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Palisades Nuclear Plant: 27780 Blue Star Memorial Highway, Covert, MI 49043

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General Manager

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Nuclear Regulatory Commission
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DOCKET 50-255 - LICENSE DPR-20 - PALISADES PLANT -
LICENSEE EVENT REPORT 92-037 - REACTOR TRIP CAUSED BY LOW STEAM GENERATOR LEVEL RESULTING FROM A BROKEN AIR LINE ON A MAIN FEEDWATER REGULATING VALVE.

Licensee Event Report (LER) 92-037 is attached. This event is reportable in accordance with 10CFR50.73(a)(2)(iv) as an event that resulted in the automatic actuation of the reactor protective system (RPS).

Gerald B Slade
General Manager

CC Administrator, Region III, USNRC
NRC Resident Inspector - Palisades

Attachment

NRC Form 366 (9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
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LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Palisades Plant	0 5 0 0 0 2 5 5	1 OF 0 4

TITLE (4) REACTOR TRIP CAUSED BY LOW STEAM GENERATOR LEVEL RESULTING FROM A BROKEN AIR LINE ON A MAIN FEEDWATER REGULATING VALVE

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (8)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAMES	DOCKET NUMBER(S)
0 8	1 4	9 2	9 2	– 0 3 7 –	0 0	0 9	1 4	9 2	N/A	0 5 0 0 0
									N/A	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	
Cris T. Hillman, Staff Licensing Engineer	6 1 6	7 6 4 - 8 9 1 3

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

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
[ ] YES *(If yes, complete EXPECTED SUBMISSION DATE)*	[X] NO				

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

On August 14, 1992 at 0017 hours, with the plant operating at 100% power, the control room received a low steam generator level alarm on the "A" steam generator. Indicated flow to the steam generator "A" was approximately 40% of the normal feedwater flow. Based on the control room indications the main feedwater regulating valve control signal and the main feedwater pumps appeared to be responding properly to the alarmed condition; however, water level in the "A" steam generator continued to drop until it reached the reactor protective system (RPS) setpoint. The reactor automatically tripped on low level in the "A" steam generator.

The cause of this event was a broken air supply line between the position operator controller and solenoid valve that supplies air for the open signal of the valve actuator, which allowed the valve to partially close and restrict flow to the "A" steam generator.

Corrective action for this event prior to plant re-start included replacing all the air lines on both main feedwater regulating valves, inspection of other key control valves in the turbine building for evidence of leaking Swagelok connections. Proposed corrective action includes reviewing the maintenance department's installation methods for compression fittings, inspecting the air lines to critical valves to determine the amount of vibration during plant operation and evaluating a method of repair, and evaluating the need for criteria for the re-use of compression fittings.

EVENT DESCRIPTION

On August 14, 1992 at 0017 hours, with the plant operating at 100% power, the control room received a low steam generator level alarm on the "A" steam generator [AB;SG]. Indicated flow to the steam generator "A" was approximately 40% of the normal feedwater flow. Based on the control room indications, the main feedwater regulating valve [JB;FCV] control signal and the main feedwater pumps [SJ;P] appeared to be responding properly to the alarmed condition; however, water level in the "A" steam generator continued to drop until it reached the reactor protective system (RPS) setpoint. The reactor automatically tripped on low level in the "A" steam generator. This in turn caused an automatic actuation of the auxiliary feedwater system. The automatic reactor trip was successfully completed with all systems responding adequately. The control room immediately initiated emergency operating procedures (EOPs) EOP-1, "Standard Post Trip Actions," and EOP-2, "Reactor Trip Recovery." The required actions for the EOPs were successfully completed and the procedures were exited. Plant response to the trip was adequate in that all safety systems responded normally.

This event is reportable to the NRC in accordance with 10CFR50.73(a)(2)(iv) as an event that resulted in the automatic actuation of the reactor protective system (RPS).

CAUSE OF THE EVENT

This event was caused by a broken air supply line between the position operator controller (POC)-0701 and solenoid valve (SV)-0701B (supply air for the open signal of the valve actuator), which allowed CV-0701 to partially close and restrict main feedwater flow to the "A" steam generator.

This event does not involve the failure of any equipment important to safety.

ANALYSIS OF THE EVENT

Prior to the event the plant was stable with all systems in a normal, full power alignment. The reactor was at 100% power. Water levels on both steam generators were at approximately 65%.

Upon receiving the low level alarm (55%) for the "A" steam generator, the control room operators immediately checked the status of the main feedwater regulating valves (CV-0701 and CV-0703) and the main feedwater pumps (P-1A and P-1B) and found that the pumps and valves appeared to be responding properly. Within thirty-nine seconds of receiving the low level alarm, water level in the "A" steam generator reached the reactor protective system (RPS) setpoint and the RPS initiated a reactor trip signal. This in turn caused an automatic actuation of the auxiliary feedwater system. The feedwater control system responded properly. Overall plant response was acceptable.
Since the control room had indication of the proper operation of both the main feedwater regulating valves and the main feedwater pumps just prior to the RPS initiated reactor trip, it was first believed that the problem was either with instrument indication or gross failure of a secondary side component downstream of the main feedwater pumps. Following the trip, a physical inspection of the plant by shift personnel did not reveal any gross failure of any secondary side components downstream of the main feedwater pumps.

Main feedwater regulating valve CV-0701 was inspected and the air supply line between position operator controller (POC)-0701 and solenoid valve (SV)-0701B (supply air for the open signal of the valve actuator) was found to be fully severed at the Swagelok fitting.

The loss of the air line to the main feedwater regulating valve actuator appeared to have resulted in the plant transient described in this event report; however, a control room operator reported that he observed the control room position indication for CV-0701 to have been "full open" just prior to the reactor trip. If the broken air line to the valve actuator was the single cause for the event, then the indication observed by the control room operator would have been in error. Therefore, more extensive troubleshooting was required.

To determine whether the loss of main feedwater flow to the "A" steam generator was caused solely by the broken air line or whether other component problems existed, a testing and inspection plan was developed. A flow test, using condensate flow through the main feedwater system and directed back to the condenser via the high pressure feedwater heater recirculation line was performed in accordance with standard operating procedure (SOP) 12, "Feedwater System." All valves in the flow path from the main feedwater pump to the main feedwater check valve (CK-0702) were cycled. All of the valves successfully stopped and started flow; however, this did not rule out the possible separation of the valve stem from the valve disk on CV-0701. Therefore, measurements of the valve operator stroke were taken for CV-0701 with the valve stem coupled to the actuator and with the valve stem uncoupled from the actuator. These measurements verified that the valve stem was coupled to the valve disk. In addition, technical specifications surveillance procedure (TSSP) QO-24, "Verify Closure of the Main Feed Water Check Valves," was performed to verify seating of the main feedwater check valves. The testing and inspection plan did not identify any other component problems beyond the CV-0701 air line failure. Lastly, the valve position indication for CV-0701 was tested and was found to be indicating properly. No explanation was determined for the control room operator's observation of CV-0701 appearing to respond properly (full open position) to the low level condition in the "A" steam generator.

In addition, the portion of the tubing containing the Swagelok fitting and ferrules removed from solenoid valve SV-0701B was sent to the Swagelok vendor for analysis. The vendor determined the air line break was a fatigue failure caused by vibration. The vendor recommended that a tube support as close to the fitting as possible may help to prevent the recurrence of this type of failure.

CORRECTIVE ACTION

All of the air lines to main feedwater regulating valves CV-0701 and CV-0703 were replaced and the valves was satisfactorily tested. Furthermore, the air lines to other key control valves in the Turbine Building were inspected for evidence of leaking Swagelok fitting connections. Six additional valves were identified as having leaking Swagelok fittings and were repaired.

Long term corrective action includes (1) reviewing the maintenance department's installation methods for compression fittings, (2) inspecting the air lines to critical valves to determine the amount of vibration during plant operation and evaluating a method of repair and (3) evaluating the need for criteria for the re-use of compression fittings is required.

ADDITIONAL INFORMATION

None