Entergy Nuclear Northeast
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Fred Dacimo
Site Vice President
Administration

August 28, 2006

Re: Indian Point Unit 3
Docket 50-286
NL-06-086

Document Control Desk
U.S. Nuclear Regulatory Commission
Mail Stop O-P1-17
Washington, DC 20555-0001

Subject: **IP-3 LER# 2006-001-000, "Reactor Trip as a result of a Main Generator Trip due to Short in the Generator Differential Protection Circuit". Event Date: 7/6/06**

Dear Sir or Madam:

The attached Licensee Event Report (LER) 2006-001-00 is the follow-up written report submitted in accordance with 10 CFR 50.73. This event is of the type defined in 10 CFR 50.73(a)(2)(iv)(A) for an event recorded in the Entergy corrective action process as Condition CR-IP3-2006-02071.

There are no commitments contained in this letter. Should you or your staff have any questions regarding this matter, please contact Mr. Patric W. Conroy, Manager, Licensing, Indian Point Energy Center at (914) 734-6668.

Very truly yours,

Fred R. Dacimo
Site Vice President
Indian Point Energy Center

cc: next page

IE22

Attachment: LER-2006-001-00

cc:

Mr. Samuel J. Collins, Regional Administrator, Region I
IPEC NRC Resident Inspector's Office, Indian Point Unit 3
Mr. Paul Eddy, New York State Department of Public Service
INPO Record Center

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

APPROVED BY OMB: NO. 3150-0104 EXPIRES: 06/30/2007

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)

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4. TITLE: Reactor Trip as a result of a Main Generator Trip due to Short in the Generator Differential Protection Circuit

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
7	6	2006	2006 -	001 -	00	8	28	2006		05000
									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)
- ☐ 20.2201(d)
- ☐ 20.2203(a)(1)
- ☐ 20.2203(a)(2)(i)
- ☐ 20.2203(a)(2)(ii)
- ☐ 20.2203(a)(2)(iii)
- ☐ 20.2203(a)(2)(iv)
- ☐ 20.2203(a)(2)(v)
- ☐ 20.2203(a)(2)(vi)
- ☐ 20.2203(a)(3)(i)
- ☐ 20.2203(a)(3)(ii)
- ☐ 20.2203(a)(4)
- ☐ 50.36(c)(1)(i)(A)
- ☐ 50.36(c)(1)(ii)(A)
- ☐ 50.36(c)(2)
- ☐ 50.46(a)(3)(ii)
- ☐ 50.73(a)(2)(i)(A)
- ☐ 50.73(a)(2)(i)(B)
- ☐ 50.73(a)(2)(i)(C)
- ☐ 50.73(a)(2)(ii)(A)
- ☐ 50.73(a)(2)(ii)(B)
- ☐ 50.73(a)(2)(iii)
- ☒ 50.73(a)(2)(iv)(A)
- ☐ 50.73(a)(2)(v)(A)
- ☐ 50.73(a)(2)(v)(B)
- ☐ 50.73(a)(2)(v)(C)
- ☐ 50.73(a)(2)(v)(D)
- ☐ 50.73(a)(2)(vii)
- ☐ 50.73(a)(2)(viii)(A)
- ☐ 50.73(a)(2)(viii)(B)
- ☐ 50.73(a)(2)(ix)(A)
- ☐ 50.73(a)(2)(x)
- ☐ 73.71(a)(4)
- ☐ 73.71(a)(5)
- ☐ OTHER

Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

NAME	TELEPHONE NUMBER *(Include Area Code)*
Vincent Andreozzi, Engineering Supervisor- Electrical	(914) 734-6816

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
B	TB	CBL1	R352	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 July 6, 2006, at 0352 hours an automatic reactor trip (RT) occurred as a result of a short circuit that occurred in the junction box of the Main Generator output phase 'B' differential protection current transformer (CT). The plant responded to the trip as designed. The Auxiliary Feed Water System (AFW) actuated and the AFW Pumps started as designed due to the Steam Generator (SG) shrink caused by the trip from 100% power. All plant systems operated normally with the exception 31 Main Boiler Feed Pump Recirculation Valve did not open due to a failed solenoid and the 32 Main Steam Line Trap was isolated due to having a minor steam leak. The root cause of this condition was determined to be design inadequacy in material application. The abrasive property of the conductor's insulation jacket was not evaluated for use in a continuous vibration environment. Corrective actions included repair of damaged cables, coaching of personnel on the lessons learned, process/procedure revisions to improve the considerations on material selection in the design process and training enhancements to address material considerations. This event had no effect on public health and safety.

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

DESCRIPTION OF EVENT:

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

On July 6, 2006, at 0352 hours, while operating at 100 percent steady state power, an automatic reactor trip (RT){JC} occurred due to a main turbine generator (MG){TB} trip as a result of a main generator protection circuit trip {EL}caused by a `short circuit in the junction box {JBX}of the Main Generator output phase 'B' differential protection current transformer (CT){XCT}.

The plant responded to the RT as designed. All control rods {AA} fully inserted and all primary 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 Auxiliary Feed Water System (AFWS) {BA} actuated and the AFW Pumps started as designed due to the Steam Generator (SG) {AB} low level as a result of shrink effect caused by the RT from 100% power. The minor problems noted during the trip were the 31 Main Boiler Feed Pump {SJ} Recirculation Valve did not open due to a failed solenoid {SOL}; and the 32 Main Steam {SB} Line Trap was isolated due to having a minor steam leak.

A 4-hour non-emergency notification was made to the NRC for actuation of the reactor protection system {JC} while critical under 10CFR50.72(b)(2)(iv)(B), and an 8-hour notification on July 6, 2006, at 0938, for a valid actuation of the AFWS under 10CFR50.72(b)(3)(iv), Event Log number 42687. The event was recorded in the IPEC corrective action program (CAP) as CR-IP3-2006-02071.

The Turbine Trip First Out Annunciator received in the control room was the Primary Lock Out Relay. Subsequent actions for the RT included a visual inspection of all Main Generator differential protection circuit connections at the CT's, Control Room and WS8 terminal box. The inspection found wires G2 and GNO in the junction box of the Main Generator output phase 'B' differential protection CT in contact with each other. At the location of contact the wires crossed over one another. The wires were separated and exposed conductors were found on both wires. The insulation on both wires was worn away. The two wires had matching wear marks indicting that the wires had been vibrating against each other. Additional wires in the junction box were found with damaged insulation due to vibration but did not have exposed conductors or were not in contact with each other. The insulation damage was also found to have been caused by the metal cable tags and the string holding the cable tags to the wire. Wire insulation damage was also found in Main Generator output phase 'C' differential protection CT junction box.

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A post transient evaluation was performed on July 06, 2006. The evaluation determined that the short circuit was located between Current Transformer (CT)wires G2 (X1) and GNO (X2/Ground). The short circuit caused Main Generator differential protection relay 87G to sense a current differential on phase 'B' of the Main Generator and trip. Relay 87G tripped Main Generator primary lockout relay 86P, which automatically tripped the Main Generator and Turbine (TT). The Main Turbine trip automatically tripped the Reactor. The turbine trip was caused by the actuation of the primary lockout relay 86P device on Main Turbine Generator. The relay was actuated by the main generator differential relay (87G) trip, which is one of the inputs to the Main Turbine Trip logic.

The wires with damaged insulation were repaired with electrical tape and the CT circuit was re-tested satisfactorily. The testing consisted of an insulation resistance test and an impedance check of the CT circuits. After the CT circuit was repaired the plant startup was commenced. The Main Generator was synchronized to the grid on July 7, 2006 at 1440 and 100 percent power was achieved on July 8, 2006 at 0130 hours.

CAUSE OF EVENT

The direct cause of the RT was a short circuit of MG output phase 'B' differential protection CT. The short circuit occurred due to two wires in the same junction box of the differential protection CT crossing and vibrating against each other. The vibration and abrasiveness of the cable insulating system caused the insulation to wear through down to the conductors. This failure occurred under the normal vibration of the Main Generator.

The root cause of this event was design inadequacy in material application. The original plant CT wiring was modified in 1993 to upgrade the temperature rating. The abrasive property of the conductor's insulation jacket was not evaluated for use in a continuous vibration environment. This resulted in insulation wear and a plant trip due to the short circuit.

The design change which was implemented in 1993, did not take into account the vibration environment in which the new CT wiring was being installed. A review of the former Unit 3 modification procedures found that the designers were not prompted to consider this factor and instead focused on the heat-resistance properties of the cable. The focus was on temperature effects as the purpose of the modification was to replace the cable due to heat degradation. In addition, the manufacture's, Rockbestos Micatemp, cable specification did not specifically address cable application in an environment exposed to constant vibration. If a cable with an abrasion resistant jacket had been installed the vibration would not have caused the insulation to wear through.

A review of the current design considerations checklist in the IPEC Design Considerations procedure revealed that the checklist does not contain a consideration for cable installation in an environment exposed to constant vibration.

Evidence of Insufficient Details in a Procedure to Perform the Task was identified as a contributing cause. The cable was installed in accordance with Electrical Workmanship Standard. The installation documents did not contain specific guidance on cable installation in an environment exposed to constant vibration. In addition the cable manufacture does not have documented recommendations for installing this type of cable.

If the cable had been installed so the wires did not contact each other where they crossed the vibration would not have caused the insulation to wear through.

A review of the current Electrical Cable Installation Standard revealed that the standard does not contain guidance for cable installation in an environment exposed to a constant vibration.

CORRECTIVE ACTIONS

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

- Immediate corrective actions were implemented to repair the damaged wires by installing electrical tape over all the degraded insulation area of the wire. Post work tests were performed comprised of surveillance testing for the CTs and meggering of the wires. All Main Generator CT junction boxes were inspected for damaged wiring as an extent of condition.
- All affected departments including Design, Maintenance, Planning/Work Control, and Construction Services will discuss the lessons learned regarding the Root Cause at department level meetings.
- Corrective actions were initiated to revise the Fleet Design Engineering procedures and IPEC Electrical Cable Installation Standard, as applicable, to consider vibration degradation during the design of modifications. Tracking items were initiated in the work control process for a modification to replace the existing degraded wiring in the CT junction boxes through to completion. Inspection work orders were created for 31 & 32 Main Transformers, Station Aux Transformer and Unit Auxiliary Transformer, respectively. Additionally the test work orders for CT's on 31 & 32 and 21 & 22 Main Transformers and the Main Generator will be tracked to completion. These corrective actions are scheduled to be completed by the end of each unit's next refueling outage in Spring 2007 for Unit 3 and Spring 2008 for Unit 2.

- The lessons learned will be incorporated from this failure mode into training. The training will include that this condition is applicable to glass braided cable and that this condition should be looked for when inspecting equipment that is exposed to a constant vibration including 60 Hz vibration. These corrective actions are scheduled to be completed by November 30, 2006.
- Extent of condition reviews will be performed on all planned and/or in progress design modifications for impact. This will include verification that the cabling being installed in a constant vibration environment is suitable for the application (abrasion resistant) and installation guidance is sufficient to prevent cable damage due to abrasion. These corrective actions are scheduled to be completed by November 30, 2006.

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 reactor trip, and the AFWS.

This event meets the reporting criteria because the RPS {JC} was actuated automatically in response to the actuation of the Generator Differential Relay (Direct Generator Trip). The AFWS was actuated and the AFW pumps started as designed as a result of Steam Generator low level from the shrink-swell effect which occurs after automatic RT from full load.

A 4-hour non-emergency notification was made to the NRC for actuation of the reactor protection system {JC} while critical under 10CFR50.72(b)(2)(iv)(B), and an 8-hour notification on July 6, 2006, at 0938, for a valid actuation of the AFWS under 10CFR50.72(b)(3)(iv), Event Log number 42687. The event was recorded in the IPEC corrective action program (CAP) as CR-IP3-2006-02071.

PAST SIMILAR EVENTS

A review of the past two years of Licensee Event Reports (LERs) was performed for events that involved a RT related to design inadequacy in material application. There were no related LERs identified that shared this common cause.

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

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no risk related components out of service at the time of the RT. The AFWS actuation was an expected reaction 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 generator is protected by the generator protection system (GPS) which is designed to protect the generator from internal and external faults by tripping the output breakers. The GPS monitors various parameters and actuates the primary (86P) and backup (86U lockout relays which initiate output breaker trip. During this event the GPS functioned as per design. 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.