

11-1-73

LAW OFFICES OF
LEBOEUF, LAMB, LEIBY & MACRAE
1757 N STREET, N.W.
WASHINGTON, D.C. 20036

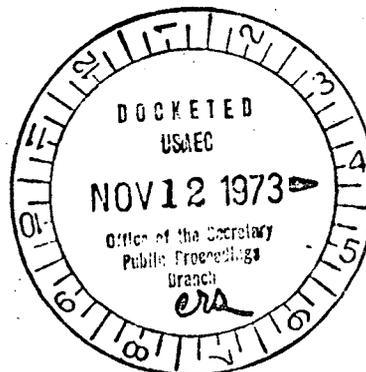
ARVIN E. UPTON
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WASHINGTON PARTNERS

ONE CHASE MANHATTAN PLAZA
NEW YORK, N.Y. 10005

November 1, 1973

WASHINGTON TELEPHONE
202-872-8668

CABLE ADDRESS
LALALU, WASHINGTON D.C.



Anthony Z. Roisman, Esq.
Berlin, Roisman & Kessler
Fourth Floor
1712 N Street, N.W.
Washington, D.C. 20036

Re: Consolidated Edison Company
of New York, Inc.
Indian Point Unit No. 2
AEC Docket No. 50-247

Dear Mr. Roisman:

Enclosed are the following documents: letter from Mr. Cobean to Mr. O'Reilly, dated October 12, 1973; letter from Mr. Cobean to Mr. O'Leary, dated October 26, 1973; letter from Mr. Cobean to Mr. O'Reilly, dated October 31, 1973; and letter from Mr. Cobean to Mr. O'Leary, dated November 1, 1973.

Very truly yours,

LEBOEUF, LAMB, LEIBY & MACRAE
Attorneys for Applicant

By Edward L. Cohen
Edward L. Cohen

BY HAND

Enclosures

cc w/encs: See next page

cc w/encs:

Samuel W. Jensch, Esq.

Mr. R. B. Briggs

Dr. John C. Geyer

Myron Karman, Esq.

Angus Macbeth, Esq.

J. Bruce MacDonald, Esq.

Hon. Louis J. Lefkowitz

Secretary, U.S. Atomic Energy Commission
Atomic Safety and Licensing Board Panel

RECEIVED

Consolidated Edison Company of New York, Inc.
4 Irving Place, New York, NY 10003

LeBoeuf, Lamb, Leiby & MacRae

RECEIVED
NOV - 8 1973

Per *J. J. [Signature]*

October 12, 1973

Re: Indian Point Unit No. 2
Facility Operating License
DPR-26

Mr. James P. O'Reilly, Director
Regulatory Operations, Region I
U. S. Atomic Energy Commission
631 Park Avenue
King of Prussia, Pennsylvania 19406



Dear Mr. O'Reilly:

In accordance with the requirements of Technical Specification 5.12.2 of Facility Operating License DPR-26, we wish to inform you of an abnormal occurrence (A.O. 3 2-'3) which was identified on October 11, 1973 at approximately 0345 hours. At 0224 hours on October 11, 1973, Unit No. 2 was tripped off the line while loaded at approximately 385 Mwe (50% power) due to loss of generator field relay action. Approximately eighty minutes later, portable instrumentation installed in the Primary Auxiliary Building indicated gaseous and particulate radioactivity in this building. This condition was confirmed by the plant vent gaseous and particulate activity monitors.

Immediately upon receipt of the alarm from the portable monitor in the P.A.B., charcoal and millipore samples of the plant vent gaseous discharge were taken and analyzed. In addition, a sample of the portable monitor filter paper was also analyzed for particulate activity. On the basis of these analyses the maximum release rate for gross radioactivity of gaseous effluents from the site was less than ten percent of the maximum permissible release rate defined in the facility Technical Specifications. The duration of the release from the site was on the order of one hour. An investigation is underway concerning the events associated with this occurrence.

James P. O'Reilly

- 2 -

October 12, 1973

Re: Indian Point Unit No. 2
Facility Operating
License IPR-26

Mr. Anthony Fasano of your office was notified of this occurrence by Mr. John M. Makepeace by telephone on October 11, 1973.

Very truly yours,

Warren R. Cobean

Warren R. Cobean, Jr.
Manager, Nuclear Power
Generation

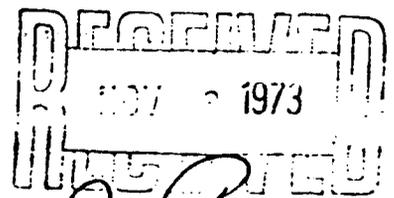
JMM:daf

Copy to: John F. O'Leary



Consolidated Edison Company of New York, Inc.
4 Irving Place, New York, N Y 10003

Loebel, Lorch, Lohy & MacRae

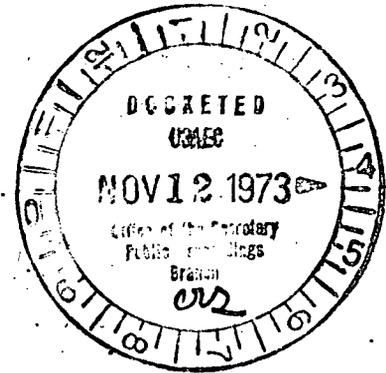


J. Goxey

October 26, 1973

Re: Indian Point Unit No. 2
AEC Docket No. 50-247
Facility Operating
License DPR-26
A.O. 3-2-13

Mr. John F. O'Leary
Directorate of Licensing
Office of Regulation
U.S. Atomic Energy Commission
Washington, D. C. 20545



Dear Mr. O'Leary:

The following report is provided pursuant to the requirements of Section 6.12.2(a) of the Technical Specifications for Facility Operating License DPR-26.

On October 11, 1973, approximately one hour after a reactor shutdown from 50% of rated power, portable instrumentation in the Primary Auxiliary Building indicated that an unplanned release of gaseous and particulate radioactivity had been made within the building. The release was limited to the P.A.B. and was removed from there via the normal plant vent system over a period of approximately one hour.

Immediately upon receipt of the alarm from the portable monitor, investigative charcoal and millipore samples of the plant vent effluent were taken and analyzed. In addition, a sample of the portable monitor filter paper was also analyzed for particulate activity.

An investigation was undertaken to determine the cause of the indicated release. Plant components and piping which handle waste or radioactive gases were examined as possible sources. No faults in equipment or operating procedures were found during the investigation which could directly account for the release. However, the PAB ventilation system had been automatically shut down in connection with a safeguards signal which followed the reactor shutdown. On the basis of this, and the fact that no equipment malfunction or improper procedure was uncovered, we have concluded that no unplanned release from the site did occur, and that the presence of airborne activity in the PAB resulted only from the building ventilation system being in a shutdown condition during a period of

Mr. John F. O'Leary

- 2 -

October 26, 1973

Re: Indian Point Unit No.2
AEC Docket No. 40-247
Facility Operating
License DPR-26
A.O. 3-2-13

time in which liquid waste was being processed. Under normal circumstances, radioactive gases liberated from the liquid waste are directed to the plant vent via the PAB ventilation system. In this case, since the ventilation system was secured, such gases accumulated within the building and this, in turn caused annunciation of the aforementioned alarms.

Based on airborne activity levels recorded by two monitors located within the PAB, the amount of activity which accumulated in the building during the period of time the ventilation system was secured is estimated to be approximately one curie, the major portion of which was a mixture of xenon and krypton gases. The accumulation of I-131 and particulates with a half-life greater than 8 days is estimated to have been 35 μ ci and 7.4ci, respectively. The maximum concentration of airborne activity measured within the building, 3.25×10^{-5} μ ci/cc, is well below the applicable limit specified in 10 CFR 20, and the rate at which the gas was released from the site following restoration of the ventilation system was less than 10% of the Technical Specification limit.

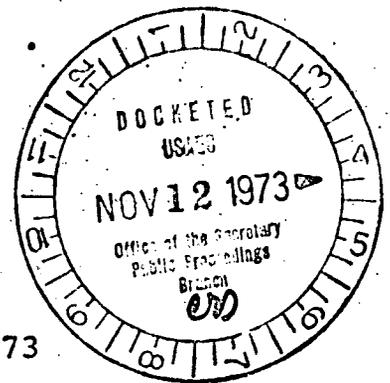
Warren R. Cobean
by John C. Couch

Warren R. Cobean, Jr.
Manager - Nuclear Power
Generation Department

cc: Mr. James P. O'Reilly
Regulatory Operations, U.S.A.E.C.



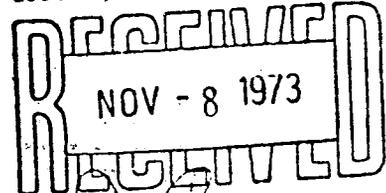
Consolidated Edison Company of New York, Inc.
4 Irving Place, New York, NY 10003



October 31, 1973

Re: Indian Point Unit No. 2
Facility Operating License
DPR-26
A.O. 3-2-15

LeBoeuf, Lamb, Leiby & MacRae



Per *J.R. [unclear]*

Mr. James P. O'Reilly, Director
Regulatory Operations, Region I
U. S. Atomic Energy Commission
631 Park Avenue
King of Prussia, Pennsylvania 19406

Dear Mr. O'Reilly,

In accordance with the requirements of Technical Specification 6.12.2 of Facility Operating License DPR-26, the following report is provided:

During the course of routine plant status inspections on October 30, 1973, leaks were observed at the 3/4 inch pipe to socket weld branch connections for vent valves S-47 and S-51 associated with the Residual Heat Removal (RHR) System. At the time of these observations, the reactor was depressurized and in the cold shutdown condition.

These venting provisions are not required for operation. Since leaks of a similar nature have occurred before on both of these connections, we plan to eliminate the pipe and valve assemblies at these particular points to eliminate any vibration induced cyclic stress which might have contributed to the failures. The vent assemblies will be replaced with socket welded plugs in the half couplings welded to the RHR System. A survey of similar connections on the RHR System has been made and no other leaks were observed.

An investigation, including, to the extent feasible, metallurgical examinations, is underway. The results of this investigation and final corrective action will be provided in a follow-up report.

Mr. Anthony Fasano of your office was apprised of the circumstances of this occurrence by Mr. John M. Makepeace, by telephone on October 30, and October 31, 1973.

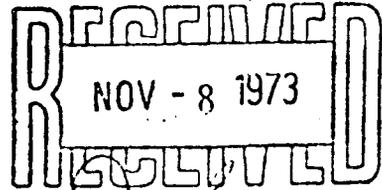
Very truly yours,

Warren R. Cobean, Jr.

Warren R. Cobean, Jr,
Manager, Nuclear Power
Generation

cc: John F. O'Leary

LeBoeuf, Lamb, Leiby & MacRae

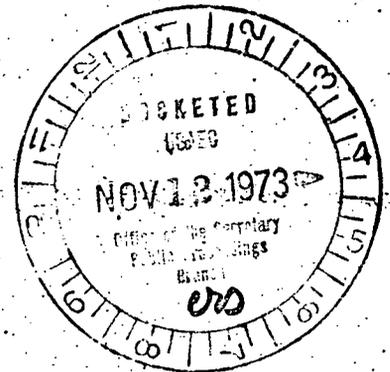


November 8, 1973

Re: Indian Point Unit No. 2
Facility Operating License
DPR-26
A.O.-3-2-14

Consolidated Edison Company of New York, Inc.
4 Irving Place, New York, NY 10003

Mr. John F. O'Leary
Directorate of Licensing
Office of Regulation
U. S. Atomic Energy Commission
Washington, D. C. 20545



Dear Mr. O'Leary:

The following report is provided pursuant to the requirements of Section 6.12.2(a) of the Technical Specifications of Facility Operating License DPR-26.

On October 17, 1973, prior to planned maintenance on the three pressurizer safety valves, to correct minor seat leakage which had been experienced, the lift pressures of these valves were checked. Two of the valves were found to be set approximately 1.0% higher than the limit set forth in section 3.1.A.3.C of the Technical Specifications. The third valve was approximately 0.6% above this limit. At the time of this observation, the reactor was in a shutdown condition.

The three Pressurizer safety valves were disassembled for inspection and reworking of the seating surfaces. Upon completion of this maintenance, they were reassembled and reset to the value specified in the Technical Specifications.

The safety valves on the Pressurizer are designed to prevent a rise in system pressure above 10% of the normal operating value and to relieve the reactor heat surge should a total loss of load occur with the plant at full power. This type of incident was examined and reported upon in the Final Safety Analysis Report under the assumptions that there is no direct control rod insertion, or pressurizer spray, and main steam safety valves and steam bypass valves to condenser do not operate. It was found that the pressurizer safety valve setpoint would not be

Mr. John F. O'Leary

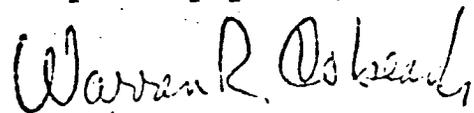
- 2 -

November 1, 1973

Indian Point Unit No. 2
Facility Operating License
DPR-26
A.O.-3-2-14

reached and that the reactor coolant system is protected against excessive pressurization due to heat surge without safety valve actuation. For these reasons, the safety implications of setpoint variances on the Pressurizer safety valves are considered to be negligible.

Very truly yours,



Warren R. Cobean, Jr.
Manager - Nuclear Power
Generation Department

cc/ J. P. O'Reilly