DUKE ENERGY®

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Attn: Document Control Desk
United States Nuclear Regulatory Commission
Washington, DC 20555-0001

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-261/RENEWED LICENSE NO. DPR-23

LICENSEE EVENT REPORT NOS. 2013-001-01, 2013-003-01, 2014-001-01 REVISIONS TO INCLUDE AND/OR CLARIFY REQUIRED INFORMATION

Ladies and Gentlemen:

Pursuant to 10 CFR 50.73, Duke Energy Progress, Inc. is submitting the attached Licensee Event Report revisions. The revisions provide additional information required by 10 CFR 50.73(b)(2)(ii)(F), 10 CFR 50.73(b)(2)(ii)(J) and NRC Form 366, Item 13. Should you have any questions regarding this matter, please contact Mr. R. Hightower, Manager - Nuclear Regulatory Affairs at (843) 857-1329.

This submittal contains no new Regulatory Commitments.

Sincerely,

R. M. Glover / for

W. R. Gideon
Site Vice President
H. B. Robinson Steam Electric Plant, Unit No. 2

WRG/jmw

Attachments:

I. LER 2013-001-01: Non-Environmentally-Qualified Splice Rendered Post Accident Monitoring Instrumentation Channel Inoperable
II. LER 2013-003-01: Reactor Trip on 4KV Bus Undervoltage During Load Transfer
III. LER 2014-001-01: Reactor Trip Due to a Two-out-of-Three Logic Signal from Steam Generator Water Level Protection Train B Logic Matrix

c: V. McCree, NRC, Region II
Ms. Martha C. Barillas, NRC Project Manager, NRR
NRC Resident Inspector, HBRSEP, Unit No. 2

IE22
NRR

H. B. ROBINSON STEAM ELECTRIC PLANT UNIT 2

LICENSEE EVENT REPORT NO. 2013-001-01
REVISION TO NON-ENVIRONMENTALLY-QUALIFIED SPLICE RENDERED POST ACCIDENT MONITORING INSTRUMENTATION CHANNEL INOPERABLE

NRC FORM 366 (02-2014) U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

(See Page 2 for required number of digits/characters for each block)

APPROVED BY OMB: NO. 3150-0104 EXPIRES: 01/31/2017

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
H. B. Robinson Steam Electric Plant, Unit No. 2	05000 261	1 OF 3

4. TITLE

Non-Environmentally-Qualified Splice Rendered Post Accident Monitoring Instrumentation Channel Inoperable

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER 05000
10	06	2013	2013 -	001 -	01	08	05	2014	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE: N

10. POWER LEVEL: 000

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check all that apply)*

☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(i)(C)	☐ 50.73(a)(2)(vii)
☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☐ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(C)	☐ OTHER
☐ 20.2203(a)(2)(vi)	☑ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

LICENSEE CONTACT	TELEPHONE NUMBER *(Include Area Code)*
R. Hightower, Manager - Nuclear Regulatory Affairs	(843) 857-1329

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
A	IP	ISV	Namco	Y					

14. SUPPLEMENTAL REPORT EXPECTED	15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
☐ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)* ☑ NO				

ABSTRACT *(Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)*

On 10/6/2013, with the plant de-fueled and vessel head removed, it was discovered that a non-environmentally-qualified butt-splice was installed in a wire connected to the 'closed' limit switch for a containment isolation valve, which rendered the Post-Accident Monitoring (PAM) Instrumentation function - Containment Isolation Valve Position Indication - inoperable. This condition has been present for an extended period of time, and it is presumed that on multiple occasions this function was inoperable for a period of time greater than allowed by Technical Specifications (TS) 3.3.3, PAM Instrumentation Limiting Conditions for Operation.

The initial investigation into the cause of this event indicates this was an isolated human performance event in which the non-licensed air-operated valve (AOV) technician failed to use proper material specified for the task per the procedure directing the task. Immediate corrective action consisted of the removal of non-environmentally-qualified splice and subsequent installation of an environmentally-qualified splice, which returned the component to operable condition.

This event did not impact the health and safety of the public.

NARRATIVE

PLANT IDENTIFICATION
Westinghouse - Pressurized Water Reactor

BACKGROUND
On 10/6/2013, with the plant de-fueled and no involvement of out-of-service structures, systems or components, it was discovered that a non-environmentally-qualified butt-splice was installed on a wire connected to the 'closed' limit switch for a containment isolation valve, which rendered the Post-Accident Monitoring (PAM) [IP] Instrumentation function - Containment Isolation Valve Position Indication - inoperable. This condition has been present for an extended period of time, and it is presumed that on multiple occasions this function was inoperable for a period of time greater than allowed by Technical Specifications (TS) 3.3.3, PAM Instrumentation Limiting Conditions for Operation (LCO). Condition Report 640902 was generated to address this violation.

EVENT DESCRIPTION
On 10/6/2013, with the plant de-fueled, personnel performing work to replace the limit switches of the Chemical Volume and Control System (CVCS) [CB] Letdown Line Isolation Valve, CVC-204B [ISV] discovered a non-environmentally-qualified (non-EQ) butt-splice installed in a wire connected to this valve's 'closed' limit switch, CVC-204B-LS-C [33]. Per TMM-036, "Environmentally Qualified (EQ) Electrical Equipment Program," the limit switches associated with CVC-204B are required to be EQ to assure compliance with H. B. Robinson Steam Electric Plant, Unit No. 2 (HBRSEP2) commitments regarding Regulatory Guide 1.97, Rev. 3, May 1983, "Instrumentation for Light Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident." The subject splice was found to be installed without the proper heat shrink insulators as required by procedure.

CVC-204A&B are air-operated globe valves [ISV] that isolate the CVCS Letdown Line, are normally open, and will close on Phase 'A' containment isolation signal or loss of supply air. These valves are arranged in series to ensure letdown can be isolated, even if one valve fails. The potential impact of the use and subsequent failure of a non-EQ splice in the 'closed' limit switch wiring would be loss of 'closed' indication for CVC-204B at the main control board [MCBD]. However, this loss of indication would not prevent the valve from functioning as designed.

The improper splice in the wiring to limit switch CVC-204B-LS-C did not meet EQ requirements, which rendered the PAM Instrumentation function - Containment Isolation Valve Position Indication - inoperable. This condition has been present for an extended period of time, and HBRSEP2 presumes that on multiple occasions this function was inoperable for a period of time greater than allowed by Technical Specifications (TS) 3.3.3, PAM Instrumentation LCO.

CAUSAL FACTORS
The investigation into the cause of this event indicates this was an isolated human performance event in which the non-licensed air-operated valve (AOV) technician failed to use the proper (Raychem) heat shrink insulators identified for the task per the work order instructions and the procedure (CM-309) directing the task.

A review of work orders (WO) for CVC-204B was performed to identify a date for installation of the non-EQ butt-splice. In May 1992, under WO 90AMPA2-CM, limit switch CVC-204B-LS-C was disconnected and reconnected with no documentation of the proper heat shrink insulators being installed for the six butt-splices performed; one of which is presumed to be the origin of the non-EQ butt-splice. There is no evidence that would suggest any time or situational pressures present during performance of WO 90AMPA2-CM.

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NARRATIVE

CAUSAL FACTORS (continued)

The same technician, which performed the improper butt-splice, performed similar splices properly less than two weeks later under a different WO, 92AESI1. The completed work documentation for this WO clearly documents the splices were performed per CM-309 using non-insulated butt-splices insulated with the proper heat shrink insulators as required per EQ guidelines. The technician's accurate performance of the butt-splices in the latter WO, and the WO under which the improper splice was performed, illustrates an established understanding of the EQ requirements for this component, the capability of the technician to satisfactorily perform CM-309, and the isolated nature of this event.

CORRECTIVE ACTIONS

Completed:

Removal of the non-EQ splice and subsequent installation of an EQ splice under WO 2066875-01, which returned the component to operable condition.

SAFETY ANALYSIS

There are three plausible events in which CVC-204B-LS-C failure would be important to safety. First, the spurious closure of the isolation valve CVC-204B could lead to loss of CVCS given failure of Volume Control Tank (VCT) [TK] makeup from the Refueling Water Storage Tank (RWST) [TK]. The spurious closure of CVC-204B has a low probability of occurrence and the resulting loss of closure indication would not prevent the VCT level indications from providing sufficient information to operations to take action in mitigating the event. Second, failure to isolate CVC-204B would only be important if the other isolation valve were to fail. Third, a break in the letdown line in pipe alley would be mitigated by the automatic closure of containment isolation valves LCV-460A&B [ISV], which are upstream of CVC-204B.

The mitigation of these events would not be impacted by the loss of closure indication for CVC-204B, and would not prevent valve CVC-204B from functioning as intended; therefore, the risk significance is low.

ADDITIONAL INFORMATION

An Operating Experience (OE) search for related events at HBRSEP2 and across the industry was conducted. None of the internal OE reviewed revealed a workmanship issue similar to that evaluated in this event. The review of the related external OE indicates that, across the industry, there have been similar occurrences to the event at RNP. The OE did not provide additional insights regarding effective corrective actions for the current event.

Energy Industry Identification System (EIIS) codes for systems and components relevant to this event are identified in the EQUIPMENT IDENTIFICATION section and in the text of this document within brackets [ ].

H. B. ROBINSON STEAM ELECTRIC PLANT UNIT 2

LICENSEE EVENT REPORT NO. 2013-003-01
REVISION TO REACTOR TRIP ON 4KV BUS UNDERVOLTAGE DURING LOAD TRANSFER

NRC FORM 366 (02-2014) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB: NO. 3150-0104 EXPIRES: 01/31/2017
LICENSEE EVENT REPORT (LER) (See Page 2 for required number of digits/characters for each block)	Estimated burden per response to comply with this mandatory collection request: 80 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the FOIA, Privacy and Information Collections Branch (T-5 F53), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to Infocollects.Resource@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
H. B. Robinson Steam Electric Plant, Unit No. 2	05000 261	1 OF 3

4. TITLE

Reactor Trip on 4KV Bus Undervoltage During Load Transfer

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER 05000
11	05	2013	2013 -	003 -	01	08	05	2014	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE	11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check all that apply)*			
1	☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(i)(C)	☐ 50.73(a)(2)(vii)
	☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
	☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
	☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
10. POWER LEVEL	☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☑ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
	☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
019	☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
	☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(C)	☐ OTHER
	☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

LICENSEE CONTACT	TELEPHONE NUMBER *(Include Area Code)*
R. Hightower, Manager - Nuclear Regulatory Affairs	(843) 857-1329

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
A	EA	BKR	WSTGHSE	Y	X	BI	P	Johnston	Y

14. SUPPLEMENTAL REPORT EXPECTED	15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
☐ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)* ☑ NO				

ABSTRACT *(Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)*

At 1801 hours EST on 11/5/2013, with the Unit in Mode 1 at 19% power, H. B. Robinson Steam Electric Plant, Unit No. 2 (HBRSEP2) experienced an automatic reactor trip while operators were transferring loads from the Startup Transformer to the Unit Auxiliary Transformer (UAT), as part of activities associated with coming on line and increasing power following Refueling Outage 28. During the coordinated breaker operation, the reactor tripped when an anomaly occurred that resulted in the actuation of two undervoltage relays associated with the loss of 2 of 3 4KV buses. The 'A' Emergency Diesel Generator (EDG) auto-started and supplied the E-1 emergency bus as a result of the undervoltage transient. The event was reported as a 4 hour Non-Emergency report per 10 CFR 50.72 (b)(2)(iv)(B) due to the valid Reactor Protection System Actuation, and as an 8 hour Non-Emergency report per 10 CFR 50.72(b)(3)(iv)(A) due to the valid actuation of Auxiliary Feed Water and EDG auto-start and subsequent starting of required undervoltage loads, save 'A' Service Water Pump.

The investigation into the cause of this event determined that advanced aging/fatigue of the phenolic operating rods of the UAT breaker (52/7) caused the failure of the 'B' phase operating rod, which prevented closure of the 'B' phase of the breaker. The cause of the failure of the 'A' Service Water Pump to sequence onto the E-1 bus during the blackout sequence was attributed to a loose wire termination in the Emergency Control Station (ECS). Immediate corrective actions consisted of assessment of the switchyard and E-1 Bus, inspection of Breaker 52/7, and securing the loose wire termination in the ECS. This event did not impact the health and safety of the public.

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		YEAR	SEQUENTIAL NUMBER	REV NO.	
H. B. Robinson Steam Electric Plant, Unit No. 2	05000 261	2013	- 003	- 01	2 OF 3

NARRATIVE

PLANT IDENTIFICATION
Westinghouse - Pressurized Water Reactor

BACKGROUND
At 1801 hours EST on 11/5/2013, with the unit in Mode 1 at 19% power and no involvement of out-of-service structures, systems or components, H. B. Robinson Steam Electric Plant (HBRSEP2) experienced an automatic reactor [RCT] trip while operators were transferring loads from the Unit Startup Transformer (SUT) [XFMR] to the Unit Auxiliary Transformer (UAT) [XFMR], in accordance with plant procedures. During the coordinated breaker operation, the reactor tripped when an anomaly occurred that resulted in the actuation of two undervoltage relays [RLY] associated with the loss of 2 of 3 4KV buses [BU]. By design, the one running 'A' Main Feedwater Pump [P] tripped and caused the auto-start of the Auxiliary Feedwater System (AFW), which maintained Steam Generator [SG] water levels within the normal operating band. The 'A' Emergency Diesel Generator (EDG) [DG] auto-started and supplied the E-1 emergency bus [BU] as a result of the undervoltage transient.

The event was reported as a 4 hour Non-Emergency report per 10 CFR 50.72(b)(2)(iv)(B) due to the valid Reactor Protection System Actuation, and as an 8 hour Non-Emergency report per 10 CFR 50.72(b)(3)(iv)(A) due to the valid actuation of AFW and EDG auto-start and subsequent starting of required undervoltage loads, save 'A' Service Water (SW) Pump [P].

EVENT DESCRIPTION
At 1801 hours EST on 11/5/2013, with the Unit in Mode 1 at 19% power, HBRSEP2 experienced an automatic reactor trip while operators were transferring loads from the SUT to the UAT, in accordance with plant procedures. When the breaker (52/7) control switch [33] connecting the UAT to 4KV Bus 1 [BU] was taken to the 'close' position, indication on the Reactor Turbine Generator Board (RTGB) [MCBD] went from 'Open' to no indication. The 52/7 breaker's 'A' and 'C' phases closed, however 'B' phase operating rod had failed thereby inhibiting closure of the 'B' phase of the breaker. When the 52/7 control switch was returned to center position, breaker 52/12 [BKR], Incoming Line – Startup Transformer to 4KV Bus 2 [BU], tripped open per normal controls. When breaker 52/12 opened, the 'B' phase current on Reactor Coolant Pumps (RCPs) [P] 'A' and 'C' dropped from approximately 500 amps to 76 amps. As designed, the relays for buses 1 and 2 tripped at a setpoint equal to 75% of nominal bus voltage. Reactor trip signal logic directs a reactor trip when 2 out of 3 4KV bus undervoltage relays trip. As a result of the 'B' phase current drop on RCPs 'A' and 'C' and the subsequent trip of their associated undervoltage relays, an automatic reactor trip occurred. The turbine [TG] automatically tripped due to the reactor trip causing the turbine stop valves [V] to close while the switchyard generator output breakers [BKR] 52/8 and 52/9 remained closed. With the turbine stop valves closed while the switchyard breakers were closed, the 60-second time delay pick-up relays initiated and timed out resulting in Generator Lockout Relay 86BU [RLY] tripping and closing breaker 52/12. Normal voltage was restored to 4KV buses 1 and 2 when breaker 52/12 closed. The voltage drop on the 4KV buses was sufficient to pick up the E-1 Bus undervoltage relays and initiate the blackout sequence on Bus E-1. The 'A' EDG auto-started and the 'A' EDG load sequencer [PMC] loaded the required Engineered Safety Features loads onto E-1 with the exception of the 'A' SW Pump, which should have started as part of the sequence.

CAUSAL FACTORS
The root cause of the reactor trip event is advanced aging/fatigue of breaker 52/7 phenolic operating rod materials. Degradation of the rod materials caused failure of the 'B' operating rod, which prevented closing of breaker 52/7. The failure of the 'A' SW Pump to start was due to a loose wire termination in the Emergency Control Station. The loose termination was attributed to a historical condition based on maintenance records searches going back more than fifteen years that did not identify any work performed in this particular Emergency Control Station.

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H. B. Robinson Steam Electric Plant, Unit No. 2	05000 261	2013	- 003 -	01	3 OF 3

NARRATIVE

CORRECTIVE ACTIONS

Completed:

1. Interim corrective action to address rod failure involved inspection and or replacement of operating rods in all four incoming breakers that are part of the fast bus transfer logic between the UAT and SUT.
2. The 'A' SW pump loose terminal connection in the Emergency Control Station was secured.

Planned:

1. Long term corrective action to prevent recurrence consists of replacing the operating rods for all 4KV incoming breakers with high tensile strength operating rods. Action Request (AR) 642282, Assignment Nos. 05 & 06.

SAFETY ANALYSIS

The risk consequences of this event were minimal based on the successful reactor trip and operators' ability to successfully start SW Pump 'D' [P] after SW Pump 'A' failed to start as expected. SW Pump 'A' remained available by a manual start and the transient was not complicated by additional equipment failures, malfunctions or human errors.

The reactor was at 19% power, and power ascension following completion of the refueling outage was in progress. Therefore, the decay heat load was well below that of a full-power trip after a period of operation and therefore well within the capabilities of the decay heat removal systems. In addition, AFW actuation had been demonstrated successfully in the previous period of plant operation, and no corrective or preventive maintenance had been performed on that system since that time. The Condensate Storage Tank [TK] capacity was adequate to remove the above decay heat load for an extended period before requiring refill. Backup sources of feedwater were available and important accident mitigation equipment remained available throughout this event.

ADDITIONAL INFORMATION

An internal Operating Experience (OE) search for related events at HBRSEP2 was conducted; no similar events were identified. An evaluation of external OE shows that this event would not have been prevented by available OE, therefore this was not considered a missed opportunity to utilize OE.

Energy Industry Identification System (EIIS) codes for systems and components relevant to this event are identified in the text of this document within brackets [ ].

H. B. ROBINSON STEAM ELECTRIC PLANT UNIT 2

LICENSEE EVENT REPORT NO. 2014-001-01
REVISION TO REACTOR TRIP DUE TO A TWO-OUT-OF-THREE LOGIC SIGNAL FROM STEAM GENERATOR WATER LEVEL PROTECTION TRAIN B LOGIC MATRIX

NRC FORM 366 (02-2014) U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

(See Page 2 for required number of digits/characters for each block)

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
H. B. Robinson Steam Electric Plant, Unit No. 2	05000 261	1 OF 3

4. TITLE

Reactor Trip Due to a Two-out-of-Three Logic Signal from Steam Generator Water Level Protection Train B Logic Matrix

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER 05000
01	09	2014	2014	- 001 -	01	08	05	2014	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE: 1

10. POWER LEVEL: 100

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check all that apply)*

☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(i)(C)	☐ 50.73(a)(2)(vii)
☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
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☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(C)	☐ OTHER
☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

LICENSEE CONTACT	TELEPHONE NUMBER *(Include Area Code)*
R. Hightower, Manager - Nuclear Regulatory Affairs	(843) 857-1329

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
X	JC	CNTR	WSTGHSE	Y					

14. SUPPLEMENTAL REPORT EXPECTED	15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
☐ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)* ☑ NO				

ABSTRACT *(Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)*

At 2234 hours EST on 1/9/2014, with the Unit in Mode 1 at 100% power, H. B. Robinson Steam Electric Plant, Unit 2 experienced an automatic reactor trip/turbine trip during the performance of Steam Generator (SG) Water Level Protection Channel testing. The 'B' reactor trip breaker opened as a result of its 2/3 SG Lo-Lo Level input logic being satisfied. This occurred when one channel contact was "open" due to foreign material lodged between the contact faces and the second channel contact was opened during channel testing. The opening of the 'B' reactor trip breaker resulted in a turbine trip followed by a reactor trip. Auxiliary Feedwater automatically started as expected. There were no other equipment performance issues.

The cause of this event was degradation of passive components (wire labels) in the Reactor Protection System (RPS) relay rack. Degraded wire labels were the source of the foreign material which became lodged in an RPS relay contact creating an undetected half-trip condition.

The foreign material was removed from the RPS relay contact. Both trains of RPS were tested to verify proper functioning of each RPS relay, and both trains of RPS relay racks were inspected to confirm no foreign material was present which could affect proper operation of the RPS relays.

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NARRATIVE

PLANT IDENTIFICATION
Westinghouse - Pressurized Water Reactor

BACKGROUND
At 2234 hours EST on 1/9/2014, with the Unit in Mode 1 at 100% power and no structures, systems or components out of service that contributed to this event, H. B. Robinson Steam Electric Plant, Unit No. 2 (HBRSEP2) experienced an automatic reactor [RCT] trip during the performance of Steam Generator (SG) [SG] Water Level Protection Channel [CHA] testing. This event is reportable under 10 CFR 50.73(a)(2)(iv)(A) due to the event resulting in an automatic actuation of the Reactor Protection System (RPS) and Auxiliary Feedwater System (AFW).

The RPS monitors all parameters related to safe operation of the reactor. The system is designed to protect the core against fuel rod cladding damage caused by departure from nucleate boiling, and to protect the Reactor Coolant System (RCS) against damage caused by overpressure. The Low-Low SG Water Level Trip circuit protects the SG in case of a sustained steam/feedwater flow mismatch of insufficient magnitude to cause a flow mismatch reactor trip. This reactor trip circuit actuates on two-out-of-three low-low water level signals in any SG.

Maintenance Surveillance Test (MST)-013, "Steam Generator Water Level Protection Channel Testing," provides the instruction necessary for performing the Surveillance Test which will determine the operability of the SG Water Level Protection Channel Sets I, II, and III.

MST-021, "Reactor Protection Logic Train 'B' at Power," provides the instructions necessary to determine the operability of Reactor Protection Logic Train 'B'.

EVENT DESCRIPTION
At 2234 on 1/9/2014, with the Unit in Mode 1 at 100% power and during the performance of MST-013, HBRSEP2 experienced an automatic reactor trip. SG Water Level Channel III [CHA] was tested in accordance with MST-013 and returned to service upon verification that all bistables [RLY] and alarms [ALM] were cleared. However, unbeknownst to technicians, contact 2-6 [CNTR] on relay LC-496A1-X(B) [RLY] was in the half-tripped condition (i.e., two-out-of-three logic with one of the two needed channels already in the 'tripped' state) and not annunciated in the Control Room. When technicians proceeded with MST-013 placing SG Water Level Channel I [CHA] in the tripped condition, a two-out-of-three logic signal was generated causing the 'B' Reactor Trip Breaker [BKR] to open, causing the turbine [TRB] trip followed by a reactor trip signal which opened the 'A' Reactor Trip Breaker [BKR]. AFW automatically started as expected. There were no other equipment performance issues.

Circuit troubleshooting revealed that the Westinghouse LC-496A1-X(B) 2-6 contact ('B' train), while appearing to be in the correct position, had 133.2 volts direct current (VDC) across it (i.e., showing an open contact). This contact failure, coupled with the LC-494A1-X(B) contacts [CNTR] being open from the Channel I test, broke continuity to the relays for the 'B' train, which broke continuity to the 'B' reactor trip breaker [BKR] undervoltage coil [CL] and automatic shunt trip coil [CL], causing the 'B' reactor trip breaker to open. The trip of the 'B' reactor trip breaker initiated a turbine trip, as expected. Per design, a turbine trip results in a reactor trip at greater than 40% reactor power and led to the reactor trip breaker opening and subsequent reactor trip.

During a visual examination of relay LC-496A1-X(B) after being removed from the RPS relay rack [RK], it was discovered that a small piece of plastic material was wedged between the plates of its 2-6 contact. The material, determined to be a piece of a degraded wire label that had become dislocated from a wire, prevented the contacts from fully closing. Since relay LC-496A1-X(B) passed surveillance test MST-021 performed on 12/02/2013, it has been concluded that the foreign material fell into the relay contacts after this test and prior to MST-013 testing on 1/9/2014.

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NARRATIVE

CAUSAL FACTORS

The direct cause of the reactor trip event is foreign material preventing closure of contacts 2-6 of relay LC-496A1-X(B). This created an undetected condition of having one channel of the 'C' SG low-low level protection matrix in a tripped condition. When another channel was placed in the tripped condition for testing, the reactor trip breaker circuit was opened causing the reactor trip.

CORRECTIVE ACTIONS

Completed:

1. The foreign material was removed from the RPS relay contact.
2. Both trains of RPS relays were tested to verify satisfactory operation and that no foreign material was present that could prevent correct relay operation.
3. Both trains of relay racks were inspected to identify and remove any potential foreign material that could adversely affect proper operation of RPS relay contacts.
4. Revision of the model work order instructions for the relay rack clean-and-inspect preventive maintenance procedures to include inspection of wiring, labels, cable raceways and other passive components for evidence of degradation.

Planned:

1. Replacement of wire labels in reactor protection and safeguards relay racks. Action Request (AR) No. 654789 Assignment Nos. 06 & 29

SAFETY ANALYSIS

The reactor was operating at full power during the performance of MST-013 when an automatic reactor trip occurred. The mitigating equipment, including AFW functioned as expected and plant shutdown proceeded normally without further challenge. The cause of the plant trip was determined to be foreign material lodged between two contacts utilized during the performance of MST-013. The trip logic performed as designed and there were no other equipment performance issues. Therefore, the risk consequence of this event was small based on a successful reactor trip with no equipment or operational challenges.

ADDITIONAL INFORMATION

A search of internal and external operating experience for the previous three years did not identify any events that could have prevented the failure of relay contacts due to degrading wire labels.

Energy Industry Identification System (EIIS) codes for systems and components relevant to this event are identified in the text of this document within brackets [ ].