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New York Power Authority

John C. Brons
Executive Vice President
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October 21, 1988
JPN-88-055

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

SUBJECT: James A. FitzPatrick Nuclear Power Plant
Docket No. 50-333
Response to NRC Bulletin 88-08 - Thermal Stresses in Piping Connected to Reactor Coolant Systems

References:
1. NRC Bulletin 88-08, dated June 22, 1988, concerning the same subject.
2. NRC Bulletin 88-08, Supplement 1, dated June 24, 1988, concerning the same subject.
3. NRC Bulletin 88-08, Supplement 2, dated August 4, 1988, concerning the same subject.

Dear Sir:

NRC Bulletin 88-08 (Reference 1) described an event at a nuclear power plant which experienced thermal fatigue cracking of unisolable piping connected to the Reactor Coolant System (RCS). The bulletin requested licensees to determine whether unisolable sections of piping connected to the RCS could be subjected to stresses from temperature stratification or temperature oscillations. Such stresses could be induced by leaking valves and may not have been evaluated in the original design analysis of the piping.

Attachment 1 provides a description of the review performed at the FitzPatrick plant. This review concludes that the condition described in Bulletin 88-08 is highly unlikely to occur in the FitzPatrick plant.

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Should you or your staff have any questions regarding this matter, please contact Mr. J. A. Gray, Jr. of my staff.

Very truly yours,

John C. Brons

John C. Brons
Executive Vice President
Nuclear Generation

State of New York
County of Westchester

Subscribed and Sworn to Before
me this 21st day of Oct. , 1988

Mina Holden
Notary Public

MINA HOLDEN
NOTARY PUBLIC, State of New York
Westchester County
No. 4829150
My Commission Expires Aug. 31, 1989

Enclosure

cc: U.S. Nuclear Regulatory Commission
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ATTACHMENT I
TO
JPN-88-055

RESPONSE TO BULLETIN 88-08

NEW YORK POWER AUTHORITY
JAMES A. FITZPATRICK NUCLEAR POWER PLANT
DOCKET NO. 50-333
DPR-59

INTRODUCTION

NRC Bulletin 88-08 notified licensees of concerns regarding systems connected to the Reactor Coolant System (RCS). Specifically, the concerns involved thermal fatigue cracking due to stresses from temperature stratification or temperature oscillations. The cause of such cracking is cold fluid leaking past a valve into a high temperature fluid system. The NRC requested licensees to identify affected piping and take action if necessary.

DISCUSSION

Systems connected to the FitzPatrick RCS/Recirculation System and, therefore, subject to the requirements of this bulletin include Residual Heat Removal (RHR)/Low Pressure Core Injection (LPCI), and Core Spray. A review of these systems, as well as the Feedwater System including High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC), was performed.

Additionally, a review of the Reactor Water Cleanup System (RWCU) was performed. The RWCU System takes suction from the Recirculation System and discharges to the Feedwater System. The temperature of the RWCU water at the discharge point is approximately 430°F. This temperature is not significantly different from the temperature of the feedwater, which is approximately 419°F. As a result, the RWCU System is not subject to the requirements of this bulletin.

RHR/LPCI and Core Spray Systems

Both of the RHR/LPCI and Core Spray systems are in the standby mode during normal plant operation except during surveillance testing. During normal operation, the reactor operating pressure is higher than the pressure in either the Core Spray or RHR/LPCI system. Therefore, leakage into the reactor due to valve leakage in these systems is unlikely. The higher reactor pressure prevents cold water from leaking through the RHR/LPCI and Core Spray valves. As a result, there will be no temperature oscillations or temperature stratification that could be induced by leaking valves in either of these systems.

Surveillance tests on both the RHR/LPCI and Core Spray system pumps are performed once per month. The test pressure remains lower than the reactor operating pressure, thus there are no temperature oscillations or temperature stratification that could be induced by leaking valves.

HPCI & RCIC Systems

Both the HPCI and RCIC systems are in the standby mode during normal plant operations except during surveillance tests. As is the case with the RHR/LPCI and Core Spray systems, both the HPCI and RCIC systems are at a lower pressure than reactor operating pressure during normal operations. Due to the differential pressure across the connecting valves, there is no leakage from the low pressure HPCI/RCIC system to the high pressure feedwater system.

In contrast to the RHR/LPCI and Core Spray systems, tests on HPCI and RCIC pumps are performed at a pressure higher than the reactor pressure. As stated in the JAF Operations Surveillance Test Procedure, the HPCI system shall deliver at least 4,250 gpm against a system head corresponding to 100 psi above reactor pressure. Similarly, the RCIC Surveillance Test states that the RCIC system shall deliver at least 400 gpm against a system head corresponding to 100 psi above the reactor pressure. Test water for both pumps is taken from the condensate tank and does not exceed 105°F during the duration of the test. Feedwater temperature is approximately 419°F (design basis). The relatively cold fluid used in the pump testing could lead to thermal fatigue in the feedwater system piping if it were to enter into the high temperature feedwater system and occur over a lengthy period of time.

Pump tests on both HPCI and RCIC occur once per month and generally last less than two hours. During the time that the systems are in operation, probability of leakage across the valves is increased. In a period of one year of reactor operation, the HPCI and RCIC pumps are in operation less than 0.3% of the time. Since the systems are in operation a relatively short period of time there is little concern of leakage across the valve.

Valves 23-MOV-19 (HPCI) and 13-MOV-21 (RCIC) isolate HPCI and RCIC from the feedwater system. These valves would have to leak in order for thermal cycling to occur.

The HPCI inboard injection valve 23-MOV-19 has been local leak rate tested during each of seven refueling outages since initial plant operation. During the tests, this valve is tested in a combined test with other containment isolation valves of penetration X-9B. Although the total leakage from this penetration has exceeded allowable leakage on three occasions, the leak has never been attributed to 23-MOV-19, nor has this valve ever required repair due to excessive leakage.

The RCIC inboard injection valve 13-MOV-21 has been local leak rate tested during each of seven refueling outages since initial plant operation. During the tests, this valve is tested in a combined test with other containment isolation valves of penetration X-9A. Although the total leakage from this penetration has exceeded allowable leakage on three occasions, the leak has never been attributed to 13-MOV-21, nor has this valve ever required repair due to excessive leakage. Also, the pipe welds and the weld heat affected zones are inspected for flaws in accordance with ASME Section XI.

CONCLUSIONS

Based on the information above, the condition described in Bulletin 88-08 is highly unlikely to occur in the FitzPatrick plant. This conclusion has been based upon the following:

1. The short amount of time that the systems of concern are in operation and are pressurized. (Pump tests are performed once a month for less than two hours.)

2. Valves 23-MOV-19 and 13-MOV-21 are tested for leakage as part of the IST program during refueling outages.

3. The pipe welds are examined in accordance with ASME Section XI which would detect flaws of the type described in Bulletin 88-08.

Based upon the above information, no additional analysis is required.