To: James P. O'Reilly
Directorate of Regulatory Operations
Region I
631 Park Avenue
King of Prussia, Pennsylvania 19406

From: Jersey Central Power & Light Company
Oyster Creek Nuclear Generating Station Docket #50-219
Forked River, New Jersey 08731

Subject: Abnormal Occurrence Report No. 50-219/74/ 10

The following is a preliminary report being submitted in compliance with the Technical Specifications paragraph 6.6.2.

Preliminary Approval:

J. T. Carroll, Jr. FDE 2/11/74
J. T. Carroll, Jr. Date

cc: Mr. A. Giambusso

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OY(R CREEK NUCLEAR GENERATING ST/ ON
FORKED RIVER, NEW JERSEY 08731

Abnormal Occurrence
Report No. 50-219/74/10

IDENTIFICATION OF OCCURRENCE: Violation of the Technical Specifications, paragraph 2.3.7. Low Pressure Main Steam Line Pressure Switches RE23A, C, and D were found to trip at pressures less than minimum required value of 860 psig.

This event is considered to be an abnormal occurrence as defined in the Technical Specifications, paragraph 1.15A.

CONDITIONS PRIOR TO OCCURRENCE:

X	Steady State Power		Routine Shutdown Operation
	Hot Standby		Load Changes During Routine Power Operation
	Cold Shutdown		Other (Specify)
	Refueling Shutdown		
	Routine Startup Operation		

The major plant parameters at the time of the occurrence were:

Power: Reactor, 1908 MWt
Electric, 672 MWe
Flow: Recirc., 60.2×10^6 lb/hr
Feed., 7.13×10^6 lb/hr
Reactor Pressure: 1020 psig
Stack Gas: 28,357 µCi/sec

DESCRIPTION OF OCCURRENCE: On Friday, February 8, 1974, at 1030, while performing a surveillance test on the four Main Steam Line Low Pressure Switches, it was discovered that RE23A, C, and D tripped at 850, 856 and 855 psig, respectively. These values are below the minimum required trip point of 860 psig which is derived by adding to the Technical Specification limit of 850 psig a 10 psig head correction factor. (See Attachment 1 for test results.)

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APPARENT CAUSE OF OCCURRENCE:	X Design ___ Manufacture ___ Installation/ Construction ___ Operator	___ Procedure ___ Unusual Service Condition Inc. Environmental ___ Component Failure ___ Other (Specify)

Sensor drift is a recognized problem and work is in progress to formulate a final solution. The steps required to achieve this end were delineated in Abnormal Occurrence Report No. 73-30 and restated in Abnormal Occurrence Report No. 74-9.

ANALYSIS OF OCCURRENCE:

As indicated in the bases of the Technical Specification, "The low pressure isolation of the Main Steam Lines at 850 psig was provided to give protection against fast reactor depressurization and the resultant rapid cooldown of the vessel. Advantage was taken of the scram feature which occurs when the Main Steam Isolation Valves are closed to provide for reactor shutdown so that high power operation at low reactor pressure does not occur, thus providing protection for the fuel cladding integrity safety limit."

The adverse consequences of reactor isolation occurring at reactor pressure approximately 10 psig below the specified minimum value of 860 psig is limited to those effects attendant to a greater than normal reactor cooldown rate. The fuel cladding integrity safety limit only comes into effect for power operation at reactor pressures less than 600 psig or for power operation greater than 354 MWt with less than 10% recirculation flow. Therefore, the consequences of a 10 psig lower than normal reactor isolation and scram setpoint has no threatening effect whatsoever( ) the fuel cladding integrity.

The effects of a too rapid cooldown due to the lower isolation pressure are inconsequential since there is approximately a 1°F difference between the saturation temperature for 860 psig and 850 psig.

CORRECTIVE ACTION:

Continuing corrective actions being taken at this time are as stated in Abnormal Occurrence Report No. 74-9 and as restated herein:

1. Investigation is being conducted into the basis for the steam line low pressure setting of 850 psig. Development of a Technical Specification change to lower the setpoint will follow if results of transient analyses indicate this possibility (see Abnormal Occurrence Report No. 73-30).
2. Vendor recommendations to possibly reduce or eliminate the sensor setpoint drift problem will be evaluated as soon as they are available (letter to Mr. A. Giambusso from Mr. D. A. Ross, dated December 24, 1973).

FAILURE DATE:

Manufacturer data pertinent to these switches are as follows:

Mercoid Corp. (subsidiary of Barksdale)
Los Angeles, California
Pressure Actuated Switch
Model 372
Catalog #372-6SS49A-293
Range 20-1400 psig
Proof Psi. 1750 G

Prepared by: Arthur H. Roal Date: 2/11/74

The as found switch settings were:

	Test Results
RE23A	850 psig
RE23B	860 psig
RE23C	856 psig
RE23D	855 psig

The pressure switches were then recalibrated and checked to actuate as follows:

	Test Results
RE23A	860 psig
RE23C	860 psig
RE23D	860 psig

Jersey Central Power & Light Company

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MEMBER OF THE
General Public Utilities Corporation
SYSTEM

February 7, 1974

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U.S. ATOMIC ENERGY COMMISSION
Regulatory
Mail Section

Mr. Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Directorate of Licensing
United States Atomic Energy Commission
Washington, D. C. 20545

50-219

Dear Mr. Ziemann:

Subject: Electromatic Relief Valve Operation Summary

This letter serves as an addendum to my previous letter addressed to you dated January 18, 1974. Its purpose is to report additional electromatic relief valve operations not accounted for in the above referenced letter. Since formal records of relief valve operations have not been maintained in the past, we relied upon operating charts, reports, and log books to determine valve actuations.

This now completes our response to your request regarding electromatic relief valve operation at our Oyster Creek Nuclear Generating Station.

Very truly yours,

Ivan R. Finfrock
Ivan R. Finfrock, Jr.
Vice President

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Records Used in Determining the Electromatic Relief Valve Operations

A compilation of scram reports was used to determine the date of those transients which would be severe enough to cause the steam pressure to exceed the setpoint of the Electromatic Relief Valves. The charts which record the downcommer temperature were then employed to verify the suspected operation of the valves on the determined dates. Due to the difficulty in reading this particular chart and a time period of downcommer temperature recorder inoperability, a cross-check was made with the steam pressure chart, in which case it was assumed that if the steam pressure reached or exceeded the setpoint of the electromatics, the valves were automatically opened. The Shift Foreman and Control Room log books were also checked on these particular dates, but no consistent record of valve operation could be found in these logs.

Automatic depressurization surveillance records were used to determine those valve operations which occurred during surveillance testing of the valve. These tests are included in the summary. In order to include other tests which might not be included in the surveillance records, the Control Room log book was examined during all periods of reactor startup and reactor shutdown operation, which is a time when manually initiated relief valve operations are most expected due to lower reactor pressure.

The operations summaries of the six-month reports were also examined, and one valve operation was found in this record; it is included in the summary.

Summary of Additional Electromatic Relief Valve Operation
January 1, 1971 to December 31, 1971

Date of Operation: September 18, 1971

Purpose of Operation: Valve operability test.

Mode of Initiation: Manual

Reactor Conditions Prior to Operation: The reactor was subcritical with the reactor pressure at 600 psig.

Comments: During a reactor shutdown, the Electromatic Relief Valves were manually tested at a pressure of 600 psig.

Summary of Additional Electromatic Relief Valve Operation
January 1, 1973 to December 31, 1973

Date of Operation: April 13, 1973

Purpose of Operation: Valve Operability Test

Mode of Initiation: Manual

Reactor Conditions Prior to Operation: The reactor was subcritical with the reactor pressure at 230 psig and coolant temperature at 400°F.

Comments: Following a scram, the Electromatic Relief Valves were manually tested at the above conditions.

Date of Operation: November 25, 1973

Purpose of Operation: Primary coolant system depressurization.

Reactor Conditions Prior to Operation:

Steam Flow	6.6×10^6 lbm/hr
Reactor Pressure	1020 psig
Electrical Output	629 MW (e)

Comments: The reactor scrammed due to a high neutron flux caused by a pressure spike which collapsed voids in the core. The pressure increased to the setpoint of the Electromatic Relief Valves of 1070 psig, causing them to open.