ENCLOSURE 1

NOTICE OF VIOLATION

New York Power Authority
James A. FitzPatrick Nuclear Power Plant

Docket No. 50-333
License No. DPR-59

During an NRC inspection conducted from September 29 to November 16, 1996, seven violations of NRC requirements was identified. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," (60 FR 34381; June 30, 1995) the violations are listed below.

A. Technical Specification 6.8.(A) requires that written procedures and administrative policies shall be established, implemented and maintained that meet or exceed the requirements and recommendations of Section 5 of American National Standards Institute (ANSI) 18.7-1972 "Facility Administrative Policies and Procedures." Section 5.1.2 of ANSI 18.7-1972 states in part, that procedures shall be followed, and the requirements for use of procedures shall be prescribed in writing. MP-004.03, CRD Removal and Replacement, describes removal and replacement of control rod drives (CRDs). The procedure requires that all CRDs to be removed are accurately located and readily identified (marked) prior to removal.

Contrary to the above, on November 11, 1996, during the work preparation phase, CRDs to be removed were not accurately located prior to removal which resulted in the incorrect removal of three CRDs.

This is a Severity Level IV Violation (Supplement 1).

B. 10 CFR Part 50, Appendix B, Criterion III, "Design Control," requires that the design basis shall be correctly translated into specifications, drawings, procedures, and instructions; that the adequacy of the design be verified; and that design changes be subject to design control measures.

Contrary to the above, on and before October 25, 1996, the design basis was not correctly translated into procedures, the adequacy of design was not verified, and design changes were not subjected to design control measures, as evidenced by the following examples:

(1) Unverified engineering judgement regarding design calculations JAF-CALC-ELEC-00426 and JAF-CALC-ELEC-00427 erroneously equated spare battery capacity with voltage at safety-related components.

(2) Electrical load added to safety-related station battery calculations JAF-CALC-ELEC-01417 and JAF-CALC-ELEC-01418 was not consistent with the load specified in Modification F1-89-158.

(3) Unverified engineering judgement concerning calculation JAF-CALC-HPCI-00840 was used to conclude that air injection would not affect high pressure coolant injection pump operability.

(4) An incorrect assumption was made in calculation JAF-CALC-DHR-03445 that a non-safety-related component could be assumed as a limiting single failure.

(5) An unverified assumption was made in calculation JAF-CAL-DHR-02380 that a plate heat exchanger could be modeled as a shell and tube heat exchanger.

This is a Severity Level IV violation (Supplement I).

C. 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," requires written test procedures which incorporate acceptance limits contained in applicable design documents.

Contrary to the above, on and before October 25, 1996, surveillance procedures did not incorporate the acceptance limits contained in applicable design documents, as exemplified by the following:

(1) Instructions in Work Request 94-02935-00, dated April 21, 1994, did not contain sufficient guidance to ensure that volt meters of appropriate accuracy and precision were used to measure voltage between the reactor protection system electrical protection assemblies and their respective power panels.

(2) Acceptance criteria in surveillance procedure ST-2X, "RHR Service Water Flow Rate," did not incorporate acceptance limits contained in applicable design documents by not accounting for instrument error associated with measuring required flow to the residual heat removal system heat exchanges.

(3) Acceptance criteria in station battery service test procedures MST-071.24, Revision 2 and MST-071.26, Revision 0, "Modified Station Battery Performance/Service Test," did not reflect the minimum battery terminal voltages for acceptable operation of safety-related equipment specified in design calculations JAF-CALC-ELEC-00426, Revision 1, dated October 16, 1992 and JAF-CALC-ELEC-00427, Revision 0, dated June 4, 1992, respectively.

This is a Severity Level IV violation (Supplement I).

D. 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requires conditions adverse to quality such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and non-conformance be promptly identified and corrected.

Contrary to the above, on and before October 25, 1996, conditions adverse to quality were not promptly identified and corrected, in that:

(1) Approximately 54 deviations and deficiencies associated with the Final Safety Analysis Report, calculations, and procedures pertaining to the residual heat removal system, identified in the design basis documentation

verification program (DBD-10) in 1994 were not evaluated for corrective action.

(2) Appropriate corrective action related to the calibration frequency was not taken to correct recurring APRM flow bias flow transmitter calibration failures.

This is a Severity Level IV violation (Supplement I).

E. 10 CFR 50.59, "Changes, tests, and experiments," permits licensees to make changes to the facility, as described in the Safety Analysis Report, without prior Commission approval, provided that the proposed changes do not involve a change in the technical specifications or involve an unreviewed safety question. Records of these changes must include a written safety evaluation which provides the bases for the determination that the change does not involve an unreviewed safety question.

Section 4.10.3 of the FitzPatrick Final Safety Analysis Report (FSAR) describes the utilization of the equipment area temperature monitoring system to detect reactor coolant pressure boundary leakage outside of the primary containment. Area temperature sensors are calibrated with the station in operation with normal ventilation patterns and ambient temperature levels to detect a seven gallon per minute leak. The residual heat removal (RHR) system equipment area temperature detector is shown in FSAR Table 4.10-1, "Summary Of Isolation/Alarm of System Monitored And The Leak Detection Methods Used."

Contrary to the above, on or about August 12, 1996, a temporary high efficiency particulate air (HEPA) filter and blower, which could have altered the accuracy of the area temperature monitoring system, were installed in the "A" RHR heat exchanger room, and no safety evaluation was performed and documented to provide the bases for the determination that the change did not involve an unreviewed safety question.

This is a Severity Level IV violation (Supplement I).

F. Technical Specification 6.11 requires, in part, that each entry into a posted locked high radiation area shall be under the control of a radiation work permit (RWP) and that a radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received be utilized. RWP 96-0411 issued for work in the drywell requires that each worker wear an alarming dosimeter.

Contrary to the above, on October 30, 1996, a contractor entered and worked in the drywell under RWP 96-0411, a posted locked high radiation area, for three hours with his alarming dosimeter turned off.

This is a Severity Level IV violation (Supplement IV).

G. Title 10, Code of Federal Regulations, Part 71.12 (10 CFR 71.12) states, in part, that shippers of licensed materials are generally licensed for shipment utilizing

packages for which the Commission has issued a certificate of compliance, provided that the licensee have in place a quality assurance program meeting the requirements contained in sections 71.101 through 71.137. 10 CFR 71.133 requires, in part, that the licensee establish means to assure that conditions adverse to quality, such as deviations, are promptly identified and corrected.

Contrary to the above, on October 22, 1996, the licensee shipped licensed radioactive material in an NRC-approved package (Certificate of Compliance No. USA/9094/A) without promptly implementing corrective actions for a prior violation of an applicable shipping procedure (involving the lack of current certification of technicians relative to the applicable computer code used to classify the shipment) as previously identified by the licensee's deviation event report (DER 96-1188).

This is a severity level IV violation (Supplement V).

Pursuant to the provisions of 10 CFR 2.201, New York Power Authority is hereby required to submit a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555 with a copy to the Regional Administrator, Region I, and a copy to the NRC Resident Inspector at the facility that is the subject of this Notice, within 30 days of receipt of the letter transmitting this Notice of Violation (Notice). This reply should be clearly marked as a "Reply to a Notice of Violation" and should include for each violation: (1) the reason for the violation, or, if contested, the basis for disputing the violation, (2) the corrective steps that have been taken and the results achieved, (3) the corrective steps that will be taken to avoid further violations, and (4) the date when full compliance will be achieved. Where good cause is shown, consideration will be given to extending the response time.

Because your response will be placed in the NRC Public Document Room (PDR), to the extent possible, it should not include any personal privacy, proprietary, or safeguards information so that it can be placed in the PDR without reduction. However, if you find it necessary to include such information, you should clearly indicate the specific information that you desire not to be placed in the PDR, and provide the legal basis to support your request for withholding the information from the public.

Dated at King of Prussia, PA
this 13th day of December, 1996.