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)	DOCKET NUMBER (2)	PAGE (3)
INDIAN POINT, UNIT 2	0 5 0 0 0 2 4 7	1 OF 4

TITLE (4): ACTUATION OF RPS

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)
0 3	0 6	8 5	8 5	0 0 4	0 0	0 4	0 4	8 5		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9):

POWER LEVEL (10): 1 0 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)	X 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)	50.73(a)(2)(iii)	50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER	
	AREA CODE	
JOHN R. ELLWANGER	9 1 4	5 2 6 - 5 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
X	S J	R L Y	P 2 9 7	Y					

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

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

On March 6, 1985 while the reactor was at 100% power and during a periodic surveillance test to determine the operability of the steam line pressure bistables, a reactor trip occurred. The Reactor Protection System (RPS) was actuated by a feedwater flow/steam flow mismatch signal in coincidence with a low steam generator level signal. The trip occurred during the process of switching steam generator level control channels, and is attributed to faulty relays which caused a feedwater regulating valve to function improperly.

There were no safety consequences, as the Reactor Protection System functioned in accordance with its design and tripped the reactor when the mismatch occurred.

IE22
1/1

Plant and System Identification

Westinghouse 4-Loop Pressurized Water Reactor - 900 MWe

Identification of Occurrence:

Actuation of Reactor Protection System (Reactor Trip) due to feedwater flow/steam flow mismatch signal coincident with low steam generator level signal.

Event Date: 3/6/85

Reportability Determination: 3/6/85

Rep. t Due Date: 4/5/85

This report was initiated by Significant Occurrence Report 85-98.

Description of Occurrence

On March 6, 1985 while at full power operation, a surveilance test PTM-11 was being performed. The objective of the test was to demonstrate the operability of the steam line pressure bistables in accordance with Technical Specification Tables 3-1 and 4.1-1. One step in the test procedure requires transferring from Channel "A" to Channel "B" as the controlling channel. In the test procedure, Channel "A" had been the control channel in an earlier test step. The Operator exercised the option of placing the steam generator level control in manual to effect the change and then return to automatic once stability had been achieved. The control channel transfer occurred without incident for Steam Generator 21 and 22. However, when control was returned to the automatic mode from the manual mode on Steam Generator 23, steam generator level and feedwater flow rapidly decreased. The Operator reverted to manual control increasing the demand for feedwater, but was not able to terminate the transient. At 8:57 a.m. on March 6, 1985, a reactor t ip occurred due to a feedwater/steam flow mismatch signal coincident wth a low steam generator level signal from Steam Generator 23. Steam Flow exceeded feedwater flow by greater than one million pounds per hour and steam generator level decreased below the 30% level.

Apparent Cause of Occurrence

At the feedwater regulating valve, a transducer converts the electrical signal to a pneumatic signal which is applied to the regulating valve

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Apparent Cause of Occurrence

At the feedwater regulating valve, a transducer converts the electrical signal to a pneumatic signal which is applied to the regulating valve

positioner. The valve positioner supplies a pneumatic signal which is capable of automatic closure of the regulating valve in the event of a high steam generator level, a Safety Injection signal, or on a Reactor Trip coincident with low primary system average temperatures. These sources were eliminated as potential contributors to the event as they would have affected all feedwater regulating valves rather than one. The regulating valve itself was checked and found to be fully operable.

When the steam generator level control is in the "manual" mode, the auto/manual station provides input to the I/P transducer which controls the regulating valve position. The master controller follows the "manual" signal. The steam generator level control is also in a tracking mode and follows the master controller. In the "auto" position, both the master controller and the level controller follow their respective input fluctuations. When the control is switched from "manual" to "automatic", the regulating valve should smoothly adjust to the position where actual steam generator level equals setpoint and feed flow equals steam flow. In the event this did not occur, instead, it appears that the regulating valve went to the closed position. However, there was no evidence that any valve demand signal called for valve closure or that the valve moved in the direction of closure despite a demand signal to open.

If the "auto" signal became erratic when the control was transferred from "auto" to "manual", the expected system response would have been reduced feedwater flow, stable steam flow, and reduced steam generator level, as observed. After the trip, the manual/auto electronic circuitry was checked. A test setup was devised which provided necessary input signals to simulate test conduct and level mode switching. The erratic behavior ws reproduced during a simulation and was traced to a 4 pole - 2 throw control relay within the level vs. setpoint controller.

Analysis of Occurrence

From a safety viewpoint all safety related systems functioned in a normal manner in accordance with their design. The steam generator level control and the feedwater regulating control function are not safety related. Their failure did represent an exercise of the Reactor Protection System. There were no consequences to the Public Health and Safety since all systems and personnel functioned as required.

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INDIAN POINT, UNIT 2	0 5 0 0 0 2 4 7	8 5	0 0 4	0 0	0 4 OF 0 4

TEXT (If more space is required, use additional NRC Form 366A's) (17)

Corrective Action

The relays (K-1 and K-2; Potter and Brumfield Model KHP17D13) have been replaced in the instrumentation system for all feedwater regulating valves. Examination of the replaced relays did not reveal significant deterioration. Only slight oxidation of the contact points was evident. Due to acceptable test results after relay replacement and the previous satisfactory experience with these relays, no further action is planned.

John D. O'Toole
Vice President

Consolidated Edison Company of New York, Inc.
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Telephone (212) 460-2533

April 4, 1985

Re: Indian Point Unit No. 2
Docket No. 50-247
LER-85-004-00

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

Dear Sirs:

The attached Licensee Event Report LER-85-004-00 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

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