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United States Nuclear Regulatory Commission
Document Control Desk
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Washington, D.C. 20555

SUBJECT: DOCKET NO. 50-333
LICENSEE EVENT REPORT: 90-026-00 - Reactor Scrams
a. During Level Instrumentation Calibration and
b. Feedwater Low Flow Control Valve Failure

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,

WILLIAM FERNANDEZ

WF:HCF:lar

Enclosure

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

Cert NO P229764414

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NRC Form 366
(9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/88

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
JAMES A. FITZPATRICK NUCLEAR POWER PLANT	0 5 0 0 0 3 3 3	1 OF 0 6

TITLE (4) Reactor Scram During Reactor Water Level Instrument Surveillance. Second Reactor Scram Due to Air Diaphragm Failure in Feed Flow Control Valve.

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)
1 2	1 2	9 0	9 0	— 0 2 6 —	0 0	0 1	1 1	9 1		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)	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	
	AREA CODE	
Hamilton C. Fish	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
X	S J	F C V	M 1 2 0	Y					

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
X YES (If yes, complete EXPECTED SUBMISSION DATE)	NO		1 2	3 1	9 1

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

EIIS Codes are in []

A reactor scram from full power occurred at 1352 on 12/12/90 during calibration of reactor water level instrumentation. The instrument being calibrated shared common reference and variable level legs with instruments of the reactor protection system [JC]. The reactor scrammed as the instrument high pressure isolation valve was being cracked open during return to service. The scram resulted from a false low reactor water level signal. During the actual level transient following the scram difficulty was experienced with restarting the reactor feedwater pumps and a failure of the reactor feedwater low flow control valve occurred. A second scram occurred due to an actual low reactor water level at 1416 due to failure of the reactor feedwater low flow control valve air operator diaphragm. The plant returned to service at 0658 on 12/17/90 after being off line for 4 days, 17 hours, and 6 minutes. A root cause investigation of this scram is in progress and is expected to be completed prior to the end of the fall 1991 refueling outage. Until then, future calibrations will be conducted during scheduled outages.

Related LERs: 90-001 and 90-027.

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Description

The plant was operating at full power on December 12, 1990. A semi-annual calibration of three safety-related narrow range reactor water level instruments in the reactor protection system [JC] was in progress in accordance with the requirements of Technical Specification Table 4.2-6, "Minimum Test and Calibration Frequency for Surveillance Instrumentation". The calibration check of level transmitter A had been completed and the instrument had been restored to service. The calibration check of level transmitter C was completed. The transmitter was vented and pressurized to reactor pressure in preparation for return to service. In accordance with approved instrument surveillance procedure (ISP-3-4), "Reactor Water Level Instrument Calibration", the technician began to slowly crack open the high pressure isolation valve. Immediately the reactor scrammed on spurious low reactor water level signal from level transmitters A and B at 1352.

The operators followed the scram response procedure and verified the primary containment group II isolation signal (reactor water level 177 inches above top of active fuel) actions were complete including initiation of both trains of the reactor building standby gas treatment system [BH] and isolation of the reactor building ventilation system [VA]. The isolation signal to the Reactor Water Cleanup (RWCU) [CE] system was received. The inboard containment isolation valve closed as designed. However, the two RWCU outboard containment isolation valves did not close because the power supply circuit breakers to the motor operators for the valves had intentionally been electrically disabled to perform an approved test procedure for area high temperature (steam leak detection) instrument functional tests. At 1410, approximately 18 minutes after the event, the isolation valves closed when the circuit breakers were manually closed. The scram was reset.

The two steam turbine driven reactor feed pumps (RFP) [SJ] initially responded normally to the transient, increasing flow to restore reactor water level. As the level increased, operators manually tripped RFP A. Reactor water level continued to rise. At 222.5 inches above Top of Active Fuel (TAF), the high reactor water level automatic trip signal was received by RFP B. However, operators observed a delay of approximately 15 to 20 seconds before the steam supply stop valve actually closed. With both RFPs in a tripped condition, reactor water level began to fall. An attempt was made to return RFP B to service. The pump turbine failed to respond. At that point the operator decided to start RFP A. With indication that the reactor feed pump was coupled, the turbine speed was increased to 3,000 rpm. No indication of feedwater flow was observed. RFP A was shutdown. RFP B was restarted. The turbine would not roll with the motor speed changer. The motor gear unit was used. The pump turbine started to turn.

However, the reactor water level continued to drop and six minutes after the scram was reset and before RFP B could be brought up to speed, a second reactor scram (on low reactor water level 177 inches above TAF) was received at 1416. A group II containment isolation occurred in accordance with design.

Operators continued to bring the pump up to speed. Reactor water level was returned to a normal range. A higher than normal feed pump speed was required to maintain reactor vessel water level. The feedwater low flow control valve "open" demand signal was at 100%, a greater than normal value. There was indication of either backflow through the RFP A discharge check valve or of the low flow control valve sticking shut. An auxiliary operator reported that the low flow control valve appeared to be open. Reactor feed pump A was checked for backward rotation and was reported not to be rotating. Operators closed the isolation valve for RFP A discharge to the low flow control valve which also isolates the RFP A discharge check valve from the RFP B discharge. An immediate increase in reactor vessel level was observed when this valve was shut which supported evidence of a stuck open RFP A discharge check valve. Following the stabilization of reactor vessel level an operator observed the low flow control valve while the valve was remotely stroked from the control room. The valve stroked smoothly. At 1420 the reactor scram signal and the group II containment isolation signals were reset.

During these scrams the following discrepancies were observed in the performance of other safety-related systems.

1. Reactor control rod 22-31, although subsequently verified to be fully inserted, was indicated as being full out on the full core display matrix.

2. The reactor process computer print-out (OD-7 Rod Position Scan) did not indicate the actual control rod positions until the scram signal was reset.

3. Source Range Monitor A (one of four) exhibited erratic spiking.

Prior to the scram the following equipment was inoperable:

1. Average Power Range Monitor F (one of six)

2. Source Range Monitor C (one of four)

Reactor start-up was commenced on December 15th at 1350. Following a scram at 2140 during the start-up (LER-90-027) due to problems with feedwater flow control, the plant was placed on line at 0658 on December 17th. Total out of service time was 4 days, 17 hours, 6 minutes.

Cause

1. Initial Scram

A false and spurious low reactor vessel water level signal initiated the reactor scram signal. It was later verified that the actual reactor water level had remained constant through the initiating event. The only parameters changing were the vessel level signals associated with the shared hydraulic instrument headers. The scram signal was initiated for low reactor water level by reactor protection system level transmitters A and B which share a common reference column and variable leg with feedwater level control transmitter C which was being restored to service.

To determine the cause of the false low level signal, the testing conditions were simulated at the instrument rack during the plant shutdown. Transmitter C was removed from service. The pressure inside the isolated transmitter was raised to 1,000 psig with test equipment. A pressure drop of only 8 psig (0.8%) over a ten-minute period demonstrated that the isolation valve packing was satisfactory. A small low pressure side valve stem packing leak, previously observed during instrument calibration, could not be duplicated. However, the available reactor pressure had been reduced to 575 psig during cooldown instead of the normal operating pressure of approximately 1,000 psig which had existed at the time of the scram. Thus, the rate of leakage and the nature of the pressure transients developed during the simulation varied from the actual conditions which existed during plant operation. A previous simulation following LER-90-001 demonstrated that leakage from low pressure side valve packing resulted in spurious level signals in the instruments that shared the common reference and variable legs. However, similar packing leaks in other level instrument isolation valves have not resulted in spurious low level scram signals. Because an almost identical event (LER-90-001) occurred in January 1990 with the same instrument, a more detailed investigation of the root cause is being conducted.

2. Secondary Scram

The cause of the second scram was the actual low reactor water level transient. This transient resulted in part from the time delay in returning reactor feed pump B to service. The delay in returning RFP B to service resulted from an improper setting of the overlap control between the pump turbine throttle valve motor speed changer (MSC) and motor gear unit (MGU).

Disassembly of the RFP A discharge check valve provided inconclusive results as to possible malfunction. It was subsequently (LER-90-027) discovered that the low flow feedwater control valve had failed to fully stroke. The valve stroke was found to be reduced to 3/4-inch compared to a normal 2-inch stroke. This reduced stroke resulted in a significant reduction in feed flow estimated to have been from approximately 4,000 gpm to less than 1,500 gpm and this in turn contributed to the low reactor water level. The valve operator failed to fully stroke because the air diaphragm had failed. Three small radial cracks were found in the periphery of the diaphragm. The diaphragm was the original fabric weave elastomeric Buna-N material. It had been in service for approximately 15 years.

Analysis

As an automatic scram, this event is reported under the provisions of 10 CFR 50.73(a)(2)(iv) which requires reporting of any event or condition that resulted in a manual or automatic actuation of any Engineered Safety Feature.

FSAR Section 14.2 "Unacceptable Safety Results for Accidents" and Section 14.5 "Analysis of Abnormal Operational Transients and Reactor Vessel Overpressure" were reviewed with respect to this event. Based on the instrument signals, the appropriate trips and isolations occurred. The standby gas treatment fans started and ran properly. The primary containment isolation system group II isolation was successful. Once the breakers were racked in the reactor water clean-up system isolation valves were successfully closed. Vessel pressure and level control were maintained within acceptable ranges using RFP B and turbine bypass valves.

Corrective Action

1. During the near term, the semi-annual calibration check of the water level instrumentation will be performed during periods of plant shutdown. The next calibration check will be performed in less than six months during the mini-outage scheduled for March 1991 and again during the refueling outage scheduled for October 1991.

2. A root cause investigation will be conducted. The goal for completion of the investigation will be the end of the fall 1991 refueling outage.

3. The possibility of increasing the surveillance interval of the feedwater control level transmitter through a Technical Specification amendment will be evaluated.

4. The instrument isolation and equalizing valves for reactor water level transmitter C were replaced.

5. The reactor feedwater low flow control valve operator was subsequently repaired after a scram on 12/15/90 (LER-90-027).

6. The reactor feed pump discharge valve was disassembled and inspected.

7. The start-up procedure, OP-65, will be updated to provide verification of full stroke operability of the reactor feedwater low flow control valve.

Additional Information

Related LERs: 90-001 and 90-027

Failed Component Data:

Function:	Reactor Feedwater Low Flow Control Valve Operator
Plant Component Identification:	34FCV-137(OP)
Manufacturer:	Masoneilan
Model:	38-2X871
Type:	Air Diaphragm Operator
NPRDS Vendor Code:	M120
NPRDS Component Code:	VALVOP