PECO NUCLEAR
A UNIT OF PECO ENERGY

PECO Energy Company
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10CFR50.73

September 15, 2000

Docket Nos. 50-352
50-353
License No. NPF-39
NPF-85

Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

SUBJECT: Licensee Event Report
Limerick Generating Station - Unit 1 and Unit 2

This LER concerns a condition outside the design basis involving a suppression chamber steam bypass leakage path.

Reference:	Docket Nos. 50-352
	50-353
Report Number:	1-97-010
Revision Number:	01
Event Date:	October 26, 1984
	August 12, 1989
Discovery Date:	October 20, 1997
Report Date:	November 19, 1997
Facility:	Limerick Generating Station
	Box 2300, Sanatoga, PA 19464-2300

This LER is being submitted pursuant to the requirements of 10CFR50.73(a)(2)(ii).

Very truly yours,

Robert C. Braun, Plant Manager, Limerick Generating Station

H. J. Miller, Administrator Region I, USNRC
A. L. Burritt, USNRC Senior Resident Inspector, LGS

IE22

NRC FORM 366 U.S. NUCLEAR REGULATORY COMMISSION
(6-1998)

LICENSEE EVENT REPORT (LER)

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

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Limerick Generating Station, Units 1 and 2	05000352/05000353	1 OF 3

TITLE (4)
Potential Containment Bypass Path Resulting in a Condition Outside of the Design Bases

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 NAME	DOCKET NUMBER
										05000
10	26	1984	1997	-- 010 --	01	11	19	1997	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9)	1
POWER LEVEL (10)	100

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

	20.2201(b)		20.2203(a)(2)(v)		50.73(a)(2)(i)		50.73(a)(2)(viii)
	20.2203(a)(1)		20.2203(a)(3)(i)	X	50.73(a)(2)(ii)		50.73(a)(2)(x)
	20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
	20.2203(a)(2)(ii)		20.2203(a)(4)		50.73(a)(2)(iv)		OTHER
	20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)		Specify in Abstract below or in NRC Form 366A
	20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
K. W. Gallogly, Manager - Experience Assessment	(610) 718-3400

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
☐ YES (If yes, complete EXPECTED SUBMISSION DATE).	☒ NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On 08/21/97, Engineering identified a potential suppression chamber steam bypass leakage path between the Drywell (DW) and the Suppression Pool (SP) air spaces. On 10/20/97, following further review of the Updated Final Safety Analysis Report, source documents and a 10CFR Part 21 Notification from the Nuclear Steam System Supplier (General Electric), Engineering concluded that a raceway or internal control cabinet failure common to the Primary Containment Isolation Valves (PCIVs) located on each nitrogen (N2) purge supply line could result in the opening of both PCIVs, thus connecting the DW and SP air spaces. This failure concurrent with a LOCA could result in primary containment pressure exceeding the analyzed DW and SP pressures. The cause of the event was a failure to adequately specify applicable design requirements for lines which connect the DW and SP air spaces. Upon discovery of this condition, one of the PCIVs was disabled to eliminate the potential for bypass leakage and procedures were revised to reflect this configuration change. Engineering is participating in the industry review of this issue. This event had no actual safety consequences and there were no previous similar events identified.

Unit Conditions at Time of Discovery

Unit 1 and Unit 2 were in the "RUN" mode at 100 percent of thermal reactor (EIIS:EA) power. There were no systems, structures or components that were inoperable that contributed to the event.

Description of the Event

On August 21, 1997, during a review of industry operating experience, engineering personnel identified a potential suppression chamber steam bypass leakage path between the air spaces of the Drywell and the Suppression Pool. The path consists of a six inch nitrogen supply line that cross connects the twenty-four (24) inch Drywell and Suppression Pool purge supply lines. The six (6) inch nitrogen supply line has one inboard Primary Containment Isolation Valve (PCIV) at the connection to each of the two purge lines. The cabling and controls for these two inboard PCIVs are located in a common control cabinet and in common raceways and are not independent. A postulated raceway or internal control cabinet failure could result in the simultaneous opening of both inboard PCIVs, connecting the Drywell and Suppression Pool air spaces. Separation/isolation of the Drywell airspace from the suppression chamber airspace is required to ensure proper pressure-suppression function of the Suppression Pool. Engineering personnel initiated a further review of this configuration to determine if this condition was previously analyzed.

On October 20, 1997, following a review of the Updated Final Safety Analysis Report (UFSAR), related source documents and a 10CFR Part 21 Notification from the Nuclear Steam System Supplier (NSSS) (General Electric), station and engineering personnel concluded;

1. The pressure suppression function of the suppression chamber is required to meet single failure criteria,

2. The use of containment sprays can be utilized to meet this requirement if adequate capability is shown,

3. The isolation of all potential communication paths between the Drywell airspace and suppression chamber airspace must meet single failure criteria or the potential bypass leakage analyzed, and

4. Since the circuitry for the two inboard PCIV's is not independent, a postulated raceway or internal control cabinet failure could result in the simultaneous opening of the two inboard PCIV's.

If this failure were to occur during a LOCA, then the potential exists for the Primary Containment air space to exceed the Drywell and Suppression Pool design pressure of 55 psig. At 1806 hours on October 21, 1997, a one hour notification to the NRC was made since it was concluded that the plant was in a condition outside the design basis. This LER is being submitted in accordance with 10 CFR 50.73(a)(2)(ii).

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Limerick Generating Station Units 1 and 2	05000 -352/-353	1997	-- 010	-- 01	3 OF 3

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

Cause of the Event

The cause of the event was an original design deficiency in that the design requirements for lines which connect the Drywell and Suppression Pool air spaces were not clearly defined. Single failure criteria or electrical independence requirements were not originally applied to the inboard nitrogen supply PCIVs with respect to each other, nor was the potential for inadvertent opening of both valves analyzed. The original design correctly applied the single failure criteria to the inboard and outboard PCIVs, however.

Analysis of Event

No actual safety consequences occurred as a result of this event. No LOCA occurred while the valves were exposed to the potential failure and the two PCIVs were not open simultaneously with the reactor critical or in hot shutdown. When the valves are in the normally closed position, a failure in the common raceway or internal to the common control cabinet could result in energizing the air operated solenoid valves which would cause the PCIVs to open. In the unlikely event of a LOCA coincident with this failure, the postulated inadvertent opening of both valves would have the potential for the design pressures of the Suppression Pool or the Drywell to be exceeded.

Corrective Actions

On August 21, 1997, one of the two air operated solenoid valves was disabled on the Unit 1 and Unit 2 six (6) inch purge valves. An administrative clearance was applied and procedures were revised to reflect this configuration change. Engineering personnel are participating with the industry review of this issue to determine if additional review and corrective actions are required.

| On September 7, 2000 a plant modification was completed on both units that prevents opening both
| nitrogen |purge valves at the same time. An electrical interlock was installed to prevent opening the valves
| if the companion valve is open. The valves with modified control circuits are HV-057-121, HV-057-131,
| HV-057-221 and HV-057-231.

Previous Similar Events

None