Entergy

Indian Point Energy Center
450 Broadway, GSB
P.O. Box 249
Buchanan, N.Y. 10511-0249
Tel (914) 254-6700

John A. Ventosa
Site Vice President
Administration

NL-14-126

October 10, 2014

U.S. Nuclear Regulatory Commission
Document Control Desk
11545 Rockville Pike, TWFN-2 F1
Rockville, MD 20852-2738

SUBJECT: Licensee Event Report # 2014-004-00, "Automatic Reactor Trip as a Result of Meeting the Trip Logic for Over Temperature Delta Temperature During Reactor Protection System Pressurizer Pressure Testing"
Indian Point Unit No. 3
Docket No. 50-286
DPR-64

Dear Sir or Madam:

Pursuant to 10 CFR 50.73(a)(1), Entergy Nuclear Operations Inc. (ENO) hereby provides Licensee Event Report (LER) 2014-004-00. The attached LER identifies an event where the reactor automatically tripped, which is reportable under 10 CFR 50.73(a)(2)(iv)(A). As a result of the reactor trip, the Auxiliary Feedwater System was actuated, which is also reportable under 10 CFR 50.73(a)(2)(iv)(A). This condition was recorded in the Entergy Corrective Action Program as Condition Report CR-IP3-2014-01903.

There are no new commitments identified in this letter. Should you have any questions regarding this submittal, please contact Mr. Robert Walpole, Manager, Regulatory Assurance at (914) 254-6710.

Sincerely,

JAV/cbr

cc: Mr. David Lew, Acting Regional Administrator, NRC Region I
NRC Resident Inspector's Office, Indian Point 3
Ms. Bridget Frymire, New York State Public Service Commission

IE22
NRR

NRC FORM 366 (01-2014) U.S. NUCLEAR REGULATORY COMMISSION

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

LICENSEE EVENT REPORT (LER)

1. FACILITY NAME: INDIAN POINT 3

2. DOCKET NUMBER: 05000-286

4. TITLE: Automatic Reactor Trip as a Result of Meeting the Trip Logic for Over Temperature Delta Temperature during Reactor Protection System Pressurizer Pressure Calibration

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE		
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV. NO.	MONTH	DAY	YEAR
8	13	2014	2014	004 -	00	10	10	2014

8. OTHER FACILITIES INVOLVED

FACILITY NAME	DOCKET NUMBER
	05000
	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)
☐ 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

NAME	TELEPHONE NUMBER (Include Area Code)
Robin Daley, Supervisor Engineering Systems-Electrical/I&C	(914) 254-6817

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

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX

14. SUPPLEMENTAL REPORT EXPECTED

☒ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)* ☐ NO

15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
	4	30	2015

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

On August 13, 2014, Instrumentation and Control (I&C) technicians started performance of an 8-hour scheduled surveillance 3-PC-OL04A (Pressurizer Pressure Loop P-455 Channel Calibration) with Loop I in test and tripped. The test was approved by I&C and operations to be stopped for a break with the bistables still tripped, Channel I in test. During the break, an automatic reactor trip (RT) occurred as a result of meeting the 2/4 trip logic for Overtemperature Delta Temperature (OTDT). All control rods fully inserted and all required safety systems functioned properly. The plant was stabilized in hot standby with decay heat being removed by the main condenser. The Auxiliary Feedwater System automatically started as expected due to SG low level from shrink effect. The Emergency Diesel Generators did not start as offsite power remained available and stable. No work was being performed at the time of the RT and no actual OTDT existed. Event investigation indicates the trip was spurious. The direct cause was a spurious spiking signal in the OTDT circuitry with another channel tripped for testing. The root cause is unknown at this time. Further troubleshooting and monitoring of the OTDT circuit will be required. The LER will be revised upon completion of troubleshooting and evaluation. Corrective actions included replacement of three bistables. Removed bistables were sent to a vendor for a failure analysis. I&C personnel will be briefed on management's expectation on minimizing break times when a channel is tripped. The event had no effect on public health and safety.

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Indian Point Unit 3	05000-286	2014	- 004 -	00	2 OF 5

NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* (17)

Note: The Energy Industry Identification System Codes are identified within the brackets {}.

DESCRIPTION OF EVENT

On August 13, 2014, during 100% steady state reactor power, Instrumentation and Control (I&C) technicians started performance of an 8-hour scheduled surveillance 3-PC-OL04A (Pressurizer Pressure Loop P-455 Channel Calibration) with Loop I in test and tripped. Loop I was left in test and tripped during an approved break by I&C and operations. During the break, an automatic reactor trip (RT) {JC} occurred at 11:57 hours as a result of meeting the 2/4 trip logic for the Reactor Protection System (RPS) {JC} function Overtemperature Delta Temperature (OTDT). All control rods {AA} fully inserted and all required safety systems functioned properly. The plant was stabilized in hot standby (Mode 3) with decay heat being removed by the main condenser {SG}. The Auxiliary Feedwater System {BA} automatically started as expected due to SG low level from shrink effect. The Emergency Diesel Generators {EK} did not start as offsite power remained available and stable. No work was being performed at the time of the RT and no actual OTDT existed. The event was recorded in the Indian Point Energy Center corrective action program (CAP) as CR-IP3-2014-01903. A post trip evaluation was completed on August 14, 2014.

The OTDT trip function is provided to ensure that the design limit Departure from Nucleate Boiling Ratio (DNBR) is met. The OTDT trip prevents the power density anywhere in the core from exceeding 118% of design power density. The inputs to the OTDT trip include pressure, coolant temperature, axial power distribution, and reactor power as indicated by loop delta Temperature assuming full reactor power. Four different temperature channels are used, one for each coolant loop. The OTDT circuitry consists of four independent channels that feed the delta Temperature (dT) and dT setpoint signals into dual bistables which drive the reactor protection relays. Contacts from these relays are connected in the proper matrix for the reactor trip relays. A trip actuation requires a two out of four logic. The indicated loop dT is used in the RPS as a measure of reactor power. This is compared with a setpoint that is automatically calculated dependent on T(avg), pressurizer pressure, and axial power tilt. When the dT signal exceeds the calculated setpoint, the affected channel is then tripped.

On August 13, 2014, at 8:50 hours, I&C Technicians started 3-PC-OL04A (Pressurizer Pressure Loop P-455 channel Calibration) for performance of a pressurizer pressure calibration. Operations entered the following Technical Specifications (TS): TS 3.3.2 (ESFAS Instrumentation) Function 7 (ESFAS Interlock-Pressurizer Pressure) Condition K (One or More Channels Inoperable), TS 3.3.1 (RPS Instrumentation) Function 5 (OTDT) and 7b (Pressurizer Pressure High) both Condition E (One Channel Inoperable), TS 3.3.2 (ESFAS Instrumentation) Function 1d (Pressurizer Pressure-Low0 Condition D (One Channel Inoperable). Required actions (Channels placed to trip) performed in body of test. Entered TS 3.3.4 PAOT Function 2a from TS Basis Table 3.3.4-1 (Remote Shutdown-Pressurizer Pressure) for calibration of Loop P-455 bistables. The test places protection channel 1 in trip and has a scheduled duration of 8 hours. Due to the long duration of the test and vulnerability to un-trip and re-trip the protection channel, it is accepted work practice to leave a channel in trip for break periods. During an approved break from calibration, with all test equipment removed, bistables still tripped, Channel I in test, a RT occurred from Channel I of OTDT.

At 11:57 hours, Control Room Operators observed First Out Alarm OTDT RT/Turbine Trip. Entered procedure 3-E-0 (Reactor Trip or Safety Injection) and transitioned to 3-ES-0.1 (Reactor Trip Response). At 13:32 hours, operators transitioned to POP 3.2.

A review of the Sequence of Events (SOE) log showed that the OTDT relay tripped and then reset within 30 milliseconds. The bistable status lights did not indicate which of the other three channels entered a trip condition due to the short nature of the signal. The trip logic make-up in another channel was considered spurious. The SOE log also shows that the RT Breakers (RTB) A and B both opened within 8 milliseconds of each other, meaning a signal was sent to both trains simultaneously. Subsequent investigation concluded no actual OTDT condition existed, as there were no changes in delta temperature, T(Ave), axial power tilt, pressurizer pressure or measured reactor power level prior to the OTDT trip.

A FMA was developed to determine the possible causes to investigate during troubleshooting. A Work order was developed to perform troubleshooting of the OTDT trip. Initial troubleshooting check was performed on relays RT3 and RT4 with satisfactory results. Resistance checks were performed on SIAM 1-X and 2-X with satisfactory results. This result eliminated high contact resistance on the safety injection relay contacts or failure of RT-3 and RT-4. Further troubleshooting focused on the OTDT circuitry. OTDT trip bistables checked satisfactory per 3-PT-Q87B. Although all of the bistables were checked satisfactorily, the FMA identified that spurious operation of a bistable as a possible cause. To eliminate this cause, all of the suspected channel bistables were replaced (TC-421A/B Foxboro, TC-431A/B NUS, and TC-441A/B Foxboro). The ripple/noise was checked for setpoint generators (TM-422B, TM-432B and TM-442B). The M20 pin connections for the three setpoint generators were checked and all pins were satisfactory. Three bistables were replaced and the removed bistables bench tested and one (TC-431A/B) intermittently tripped the GFI. The AC noise at the output of the average THL was measured and the results indicated a possible loose wire connection or card problem. This finding will be evaluated further in the Failure modes analysis (FMA). Five days after the RT, the OTDT trip or Rod Stop Control Room (CR) alarm was actuated (CR-IP3-2014-01953 recorded event). Monitoring of the OTDT logic relays was added to the action plan and the alarm can for OTDT Channel trip or Rod stop CR alarm was replaced. On September 6, 2014, at 22:16 hours, CR-IP3-2014-02111 recorded that a OTDT Channel Trip or Rod stop CR alarm was received. Monitoring of OTDT logic relays did not capture any change of state.

An extent of condition investigation determined that Unit 3 is susceptible to spurious OTDT trip during RPS testing which reduces the trip logic to 1 out of 3. Unit 2 does not have this vulnerability as the Unit 2 RPS channels have a test bypass feature which maintains the trip logic and only reduces the number of channels.

The Cause of Event

The direct cause of the RT was a spurious spiking signal in the OTDT circuitry with another channel tripped for testing. One channel was made up for testing, then a spurious intermittent spike caused another channel to make up thereby satisfying the trip logic. No direct cause of equipment failure mechanism which is generating the spurious channel spiking has been confirmed. The origin of the spiking is limited to the analog or process OTDT channel 2, 3, 4. Because RPS logic make-up was caused by a spurious intermittent spike, the trip would still have occurred even if the surveillance was worked continuously. However, it was determined that the site has become de-sensitized to working surveillances at unit 3 in which there is an increased risk for a RT due to reduced RPS logic.

NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* (17)

The root cause is unknown at this time. Further troubleshooting and monitoring of the OTDT circuit will be required to determine the root cause. Channel 4 has a bypass capability. Channel 1 for OTDT was tripped at the time for surveillance testing. The Unit 3 design does not have the capability to place a channel in bypass except for channel 4. The condition results in a trip risk during surveillance testing since the reactor protection logic is reduced from a 2 out of 4, to a 1 out of 3 trip signals required for a RT.

Corrective Actions

The following corrective actions have been or will be performed under the Corrective Action Program (CAP) to address the causes of this event.

- Three bistables.TC-421A/B (Channel II-Foxboro), TC-431A/B (Channel III-NUS), and TC-441A/B (Channel IV-Foxboro) were replaced with three NUS bistables.
- The removed bistables were sent to a vendor for a failure analysis.
- Temporary procedure changes will be issued to maintain logic relays energized when an associated analog OTDT channel is placed in trip for reactor protection quarterly surveillance tests. The affected tests are 3-PT-Q87A/B/C, 3-PC-Q109A/B/C, and 3-PT-Q95A/B/C.
- The existing troubleshooting work order (WO) was revised to add tasks to validate/eliminate the remaining possible causes in the Failure Modes Analysis.
- LER-2014-004 will be revised based on the revised RCA per input from troubleshooting, FMA, vendor evaluation and monitoring results.
- The ODMI for OTDT will be revised to include monitoring of OTDT bistables for each channel using the SOE recorder.
- I&C personnel will be briefed on management's expectation on minimizing break times when a channel is tripped.

Event Analysis

The event is reportable under 10CFR50.73(a)(2)(iv)(A). The licensee shall report any event or condition that resulted in manual or automatic actuation of any of the systems listed under 10CFR50.73(a)(2)(iv)(B). Systems to which the requirements of 10CFR50.73(a)(2)(iv)(A) apply for this event include the Reactor Protection System (RPS) including RT and AFWS actuation. This event meets the reporting criteria because an automatic RT was initiated at 11:57 hours, on August 13, 2014, and the AFWS actuated as a result of the RT. On August 13, 2014, a 4-hour non-emergency notification was made to the NRC at 13:06 hours, for an actuation of the reactor protection system {JC} while critical and included an 8-hour notification under 10CFR50.72(b)(3)(iv)(A) for a valid actuation of the AFW System (Event Log #50361). As all primary safety systems functioned properly there was no safety system functional failure reportable under 10CFR50.73(a)(2)(v).

Past Similar Events

A review was performed of the past three years for Licensee Event Reports (LERs) reporting a RT as a result of testing. No LERs were identified.

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Indian Point Unit 3	05000-286	2014	- 004	- 00	5 OF 5

NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* **(17)**

Safety Significance

This event had no effect on the health and safety of the public.
There were no actual safety consequences for the event because the event was an uncomplicated reactor trip with no other transients or accidents. Required primary safety systems performed as designed when the RT was initiated. The AFWS actuation was an expected reaction as a result of low SG water level due to SG void fraction (shrink), which occurs after a RT and main steam back pressure as a result of the rapid reduction of steam flow due to turbine control valve closure. For this RT there was no actual OTDT condition.

There were no significant potential safety consequences of this event. The RPS is designed to actuate a RT for any anticipated combination of plant conditions to include low SG level. The reduction in SG level and RT is a condition for which the plant is analyzed. A low water level in the SGs initiates actuation of the AFWS. Redundant safety SG level instrumentation was available for a low SG level actuation which automatically initiates a RT and AFWS start providing an alternate source of FW. The AFW System has adequate redundancy to provide the minimum required flow assuming a single failure. The analysis of a loss of normal FW (UFSAR Section 14.1.9) shows that following a loss of normal FW, the AFWS is capable of removing the stored and residual heat plus reactor coolant pump waste heat thereby preventing either over pressurization of the RCS or loss of water from the reactor. All components in the RCS were designed to withstand the effects of cyclic loads due to reactor system temperature and pressure changes. For this event, rod control was in automatic and all rods inserted upon initiation of a RT. The AFWS actuated and provided required FW flow to the SGs. RCS pressure remained below the set point for pressurizer PORV or code safety valve operation and above the set point for automatic safety injection actuation. Following the RT, the plant was stabilized in hot standby.