Stephen B. Bram
Vice President

Consolidated Edison Company of New York, Inc.
Indian Point Station
Broadway & Bleakley Avenue
Buchanan, NY 10511
Telephone (914) 737-8116

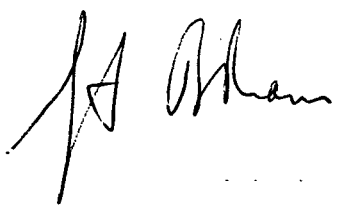
September 30, 1992

Re: Indian Point Unit No. 2
Docket No. 50-247
LER 92-17-00

Document Control Desk
US Nuclear Regulatory Commission
Mail Station P1-137
Washington, DC 20555

The attached Licensee Event Report LER 92-17-00 is hereby submitted in accordance with the requirements of 10 CFR 50.73.

Very truly yours,



Attachment

cc: Mr. Thomas T. Martin
Regional Administrator - Region I
US Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Mr. Francis J. Williams, Jr., 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
PO Box 38
Buchanan, NY 10511

130067

IE22
1/1

NRC FORM 366 (6-89)

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104
EXPIRES: 4/30/92

ESTIMATED BURDEN PER RESPONSE TO COMPLY WTH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE RECORDS AND REPORTS MANAGEMENT BRANCH (P-530), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Indian Point Unit No. 2	0 5 0 0 0 2 4 7	1 OF 0 4

TITLE (4)

Re-Evaluation of the Auxiliary Feedwater System Seismic Design Basis

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 5 0 0 0
0 8	3 1	9 2	9 2	— 0 1 7	— 0 0	0 9	3 0	9 2		0 5 0 0 0

OPERATING MODE (9)	N	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)			
POWER LEVEL (10)	1 0 0	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)	X 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
Pedro J. Franceschi	9 1 4 5 2 6 - 5 6 7 0

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

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS		CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS	

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)

Based on the preliminary results of an engineering re-evaluation of the Auxiliary Feedwater System (AFW) Seismic Design Bases, it was determined on August 31, 1992, that the trip setpoint on the Low Suction Pressure Protection Switch was nonconservatively high and would have resulted in tripping both motor-driven pumps under design bases assumptions. A re-evaluation of an engineering analysis of a postulated seismically induced break in the non-seismic portion of a condensate line connected to the suction side of the AFW pumps indicated that for momentary pressure drops induced during start-up of the motor driven AFW pumps, the suction pressure for these pumps is below the pump trip setpoints. This would cause the tripping of both motor driven auxiliary feedwater pumps during the event and prevent restart until the break was isolated. The low suction pressure analyzed for this event is well above the Net Positive Suction Head required for the back-up steam-driven AFW pump, which does not have a suction low pressure trip.

The immediate compensatory action was to maintain an elevated level in the Condensate Storage Tank until the setpoints of the pressure switches in the pump suction lines were reduced to a level that gave adequate margin to pump trip (i.e., a setpoint of 4 psig). This setpoint will continue to provide pump protection.

PLANT AND SYSTEM IDENTIFICATION:

Westinghouse 4-Loop Pressurized Water Reactor

IDENTIFICATION OF OCCURRENCE:

Re-evaluation of the Auxiliary Feedwater System Seismic Design Bases

EVENT DATE:

August 31, 1992

REPORT DUE DATE:

September 30, 1992

REFERENCES:

Significant Occurrence Report (SOR) 92-425

PAST SIMILAR OCCURRENCE:

None

DESCRIPTION OF OCCURRENCE:

On August 31, 1992, the preliminary results of a re-evaluation of the Auxiliary Feedwater (AFW) system seismic break analysis indicated that, with the existing low suction pressure switch setpoint, both motor-driven AFW pumps would trip on a seismically induced break of the non-seismic line interconnected with the suction line and a concurrent demand-to-start signal. The re-evaluation was the result of changes made to AFW system model used in the previous seismic break analysis and a need to ensure that the present low suction pressure switch setpoint provided adequate margin for the seismic design bases of the AFW system.

ANALYSIS OF OCCURRENCE:

As a result of a Technical Evaluation Report prepared in connection with Indian Point Unit 3, Indian Point 2 was requested to address an issue in the seismic design of the AFW System. On December 1, 1986, the NRC issued a SER for the Indian Point Unit 2 AFW system based on a pipe failure of the non-seismic condensate line to the hotwells, which is connected to the AFWS pumps suction line. This seismic induced failure was evaluated with the concurrent random single failure of the seismic isolation valve, LCV-1158,

ANALYSIS OF OCCURRENCE: (continued)

to automatically close. A thirty (30) minute penalty was assumed for operator action to manually close the valve. The result was that the Condensate Storage Tank (CST) would have sufficient water inventory to reach Residual Heat Removal (RHR) cut-in temperature. At that time, the motor-driven pump low suction protection was provided by a flow measuring device with a time delay which was insensitive to the actual pressure in the line.

In 1987, the flow measuring device was changed to a pressure measuring device because of problems relating to maintaining the Environmental Qualification of the device while providing normal calibration and maintenance activities. The setpoint of the new pressure device was determined by the use of a steady state static analysis. This was considered acceptable because the margin between the analyzed suction pipe pressure and the setpoint pressure was so large that it could easily accept the transitory dynamic effect of the pumps accelerating from zero flow to full flow and the resultant steady state suction pressure from both pumps delivering their full flow.

In April 9, 1990, the break analysis was redone for the Stretch Power Rating of 3071.4 MWt and the higher service water design temperature of 95 degrees F. The system was analyzed using a more sophisticated model which accounted for the higher decay heat load, the lower RHR Heat Exchanger heat removal capability, and the higher system flow delivery effect on pump low suction pressure switch setpoint. The result was that the plant continued to adequately meet the December 1, 1986 NRC SER.

As a result of a plant transient trip event on April 13, 1992 (LER #92-007), a new hotwell condensate line interaction scenario (non-seismic) was found to exist which with an independent concurrent demand-to-start of the AFW pumps, resulted in actuation of both pump low suction trips. The system model was further refined to account for this interaction which involved both changes in the static and dynamic responses. Plant process computer data from the April 13, 1992 event indicated the need for further system remodeling due to changes in AFW system piping physical characteristics. After the model was finalized for the hotwell interaction problem, the seismic break analysis was re-evaluated to determine if the previous AFW pump suction pressure switch setpoint continued to be valid. On June 15, 1992, the analysis was reverified with a finding found that the low suction pressure switch setpoints continued to be acceptable. However, due to the reduced margin of the results, further transient/acceleration analyses of the pumps start-up conditions were determined to be warranted.

ANALYSIS OF OCCURRENCE: (continued)

On August 31, 1992, the preliminary results indicated that, with the new model, static and dynamic momentary pressure drops induced by pump acceleration would result in both suction pressure switches tripping their respective pumps for the seismic design basis event of the AFW system.

This condition was determined to be outside the plant design basis. Therefore this LER is being submitted under the provision of 10 CFR 50.73 (A)(2)(ii)(B).

CAUSE OF OCCURRENCE:

During a plant transient/trip event on April 13, 1992, it was discovered that both motor-driven AFW pumps tripped on low suction pressure when a demand to-start occurred concurrent with the interconnected hotwell vacuum drag make-up line fully open. Investigation of the event led to refining the AFW system modeling to account for this condition. After finalizing the new model, a re-evaluation of the most recent design basis seismic break engineering analysis for the AFW system was performed to ascertain the validity of the existing low suction pressure setpoint.

On August 31, 1992, preliminary results indicated that the existing setpoint would result in the low suction pressure switches tripping both AFW pumps for a design basis earthquake that induced a pipe failure in the non-seismic hot well vacuum drag make-up line concurrent with an AFW system demand-to-start condition.

CORRECTIVE ACTION:

The corrective action was to maintain an elevated level in the Condensate Storage Tank until the setpoint of the low suction pressure protection switches was reduced to a level that gave adequate margin to pump trip. The trip setpoint was reduced and tested for both pumps in less than 90 minutes after notification to the Control Control Room.