NRC FORM 366
(7-77)

U. S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT

CONTROL BLOCK: | | | | | | | (1) (PLEASE PRINT OR TYPE ALL REQUIRED INFORMATION)

0 1	LICENSEE CODE (2)	M A P P S 1
	LICENSE NUMBER (3)	0 0 - 0 0 0 0 0 - 0 0
	LICENSE TYPE (4)	4 1 1 1 1
	CAT (5)	
CON'T 0 1	REPORT SOURCE (6)	L
	DOCKET NUMBER (7)	0 5 0 - 0 2 9 3
	EVENT DATE (8)	0 7 1 3 8 1
	REPORT DATE (9)	0 9 0 3 8 1

EVENT DESCRIPTION AND PROBABLE CONSEQUENCES (10)

0 2 On July 13, 1981 during reactor startup, while conducting surveillance procedure

0 3 No. 8.7.1.7, the outer drywell personnel access door was discovered to have a leak

0 4 rate in excess of T.S. limits. The inner door seal had been proven to be intact.

0 5 The seal was replaced in-kind and a satisfactory leak rate test conducted at 1335

0 6 hrs. the same day. See Attachment

0 7

0 8

0 9	SYSTEM CODE (11)	S A
	CAUSE CODE (12)	E
	CAUSE SUBCODE (13)	B
	COMPONENT CODE (14)	V E S S E L
	COMP. SUBCODE (15)	C
	VALVE SUBCODE (16)	Z
(17) LER/RO REPORT NUMBER	EVENT YEAR	8 1
		—
	SEQUENTIAL REPORT NO.	0 4 3
		/
	OCCURRENCE CODE	0 1
	REPORT TYPE	T
		—
	REVISION NO.	0
	ACTION TAKEN (18)	A
	FUTURE ACTION (19)	G
	EFFECT ON PLANT (20)	Z
	SHUTDOWN METHOD (21)	Z
	HOURS (22)	0 0 0 0
	ATTACHMENT SUBMITTED (23)	Y
	NPRD-4 FORM SUB. (24)	N
	PRIME COMP. SUPPLIER (25)	N
	COMPONENT MANUFACTURER (26)	C 3 1 0

CAUSE DESCRIPTION AND CORRECTIVE ACTIONS (27)

1 0 The cause of the seal leakage has been determined to be normal in-service seal

1 1 degradation. A change to procedure 2.1.1, which requires a local leak rate test prior

1 2 to Reactor Coolant Temperature exceeding 212°F. has been made to preclude recurrence

1 3 of this event.

1 4

1 5	FACILITY STATUS (28)	X
	% POWER (29)	0 0 1
	OTHER STATUS (30)	Startup
	METHOD OF DISCOVERY (31)	B
	DISCOVERY DESCRIPTION (32)	Surveillance Test
1 6	ACTIVITY RELEASED (33)	Z
	CONTENT OF RELEASE (34)	Z
	AMOUNT OF ACTIVITY (35)	N.A.
	LOCATION OF RELEASE (36)	N.A.
1 7	PERSONNEL EXPOSURES NUMBER (37)	0 0 0
	TYPE (38)	Z
	DESCRIPTION (39)	N.A.
1 8	PERSONNEL INJURIES NUMBER (40)	0 0 0
	DESCRIPTION (41)	N.A.
1 9	LOSS OF OR DAMAGE TO FACILITY TYPE (42)	Z
	DESCRIPTION (43)	N.A.
2 0	PUBLICITY ISSUED (44)	N
	DESCRIPTION (45)	N.A.
	NRC USE ONLY	

8109150515 810903
PDR ADOCK 05000293
S PDR

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BOSTON EDISON COMPANY
PILGRIM NUCLEAR POWER STATION
DOCKET NO. 50-293

Attachment to LER 81-043/01T-0

On July 15, 1981, the Operations Review Committee (ORC) determined that degradation of the outer drywell access door seal was reportable per 10 CFR 50 Appendix J.

This determination was based on available information. However, the ORC requested the On-site Safety Group Leader to prepare a detailed report of the criteria for testing the drywell door seals. On August 19, 1981 the ORC reviewed this report and determined an immediate event report be issued since operating procedure 2.1.1 permitted (and operating personnel performed) a drywell entrance for a routine, scheduled inspection after reactor coolant temperature exceeded 212° F and before a satisfactory surveillance test of the drywell access door seals. To preclude a recurrence of this event, procedure 2.1.1 has been revised to require a local leak rate test of the door seals prior to a coolant temperature in excess of 212°F as well as after the required drywell inspection.