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 No. 2

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

PAGE (3): 1 OF 0 3

TITLE (4): Technical Specification Limit Exceeded; IVSWS

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	2 3	8 8	8 8	0 0 3	0 0	0 4	2 3	8 8		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): N

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)	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)	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: John Ellwanger, Principal Engineer

TELEPHONE NUMBER: AREA CODE 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
B	C B	V	A 3 9 6	N	B	B 0	O	A 3 9 5	N
B	C B	V	A 3 9 6	N					

SUPPLEMENTAL REPORT EXPECTED (14)

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

EXPECTED SUBMISSION DATE (15): MONTH DAY YEAR

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

Abstract

On March 23, 1987, during a review of completed test results, it was determined that a Technical Specification limit had been exceeded. During the 1987 refueling outage, a surveillance test of type "B" and "C" (Appendix J) valves was conducted from October through December of 1987. When test results were totaled, it was discovered that the aggregate of individual valve leakage exceeded that permitted by the Technical Specification pertaining to the limit for the Isolation Valve Seal Water System. Excessive leakage occurred across two Letdown Containment isolation valves (201 and 202) as well as a Containment isolation valve in the Safety Injection System (851A). The valves have now been repaired and the leakage is currently well within acceptable limits.

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Plant and System Identification:

Westinghouse 4-loop pressurized water reactor.

Identification of Occurrence:

Total leakage of Isolation Valve Seal Water System (IVSWS) exceeds Technical Specification Limit.

Event Date: March 23, 1988

Past Similar Occurrence: LER 84-006

Significant Occurrence Report: SOR 88-137

Description of Occurrence:

During the 1987 Refueling outage surveillance testing of Containment isolation valves was performed throughout the period October through December of 1987. In March of 1988, as part of the effort in preparing the overall Containment Leak Rate Test report, it was discovered that leakage from the Isolation Valve Seal Water System (IVSWS) exceeded the maximum permitted by the Technical Specification. The total measured leakage was 68,321 c.c per hour, whereas the Technical Specification limit is 14,700 c.c per hour. The valves contributing to the leakage were subjected to corrective maintenance and "as left" leakage rates were substantially below the maximum permitted.

Analysis of Occurrence:

This event is reportable since a Technical Specification Limit had been exceeded. Since the IVSWS is not relied upon in accident mitigation scenarios, at no time was there any impact upon plant safety.

Cause of the Occurrence:

The cause of the leakage was traced to excessive leakage across three Containment isolation valves. These are valves 201 and 202 which are part of the Letdown system and 851A which is in the Safety Injection System.

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

Corrective Action:

The leakage across these valves was evaluated by Engineering and the "as found" data, although excessive from the viewpoint of IVSWS, was not found to be unacceptable in terms of Containment leakage. The valves were repaired, tested and returned to service.

Stephen B. Bram
Vice President

Consolidated Edison Company of New York, Inc.
Indian Point Station
Broadway & Bleakley Avenue
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Telephone (914) 737-8116

April 23, 1988

Re: Indian Point Unit No. 2
Docket No. 50-247
LER 88-003

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

The attached Licensee Event Report LER 88-003 is hereby submitted in accordance with the requirements of 10 CFR 50.73.

Very truly yours,

attachment

cc: Mr. William Russell
Regional Administrator - Region I
U.S. Nuclear Regulatory Commission
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