Entergy

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Fred Dacimo
Vice President, Operations

October 14, 2003

Re: Indian Point Unit No.3
Docket No. 50-286
NL-03-156

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

Subject: Licensee Event Report No. 2003-005-00
Automatic Reactor Trip due to Reactor Coolant Pump Trip on Under-Frequency Caused by a Degraded Off-Site Grid

Dear Sir:

Entergy Nuclear Operations, Inc. (Entergy) hereby submits the attached Licensee Event Report (LER), 2003-005-00, 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 Entergy's Corrective Action Process as Condition Report CR-IP3-2003-04698.

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

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

Mr. Paul Eddy
Public Service Commission
3 Empire State Plaza, 10 Fl
Albany, NY 12223-1350

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 4

4. TITLE

Automatic Reactor Trip due to Reactor Coolant Pump Trip on Under-Frequency Caused by a Degraded Off-Site Grid

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-
08	14	2003	2003	- 05	- 00	10	14	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)
Earl R. Libby, Supervisor-Operations	(914) 736-8514

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

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 August 14, 2003, at approximately 1611 hours, during 100% steady state power, Indian Point Unit 3 experienced an automatic reactor trip initiated as a result of low reactor coolant loop flow due to the trip of the 34 Reactor Coolant Pump (RCP) breaker. The 34 RCP breaker tripped due to electrical supply bus under-frequency caused by an unstable off-site power grid (Northeast blackout). Off-site power was lost and all three Emergency Diesel Generators started and energized their assigned safety buses. Main feedwater isolated and the Auxiliary Feedwater (AFW) pumps automatically started. A Notification of Unusual Event (NUE) was declared at 1623 hours, in accordance with the Emergency Plan when off-site power was unavailable for greater than 15 minutes. The NUE was terminated on August 15, at 0210 hours, when off-site power was restored. The cause of the event was a loss of off-site power due to an unstable power grid. Corrective actions to address the cause of the event included a post trip review, root cause evaluation and plant assessment. There were no nuclear safety concerns exhibited during the event and all fission product barriers remained intact. There was no impact on the health and safety of the general public.

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Indian Point Unit 3	05000-286	2003	- 05 -	00	2 OF 4

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 the brackets {}.

On August 14, 2003, at approximately 1611 hours, during 100% steady state power, Indian Point Unit 3 experienced an automatic reactor trip (RT) {JE} initiated by a loss of off-site power due to a grid disturbance. The loss of off-site power (LOOP) was associated with the blackout that affected parts of northeastern United States and Ontario, Canada. The degraded grid caused an under-frequency breaker trip on the 34 Reactor Coolant Pump (RCP). The trip of 34 RCP breaker initiated a RT on low reactor coolant loop flow (1 of 4 low loop flow with reactor power above the P-8, approximately 35% power, permissive set point). The plant stabilized in natural circulation and the Emergency Diesel Generators (EDGs){EK} 31, 32, and 33 started automatically and energized the 480V buses. Main feedwater system isolated and the Auxiliary Feed Water System (AFW){BA} pumps automatically started. The AFW flow control valves associated with AFW pumps 31 and 33 subsequently lost pneumatic control and manual control was assumed. Other equipment that failed to operate properly included the 34 Main Steam Line (MSL) Safety Valve (lifted prematurely), 32 Source Range Monitor Nuclear Instrument failed, Spent Fuel Cooling was lost for 8 hours, 32 CRDM fan tripped and the Technical Support Center (TSC) Diesel tripped on over speed. The condition for the TSC Diesel was recorded as CR-IP3-2003-04706. The TSC Diesel failure did not prevent activation of the emergency plan when required. None of the equipment issues precluded the return of the Unit to power.

No actuation of the Safety Injection System occurred nor was required as a result of this trip and no Power Operated Relief Valves actuated during this event. The Pressurizer Code Safety Valves remained closed throughout this transient. This event was entered into the Entergy Corrective Action Process under CR-IP3-2003-04698.

CAUSE OF EVENT

The cause of the reactor trip was a loss of off-site power due to grid disturbance. The root cause of the grid disturbance which resulted in a blackout for parts of northeastern United States and Ontario, Canada is under investigation by a joint United States and Canadian government special task force. The grid disturbance caused the main generator to have lower frequency. The 34 RCP breaker tripped on under-frequency, which resulted in a RPS logic trip of the reactor on loss of RCS loop flow.

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

CORRECTIVE ACTIONS

The reactor experienced an automatic trip and the plant shutdown as designed. All emergency systems initiated as required. Corrective actions for the event included a post trip review, a root cause evaluation, and plant walkdown. No specific corrective actions to preclude loss of off-site power due to a similar event were identified.

EVENT REPORTING

This event is reportable under 10 CFR 50.73 (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 in 10 CFR 50.73 (a) (2) (iv) (B). Systems to which the requirements of 10 CFR 50.73 (a) (2) (iv) (A) apply includes the Reactor Protection System including reactor scram or reactor trip, AFW system and the EDGs.

PAST SIMILAR EVENTS

A review of previous occurrences when IP3 had experienced unit trip due to a loss of off-site power was performed. Within the past three years, no such occurrences were identified. However, unit trips as a result of loss of off-site power were reported to the NRC for Indian Point 2 via LER 2003-004-00, 2003-003-00 and 2001-007-00. These trips were for a single unit unlike the present event where both Indian Point 2 and 3 tripped simultaneously.

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

EVENT SAFETY SIGNIFICANCE

There were no significant safety consequences for this event because the plant systems responded as expected except as noted. No pressurizer safety valves lifted and no actuation of the safety injection system was required. There were no nuclear safety concerns exhibited during the event and all fission product barriers remained intact. There was no significant impact on the health and safety of the general public.

The loss of a reactor coolant pump is described in the UFSAR Section 14.1.6, "Loss of Reactor Coolant Flow." This event was initiated when the Unit was operating at 100 % power and is bounded by the UFSAR analysis.

The loss of power to station auxiliaries is described in UFSAR Section 14.1.12, "Loss of Station Auxiliaries." The design event as described in the UFSAR results in a loss of offsite power to both 6.9kV and 480V busses. In this event, the loss of power was per this design event and was bounded by the UFSAR analysis.