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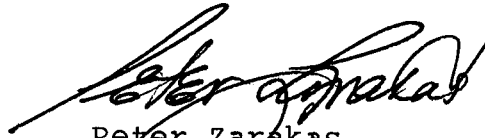
Mr. Richard Snaider
Generic Issues Branch
Nuclear Regulatory Commission
Washington, D. C. 20555

Subject: NUREG-0577, "Potential for Low Fracture
Toughness and Lamellar Tearing on PWR
Steam Generator and Reactor Coolant
Pump Supports" ("For Comment" edition)

Dear Sir,

In November 1979 NRC transmitted the "For Comment" edition of NUREG-0577. Consolidated Edison Company of New York Inc. (Con Edison), as holder of Facility Operating License No. 50-247 for Indian Point Unit No. 2 wishes to provide its comments, contained in attachment A to this letter, concerning the subject document.

Respectfully,


Peter Zarakas
Vice President

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Attachment A

Comments on NUREG-0577-"Potential for Low Fracture Toughness and Lamellar Tearing on PWR Steam Generator and Reactor Coolant Pump Supports"-("For Comment" Edition published October 1979)

In their conclusions, the authors of NUREG-0577 properly recognize that "Many factors (initiating event, low fracture toughness in a critical support member in tension, low operating temperature, large flaw) must be simultaneously present for failure of the support system to ensue."

However, the report by Sandia Laboratories considers only the materials used in component supports at the responding nuclear power plants, and proceeds to classify the plants in groups of levels of susceptibility to brittle fracture without adequate consideration of other factors such as stress level.

The current edition of the ASME Boiler and Pressure Vessel Code Section III paragraph NF2311 exempts from impact testing material for supports when the maximum stress does not exceed 6000 psi or is compressive. Further, Appendix G to that code, "Protection Against Brittle Failure", presents a procedure for obtaining the allowable loadings for ferritic pressure-retaining materials. It is suggested that analyses of support structures be based on these sections of the Codes, which have been generally accepted.