Stephen E. Quinn
Vice President

Consolidated Edison Company of New York, Inc.
Indian Point Station
Broadway & Bleakley Avenue
Buchanan, NY 10511
Telephone (914) 734-5340

June 2,1997

Re: Indian Point Unit No. 2
Docket No. 50-247
LER 97-10-00

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

The attached Licensee Event Report 97-10-00 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

090057

Senior Resident Inspector
US Nuclear Regulatory Commission
PO Box 38
Buchanan, NY 10511

IE22 /1

NRC FORM 366 (6-89)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

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

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH 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.

FACILITY NAME (1): Indian Point Unit No. 2

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

PAGE (3): 1 OF 0 6

TITLE (4): Safety Injection Actuation While Testing Main SteamSafety Valves

EVENT DATE (5) MONTH	DAY	YEAR	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	REPORT DATE (7) MONTH	DAY	YEAR	OTHER FACILITIES INVOLVED (8) FACILITY NAMES	DOCKET NUMBER(S)
0 5	0 1	9 7	9 7	0 1 0	0 0	0 6	0 2	9 7		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): N

POWER LEVEL (10): 0 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)	[XX] 50.73(a)(2)(iv)	[ ] 73.71(b)
[ ] 20.405(a)(1)(i)	[ ] 50.36(c)(1)	[XX] 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	
Richard Louie, Engineer	9 1 4	7 3 4 - 5 6 7 8

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
B	S B	R V	C 6 3 5	Yes					
B	E C	B K R	W 1 2 0	Yes					

SUPPLEMENTAL REPORT EXPECTED (14)

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

EXPECTED SUBMISSION DATE (15): MONTH 0 9, DAY 0 1, YEAR 9 7

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

On May 1, 1997 at 00:01 the main generator at Indian Point Unit No. 2 was taken off line for the 1997 refueling outage. With the plant at hot shutdown and during the performance of test PT-R6, "Main Steam Safety Valve Setpoint Determination," one main steam safety valve, MS-46C, on the main steam line for steam generator 23, stuck open. This caused the initiation of a Safety Injection (SI) signal at 03:08:19 due to the differential pressure between one steam generator and two others. The SI signal generated a "stripping" signal to the 480 VAC buses and a start signal to the emergency diesel generators (EDGs). Loads "stripped" from the 480 VAC buses and the EDGs started as expected. The SI logic circuit then commenced to sequentially reload the buses. During the EDG load sequence, a motor-driven auxiliary feedwater pump (23AFWP) and containment fan cooler unit (25FCU) failed to start as required. In accordance with the emergency operating procedures (EOPs), operators manually started 23AFWP and 25FCU. At 03:21:51 the SI was reset in accordance with procedure. NRC approval for relief from the ASME Section XI testing requirements pertaining to the main steam safety valves was requested. Subsequent to receiving NRC verbal approval, the reactor was safely brought to cold shutdown conditions. During this event, the Isolation Valve Seal Water System (IVSWS) did not perform as described within the UFSAR. The health and safety of the public were not affected by this event.

PLANT AND SYSTEM IDENTIFICATION:

Westinghouse 4-Loop Pressurized Water Reactor

IDENTIFICATION OF OCCURRENCE:

Safety Injection Actuation While Testing Main Steam Safety Valves

EVENT DATE:

May 1, 1997

REPORT DUE DATE:

June 2, 1997

REFERENCES:

Condition Identification and Tracking System (CITRS) No. 97-E01305

PAST SIMILAR OCCURRENCE:

None

DESCRIPTION OF OCCURRENCE:

On May 1, 1997 at 00:01 the main generator at Indian Point Unit No. 2 was taken off line for the 1997 refueling outage. Control rods were driven into the core and the reactor trip breakers opened at 00:23. Plant shutdown was proceeding normally in accordance with established plant procedures. At 02:45 commencement of test PT-R6, "Main Steam Safety Valve Setpoint Determination," was authorized. This test determines the actual setpoint of the twenty main steam safety valves (MSSVs). To determine valve set pressure, an air pressure assist device is used. In order to perform this test, the hand lifting assembly is removed from the top of the MSSV. An air-operated lifting motor (diaphragm sealed air cylinder having a known effective area) is mounted on the MSSV bonnet. With steam pressure below the MSSV set pressure, air is supplied to the air motor until the valve starts to open. This results in a force added to the steam pressure, which is used to determine the valve set pressure. Normally during this test the MSSV is "popped" open for approximately one second and the resultant pressure drop this causes in the steam generators is on the order of a few psi.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Indian Point Unit No. 2	0 5 0 0 0 2 4 7	9 7	— 0 1 0	— 0 0	0 3 OF 0 6

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

During the performance of PT-R6, one MSSV (MS-46C) on the main steam line for steam generator 23, did not immediately reclose after opening. Four MSSVs had successfully been tested prior to MS-46C. The test technician expected to see an air motor pressure of approximately 50 psig at which the MSSV should "pop" open. This did not occur. The technician then continued to increase the pressure to 65 psig with no success. He subsequently reduced the air pressure to inspect the hose connections. Confirming that all connections were correct, the technician increased air motor pressure to 96 psig, equivalent to a setpoint of 1200 psig. MS-46C opened and remained in that position. The steam discharge from MS-46C resulted in a plant cooldown. At 03:03:52, the temperature of reactor coolant system (RCS) loop 23 decreased below 541°F followed seconds later by similar decreases in the other three loops. RCS loop 23 is associated with steam generator 23 and would cool first due to the stuck open MS-46C on that steam generator. At 03:08:19, a Safety Injection (SI) signal occurred due to steam generator differential pressure, since the pressure in steam generator 23 was 155 psig lower than two of the other three steam generators. This SI signal generated a "stripping" signal to the 480 VAC buses and starts all three emergency diesel generators (EDGs). As expected, electrical loads "stripped" from the 480 VAC buses. The SI logic then commenced to sequentially reload the buses. In the event that a SI occurs concurrently with a loss of off-site power, the running EDGs are then loaded onto their deenergized buses that in turn power the loads necessary for the SI. Since off-site power was available during this event, the EDGs did not load onto their buses. At 03:08:42, MS-46C closed. MS-46C had been open for 334 seconds. After MS-46C closed, the cooldown of the RCS stopped.

During the SI loading sequence, a motor-driven auxiliary feedwater pump (23 AFWP) and containment fan cooler unit (25FCU) failed to start. The control room (CCR) breaker indication for 23AFWP showed an amber breaker "mismatch" light on the control panel indicating either a trip of the amptector circuit or that the breaker is in a mechanically tripped-free condition. This light indicates whenever a breaker trips from any means other than the shunt trip attachment. In accordance with EOPs, operators manually started 23AFWP and 25FCU.

At 03:09:46 operators manually reestablished charging flow to the RCS. This ensured charging flow to the seal packages of the reactor coolant pumps (RCPs) and added inventory to the RCS. At the time, letdown had not yet been reestablished. Without letdown to remove excess RCS inventory, pressurizer level increased due to plant heatup and charging flow. This level increase caused an increase of the pressurizer pressure. At 03:11:08 pressurizer pressure returned to 2235 psig and remained constant for approximately four minutes. The opening of the pressurizer spray valves permitted the pressure to be maintained at its normal control point. As designed, the SI logic isolated the normal instrument air supply to containment. When this occurred the air-operated pressurizer spray valves failed close. Although instrument air to containment was isolated, the air ring header inside the containment maintained a volume of compressed air for a short time after isolation. There was sufficient air pressure available for the pressurizer spray valves to open and pressurizer pressure was controlled following the isolation of air to containment. At 03:14:32 the pressurizer spray valves closed as the limited volume of air was expended and pressurizer pressure began to increase sharply. The operators recognized the pressure increase and opened the PORV block valves. Pressurizer pressure was controlled by the power operated relief valves (PORVs), which have their own nitrogen accumulators. At 03:17:25 the pressurizer PORV lifted at 2335 psig and relieved to the pressurizer relief tank (PRT). The open PORV reduced pressurizer pressure to 2310 psig and then closed. Pressure then began to increase again until another PORV actuation occurred. A total of five PORV actuations occurred at approximately ninety second intervals.

At 03:21:51 the SI was manually reset by procedure. The containment isolation signal caused by the SI was reset at 03:25:27 and instrument air was reestablished to the containment. Pressurizer spray valves immediately went open and restored pressurizer pressure to its normal 2235 psig.

Subsequent to this event, the isolation valve seal water system (IVSWS) tank level and pressure were observed by a plant operator to be 75% and 38 psig, respectively. The operator restored the level to 94% and the pressure to 56 psig by adding primary water to the tank.

Subsequent to this event, emergency relief from the "as found" condition testing requirements of ASME Section XI pertaining to the main steam safety valves was requested from NRC. Subsequent to receiving NRC verbal approval, the plant was safely brought to cold shutdown conditions.

ANALYSIS OF OCCURRENCE :

This report is provided pursuant to the requirements of 10 CFR 50.73(a)(2)(iv) because the automatic actuation of a engineered safety features (ESF) occurred during the plant shutdown which began on May 1, 1997. During the shutdown, the main steam code safety valves were being tested to obtain "as found" set pressure data. PT-R6, "Main Steam Safety Valve Setpoint Determination," is the test performed to determine the actual setpoints of the main steam safety valves. While performing PT-R6, one main steam safety valve, MS-46C did not re-close after opening. This caused a cooldown of the RCS resulting in the automatic initiation of a SI signal due to steam generator differential pressure. To prevent an additional undesirable transient and subsequent challenge to safety systems if another safety valve failed to re-close, emergency relief from the Code requirement to obtain "as found" data was requested. NRC verbal approval was received. Further testing of the main steam safety valves was discontinued and the plant was brought to cold shutdown conditions in accordance with procedures.

Following the disassembly and inspection of MS-46C, several discrepancies were noted regarding the condition of the valve internal components. Corrosion was discovered on the outside diameter of the disk holder and inside surface of the guide assembly. Dimensional clearances between the spindle rod and several internal components were determined to be reduced and out of tolerance. This condition imposed additional frictional forces on the valve which resulted in a higher opening set pressure. Similar conditions were also observed on valve MS-45A which was not tested during the shutdown.

During the SI signal sequence, both 23AFWP and 25FCU failed to start as required. Post-event investigations into the cause for misoperation of 23AFWP and 25FCU centered upon the 480 VAC DB-50 breakers. The cause of the failure of 25FCU to start is attributed to excessive friction between the roller and a latch face within the breaker mechanism. This friction results in binding within the operating mechanism preventing the closing coil core from dropping down as required. The breaker will not close mechanically or electrically when the operating mechanism is in this position. This is an apparent intermittent condition as the breaker could be operated many times without this observed binding condition. The existing breaker preventive maintenance procedure and vendor advisories may not adequately inspect for this condition.

Based upon our investigation, no conclusive root cause has been identified for the failure of 23AFWP to start. Prior to the SI signal actuation, 23AFWP had been in service. Following the SI signal actuation, the breaker tripped as indicated by the amber breaker "mismatch" light on the control panel. On May 1, 1997 at 03:10 the control room operator started 23AFWP and reset the breaker controls manually by positioning the control switch from "Auto" to "Start." Subsequently, the breaker and amptector circuitry was satisfactorily inspected and tested.

During this event the IVSWS did not perform as described within the UFSAR. The IVSWS seal water tank is designed to provide at least a 24 hour supply of seal water under the most adverse circumstances assuming an isolation valve design leakage rate of 50 cm^3/hr-in., plus the failure of the largest containment isolation valve to seat and leaking at the maximum rate of 1000 cm^3/hr-in. Subsequent to the event, a leak was identified on the IVSWS line to the containment penetration for the city water system at valve MW-17. This containment penetration is normally closed and does not receive IVSWS seal water until IVSWS is actuated. The seal water tank level decrease observed following the SI signal actuation has been attributed to the leak identified at this penetration. The IVSWS seal water tank is pressurized by nitrogen gas which is regulated by two pressure regulators. Only one nitrogen pressure regulator is in service while the other is isolated. The inability to maintain adequately the required pressure within the IVSWS seal water tank is attributed to the nitrogen pressure regulator. Analysis of the pressure regulator will be performed to determine the root cause for this failure.

The procedure for test PT-R6 did not sufficiently specify the test termination criteria. No upper limit test criteria for the air motor pressure was identified, which would have terminated the testing of a main steam safety valve and prevented its lifting at a higher pressure than desired.

Finally, during the performance of PT-R6, the verbal communications established between the test technician and the control room were not consistent. This was contrary to the requirements of the test and the expectations of management. Although the test crew did establish communication with the control room during the beginning portions of the test, this communication was discontinued after the third valve test. Also, the pre-job briefings were not complete in addressing potential failure conditions, termination criteria, and potential safety consequences to personnel within the working area. Consequently, the control of this evolution was less than adequate with respect to the monitoring of potential adverse responses.

CORRECTIVE ACTION:

Due to the nature and complexity of this event, a multi-disciplinary Con Edison team was immediately formed to investigate this event. The scope of the team's investigation included the performance of a systematic and thorough root cause analysis of the event and the identification of corrective actions.

Subsequently, a team including industry experts knowledgeable in the operation and maintenance of the Westinghouse DB-50 breakers was formed to utilize all available industry experience to revise our breaker inspection procedures.

The above mentioned investigations are on-going. When these investigations are completed, we will provide a summary of conclusions and corrective actions in a supplement to this report. This summary will include the results of our investigations of the IVSWS nitrogen gas pressure regulator and the DB-50 breakers.