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April 18, 1990
JAFP-90-0339

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

SUBJECT: DOCKET NO. 50-333
LICENSEE EVENT REPORT: 90-010-00
HPCI Control Oscillations

Dear Sir:

This Licensee Event Report is submitted in accordance with 10 CFR 50.73(a)(2)(v).

Questions concerning this report may be addressed to Mr. Hamilton Fish at (315) 349-6013.

Very truly yours,

WILLIAM FERNANDEZ

WF:HCF:lar

Enclosure

cc: USNRC, Region I
USNRC Resident Inspector
INPO Records Center
American Nuclear Insurers

Cert No P148 799849

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NRC Form 366 (9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/85

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1): JAMES A. FITZPATRICK NUCLEAR POWER PLANT

DOCKET NUMBER (2): 0 5 0 0 0 3 3 3

PAGE (3): 1 OF 0 5

TITLE (4): High Pressure Coolant Injection System Control Oscillations Caused by Lack of Damping Due to Not Readjusting Needle Valve and Test Procedure Inadequacy

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAMES	DOCKET NUMBER(S)
0 3	1 9	9 0	9 0	— 0 1 0 —	0 0	0 4	1 8	9 0		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): N

POWER LEVEL (10): 1 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)

20.402(b)	20.405(c)	50.73(a)(2)(iv)	73.71(b)
20.405(a)(1)(i)	50.36(c)(1)	X 50.73(a)(2)(v)	73.71(c)
20.405(a)(1)(ii)	50.36(c)(2)	50.73(a)(2)(vii)	OTHER (Specify in Abstract below and in Text, NRC Form 366A)
20.405(a)(1)(iii)	50.73(a)(2)(i)	50.73(a)(2)(viii)(A)	
20.405(a)(1)(iv)	50.73(a)(2)(ii)	50.73(a)(2)(viii)(B)	
20.405(a)(1)(v)	50.73(a)(2)(iii)	50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME: Hamilton C. Fish

TELEPHONE NUMBER: AREA CODE 315 349-6013

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)

YES (If yes, complete EXPECTED SUBMISSION DATE) — X NO

EXPECTED SUBMISSION DATE (15): MONTH | DAY | YEAR

ABSTRACT (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines) (16)

EIIS Codes are in []

During a reactor scram (LER-90-009) from full power at 1554 on 3/19/90 the High Pressure Coolant Injection (HPCI) [BJ] correctly responded to low reactor water level by automatic start and injection into the reactor vessel. Unstable operation as indicated by flow rate oscillations prompted the operator to take manual control to restore reactor water level. Four days of testing did not reproduce oscillations at 150 psig reactor pressure. Oscillations were replicated at normal operating pressure of 950 psig. HPCI had been extensively tested and the control circuits adjusted in 12/89 (LER-89-025). During 2/89 the HPCI hydraulic servo was cleaned (LER-90-005) resulting in significant improvement in response times compared to 12/89. While not noticed at that time, improved response time reduced the control system damping and the ability of the system to respond to significant flow signal fluctuations. Existing post-work test procedures failed to reveal this condition. Therefore, the need to increase damping by readjusting a hydraulic speed control loop needle valve was not identified and resulted in unstable operation of HPCI in the automatic mode. The needle valve was properly adjusted. Post-work test procedures will be revised. A full flow injection test to the vessel will be performed during start-up from the next refueling outage. Related LERs: 90-009, 90-005, 89-025, 89-018, and 89-002.

NRC Form 366 (9-83)

EIIS Codes are in []

Description

The reactor automatically scrammed (LER-90-009) from full power at 3:54 P.M. on March 19, 1990 when a false low reactor vessel water level signal resulted in a rapid rise of reactor water to 238 inches above Top of Active Fuel (TAF). In accordance with design, the turbine driven feed pumps tripped at a level of 222.5 inches to protect the turbines from damage. The trip of the feed pump turbines resulted in a rapid decrease in water level to 131 inches TAF and automatic initiation of the High Pressure Coolant Injection system (HPCI) [BJ] to inject make-up water to the reactor vessel. Upon receipt of the start signal, the HPCI system exhibited rapid oscillations in flow. The operator then took manual control of HPCI to assure a controlled restoration of reactor water level. Subsequently, HPCI tripped on high reactor water level in accordance with design. A turbine driven reactor feedwater pump was restarted to maintain vessel level. A normal plant cooldown was initiated.

HPCI was declared to be inoperable at 9:00 P.M. because of the observed oscillations during vessel injection. One residual heat removal (RHR) [BO] pump C was out of service for maintenance prior to the scram.

While the plant was shut down, troubleshooting and standard in-service response testing of the HPCI control systems were performed. No problems were identified. A minimum reactor steam pressure of 145 to 150 psig is required to operate the HPCI turbine for operational testing. RHR pump C was restored to service to permit reactor start up. Reactor start-up was commenced at 5:10 A.M. on March 21, 1990. Criticality was achieved at 7:29 A.M. Reactor pressure of approximately 150 psig was achieved and maintained for the duration of low pressure HPCI testing to demonstrate operability through March 22nd.

Engineering personnel reviewed available computer and strip chart recordings of HPCI performance for evidence of flow oscillation. Preoperational test records of original HPCI injection to the reactor vessel were reviewed. Large oscillations were present during the test. However, subsequent damping was adequate to control the oscillations at that time. They consulted with the vendor and other nuclear plant sites concerning settings of the speed controller. On March 21, using Temporary Operating Procedure (TOP) 107, "Testing of the High Pressure Coolant Injection System", three test runs (at 150 psig reactor pressure) were performed over a six-hour period using manually adjusted step changes in controller settings. The attempts to reproduce instability and flow oscillations similar to those observed during HPCI response to the scram were not successful.

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TEXT (If more space is required, use additional NRC Form 366A's) (17)

On March 22 Surveillance Test ST-4N, "HPCI Flow Rate and Inservice Test (IST)", was performed at 150 psig and HPCI was declared operable at 4:00 P.M. At 4:10 P.M. reactor heat-up was resumed to raise reactor pressure to approximately 950 psig for full pressure testing of HPCI. The reactor mode switch was placed in the RUN position at 9:05 P.M. The "HPCI System Inoperable Test" (ST-4G) was performed and at 9:50 P.M. HPCI was declared inoperable to connect additional temporary flow recording instrumentation and perform TOP-107.

At 12:45 A.M. on March 23, HPCI was placed in service for testing using TOP-107. Step changes were applied to the controller in both the manual and automatic modes. At this reactor pressure, flow oscillations were observed with HPCI in the automatic flow control mode. The controller failed to restore and maintain HPCI in a stable operating condition. The HPCI turbine was removed from testing status at 1:43 A.M.

Additional tests of HPCI were conducted from 8:43 A.M. to 10:18 A.M. and from 1:07 P.M. to 1:24 P.M. Flow oscillations were induced and adjustments were made to the controller and process setpoints to obtain proper controller response. Because adjustments to the full flow controller did not result in elimination of flow instability, setpoints were returned to their previous settings. The actuator needle valve was then systematically adjusted followed by flow controller step changes to obtain proper HPCI response to flow oscillations. Additional testing of the flow controller demonstrated that proper damping of hydraulic signals to governor servo was obtained and that the previous flow instability had been eliminated.

During normal HPCI testing, the test return discharge valve to the condensate storage tank is prepositioned in a partially open position to develop pump discharge pressure of approximately 1200 psig at design flow. Therefore, a full flow path always exists through the pump, including initial fast start. During an actual vessel injection, the pump starts but there is only minimum recirculation flow until the developed HPCI pump discharge pressure exceeds reactor pressure and the feedwater injection check valve is forced open. Under these conditions the HPCI flow increases from zero to full flow almost instantaneously. To better simulate these conditions a final special test was run. During this test the HPCI test return discharge valve was initially in the full closed position and not opened until pump discharge pressure rose to approximately 1200 psig. When the valve was manually opened, HPCI responded properly without evidence of flow oscillation.

At 4:15 P.M. on March 23, HPCI was started for ST-4N. The test was completed at 7:30 P.M. and HPCI was declared to be operable.

Cause

The speed control system for HPCI was extensively tested and readjusted in December 1989 (LER-89-025). The needle valve adjustment affects system stability by varying the amount of damping and was properly adjusted for total system response at that time. During a subsequent surveillance test of February 20, 1990 (LER-90-005) it was noted that HPCI overall time to achieve full flow was marginal and steam stop valve opening time was too long. At that time the cause was found to be sluggish performance of the hydraulic remote servo. The servo was cleaned and response time of all parameters was significantly improved. This significantly shortened (faster) response time meant that there was less damping in the control system compared to December 1989. However, following the cleaning of the servo in February, the needle valve was not readjusted to provide the necessary increase in damping to compensate for more rapid response times. As demonstrated during the automatic injection of this event on March 19, 1990, response times were beyond the capacity of the hydraulic damping in the control system (as then adjusted) and resulted in unstable flow oscillations. However, there had been no indication of stability problems during any of the in-service testing which had been performed in December, February, or March.

The cause of this event was procedural inadequacy, Cause Code [D]. The existing surveillance and in-service tests for the HPCI system had been reviewed in detail for adequacy by the system vendor, the turbine vendor, and INPO. However, this event clearly demonstrates that the existing procedures do not sufficiently simulate the actual conditions of an injection to the reactor vessel. They did not produce the oscillations which would have revealed the need for adjustment of the control system. On the contrary, the results of these tests indicated that the system was stable and could be relied upon to perform properly during an automatic initiation. To induce oscillations in the system it was necessary to introduce step changes to the control circuit at full operating pressure, while testing under a temporary operating procedure. Further, only trial and error feedback methods were successful in establishing an appropriate adjustment for the needle valve. Each adjustment may require development of a unique trial and error technique (procedure) to adequately test the system. This type of situation is not readily addressed or anticipated by existing procedures.

Analysis

The HPCI system is an engineered safety feature designed to inject a highly reliable source of water into the reactor at rated pressure and in sufficient volume to maintain core coverage through a broad spectrum of hypothetical accident conditions. The principal component is a

turbine driven, high pressure, high volume, multi-stage centrifugal pump. The steam supply to the turbine comes directly from the reactor vessel thus ensuring availability regardless of the availability of the AC electrical power supplies.

The initial start-up flow instability of HPCI was of sufficient magnitude that the operator determined that it was necessary to take manual control. It is therefore reported under the provisions of 10 CFR 50.73(a)(2)(v) as an event or condition that alone could have prevented the fulfillment of the safety function of a system needed to remove residual heat or mitigate the consequences of an accident.

Surveillance tests verified that the back-up emergency core cooling systems were operable. While HPCI was not available, core coverage was assured by the automatic depressurization system together with the low pressure emergency core cooling systems including the one core spray system [BM] and one residual heat removal (low pressure coolant injection) subsystem [BO].

Furthermore, the HPCI system remained operable in the manual mode except for those intervals when servo adjustment or testing were in progress. The SAFER/GESTER LOCA sensitivity analysis shows that the HPCI response time (which was within FSAR limits) has little effect on peak clad temperatures.

Corrective Action

1. The needle valve in the HPCI speed control hydraulic loop was readjusted to provide more damping control (slower speed control circuit response time).
2. The HPCI system will be subjected to a full flow test with actual injection to the reactor vessel during the start-up from the 1990 refueling outage. Additional data will be collected and control circuits adjusted as appropriate.
3. Post-work test and maintenance procedures will be reviewed to develop appropriate steps to test for system stability following work on control system components.

Additional Information

Other LERs involving HPCI speed control problems:

90-009	02/19/90	Reactor scram
90-005	02/20/90	Speed control servo - sluggish response due to dirt in servo oil
89-025	11/30/89	High steam flow isolation - change in start-up speed control ramp generator adjustment
89-018	10/08/89	High steam flow isolation - change in start-up speed control ramp generator adjustment
89-002	03/02/89	Wiring error control system start-up speed ramp generator