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)
Calvert Cliffs Unit 2	0 5 0 0 0 3 1 8	1 OF 0 6

TITLE (4) Plant Shutdown Resulting From an Air Leak on Feedwater Regulating Valve Caused by Failure to Identify the Need to Perform Adequate Preventive Maintenance

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 0 0 0
0 3	0 7	8 9	8 9	— 0 0 3	— 0 0	0 4	0 6	8 9		0 5 0 0 0

OPERATING MODE (9)	1	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)			
POWER LEVEL (10)	1 0 0	20.402(b)	20.405(c)	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)	X 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
John M. Osborne, Licensing Engineer	3 0 1 2 6 0 - 4 3 8 4

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	
D	S J	F C V	F 1 3 0	Y							

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 March 7, 1989, at 1910, Calvert Cliffs Unit 2 completed a shutdown as required by Technical Specification 3.0.3. This shutdown was required to time response test the #22 Steam Generator (SG) Feedwater Regulating Valve (FRV) which had failed due to an air leak in its actuator positioner. Unit 2 was operating in Mode 1 at 100% power prior to the shutdown. The total duration of the event was thirteen hours and 50 minutes.

The root cause of the event was the failure to identify the need to perform adequate preventive maintenance on the positioner for the #22 SG FRV. Neither vendor information nor in-plant experience indicated that the positioner pedestal gasket required periodic replacement.

As an immediate corrective action, operators placed the FRV controller in manual when #22 SG level dropped to minus (-)31 inches. Additional corrective actions to prevent recurrence include:

(1) Develop a periodic preventive maintenance schedule for the positioner pedestal gaskets based on vendor recommendations and in-plant experience.

(2) Inspect the #21 FRV positioner pedestal gaskets during the next plant shutdown. The Unit 1 FRVs, #11 and #12, had their positioner pedestal gaskets replaced on March 6, 1989 and January 11, 1989, respectively, during the performance of troubleshooting and corrective maintenance to their positioners.

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

I. DESCRIPTION

On March 7, 1989, at 1910, Calvert Cliffs Unit 2 completed a shutdown as required by Technical Specification (T.S.) 3.0.3. This shutdown was required to time response test the #22 Steam Generator (SG) Feedwater Regulating Valve (FRV) (2-FW-1121-CV) which had failed due to an air leak in its actuator positioner. Unit 2 was operating in Mode 1 at 100% power prior to the shutdown.

On March 7, 1989, at approximately 0940, the #22 SG began to experience level oscillations of approximately ± 5 inches from the zero reference at a period of approximately 5 minutes. The operators performed local inspections of equipment in an attempt to identify a potential cause of these oscillations. This equipment included SG FRVs, Feedwater Heaters, and Steam Generator Feed Pumps. The cause was not discovered.

At approximately 1144, the #22 SG level lowered to approximately minus (-)31 inches due to an air leak on the actuator positioner for the #22 SG FRV (see diagram). The Reactor Protective Systems (RPS) pretrip alarms for Channel A, B, C, and D for low SG level were received. No RPS or Engineered Safety Features Actuation System (ESFAS) was manually or automatically started or actuated. No RPS or ESFAS actuation was required to place the plant in a stable condition.

The SG level was restored by opening the #22 SG FRV by remote manual control from the Control Room in accordance with Abnormal Operating Procedure (AOP)-3G: Malfunctions of Main Feed System. The plant was stabilized at 1147 hours.

At 1220, local manual control of the #22 SG FRV was taken in accordance with OI-12A: Feedwater System to allow repair of the actuator. This included locally stationing licensed operators at the FRV who were in constant communication with the Control Room. Problems associated with establishing these communications delayed taking local manual control of the #22 SG FRV. Plant management had concluded that continued steady state operation while attempting to repair the actuator was more prudent than attempting a plant shutdown with known failed equipment. A controlled shutdown was considered difficult to achieve in that a plant trip on abnormal SG level was considered likely due to the failed FRV.

By 1420, the actuator was inspected and repaired. The FRV was returned to automatic control in accordance with OI-12A: Feedwater System. At 1555, plant management concluded that the FRV appeared to be operating properly but its requirement to shut within 20 seconds following a turbine trip needed to be verified. They concluded the response time must be verified by surveillance testing. The Unit-2 shutdown was commenced. At 1655, an Unusual Event was declared. The shutdown was complete at 1910. The Unusual Event was terminated at 2330 when Surveillance Test Procedure STP-M-521-2 ESFAS Time Response Test was completed satisfactorily and verified that the FRV would shut in 18.65 seconds.

There were no other components or systems which were inoperable and/or out-of-service which contributed to this event. No other system or component failures resulted from this event. Operator actions throughout the event were deliberate and correct.

II. CAUSE OF EVENT

The root cause of the event was the failure to identify the need to perform adequate preventive maintenance on the positioner for the #22 SG FRV. A neoprene gasket between the positioner pedestal and positioner housing was found brittle and cracked. The positioner pedestal gasket (Fisher Controls, part number 1H854703012) was replaced. Maintenance Technician actions throughout the event were deliberate and correct. Preventive Maintenance to verify proper operation of valve function was in place but it did not require periodic replacement of any gasket or o-rings. The Fisher Controls 3570 Series Pneumatic Valve Positioner Instruction Manual (Form 1837) does not specify any periodic replacement of gaskets or o-rings. However, it does recommend that when troubleshooting the positioner for sluggish or erratic operation, the o-rings and gaskets should be checked for wear or damage. This is our standard practice when troubleshooting mechanical devices. The local Fisher Controls representative was contacted to obtain an expected life for the gasket. No recommendation for periodic replacement of the gasket is available since these valves are designed and commonly used in environments such as out-door use and corrosive atmospheres or where air supply quality varies greatly. The best available recommendation was to base the periodic replacement on in-plant experience. The representative considered our installed environment as relatively benign since our air is of high quality and the valve is located in an area of constant temperature. Another vendor for valves that we use in containment environments, suggested four years as an acceptable period for o-ring or gasket replacement. However, this value is quite conservative as it assumes a post-LOCA environment during the four year period.

The positioner pedestal gasket for the #22 SG FRV has historically been replaced as a subtask to replacing or repairing the positioner. There are no known failures of only the positioner pedestal gasket. Therefore, based on our plant experience and vendor information, we have concluded that we had no reason to institute a PM prior to the March 7, 1989, gasket failure.

III. ANALYSIS OF EVENT

This event is considered reportable in accordance with 10 CFR 50.73(a)(2)(i)(A). "The completion of any nuclear plant shutdown required by the plant's Technical Specifications." In accordance with T.S. 3.0.3 when the 20-second response time test of the FRV to shut following a turbine trip, T.S. 3.3.2.1, was determined to require verification, a shutdown was required. The total duration of the event was thirteen hours and 50 minutes.

The FRV receives information from steam flow, feed flow and SG level and positions itself appropriately to ensure SG level is correct. In the event of a reactor or turbine trip, 5 percent of full feedwater flow to the SG is required for decay heat removal but overly rapid addition of feedwater could result in a reduction of shutdown margin. This requirement is met by the FRV shutting fully (less than 20 seconds required by T.S. 3.3.2.1, Table 3.3-5) and the FRV bypass valve opening to 5 percent of full feedwater flow. In the event of a loss of air supply to the FRV positioner, the FRV fails-as-is to minimize plant transients that could occur due to a loss of air supply. When the positioner pedestal gasket failed, the FRV began to hunt its appropriate position which resulted in the ± 5 in. SG level changes.

Following repair, subsequent testing showed the FRV would have performed its intended function and shut in 18.65 seconds following a reactor or turbine trip. Although the FRV may not have been able to perform its intended function prior to its repair, there is no significant safety consequence resulting from this event. The Steam Line Break Event, Updated Final Safety Analysis Report (UFSAR) Chapter 14, pg. 14.14-6, is the limiting event; however, the failure of one High Pressure Safety Injection pump to start is identified as the limiting single component failure. The failure of the FRV to ramp shut following a turbine trip was also analyzed but it was not the limiting case. Therefore, the UFSAR analysis of the Steam Line Break Event bounds the event discussed in this LER.

IV. CORRECTIVE ACTIONS

The immediate corrective action was to place the FRV controller in manual as discussed in the DESCRIPTION section. The FRV was repaired and tested satisfactorily.

Additional corrective actions to preclude recurrence include the following:

1. Develop a periodic preventive maintenance schedule for the positioner pedestal gaskets based on vendor recommendations and in-plant experience. A Reliability Centered Maintenance (RCM) project is currently underway and will incorporate RCM techniques in various plant systems. The FRVs will be included in this project based on an approach involving phased implementation of RCM to plant systems. Plant systems are selected for RCM application based on their importance ranking to reactor safety as reflected in the results of the CCNPP Unit 1 Interim Reliability Evaluation Program Report.

2. Inspect the #21 SG FRV positioner pedestal gaskets during the next plant shutdown. The Unit 1 FRVs, #11 and #12, had their positioner pedestal gaskets replaced on March 6, 1989 and January 11, 1989, respectively, during the performance of troubleshooting and corrective maintenance to their positioners.

3. Verify the adequacy of and complete the upgrade to the preventive maintenance program for the plant site internal communication system.

4. Resolve the apparent discrepancy between the T.S. requirement for the FRV to shut in less than 20 seconds following a reactor or turbine trip and the potential to violate this requirement when the FRV fails-as-is following a loss of air supply.

There has been one similar event that involved FRV failure from an air supply line rupture. This event is described in LER 317/88-09.

Identification of components referred to in this LER are:

Component	IEEE 803 EIIS Funct	IEEE 805 System ID
Steam Generator	SD	SB
Feed Regulating Valve	FCV	SJ
High Pressure Safety Injection Pump	P	BQ
Steam Generator Feed Pump	P	SJ
Feedwater Heater	HX	SJ
Reactor Protection System	N/A	JC
Engineered Safety Feature Actuation System	N/A	JE

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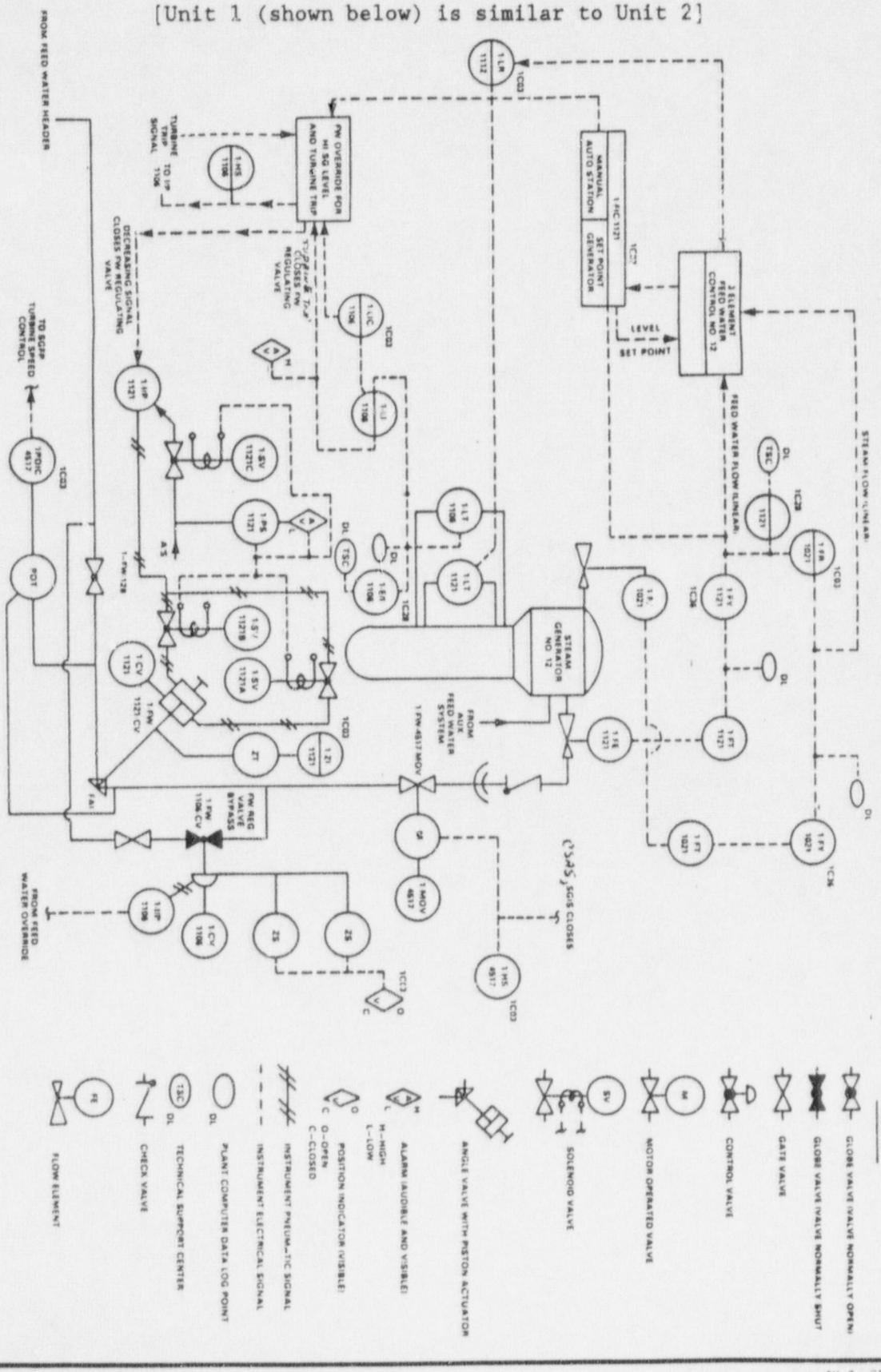
TEXT *(If more space is required, use additional NRC Form 366A's)* (17)

DIAGRAM

[Unit 1 (shown below) is similar to Unit 2]



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CALVERT CLIFFS NUCLEAR POWER PLANT DEPARTMENT
CALVERT CLIFFS NUCLEAR POWER PLANT
LUSBY, MARYLAND 20657

April 6, 1989

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

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License No. DPR 69

Dear Sirs:

The attached LER 89-003 is being sent to you as required by 10 CFR 50.73.

Should you have any questions regarding this report, we would be pleased to discuss them with you.

Very truly yours,

L B Russell

L. B. Russell
Manager-Calvert Cliffs Nuclear Power Plant Department

LBR:JMO:tls

cc: William T. Russell
Director, Office of Management Information
and Program Control
Messrs: G. C. Creel
W. J. Lippold

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