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John C. Brons
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November 1, 1989
JPN-89-071

U. S. Nuclear Regulatory Commission
Mail Station 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, Supplement 3,
Thermal Stresses in Piping Connected to Reactor Coolant Systems

References:

1. NRC Bulletin 88-08, dated June 22, 1988, regarding thermal stresses in piping connected to reactor coolant systems.
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.
4. NYPA letter, J. C. Brons to NRC, dated October 21, 1988, providing response to NRC Bulletin 88-08, (JPN-88-055).
5. NRC Bulletin 88-08, Supplement 3, dated April 11, 1989, concerning the same subject as Bulletin 88-08.

Dear Sir:

NRC Bulletin 88-08 (Reference 1) described an event at a nuclear power plant involving 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. Reference 2 provided preliminary information about a similar event at a foreign plant and emphasized the need for sufficient examinations. Reference 3 emphasized the need for enhanced ultrasonic testing and for experienced examination personnel to detect cracks in stainless steel piping. In Reference 4, the Authority described the review performed at the FitzPatrick plant. This review concluded that the condition described in Bulletin 88-08 is highly unlikely to occur in the FitzPatrick plant.

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Reference 5 provided information about a similar event at another foreign reactor and emphasized the need for sufficient review and the importance of taking action where necessary. Since the event described in Reference 5 differs somewhat from the events described previously, the Authority has readdressed the systems affected. The review is discussed in detail in Attachment 1. The Authority has concluded that there is negligible risk of pipe failure which would lead to an unisolable leak of primary coolant, and that no additional action is required.

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

Very truly yours,

for John C. Brons
Executive Vice President
Nuclear Generation

Attachment

STATE OF NEW YORK
COUNTY OF WESTCHESTER

Subscribed and sworn to before me
this 1st day of November 1989.

Notary Public
ANNE HOLDEN
NOTARY PUBLIC, State of New York
Westchester County
No. 4829150
My Commission Expires Aug. 31, 1991

cc: U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

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ATTACHMENT I
TO
JPN-89-071

Response to NRC Bulletin 88-08 Supplement 3 - Thermal Stresses in Piping Connected to Reactor Coolant Systems

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

ATTACHMENT I TO JPN-89-071

Response to NRC Bulletin 88-08 Supplement 3, Thermal Stresses in Piping Connected to Reactor Coolant Systems

I. INTRODUCTION

NRC Bulletin 88-08, Supplement 3 notified licensees of concerns regarding systems connected to the Reactor Coolant System (RCS). Specifically, these concerns involved thermal fatigue cracking caused by stresses resulting from temperature stratification or temperature oscillations. Such cracking may occur when fluid from a hot system, such as feedwater (FW) or recirculation, cools in a stagnant leg. This subcooled fluid unseats the normally shut isolation valve causing leakage to occur. Hot makeup fluid to this valve causes the valve disk to expand and subsequently stops the leak. This temperature oscillation may lead to fatigue-induced cracking. The NRC requested licensees to identify affected piping and to take action if necessary.

II. DISCUSSION

During normal operation, the recirculation and FW systems contain hot pressurized reactor water. Isolated from these systems are various standby systems which have stagnant legs similar to the failed pipe configuration cited in Supplement 3. Systems that interface with the hot pressurized systems are High Pressure Core Injection (HPCI), Reactor Core Isolation Cooling (RCIC), Residual Heat Removal (RHR) in the Low Pressure Coolant Injection (LPCI) mode, Reactor Water Cleanup (RWCU), and Core Spray (CS). The Authority evaluated each of these systems to determine their susceptibility to thermal stratification fatigue-induced cracking as described in Supplement 3. The following summarizes the results of this evaluation.

III. CS/HPCI/RCIC/RHR (LPCI)

During normal plant operation, these four systems are in the standby mode. They each have a generic valve arrangement and a unique piping configuration. The generic valve configuration is shown schematically below (Figure 1). (The second isolation valve shown in the figure is not considered in the scope of this evaluation.)

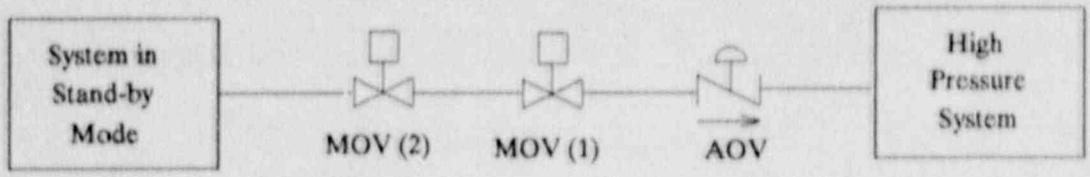

FIGURE 1

In all cases the motor operated valves (MOVs) are solid disk wedge-type gate valves. It is this type of valve that caused the thermal stratification fatigue-induced cracking described in Supplement 3. The air-operated check valve upstream of the MOV offers the piping between these valves some protection from pipe fatigue failure. Additionally, the RHR (LPCI), HPCI, and RCIC systems have long pipe runs (>25 feet) between the flow of hot fluid and the first motor-operated valve. This eliminates the effects of thermal stratification, thus reducing the probability of pipe fatigue cracking.

The piping susceptible to failure is between the air-operated valve and the MOVs. A failure in this pipe run, however, would not cause an unisolable leak from the high pressure system.

In order for thermal cycling to occur as presented in Supplement 3, the following MOVs would have to leak. (These valves are designated MOV(1) in Figure 1).

14MOV-12A & 12B (Core Spray)
23MOV-19 (HPCI)
13MOV-21 (RCIC)
10MOV-25A & 25B (RHR/LPCI)

An analysis of the specific valves follows below:

A. CORE SPRAY

Core spray inboard injection valves 14MOV-12A and 14MOV-12B have been local leak rate tested during each of the seven refueling outages since initial plant operation. During the tests (ST-39B), these valves are tested in combination with penetrations X-16A and X-16B. These penetrations have never exceeded allowable leakage, and repairs to these valves have never been required as a result of excessive leakage. The piping welds between these valves and the reactor vessel nozzle are inspected for flaws in accordance with ASME Section XI.

During the 1981 refueling outage, the Authority replaced the "A" loop piping between the reactor vessel nozzle safe-end and 14AOV-13A.

B. HPCI

The HPCI inboard injection valve, 23MOV-19, has also been local leak-rate tested during each of seven refueling outages since initial plant operation. During the test (ST-39B), this valve is tested in combination with other containment isolation valves of penetration X-9B. Although the total leakage from this penetration has exceeded allowable leakage on three occasions, the leakage has never been attributed to 23MOV-19, nor has this valve ever required repair due to excessive leakage. In addition the piping welds and heat-affected zone downstream from 23MOV-19 are inspected for flaw indications in accordance with ASME, Section XI.

C. RCIC

The Authority has local leak-rate tested RCIC inboard injection valve, 13MOV-21, during each refueling outage. During the test (ST-39B), this valve is tested in combination 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 13MOV-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.

D. RHR/LPCI

The RHR/LPCI inboard injection valves, 10MOV-25A and 10MOV-25B, have been leak-rate tested during each refueling outage. During the test (ST-33B), these valves are tested in combination with penetrations 13A and 13B, respectively. Although the total leakage from this penetration has exceeded allowable leakage on three occasions, the leak has never been attributed to 10MOV-25A or 10MOV-25B. Valve 10MOV-25A was repacked in 1985, but this was not due to excessive leakage. Valve 10MOV-25B had a live load stem packing installed during the 1988 refueling outage which is designed to prevent stem leakage during operation. Additionally, the pipe welds and heat-affected zones are inspected for flaw indications in accordance with ASME, Section XI.

E. RWCU

The RWCU system currently has the B pump in operation and the A pump isolated. The pipe from the header tee in the cleanup pump room to the manually operated isolation valves was evaluated for fatigue cracking. Leakage through the A pump isolation valves, 12RWC-19 and 12RWC-73 stem packing would be detected readily on the floor. Small leakage through two manually operated gate valves, caused by heat-up and expansion, is unlikely. Manually operated gate valves do not have the problem of preset torque valves or preset stem engagement which may cause a valve disk to expand and lift, causing leakage, when heated up as discussed in Supplement 3. Additionally, the 4" pipe line is less susceptible to fluid stratification due to the small pipe diameter and the vertical orientation of the valves.

No leak-rate data is available on these valves, however, live load stem packing has been installed on 12RWC-73 to prevent valve stem leakage.

F. RHR/ SHUTDOWN COOLING

The shutdown cooling mode, including the reactor vessel head spray, is a function of the RHR system and operates during normal cooldown and shutdown.

During reactor cooldown, coolant is pumped from the B recirculation loop, cooled in the RHR heat exchangers, and pumped back into the recirculation loops. Valve 10MOV-18 on the suction line is a wedge-type gate valve located close to the recirculation line 02-WH-GE-1A. Because of poor valve orientation (stem pointing down), 10MOV-18 experienced many maintenance problems before 1985. During the 1985 refueling outage, this valve was replaced along with the piping between the drywell penetration and 10RHR-88. The balance of pipe between the high-energy recirculation pipe and valve 10RHR-88 was subsequently replaced during the 1987 refueling outage.

Live load stem packing was installed on 10MOV-18 during the 1988 refueling outage, which is designed to prevent stem leakage. The inboard suction valve 10MOV-18 was leak-rate tested during the 1986 refueling outage. At that time, this valve was tested in combination with penetration X-12. The penetration did not exceed allowables, and no repairs were required due to excessive leakage.

For the RHR head spray sub-system, the Authority evaluated valves 10MOV-32 and 10RHR-29. The valve design configuration is similar to that shown in Figure 1, but without an air operator on the first isolation check valve. The isolation check valve protects against steam coming from the reactor head. This prevents fluid stratification. Leakage past the first isolation check valve, would initially flash and then condense through the pipe run to the downstream MOV. The 4 inch pipe size and extended run of pipe will eliminate the possibility of stratification-induced thermal fatigue cracking.

IV. CONCLUSION

The Authority has evaluated FitzPatrick plant piping systems with the potential for thermal stratification fatigue-induced cracking as described in Supplement 3. It concluded that due to generic valve design, piping geometry, penetration leak-rate data, and current inspection programs, no additional action is required. There is little risk of pipe failure which would lead to an unisolable lead of primary coolant. The most susceptible location was identified as the piping upstream of 10MOV-18. However, the Authority replaced this piping in its entirety in 1985 and 1987.

As a conservative maintenance effort, the Authority is considering installing live load stem packing on valves to preclude the already low probability of cracking in piping upstream of the valves, if oscillating leakage were to occur. Those valves are: 14MOV-12A, 13MOV-21, 10MOV-25A, 12RWC-19, 10MOV-32, 10RHR-29.