(7-77)

LICENSEE EVENT REPORT

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

0 1 | I L L S C 1 (2) LICENSEE CODE | 0 0 - 0 0 0 0 0 - 0 0 (3) LICENSE NUMBER | 4 1 0 0 0 (4) LICENSE TYPE | CAT (5)

CON'T 0 1 | REPORT SOURCE L (6) | 0 5 0 0 0 3 7 3 (7) DOCKET NUMBER | 0 7 1 8 8 3 (8) EVENT DATE | 0 8 1 2 8 3 (9) REPORT DATE

EVENT DESCRIPTION AND PROBABLE CONSEQUENCES (10)

0 2 On July 18, 1983 at 1137 hours a cooldown rate in excess of 100°F/hr was observed when
0 3 a Full Core SCRAM occurred as a result of a bypass valve transient during EHC
0 4 troubleshooting. T.S. 3.4.6.1.a was violated. Engr. evaluation was performed by
0 5 G.E. San Jose and CECO Engr. They concluded that no structural effects on the vessel
0 6 had occurred.
0 7
0 8

0 9	SYSTEM CODE	CAUSE CODE	CAUSE SUBCODE	COMPONENT CODE	COMP. SUBCODE	VALVE SUBCODE
	R A (11)	A (12)	C (13)	Z Z Z Z Z Z (14)	Z (15)	Z (16)

(17) LER/RO REPORT NUMBER

EVENT YEAR		SEQUENTIAL REPORT NO.		OCCURRENCE CODE	REPORT TYPE		REVISION NO.
8 3	—	0 8 5	/	0 3	L	—	0

ACTION TAKEN	FUTURE ACTION	EFFECT ON PLANT	SHUTDOWN METHOD	HOURS (22)	ATTACHMENT SUBMITTED	NPRD-4 FORM SUB.	PRIME COMP. SUPPLIER	COMPONENT MANUFACTURER
X (18)	H (19)	Z (20)	Z (21)	0 0 0 0	Y (23)	N (24)	Z (25)	Z 9 9 9 (26)

CAUSE DESCRIPTION AND CORRECTIVE ACTIONS (27)

1 0 Excessive cooldown was a result of rapid depressurization, full core SCRAM and HPCS
1 1 & RCIC injection. Operations recovered from the transient and restored Unit para-
1 2 meters to normal. Corrective actions to the transient are addressed in DVR 1-1-83-270
1 3
1 4

1 5	FACILITY STATUS	% POWER	OTHER STATUS (30)	METHOD OF DISCOVERY	DISCOVERY DESCRIPTION (32)
	B (28)	0 0 0 (29)	Z	A (31)	Observation

1 6	ACTIVITY RELEASED	CONTENT OF RELEASE	AMOUNT OF ACTIVITY (35)	LOCATION OF RELEASE (36)
	Z (33)	Z (34)	NA	NA

PERSONNEL EXPOSURES

1 7	NUMBER	TYPE	DESCRIPTION (39)
	0 0 0 (37)	Z (38)	NA

PERSONNEL INJURIES

1 8	NUMBER	DESCRIPTION (41)
	0 0 0 (40)	NA

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LOSS OF OR DAMAGE TO FACILITY (43)

1 9	TYPE	DESCRIPTION
	Z (42)	NA

PUBLICITY

2 0	ISSUED	DESCRIPTION (45)	NRC USE ONLY
	N (44)	NA	

NAME OF PREPARER D. R. Winterhoff PHONE: 815/357-6761

8308220240 830812
PDR ADOCK 05000373
S PDR

GPO 917-828

I. LER NUMBER: 83-085/03L-0

II. LASALLE COUNTY STATION: Unit 1

III. DOCKET NUMBER: 050-373

IV. EVENT DESCRIPTION:

Technical Specifications 3.4.6.1.a states that the Reactor Coolant temperature shall not exceed a maximum cooldown rate of less than or equal to 100°F per hour. On July 18, 1983 at 1137 hours a cooldown rate in excess of 100°F/hr (113°F/hr) was observed when a full core scram occurred as a result of a bypass valve transient during EHC troubleshooting.

V. PROBABLE CONSEQUENCES OF THE OCCURRENCE:

To satisfy the action statement of Technical Specification 3.4.6.1, an engineering evaluation was conducted by CECO engineering and General Electric in San Jose. They were informed of the occurrence and supplied with the following data:

1. Cooldown rate as calculated from steam space pressure and temperature of approximately 113°F/hour.

2. Maximum metal temperature change of 10°F during transient.

3. Rx vessel level remained below shell flange.

It was concluded by the engineering evaluation that no structural effects on the reactor vessel had occurred, and that no nil ductility limits had been approached. These conclusions were reached based on the following facts:

1. Rx pressure vessel flange bolts were the limiting component.

2. Based on events, this occurrence was less severe than a normal shutdown during which water quenching of the flange occurs due to flooding.

3. Total fatigue usage for the limiting component was not affected as determined by the General Electric analysis.

VI. CAUSE:

LaSalle Unit 1 scrammed on low reactor water level, +12.5", as a result of a bypass valve transient during EHC troubleshooting. The excessive vessel cooldown addressed in this report was a consequence of rapid vessel depressurization, full core SCRAM and HPCS & RCIC injection. A detailed description of those events are discussed in DVR 1-1-83-270.

VII. CORRECTIVE ACTION:

Immediate response by the Operations Department was to return unit parameters to normal and to recover from the transient. An engineering evaluation was

VII. CORRECTIVE ACTION (Cont'd):

performed as outlined in Section V of this report to determine the impact of the event.

Corrective actions to the cause of the transient and the resultant SCRAM are addressed in DVR 1-1-83-270.

Prepared by: D. R. Winterhoff

Commonwealth Edison
LaSalle County Nuclear Station
Rural Route #1, Box 220
Marseilles, Illinois 61341
Telephone 815/357-6761

August 12, 1983

James G. Keppler
Regional Administrator
Region III
U.S. Nuclear Regulatory Commission
799 Roosevelt Road
Glen Ellyn, IL 60137

Dear Sir:

Reportable Occurence Report #83-085/03L-0 Docket #050-373 is being submitted to your office in accordance with LaSalle County Nuclear Power Station Technical Specification 6.6.B.2.(b), conditions leading to operation in a degraded mode permitted by a limiting condition for operation or plant shutdown required by a limiting condition for operation.

C E Sargent
for G. J. Diederich
Superintendent
LaSalle County Station

GJD/GW/rg

Enclosure

cc: Director of Inspection & Enforcement
Director of Management Information & Program Control
U.S. NRC Document Management Branch
INPO-Records Center
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