FEB 26 1973

Richard C. DeYoung, Assistant Director
for Pressurized Water Reactors
Directorate of Licensing

CONSOLIDATED EDISON OF NEW YORK, INC., INDIAN POINT NUCLEAR GENERATING STATION, UNIT NO. 3, (OL), DOCKET NO. 50-286

Plant Name: Indian Point Nuclear Generating Station, Unit 3
Licensing Stage: OL
Docket Number: 50-286
Responsible Branch and Project Manager: PWR-1; H. Specter
Requested Completion Date: Not Applicable
Applicant's Response Date Necessary for Completion of Next Planned Action on Project: Not Applicable.
Description of Response: Safety Evaluation Revision
Review Status: Complete

Enclosed is a revision to our section of the Safety Evaluation Report which was submitted to you on February 22, 1973. Changes have been made to the item on "Evaluation of the Integrity of the Reactor Vessel," (see pages 5 and 6 of the report), which include a statement indicating the absence of "Special Considerations," such as described in J. F. O'Leary's letter of January 12, 1972, to A. Giambusso and J. M. Hendrie, "Consideration of Reactor Pressure Vessel Integrity for Light Water Reactors."

Original signed by
R. Maccary

R. R. Maccary, Assistant Director
for Engineering
Directorate of Licensing

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Docket 50-286
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Enclosure:
Materials Engineering Branch Safety Evaluation - Item on "Evaluation of the Integrity of the Reactor Vessel"

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DATE ▶	2/26/73	2/26/73	2/26/73	2/26/73	2/26/73

REACTOR COOLANT SYSTEM

Evaluation of the Integrity of the Reactor Vessel

During installation of the reactor vessel, a hoist failed, and the vessel was dropped. A reinspection of the vessel was performed, which involved dimensional checks, visual examination, and nondestructive examination by magnetic particle, liquid penetrant, and ultrasonic methods. The results obtained from the nondestructive examinations subsequently served as a basis for assessment of possible damage to the vessel using stress analysis and fracture mechanics criteria.

A report prepared by Oak Ridge National Laboratory entitled, "Summary Report and Reinspection and Appraisal of the Indian Point Unit No. 3 Reactor Pressure Vessel Subsequent to Hoist Failure on January 12, 1971," covering the above incident and the subsequent reinspection and evaluation has been submitted to Licensing by the applicant.

Our review of the report revealed that the nondestructive examination techniques which were used were equal or better than those specified by the ASME Boiler and Pressure Vessel Code, Section III, and in fact permitted a more comprehensive examination than that originally performed which used the Code specified methods. No rejectable defects were disclosed as a result of the above indicated inspection, even though additional discontinuities were shown to be present in excess of those originally reported.

Appendix "C" of the report, which is in two parts, contains an assessment of the effects of this incident based on stress analysis and fracture mechanics. This appendix has been reviewed and evaluated.

The procedure in the first part of this appendix is inappropriate due to assumptions made relating to the stress, the imposed stress intensity, and the toughness. In the second part the toughness value that was used agrees well with an estimated lower bound reference toughness from the ASME Code, Section III, Appendix G, 1972 Summer Addenda. We believe that the calculated maximum bending stress is realistic. A critical flaw depth of approximately 4 inches was calculated. Our independent calculations, performed according to the procedures of Welding Research Council Bulletin No. 175, PVRC Recommendations on Toughness Requirements for Ferritic Materials, August 1972, confirm the results of this calculation. Further, using conservative assumptions, we have estimated that a 4 inch deep flaw, assumed to exist in the most deleterious location and orientation, would have grown less than 0.001 inch due to this incident.

We concur with the findings of the report that no rejectable defects were disclosed, and that any existing flaws would not have been significantly extended as a consequence of this incident. There was no mechanical damage to the reactor vessel and, therefore, its integrity was not impaired by the drop which resulted from the hoist failure. On this basis we conclude there are no special considerations that make it necessary that potential pressure vessel failure be considered for Indian Point Nuclear Generating Station, Unit No. 3.