James A. FitzPatrick
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New York Power Authority

Harry P. Salmon, Jr.
Site Executive Officer

February 23, 1996
JAFP-96-0080

United States Nuclear Regulatory Commission
Document Control Desk
Mail Station P1-137
Washington, D.C. 20555

SUBJECT: James A. FitzPatrick Nuclear Power Plant
Docket No. 50-333
Annual Report
Safety Relief Valve Challenges and Failures

Dear Sir:

FitzPatrick Plant Technical Specification 6.9.A.2.b requires that challenges to, and failures of, Safety Relief Valves (SRVs) be reported annually.

During calendar year, 1995, no challenges to Safety Relief Valves (SRVs) from automatic control circuits or due to pressure transients occurred.

During calendar year, 1995, a total of 30 operator demand challenges occurred to satisfy required Technical Specification surveillance or post-maintenance operability demonstrations. On March 25, 1995 one failure of an SRV to respond to an operator demand occurred. Safety Relief Valve 02RV-71E failed to open from the remote shutdown panel during testing. Investigation revealed a high electrical resistance at the valve solenoid coil to field lead splice. A new valve solenoid (which is provided by the vendor with field leads) was installed and tested with satisfactory results. It should be noted that the SRV design includes two solenoids. One solenoid is used for remote-manual valve actuation from the remote shutdown panel while the other solenoid is used for remote-manual valve actuation from the main Control Room or Automatic Depressurization System logic circuits. The function of the second solenoid was not affected.

During the 1994/1995 refuel outage, all 11 SRV pilot valve assemblies were removed in December 1994 for replacement with refurbished and recertified pilot valve assemblies. On January 6 and March 15, 1995, the test laboratory performing the tests on the pilot valve assemblies informed the New York Power Authority that 10 of 11 pilot valves had not lifted within 1% of the setpoint. Licensee Event Report (LER) 95-006 provided a discussion of the safety significance and effects of the setpoint drift of the 10 SRVs. The discussion in LER-95-006 also noted that analyses had shown that the setpoint drift would not have had an adverse effect on Emergency Core Cooling System performance and that the American Society of Mechanical Engineers (ASME) reactor vessel peak allowable pressure limit would not have been violated.

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If you have any questions regarding this report, please contact Mr. Arthur Zaremba at (315) 349-6365.

Very truly yours,

Harry P. Salmon, Jr.
Site Executive Officer

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