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)
Shoreham Nuclear Power Station Unit #1	0 5 0 0 0 3 2 2	1 OF 0 5

TITLE (4) Results of LLRT of Penetration Showed Leakage that, when Combined with All Type B and C Penetration Leakages, Exceeded the Tech. Spec. Limit of 0.6 La.

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 9	0 3	8 7	8 7	— 0 2 9	— 0 1	1 2	0 4	8 7		0 5 0 0 0

OPERATING MODE (9)		THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)			
4		20.402(b)	20.405(c)	50.73(a)(2)(iv)	73.71(b)
POWER LEVEL (10)		20.405(a)(1)(i)	50.36(c)(1)	50.73(a)(2)(v)	73.71(c)
0 0 0		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)	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
Robert W. Grunseich, Operational Compliance Engineer	AREA CODE 5 1 6 9 2 9 - 8 3 0 0

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 D	I S V	V 0 8 5	No		X	B O	I S O	A 3 9 1	No	
B	B D	I S V	F 1 3 0	No		X	B O	I S V	A 3 9 1	No	

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
X YES (If yes, complete EXPECTED SUBMISSION DATE)	NO		0 4	3 0	8 8

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

On September 8, 1987 at 1330, it was determined by the Maintenance section that the results of a Local Leak Rate Test (LLRT) of check valve 1T48*O1V-0016A (Post Accident Sampling Containment Atmosphere Sample Return), showed leakage that, when combined with all "B" and "C" penetration leakages exceeded the Tech. Spec. limit of 0.6 La. During the LLRT program, the plant was in Operational Condition 4 (Cold Shutdown) with the mode switch in Shutdown and all rods inserted in the core. Subsequent testing identified six additional valves which had excessive leakage. Maintenance Work Requests (MWRs) were generated to investigate the leakage and to repair the valves as needed. The 1T48 valve was repaired and retested with acceptable results. The remaining valves are scheduled to be repaired. A supplemental report will be generated upon completion of the LLRT and will identify all valves which significantly contributed to the combined leakage along with any corrective action taken.

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PDR ADOCK 05000322
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IE22,1

PLANT AND SYSTEM IDENTIFICATION

General Electric - Boiling Water Reactor

Energy Industry Identification System (EIIS) codes are identified in the text as [xx].

IDENTIFICATION OF THE EVENT

Results of LLRT for check valve showed leakage that, when combined with all Type B and C penetration leakage, exceeded the Tech. Spec. limit of 0.6 La.

Event Date: 9/8/87

Report Date: 12/4/87

CONDITIONS PRIOR TO THE EVENT

Operational Condition 4 (Cold Shutdown)

Mode Switch - Shutdown

RPV Pressure = 0 psig RPV Temperature = 101 Degrees F

POWER LEVEL - 0

All rods inserted in the core

DESCRIPTION OF THE EVENT

On September 8, 1987 at 1330, it was determined by the Maintenance section that the results of a Local Leak Rate Test (LLRT) of check valve 1T48*01V-0016A [BD] (Post Accident Sampling Containment Atmosphere Sample Return), showed leakage that, when combined with all "B" and "C" valve leakages exceeded the Tech. Spec. limit of 0.6 La. The penetration was tested on September 3, 1987 to satisfy Tech. Spec. requirement 3.6.1.2.b. Upon identification of the

excessive leakage, a Maintenance Work Request was generated to investigate the cause and subsequent repair of the valve. Subsequent testing identified six additional valves which had excessive leakage. They are;

1T46*AOV-039D [BD] (Purge Air from Suppression Pool Isolation Valve)

1B21*18V-1103A [SJ] ("A" Inboard Feed Water Check Valve)

1E11*01V-047 [BO] (PASS Sample Return Check Valve)

1E11*MOV-031A [BO] (RHR "A" Pump Suppression Pool Suction Valve)

1E11*AOV-081A [BO] (RHR LPCI "A" to Recirc. Syst. Valve)

1E11*MOV-042A [BO] (RHR Containment Spray Valve)

A supplemental report will be issued upon completion of the LLRT to identify all valves which significantly contributed to the combined leakage.

CAUSE OF THE EVENT

The cause of the leakage for the 1T48 valve was due to normal valve degradation (excessive valve disc to seat leakage). The 1T46 valve had oxidation products lodged in the seat of the valve, preventing a tight seal. The cause for the leakage for the remaining five valves has not yet been determined. This will be discussed upon completion of the LLRT in the supplemental report.

ANALYSIS OF THE EVENT

This event is reportable per 10CFR50.73(a)(2)(ii) as identified in NUREG 1022, page 12. There is minimal safety significance to this event. All seven valves are set up in a double valve configuration and their adjacent valves leak tested satisfactorily, and would have provided a means for containment isolation upon an accident signal. Had a LOCA occurred during 5% power operation with the leakage rate above the allowable rate, the radiation doses would have been well below the limits as required by 10CFR100 and the General Design Criterion 19 of Appendix A of 10CFR50.

CORRECTIVE ACTIONS

MWRs were generated for all seven valves. The T48 valve was repaired and retested with acceptable results. The remaining valves are scheduled to be repaired and will be retested upon completion of the repairs. The supplemental report will identify those actions taken to repair the valves along with corrective actions taken for any additional valves which significantly contributed to the combined leakage.

ADDITIONAL INFORMATION

a. Manufacturer and model number of failed component (s)

Valve No.	Manufacturer	Model No.
1T48*01V-0016A	Velan Valve Co.	299-1
1T46*AOV-039D	Fisher Controls Co.	9220
1B21*18V-1103A	Anchor-Darling Valve Co.	E5836-14-2
1E11*01V-047	Velan Valve Co.	3026X
1E11*MOV-031A	Anchor-Darling Valve Co.	E5836-1-1
1E11*AOV-081A	Velan Valve Co.	B24-7114-255P
1E11*MOV-042A	Anchor-Darling Valve Co.	E5836-4-1

b. LER numbers of previous similar events
85-049

NRC Form 366B
(9-83)

LICENSEE EVENT REPORT (LER) FAILURE CONTINUATION

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FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Shoreham Nuclear Power Station	0 5 0 0 0 3 2 2	8 7	0 2 9	0 1	0 5 OF 0 5

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	B O	I S V	V 0 8 5	No							
X	B O	I S V	V 0 8 5	No							
X	S J	I S V	A 3 9 1	No							

NRC FORM 366B
(9-83)

LONG ISLAND LIGHTING COMPANY

SHOREHAM NUCLEAR POWER STATION • P.O. BOX 628 • WADING RIVER, NEW YORK 11792

TEL. (516) 929-8300

December 4, 1987 PM-87-293

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

Dear Sir:

In accordance with 10CFR50.73, enclosed is a revision to Shoreham Nuclear Power Station's Licensee Event Report LER 87-029 Rev. 1. This revision is being submitted to identify additional containment isolation valves which significantly contributed to the combined leakage during their respective LLRTs

Sincerely yours,

William E. Steiger, Jr.
Plant Manager

WES/pz

Enclosure

cc: William T. Russell, Regional Administrator
Frank Crescenzo, Resident Inspector
Institute of Nuclear Power Operations, Records Center
American Nuclear Insurers

SR.A21.0200