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

TITLE (4)

ISOLATION VALVE SEAL WATER SYSTEM LEAKAGE

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 6	1 1	8 4	8 4	— 0 0 6	— 0 0	0 7	1 1	8 4		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9)		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)
POWER LEVEL (10)	0 0 0	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)	X 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
MICHAEL BLATT, DIRECTOR - REGULATORY AFFAIRS	AREA CODE 9 1 4 5 2 6 - 5 1 2 7

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	
E	H J	I S V	A 3 9 5	Y							

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

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

On June 11, 1984 while the plant was shutdown for refueling, routine surveillance tests were being conducted. During surveillance of the Isolation Valve Seal Water System, it was reported that the overall system leakage was in excess of acceptance criteria. This was attributed to excessive leakage through the isolation valves for the auxiliary steam supply line and condensate return line.

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On June 11, 1984 during a refueling outage it was reported that the overall leakage detected in a test of the Containment Isolation Valve Seal Water System was 15,543 cc/hr. The maximum permitted for this test is 14,700 cc/hr. (Technical Specification 4.4.D.2.C).

The Isolation Valve Seal Water System assures the effectiveness of those containment isolation valves that are located in lines connected to the Reactor Coolant System, or that could be exposed to the containment atmosphere during any condition which requires containment isolation, by providing a water seal at the valves. The system provides a simple and reliable means for injecting seal water between the seats and stem packing of the globe and double disc types of isolation valves, and into the piping between closed diaphragm type isolation valves. This system operates to limit the fission product release from the containment. Although no credit is taken for the operation of this system in the calculation of off-site accident doses, it does provide assurance that the containment leak rate is lower than that assumed in the accident analysis should an accident occur.

The excessive leakage detected during the test was attributed primarily to the isolation valves in the auxiliary steam supply line and condensate return line. These two valves together exhibited a leakage of 12,480 cc/hr. These valves will be disassembled and examined, parts will be repaired or replaced as appropriate, and the system will be retested before startup of the plant.

At the time of the test, the plant was in the cold shutdown condition and therefore the operability of the Isolation Valves Seal Water System was not required. There would have been no undue risk to the public even if the plant had been in operation since, as discussed above, the FSAR analysis of accidents having radiological consequences takes no credit for the operation of this system.

No previous similar events.

John D. O'Toole
Vice President

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

July 11, 1984

Re: Indian Point Unit No. 2
Docket No. 50-247
LER-84-006-00

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

Dear Sirs:

The attached Licensee Event Report LER-84-006-00 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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