August 2, 2000

Mr. James Knubel
Chief Nuclear Officer
Power Authority of the State of
New York
123 Main Street
White Plains, NY 10601

SUBJECT: JAMES A. FITZPATRICK NUCLEAR POWER PLANT- REVIEW OF LICENSEE RESPONSES TO GENERIC LETTER (GL) 96-06, "ASSURANCE OF EQUIPMENT OPERABILITY AND CONTANMENT INTEGRITY DURING DESIGN-BASIS ACCIDENT CONDITIONS" (TAC NO. M96812)

Dear Mr. Knubel:

By letter dated January 27, 1997, as supplemented by letters dated May 27, 1997, July 30, 1998, January 25, 1999, and October 26, 1999, you responded to GL 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions." Your submittals address the issues of water hammer, two-phase flow and thermally-induced pressurization of piping penetrating containment for the FitzPatrick Nuclear Power Plant. The NRC staff review was done in two parts - (1) Water Hammer and Two-phase Flow Issues and (2) Thermally-induced Pressurization of Piping Penetrating Containment.

1. <u>Water Hammer and Two-phase Flow Issues</u>

Your assessment of the water hammer and two-phase flow issues for the FitzPatrick plant was provided in a letter dated May 27, 1997, and additional information was submitted in a letter dated July 30, 1998. Based on the NRC staff's review of the information that was provided, it is our understanding that: (a) the drywell cooling units are not required for accident mitigation, and (b) procedure revisions have been made to prevent restoration of cooling water flow to the containment drywell coolers during a loss-of-coolant accident (LOCA) concurrent with a loss-of-offsite power (LOOP), thereby eliminating the potential for waterhammer or two-phase flow conditions. We also understand that during a LOCA event where cooling water flow to the drywell coolers is not lost, system design and operating characteristics prevent the formation of steam such that waterhammer and two-phase flow conditions are not possible. We are satisfied with your response and consider the waterhammer and two-phase flow elements of GL 96-06 to be closed.

2. <u>Thermally-Induced Pressurization of Piping Penetrating Containment</u>

In your submittals of January 27, 1997, May 27, 1997, July 30, 1998, and January 25, 1999, you identified 23 pipe segments penetrating containment at FitzPatrick, susceptible to thermally-induced pressurization which were evaluated for operability. Two penetrations (X-14 and X-41) were eliminated from further evaluation because the water temperature would be the same or higher than the containment temperature under design-basis accident conditions, and another penetration (X-12) was eliminated due to the valve by-pass design feature. The remaining 20 pipe segments are potentially susceptible to thermal pressurization and are associated with penetrations for the Reactor Building Closed Loop Cooling (RBCLC) lines, Main Steam Drain line, Drywell Equipment Drain Sump Discharge line, Residual Heat Removal (RHR) to Suppression Pool lines, RHR to Spray Header lines, RHR to Containment Spray lines, Reactor Core Isolation Cooling Pump Suction line, and High Pressure Coolant Injection (HPCI) Pump Suction line.

In your July 30, 1998, submittal, it was stated that 11 of the remaining 20 pipe segment penetrations (X-8, X-18, X-19, X-39A/B, X-210A/B, X-211A/B, X-224, and X-226) are evaluated for the thermal pressure conditions which would occur with the consideration that an isolation valve would lift off its seat to limit the pressure. The calculated valve lifting pressures are compared to the maximum allowable internal pressures to determine if the pipe segment's integrity is maintained. The maximum allowable internal pressures are developed in accordance with ANSI B31.1 Code using criteria allowed by Appendix F of American Society of Mechanical Engineers Boiler and Pressure Vessel Code Section III. Based on the results of the evaluation, you determined that for all 11 penetrations, the valve lifting or leak pressures were lower than the maximum allowable pressures. However, for penetration X-18, the single active failure criterion of the affected containment isolation valves was not addressed. In a subsequent letter dated October 26, 1999, it was clarified that due to consideration of single active failure criterion, a modification such as a rupture disk was to be installed during the next refueling outage to relieve the pressure. The staff finds the committed modification acceptable for penetration X-18.

The nine penetrations (X-23, X-24, X-62, X-63, X-64, X-65, X-66, X-67 and X-68) associated with RBCLC employ normally open air operated globe valves. In the same July 30, 1998, submittal, it was stated that there are two factors that prevented thermal pressurization of these piping segments. The first is the pressure relieving capability of the isolation valves and the second is the condition of high fluid temperatures in the lines at the time the valves would be isolated. Although the staff has expressed concerns of uncertainty of high fluid temperatures following the LOCA, you state that the calculations performed by the valve actuator manufacturer have shown that the fluid pressure will lift the air operated globe valves and prevent damage to the RBCLC piping due to thermally-induced pressurization.

The staff finds that installation of a rupture disc for penetration X-18 is an acceptable means for relieving pressure of a solid water volume. The staff also finds that using Appendix F criteria and calculations of valve lifting pressure is an acceptable means to demonstrate the operability of affected piping segments. Therefore, the staff concludes that your corrective actions, proposed plant modifications, the use of Appendix F criteria and calculations of valve lifting pressures provide an acceptable resolution for the issue of thermally-induced pressurization of piping runs penetrating the containment, and the closed piping segments inside the containment. This completes the staff's review of GL 96-02 issues and thus this TAC is closed.

Sincerely,

/RA/

Guy S. Vissing, Senior Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-333

cc: See next page

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Sincerely,

/RA/

Guy S. Vissing, Senior Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

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* SE input dated 6/5/00 was provided and no major changes were made.

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