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)
Perry Nuclear Power Plant, Unit 1	0 5 0 0 0 4 4 0	1 OF 0 3

TITLE (4) System Vibration Results In Failure Of A Condensate Drain Line and the Subsequent Initiation Of A Manual Reactor Scram

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 5	2 4	8 7	8 7	— 0 3 5 —	0 0	0 6	2 3	8 7		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1
POWER LEVEL (10): 0 5 5

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)
Gregory A. Dunn, Compliance Engineer, Extenstion 6484	2 1 6 2 5 9 - 3 7 3 7

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 D	F C V	F 1 3 0	N					

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)

On May 24, 1987 at 0304, a Reactor Protection System (RPS) manual reactor scram was initiated by control room operators in response to a leak in the Condensate system. Since the condensate leak could not be isolated with the Condensate system in service, the operations section Shift Supervisor ordered a reactor shutdown. Investigation following the scram revealed that the condensate leak was originating from the 1N21-F701B valve (1N21-F230 Upstream Piping Drain). The 1N21-F701B valve and drain line had severed from the condensate piping.

The cause of this event was excessive vibration of the Condensate system resulting in the failure of the piping for the 1N21-F701B drain valve. An engineering design change had been initiated due to a previous event (see LER 87-030) to add a second condensate control valve in parallel with 1N21-F230 and add additional pipe supports/guides to reduce the vibration experienced in the vicinity of the 1N21-F230 valve. This design change is under development to be implemented in an upcoming outage. An evaluation of vibration effects on Condensate and Feedwater system vent and drain valves will also be performed to determine further corrective actions. In addition, an engineering design change is being studied to reduce the pressure in the system piping.

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

On May 24, 1987 at 0304, a Reactor Protection System (RPS) [JC] manual reactor scram was initiated by control room operators in response to a leak in the Condensate system [SD]. Prior to the event, the plant was in Operational Condition 1 (Power Operation) with Feedwater Control system [JB] startup testing in progress. Reactor thermal power was approximately 55 percent of rated, reactor coolant temperature was approximately 535 degrees, and reactor vessel [RPV] pressure was approximately 930 psig.

On May 24 at 0302, control room operators received a report from the Radwaste control room technicians that the floor drain sump inleakage for the Turbine Power Complex [NM] and Heater Bay had increased dramatically. Both the normal and standby sump pumps were running. Investigation by the operations Shift Supervisor of the Heater Bay revealed that a condensate leak in the vicinity of the condensate flow control valve 1N21-F230 existed. In response to the condensate leak at the 1N21-F230 valve (cannot be bypassed or isolated with the Condensate system in service), the Shift Supervisor ordered that a fast reactor shutdown be performed.

In accordance with the Shift Supervisor's order, control room operators reduced reactor power to approximately 25 percent of rated by running back the Reactor Recirculation system [AD] flow control valves to their minimum position and transferring the Reactor Recirculation pumps to slow speed. Control room operators then manually actuated a RPS reactor scram at 0304:15 and entered Off-Normal Instruction (ONI)-C71, "Reactor Scram (Unit 1)." Further investigation by the Shift Supervisor following the scram revealed that the condensate leak was originating from the 1N21-F701B valve (1N21-F230 Upstream Piping Drain). The 1N21-F701B valve and drain line had severed from the condensate piping.

At 0304:24 reactor vessel level decreased to less than Level 3 (+177.7 inches above top of active fuel (TAF)) and Control Room operators entered Plant Emergency Instruction (PEI)-B13, "Reactor Pressure Vessel Control." During the transfer of plant loads from the Auxiliary Transformer to the Startup Transformer, the Reactor Recirculation pump A tripped and Control Room operators entered ONI-B33-2, "Loss of One or Both Recirculation Pumps." At 0304:42, reactor vessel level increased above Level 3. The reactor scram was reset at 0309:20 and PEI-B13 was exited. The minimum reactor vessel level reached during this event was +166 inches above TAF.

At 0352, Reactor Recirculation pump A was restarted in slow speed and ONI-B33-2 was exited. ONI-C71 was exited at 0430. The Feedwater system [SJ] was secured at 0524 and the Condensate system was placed in short cycle cleanup bypassing the condensate leak. Reactor Core Isolation Cooling system (RCIC) [BN] was used for vessel makeup for approximately 40 minutes. Reactor Water Cleanup system (RWCU) [CE] was then used for reactor vessel level control.

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

The 1N21-F701A (1N21-F230 Downstream Piping Drain) and 1N21-F701B drain valves were removed and replaced with pipe caps. The transient analysis was completed and approval to restart was granted by the Plant Manager on May 25. The plant entered Operational Condition 2 (Startup) on May 25 at 2310.

The cause of this event was excessive vibration of the Condensate system resulting in the failure of the piping for the 1N21-F701B drain valve. A previous event (see LER 87-030) involved the failure of the 1N21-F230 valve air control supply lines due to excessive vibration. As a result of the previous event, an engineering design change had been initiated to add a second condensate control valve in parallel with 1N21-F230 and add additional pipe supports/guides to reduce the vibration experienced in the vicinity of the 1N21-F230 valve. This engineering design change is under development to be implemented in an upcoming outage.

In addition, inspection of the 1N21-F230 valve (Manufacturer: Fisher Controls, Model #: 16" BWE 300# 496L-5 ELT-3570P) revealed that the valve could not be operated manually due to stem rotation. This had occurred also during the May 1 reactor scram (see LER 87-030). Investigation has shown that this problem was caused by a weak stem guide. Investigation of the Reactor Recirculation pump trip during the transfer of plant loads is continuing.

Reactor power level was relatively low at the time of the reactor scram and the resulting transient was small. Plant systems, with the exception of those described above responded to the plant transient as designed to maintain the plant in a safe condition. For high power levels, FSAR Section 15.2.7 analyzed the loss of feedwater flow which results in a reactor scram at Level 3. The low Level 3 scram trip function meets single failure criterion. This analysis is a more limiting event than experienced during the manual reactor scram. Valves 1N21-F701A & B are drain valves located on either side of the 1N21-F230 condensate control valve. Valve 1N21-F230 regulates flow of condensate water to the direct contact heater and the Hot Surge Tank (HST). The HST acts as a surge volume and reservoir to accommodate changes in system demand and to provide the required NPSH for the RFBPs. The condensate system is non-safety class and performs no safety function. For these reasons, this event is not considered safety significant.

Additionally as a result of this event, an evaluation of vibration effects on Condensate and Feedwater system vent and drain valves will be performed to determine further corrective actions. Also, the stem guide for the 1N21-F230 valve will be replaced with a thicker stem guide manufactured with a stronger material. In addition, an engineering design change is being studied to reduce the pressure in the system piping.

Energy Industry Identification System Codes are identified in the text as [XX].

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MURRAY R. EDELMAN
SR. VICE PRESIDENT
NUCLEAR

June 23, 1987
PY-CEI/NRR-0671 L

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Perry Nuclear Power Plant
Docket No. 50-440
LER 87-035-00

Dear Sir:

Enclosed is Licensee Event Report 87-035-00, for the Perry Nuclear Power Plant.

Very truly yours,

A. Kaplan for

Murray R. Edelman
Senior Vice President
Nuclear Group

MRE:njc

Enclosure: LER 87-035-00

cc: T. Colburn
K. Connaughton

U.S. Nuclear Regulatory Commission
799 Roosevelt Road
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