Charles H. Cruse
Plant General Manager
Calvert Cliffs Nuclear Power Plant

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BGE

August 10, 1994

U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

ATTENTION: Document Control Desk

SUBJECT: Calvert Cliffs Nuclear Power Plant
Unit No. 2; Docket No. 50-318; License No. DPR 69
Licensee Event Report 94-003
Reactor Shutdown for Leak From Fatigue Crack in Safety Injection Tank Line

The attached report is being sent to you as required under 10 CFR 50.73 guidelines. Should you have any questions regarding this report, we will be pleased to discuss them with you.

Very truly yours,

John T Carroll
for
C. H. Cruse
Plant General Manager

CHC/DWM/bjd
Attachment

cc: D. A. Brune, Esquire
J. E. Silberg, Esquire
P. T. Kuo, NRC
D. G. McDonald, Jr., NRC
T. T. Martin, NRC
P. R. Wilson, NRC
R. I. McLean, DNR
J. H. Walter, PSC
Director, Office of Management Information
and Program Control

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NRC FORM 366 (5-92)

U. S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

APPROVED BY OMB NO. 3150-0104
EXPIRES: 5/31/952

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Calvert Cliffs, Unit 2	05000 318	1 OF 07

TITLE (4)
Reactor Shutdown for Leak From Fatigue Crack in Safety Injection Tank Line

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 NUMBERS(S)
										05000
07	11	94	94	-- 003 --	00	08	10	94		05000

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR: (Check one or more) (11)

OPERATING MODE (9)	1		20.402(b)		20.405(c)		50.73(a)(2)(iv)		73.71(b)
POWER LEVEL (10)	100		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 (include Area Code)
D. W. Muth, Compliance Engineer	410-260-3592

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 15 single-space typewritten lines) (16)

On Tuesday, July 11, 1994 at about 1115 hours, a non-isolable Reactor Coolant System pressure boundary leak was discovered at Calvert Cliffs Unit 2. The leak was found to be caused by a 150 degree circumferential crack in a weld in the 22A Safety Injection Tank discharge test connection. In accordance with Technical Specification 3.4.6.2, "Reactor Coolant System Leakage," Action A, the Unit commenced a shutdown to HOT SHUTDOWN (MODE 4).

The line had been removed and replaced in the spring of 1993. It is suspected that minor changes made at this time resulted in harmonic oscillation causing high cycle fatigue failure.

The support for this line was redesigned and similar lines were examined to verify that similar conditions do not exist. The circumstances of this event will be reviewed with Design Engineering personnel. We will evaluate whether additional generic measures to prevent high frequency fatigue failure are appropriate.

I. DESCRIPTION OF EVENT

On Tuesday, July 11, 1994 at about 1115 hours, a non-isolable Reactor Coolant System (RCS) pressure boundary leak was discovered at Calvert Cliffs Unit 2. In accordance with Technical Specification 3.4.6.2, "Reactor Coolant System Leakage," Action A, the Unit commenced a shutdown to HOT SHUTDOWN (MODE 4). At the time the condition was discovered, the Unit was operating at 100 percent rated thermal power at normal operating temperature and pressure.

On July 7, 1994, 22A Safety Injection Tank (SIT) outleakage increased noticeably. At about the same time frame, an increase in containment sump discharge frequency was noted indicating increased leakage to containment. On July 11, 1994, an operator entered containment to determine why the 22A SIT leakage rate had increased to approximately 20 gallons per hour (0.33 gpm). Upon arriving in the vicinity of the 22A SIT outlet Motor Operated Valve, the operator discovered an active leak from the inlet weld on a 3/4 inch to 1 inch reducer for vent valve 2-SI-556 on the 22A SIT discharge test connection (see Figure 1). He noted boric acid buildup around the leak and estimated that the leak was spraying a mist about two to three feet outward for about a 150 degree circumference around the pipe.

The SIT discharge test connection consists of a 3/4 inch segment of austenitic stainless steel schedule 160 pipe running from the SIT discharge line and socket welded to a 3/4 inch to 1 inch reducer attached to a 1 inch globe valve. The crack was located in this socket weld (see Figure 2). The line is isolated from the Reactor Coolant loop by the Safety Injection Loop Check Valve (see Figure 1).

Plant System Engineers, a Weld Shop planner, and Materials Engineering personnel entered containment to verify the location, extent of leakage, and perform an initial characterization of the leak. They also performed an assessment of similar installations on the other three SITs in Unit 2. They noted no other leaks at these locations. They determined that the leakage was coming from an approximately 150 degree circumferential crack, the middle 50 degrees of which was through-wall and passing the leakage. The team's initial assessment was that the crack resulted from high cycle fatigue.

Upon notification that the leakage was non-isolable and within the RCS pressure boundary, the Control Room Operators invoked Technical Specification 3.4.6.2, Action A, which requires that the Unit be placed in HOT STANDBY (MODE 3) within six hours, and in COLD SHUTDOWN (MODE 5) within the following 30 hours. Unit 2 shutdown commenced at 1315 hours. Since the repair to the line was effected in less than 30 hours, it was performed with the plant in HOT SHUTDOWN (MODE 4). The Unit was not actually taken into MODE 5.

II. CAUSE OF EVENT

Metallurgical analysis of the cracked weld concluded that this was a high cycle fatigue failure. Visual examination of the fracture surface under low power magnification revealed faint clamshell marks indicative of fatigue. Vibration measurements were taken on the 22A SIT discharge test connection and compared to the other three SIT test connections in Unit 2. Vibration levels on the cracked line were 2-3 times higher than those on the similar lines. Engineering analysis indicates that, although the design and installation of the test connection fully met applicable code requirements, the configuration introduced a high potential for harmonic oscillation to occur, coinciding with the excitation frequency of the RCS. This would create a higher driving stress sufficient to account for the high cycle fatigue. Although vibration measurements were taken prior to repair of the test connection, these were not sufficient to confirm or deny harmonic oscillation. The vulnerability of the line to harmonic oscillation was not realized until after the measurements had been taken.

The 22A SIT discharge test connection was removed during the Spring 1993 Unit 2 Refueling Outage to allow for the replacement of the 22A SIT Outlet Check Valve. After the check valve was replaced, the line was rewelded in place. The placement of the support was slightly modified (it was within specified tolerances) at this time. It is suspected that these changes resulted in harmonic oscillation causing high cycle fatigue failure.

III. ANALYSIS OF EVENT

There is no safety significance or consequence associated with this event. The plant was shut down safely in accordance with applicable procedures and as a result of leakage found via routine use of appropriate leakage detection techniques, and as required by Technical Specifications. The leakage was not large enough to cause the 22A SIT to be incapable of performing its safety function in response to a large break loss-of-coolant accident. The maximum resultant leakage (0.33 gpm) was well within the allowable safety injection leakage assumed in the small break loss-of-coolant accident analysis. This and the other three SITs on Unit 2 were available to perform their safety function, as assumed in the Calvert Cliffs Accident Analysis, during the time the test connection was leaking.

This item is reportable under the provisions of 10 CFR 50.73(a)(2)(i)(A) as a shutdown required by Technical Specifications.

IV. CORRECTIVE ACTIONS

Short Term:

A. The support for 22A SIT discharge test connection was redesigned and installed to reduce levels of vibration to an acceptable level.

Measurements following the modification confirmed that vibration was reduced and verified not to be in the harmonic region.

B. Walkdowns were performed on the other three SIT discharge test connections to ensure their acceptability. All are sufficient. Measurements on other lines confirmed lower amplitude vibration.

C. The design of the Unit 1 SIT configurations were verified not be prone to the same problem.

D. The circumstances of this event will be reviewed with appropriate Engineering personnel.

Long Term:

E. We will evaluate whether additional generic measures to prevent high frequency fatigue failure are appropriate.

V. ADDITIONAL INFORMATION

A. Affected Component Identification:

Component or System	IEEE 803 EIIS Funct	IEEE 805 System ID
Safety Injection Tank	TK	AB
SIT Vent Valve	VTV	AB
SIT Motor Operated Valve	20	AB
SIT Check Valve	V	AB

B. Previous Similar Events:

There have been 5 similar events reported via Licensee Event Report involving small bore piping/tubing cracked due to high cycle fatigue.

1. LER 318/87-003 reported a crack in a 1/2 inch line to a Low Pressure Safety Injection relief valve. The valve was replaced and subsequently cracked again, as reported in 318/87-004. The cause of the failures is suspected to be torsional/bending loads from a single transient vibration condition.

2. LER 318/87-006 reported a crack in a 1/2 inch Electrohydraulic Control System line due to lack of sufficient support.

3. LER 317/88-009 reported a trip caused by a sheared 1/4 inch instrument air line to the 12 Main Feed Regulating Valve resulting from vibration induced by an improperly located pressure switch.

4. LER 317/90-030 reported cracks found in the SIT nitrogen test connections due to an undetected cantilever condition introduced during replacement of the valves.

None of the above events involved suspected harmonic oscillation.

NRC FORM 366A (5-92)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 5/31/95
LICENSEE EVENT REPORT (LER) TEXT CONTINUATION		ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)	PAGE (3)
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TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

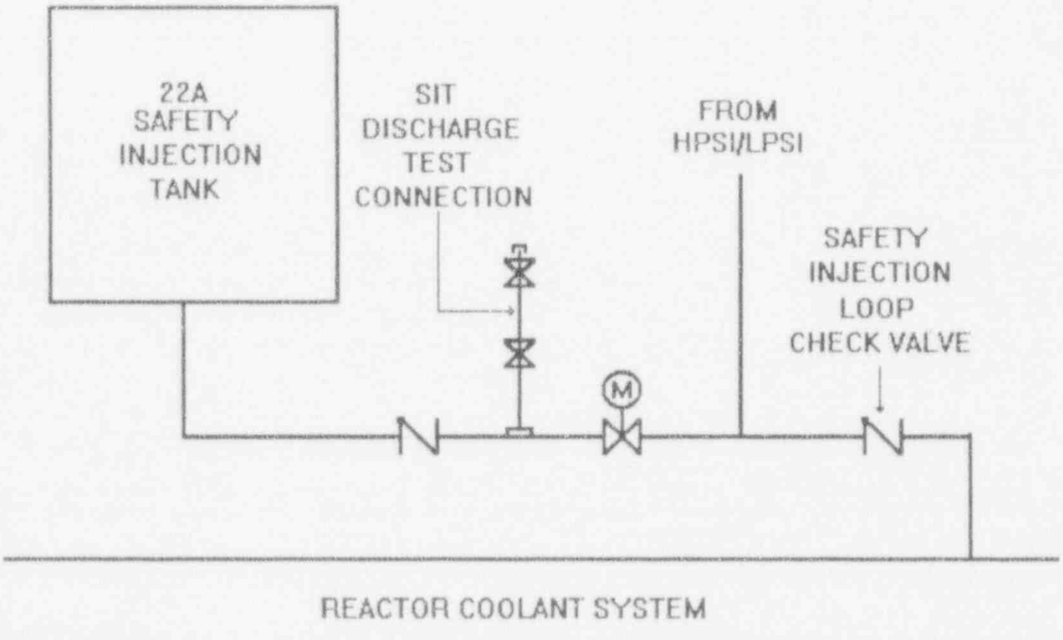

FIGURE 1

22A SAFETY INJECTION TANK DISCHARGE LINE

NRC FORM 366A (5-92) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 5/31/95
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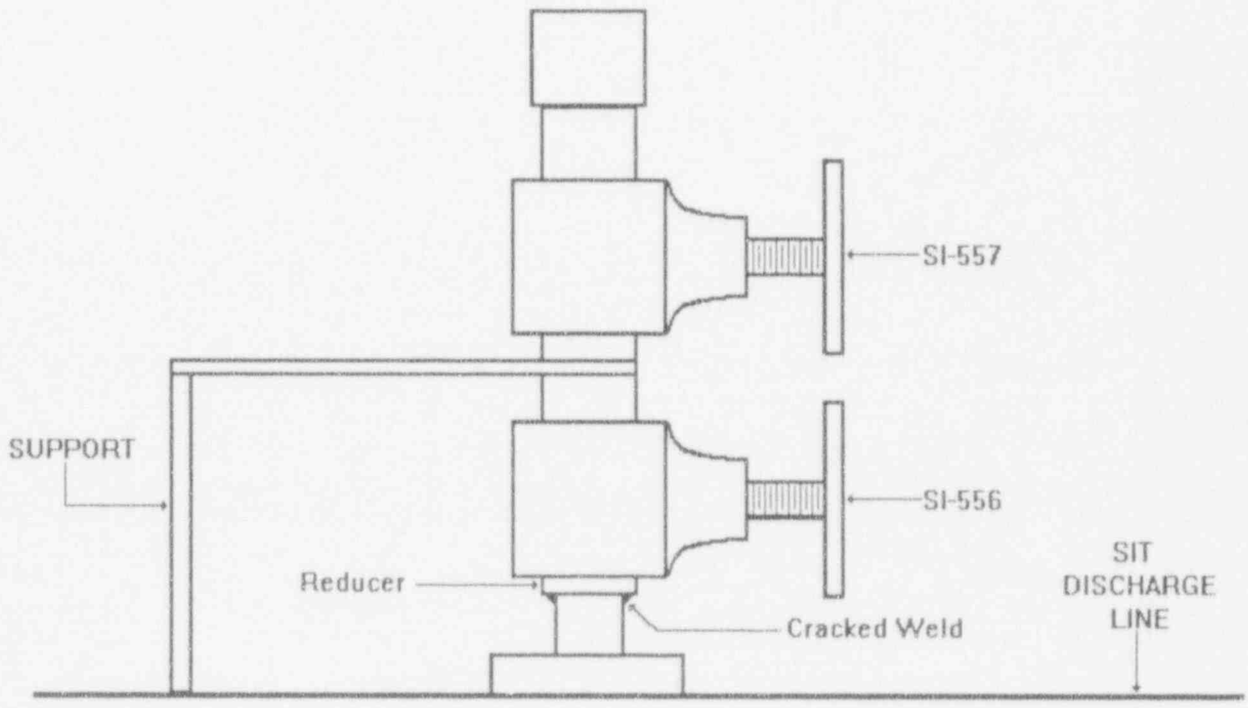

FIGURE 2

22A SAFETY INJECTION TANK DISCHARGE TEST CONNECTION