Entergy

Entergy Nuclear Northeast
Indian Point Energy Center
295 Broadway, Suite 1
P.O. Box 249
Buchanan, NY 10511-0249
Tel 914 734 5340
Fax 914 734 5718

Fred Dacimo
Vice President, Operations

August 21, 2003

Re: Indian Point Unit No. 3
Docket No. 50-286
NL-03-136

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

SUBJECT: Licensee Event Report No. 2003-003-00
Automatic Turbine / Reactor Trip Due to Fault in
345kV Generator Output Breaker 3

Dear Sir:

Entergy Nuclear Operations, Inc. (Entergy) hereby submits the attached Licensee Event Report (LER), 2003-003-00, in accordance with the requirements of 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 Entergy's corrective action process as Condition Report CR-IP3-2003-03809.

Entergy is making no new commitments in this LER. Should you have any questions regarding this submittal, please contact Mr. John McCann, Manager, Licensing, Indian Point Energy Center at (914) 734-5074.

Sincerely,

Fred R. Dacimo
Vice President, Operations
Indian Point Energy Center

Attachment

cc: see next page

IE22

cc: Mr. Hubert J. Miller
Regional Administrator, Region 1
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406-1415

Mr. Patrick D. Milano, Sr. Project Manager
Project Directorate I
Division of Licensing Project Management
U. S. Nuclear Regulatory Commission
Mail Stop: O-8-C2
Washington, DC 20555-0001

INPO Record Center
700 Galleria Parkway
Atlanta, GA 30339-5957

U.S. Nuclear Regulatory Commission
Resident Inspector's Office
Indian Point 3
P. O. Box 337
Buchanan, NY 10511-0337

NRC FORM 366 (7-2001)

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104 EXPIRES 7-31-2004

LICENSEE EVENT REPORT (LER)

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

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
Indian Point Unit 3	05000- 286	1 OF 5

4. TITLE
Automatic Turbine Trip / Reactor Trip Due to Fault in 345kV Generator Output Breaker 3

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MO	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MO	DAY	YEAR	FACILITY NAME	DOCKET NUMBER 05000-
06	22	2003	2003	03	00	08	21	2003	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)(ii)		50.73(a)(2)(ii)(B)	50.73(a)(2)(ix)(A)
	20.2201(d)		20.2203(a)(4)		50.73(a)(2)(iii)	50.73(a)(2)(x)
	20.2203(a)(1)		50.36(c)(1)(i)(A)	X	50.73(a)(2)(iv)(A)	73.71(a)(4)
	20.2203(a)(2)(i)		50.36(c)(1)(ii)(A)		50.73(a)(2)(v)(A)	73.71(a)(5)
	20.2203(a)(2)(ii)		50.36(c)(2)		50.73(a)(2)(v)(B)	OTHER
	20.2203(a)(2)(iii)		50.46(a)(3)(ii)		50.73(a)(2)(v)(C)	Specify in Abstract below or in NRC Form 366A
	20.2203(a)(2)(iv)		50.73(a)(2)(i)(A)		50.73(a)(2)(v)(D)	
	20.2203(a)(2)(v)		50.73(a)(2)(i)(B)		50.73(a)(2)(vii)	
	20.2203(a)(2)(vi)		50.73(a)(2)(i)(C)		50.73(a)(2)(viii)(A)	
	20.2203(a)(3)(i)		50.73(a)(2)(ii)(A)		50.73(a)(2)(viii)(B)	

12. LICENSEE CONTACT FOR THIS LER

NAME	TELEPHONE NUMBER (Include Area Code)
Richard Louie. Licensing Engineer	(914) 734-5678

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	EL	BKR	I005	N					

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

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

On June 22, 2003, at approximately 1742 hours, following completion of maintenance activities on the 345kV Buchanan distribution system {FK}, Indian Point Unit 3 experienced an automatic turbine / reactor trip event {JE}. Prior to this event, Indian Point Unit 3 was in Mode 1, operating at 100 percent, steady-state power conditions. Generator output breaker No. 3 had been opened earlier at approximately 1654 hours to support maintenance activities on 345kV offsite feeder W-98 {FDR}. Indian Point Unit 3 remained in service utilizing generator output breaker No. 1. While attempting to close breaker No. 3, a fault occurred which resulted in the tripping of both generator output breakers No. 1 and 3. The cause for this event was attributed to a ground fault on Phase B of generator output breaker No. 3 {EL}. Control room personnel immediately entered Emergency Operating Procedure (EOP) E-0, "Reactor Trip or Safety Injection" to initiate plant recovery. No radiological releases or adverse safety implications to the public occurred as a result of this event. In accordance with 10 CFR 50.72(b)(2)(iv)(B), the NRC was notified (Event Number 39955) of this event at 2118 hours on June 22, 2003. This report is submitted pursuant to 10 CFR 50.73(a)(2)(iv)(A) as an event or condition that resulted in a manual or automatic actuation of the reactor protection system.

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

DESCRIPTION OF EVENT

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

On June 22, 2003, at approximately 1742 hours, following completion of maintenance activities on the 345kV Buchanan distribution system {FK}, Indian Point Unit 3 experienced an automatic turbine / reactor trip event {JE}. Prior to this event, Indian Point Unit 3 was operating in Mode 1, at 100 percent, steady-state power conditions. Generator output breaker No. 3 {BKR} had been opened earlier at approximately 1654 hours to support maintenance activities on 345kV offsite feeder W-98 {FDR}. Indian Point Unit 3 remained in service utilizing generator output breaker No. 1. At approximately 1740 hours, upon restoration of feeder W-98, a request was received from the Con Edison District Operator to close generator output breaker No. 3. While attempting to close breaker No. 3, a fault occurred which resulted in the tripping of both generator output breakers No. 1 and 3. A main turbine trip occurred as a result of the actuation of the main generator primary (86P) and backup (86BU) lockout relays {86}. The main generator trip actuation was caused by the actuation of electrical protective relaying direct trip circuitry from the 345kV {FK} main output breakers {BKR}. The cause for this event was attributed to a ground fault on Phase B of generator output breaker No. 3. Control room personnel immediately entered Emergency Operating Procedure (EOP) E-0, "Reactor Trip or Safety Injection" to initiate plant recovery. Subsequently, operators transitioned to ES-0.1, "Reactor Trip Response," and then to POP-3.1, "Plant Shutdown From 45 Percent Power." The plant was stabilized in the hot shutdown condition and the transient terminated. Subsequently, the plant was brought to cold shutdown conditions at approximately 1304 hours on June 23, 2003. Station offsite power was maintained throughout this event. There was no automatic start of the Emergency Diesel Generators {EK}. All control rods {AA} fully inserted. The Auxiliary Feedwater System {BA} pumps automatically started. In accordance with 10 CFR 50.72(b)(2)(iv)(B), the NRC was notified (Event Number 39955) of this event at 2118 hours on June 22, 2003. This event was entered into the Entergy Corrective Action Program under CR-IP3-2003-03809. A post transient evaluation was performed (No. 03-04) on June 23, 2003. This evaluation did not identify any personnel or procedural issues during the event. Upon completion of plant restoration activities, Indian Point Unit 3 was re-started and synchronized to the grid on June 27, 2003 utilizing Breaker No. 1 only.

Generator output breaker No. 3 is a 345kV, Type 345GA25-30, manufactured by ITE Imperial Corporation. There are six sets of contacts (fixed and moving) mounted inside each tank. The contacts are enclosed in a high gas pressure chamber to quench the arc during contact disengagement between stationary and moving contacts.

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

CAUSE OF EVENT

This event was initiated by the failure of Breaker No. 3, Phase B, and was a result of compromised dielectric. It is believed that moisture and/or a contaminated pull rod assembly compromised the SF6 dielectric gas. The compromised dielectric reduced the breaker's capability to withstand a heavy voltage surge during breaker closure. The closure of breaker No. 3 and subsequent re-closure of Millwood breaker No. 18 generated a larger than usual voltage spike. This resulted in an arc to ground along the bottom of a defective pull rod assembly in lower support insulator No. 3 causing the lower support assembly to fail. The apparent cause for this event is human performance-related, and is attributed to limited in-house knowledge of the maintenance and operational requirements of the ITE Imperial Corporation breaker design. As a result, a heavy reliance on the expertise provided by outside contractors to perform activities such as breaker repair or refurbishment existed. A contributing factor was inadequate supervision and oversight of contractor activities, primarily following recent breaker refurbishment activities.

CORRECTIVE ACTIONS

The following corrective actions have been or will be performed to prevent a recurrence of this event.

1. The immediate corrective actions taken prior to plant restart were to assess the damage to breaker No. 3, re-verify the capability of breaker No. 1 to support 100 percent plant power operation, and restore those Balance of Plant components which did not operate as expected following the reactor trip event. Various equipment repairs were performed prior to the plant's return to service on June 27, 2003. (Completed)

2. The dielectric moisture level in generator output breaker No. 1 was tested and found to be within acceptable limits. (Completed)

3. Breaker No. 3 will be replaced with newer designed breaker (GE-Hitachi HVB). (Estimated Completion Date: August 2003)

4. The Training Department will perform a site wide needs analysis to determine the appropriate level of training and qualification needed for personnel to provide supervisory oversight of contractor/vendor personnel performing work on plant equipment. (Estimated Completion Date: December 2003)

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

EVENT REPORTING

This report is submitted pursuant to 10 CFR 50.73(a)(2)(iv)(A) as an event or condition that resulted in a manual or automatic actuation of the reactor protection system. In accordance with 10 CFR 50.72(b)(2)(iv)(B), the NRC was notified (Event Number 39955) of this event at 2118 hours on June 22, 2003.

PAST SIMILAR EVENTS

A review of previous Licensee Event Reports generated within the past three years, which involved a reactor trip caused by a Buchanan Switchyard related fault identified two events.

LER 2002-003: On November 15, 2002, while at 100 percent steady-state reactor power, an automatic reactor trip occurred. Main generator output breaker No. 3 faulted resulting in breakers Nos. 1, 3, and 6 opening and electrically isolating the plant output. The apparent cause for the phase-to-ground fault in breaker No. 3 was high resistance at the breaker contact surfaces. The high resistances at the breaker contacts were a result of breaker contact misalignment during previous maintenance performed during the 2001 refueling outage. Subsequent inadequate corrective actions associated with maintenance performed on breaker 3 in December 2002 contributed to the June 22, 2003 failure. In general, a limited knowledge of these breakers and the difficulty in adequate oversight of contractors due to this limited knowledge were identified as factors.

LER 2000-008: On June 9, 2000, with the reactor at approximately 100 percent power, an automatic reactor trip occurred as a result of a main turbine trip. The cause for the main turbine trip was a direct trip from the 345kV Substation. Degraded electrical cable insulation between several conductors in the direct trip circuitry within underground cable between the plant and the 345kV Substation was the probable cause. The corrective actions for that event did not prevent this event because the cause was different. This event was due to a faulted generator output breaker and was not related to faults in the direct trip cable.

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EVENT SAFETY SIGNIFICANCE

During this event, the turbine received a direct trip signal from the Buchanan Switchyard. The turbine trip caused a reactor trip due to signals derived from the turbine auto stop oil pressure switches since the plant was above P-8 (approximately 35% power). There was no loss of offsite power.
There were no safety significant consequences during this event. A loss of external electrical load and/or turbine trip event is described in the Indian Point Unit 3 FSAR Section 14.1.8. As discussed in the FSAR, the results of the analyses performed for a total loss of external electrical load without a direct or immediate reactor trip from full power conditions show that the plant design is such that there would be no challenge to the integrity of the reactor coolant system or the main steam system. Pressure relieving devices incorporated in the design of the plant would be adequate to limit the maximum pressures to within the design limits. In addition, the integrity of the core would be maintained by operation of the reactor protection system; i.e., the DNBR would be maintained above the safety analysis limit value. Thus, no core safety limit would be violated. Furthermore, these results, in conjunction with the results for the complete loss of flow event from full power, bound the results for a complete loss of load from 50% power without a direct reactor trip on turbine trip. In this event, offsite power was not lost and the reactor did trip immediately. Therefore the Chapter 14 analysis bounds this event.