POWER AUTHORITY OF THE STATE OF NEW YORK

10 COLUMBUS CIRCLE NEW YORK, N. Y. 10019

(212) 397-6200

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IPN-80-69

Director of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Mr. Richard P. Snaider
Generic Issues Branch
Division of Safety Technology

Subject: Indian Point 3 Nuclear Power Plant
Docket No. 50-286
Comments on NUREG-0577

Dear Sir:

The Power Authority of the State of New York has the following comments on NUREG-0577:

The work by Sandia Laboratories has resulted in classifying certain plants (including Indian Point Unit 3) according to the potential susceptibility of Steam Generator and Reactor Coolant Pump Supports to low fracture toughness, and therefore, brittle failure.

Results of this classification places 21 PWR's into "Group I" (most susceptible), which will require additional study to ascertain the fracture toughness of several Steam Generator and Reactor Coolant Pump Support materials.

NUREG-0577 further states that "other PWR supports and the supports used in BWR's will be included later in an expanded review" because the staff found no technical basis to exclude such supports. Based on only the material properties survey by Sandia, there appears to be little technical justification to include other PWR supports or BWR supports in this generic issue.

In addition to the above, the staff recognized that there are no requirements in either NRC regulations or the ASME Code for inservice inspection of these supports other than a "visual inspection" once every ten years. However, the staff informs us in NUREG-0577 that a regular inservice inspection, to assure that flaw size remains at an acceptable level, may be required.

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The classification of the plants into groups most susceptible to low fracture toughness, based on a materials property survey addressing only charpy V notch and estimated NDT properties, is not technically justified to precipitate such concern or future work. Although the report is a noteworthy academic exercise in estimating the fracture toughness and nil ductility transition temperatures for a good portion of ASTM structural steels, the work does not adequately address stress levels, the effects of section thickness on fracture toughness of a material, or the ability of a material to function in its intended use at its nil ductility transition temperature, depending on the type of stress or stress level encountered.

The current edition of the ASME Boiler and Pressure Vessel Code Section III, paragraph NF-2311, exempts support material from impact testing when the maximum stress does not exceed 6,000 psi, or when the stress is compressive.

Further, Appendix G, Protection Against Nonductile 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 ASME Codes, which have been generally accepted. Such an approach may show that there is no need for "Group I" classification or extended work in nondestructive examinations to determine existing flaw sizes in supports.

In the Indian Point Unit 3 licensing review of supports, emphasis was placed on the requirements of 10CFR50, Appendix - G - Fracture Toughness and these analyses are equivalent to the requirements of Subsection NF of ASME Section III. These analyses have, as previously mentioned, been accepted by both the industry and the NRC.

Very truly yours,

J. P. Bayne
Senior Vice President
Nuclear Generation