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JAFP-90-0147

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

SUBJECT: DOCKET NO. 50-333
LICENSEE EVENT REPORT: 90-002-00
Shutdown Cooling System
Isolation

Dear Sir:

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

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

Very truly yours,

R. Lisero by direction

WILLIAM FERNANDEZ

WF:HCF:lar

Enclosure

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

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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)	DOCKET NUMBER (2)	PAGE (3)
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TITLE (4) Isolation of Shutdown Cooling System

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 1	2 0	9 0	9 0	— 0 0 2 —	0 0	0 2	1 5	9 0		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): N

POWER LEVEL (10): 0 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)	[X] 50.73(a)(2)(iv)	[ ] 73.71(b)
[ ] 20.405(a)(1)(i)	[ ] 50.36(c)(1)	[ ] 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	TELEPHONE NUMBER
Hamilton C. Fish	AREA CODE 3 1 5 — 3 4 9 - 6 0 1 3

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 []

At 6:24 A.M. on January 20, 1990, while the reactor was hot shutdown following a scram (LER-90-001), the reactor high pressure isolation logic tripped and isolated Residual Heat Removal (RHR) [BO] B Shutdown Cooling (SDC) system as the system was being started. This logic is set to trip at less than or equal to 75 psig to protect and isolate the low pressure piping of the SDC from the reactor pressure. At the time of the trip the reactor pressure was 27.5 psig, which is less than one half of the trip setpoint. Investigation found that the trip logic pressure sensors were properly calibrated and set at approximately 60 psig equivalent reactor pressure and that the reactor pressure reading of 27.5 psig was accurate.

No definitive cause was found. A hydraulic pressure transient is suspected as the cause. The operating procedure was revised to recommend starting the SDC at less than 10 psig rather than less than 75 psig reactor pressure.

EIIS Codes are in []

Description

On January 20, 1990 the reactor was hot shutdown following an automatic scram (LER-90-001) from full power the previous day (1/19/90) and was in the process of being brought to cold condition.

To maintain the reactor water temperature below 212° F, the shutdown cooling mode (SDC) of the Residual Heat Removal (RHR) [BO] is started. During initiation of the system it was isolated due to an apparently spurious high reactor pressure signal. The piping system and heat exchangers of the SDC are rated for low pressure (compared to reactor pressure) service. Because the system takes suction from and discharges to the reactor recirculation system [AD] piping which is connected directly to the reactor vessel, the low pressure RHR system must be protected from exposure to reactor pressure. This overpressure protection is accomplished by isolation signals which close isolation valves on the SDC system when a reactor pressure of less than or equal to 75 psig reactor dome pressure (Technical Specification Table 3.2-1) is sensed.

The operators verified that the reactor pressure was less than 75 psig. They continued to reduce reactor pressure to approximately 27 psig as determined from the plant process computer. The operator also verified that the reactor high pressure protective isolation (75 psig) relays were reset. When sufficient RHR water purity was obtained, the B RHR system was lined up for the SDC mode. The B RHR pump was started. The Low Pressure Coolant Injection (LPCI) outboard injection valve 10MOV-27B was jogged open. Flow in the RHR system increased to approximately 6,000 gpm. The injection valve, 10MOV-27B, was then throttled in the close direction to restrict flow to 3,500 gpm. At 6:24 A.M. as the control switch for the valve was being jogged, annunciator 09-4-3-22, "Shutdown Cooling Suction Header Pressure High", activated, isolation valves 10MOV-17 and -18 closed, and the B RHR pump tripped. The annunciator cleared immediately. The operators verified that a proper isolation had taken place and that the isolation relays 16A-K28 and 50 had reset. The keepfull system was restored to service on RHR B system.

The pressure switches for the reactor high pressure protection isolation to the RHR system had been functionally tested on January 17, 1990, three days before this event, and found to be within calibration tolerances. No adjustments were made. Following the event they were again functionally tested and found to be satisfactory with trip points set an equivalent (allowing for a 35 psig head correction factor) reactor pressure of approximately 60 psig which is 15 psi less than the Technical Specification limit.

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

The reactor pressure was further reduced to approximately 12 psig. The SDC system was restarted at 10:14 A.M., 3 hours and 50 minutes after the isolation. There was no interruption in the removal of decay heat.

Cause

Investigation of this event considered as possible causes the potential for the pressure sensing instruments to be out of calibration and the possibility of a pressure surge from initial pump operation and flow regulation causing a hydraulic transient at the isolation logic pressure sensor. The calibration check of the pressure switches for the isolation logic showed that they were in calibration and they were left as found.

The reactor pressure indication of 27.5 psig was obtained by the operator from the computer system display. Investigation of the validity of this value found that it was a composed point which is calculated from the input from ten wide-range pressure transmitters. The computer system for this point is provided with signal checking and validation logic which would have provided indication to the operator for a condition where the validity of the pressure reading was questionable. Therefore, this reading is considered to be highly reliable.

Strip chart recordings of reactor pressure are available and show a stable pressure of approximately 40 psig at the time of the isolation. They are recorded on slow speed of one inch per hour. The range for these charts is from 0 to 1,500 psig compressed to a vertical scale of approximately 4.25 inches where a 0.057 inch division equals 20 psig. On this scale, the resolution of a small spike from 27 psig to 60 psig would be insufficient to determine if the spike had in fact taken place. Further confirmation that the actual reactor pressure was approximately 27 psig was obtained by using the reactor coolant temperatures recorded in Surveillance Test ST-26J, "Heatup and Cooldown Temperature Checks" at 15 minute intervals and the strip chart recorded temperatures to compute the reactor pressure by use of steam tables.

Further, both of the pressure transmitters for the isolation logic are located on the suction line to reactor recirculation pump A. This line is hydraulically isolated from the B RHR system (which was the system in use) by the volume of water in the reactor vessel. Although it is unlikely that a hydraulic transient resulting from the starting of RHR system B pump in the SDC mode and throttling of the injection flow could have been sensed by the isolation logic in the A recirculation system, this appear to be the most reasonable cause. Nevertheless, no definitive cause has been found for this event.

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

Analysis

Isolation of the shutdown cooling system by the reactor high pressure isolation logic is reportable under the provisions of 10 CFR 50.73(a)(2)(iv) as an activation of an engineered safety feature [JE]. There were no system or equipment failures. The shutdown cooling system isolation was performed in accordance with design. Although the isolation reset immediately, it was decided to delay attempted restart of the SDC until the reactor pressure had dropped to a lower value. The SDC was restarted successfully at 10:14 A.M., 3 hours and 50 minutes after the isolation. The system could have been restarted before that time if it had been required. There was no interruption in the removal of decay heat from the reactor.

Corrective Action

A revision was made to Operating Procedure OP-13, "Residual Heat Removal System", under the section for shutdown cooling configurations in the first step to change the existing instruction from "reactor pressure must be less than 75 psig" to "reactor pressure should be as low as possible, preferably less than 10 psig when entering shutdown cooling." This revision was approved on January 24, 1990.