NRC Form 366
(9-83)

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/85

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)

INDIAN POINT UNIT 2

DOCKET NUMBER (2): 0 5 0 0 0 2 4 7

PAGE (3): 1 OF 1 2

TITLE (4)

SPURIOUS SAFETY INJECTION SIGNAL

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAMES	DOCKET NUMBER(S)
1 2	2 8	8 4	8 4	— 0 2 6	— 0 0	0 1	2 8	8 5		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 0

POWER LEVEL (10): 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)

20.402(b)	20.405(c)	50.73(a)(2)(iv)	73.71(b)
20.405(a)(1)(i)	50.36(c)(1)	50.73(a)(2)(v)	73.71(c)
20.405(a)(1)(ii)	50.36(c)(2)	50.73(a)(2)(vii)	OTHER (Specify in Abstract below and in Text, NRC Form 366A)
20.405(a)(1)(iii)	50.73(a)(2)(i)	50.73(a)(2)(viii)(A)	
20.405(a)(1)(iv)	50.73(a)(2)(ii)	50.73(a)(2)(viii)(B)	
20.405(a)(1)(v)	X 50.73(a)(2)(iii)	50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER: AREA CODE	
JOHN EILWANGER	9 1 1 4	9 2 6 1 - 1 9 1 1 8 2

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

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)

☐ YES (If yes, complete EXPECTED SUBMISSION DATE) ☐ NO

EXPECTED SUBMISSION DATE (15): MONTH | DAY | YEAR

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

On December 28, 1984, while the reactor was subcritical and an orderly cooldown in progress, a spurious Safety Injection signal occurred. All three Safety Injection pumps were de-energized due to a previous incident. The reactor coolant system was at 345°F and 2000 psia.

At 7:30 p.m. one instrument channel monitoring steam flow from Steam Generator #24 was observed to be drifting excessively in the high (conservative) direction. At 10:20 p.m. a high steam flow signal was generated from #23 Steam Generator. For the existing reactor coolant temperature and pressure, a second high steam flow signal was sufficient to result in a Safety Injection signal.

The behavior of the two channels is not unique although the magnitude of the drift was excessive. A drift of this magnitude is considered to be an isolated case. There was no impact on plant safety as the Safety Injection pumps would have normally been de-energized for this reactor coolant temperature and pressure.

IE22
1/1

On December 28, 1984, while the reactor was subcritical and an orderly cooldown in progress, a spurious Safety Injection signal occurred. All three Safety Injection pumps were de-energized due to a previous incident (LER 84-025). The reactor coolant system was at 345°F and 2000 psia.

At 7:30 p.m. one instrument channel monitoring steam flow from Steam Generator #24 was observed to be drifting excessively in the high (conservative) direction. The operator placed this channel in the tripped condition in accordance with procedures. At 10:20 p.m. a high momentary steam flow signal was generated from #23 Steam Generator line. For the existing reactor coolant system temperature and pressure, the two high steam flow signals were sufficient to result in a Safety Injection signal.

The instrumentation loop associated with the first channel drift was checked and found to be satisfactory. The transmitter sensing tubes were blown down, vented and placed backed in service.

The response of the steam flow channels under these conditions is due to inherent non-linearity of errors which become significant at lower scale ranges. The setpoint, associated with high steam flow, is constant from zero to 20% load. For the steam generator conditions at the time of the second high steam flow signal, a mass flow rate of 14.8% of full steam flow would have been sufficient to produce the same ΔP as for full steam flow. Steam flow signals are susceptible to system errors at the low (less than 25% full scale) end of the scale. These errors are due to instrument drift, air in impulse lines and electronic instability. No further action is planned.

There was no impact upon plant safety as both High Flow channels were activated by steam flows which are below the minimum required. From the viewpoint of plant safety, the behavior of both instrument channels was conservative.

John D. O'Toole
Vice President

Consolidated Edison Company of New York, Inc.
4 Irving Place, New York, NY 10003
Telephone (212) 460-2533

January 28, 1985

Re: Indian Point Unit No. 2
Docket No. 50-247
LER-84-026-00

Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Sirs:

The attached Licensee Event Report LER-84-026-00 is hereby submitted in accordance with the requirements of 10 CFR Part 50.73.

Very truly yours,

John D. O'Toole

attach.
cc: Dr. Thomas E. Murley,
Regional Administrator-Region I
U. S. Nuclear Regulatory Commission
631 Park Avenue
King of Prussia, Pa. 19406

Senior Resident Inspector
U. S. Nuclear Regulatory Commission
P. O. Box 38
Buchanan, New York 10511

IE22
1/1