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/ 1

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

Preliminary Approval:

J. T. Carroll 1/7/74
J. T. Carroll, Jr. Date

cc: Mr. A. Giambusso

B/684

Report Date: 1/7/74 Occurrence: 1105

OYSTER CREEK NUCLEAR GENERATING STATION
FORKED RIVER, NEW JERSEY 08731

Abnormal Occurrence
Report No. 50-219/74/ 1

IDENTIFICATION OF OCCURRENCE: Violation of the Technical Specifications, paragraph 2.3.7, Low Pressure Main Steam Line Pressure Switch (RE23C) was found to trip at a pressure less than 850 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
- Hot Standby
- Cold Shutdown
- Refueling Shutdown
- Routine Startup Operation
- Routine Shutdown Operation
- Load Changes During Routine Power Operation
- Other (Specify)

The major plant parameters at the time of the event were as follows:

Power - Core, 1830 MWt
Elec., 642 MWe (g)
Flow - Recirc., 60.2 $X10^6$ #/hr
Feed., 6.77 $X10^6$ # hr
Stack Gas - 24,000 µCi/Sec

DESCRIPTION OF OCCURRENCE: On Friday, January 4, 1974 at 1105, while performing surveillance testing on the four Main Steam Line Low Pressure Switches, RE23A, B, C, and D, RE23C was found to trip at 841 psig which is 9 psig below the minimum required setpoint of 850 psig. The other pressure switches were found to trip at the following values:

RE23A - 850 psig
RE23B - 851 psig
RE23D - 851 psig

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APPARENT CAUSE OF OCCURRENCE:

___ Design
___ Manufacture
___ Installation/ Construction
___ Operator
___ Procedure
___ Unusual Service Condition Inc. Environmental
Component Failure
___ Other (Specify)

At this time, the cause of this event, which has occurred in the past, has yet to be determined. In the past, the manufacturer had been contacted and had informed plant personnel that the problem of setpoint drift has been recognized with this type of instrument. An investigative program has been initiated by the manufacturer, but no formal report on the results issued as of yet.

ANALYSIS OF OCCURRENCE:

As indicated in the bases of the Technical Specifications, "The low pressure isolation of the Main Steam Lines at 850 psig was provided to give protection against fast reactor depressurization and the resulting rapid cooldown of the vessel. Advantage was taken of the scram feature which occurs when the Main Steam Line 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."

With regards to power operation below 850 psig and the attendant effects on the fuel cladding integrity safety limit, power level must be limited when pressure is less than 600 psig or flow is less than 10% to 354 MWt or approximately 18.3% of rated. As stated in the Technical Specifications, "The value is applicable to ambient pressure and no flow conditions. For any greater pressure or flow conditions there is increased margin." The fuel cladding integrity safety limit curve has been developed and is applicable for pressure in excess of 600 psig. Therefore, whether a reactor scram occurs at 850 psig or 841 psig has little safety significance since no severe restrictions on critical heat flux are imposed until pressure is less than 600 psig. In addition, the remaining three (3) sensors were operable and would have performed the protective action, if required and at the proper setpoint of ≥850 psig.

CORRECTIVE ACTION: The pressure switch (RE23C), upon discovery of this condition, was reset to 850 psig to conform with the Technical Specifications.

Recommendations to correct the abnormal occurrence and prevent repetition of the occurrence and of similar occurrences will be provided after review by the Plant Operations Review Committee.

FAILURE DATA: Manufacturer data pertinent to these switches are as follows:

Meletron Corp. (subsidiary of Barksdale)
Los Angeles, California
Pressure Actuated Switch
Model 372
Catalog #372-6SS49A-293
Range 850 G Dec.
Proof Psi. 1750 G

Prepared by: James P. Batman Date: Jan. 4, 1974

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Jersey Central Power & Light Company

MADISON AVENUE AT PUNCH BOWL ROAD • MORRISTOWN, N. J. 07960 • 201-539-6111

MEMBER OF THE General Public Utilities Corporation SYSTEM

January 7, 1974

Mr. A. Giambusso
Deputy Director for Reactor Projects
Directorate of Licensing
United States Atomic Energy Commission
Washington, D. C. 20545

Dear Mr. Giambusso:

Subject: Oyster Creek Station
Docket No. 50-219
Reset Problem with
Test Spool Valve for
Main Steam Isolation Valve NS04A

This letter serves to report the failure of the test spool valve for NS04A to properly reset after a 5% closure test. The faulty operation of the test valve resulted in a condition which may have caused the main steam isolation valve (MSIV) to close slower than normal should an isolation signal have occurred during the test interval. This event is considered to be an abnormal occurrence as defined in the Technical Specifications, paragraph 1.15D. Notification of this event, as required by the Technical Specifications, paragraph 6.6.2.a, was made to AEC Region I, Directorate of Regulatory Operations by telephone and telecopier on Thursday, December 20, 1973.

During the daily 5% closure exercises conducted on MSIV NS04A on December 18 and 19, pressure transients of 6 psig and 10 psig, respectively, were observed due to apparent valve overtravel. An analysis of the problem indicated that possibly the overtravel was caused by the test spool valve not resetting properly. On December 19, the reactor power was reduced to 853 MWt (<40% of rated power) to facilitate further examination of the test spool valve assembly. A 5% valve closure test was performed with an instrument technician observing the valve. The sticking condition had apparently worsened, allowing the MSIV to go approximately 50% closed before reopening in a normal manner.

During the test interval (approximately 2-3 minutes), if a situation existed whereby had MSIV NS04A received a signal to isolate, it is unlikely that the valve would have met the required closure time of ≤10 seconds since the exhaust air from the underside of the valve operator piston must pass through the test valve assembly. Because of the malpositioned valve assembly, the exhaust air from the operator would have been restricted, thereby increasing

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the valve closure time. The consequences of this event are minimized since NS04A would eventually close and the redundant valve, NS03A, would have provided isolation and closed in the required time, in the event of a reactor isolation. Furthermore, the "slow closure condition" was only present for the length of time the 5% closure test was in progress (2-3 minutes).

The test spool valve assembly for NS04A was replaced. A 5% closure test and also a full closure timing test (6.7 seconds) were satisfactorily completed.

Upon removal and disassembly of the test spool valve, foreign material resembling red powder was found on and between the spool and sleeve assembly. It has not been conclusively determined that this material was the cause of the valve sticking. The spool assembly is being sent to a testing lab for further examination.

Until the laboratory analysis on the defective test spool assembly is completed and a cause for the valve sticking is conclusively determined, the operation of the test valves on all four MSIVs will be closely observed for further evidence of this problem.

It is recognized that sticking is not a new problem with the 4-way air-and-spring operated valves (Numatics) used in the control circuits of the MSIVs at Oyster Creek, and work is in progress to formulate a modification to provide final resolution of this situation. The General Electric Company has recommended a modification to the MSIVs in which the present "Numatic" valves would be replaced with "Automatic" valves and manifold assemblies. This modification is currently being reviewed by appropriate groups within Jersey Central Power & Light Company's Generating Department and, if adopted, an attempt will be made to have the installation work completed during the next refueling outage presently scheduled for April 1974.

Enclosed are forty copies of this report.

Very truly yours,

Donald A. Ross

Donald A. Ross
Manager, Nuclear Generating Stations

cs

cc: Mr. J. P. O'Reilly, Director
Directorate of Regulatory Operations, Region I

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Jersey Central Power & Light Company

MADISON AVENUE AT PUNCH BOWL ROAD • MORRISTOWN, N. J. 07960 • 201-539-6111

MEMBER OF THE General Public Utilities Corporation SYSTEM

January 3, 1974

Mr. A. Giambusso
Deputy Director for Reactor Projects
Directorate of Licensing
United States Atomic Energy Commission
Washington, D. C. 20545

Dear Mr. Giambusso:

Subject: Oyster Creek Station
Docket No. 50-219
Spill of Chromated Water

On December 20, 1973 at 7:00 a.m., a spill of approximately 3400 gallons of lightly contaminated radioactive chromated water occurred at the Oyster Creek Nuclear Generating Station. Mr. Caphton of the Region I Regulatory Operations Office was notified at 9:00 a.m. on December 20, 1973 of this incident and requested that a letter of explanation be forwarded for their review.

The spill occurred as a result of a drain line from a temporary storage tank containing the water freezing and subsequently cracking during the severe cold weather over the past several days. Warm weather on December 20 thawed the drain line and allowed a portion of the water stored in the tank to drain onto the ground inside the plant security area. Upon discovery by operating personnel during a routine plant tour, the leak was stopped and the water contained in a relatively small area (40' x 150') adjacent to the radwaste building. Immediate efforts to clean up the spill were made by pumping the water laying on the surface of the ground into containers. A total of approximately 1300 gallons was recovered in this manner. The remaining water in the storage tank was pumped into containers. A total of approximately 1300 gallons was recovered from the tank. The amount of water that leaked into the soil amounted to approximately 2100 gallons, containing a total of 1.7 pounds of $CrO_4^=$. This water also contained approximately 5.17×10^{-4} µCi/cc or a total of 0.004 Ci of activity.

Shallow wells are being drilled to permit subterranean water analyses, core samples taken, and puddle water analyzed to determine the extent of the contamination and establish the best means of completing the clean up efforts.

On December 22, four new shallow wells were drilled in the area of the spill. Samples from these wells and an existing well in the nearby vicinity were analyzed for chromate. The well samples indicated that chromate was not present.

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On December 20, representatives of the USAEC, State of New Jersey, Department of Environmental Protection and the Federal EPA were advised of this event. The following day, a news release was mailed to area newspapers, copy of which is attached.

Mr. David Longstreet of the New Jersey Department of Environmental Protection came to the site on December 20, 1973 and obtained samples of the water in the leaking tank, the water underneath the tank, the soil underneath the tank and background soil and water samples some distance from the area of concern. Mr. Fisher of the New Jersey Bureau of Radiation Protection also came to the site and took a sample of water from the tank and a sample of the soil underneath the tank. Analysis results from these samples have not been made available as of this time.

Surveill progr ?

Plans are being made to install some temporary heating in the remaining eleven tanks until a rubber liner can be properly set up which will eventually be capable of containing the water presently stored in temporary tanks. Because of unanticipated difficulties in developing an acceptable chromate disposal method, we intend to proceed with an alternate storage method to replace the temporary tanks. This will consist of a 50,000-gallon rubber bag placed in a trench. This trench will be lined to prevent water seeping into the ground in the unlikely event of a leak developing in the rubber bag. Our original approach was to use a precipitation method for the ultimate disposal of this chromate. This method has not been able to meet the State of New Jersey, Department of Environmental Protection's requirement of 0.05 mg/l concentration prior to discharge. The Jersey Central Power & Light Company's Generation Engineering Department is presently investigating the feasibility of an activated charcoal absorption method to meet the State's discharge requirements.

Enclosed are forty copies of this report.

Very truly yours,

Donald A. Ross

Donald A. Ross
Manager, Nuclear Generating Stations

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Attachment

cc: Mr. J. P. O'Reilly, Director
Directorate of Regulatory Operations, Region I

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Jersey Central Power & Light Company

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MEMBER OF THE General Public Utilities Corporation SYSTEM

January 4, 1974

Mr. A. Giambusso
Deputy Director for Reactor Projects
Directorate of Licensing
United States Atomic Energy Commission
Washington, D. C. 20545

RECEIVED JAN 1974 U.S. ATOMIC ENERGY COMMISSION

Dear Mr. Giambusso:

Subject: Oyster Creek Station
Docket No. 50-219
Drift in Set Point of
Isolation Condenser High Flow Sensors

This letter serves to report the failure of four isolation condenser high flow sensors to trip on ΔP values that conform to those stated in Technical Specification Table 3.1.1.H ($\leq$20 psid - steam, $\leq$27 inches ΔP H_2O - condensate).

This event is considered to be an abnormal occurrence as defined in the Technical Specifications, paragraph 1.15A. Notification of this event, as required by the Technical Specifications, paragraph 6.6.2.a, was made to AEC Region I, Directorate of Regulatory Operations, by telephone on Friday, December 21, 1973, and by telecopier on Wednesday, December 26, 1973.

On Friday, December 21, 1973, while performing surveillance testing on the isolation condenser high flow sensors (two each per condensate and steam line per condenser), four of the sensors were found to trip at set points in excess of their respective limits. Specifically, the sensors of concern and their corresponding "as found" set points were as follows:

<u>Condensate</u>

"A" Condenser 1B11A1 - 29" of water
1B11A2 - 31" of water
1B11B2 - 29" of water

<u>Steam</u>

"B" Condenser 1B05B1 - 22 psid

The reason for the drift in set points is yet to be determined.

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Upon discovery of the condition, the instrument technician performing the surveillance test reset the affected differential pressure (d/p) sensors to their required values ($\leq$27 inches H_2O on condensate and $\leq$20 psid on the steam).

The safety implication of this event is minimal, since the sensors were operable and would have performed their function. Past data has shown that upon initiating the condensers, the condensate line senses d/p in excess of 60 inches of water; therefore, although the set point had drifted, the new value was well within the range of the expected signal and the sensor was capable of performing its protective action.

A review of the surveillance procedures will be made to insure that the test and calibration methods and instruments are consistent with the capability of the instruments under surveillance.

In addition, an investigation will be made into the feasibility of submitting a Technical Specification change to include tolerances on instrument trip set points to account for instrumentation design accuracies where applicable.

Enclosed are forty copies of this report.

Very truly yours,

Donald A. Ross

Donald A. Ross
Manager, Nuclear Generating Stations

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Enclosures

cc: Mr. J. P. O'Reilly, Director
Directorate of Regulatory Operations, Region I

Jersey Central Power & Light Company

MADISON AVENUE AT PUNCH BOWL ROAD • MORRISTOWN, N. J. 07960 • 201-539-6111
MEMBER OF THE
General Public Utilities Corporation

January 7, 1974

Mr. A. Giambusso
Deputy Director for Reactor Projects
Directorate of Licensing
United States Atomic Energy Commission
Washington, D. C. 20545

Dear Mr. Giambusso:

Subject: Oyster Creek Station
Docket No. 50-219
Reset Problem with
Test Spool Valve for
Main Steam Isolation Valve NS04A

This letter serves to report the failure of the test spool valve for NS04A to properly reset after a 5% closure test. The faulty operation of the test valve resulted in a condition which may have caused the main steam isolation valve (MSIV) to close slower than normal should an isolation signal have occurred during the test interval. This event is considered to be an abnormal occurrence as defined in the Technical Specifications, paragraph 1.15D. Notification of this event, as required by the Technical Specifications, paragraph 6.6.2.a, was made to AEC Region I, Directorate of Regulatory Operations by telephone and telecopier on Thursday, December 20, 1973.

During the daily 5% closure exercises conducted on MSIV NS04A on December 18 and 19, pressure transients of 6 psig and 10 psig, respectively, were observed due to apparent valve overtravel. An analysis of the problem indicated that possibly the overtravel was caused by the test spool valve not resetting properly. On December 19, the reactor power was reduced to 853 MWt (<40% of rated power) to facilitate further examination of the test spool valve assembly. A 5% valve closure test was performed with an instrument technician observing the valve. The sticking condition had apparently worsened, allowing the MSIV to go approximately 50% closed before reopening in a normal manner.

During the test interval (approximately 2-3 minutes), if a situation existed whereby had MSIV NS04A received a signal to isolate, it is unlikely that the valve would have met the required closure time of <10 seconds since the exhaust air from the underside of the valve operator piston must pass through the test valve assembly. Because of the malpositioned valve assembly, the exhaust air from the operator would have been restricted, thereby increasing

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the valve closure time. The consequences of this event are minimized since NS04A would eventually close and the redundant valve, NS03A, would have provided isolation and closed in the required time, in the event of a reactor isolation. Furthermore, the "slow closure condition" was only present for the length of time the 5% closure test was in progress (2-3 minu~es).

The test spool valve assembly for NS04A was replaced. A 5% closure test and also a full closure timing test (6.7 seconds) were satisfactorily completed.

Upon removal and disassembly of the test spool valve, foreign material resembling red powder was found on and between the spool and sleeve assembly. It has not been conclusively determined that this material was the cause of the valve sticking. The spool assembly is being sent to a testing lab for further examination.

Until the laboratory analysis on the defective test spool assembly is completed and a cause for the valve sticking is conclusively determined, the operation of the test valves on all four MSIVs will be closely observed for further evidence of this problem.

It is recognized that sticking is not a new problem with the 4-way air-and-spring operated valves (Numatics) used in the control circuits of the MSIVs at Oyster Creek, and work is in progress to formulate a modification to provide final resolution of this situation. The General Electric Company has recommended a modification to the MSIVs in which the present "Numatic" valves would be replaced with "Automatic" valves and manifold assemblies. This modification is currently being reviewed by appropriate groups within Jersey Central Power & Light Company's Generating Department and, if adopted, an attempt will be made to have the installation work completed during the next refueling outage presently scheduled for April 1974.

Enclosed are forty copies of this report.

Very truly yours,

Donald A. Ross

Donald A. Ross
Manager, Nuclear Generating Stations

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cc: Mr. J. P. O'Reilly, Director
Directorate of Regulatory Operations, Region I

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Jersey Central Power & Light Company

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MEMBER OF THE General Public Utilities Corporation SYSTEM

January 2, 1974

Mr. A. Giambusso
Deputy Director for Reactor Projects
Directorate of Licensing
United States Atomic Energy Commission
Washington, D. C. 20545

Dear Mr. Giambusso:

Subject: Oyster Creek Station
Docket No. 50-219
Unexplained Isolation of the "B" Isolation Condenser

The purpose of this letter is to report to you, as a matter of interest, an unexplained isolation of the "B" isolation condenser during a routine reactor cooldown operation that took place on November 25, 1973. Notification of this incident was verbally made to Mr. E. Greenman, AEC Region I, Directorate of Regulatory Operations on Monday, November 26. 1973.

Each of the two isolation condensers is equipped with flow sensing differential pressure (d/p) instrumentation on the condensate return and steam supply elbows. The system is designed to isolate when the d/p instrumentation senses a flow three times rated which would be indicative of a ruptured lire. This condition corresponds to a d/p signal of 27 inches of water on the condensate return line and 20 psi on the steam supply.

Experience has shown that during normal system initiation, the inrush flow through the condensate elbow exceeds the isolation d/p set point significantly. The peak had, in the past, caused the d/p sensor to exceed its uppermost limit causing mechanical damage to the Barton sensor. To correct this problem, dampening snubbers were added to the instrument sensing lines which suppressed the peak to within the instrument limits. The peak differential pressure sensed on the condensate elbow with the snubbers installed was measured to be approximately 64 inches of water and the time the set point was exceeded was approximately 15.5 seconds. Since the inrush transient is intrinsic to the system, it was necessary to delay condenser isolation for a time period in excess of the pulse duration of 15.5 seconds. A time delay relay set for 35 seconds was used to satisfy this condition and still permit isolation in the required 60 seconds.

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It should be noted that the duration of the pulse is directly attributable to the time constant of the snubbers used and if the snubber characteristics change with time the apparent pulse duration can be increased.

After the November 25, 1973 scram, the "B" isolation condenser was initiated in order to control reactor pressure by opening the DC drain valve, V-14-35. The "rupture alarm" was reported to have sounded immediately upon initiation but an isolation of the condenser did not occur at this time. Normally, isolation would commence exactly the same time as the alarm comes in, since both the alarm and isolation are actuated when the time delay relay completes its cycle without the d/p sensor returning to below 24 inches of water (the reset point). It was then reported that as the valve was being closed (approximately one-half minute after initiation), an isolation occurred and the rupture alarm again sounded. The first alarm is thought to be spurious due to adjacent relay "noise", a condition which has been observed in the past. However, based on past experience with the system, it is felt that a true isolation signal was present coincidently at this same time which caused the condenser isolation time delay relay to start its cycle. The relay timed out just as the condenser was being returned to the standby condition, hence the isolation and the second alarm actuation. This event indicated that the pulse duration, as sensed by the d/p instrumentation, exceeded the isolation set point for a period in excess of 35 seconds which is a marked increase over the normal pulse duration of 15.5 seconds as measured previously.

The increase in pulse duration was traced to a corresponding increase in the condensate instrument sensing line snubbers time constant. There are two snubbers per d/p sensor and there are four separate d/p sensors per condenser, two on each of the condensate and steam legs. A pressure decay test was performed on each sensor for the condensate legs on both the "A" and "B" isolation condensers. The test basically consisted of imposing a d/p of 60 inches of water on each sensor and measuring the time required to decay down to the reset d/p of 24 inches of water through the two snubbers in the sensing lines. The following are the results of these tests:

Isolation Condenser "A"
- Sensor B11A1 - 2 seconds
- Sensor B11A2 - 35 seconds

Isolation Condenser "B"
- Sensor B11B1 - 17 seconds
- Sensor B11B2 - 10 seconds

The time constants for both the "A" and "B" isolation condensers are felt to be large enough to cause an isolation during normal initiation. It was initially reported that both condensers had isolated but it has since been determined that only the "B" isolation condenser did indeed isolate.

The snubbers were replaced and the pressure decay tests yielded results which are as follows:

Isolation Condenser "A"
Sensor B11A1 - 2 seconds
Sensor B11A2 - 4 seconds

Isolation Condenser "B"
Sensor B11B1 - 3.5 seconds
Sensor B11B2 - 3 seconds

Had a bonifide condition existed whereby the isolation condensers were required to serve as a back-up heat sink under reactor isolation conditions, the "B" condenser would have isolated, thereby eliminating it as a back-up heat sink. The "A" condenser would have performed this function, however, since a single operable condenser is capable of removing the decay heat after a scram from 1950 MWt for 45 minutes before makeup is required and with approximately 107,500 gallons of makeup water ull cooldown could be achieved. The safety significance, therefore, is in the loss of redundancy in the system. It should be noted that the "B" isolation condenser was reinitiated and it again isolated under very much the same circumstances. After the third initiation, however, the condenser did not isolate and it was successfully used many times, along with the "A" condenser, throughout the reactor cooldown operation. This was due to establishing "hot" condensate in the condenser which, because of its lower density reduces the magnitude and duration of the inrush flow transient. This was evidenced by a test run on November 14, 1973 which showed that the transient peak was reduced from a cold condenser initial peak d/p of 64 inches of water and a duration of 15.5 seconds over the trip point to a peak of 32.2 inches of water and 4.1 seconds for a "hot" condenser.

Furthermore, had a bonifide reactor isolation occurred, it should have been possible to manually initiate and reset the "B" condenser until isolation no longer occurred.

In the future, we plan to include the pressure decay tests with the regular line break sensor surveillance test to permit early detection and correction of slow instrument response.

Enclosed are forty copies of this report.

Very truly yours,

Donald A. Ross

Donald A. Ross
Manager, Nuclear Generating Stations

cs

cc: Mr. J. P. O'Reilly, Director
Directorate of Regulatory Operations, Region I