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

July 7, 2015

EA-15-043

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 SPECIAL INSPECTION REPORT 05000458/2015009; PRELIMINARY WHITE FINDING

Dear Mr. Olson:

On June 29, 2015, the U.S. Nuclear Regulatory Commission (NRC) completed a Special Inspection at the River Bend Station to evaluate the facts and circumstances surrounding an unplanned reactor trip. Based upon the risk and deterministic criteria specified in NRC Management Directive 8.3, "NRC Incident Investigation Program," the NRC initiated a Special Inspection in accordance with Inspection Procedure 93812, "Special Inspection." The basis for initiating the special inspection and the focus areas for review are detailed in the Special Inspection Charter (Attachment 2). The NRC determined the need to perform a Special Inspection on January 15, 2015, and the onsite inspection started on January 26, 2015. The enclosed report documents the inspection findings that were discussed on May 21 and June 29, 2015, with you and members of your staff. The team documented the results of this inspection in the enclosed inspection report.

The enclosed inspection report documents a finding that has preliminarily been determined to be White, a finding with low to moderate safety significance that may require additional NRC inspections, regulatory actions, and oversight. The team identified an apparent violation for failure to maintain the simulator so it would accurately reproduce the operating characteristics of the facility. Specifically, the River Bend Station simulator failed to accurately model feedwater flow and reactor vessel level response following a scram, failed to provide the correct alarm response for loss of a reactor protection system motor generator set, and failed to correctly model the operation of the startup feedwater regulating valve. As a result of the simulator deficiencies, operations personnel were presented with additional challenges to control the plant and maintain plant parameters following a reactor scram on December 25, 2014. Because actions have been taken to initiate discrepancy reports, to investigate and resolve the potential fidelity issues and to provide training to operations personnel, the simulator deficiencies do not represent a continuing safety concern. The NRC assessed this finding using the best available information, and Manual Chapter 0609, "Significance Determination Process." The basis for the NRC's preliminary significance determination is described in the enclosed report. The finding is also an apparent violation of NRC requirements and is being considered for escalated enforcement action in accordance with the Enforcement Policy, which can be found on the

NRC's website at http://www.nrc.gov/about-nrc/regulatory/enforcement/enforce-pol.html. The NRC will inform you in writing when the final significance has been determined.

Before we make a final decision on this matter, we are providing you with an opportunity to (1) attend a Regulatory Conference where you can present your perspective on the facts and assumptions used to arrive at the finding and assess its significance, or (2) submit your position on the finding to the NRC in writing. If you request a Regulatory Conference, it should be held within 30 days of your receipt of this letter. We encourage you to submit supporting documentation at least one week prior to the conference in an effort to make the conference more efficient and effective. The focus of the Regulatory Conference is to discuss the significance of the finding and not necessarily the root cause(s) or corrective action(s) associated with the finding. If you choose to attend a Regulatory Conference, it will be open for public observation. The NRC will issue a public meeting notice and press release to announce the conference. If you decide to submit only a written response, it should be sent to the NRC within 30 days of your receipt of this letter. If you decline to request a Regulatory Conference or to submit a written response, you relinquish your right to appeal the NRC's final significance determination, in that by not choosing an option, you fail to meet the appeal requirements stated in the Prerequisites and Limitations sections of Attachment 2, "Process for Appealing NRC Characterization of Inspection Findings (SDP Appeal Process)," of NRC Inspection Manual Chapter 0609.

Please contact Greg Warnick at (817) 200-1144, and in writing, within 10 days from the issue date of this letter to notify us of your intentions. If we have not heard from you within 10 days, we will continue with our final significance determination and enforcement decision. The final resolution of this matter will be conveyed in separate correspondence.

Because the NRC has not made a final determination in this matter, no Notice of Violation is being issued for this inspection finding at this time. In addition, please be advised that the number and characterization of the apparent violation described in the enclosed inspection report may change based on further NRC review.

In addition, the NRC inspectors documented four findings of very low safety significance (Green) in this report. Three of these findings were determined to involve violations of NRC requirements. The NRC is treating these violations as non-cited violations consistent with Section 2.3.2.a of the Enforcement Policy.

If you contest the violations or significance of these 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 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 Regional Administrator, Region IV; and the NRC resident inspector at the 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's 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/

Troy W. Pruett
Director
Division of Reactor Projects

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

Enclosure:
Inspection Report 05000458/2015009
w/ Attachments:
1. Supplemental Information
2. Special Inspection Charter

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's 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,

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Troy W. Pruett
Director
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Docket No. 50-458
License No. NPF-47

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Letter to Eric Olson from Troy Pruett dated July 7, 2015.

SUBJECT: RIVER BEND STATION – NRC SPECIAL INSPECTION REPORT 05000458/2015009; PRELIMINARY WHITE FINDING

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Deputy Regional Administrator (Kriss.Kennedy@nrc.gov)
DRP Director (Troy.Pruett@nrc.gov)
DRP Deputy Director (Ryan.Lantz@nrc.gov)
DRS Director (Anton.Vegel@nrc.gov)
DRS Deputy Director (Jeff.Clark@nrc.gov)
Senior Resident Inspector (Jeffrey.Sowa@nrc.gov)
Resident Inspector (Andy.Barrett@nrc.gov)
RBS Administrative Assistant (Lisa.Day@nrc.gov)
Branch Chief, DRP/C (Greg.Warnick@nrc.gov)
Senior Project Engineer (Ray.Azua@nrc.gov)
Project Engineer (Michael.Stafford@nrc.gov)
Project Engineer (Paul.Nizov@nrc.gov)
Public Affairs Officer (Victor.Dricks@nrc.gov)
Public Affairs Officer (Lara.Uselding@nrc.gov)
RIV RSLO (Bill.Maier@nrc.gov)
Project Manager (Alan.Wang@nrc.gov)
Team Leader, DRS/TSS (Don.Allen@nrc.gov)
RITS Coordinator (Marisa.Herrera@nrc.gov)
ACES (R4Enforcement.Resource@nrc.gov)
Regional Counsel (Karla.Fuller@nrc.gov)
Technical Support Assistant (Loretta.Williams@nrc.gov)
Congressional Affairs Officer (Jenny.Weil@nrc.gov)
RIV Congressional Affairs Officer (Angel.Moreno@nrc.gov)
RIV/ETA: OEDO (Michael.Waters@nrc.gov)
Senior Staff Engineer, TSB (Kent.Howard@nrc.gov)
Enforcement Specialist, OE/EB (Robert.Carpenter@nrc.gov)
Senior Enforcement Specialist, OE/EB (John.Wray@nrc.gov)
Branch Chief, OE (Nick.Hilton@nrc.gov)
Enforcement Coordinator, NRR/DIRS/IPAB/IAET (Lauren.Casey@nrc.gov)
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U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Docket:	05000458
License:	NPF-47
Report:	05000458/2015009
Licensee:	Entergy Operations, Inc.
Facility:	River Bend Station, Unit 1
Location:	5485 U.S. Highway 61N St. Francisville, LA 70775
Dates:	January 26 through June 29, 2015
Inspectors:	T. Hartman, Senior Resident Inspector D. Bradley, Resident Inspector J. Drake, Senior Reactor Inspector
Approved By:	T. Pruett, Director Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000458/2015009; 01/26/2015 – 06/29/2015; River Bend Station; Special inspection for the scram with complications that occurred on December 25, 2014.

The report covered one week of onsite inspection and in-office review through June 29, 2015, by inspectors from the NRC's Region IV office. One preliminary White apparent violation, three Green non-cited violations, and one Green finding were identified. The significance of most findings is indicated by their color (Green, White, Yellow, or Red) using Inspection Manual Chapter 0609, "Significance Determination Process." Findings for which the significance determination process does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

Cornerstone: Initiating Events

- Green. The team reviewed a self-revealing, non-cited violation of Technical Specification 5.4.1.a for the licensee's failure to establish adequate procedures to properly preplan and perform maintenance that affected the performance of the B reactor protection system motor generator set. Specifically, due to inadequate procedures for troubleshooting on the B reactor protection system motor generator set, the licensee failed to identify a degraded capacitor that caused the B reactor protection system motor generator set output breaker to trip, which resulted in a reactor scram. The licensee entered this issue into their corrective action program as Condition Report CR-RBS-2014-06605 and replaced the degraded field flash card capacitor.

  This performance deficiency is more than minor, and therefore a finding, because it is 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 and challenge critical safety functions during shutdown as well as power operations. Using Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," Exhibit 1, "Initiating Event Screening Questions," this finding is determined to have a very low safety significance (Green) because the transient initiator did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions would not have been available. This finding has an evaluation cross-cutting aspect within the problem identification and resolution area because the licensee failed to thoroughly evaluate this issue to ensure that the resolution addressed the cause commensurate with its safety significance. Specifically, the licensee failed to thoroughly evaluate the condition of the field flash card to ensure that the cause of the trip had been correctly identified and corrected prior to returning the B reactor protection system motor generator set to service [P.2]. (Section 2.7.a)

Cornerstone: Mitigating Systems

- Green. The team 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, 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 startup feedwater regulating valve as part of the post-scram actions. The startup feedwater regulating valve operator characteristics are non-linear and not designed to operate in the dynamic conditions immediately following a reactor scram. To correct the inadequate procedure, the licensee implemented a change to direct operations personnel to utilize one of the main feedwater regulating valves until the plant is stabilized. This issue was entered in the licensee’s corrective action program as Condition Report CR-RBS-2015-00657.

This performance deficiency is more than minor, and therefore a finding, because it is 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, the procedure directed operations personnel to isolate the main feedwater regulating valves and control reactor pressure vessel level using the startup feedwater regulating valve, whose operator was not designed to function in the dynamic conditions associated with a post-scram event from high power, and this challenged the capability of the system. The team performed an initial screening of the finding in accordance with Inspection Manual Chapter 0609, Appendix A, “The Significance Determination Process (SDP) for Findings At-Power.” Using Inspection Manual Chapter 0609, Appendix A, Exhibit 2, “Mitigating Systems Screening Questions,” the team determined that the finding is of very low safety significance (Green) because it: (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 non-technical specification trains of equipment designated as high safety-significant in accordance with the licensee’s maintenance rule program. This finding has an evaluation cross-cutting aspect within the problem identification and resolution area because the licensee failed to thoroughly evaluate this issue to ensure that the resolution addressed the cause commensurate with its safety significance. Specifically, the licensee failed to properly evaluate the design characteristics of the startup feedwater regulating valve operator before implementing the procedure to use the valve for post-scram recovery actions [P.2]. (Section 2.7.b)

- <u>Green</u>. The team identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Action,” for the licensee’s failure to assure a condition adverse to quality was promptly identified. Specifically, the licensee failed to identify, that reaching the reactor pressure vessel water Level 8 (high) setpoint, on December 25, 2014, was an adverse condition, and as a result, failed to enter it into the corrective action program. To restore compliance, the licensee entered this issue into their corrective action program as Condition Report CR-RBS-2015-00620 and commenced a causal analysis for Level 8 (high) trips.

This performance deficiency is more than minor, and therefore a finding, because it is associated with the equipment performance 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, failure to identify Level 8 (high) conditions and unplanned automatic actuations as conditions adverse to quality, would continue to result in the undesired isolation of mitigating equipment including reactor feedwater pumps, the high pressure core spray pump, and the reactor core isolation cooling pump. The team performed an initial screening of the finding in accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power." Using Inspection Manual Chapter 0609, Appendix A, Exhibit 2, "Mitigating Systems Screening Questions," the team determined that the finding is of very low safety significance (Green) because it: (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 non-technical specification trains of equipment designated as high safety-significant in accordance with the licensee's maintenance rule program. This finding has an avoid complacency cross-cutting aspect within the human performance area because the licensee failed to recognize and plan for the possibility of mistakes, latent issues, and inherent risk, even while expecting successful outcomes. Specifically, the licensee tolerated leakage past the feedwater regulating valves, did not plan for further degradation, and the condition ultimately resulted in the Level 8 (high) trip of the running reactor feedwater pump on December 25, 2014 [H.12]. (Section 2.7.c)

- TBD. The team identified an apparent violation of 10 CFR 55.46(c)(1), "Plant-Referenced Simulators," for the licensee's failure to maintain the simulator so it would demonstrate expected plant response to operator input and to normal, transient, and accident conditions to which the simulator has been designed to respond. As of January 30, 2015, the licensee failed to maintain the simulator consistent with actual plant response for normal and transient conditions related to feedwater flows, alarm response, and behavior of the startup feedwater regulating valve controller. Specifically, the River Bend Station simulator failed to correctly model feedwater flows and resulting reactor vessel level response following a scram, failed to provide the correct alarm response for a loss of a reactor protection system motor generator set, and failed to correctly model the behavior of the startup feedwater regulating valve controller. As a result, operations personnel were challenged in their control of the plant during a reactor scram that occurred on December 25, 2014. This issue has been entered into the corrective action program as Condition Report RBS-CR-2015-01261, which includes actions to initiate simulator discrepancy reports, investigate and resolve the potential fidelity issues, and provide training to operations personnel on simulator differences.

  This performance deficiency is more than minor, and therefore a finding, because it is associated with the human performance attribute of the Mitigating Systems Cornerstone and adversely affected the cornerstone objective of ensuring availability, reliability, and capability of systems needed to respond to initiating events to prevent undesired consequences. Specifically, the incorrect simulator response adversely affected the operations personnel's ability to assess plant conditions and take actions in accordance with approved procedures during the December 25, 2014, scram. The team performed an initial screening of the finding in accordance with inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," Attachment 4, "Initial Characterization of Findings." Using Inspection Manual Chapter 0609, Attachment 4, Table 3, "SDP Appendix Router," the team answered 'yes' to the following question: "Does the finding involve the operator licensing requalification program or simulator fidelity?" As a result, the team used Inspection Manual Chapter 0609, Appendix I, "Licensed Operator Requalification Significance Determination Process (SDP)," and preliminarily determined the finding was of low to moderate safety significance (White) because the deficient simulator performance negatively impacted operations personnel performance in the actual plant during a reportable event (reactor scram). This finding has an evaluation cross-cutting aspect within the problem identification and resolution cross-cutting area because the licensee failed to thoroughly evaluate this issue to ensure that the resolution addressed the extent of condition commensurate with its safety significance. Specifically, the licensee's evaluation of the fidelity issue identified by the NRC in March 2014, focused on other training areas that used simulation, rather than evaluating the simulator modelling for additional fidelity discrepancies [P.2]. (Section 2.7.d)

- Green. The team identified a finding for the licensee's failure to follow written procedures for classifying deficient plant conditions as operator workarounds and providing compensatory measures or training in accordance with fleet Procedure EN-OP-117, "Operations Assessment Resources," Revision 8. A misclassification of these conditions resulted in the failure of the operations department to fully assess the impact these conditions had during a plant transient. The failure to identify operator workarounds contributed to complications experienced during reactor scram recovery on December 25, 2014. The licensee entered this issue into their corrective action program as Condition Report CR-RBS-2015-00795.

  This performance deficiency is more than minor, and therefore a finding, because it had the potential to lead to a more significant safety concern if left uncorrected. Specifically, the performance deficiency contributed to complications experienced by the station when attempting to restore feedwater following a scram on December 25, 2014. The team performed an initial screening of the finding in accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power." Using Inspection Manual Chapter 0609, Appendix A, Exhibit 2, "Mitigating Systems Screening Questions," the team determined this finding is of very low safety significance (Green) because it: (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 non-technical specification trains of equipment designated as high safety-significant in accordance with the licensee's maintenance rule program. This finding has a consistent process cross-cutting aspect in the area of human performance because the licensee failed to use a consistent, systematic approach to making decisions and failed to incorporate risk insights as appropriate. Specifically, no systematic approach was enacted in order to properly classify deficient conditions [H.8]. (Section 2.7.e)

REPORT DETAILS

1. Basis for Special Inspection

On December 25, 2014, at 8:37 a.m., River Bend Station scrammed from 85 percent power following a trip of the B reactor protection system (RPS) motor generator (MG) set. At the time of the MG set trip, a Division 1 half scram existed due to an unrelated equipment issue with a relay for the Number 2 turbine control valve fast closure RPS function. The combination of the B RPS MG set trip and the Division 1 half scram resulted in a scram of the reactor.

The following equipment issues occurred during the initial scram response.

- An unexpected Level 8 (high) reactor water level signal at +51” was received which resulted in tripping the running reactor feedwater pumps (RFPs).

- Following reset of the Level 8 (high) reactor water level signal, operations personnel were unable to start RFP C. They responded by starting RFP A at a vessel level of +25”. The licensee subsequently determined that the circuit breaker (Magne Blast type) for RFP C did not close.

- Following the start of RFP A, the licensee attempted to open the startup feedwater regulating valve (SFRV) but was unsuccessful prior to the Level 3 (low) reactor water level trip setpoint at +9.7”. The licensee then opened main feedwater regulating valve (FRV) C to restore reactor vessel water level. The lowest level reached was +8.1”. Subsequent troubleshooting revealed a faulty manual function control card. The card was replaced by the licensee and the SFRV was used on the subsequent plant startup.

Following restoration of reactor vessel water level, the plant was stabilized in Mode 3. A plant startup was conducted on December 27, 2014, with RPS bus B being supplied by its alternate power source. During power ascension following startup, RFP B did not start. The licensee re-racked its associated circuit breaker and successfully started RFP B. The licensee did not investigate the cause of RFP B failing to start.

Management Directive 8.3, “NRC Incident Investigation Program,” was used to evaluate the level of NRC response for this event. In evaluating the deterministic criteria of Management Directive 8.3, it was determined that the event: (1) included multiple failures in the feedwater system which is a short term decay heat removal mitigating system; (2) involved two Magne Blast circuit breaker issues which could possibly have generic implications regarding the licensee’s maintenance, testing, and operating practices for these components including safety-related breakers in the high pressure core spray system; and (3) involved several issues related to the ability of operations to control reactor vessel level between the Level 3 (low) and Level 8 (high) trip setpoints following a reactor scram. Since the deterministic criteria were met, the trip was evaluated for risk. The preliminary Estimated Conditional Core Damage Probability was determined to be 1.2E-6.

Based on the deterministic criteria and risk insights related to the multiple failures of the feedwater system, the potential generic concern with the Magne Blast circuit breakers, and the issues related to the licensee's operations department's inability to control reactor vessel level between the Level 3 (low) and Level 8 (high) setpoints following a reactor scram, Region IV determined that the appropriate level of NRC response was to conduct a Special Inspection.

This Special Inspection is chartered to identify the circumstances surrounding this event, determine if there are adverse generic implications, and review the licensee's actions to address the causes of the event.

The team used NRC Inspection Procedure 93812, "Special Inspection Procedure," to conduct the inspection. The inspections included field walkdowns of equipment, interviews with station personnel, and reviews of procedures, corrective action documents, and design documentation. A list of documents reviewed is provided in Attachment 1 of this report; the Special Inspection Charter is included as Attachment 2.

2. <u>Inspection Results</u>

2.1 Charter Item 2: Develop a complete sequence of events related to the reactor scram that occurred on December 25, 2014.

a. <u>Inspection Scope</u>

The team developed and evaluated a timeline of the events leading up to, during, and after the reactor scram. This includes troubleshooting activities and plant startup. The team developed the timeline, in part, through a review of work orders, action requests, station logs, and interviews with station personnel. The team created the following timeline during their review of the events related to the reactor trip that occurred on December 25, 2014.

Date/Time	Activity
December 6, 2014	
10:12 a.m.	A Division 2 half-scram was received from loss of the B RPS MG set, licensee initiated Condition Report CR-RBS-2014-06233
10:17 a.m.	The RPS bus B was transferred to the alternate power supply, Division 2 half-scram was reset

Date/Time	Activity
December 13, 2014	
12:35 p.m.	The B RPS MG set was restored
December 16, 2014	
9:30 p.m.	The RPS bus B was placed on B RPS MG set
December 23, 2014	
7:59 a.m.	The licensee commenced a reactor downpower to 85 percent to support maintenance on RFP B
08:30 a.m.	The RFP B was secured to support maintenance
10:28 a.m.	A Division 1 half-scram signal from the turbine control valve 2 fast closure relay was received, licensee initiated Condition Report CR-RBS-2014-06581
2:21 p.m.	The Division 1 half-scram signal was reset by bypassing the turbine control valve fast closure signal
10:00 p.m.	RPS channel A placed in trip condition to satisfy Technical Specification 3.3.1.1
December 25, 2014	
8:37 a.m.	Reactor scram due to loss of RPS bus B
8:39 a.m.	Feedwater master controller signal caused all FRVs to close, feedwater continued injecting at 520,000 lbm/hr (leakby through valves), reactor pressure vessel (RPV) water level at 27.8”
8:40 a.m.	RFP A was secured per procedure, RPV water level ~ 43”, feedwater flow lowered to 426,400 lbm/hr (leakby through valves)

Date/Time	Activity
8:41 a.m.	Reactor water level reached Level 8 (high) condition, RFP C (only running RFP) trips
8:42 a.m.	All FRV's and associated isolation valves were closed by operations personnel and the SFRV placed in AUTO with a setpoint at 18" per procedure
8:45 a.m.	Reactor water level dropped below 51" allowing reset of Level 8 (high) signal and restart of RFPs
8:50 a.m.	RFP C failed to start, no trip flags on RFP breaker, RPV water level ~ 33" and lowering, licensee initiated Condition Report CR-RBS-2014-06601
8:52 a.m.	Operations personnel started RFP A
8:54 a.m.	Operations personnel reset the reactor scram signal on Division 2 of RPS only, RPV water level ~ 17" and lowering
8:54 a.m.	The SFRV did not respond as expected in the automatic mode. Operations personnel attempted to control the SFRV in Manual, however it did not respond. As a result, operations personnel began placing the FRV C in service, licensee initiated Condition Report CR-RBS-2014-06602
8:56 a.m.	Water level reached Level 3 (low) and actuated a second reactor scram signal, RPV water level reached ~ 8.1", operations personnel completed placing FRV C in service and reactor water level began to rise
8:57 a.m.	RPV water level rose above 9.7", reactor scram signal clear
8:58 a.m.	Operations personnel reset the reactor scram signal on Division 2 of RPS only, RPV water level ~ 15.7"
December 27, 2014	
12:53 a.m.	The plant entered Mode 2 and commenced a reactor startup

Date/Time	Activity
10:00 a.m.	RFP C failed to start due to the associated minimum flow valve not fully opening, licensee initiated Condition Report CR-RBS-2014-06653
10:18 a.m.	Operations personnel started RFP A
5:41 p.m.	The plant entered Mode 1
December 28, 2014	
7:23 p.m.	RFP B failed to start, licensee initiated Condition Report CR-RBS-2014-06649
8:43 p.m.	The RFP B breaker was racked out and then racked back in
8:49 p.m.	RFP B was successfully started

b. Findings and Observations

In reviewing the sequence of events and developing the timeline, the team reviewed the licensee's maintenance and troubleshooting activities associated with the B RPS MG set failure on December 6, 2014. Additionally, the team reviewed the operability determination to evaluate the licensee's basis for returning the B RPS MG set to service.

The licensee's troubleshooting practices lacked the technical rigor and attention to detail necessary to identify and correct the deficient B RPS MG set conditions. On several occasions, the team noted that the licensee chose the expedient solution rather than complete an evaluation to determine that corrective actions resolved the deficient condition. Specifically, the licensee chose to restore the B RPS MG set to service without fully understanding the failure mechanism. Other examples included the licensee's choice to have operations personnel rack in and out breakers, and have maintenance personnel manually operate a limit switch, on the makeup and start logic for the RFP C minimum flow valve, when the RFP did not start. As indicated above, the licensee performed these compensatory actions instead of evaluating and correcting the issue.

Based upon a review of the events leading up to the reactor scram, the team determined the licensee failed to properly preplan and perform maintenance on the B RPS MG set after the failure that occurred on December 6, 2014. Further discussion involving the licensee's failure to adequately troubleshoot, identify, and correct degraded components on the B RPS MG set, prior to returning it to service, is included in Section 2.7.a. of this report.

Additionally, the team reviewed the procedures that operations personnel used to respond to the reactor scram and determined the licensee failed to provide adequate procedures to respond to a post-trip transient. Further discussion on the procedure prescribing activities affecting quality not being appropriate for the circumstances is included in Section 2.7.b. of this report.

2.2 Charter Items 3 and 8: Review the licensee's root cause analysis and corrective actions from the current and previous scrams with complications.

a. Inspection Scope

At the time of the inspection, the root cause report for the December 25, 2014, scram had not been completed. To ensure the licensee was conducting the cause evaluation at a level of detail commensurate with the significance of the problem, the team reviewed corrective action procedures, met with members of the root cause team, and reviewed prior related corrective actions.

The procedures reviewed by the team included quality related Procedure EN-LI-118, "Cause Evaluation Process," Revision 21, and quality related Procedure EN-LI-102, "Corrective Action Program," Revision 24.

The licensee's approach for the December 25, 2014, scram causal evaluation was to use several detailed evaluations as input to the overall root cause. Specifically, the licensee performed an apparent cause evaluation, under Condition Report CR-RBS-2014-06696, to understand the failure of Division 2 RPS equipment. The licensee performed an apparent cause evaluation under Condition Report CR-RBS-2014-06602, to review the conditions that resulted in the additional reactor water Level 3 (low) trip, after the initial scram. The licensee also performed an apparent cause evaluation, under Condition Report CR-RBS-2014-06581, to review the turbine control valve fast closure circuit failure that resulted in the Division 1 half-scram signal. All of these evaluations were reviewed under the parent root cause Condition Report CR-RBS-2014-06605.

The licensee used multiple methods in their causal evaluations that included: event and causal factor charting, barrier analysis, and organizational and programmatic failure mode trees. The licensee's charter for the root cause evaluation required several periodic meetings with the members of the different causal analysis teams. It also required a pre-corrective action review board update and review, a formal corrective action review board approval, and an external challenge review of the approved root cause report.

The NRC team also reviewed corrective actions to address complications encountered during previous reactor scrams. Specifically, the following NRC inspection reports were reviewed and the related licensee corrective actions were assessed:

- 05000458/2002002, Integrated Inspection Report, July 24, 2002, ML022050206

- 05000458/2006013, Special Inspection Team Report, March 1, 2007, ML070640396

- 05000458/2012009, Augmented Inspection Team Report, August 7, 2012, ML12221A233

- 05000458/2012012, Supplemental Inspection Report, December 28, 2012, ML12363A170

b. Findings and Observations

The NRC team found the licensee's root cause team members had met the organizational diversity and experience requirements of their procedures. The team reviewed the qualifications of the members of the root cause team and determined they were within the correct periodicity.

At the time of the inspection, there were 4 root cause and 10 apparent cause evaluations in progress. The team determined the root cause analyses were conducted at a level of detail commensurate with the significance of the problems.

In reviewing corrective actions for prior scrams, the team noted that there have been five unplanned reactor scrams in the past five years, including the December 25, 2014, event. Of those five scrams, two involved Level 8 (high) reactor water level signal trips of all running feedwater pumps. Based upon a review of prior scrams and associated corrective actions, the team determined that the licensee does not have an appropriately low threshold for recognizing Level 8 (high) reactor water level signal trips as an adverse condition, and entering that adverse condition into their corrective action program. Otherwise, the team determined that the licensee's corrective actions to address complications, encountered during previous reactor scrams, were adequate. Further discussion involving the licensee's failure to identify Level 8 (high) reactor water level signal trips as adverse conditions is included in Section 2.7.c of this report.

2.3 Charter Item 4: Determine the cause of the unexpected Level 8 (high) water level trip signal.

a. Inspection Scope

To determine the cause of the unexpected Level 8 (high) reactor water level trip on December 25, 2014, the NRC team reviewed control room logs and graphs of key reactor parameters to assess the plant's response to transient conditions. This information was then compared to the actions taken by operations personnel in the control room per abnormal and emergency operating procedure requirements.

Section 5.1 of Procedure AOP-0001, "Reactor Scram," Revision 30, required operations personnel to verify that the feedwater system was operating to restore reactor water level. This was accomplished using an attachment of Procedure OSP-0053, "Emergency and Transient Response Support Procedure," Revision 22. Specifically, Attachment 16, "Post Scram Feedwater/Condensate Manipulations Below 5% Reactor

Power," required transferring reactor water level control to the startup feedwater system after reactor water level had been stabilized in the prescribed band.

Only four minutes elapsed from the time of the scram until the time the Level 8 (high) reactor water level isolation signal was reached. Consequently, operations personnel did not have sufficient time to gain control and stabilize reactor vessel level in the required band.

To gain an understanding of issues affecting systems at the time of the scram, the NRC team met with system engineers for the feedwater system, feedwater level control system, and remotely operated valves. Discussions with engineering included system health reports, open corrective actions from condition reports, licensee event reports, design data for systems, startup testing and exceptions, post-trip reactor water level setpoint setdown parameters, open engineering change packages, and requirements for engineering to analyze post-transient plant data.

b. Findings and Observations

Operations personnel responded to the events in accordance with procedure requirements. The NRC did not identify any performance deficiencies related to immediate or supplemental actions taken by control room staff during the transient. However, operations personnel stated that the plant did not respond in a manner consistent with their simulator training.

Based on review of operations personnel response to the event and the training received from the simulator, the NRC team determined that the licensee did not maintain the simulator in a condition that accurately represented actual plant response. On April 10, 2015, the licensee provided a white paper with additional information related to the modeling of the plant-referenced simulator. Further discussion involving the licensee's failure to maintain the simulator is included in Section 2.7.d of this report.

The NRC team determined that the plant did not respond per the design as described in the final safety analysis report. Specifically, the feedwater level control system and feedwater systems were designed to automatically control reactor water level in the programmed band post-scram. During the December 25, 2014 scram, reactor water level quickly (within 4 minutes) rose to a Level 8 (high) trip. By design, reactor water level should rapidly lower after the initial level transient from core void collapse, rise as feedwater compensates for the level change, and then return to the programed setpoint. A Level 8 (high) trip should not occur. The team determined that significant leakage past the feedwater isolation valves caused the rapid rise in reactor water level. Operations personnel were unable to compensate for the rapid change in reactor vessel level. The licensee initially discovered the adverse condition during startup testing in 1986, and allowed the condition to degrade without effective corrective actions.

The team noted that significant post-trip or post-transient plant performance data was available to system engineers, but review of this data was not prioritized by the licensee. The review of plant transient data was primarily driven by the licensee's root cause team charter or by self-assigned good engineering practices. At the time of this inspection, the licensee had not quantified the amount of leakage past the FRVs, although the scram and subsequent startup had occurred one month earlier. The NRC team observed that there was a potential to miss important trends in plant performance without a more timely review.

2.4 Charter Item 5: Review the effectiveness of licensee actions to address known equipment degradations that could complicate post-scram response by operations personnel.

a. Inspection Scope

The NRC team reviewed licensee procedures for classifying and addressing plant conditions that may challenge operations personnel while performing required actions per procedures during normal and off-normal conditions.

The team reviewed the licensee's current list of operator workarounds and operator burdens. Specifically, the team was looking for any known equipment issues that could complicate post-scram response by operations personnel.

b. Findings and Observations

The team determined the licensee did not properly classify several deficient plant conditions as operator workarounds in accordance with fleet Procedure EN-OP-117, "Operations Assessment Resources," Revision 8. Further discussion related to the failure to classify plant deficiencies as operator workarounds is included in Section 2.7.e of this report.

2.5 Charter Items 6 and 7: Review the licensee's maintenance, testing and operating practices for Magne Blast circuit breakers including the causes and corrective actions taken to address the failure of the RFPs to start.

a. Inspection Scope

The team reviewed the final safety analysis report, system description, the current system health report, selected drawings, maintenance and test procedures, and condition reports associated with Magne Blast breakers. The team also performed walkdowns and conducted interviews with system engineering and design engineering personnel to ensure circuit breakers were capable of performing their design basis safety functions. Specifically, the team reviewed:

- Vendor and plant single line, schematic, wiring, and layout drawings
- Circuit breaker preventive maintenance inspection and testing procedures
- Vendor installation and maintenance manuals
- Preventive maintenance and surveillance test procedures
- Completed surveillance test and preventive maintenance results
- Corrective actions and modifications

b. Findings and Observations

Unresolved Item (URI) – Vendor and Industry Recommended Testing Adequacy on Safety-related and Safety-significant Circuit Breakers

Introduction. The team identified an unresolved item related to the licensee's breaker maintenance and troubleshooting programs for safety-related and safety-significant circuit breakers. The charter tasked the team with inspecting the issues associated with Magne Blast breaker problems that occurred during and after the December 25, 2014, scram. The NRC team determined that breaker maintenance and troubleshooting practices extended beyond the Magne Blast breakers. The team identified that there were potential issues with safety-related Master Pact breakers and determined that maintenance procedures used to ensure that 4160 V and 13.8 kV safety-related and safety-significant breakers were being maintained and overhauled in a timely manner may not conform to industry recommended standards.

Description. The team identified that the licensee's maintenance programs for Division I, II, III, and non-safety 4160 V and 13.8 kV breakers installed in the plant may not meet the standards recommended by the vendor, corporate, or Electric Power Research Institute (EPRI) guidelines. The licensee's programs were based on EPRI documents TR-106857-V2 and TR-106857-V3, which were preventive maintenance program bases for low and medium voltage switchgear. However, the licensee appeared to only implement portions of the recommended maintenance program, and were not able to provide the team with engineering analyses or technical bases to justify the changes. The EPRI guidance was developed specifically for Magne Blast breakers based on industry operating experience, NRC Information Notices, and General Electric SILs/SALs. The NRC team was concerned that the licensee may not have performed the entire vendor or EPRI recommended tests, inspections, and refurbishments on the breakers since they were installed. The aggregate impact of missing these preventive maintenance tasks needs to be evaluated to determine if the reliability of the affected breakers has been degraded.

Pending further evaluation of the above issue by the licensee and subsequent review by NRC inspectors, this issue will be tracked as URI 05000458/2015009-01, "Vendor and Industry Recommended Testing Adequacy on Safety-related and Safety-significant Circuit Breakers."

2.6 Charter Item 9: Evaluate pertinent industry operating experience and potential precursors to the event, including the effectiveness of any action taken in response to the operating experience.

a. Inspection Scope

The team evaluated the licensee's application of industry operating experience related to this event. The team reviewed applicable operating experience and generic NRC communications with a specific emphasis on Magne Blast breaker maintenance practices, to assess whether the licensee had appropriately evaluated the notifications for relevance to the facility and incorporated applicable lessons learned into station programs and procedures.

b. Findings and Observations

Other than the URI described in Section 2.5, of this report, no additional findings or observations were identified.

2.7 Specific findings identified during this inspection.

a. Failure to Establish Adequate Procedures to Perform Maintenance on Equipment that can Affect Safety-Related Equipment

Introduction. The team reviewed a Green, self-revealing, non-cited violation of Technical Specification 5.4.1 for the licensee's failure to establish adequate procedures to properly preplan and perform maintenance that affected the performance of the B RPS MG set. Specifically, due to inadequate procedures for troubleshooting on the B RPS MG set, the licensee failed to identify a degraded capacitor that caused the B RPS MG set output breaker to trip, which resulted in a reactor scram.

Description. On December 6, 2014, during normal plant operations, RPS bus B unexpectedly lost power because of a B RPS MG set failure, which resulted in a Division 2 half scram and a containment isolation signal. The RPS system is designed to cause rapid insertion of control rods (scram) to shut down the reactor when specific variables exceed predetermined limits. The RPS power system, of which the B RPS MG set is a component, is designed to provide power to the logic system that is part of the reactor protection system.

The licensee's troubleshooting teams identified both the super spike suppressor card and the field flash card as the possible causes of the B RPS MG set failure. The licensee replaced the super spike suppressor card. While inspecting the field flash card, a strand of wire from one of the attached leads was found nearly touching a trace on the circuit board. A continuity test was performed while the field flash card was being tapped and no ground was observed. A ground was observed when forcibly pushing down on the wire. The licensee believed that the wire strand most likely caused the B RPS MG set trip. The licensee removed the wire strand and re-installed the field flash card without any further troubleshooting. Operations personnel returned the B RPS MG set to service on December 16, 2014.

On December 25, 2014, while operating at 85 percent power, a reactor scram occurred due to a Division 2 RPS trip concurrent with a Division 1 RPS half-scram signal that was present at the time. The Division 1 half-scram signal was received on December 23, 2014, because of a turbine control valve fast closure signal. Troubleshooting for the cause of the Division 1 half-scram was ongoing when the Division 2 RPS trip occurred. This resulted in a full RPS actuation and an automatic reactor scram. Electrical protection assembly breakers 3B/3D and the B RPS MG set output breaker were found tripped, similar to the conditions noted following the loss of the B RPS MG set on December 6, 2014. The subsequent failure modes analysis and troubleshooting teams identified the probable cause of the failure of the B RPS MG set output breaker was an intermittent failure of the field flash card. A more detailed inspection of the field flash card revealed that a 10 microfarad capacitor had been subjected to minor heating over a long period of time. As a result, the degraded component contributed to a reactor scram. The capacitor on the field flash card in the Division 2 RPS MG set was replaced.

Analysis. Failure to establish and implement procedures to perform maintenance to correct adverse conditions on B RPS MG set equipment that can affect the performance of the safety-related reactor protection system was a performance deficiency. This performance deficiency is more than minor, and therefore a finding, because it is 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 and challenge critical safety functions during shutdown as well as power operations.

The team performed an initial screening of the finding in accordance with Inspection Manual Chapter (IMC) 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power." Using IMC 0609, Appendix A, Exhibit 1, "Initiating Event Screening Questions," this finding is determined to have very low safety significance because the transient initiator did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions would not have been available. This finding has an evaluation cross-cutting aspect within the problem identification and resolution area because the licensee failed to thoroughly evaluate the failure of the B RPS MG set to ensure that the resolution addressed the cause commensurate with its safety significance. Specifically, the licensee failed to thoroughly evaluate the condition of the field flash card to ensure that the cause of the trip had been correctly identified and corrected prior to returning the B MG set to service [P.2].

Enforcement. Technical Specification 5.4.1.a states, 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, February 1978. Regulatory Guide 1.33, Appendix A, Section 9.a., states, in part, that, "maintenance that can affect the performance of safety-related equipment should be properly preplanned and performed in accordance with written procedures, documented instructions, or drawings appropriate to the circumstances." Contrary to the above, on December 6, 2014, the licensee failed to establish adequate procedures to properly preplan and perform maintenance on the B RPS MG set that ultimately affected the performance of safety-related B RPS equipment. Specifically, due to inadequate procedures for troubleshooting on the B RPS MG set, the licensee failed to identify a degraded capacitor on the B RPS MG set that caused its output breaker to trip, prior to returning it to service. On December 25, 2014, this degraded capacitor caused the B RPS MG set breaker to trip causing a loss of power to the B RPS bus which resulted in a reactor scram. The licensee entered this issue into their corrective action program as Condition Report CR-RBS-2014-06605 and replaced the degraded field flash card capacitor. Because this finding is determined to be of very low safety significance and has been entered into the licensee's corrective action program this violation is being treated as a non-cited violation consistent with Section 2.3.2.a of the NRC Enforcement Policy:

NCV 05000458/2015009-02, “Failure to Establish Adequate Procedures to Perform Maintenance on Equipment that can Affect Safety-Related Equipment.”

b. Failure to Provide Adequate Procedures for Post-Scram Recovery

Introduction. The team reviewed a Green, 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, February 1978. 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 SFRV as part of the post-scram actions. The SFRV operator characteristics are non-linear and not designed to operate in the dynamic conditions immediately following a reactor scram from power.

Description. On November 18, 2013, the licensee modified Procedure OSP-0053, Attachment 16, due to excessive leakage across the main FRVs and verified the adequacy of the change using the simulator. The licensee did not realize that the simulator incorrectly modeled the operating characteristics of the SFRV.

On December 25, 2014, following a reactor scram, operations personnel attempted to implement Procedure OSP-0053, Attachment 16, “Post Scram Feedwater/Condensate Manipulations Below 5% Reactor Power.” When the SFRV did not begin to open as RPV level approached the level setpoint, operations personnel thought the SFRV had failed in automatic and placed the valve controller in manual. Unknown to operations personnel, the manual control of the valve was inoperable due to a faulty card. Unable to control the SFRV, operations personnel then began placing one of the main FRVs back in service. The isolation valves for the FRV are motor-operated and take approximately 90 seconds to reposition. Because of the delay in restoring feedwater to the RPV, a second Level 3 (low) water level reactor scram signal occurred.

The NRC team determined that plant data indicated the SFRV does not open on a slowly decreasing RPV water level until the controller signal reaches approximately 12.5 percent error or about 3 inches below the RPV water level setpoint on the controller. The SFRV in the simulator opens as soon as the controller open signal is greater than 0.0 percent error. When the licensee became aware of the SFRV design operating parameters, they determined that the SFRV was not designed to respond to the dynamic conditions that exist during post-scram recovery, and revised Procedure OSP-0053, Attachment 16, to continue using the main FRVs during post-scram recovery actions.

Analysis. The licensee’s failure to provide adequate guidance in Procedure OSP-0053 for post-scram recovery actions was a performance deficiency. This performance deficiency is more than minor, and therefore a finding, because it is 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, the procedural guidance that directed operations personnel to establish feedwater flow to the RPV using the SFRV as part of the post-scram actions adversely affected the capability of the feedwater systems that respond to prevent undesirable consequences. The system capability was adversely affected since the valve operator characteristics are non-linear and not designed to operate in the dynamic conditions immediately following a reactor scram from high power levels.

The team performed an initial screening of the finding in accordance with IMC 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power." Using IMC 0609, Appendix A, Exhibit 2, "Mitigating Systems Screening Questions," the finding was of very low safety significance (Green) because it: (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 non-technical specification trains of equipment designated as high safety-significant in accordance with the licensee's maintenance rule program.

This finding has an evaluation cross-cutting aspect within the problem identification and resolution area because the licensee failed to thoroughly evaluate this issue to ensure that the resolution addressed the cause commensurate with its safety significance. Specifically, the licensee failed to properly evaluate the design characteristics of the SFRV operator before implementing procedural guidance for post-scram recovery actions [P.2].

Enforcement. Technical Specification 5.4.1.a states, 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, 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 March 3, 2010, until January 30, 2015, the licensee failed to establish, implement and maintain Procedure OSP-0053, which directs operator actions for a reactor trip. Specifically, Procedure OSP-0053 inappropriately directed operations personnel to establish feedwater flow to the reactor pressure vessel using the SFRV as part of the post-scram actions. The SFRV operator characteristics are non-linear and not designed to operate in the dynamic conditions immediately following a reactor scram from high power. Subsequent to the event, the licensee changed the procedure, directing operations personnel to utilize one of the main FRVs until the plant was stabilized. Because this finding is determined to be of very low safety significance and has been entered into the licensee's corrective action program as Condition Report CR-RBS-2015-00657, this violation is being treated as a non-cited violation consistent with Section 2.3.2.a of the NRC Enforcement Policy:
NCV 05000458/2015009-03, "Failure to Provide Adequate Procedures for Post-scram Recovery."

c. Failure to Identify High Reactor Water Level as a Condition Adverse to Quality

Introduction. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," for the licensee's failure to assure a condition adverse to quality was promptly identified. Specifically, the licensee failed to identify that reaching the reactor pressure vessel water Level 8 (high) setpoint, on December 25, 2014, was an adverse condition and enter it into the corrective action program.

Description. On December 25, 2014, the licensee experienced a scram with complications. The team reviewed the post-scram report as documented in Procedure GOP-0003, "Scram Recovery," Revision 24. During the scram, the licensee experienced a Level 8 (high) reactor water condition approximately four minutes after the scram. This high water level condition should not occur for a scram when main steam isolation valves remain open and safety relief valves do not actuate.

The team noted that operations personnel followed their training and performed the required post-scram actions. Those actions did not prevent the overfeeding of the reactor vessel (which reached the Level 8 (high) setpoint), causing the RFPs to trip off and would have caused isolation of other emergency core cooling systems, if actuated, such as high pressure core spray and reactor core isolation cooling. The loss of all feedwater contributed to the RPV water level lowering to a Level 3 (low) condition that actuated a second reactor scram signal.

The team interviewed control room operations personnel, system engineers, and corrective action staff regarding the plant's response to the scram. Further, the team reviewed plant parameter graphs, control room logs, alarm logs, design history, and licensing basis documents, and determined that excessive leakage past the FRVs caused the Level 8 (high) trip of all RFPs.

In reviewing the feedwater system data from the December 24, 2014, scram, the licensee estimated 500,000 lbm/hr leaked past the closed FRVs. This represents approximately 3 percent of the full-power feedwater flow and significantly exceeds the design specification for leakage of 135,000-150,000 lbm/hr.

The licensee identified excessive leakage past the FRVs during testing in 1986. At the time of inspection, the licensee could not produce any corrective actions taken to identify or correct leakage past the FRVs. Further, the licensee had not quantified the amount of leakage past the FRVs prior to the December 24, 2014, event and NRC Special Inspection.

Procedure GOP-0003 provided a post-scram checklist to operations personnel to help identify equipment and procedure problems that should be corrected prior to the reactor startup. This document was then reviewed by the Offsite Safety Review Committee in order to understand and confirm that the plant was safe to restart. Step 1.1 stated the following:

> Following a reactor scram from high power levels, there is an initial RPV level "Shrink" of 20 to 40 inches followed by a "Swell" of approximately 10 to 20 inches. The Feedwater Level Control System is programmed to "ride out" this shrink and swell without overfilling the RPV.

In section 6.7 of Procedure GOP-003, the licensee documented that there was a control system trip of RFPs due to reaching Level 8 (high). In section 6.12, however, the licensee failed to document any off-normal trips (Level 8 (high) feed pump trips). In Attachment 3 of GOP-003 Procedure, "Analysis and Evaluations," Level 8 (high) was mentioned as part of a timeline discussion but was not listed in the final section labeled "Corrective Actions Required Prior to Returning Unit to Service." This final section was where condition reports were required for all items listed. By omitting Level 8 (high) from the discussion, no corrective action document was generated for that condition.

The licensee did not identify that reaching reactor water Level 8 (high) was an adverse condition. Therefore, the unexpected Level 8 (high) trip was not addressed prior to startup on December 28, 2014.

The team reviewed the history of Level 8 (high) RFP trips and noted that similar issues of concern were raised by the NRC in 2012. Specifically, a Supplemental Inspection, performed in 2012, for a White performance indicator associated with reactor scrams with complications documented the failure to recognize a Level 8 (high) trip as an adverse condition and enter it into the corrective action program. This non-cited violation was documented in NRC Inspection Report 05000458/2012012.

The team determined that the licensee did not have a sufficiently low threshold for entering issues into their corrective action program for reactor water level transients. Specifically, long-standing equipment issues associated with FRV leakage has led to the licensee reaching reactor water Level 8 (high) during two reactor scrams in a three-year period.

Analysis. The failure to identify Level 8 (high) reactor water level trips as adverse conditions was a performance deficiency. This performance deficiency is more than minor, and therefore a finding, because it is associated with the equipment performance 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, failure to identify Level 8 (high) conditions and resulting actuations as conditions adverse to quality, would continue to result in the undesired isolation of mitigating equipment including RFPs, the high pressure core spray pump, and the reactor core isolation cooling pump.

The team performed an initial screening of the finding in accordance with IMC 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power." Using IMC 0609, Appendix A, Exhibit 2, "Mitigating Systems Screening Questions," the finding was of very low safety significance (Green) because it: (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 non-technical specification trains of equipment designated as high safety-significant in accordance with the licensee's maintenance rule program.

This finding has an avoid complacency cross-cutting aspect within the human performance area because the licensee failed to recognize and plan for the possibility of mistakes, latent issues, and inherent risk, even while expecting successful outcomes. Specifically, the licensee tolerated excessive leakage past the FRVs, did not plan for further degradation, and the condition ultimately resulted in the Level 8 (high) trip of the running RFP on December 25, 2014 [H.12].

Enforcement. Title 10 of the *Code of Federal Regulations*, Part 50, Appendix B, Criterion XVI, "Corrective Action," 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 non-conformances are promptly identified and corrected. Contrary to the above, from December 25, 2014, to January 29, 2015, the licensee failed to assure that a condition adverse to quality was promptly identified. Specifically, the licensee failed to identify that reaching the reactor pressure vessel water Level 8 (high) setpoint, on December 25, 2014, was an adverse condition and enter it into the corrective action program. To restore compliance, the licensee entered this issue into their corrective action program as Condition Report CR-RBS-2015-00620 to perform a causal analysis for Level 8 (high) trips. Since the violation was of very low safety significance (Green), this violation is being treated as a non-cited violation, consistent with Section 2.3.2.a of the Enforcement Policy: NCV 05000458/2015009-04, "Failure to Identify High Reactor Water Level as a Condition Adverse to Quality."

d. Failure of the Plant-Referenced Simulator to Demonstrate Expected Plant Response

Introduction. The team identified an apparent violation of 10 CFR 55.46(c)(1), "Plant-Referenced Simulators," for the licensee's failure to maintain the simulator so it would demonstrate expected plant response to operator input and to normal, transient, and accident conditions to which the simulator has been designed to respond. As of January 30, 2015, the licensee failed to maintain the simulator consistent with actual plant response for normal and transient conditions related to feedwater flows, alarm response, and behavior of the SFRV controller. As a result, operations personnel were challenged in their control of the plant during a reactor scram that occurred on December 25, 2014.

Description. On December 25, 2014, River Bend Station was operating at 85 percent power when a reactor scram occurred. On January 26, 2015, a Special Inspection was initiated in response to this event. The Special Inspection team reviewed the event and identified several simulator fidelity issues. Licensee Procedure EN-TQ-202, "Simulator Configuration Control," Revision 9, provided the process requirements necessary to satisfy the guidelines for simulator testing, performance, and configuration control specified by ANSI/ANS-3.5-2009. Standard ANSI/ANS-3.5-2009, "Nuclear Power Plant Simulators for Use in Operator Training and Examination," provides the simulator testing requirements, as well as simulator configuration management to ensure simulator fidelity. Specifically, as of January 30, 2015, the River Bend Station simulator failed to model feedwater accurately and failed to model resulting reactor vessel level response following a scram, failed to provide the correct alarm response for a loss of a RPS MG set, and failed to correctly model the behavior of the SFRV controller. The simulator modeling discrepancies and how these discrepancies affected plant response during the plant trip are discussed below:

- The licensee stated their simulator modeled zero leakage across the FRV rather than the actual leakage in the plant. General Electric record 0247.230-000-016, "Feedwater Control Valve Assembly – Purchase Specification," described the total design leakage across all the FRVs was approximately 135,000 lbm/hr. This is equal to approximately 1.1 percent full feedwater flow. The flow rate across the FRVs measured in the plant on December 25, 2014, was approximately 500,000 lbm/hr, which is approximately 3 percent full feedwater flow. The rate of level change of the reactor vessel in the plant was larger than operations personnel anticipated based on training received in the simulator. ANSI/ANS-3.5-2009, Section 4.1.4(3), states, "The simulator shall not fail to cause an alarm or automatic action if the reference unit would have caused an alarm or automatic action under identical circumstances." In this case, the simulator under similar conditions did not reach the RPV water Level 8 (high) condition and trip the RFPs, when the actual plant did.

- The licensee's simulator did not correctly model all alarms that would be received on a loss of power to the RPS. ANSI/ANS-3.5-2009, Section 4.1.4(3), states, "The simulator shall not fail to cause an alarm or automatic action if the reference unit would have caused an alarm or automatic action under identical circumstances." Although the licensee had identified this discrepancy on December 11, 2014, and implemented a correction in the simulator model, operations personnel had not received training nor were they notified of the discrepancy. As a result, during the plant scram on December 25, 2014, the alarms for drywell high pressure and RPV high pressure annunciated per the facility design, operations personnel were not expecting the alarms because they did not alarm in the simulator during training.

- The simulator SFRV responded differently than the actual SFRV in the reference plant. ANSI/ANS-3.5-2009, Section 4.1.4(2) [for malfunctions], stated, "Any observable change in simulated parameters corresponds in direction to the change expected from actual or best estimate response of the reference unit to the malfunction." Plant data indicated the SFRV does not open on a slowly decreasing RPV water level until the controller signal reaches approximately 12.5 percent or about 3 inches below the RPV water level setpoint of the controller. The SFRV in the simulator opens as soon as the controller open signal is greater than 0.0. Because the SFRV did not respond as expected, operations personnel incorrectly believed the SFRV had failed in automatic operation and placed the controller in manual. Due to an unrelated issue, the manual function of the SFRV was unavailable.

Collectively, these modeling discrepancies negatively impacted licensed operations personnel performance in the actual control room, during the event of December 25, 2014. Specifically, operations personnel were not able to control reactor vessel water level during the reactor scram.

The team noted that the licensee similarly stated in Condition Report CR-RBS-2015-00641 that, "During an investigation into the report at the OSRC (Onsite Safety Review Committee) for the SCRAM on December 25, 2014, that feed regulating valve leakage (FRV) contributed to the Level 8 received reactor vessel, it was determined by analysis that there is sufficient evidence that leakage by the Feedwater Regulating Valves presents a significant challenge to Operations during a scram event."

On April 10, 2015, the licensee provided a white paper with additional information related to the modeling of the plant-referenced simulator. Specifically, it provided the licensee's perspective with regard to the following issues raised by the NRC:

1. Two unexpected alarms on loss of Division II Reactor Protection System Power
2. Main Feedwater Regulating Valve Seat Leakage
3. Start-up Feedwater Regulating Valve Response

The licensee concluded that although they perceived that there were differences between the simulator and the actual plant, they were considered to be minor. For each of the items in question, the paper summarized that operator performance was not impacted by simulator modeling. The team considered the information in the white paper, and disagreed with the licensee's conclusions. Some of the information provided, however, did improve the team's understanding of the modeling deficiencies.

<u>Analysis</u>. The failure to maintain the plant-referenced simulator so that it would demonstrate expected plant response to operator input and to normal and transient conditions was a performance deficiency. This performance deficiency is more than minor, and therefore a finding, because it is associated with the human performance attribute of the Mitigating Systems Cornerstone and adversely affected the objective of ensuring availability, reliability, and capability of systems needed to respond to initiating events to prevent undesired consequences. Specifically, the incorrect simulator response adversely affected the operating crew's ability to assess plant conditions and take actions in accordance with approved procedures during the December 25, 2014, scram.

The team performed an initial screening of the finding in accordance with IMC 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," Attachment 4, "Initial Characterization of Findings." Using IMC 0609, Attachment 4, Table 3, "SDP Appendix Router," the team answered 'yes' to the following question: "Does the finding involve the operator licensing requalification program or simulator fidelity?" As a result, the team used IMC 0609, Appendix I, "Licensed Operator Requalification Significance Determination Process (SDP)," and preliminarily determined the finding was of low to moderate safety significance (White) because the deficient simulator performance negatively impacted operations personnel performance in the actual plant during a reportable event. This modeling deficiency resulted in actual impact on operations personnel performance during response to a reactor scram that occurred on December 25, 2014.

The NRC recently issued a non-cited violation related to simulator fidelity in March 2014 documented in Inspection Report 05000458/2014301. Since the licensee recently verified simulator fidelity, this issue is indicative of current plant performance and has an evaluation cross-cutting aspect within the problem identification and resolution area because the licensee failed to thoroughly evaluate this issue to ensure that the resolution addressed the extent of condition commensurate with its safety significance. Specifically, the licensee's evaluation of the fidelity issue focused on other training areas that used simulation, rather than evaluating the simulator modelling for additional fidelity discrepancies [P.2].

<u>Enforcement</u>. Title 10 of the *Code of Federal Regulations*, Part 55.46(c)(1), "Plant-Referenced Simulators," requires in part, that a simulator "must demonstrate expected plant response to operator input and to normal, transient, and accident conditions to which the simulator has been designed to respond."

Contrary to the above, as of January 30, 2015, the simulator failed to demonstrate expected plant response to operator input and to normal, transient, and accident conditions to which the simulator has been designed to respond. Specifically, the River Bend Station simulator failed to correctly model leakage flow rates across the FRVs; failed to provide the correct alarm response for a loss of a RPS MG set; and failed to correctly model the behavior of the SFRV controller. These simulator modeling issues led to negative training of operators. This subsequently complicated the operator's response to a reactor scram in the actual plant on December 25, 2014. This issue has been entered into the corrective action program as Condition Report CR-RBS-2015-01261. The licensee's condition report included actions to initiate simulator discrepancy reports, to investigate and resolve the potential fidelity issues, and to provide training to operations personnel on simulator differences. This is a violation of 10 CFR 55.46(c)(1), "Plant-Referenced Simulators": AV 05000458/2015009-05, "Failure of the Plant-Referenced Simulator to Demonstrate Expected Plant Response."

e. <u>Failure to Identify and Classify Operator Workarounds that Impacted Scram Recovery Actions</u>

<u>Introduction</u>. The team identified a Green finding for the licensee's failure to follow written procedures for classifying deficient plant conditions as operator workarounds and providing compensatory measures or training in accordance with fleet Procedure EN-OP-117. A misclassification of these conditions resulted in the failure of the operations department to fully assess the impact these conditions had during a plant

transient. The failure to identify operator workarounds contributed to complications experienced during reactor scram recovery on December 25, 2014.

Description. The team reviewed the recovery actions taken by the main control room staff following the reactor scram on December 25, 2014, from 85 percent power. During the review, the team observed the station had zero conditions identified as operator workarounds. The team reviewed fleet Procedure EN-FAP-OP-006, "Operator Aggregate Impact Index Performance Indicator," Revision 2. This procedure defined an operator workaround as:

> Any plant condition (equipment or other) that would require compensatory operator actions in the execution of normal operating procedures, abnormal operating procedures, emergency operating procedures, or annunciator response procedures during off-normal conditions. This indicator provided a measure of plant safety. It provided a measure of the likelihood that a plant transient may be complicated by equipment and human performance problems.

During their review, the team identified the following three conditions which met the definition of an operator workaround as described in Procedure EN-FAP-OP-006, and which were in effect prior to the December 25, 2014, event:

- Work Order WO-RBS-00404323: RFP B supply breaker repetitive failures to close potentially reduces the number of feedwater pumps available to operations personnel during a transient following reactor pressure vessel water Level 8 (high). Operations personnel would rack out and then rack the breaker back in until the breaker would function properly. This work order was initiated on February 3, 2015, following discussions with the NRC inspection team.

- Work Order WO-RBS-00396449: RFP C minimum flow valve does not stroke fully open which prevents starting the C feed pump. Maintenance personnel would manually operate a limit switch on the valve to make up the start logic for the RFP. This work order was initiated on October 10, 2014.

- Work Order WO-RBS-00346642: leakage past FRVs when closed complicated post-scram reactor water level control. Operations personnel proceduralized the closure of the main feedwater isolation valves to stop the effect of the leakage. This work order was initiated on March 27, 2013.

The deficient conditions in WO-RBS-00346642 and WO-RBS-00396449 contributed to complications experienced by the station when attempting to restore feedwater following a scram and loss of all feedwater pumps on a reactor pressure vessel water Level 8 (high).

Fleet Procedure EN-OP-117, Attachment 9.4, "Operator Aggregate Assessment of Plant Deficiencies," provides a method to assess and document the impact of plant deficiencies on operations personnel response during off-normal and emergency conditions. In order to assess the cumulative impact of outstanding operator aggregate impact deficiencies, several deficiency types were evaluated, including operator workarounds. Following assessment of deficiencies, Attachment 9.4, step 5, directed the station to provide compensatory measures or training as appropriate until the deficiencies could be corrected.

The resident inspectors engaged operations department management in January 2015, and informed the licensee that the three conditions appeared to meet the definition of an operator workaround as described in Procedure EN-FAP-OP-006. Upon learning of the misclassification of these issues, the station revised their operator aggregate index on February 6, 2015, to account for the three operator workaround conditions and the indicator turned red. As a result, the station issued guidance for post-scram reactor water level control and required operating crews to attend simulator training on vessel level control and feedwater system recovery following a Level 8 (high) trip of feedwater pumps. Additionally, the station wrote Condition Report CR-RBS-2015-00795 to document the issue.

Analysis. The failure to follow written procedures for classifying deficient plant conditions as operator workarounds and providing compensatory measures or training in accordance with fleet Procedure EN-OP-117 was a performance deficiency. This performance deficiency is more than minor, and therefore a finding, because it had the potential to lead to a more significant safety concern if left uncorrected. Specifically, the performance deficiency contributed to complications experienced by the station when attempting to restore feedwater following a scram on December 25, 2014.

The team performed an initial screening of the finding in accordance with IMC 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power." Using IMC 0609, Appendix A, Exhibit 2, "Mitigating Systems Screening Questions," the finding was of very low safety significance (Green) because it: (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 non-technical specification trains of equipment designated as high safety-significant in accordance with the licensee's maintenance rule program.

This finding has a consistent process cross-cutting aspect within the human performance area because the licensee failed to use a consistent, systematic approach to making decisions and incorporate risk insights as appropriate. Specifically, no systematic approach was enacted in order to properly classify deficient conditions [H.8].

Enforcement. Enforcement action does not apply because the performance deficiency did not involve a violation of regulatory requirements. Because this finding does not involve a violation and is of very low safety significance, this issue was entered into the licensee's corrective action program as Condition Report CR-RBS-2015-00795: FIN

05000458/2015001-06, “Failure to Identify and Classify Operator Workarounds That Impacted Scram Recovery Actions.”

4OA6 Meetings, Including Exit

Exit Meeting Summary

On January 20, 2015, the team initially debriefed Mr. E. Olson, Site Vice President, and other members of the licensee's staff. The licensee representatives acknowledged the findings presented.

On June 29, 2015, the team conducted an exit briefing with Mr. E. Olson, Site Vice President, and other members of the licensee’s staff. The licensee representatives acknowledged the findings presented.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

E. Olson, Site Vice President
D. Bergstrom, Senior Operations Instructor
M. Browning, Senior Operations Instructor
T. Brumfield, Director, Regulatory & Performance Improvement
S. Carter, Manager, Shift Operations
M. Chase, Manager, Training
J. Clark, Manager, Regulatory Assurance
F. Corley, Manager, Design & Program Engineering
T. Creekbaum, Engineer
G. Degraw, Manager, Training
G. Dempsey, Senior Operations Instructor
S. Durbin, Superintendent, Operations Training
R. Gadbois, General Manager, Plant Operations
T. Gates, Manager, Operations Support
J. Henderson, Assistant Manager, Operations
K. Huffstatler, Senior Licensing Specialist, Licensing
K. Jelks, Engineering Supervisor
G. Krause, Assistant Manager, Operations
T. Laporte, Senior Staff Operations Instructor
R. Leasure, Superintendent, Radiation Protection
P. Lucky, Manager, Performance Improvement
J. Maher, Manager, Systems & Components Engineering
W. Mashburn, Director, Engineering
W. Renz, Director, Emergency Planning, Entergy South
J. Reynolds, Senior Manager, Maintenance
T. Shenk, Manager, Operations
T. Schenk, Manager, Operations
S. Vazquez, Director, Engineering
D. Williamson, Senior Licensing Specialist
D. Yoes, Manager, Quality Assurance

NRC Personnel

G. Warnick, Branch Chief
J. Sowa, Senior Resident Inspector
R. Deese, Senior Reactor Analyst

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000458/2015009-01	URI	Vendor and Industry Recommended Testing Adequacy on Safety-related and Safety-significant Circuit Breakers (Section 2.5.b)

Opened and Closed

05000458/2015009-02	NCV	Failure to Establish Adequate Procedures to Perform Maintenance on Equipment that can Affect Safety-Related Equipment (Section 2.7.a)
05000458/2015009-03	NCV	Failure to Provide Adequate Procedures for Post-scram Recovery (Section 2.7.b)
05000458/2015009-04	NCV	Failure to Identify High Reactor Water Level as a Condition Adverse to Quality (Section 2.7.c)
05000458/2015009-05	AV	Failure of the Plant-Referenced Simulator to Demonstrate Expected Plant Response (Section 2.7.d)
05000458/2015009-06	FIN	Failure to Identify and Classify Operator Workarounds that Impacted Scram Recovery Actions (Section 2.7.e)

LIST OF DOCUMENTS REVIEWED

DRAWINGS

NUMBER	TITLE	REVISION
GE-828E445AA, Sheet 7	Elementary Diagram – Nuclear Steam Supply Shutoff System	34
GE-828E445AA, Sheet 8	Elementary Diagram – Nuclear Steam Supply Shutoff System	33
GE-828E445AA, Sheet 10	Elementary Diagram – Nuclear Steam Supply Shutoff System	31
GE-828E445AA, Sheet 11	Elementary Diagram – Nuclear Steam Supply Shutoff System	30
GE-828E445AA, Sheet 12	Elementary Diagram – Nuclear Steam Supply Shutoff System	30
GE-828E445AA, Sheet 15	Elementary Diagram – Nuclear Steam Supply Shutoff System	37
GE-944E981	Elementary Diagram – RPS MG Set Control System	11

DRAWINGS

NUMBER	TITLE	REVISION
PID-25-01A	Engineering P&I Diagram – System 051, Nuclear Boiling Instrumentation	19
PID-25-01B	Engineering P&I Diagram – System 051, Nuclear Boiling Instrumentation	7
828E531AA, Sheet 4	Elementary Diagram – Reactor Protection System	25
828E531AA, Sheet 4A	Elementary Diagram – Reactor Protection System	22
828E531AA, Sheet 6	Elementary Diagram – Reactor Protection System	27

PROCEDURES

NUMBER	TITLE	REVISION
AOP-0001	Reactor Scram	30
AOP-0003	Automatic Isolations	33
AOP-0006	Condensate/Feedwater Failures	19
AOP-0010	Loss of One RPS Bus	19
EN-FAP-OM-004	Fleet and Site Business Plan Process	0
EN-FAP-OM-012	Prompt Investigation, Notifications and Duty Manager Responsibilities	6
EN-FAP-OP-006	Operator Aggregate Impact Index Performance Indicator	2
EN-LI-102	Corrective Action Program	24
EN-LI-118	Cause Evaluation Process	21
EN-MA-125	Troubleshooting Control of Maintenance Activities	17
EN-OP-104	Operability Determination Process	7
EN-OP-115	Conduct of Operations	15
EN-OP-117	Operations Assessment Resources	8
EN-OP-115-09	Log Keeping	1
EN-TQ-202	Simulator Configuration Control	9
EOP-0001	RPV Control	26
EOP-0003	Secondary Containment and Radioactive Release Control	16

PROCEDURES

NUMBER	TITLE	REVISION
EPSTG-0001	Emergency Operating and Severe Accident Procedures – Plant Specific Technical Guidelines (PSTG)	16
EPSTG-0002	EPGs/SAGs to PSTG to EOP/SAP Flowcharts Comparison	16
EPSTG-0002, Appendix B	Emergency Operating and Severe Accident Procedures - Bases	16
GOP-0001	Plant Startup	83
GOP-0002	Plant Shutdown	70
GOP-0003	Scram Recovery for December 27, 2014	24
OSP-0001	Control of Operator Aids	13
OSP-0053	Emergency and Transient Response Support Procedure	22

CONDITION REPORTS

CR-RBS-1998-00384	CR-RBS-2002-00672	CR-RBS-2002-00688	CR-RBS-2006-04078
CR-RBS-2011-02209	CR-RBS-2011-09053	CR-RBS-2012-02249	CR-RBS-2012-03434
CR-RBS-2012-03439	CR-RBS-2012-03440	CR-RBS-2012-03665	CR-RBS-2012-03739
CR-RBS-2012-03816	CR-RBS-2012-03817	CR-RBS-2012-05894	CR-RBS-2012-06015
CR-RBS-2012-07249	CR-RBS-2012-07250	CR-RBS-2012-07251	CR-RBS-2012-07253
CR-RBS-2012-07254	CR-RBS-2013-04419	CR-RBS-2014-05200	CR-RBS-2014-05209
CR-RBS-2014-06233	CR-RBS-2014-06357	CR-RBS-2014-06561	CR-RBS-2014-06581
CR-RBS-2014-06602	CR-RBS-2014-06605	CR-RBS-2014-06649	CR-RBS-2014-06696
CR-RBS-2015-00030	CR-RBS-2015-00043	CR-RBS-2015-00153	CR-RBS-2015-00318
CR-RBS-2015-00365	CR-RBS-2015-00480	CR-RBS-2015-00482	CR-RBS-2015-00483
CR-RBS-2015-00484	CR-RBS-2015-00486	CR-RBS-2015-00487	CR-RBS-2015-00579
CR-RBS-2015-00620	CR-RBS-2015-00626	CR-RBS-2015-00641	CR-RBS-2015-00657
CR-RBS-2015-00795	CR-RBS-2015-01261	CR-RBS-2015-02810	

WORK ORDERS

WO-RBS-00346642	WO-RBS-00396449	WO-RBS-00401085	WO-RBS-00404323

MISCELLANEOUS DOCUMENT

NUMBER	TITLE	REVISION / DATE
EC 50374	Engineering Change – Feedwater Level Control Setpoint Setdown Modification	0
EN-LI-100-ATT-9.1	Process Applicability Determination Form for AOP-0001, Reactor Scram, Revision 24	August 6, 2007
LI-101	50.59 Review Form for GOP-0002, Power Decrease/Plant Shutdown, Revision 30	August 26, 2004
GE-22A3778	Feedwater Control System (Motor Driven Feed Pumps) Design Specification	4
GE-22A3778AB	Feedwater Control System (Motor Driven Feed Pumps) Design Specification Data Sheet	7
RLP-LOP-0511	Licensed Operator Requalification – Industry Events/Operating Experience and Plant Modifications	August 1, 2002
1-ST-27-TC6	Startup Procedure and Results – Turbine Trip and Generator Load Reject	June 27, 1986
107-Feedwater	System Health Report – Feedwater	Q2 2014
0247.230-000-16	Feedwater Control Valve Assembly – Purchase Specifications	301
	List of Actuations/Isolations That Occur From Loss of RPS Bus B	January 29, 2015
	Main Control Room Log	December 6, 2014
	Main Control Room Log	December 13, 2014
	Main Control Room Log	December 16, 2014
	Main Control Room Log	December 27, 2014
	Main Control Room Log	December 28, 2014

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

January 15, 2015

MEMORANDUM TO: Tom Hartman, Senior Resident Inspector
Reactor Projects Branch B
Division of Reactor Projects

FROM: Troy Pruett, Director **/RA/**
Division of Reactor Projects

SUBJECT: SPECIAL INSPECTION CHARTER TO EVALUATE CAUSES OF THE UNPLANNED REACTOR TRIP WITH COMPLICATIONS AT THE RIVER BEND STATION

In response to the unplanned reactor trip with complications at the River Bend Station, a special inspection will be performed. You are hereby designated as the special inspection team leader. The following members are assigned to your team:

- Jim Drake, Senior Reactor Inspector, Division of Reactor Safety
- Dan Bradley, Resident Inspector, Division of Reactor Projects

A. Basis

On December 25, 2014, at 8:37 AM, River Bend Station scrammed from 85 percent power following a trip of the B reactor protection system (RPS) motor generator (MG) set. At the time of the MG set trip, a Division 1 half scram existed due to an unrelated equipment issue with a relay for the No. 2 turbine control valve fast closure RPS function. The combination of the B RPS MG set trip and the Division 1 half scram resulted in a scram of the reactor.

The following equipment issues occurred during the initial scram response.

- An unexpected Level 8 (high) reactor water level signal was received which resulted in tripping of all RFPs.
- Following reset of the Level 8 high reactor water level signal, plant operators were unable to start RFP C. Plant operators responded by starting RFP A at a vessel level of 25”. The licensee subsequently determined that the circuit breaker (Magne Blast type) for RFP C did not close because an interlock lever for a microswitch that controls the breaker close permissive was not fully engaged in the cubicle.
- Following the start of RFP A, the licensee attempted to open the startup feed regulating valve but was unsuccessful prior the Level 3 low reactor water level trip setpoint at +9.7”. The licensee then opened the C main feedwater regulating valve to

restore reactor vessel water level. The lowest level reached was +7.5". Subsequent troubleshooting revealed a faulty manual function control card. The card was replaced by the licensee and the startup feedwater regulating valve was used on the subsequent plant startup.

Following restoration of reactor vessel water level, the plant was stabilized in Mode 3. A plant startup was conducted on December 27, 2014 with RPS bus B being supplied by its alternate power source. During power ascension following startup, RFP B did not start. The licensee re-racked its associated circuit breaker and successfully started RFP B.

Management Directive 8.3, "NRC Incident Investigation Program," was used to evaluate the level of NRC response for this event. In evaluating the deterministic criteria of MD 8.3, it was determined that: (1) The event included multiple failures in the feedwater system which is a short term decay heat removal mitigating system; (2) involved two Magna Blast circuit breaker issues which could possibly have generic implications regarding the licensee's maintenance, testing, and operating practices for these components including safety-related breakers in the high pressure core spray system; and, (3) involved several issues related to the ability of operations to control reactor vessel level between the Level 3 low and Level 8 high trip set points following a reactor scram. Since the deterministic criteria was met, the trip was evaluated for risk. The preliminary Estimated Conditional Core Damage Probability was determined to be 1.2E-6.

Based on the deterministic criteria and risk insights related to the multiple failures of the feedwater system, the potential generic concern with the Magna Blast circuit breakers, and the issues related to the licensee's Operations department's inability to control reactor vessel level between the Level 3 and Level 8 setpoints following a reactor scram, Region IV determined that the appropriate level of NRC response was to conduct a Special Inspection.

This Special Inspection is chartered to identify the circumstances surrounding this event, determine if there are adverse generic implications, and review the licensee's actions to address the causes of the event.

B. Scope

The inspection is expected to perform data gathering and fact-finding in order to address the following:

1. Provide a recommendation to Region IV management as to whether the inspection should be upgraded to an augmented inspection team response. This recommendation should be provided by the end of the first day on site.

2. Develop a complete sequence of events related to the reactor scram that occurred on December 25, 2014. The chronology should include the events leading to the reactor scram, the licensee's immediate scram response and the licensee's post-scram recovery actions including troubleshooting and reactor startup.

3. Review the licensee's root cause analysis and determine if it is being conducted at a level of detail commensurate with the significance of the problem.

4. Determine the causes for the unexpected Level 8 high water level trip signal that was experienced following the reactor scram.

5. Review the effectiveness of licensee actions to address known equipment degradations that could complicate post scram operator response.

6. Review the causes and corrective actions taken to address the failure of RFP C to start during the initial scram response and RFP B during the subsequent reactor startup. For issues related to Magne Blast circuit breakers, verify that the licensee's corrective actions have addressed extent of condition and extent of cause.

7. Review the licensee's maintenance, testing and operating practices for Magne Blast circuit breakers. Promptly communicate any potential generic issues to regional management.

8. Review the licensee's corrective actions to address complications encountered during previous reactor scrams. Reference previously docketed correspondence regarding complicated reactor scrams in NRC inspection reports 05000458/2002002, 05000458/2006013, 05000458/2012009 and 05000458/2012012.

9. Evaluate pertinent industry operating experience and potential precursors to the event, including the effectiveness of any action taken in response to the operating experience.

10. Collect data necessary to support completion of the significance determination process.

C. Guidance

Inspection Procedure 93812, "Special Inspection," provides additional guidance to be used by the Special Inspection Team. Your duties will be as described in Inspection Procedure 93812. The inspection should emphasize fact-finding in its review of the circumstances surrounding the event. It is not the responsibility of the team to examine the regulatory process. Safety concerns identified that are not directly related to the event should be reported to the Region IV office for appropriate action.

You will formally begin the special inspection with an entrance meeting to be conducted no later than January 26, 2015. You should provide a daily briefing to Region IV management during the course of your inspections and prior to your exit meeting. A report documenting the results of the inspection should be issued within 45 days of the completion of the inspection.

This Charter may be modified should you develop significant new information that warrants review. Should you have any questions concerning this Charter, contact Jeremy Groom at (817) 200-1144.

cc via E-mail:
M. Dapas
K. Kennedy
T. Pruett
A. Vegel
J. Clark
V. Dricks
W. Maier
J. Groom
J. Sowa
R. Azua
N. Taylor
T. Hartman
J. Drake
D. Bradley

ADAMS ACCESSION NUMBER ML15015A634

SUNSI Rev Compl.	☑ Yes ☐ No	ADAMS	☑Yes ☐ No	Reviewer Initials	JRG
Publicly Avail	☑ Yes ☐ No	Sensitive	☐ Yes ☑No	Sens. Type Initials	JRG
Keyword	MD 3.4/A.7				

RIV/DRP: BC	RIV/DRP: DIR			
JRGroom	TWPruett			
/RA/RAzua for	***/RA/***			
1/15/15	1/15/15			

OFFICIAL RECORD