7-77)

LICENSEE EVENT REPORT

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

0	1	I L L S C 1 (2)	0 0 - 0 0 0 0 0 - 0 0 (3)	4 1 0 0 0 (4)		(5)

LICENSEE CODE 9–14; LICENSE NUMBER 15–25; LICENSE TYPE 26–30; CAT 57–58

CON'T

0	1	REPORT SOURCE L (6)	0 5 0 0 0 3 7 3 (7)	0 6 0 1 8 3 (8)	0 6 2 9 8 3 (9)

REPORT SOURCE 60; DOCKET NUMBER 61–68; EVENT DATE 69–74; REPORT DATE 75–80

EVENT DESCRIPTION AND PROBABLE CONSEQUENCES (10)

0 2 Following a Reactor scram, which occurred on 6/1/83 at 0230 hrs., the operator decreased

0 3 reactor pressure to about 250 psig to allow condensate makeup to the reactor to correct

0 4 the low RPV level. The moderator temperature decreased from 536°F to 418°F in one hr.

0 5 Since the limiting components in the stress report are the reactor vessel flange bolts

0 6 G.E. reviewed the event and determined that the metal temperature had not decreased

0 7 significantly during the initial cooldown, typical of a normal shutdown.

0 8

	SYSTEM CODE	CAUSE CODE	CAUSE SUBCODE	COMPONENT CODE	COMP. SUBCODE	VALVE SUBCODE
0 9	R A (11)	A (12)	A (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 5 7	/	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
Z (18)	Z (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 The reactor scrammed on low reactor vessel level when the TDRFP's and the CB pumps

1 1 tripped. The operator used the main turbine bypass valves to relieve pressure and

1 2 allow makeup via the condensate pumps. Vessel pressure dropped past the target

1 3 pressure of 600 psig due to steam supply to Off Gas. G.E.'s evaluation determined

1 4 the event had negligible effect on the Rx vessel.

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

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

PERSONNEL EXPOSURES

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

PERSONNEL INJURIES

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

LOSS OF OR DAMAGE TO FACILITY (43)

	TYPE	DESCRIPTION
1 9	Z (42)	NA

8307060054 830629
PDR ADOCK 05000373
S PDR

PUBLICITY

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

NAME OF PREPARER Kermit Wittenberg PHONE: (815) 357-6761

GPO 817-920

Commonwealth Edison
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Marseilles, Illinois 61341
Telephone 815/357-6761

June 29, 1983

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

Dear Sir:

Reportable Occurrence Report #83-057/03L-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.

G.J. Diederich

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
File/NRC

JUN 30 1983

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I. LER NUMBER: 83-057/03L-0

II. LASALLE COUNTY STATION: Unit 1

III. DOCKET NUMBER: 050-373

IV. EVENT DESCRIPTION:

Following a Reactor Scram on low RPV level, which occurred on June 1, 1983 at 0230 hours, the operator decreased reactor pressure to about 600 psig to allow condensate makeup to the reactor as neither the Reactor Core Isolation Cooling system nor the Motor Driven Reactor Feed Pump were available. The reactor level rose from about -33 inches as indicated on the wide range RPV level recorder, 1B21-R884, to almost 46 inches, per the upset level recorder 1C34-R608. The moderator temperature decreased from about 536°F to 418°F in one hour, measured at the Rx Recirc. pump suction. This exceeds Technical Specification 3.4.6.1.6 which sets a maximum cooldown limit of 100°F in one hour.

V. PROBABLE CONSEQUENCES OF THE OCCURRENCE:

General Electric reviewed the Combustion Engineering stress report for the Reactor Pressure Vessel which shows that the reactor vessel flange bolts are the limiting component based on fatigue usage factor calculated in the report. Based on the information that the water level never reached the vessel flange and the saturated steam temperature taken from Attachment A, which shows reactor pressure during this event, it was concluded that this transient was less severe than a regular shutdown with vessel flooding event in which water quenching of the reactor pressure vessel flange occurs.

After the initial temperature drop to 418°F, the water temperature stabilized. indicating that reactor metal temperature had not decreased significantly during the initial cool down. It was also observed that the reactor vessel upper flange metal temperature exhibited only a gradual decrease typical of a normal shut down, during this transient.

VI. CAUSE:

The reactor scrammed on low reactor vessel level when the Turbine Driven Reactor Feed Pumps and the Condensate Booster Pumps tripped. As the Motor Driven Reactor Feed Pump and the Reactor Core Isolation Cooling System were not available, the operator used the Main Turbine Bypass Valves to relieve pressure in the vessel. This allowed makeup to the vessel through the feed water system from the condensate/condensate booster. Vessel pressure continued to drop past the target pressure of 600 psig due to steam supply to the Steam Seal and Off Gas Systems. This pressure drop to approximately 250 psig caused the Reactor Recirculation Pumps suction temperature to cool from 536°F to 418°F in 1 hour. Thereafter reactor level and temperature were maintained at a more stable condition.

LER NUMBER: 83-057/03L-0

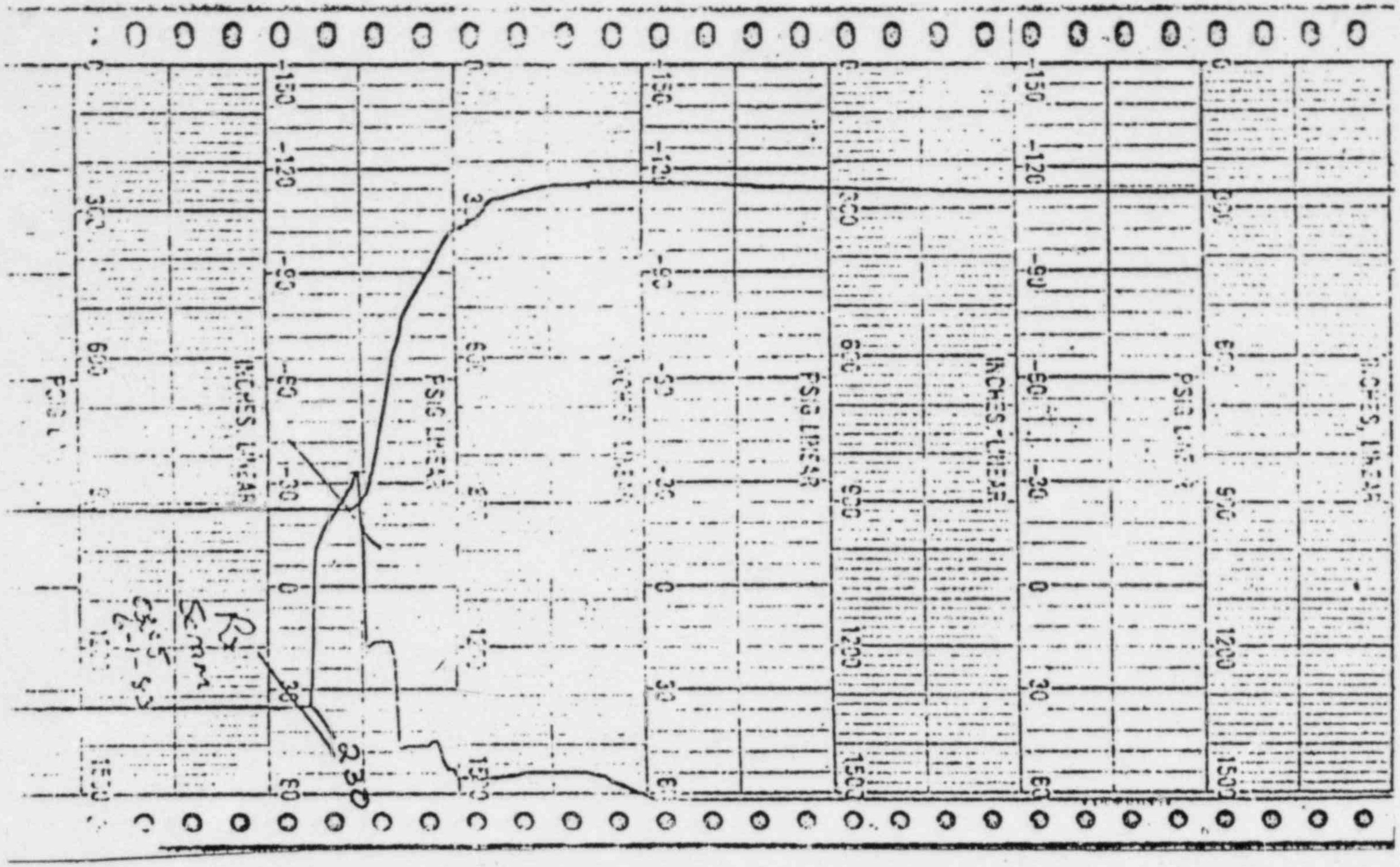
VII. CORRECTIVE ACTION:

General Electric performed an engineering evaluation to determine the effects on the structural integrity of the reactor coolant system. This evaluation determined that the event had negligible effect on the reactor vessel.

AIR 1-83-88 has been written to determine if a modification to the plant is necessary to prevent this from occurring again.

Prepared by: Kermit C. Wittenburg/John Damron

ATTACHMENT A



1B21-R884 Wide Range Reactor Pressure and Level