NRC Form 366
(9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/85

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Calvert Cliffs, Unit 1	0 5 0 0 0 3 1 7	1 OF 0 4

TITLE (4)

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 1	1 6	8 5	8 5	0 0 1	0 0	0 2	0 8	8 5		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1

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)	[ ] 50.73(a)(2)(iv)	[ ] 73.71(b)
[ ] 20.405(a)(1)(i)	[ ] 50.36(c)(1)	[X] 50.73(a)(2)(v)	[ ] 73.71(c)
[ ] 20.405(a)(1)(ii)	[ ] 50.36(c)(2)	[X] 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)	[X] 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	
Brian E. Holian, 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
B	B P	S E A L	A 5 8 5	Y					

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)

At 1416 on January 16, 1985 a ten minute Safety Injection Tank (SIT) check valve inleakage test was completed. Initial results indicated excessive inleakage into two SITs. The results were presented to the Plant Operations and Safety Review Committee (POSRC). It was decided that the high pressure Safety Injection flow rate specified in the Technical Specifications could not be assured. Additionally, under certain circumstances, the SIT inleakage could render the tanks inoperable. Based on this information the POSRC recommended that both Unit 1 High Pressure Safety Injection headers be declared inoperable. Reactor shutdown was completed at 1845.

Two SIT outlet check valves were overhauled on January 17, 1985. Each valve's seating surface o-ring was found approximately one-third degraded. The Ethylene Propylene o-rings had been upgraded previously due to their inability to withstand the temperature environment in which these valves operate. Both o-rings were replaced with a more heat resistant o-ring. Both check valves were subsequently satisfactorily leak tested.

The following corrective actions will be taken as a result of this event: (1) All SIT outlet check valves will be leak tested quarterly. (2) The remaining six SIT outlet check valves will be overhauled during their respective 1985 refueling outage. The Ethylene Propylene o-rings will be replaced by the higher temperature resistant Dupont Kalrez compound. (3) A change to the Technical Specifications justifying a more flexible minimum combined flow rate for the lowest three High Pressure Safety Injection leg flows will continue to be pursued.

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On the afternoon of January 15, 1985, a leak test was performed on Unit 1 and Unit 2 Safety Injection Tank (SIT) (EIIS BP-TK) Outlet Check Valves (EIIS BP-V). This test was performed to obtain background data necessary for a revision to Surveillance Test Procedure (STP) 0-65, "Quarterly Valve Operability Verification". The test consisted of pressurizing the High Pressure Safety Injection (HPSI) (EIIS BQ) Header for ten minutes, simultaneously monitoring SIT inleakage. There are two possible leakage paths. The first path is through two 1" Isolation Valves (EIIS BP-ISV): the SIT "Check Valve Leakage Drain Valve" and the "Fill Valve". The second path is simply reverse flow through the SIT outlet check valve. Unit 2 results were considered negligible whereas Unit 1 results warranted further investigation to more accurately verify and quantify SIT check valve inleakage.

At 1406 on January 16, 1985, with Unit 1 in **MODE** 1, Number 13 High Pressure Safety Injection Pump (EIIS BQ-P) was started commencing a second inleakage test. This test was patterned after the Calvert Cliffs' Operating Instruction for leak testing SIT fill, drain and tank outlet check valves. Prior to starting the HPSI pump, one potential leakage path was isolated by closing the manual isolation valves for the SIT fill header. At 1416 the HPSI pump was secured. A marked rise was noted in #11A SIT during the ten minute pump run. (This identical tank also had the highest indicated inleakage on the previous day's test.) The following rates of volume change were calculated:

SIT	INLEAKAGE (GPM)
11A	27.2
11B	7.6
12A	1.6
12B	2.0

All applicable data was assembled and presented to the Plant Operations and Safety Review Committee (POSRC). The ensuing discussion centered on the safety consequences of this inleakage. Two over-riding concerns dominated the discussion. First, the HPSI flow rate specified in the Technical Specifications could not be assured, thereby potentially worsening the consequences of the limiting small break Loss of Coolant Accident (LOCA). Secondly, during prolonged operation of HPSI, the SIT inleakage could cause the tank's relief valve (EIIS BP-RV) to lift (250 ± 8 psig setpoint), reducing the nitrogen inventory and thereby rendering the tanks inoperable.

Based on this information the POSRC recommended that both Unit 1 HPSI Headers be declared inoperable. At 1600 on January 16, 1985, the Plant Superintendent ordered a Unit 1 shutdown in accordance with the requirements of Technical Specification 3.0.3. Operators commenced reducing reactor power at 1640 and declared an Unusual Event due to Unit 1 shutdown. At 1835 the Main Turbine (EIIS SB-TRB) was taken off-line. Reactor (EIIS AC) shutdown was completed at 1845.

At 0340 on January 17, 1985, Unit 1 entered **MODE** 4, thereby downgrading from the Emergency Action Level "Unusual Event". Preparations were made for draining and venting 11A and 11B SITs. Work packages were prepared for overhauling both of these tanks' outlet check valves. These valves are 12" - 1500#, swinging disc check valves with an inclined, "soft" seat. They provide pressure isolation for the tanks from both primary coolant pressure and HPSI pump discharge pressure. An Ethylene Propylene o-ring (Type E-832-9) (EIIS BP-SEAL) is utilized at the seating surface. This material has been found to deteriorate at a rate greater than that specified by the manufacturer. Three different types of ethylene propylene have been used since initial operation. Although successive o-ring materials have had better temperature and radiation resistant qualities, each type has experienced degradation. A facility modification was approved in December 1983 to allow for the use of Dupont Kalrez 4079 as replacement o-ring material. Kalrez , a perfluoroelastomer, will better withstand the temperature environment in which these valves operate. This modification had been scheduled for completion during 1985 on both units.

Repairs commenced concurrently on both SIT outlet check valves at approximately 1200 on January 17, 1985. The valves were disassembled and valve internals were cleaned and inspected. Approximately one-third of the o-ring seats were found degraded. Both o-rings were replaced with Kalrez Compound 4079. Valve repairs took approximately 20 hours. Both SITs were filled by 1556 on January 18, 1985. Both check valves were subsequently satisfactorily tested for zero leakage. Unit 1 was taken critical at 1825 on January 19, 1985. At 2258 Unit 1 was paralleled to the grid.

The following corrective actions will be taken as a result of this event:

1. All SIT outlet check valves will be leak tested quarterly.

2. The remaining six SIT outlet check valves will be overhauled during their respective 1985 refueling outage. The Ethylene Propylene o-rings will be replaced by the higher temperature resistant Kalrez compound.

3. A change to the Technical Specifications justifying a more flexible minimum combined flow rate for the lowest three High Pressure Safety Injection leg flows will continue to be pursued.

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

ASSESSMENT OF SAFETY CONSEQUENCES AND IMPLICATIONS OF THIS EVENT

Concerning the safety implications of this event, given the limiting small break Loss of Coolant Accident (LOCA), assuming the worst single failure and the lowest combination of three injection leg flows, and crediting no flow from the Chemical Volume and Control System (CVCS) (EIIS CB), the flow reaching the core could have fallen short of the value assumed in the small break LOCA analysis. However, the hot full power moderator temperature coefficient, the peak linear heat rate, and the axial shape index have been significantly less adverse than those assumed in the accident analysis. It is also likely that some CVCS flow would exist. A specific small break LOCA calculation using the most adverse conditions that existed throughout cycle life (in conjunction with modeling the pressure dependent SIT inleakage) might, therefore, show acceptable results for peak clad temperature. Such a calculation was not performed. Finally, NRC test programs have shown that significant conservatism exists in the mandated LOCA methodology.

Besides degraded HPSI flow concerns, the consequences of SIT inoperability must be addressed. SIT inleakage could cause a situation whereby one or more SITs may become inoperable. Substantial inleakage could pressurize a tank sufficiently to lift its relief valve, thereby reducing the nitrogen inventory. This consequence would not present a concern in the large break LOCA analyses where SIT discharge occurs prior to the commencement of HPSI flow. Additionally, our most limiting small break LOCA shows clad temperature peaking prior to SIT discharge. SIT water in this case is not required to limit peak clad temperature, although it may accelerate the rate of subsequent cooldown.

Three additional Safety Implications were considered. The first, the possibility of the overpressurization of a SIT, was discounted due to the capability of the associated relief valve to adequately pass at least twice the amount of this event's measured inleakage. The second, possible HPSI pump run out, is not possible in this case due to the presence of preset throttled HPSI flow valves in the discharge piping of the HPSI pumps. Lastly, the effect of degraded HPSI flow was evaluated for the main steam line break event analysis. In this event steam generator blowdown is complete, and peak reactivity and return to power occur prior to boration from HPSI reaching the core. Therefore, the minor diversion to the SITs has no impact.

An examination of previous LER's dealing with Safety Injection System problems revealed the following similar Events: 82-033, and 78-031.

The contact person for this Event is B. E. Holian (301) 260-4384.

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

February 8, 1985

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

Docket No. 50-317
License No. DPR 53

Dear Sirs:

The attached LER 85-01 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
Plant Superintendent

BEH
LBR/BEH/pah

cc: Dr. Thomas E. Murley
Director, Office of Management Information and Program Control
Messrs: A. E. Lundvall, Jr.
J. A. Tiernan

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