123 Main Street
White Plains, New Y[illegible]0601
914 681.6240

NewYorkPower Authority

John C. Brons
Executive Vice President
Nuclear Generation

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JPN-89-069
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U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Station P1-137
Washington, D.C. 20555

SUBJECT: James A. FitzPatrick Nuclear Power Plant
Docket No. 50-333
Indian Point 3 Nuclear Power Plant
Docket No. 50-286
Individual Plant Evaluations

REFERENCES:
1. NRC Generic Letter 88-20 "Individual Plant Examination for Severe Accident Vulnerabilities," dated November 23, 1988.
2. NYPA Letter, J.C. Brons to the NRC, dated September 28, 1987 (JPN-87-051/IPN-87-045), as amended January 12, 1988 (JPN/IPN-88-001), regarding comments on Draft NUREG-1150.

Dear Sir:

In Reference 1, the NRC required all utilities to perform a systematic examination of their nuclear power plants to identify plant-specific features which may constitute a vulnerability to severe accidents. The Authority has evaluated several examination methodologies and has chosen to perform a Level II Probabilistic Risk Assessment (PRA) for the FitzPatrick Nuclear Power Plant and to update and revise the existing full scope Level III PRA for the Indian Point 3 Nuclear Power Plant.

The Authority has established an in-house Systems Analysis group to perform these evaluations with limited support from outside consultants. The current members of this group have expertise in probabilistic risk assessment, reliability, nuclear engineering, and analytical reactor transient and accident analysis. Available to this group on an as needed basis are an operations and maintenance engineer familiar with the original Zion/Indian Point PRA study, a licensing engineer with a Shift Technical Advisor background, and Senior Licensed Operator licensed site personnel.

The Authority's experience with Nuclear Power Plant Risk Assessment began with the pioneering Indian Point Probabilistic Safety Study (IPPSS) conducted in the early 1980's in conjuction with Consolidated Edison's Indian Point 2 and Commonwealth Edison's Zion plants. This study was performed to support an adjudicatory hearing. That study, reviewed by the NRC, concluded that operation of the Authority's Indian Point 3 nuclear

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plant posed no undue risks to the general public. The NRC often uses this study in its severe accident research.

Since that time, the Authority has led the industry in the development of severe accident computer codes and models. Based upon the NRC's Source Term Code Package (STCP), the Authority has sponsored the development of the much improved STCP-RMA code suite in conjunction with Risk Management Associates. This code, run in-house on a desktop parallel processing microcomputer, is being used by the Authority to develop success criteria, verify phenomenological assumptions, and study accident sequence development.

The Authority also used this code to evaluate the Peach Bottom plant as part of the Authority's detailed comments on the draft NUREG-1150 (Reference 2). Besides indicating that several accident sequences do not necessarily lead to "core melt" as stated in NUREG-1150, subtle plant differences (such as the exact model of SRVs) can make significant differences in the response of supposedly "similar" plants.

J. A. FITZPATRICK PRA STUDY

The FitzPatrick PRA will use the fault tree/event tree methodology as described in NUREG/CR-4550. Certain enhancements to that methodology are being included to take advantage of state-of-the-art knowledge and methods. The system fault trees and event trees are constructed using SAIC's CAFTA Workstation computer code run on desk top microcomputers. System and sequence quantification is to be performed in-house using the SETS code.

This effort is well under way and is expected to follow this schedule of significant milestones:

TASK	COMPLETION DATE
Level I PRA Complete	3/30/90
Level II PRA Complete	8/31/90
Review of Draft Report Complete	10/31/90
Submittal to the NRC	12/30/90

Although the study is not yet complete, the Authority has applied several insights gained during its development to effect and/or verify cost beneficial changes to the plant to include:

- Not modifying the Recirculation Pump Trip (RPT) coil configuration as requested by the NRC;
- Raising the Reactor Core Isolation Cooling (RCIC) turbine exhaust pressure trip point to accommodate operation at elevated containment pressure;
- Confirming and corroborating the safety implications of emergency operating procedures revised in conjunction with the Rev. 4 Emergency Procedures Guidelines (EPGs). Specific procedural changes include operation of the High Pressure Coolant Injection (HPCI) system such that the suction source of water

to the pump remains the Condensate Storage Tanks (CST) as long as possible to avoid potentially damaging high water temperature from the suppression pool source; and

- Supporting the development of station blackout (SBO) procedures.

INDIAN POINT 3 PRA STUDY

The low risk associated with the Authority's Indian Point 3 Nuclear Power Plant is well documented in the existing IPPSS study. To take advantage of advances in the state-of-the-art methodology since this study was originally completed and accomodate future application of the PRA, the Authority will revise the Indian Point 3 Level I and II portion of the study. This involves replacing the existing methodology with the more conventional large fault tree/event tree methodology as used in the FitzPatrick PRA. The study will also be updated to reflect the current plant design and operation in place since the original study, and will rely upon a larger body of generic and plant specific component reliability data.

The Systems Analysis group will begin the Indian Point 3 reevaluation in mid 1990 as resources become available from the FitzPatrick study. The full complement of Systems Analysis resources will be available upon completion of the FitzPatrick study. This schedule does not delay the identification of potential vulnerabilities, since any significant vulnerability would have already been identified in the existing study. Modifications installed since the time of the original study were designed to increase plant safety and are not expected to result in the identification of new vulnerabilities. The revision and update of the Indian Point 3 PRA is expected to follow this schedule of significant milestones:

TASK	COMPLETION DATE
Level I PRA Complete	2/28/92
Level II PRA Complete	5/29/92
Review of Draft Report Complete	7/31/92
Submittal to the NRC	9/1/92

Should you or your staff have any questions regarding this matter, please contact Ms. S.M. Toth or Mr. P. Kokolakis of my staff.

Very truly yours,

for John C. Brons
Executive Vice President
Nuclear Generation

cc: See next page

cc:

U.S. Nuclear Regulatory Commission
Region I
475 Allendale Road
King of Prussia, PA 19406

Office of the Resident Inspector
U.S. Nuclear Regulatory Commission
P.O. Box 136
Lycoming, NY 13093

Mr. David E. LaBarge
Project Directorate I-1
Division of Reactor Projects - I/II
U.S. Nuclear Regulatory Commission
Mail Stop 14 B2
Washington, D.C. 20555

Resident Inspector's Office
Indian Point Unit 3
U.S. Nuclear Regulatory Commission
P.O. Box 337
Buchanan, NY 10511

Mr. Joseph D. Neighbors
Project Directorate I-1
Division of Reactor Projects - I/II
U.S. Nuclear Regulatory Commission
Mail Stop 14 B2
Washington, D.C. 20555