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March 30, 1990

Re: Indian Point Unit No. 2
Docket No. 50-247

Document Control Desk
US Nuclear Regulatory Commission
Mail Station P1-137
Washington, DC 20555

SUBJECT: NRC Generic Letter No. 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and its Impact on Plant Operations"

By letter dated January 5, 1989 the NRC staff clarified the requirements of Generic Letter 88-11, entitled "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and its Impact on Plant Operations," dated July 12, 1988, and requested us to revise our November 30, 1988 response. Accordingly, transmitted as Attachment I to this letter is our revised response to the subject generic letter. Also included in the revised response is clarification requested by the NRC staff during a telephone conference call on October 27, 1989.

Additionally, pursuant to 10 CFR §50.61(b)(1), transmitted as Attachment II to this letter, are our current and projected values of the nil ductility transition reference temperature for pressurized thermal shock evaluation ("RT_{PTS}") of the Indian Point Unit No. 2 ("IP-2") reactor vessel beltline materials. Our current projection , which incorporates the licensed stretch power rating of 3071.4 MWt, is an update to our January 22, 1986 projection. Updates are required by 10 CFR §50.61(b)(1)(2) which, in the pertinent part, states that "[analysis] must be updated whenever changes in core loadings, surveillance measurements, or other information indicate a significant change in projected values."

Should you or your staff have any questions regarding this matter, please contact Mr. Charles W. Jackson, Manager, Nuclear Safety and Licensing.

Very truly yours,

Attachments
cc: Mr. William Russell
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ATTACHMENT I

REVISED RESPONSE TO NRC GENERIC LETTER NO. 88-11

"NRC POSITION ON RADIATION EMBRITTLEMENT OF REACTOR VESSEL MATERIALS AND ITS IMPACT ON PLANT OPERATIONS"

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.
INDIAN POINT UNIT NO. 2
DOCKET NO. 50-247
MARCH, 1990

NRC Generic Letter No. 88-11 requested that licensees use the methods described in Revision 2 to Regulatory Guide 1.99 to predict the effect of neutron radiation on reactor vessel materials as required by Paragraph V.A. of 10 CFR Part 50 Appendix G.

As part of Indian Point Unit No. 2's reactor vessel surveillance program, a capsule was removed at the end of cycle 8 operation (October, 1987). The analysis of this capsule included calculations using Regulatory Guide 1.99 Revision 1 and Revision 2. By letter dated October 12, 1988 we transmitted a report, entitled "Reactor Vessel Material Surveillance Program for Indian Point Unit No. 2 - Analysis of Capsule V," pursuant to the requirements of 10 CFR Part 50 Appendix H. This report which contains the calculations and technical analysis requested by the Generic Letter was enclosed in our November 30, 1988 response. A revised version of this report which reflects the NRC staff clarifications and guidance (by letter dated January 5, 1989 and telephone conference call on October 27, 1989) is enclosed in Attachment II. Prior to startup from our Cycle 10/11 refueling outage, we will complete all required actions to implement the results of this revised analysis so as to continue to satisfy the requirements of Section V of 10 CFR Part 50, Appendix G. These actions will include the submittal of a Technical Specification amendment application to modify the pressure-temperature limits contained therein.

Additionally, in Generic Letter No. 88-11 the consideration of using Regulatory Guide 1.99 Revision 2 equations for RT_{PTS} is discussed without changing the screening criteria. A comparison of the Revision 2 results contained in the revised report with the screening criteria has been performed. The screening criteria given in the PTS Rule, 10 CFR §50.61, is 270°F for plates, forgings and axial weld materials, and 300°F for circumferential weld materials. Employing the NRC suggested methodology set forth in Regulatory Guide 1.99 Rev. 2, the comparison showed that the reactor vessel material (plate, weld, and HAZ) are still below the screening criteria of 270°F at 32 Effective Full Power Years (EFPYs). The use of the suggested Regulatory Guide 1.99 Revision 2 equations for RT_{PTS} indicates that the impact will be beyond 32 EFPYs of operation.

Regulatory Guide 1.99 guidance has always been provided to assure compliance with the requirements of 10 CFR 50 Appendix G. These requirements apply to normal heatup and cooldown limitations for iterations within the control of the operator. The PTS rule (10 CFR 50.61), however, is intended to address limits associated with hypothetical accident transient conditions. Thus, the calculation methods and criteria are different for the normal vs. accident situations. Application of the new, more conservative calculational techniques to the normal heatup and cooldown limitations addressed by Regulatory Guide 1.99 will cause a further narrowing of the operating window as indicated by Generic Letter 88-11. However, to utilize the NRC's suggested application of Regulatory Guide 1.99, Rev. 2 in the calculation of RT_{PTS} for compliance with the PTS rule for hypothetical accidents would be inappropriate without a concurrent re-evaluation by the NRC of the overall conservatisms of the analytical basis of the PTS rule. This re-evaluation would also have to consider appropriate revisions of the PTS screening criteria. Only then could a valid re-evaluation of remaining vessel service life be conducted.

ATTACHMENT II

PROJECTION OF RT_{PTS} VALUES PURSUANT TO 10 CFR 50.61

" FRACTURE TOUGHNESS REQUIREMENTS FOR PROTECTION AGAINST PRESSURIZED THERMAL SHOCK EVENTS"

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The recalculation of RT_{PTS}, as required by 10 CFR §50.61(b)(1), is contained in Table V-1 of the attached revised report which was originally submitted to the NRC by Con Edison on October 12, 1988. The results for RT_{PTS} are still well below the screening criteria at 32 EFPY.