Entergy

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Fred Dacimo
Site Vice President
Administration

April 30, 2007
Indian Point 2
Docket No. 50-247
NL-07-044

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Stop O-P1-17
Washington, D.C. 20555-0001

Subject: Licensee Event Report # 2007-004-00, "Manual Reactor Trip Due to Decreasing Steam Generator Levels Caused by Loss of Feedwater Flow as a Result of Feedwater Pump Suction Pressure Transmitter Power Supply Failure"

Dear Sir or Madam:

Pursuant to 10 CFR 50.73(a)(1), Entergy Nuclear Operations Inc. (ENO) hereby provides Licensee Event Report (LER) 2007-004-00. The enclosed LER identifies an event where the reactor was manually tripped, which is reportable under 10 CFR 50.73(a)(2)(iv)(A) . This condition has been recorded in the Entergy Corrective Action Program as Condition Report CR-IP2-2007-01046.

There are no new commitments identified in this letter. Should you have any questions regarding this submittal, please contact Mr. T. R. Jones, Manager, IPEC Licensing at (914) 734-6670.

Sincerely,

Fred R. Dacimo
Site Vice President
Indian Point Energy Center

cc: Mr. Samuel J Collins, Regional Administrator, NRC Region I
NRC Resident Inspector's Office, Indian Point 2
Mr. Paul Eddy, New York State Public Service Commission
INPO Record Center

IE22

NRC FORM 366 (6-2004) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104 EXPIRES: 6/30/2007

Estimated burden per response to comply with this mandatory collection request: 50 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records and FOIA/Privacy Service Branch (T-5 F52), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-I0202, (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.

LICENSEE EVENT REPORT (LER)

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
INDIAN POINT 2	05000-247	1 OF 4

4. TITLE Manual Reactor Trip Due to Decreasing Steam Generator Levels Caused by Loss of Feedwater Flow as a Result of Feedwater Pump Suction Pressure Transmitter Power Supply Failure

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
2	28	2007	2007-	004 -	00	04	30	2007	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)
☐ 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)*
Christopher Ingrassia, System Engineer	(914) 271-7047

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
X	JK	JX	F180	Y					

14. SUPPLEMENTAL REPORT EXPECTED

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

15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR

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

On February 28, 2007, control room operators manually initiated a reactor trip (RT) at approximately 0633 hours, after receiving speed control trouble alarms for the 21 and 22 Main Boiler Feedwater Pumps (MBFPs) and observing decreasing MBFP speed and steam generator (SG) levels. 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. There was no radiation release. The Emergency Diesel Generators did not start as offsite power remained available. The Auxiliary Feedwater System automatically started as expected due to Steam Generator low level from shrink effect. The cause of the RT was loss of main feedwater (FW) flow due to the failure of the power supply (PQ-408B) for the MBFP suction pressure transmitter (PT-408B). The power supply failed due to a failure of its filter capacitors as a result of age degradation. The root cause of the power supply failure was insufficient verification of the existing plant programs to address capacitor age degradation due to human error. Significant corrective actions include replacement of power supply PQ-408B and pressure transmitter PT-408B. An instrument power supply Preventive Maintenance (PM) will be implemented in accordance with the ENS PM template, and a capacitor program will be developed and implemented to address age degradation of capacitors. The event had no effect on public health and safety.

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Indian Point Unit 2	05000-247	2007	- 04 -	00	2 OF 4

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 February 28, 2007, while at approximately 100% steady state reactor power, control room operators manually initiated a reactor trip (RT) {JC} at approximately 0633 hours, after receiving trouble alarms {ALM} for the 21 and 22 Main Boiler Feedwater Pump (MBFP) {SJ} speed control {JK} and observing decreasing MBFP speed and steam generator (SG) {AB} levels. The Balance of Plant (BOP) reactor operator noticed the MBFP suction pressure indicator lowering to near zero. All control rods {AA} 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 {SG}. There was no radiation release. The Emergency Diesel Generators {EK} did not start as offsite power remained available. The Auxiliary Feedwater System {BA} automatically started as expected due to Steam Generator low level from shrink effect. An investigation into the cause of the event and a post transient evaluation was initiated.

Prior to the event all Control Rods were fully withdrawn from the reactor core and in Auto, both Main Boiler Feedwater Pumps (MBFPs) were in service, Auxiliary Feedwater Pumps (AFWPs) were in standby, the Emergency Diesel Generators were in standby, and off-site power was in service. After the manual RT, Control Room operators entered emergency operating procedure (EOP) E-0, "Reactor Trip or Safety Injection," and transitioned to EOP ES-0.1, "Reactor Trip Response," and stabilized the plant to hot standby using normal plant operating procedures. On February 28, 2007, at 1001 hours, a four hour non-emergency notification was made to the NRC (Log Number 43199) for a reactor trip while critical and the notification included the eight hour non-emergency notification for actuation of the AFW system. The RT was reported under 10 CFR 50.72(b)(2)(iv)(B) and the AFW actuation reported under 10 CFR 50.72(b)(3)(iv)(A). The event was recorded in the Indian Point Energy Center corrective action program (CAP) as CR-IP2-2007-01046.

Instrumentation and Control (I&C) personnel performed troubleshooting of MBFP suction pressure transmitter PT-408B and found that the power supply (PQ-408B) for PT-408B had failed. The power supply was replaced and although the pressure transmitter (PT-408B) was functioning properly it was also replaced. During normal plant operations the speed of each turbine driven MBFP is controlled automatically by its own auto-manual speed controller. Each MBFP speed signal is compared to a MBFP suction pressure signal by low current selectors which choose the lower of the two signals to pass on to the MBFP speed changers. Pressure transmitter PT-408B senses MBFP suction pressure and sends a signal to pressure controller PC-409A which is programmed to generate a full output signal (50 milliamps) for any suction pressure above 265 psi and ramp down to a minimum output (10 milliamps) for a suction pressure of 230 psi. During normal operation, the MBFP suction pressure is approximately 370 psi, but if the MBFP suction pressure falls below 265 psi, the pressure signal will begin decreasing until it falls below the value of the FW speed signal. At that point, the low current selector will pass the suction pressure signal to the signal processor thereby limiting MBFP speed to maintain suction pressure. This design protects the MBFPs from cavitation by limiting speed to maintain suction pressure above a conservative limit. The failure of power supply PQ-408B resulted in the signal from MBFP suction pressure transmitter (PT-408B) failing low (11 milliamps) which resulted in the MBFP suction pressure signal being lower than the MBFP speed signals.

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Indian Point Unit 2	05000-247	2007	- 04 -	00	3 OF 4

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

The low current selectors sensed the condition as low MBFP suction pressure and cutback the speed of both MBFPs. The power supply (PQ-408B) is a Foxboro {F180} Model 610AR-0 determined to be original plant equipment.

A review of power supply and capacitor aging and obsolescence program history identified a number of condition reports (CR) on Foxboro equipment and capacitor failures with corrective actions (CA) generated to address age degradation of capacitors and failures of Foxboro equipment. In September 1999, CRs were initiated to develop a capacitor program and the need for a power supply Preventive Maintenance (PM) program. In 2002, CAs were identified to develop and implement a capacitor program but were closed in May 2003 based on a response that the existing drift monitoring, calibration and PM programs adequately addressed the issue of capacitor aging and/or replacement. The conclusion drawn in the response to the CR was insufficient since it failed to verify that aging capacitors, which were not being tested by existing plant programs, were being addressed. A CA to a 2002 CR which generated a PM Change Request (PMCR) to periodically replace power supplies was cancelled based on a conclusion the change was historical and contained insufficient information. Existing plant PM did not adequately address Foxboro units.

Cause of Event

The direct cause of the RT was loss of main feedwater (FW) flow. The cause of the low FW flow was failure of the power supply (PQ-408B) for the MBFP suction pressure transmitter (PT-408B). The power supply failed due to a failure of its filter capacitors (C1 and C2) as a result of age degradation. The root cause of the power supply failure was insufficient verification of the existing plant programs to address capacitor age degradation due to human error. The cause analysis determined that the verification of existing plant programs to address capacitor age degradation was insufficient. There was a failure to verify that aging capacitors, which were not tested by existing programs, were being addressed. Power supply PQ-408B was not being tested by existing programs. The insufficient verification in response to CR-IP2-2002-08120 was due to lapses in technical rigor. Issues with lapses in technical rigor have been identified through INPO and self assessments which were addressed by completing a technical rigor improvement plan (2006).

Corrective Actions

The following corrective actions have been or will be performed under Entergy's Corrective Action Program to address the cause and prevent recurrence

- Replacement of power supply PQ-408B and pressure transmitter PT-408B has been completed.
- An Instrument Power Supply PM will be developed and implemented at IPEC in accordance with the Entergy Nuclear South (ENS) PM Template. Actions are scheduled for completion by August 2007.
- A capacitor program will be developed and implemented. Actions are scheduled for completion by August 2007.

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Indian Point Unit 2	05000-247	2007	- 04 -	00	4 OF 4

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

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 a manual RT was initiated at 0633 hours, on February 28, 2007, and the AFWS actuated as a result of the RT. The failure of the MBFP suction pressure transmitter power supply PQ-408B did not result in the loss of any safety function. Therefore, there was no safety system functional failure reportable under 10CFR50.73(a)(2)(v).

PAST SIMILAR EVENTS

A review of the past two years of Licensee Event Reports (LERs) for events that involved a RT from loss of a power supply was performed. There were two LERs identified that reported a RT due to a power supply failure. LER-2006-003 reported a manual RT due to a mismatch between reactor power and turbine load caused by cycling of steam dump valves after a power reduction for loss of Heater Drain Tank (HDT) pumps. The cause of the loss of the HDT pumps was a failed HDT level controller power supply. LER-2006-003 was initiated by a level controller whose aging power supply failed. LER-2006-005 reported an automatic RT due to a turbine trip as a result of a Main Generator Exciter protective trip caused by a Generrex Power Supply loss. The power supply failed due to a loose terminal screw where the grounds are mounted. The cause of the event for LER-2006-005 was a loose power supply connection associated with a degraded power supply due to high resistance connections as a result of oxidation residue (aging).

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 RT with no other transients or accidents. Required primary safety systems performed as designed when the RT was initiated. There were no risk related components out of service at the time of the RT. The AFWS actuation was expected as a result of low SG water level due to SG void fraction (shrink), which occurs after automatic RT from full load. There were no significant potential safety consequences of this event under reasonable and credible alternative conditions. The AFWS actuated and provided required FW flow to the SGs. There are two motor driven AFW pumps and one steam driven pump, any one of which could provide the minimum required FW flow to the SGs. Main FW remained available but due to the cutback signal from PT-408B (failed low) was at minimum flow. 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.