EC-75588	Accept As-Is Evaluation for Remainder of Cycle 20: Sparger N4C Nozzles 7 and 8 Damaged	0 & 1

Procedures

Number	Title	Revision
OSP-0053	Emergency and Transient Response Support Procedure	20-25
STP-000-0001	Daily Operating Logs	082
DBR-0035279	GEH Comment Resolution Form	0
4221.110-000-043	Operability Assessment of the River Bend Station Feedwater Sparger Assembly in the January 2018 As-Found Condition	0

71152 – Problem Identification and Resolution

Condition Reports (CR-RBS-)

CR-RBS-2018-00358	CR-RBS-2018-00613	CR-RBS-2018-00633	CR-RBS-2018-00733
CR-RBS-2018-00895	CR-RBS-2018-00294	OE-NOE-2004-00008	OE-NOE-2004-00084

Engineering Changes

Number	Title	Revision
EC-75663	Loose Parts Evaluation for Material Lost From Feedwater Spargers Identified During PO-18-01 Foreign Material FME LPA-000	0

Miscellaneous Documents

Number	Title	Revision/Date
	OSRC Meeting 2018-0001 Minutes	
	OSRC Meeting 2018-0002 Minutes	
	Action Item OE33308-20110507-A2-RBS-001	
CNR RBS PO-18-01-01	Foreign Material Customer Notification Report	0
ECH-NE-17-00039	River Bend MOC-20a Fuel Inspection Plan	0
NEDC-31336P-A	General Electric Instrument Setpoint Methodology	0
NEDE-21821-A	Boiling Water Reactor Feedwater Nozzle/Sparger Final Report	0
NEI 96-07	Guidelines for 10 CFR 50.59 Implementation	1
OE33308-20110507	Sampling Probe Found in Feedwater Sparger	August 17, 2011

Miscellaneous Documents

Number	Title	Revision/Date
PO 18-01	BOP Foreign Material Inspection Report	
RBS-ER-99-0539	Engineering Response Associated with Loose Part in the Feedwater System	0

Procedures

Number	Title	Revision
AOP-0001	Reactor Scram	37
AOP-0024	Thermal Hydraulic Stability Controls	30, 31, & 32
EN-NF-102	Corporate Fuel Reliability	6
EN-OP-104	Operability Determination Process	14
EN-OP-111	Operational Decision Making Issue Process	15
EN-OP-117	Operations Assessments	4
EOP-0001	Emergency Operating Procedure – RPV Control	28
GOP-0001	Plant Startup	99
GOP-0002	Power Decrease/Plant Shutdown	78
GOP-0003	Scram Recovery	31
GOP-0004	Single Loop Operation	25
OE-100	Operating Experience Program	1
R-PL-012	Corrective Action Program	1
STP-000-0001	Daily Operating Logs	082

Work Order

52599498

71153—Follow-up of Events and Notices of Enforcement Discretion

Procedures

Number	Title	Revision
EN-OP-115	Conduct of Operations	23
GOP-0004	Single Loop Operation	23

Condition Reports (CR-RBS-)

2018-03149	2018-03953

RIVER BEND STATION – REVISED NRC BASELINE INSPECTION REPORT 05000458/2018012 – DATED JUNE 25, 2019

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ADAMS ACCESSION NUMBER: ML19179A153

☒ SUNSI Review By: **CHY**	ADAMS: ☒ Yes ☐ No	☐ Non-Publicly Available ☒ Publicly Available	☒ Non-Sensitive ☐ Sensitive	Keyword: NRC-002

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