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September 30, 1998

Re: Indian Point Unit No. 2
Docket No. 50-247
LER 97-012-01

Document Control Desk
US Nuclear Regulatory Commission
Mail Station PI-137
Washington, DC 20555

The attached Licensee Event Report 97-12-01 is hereby submitted in accordance with the requirements of 10 CFR 50.73.

Very truly yours,

Attachment

cc: Mr. Hubert J. Miller
Regional Administrator - Region I
US Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Mr. Jefferey Harold, Project Manager
Project Directorate I-1
Division of Reactor Projects I/II
US Nuclear Regulatory Commission
Mail Stop 14B-2
Washington, DC 20555

Senior Resident Inspector
US Nuclear Regulatory Commission
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NRC FORM 366 (6-1998) | U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Indian Point Unit No. 2	50-247	1 OF 3

TITLE (4)
IVSWS Leakage Greater Than Technical Specification Limit

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 NAME	DOCKET NUMBER
										05000
05	09	97	97	-- 12 --	01	09	05	98	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9)	N
POWER LEVEL (10)	000

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

	20.2201(b)		20.2203(a)(2)(v)		50.73(a)(2)(i)		50.73(a)(2)(viii)
	20.2203(a)(1)		20.2203(a)(3)(i)	X	50.73(a)(2)(ii)		50.73(a)(2)(x)
	20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
	20.2203(a)(2)(ii)		20.2203(a)(4)		50.73(a)(2)(iv)		OTHER
	20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)		Specify in Abstract below or in NRC Form 366A
	20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Philip Griffith, Sr. Licensing Engineer	(914)734-5190

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

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX
X	CC	ISV	A391	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 15 single-spaced typewritten lines) **(16)**

Surveillance tests of containment isolation valves were performed during the 1997 refueling outage, with containment integrity not required. The as-found leakage rate for valve FCV-625 sealed with water from the Isolation Valve Seal Water System exceeded the 14,700 cubic centimeters per hour leakage rate Technical Specification limit for containment isolation valves sealed by this system.
This valve was repaired and tested prior to plant startup to ensure operability.

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

PLANT AND SYSTEM IDENTIFICATION:

Westinghouse 4-Loop Pressurized Water Reactor

IDENTIFICATION OF OCCURRENCES:

Total leakage rate of Isolation Valve Seal Water System (IVSWS) exceeded the Technical Specification 4.4.D.2.c limit of 14,700 cubic centimeters per hour due to assumed excessive leakage through FCV-625 containment isolation valve sealed with water.

EVENT DATE:

May 9, 1997

REPORT DUE DATE:

September 30, 1998

REFERENCES:

Indian Point 2 Condition Identification and Tracking System 97-E01476

PAST SIMILAR OCCURRENCES:

LER 84-06, LER 88-03, LER 89-08, LER 93-03, LER 95-06

DESCRIPTION OF OCCURRENCES:

During the 1997 refueling outage, with containment integrity not required, it was assumed that the as-found leakage through the containment isolation valve (CIV) FCV-625, which is supplied seal water from the IVSWS, was greater than the Technical Specification 4.4.D.2.c limit of 14,700 cubic centimeters per hour. The leakage rate through CIV FCV-625 is tested in refueling surveillance test PT-R26A, "Local IVSWS Test Type 'C,'" and since the leakage rate exceeded the ability of the test rig to maintain flow and pressure through FCV-625 the Technical Specification limit was considered to have been exceeded.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

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

ANALYSIS OF OCCURRENCES:

The IVSWS 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 that 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. The resulting water seal blocks any potential leakage of the containment atmosphere through the valve seats and stem packing. The water is introduced at a pressure slightly higher than the containment design pressure of 47 psig. The possibility of leakage from the containment or reactor coolant system past the first isolation point is thus prevented by assuring that if leakage occurs, it will be from the IVSWS into containment. Containment isolation valve FCV-625 is sealed with IVSWS water.

CAUSE OF OCCURRENCES:

Technical Specification 4.4.D.2.c leakage rate limit was considered to be exceeded due to excessive leakage through the IVSWS sealed containment isolation valve FCV-625, component cooling water return from RCP thermal barrier. The leakage through FCV-625 was determined to be past the valve seating surface.

CORRECTIVE ACTIONS:

The repair of valve FCV-625 seat and disk was completed under Work Order NP-9791152 to correct the leakage. The valve was retested to ensure operability and the work order closed out on June 3, 1997.

An evaluation of the compatibility of the valve for suitability of design and materials was completed on May 19, 1998 with a determination that the materials and design were suitable for the valve application.

An evaluation was performed that verified the suitability of design and materials for valves in similar applications to FCV-625; this evaluation was completed on June 8, 1998.