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)
LaSalle County Station Unit 1	0 5 0 0 0 3 7 3	1 OF 0 6

TITLE (4)
Unit Scram

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 2	0 8	8 5	8 5	— 0 1 7	— 0 0	0 3	0 7	8 5	NA	0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9)	1	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)									
POWER LEVEL (10)	0 7 5		20.402(b)		20.405(c)	X	50.73(a)(2)(iv)		73.71(b)		
			20.405(a)(1)(i)		50.36(c)(1)	X	50.73(a)(2)(v)		73.71(c)		
			20.405(a)(1)(ii)		50.36(c)(2)		50.73(a)(2)(vii)	X	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)		50.73(a)(2)(ii)		50.73(a)(2)(viii)(B)		Special		
			20.405(a)(1)(v)		50.73(a)(2)(iii)		50.73(a)(2)(x)				

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER
	AREA CODE
R. D. Koenig, extension 575	8 1 5 3 5 7 - 6 7 6 1

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 N	T R B	T 1 4 7	Y		X	A D	P S	B 0 7 0	Y	
X	B O	B K R	K 1 8 2	Y		X	V A	D M P	T 0 7 4	N	

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

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

On 2/8/85, at 1018 hours, with Unit 1 at 75% power the reactor scrammed on low reactor vessel level.

A low feedwater pump suction pressure signal tripped the A and B turbine driven reactor feedwater pumps. An unsuccessful attempt was made to restart the A TDRFP. When vessel level reached about 12.5 inches, the reactor scrammed and the recirculation (AD) pumps shifted to low speed. When the vessel level reached about -50 inches, the High Pressure Core Spray system initiated, and the recirculation pumps tripped as required. The Reactor Core Isolation Cooling system also initiated but tripped due to overspeed.

The main steam isolation valves closed and other isolations occurred. Safety valve E opened to relieve vessel pressure at 1125 psig. Vessel level dropped to approximately -60 inches before it was restored to normal.

The primary cause of the unit scram was placing a jumper in the feedwater pump NPSH trip circuit which did not serve its intended function. The jumper was installed utilizing a revision of the print which did not reflect the present system configuration. The circuit had been modified, Mod #1-1-82-293, during the September, 1984, outage and the Drawing Change Request had not yet been processed.

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NRC Form 366B
(9-83)

LICENSEE EVENT REPORT (LER) FAILURE CONTINUATION

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

CAUSE	SYSTEM	COMPONENT	MANUFAC TURER	REPORTABLE TO NPRDS		CAUSE	SYSTEM	COMPONENT	MANUFAC TURER	REPORTABLE TO NPRDS	
X	S J	R T V	A 4 1 5	N							

NRC FORM 366B
(9-83)

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
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TEXT (If more space is required, use additional NRC Form 366A's) (17)

I. EVENT DESCRIPTION

On February 8, 1985, at 1018 hours, the Unit 1 reactor scrammed (JM) on low reactor vessel level.

At approximately 0600, 2/8/85, the Operating Department placed a jumper in the feedwater (JK) NPSH trip logic to prevent the tripping of the feedwater (SJ, FW) pumps while the Maintenance Department repaired a steam leak in the instrument root valve of the FW NPSH pressure switch. The jumper was placed in accordance with Station Administrative Procedure, LAP-240-6, Temporary System Changes.

At 1018 the pressure switch was valved out for maintenance causing a low suction pressure signal to be transmitted to the running feed pumps which resulted in the tripping of the 1A and 1B Turbine Driven Reactor Feed Pumps.

The following sequence of events was identified after reviewing the sequence of events print out and a review of the wide range level indication and pressure indication.

10:18:24 Net positive suction head to feedpumps low.
10:18:28 1A & 1B feedwater pump trip causing loss of feedwater.
10:18:35 The feedwater control circuit alarms level 4 (31.5 inches).
10:18:41 Division 2 Reactor Level Low Alarm received.
10:18:44 Division 1 Reactor Level Low Alarm received.
10:18:45 Automatic reactor scram on low reactor water level of 12.5 inches.
10:18:48 Reactor water level reaches level 2 (-50 inches) and High Pressure Core Spray (HPCS, BG) actuates, Reactor Core Isolation Cooling (RCIC, BN) starts and trips on overspeed, Division 3 diesel generator (EK) starts, reactor recirculation (AD) pumps trip off, the Main Steam Isolation Valves (MSIV, SB) close on Group 1 isolation signal and Reactor Building ventilation (VA) for both units isolates on Group 4. In addition Groups 2, 3 and 5 isolated at this level. Review of the wide range level indication showed the reactor vessel water level reached -60 inches before the level transient turned.

10:21:29 RCIC was reset and manually restarted for level and pressure control.
10:22:50 to 10:25 Automatic Depressurization System (ADS, AD) valve 1B21-N013E opened, at approximately 1125 psi, and closed twice. Normal lift point is 1175 psi.
10:25:26 Reactor water level reached level 8 (55.5")
10:35:35 The Reactor Building ventilation was unisolated and restarted.
10:35:39 Scram Discharge Volume (AA) Hi Level trip bypassed and the scram was reset.

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TEXT (If more space is required, use additional NRC Form 366A's) (17)

I. EVENT DESCRIPTION (CONTINUED)

Unit 1 was in Condition 1 (Run) at approximately 900 MWe. Maintenance activities were in progress to repair a leak on a valve in the feedwater pump NPSH trip switch. The Operating Department had earlier placed a jumper in the NPSH trip circuit to prevent the tripping of the feedwater pumps due to the maintenance activity. The Motor Driven Reactor Feed Pump was out of service for maintenance and unavailable for service. The process computer was unavailable due to computer maintenance activities.

The following equipment failures occurred during the event:

1. RCIC tripped on overspeed auto initiation; the system was reset and manually started later in the transient. The oil in RCIC was changed prior to the unit startup.

2. Upon return of the computer it was noted that the 1E12-F040A valve had tripped in the closed position. The valve was closed prior to the isolation signal.

3. Upon reset of the Group 4 isolation, VR damper 1VR04YA required three attempts to obtain open indication.

4. Safety relief valve 1B21-F013E opened before reactor pressure reached the required setpoint of 1086 psi.

II. CAUSE

The primary cause of the unit scram was placing a jumper in the feedwater pump NPSH trip circuit which did not serve its intended function. The jumper was installed utilizing a revision of the print which did not reflect the present system configuration. The feedwater pump NPSH trip circuit had been modified (Modification 1-1-82-293) during the September, 1984, outage and the Drawing Change Request had not yet been processed.

The Station Modification Procedure, LAP-1300-2, required the revised drawings to be maintained in the Modification Package until the final review had been complete. The Document Control Procedure, LAP-810-11, which provides for document control for drawings assigned to modifications, required only the Central File print file to be marked with the modification in progress. The procedure did not provide for marking the satellite files until such time as a Drawing Change Request had been processed. The print which was used for placing the jumper in the feedwater pump NPSH logic was reproduced from the shift satellite file.

The delay in processing the final Drawing Change Request resulted from an increased review effort on approximately 150 modifications which were rejected by Quality Assurance for uninitiated corrections to the documents. This increased scrutiny of documents occurred because of an NRC open item on changes of this type in November, 1984.

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TEXT (If more space is required, use additional NRC Form 366A's) (17)

III. PROBABLE CONSEQUENCES OF THE OCCURRENCE

All components (with the exception of RCIC) responded as required during this event. RCIC was backed up since HPCS was in operation. The Low Pressure Core Spray (BM) system and Low Pressure Coolant Injection System (BO) were also available. The ADS relief valve lifted at a conservative setpoint. The failed thermal overload on valve 1E12-F040A did not impact on the operation of the RHR system since the valve was already closed.

The limit switch indication problem did not affect the operation of the Reactor Building ventilation damper (1VR04YA) in the closed (safety) direction.

The accumulated cycles on the HPCS nozzle are six cycled and the nozzle usage factor is less than 0.70.

IV. CORRECTIVE ACTIONS

A combination meeting of the Scram Reduction Committee and Restart Review was conducted the same day.

All necessary repairs to equipment which failed was completed prior to restart on 2/9/85.

It was found that the oil system of the RCIC turbine was dirty. The oil system was flushed and new oil and filters were installed. LST 85-22 was performed on RCIC before restart to verify RCIC operability. The problem with the Reactor Building ventilation damper (1VR04YA) was found to be associated with a limit switch. The limit switch was readjusted and the damper restored to fully operable status.

The problem with the 1E12-F040A valve was found to be due to a burned out thermal overload device. The thermal overload device was replaced, and the motor windings checked. The valve was left fully operable. The ADS valve pressure switch was found to be set at a lower pressure than required. The switch was recalibrated and left fully operable. The leak on the feedwater suction pressure switch root valve was repaired. All the above actions were performed prior to entering Mode 2 (Startup). The Motor Driven Reactor Feed Pump was also restored to service prior to restart.

All necessary surveillances to demonstrate operability were performed. On 2/8/85 the modification procedure was revised to require processing the Drawing Change Request following declaring the system operational but prior to the final documentation review. Processing the Drawing Change package at this time will insure the dissemination of system configuration and will also allow for subsequent review of the revised drawings in Central File.

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TEXT (If more space is required, use additional NRC Form 366A's) (17)

IV. CORRECTIVE ACTIONS (CONTINUED)

A review of the Temporary System Change procedure, LAP-240-6, was conducted and the procedure does not require revision. As a provision against utilizing incorrect prints until they could be verified correct, warning signs were temporarily posted on the satellite file in the Shift Engineer's office to direct all plant personnel to the Central File prints for official copies.

The Shift Overview Superintendent reviewed all outstanding Temporary System Changes to identify those which were associated with safety related systems. Two Temporary System Changes affecting safety related systems were identified which required review against the modification status. The two system changes in question are installed in the hydrogen recombiner system for environmental qualification considerations and the Reactor Water Cleanup system to bypass temperature switches for leak detection which have been removed from the Technical Specifications. The only modifications against the prints in question are those modifications which will permanently clear the Temporary System Changes.

On 2/9/85 a review of all modifications in a similar point of review was identified utilizing the Plant Modification Status Program. As a result of this review there was a possible 56 modifications in this status. Increased efforts were initiated to process all necessary DCR's. This was completed.

In order to obtain a status of other active modifications, i.e., modifications being installed in the plant, all modifications which are assigned to the working departments are being reviewed (AIR #1-85-67040). When the final status of modifications in progress and the completed modifications' DCR's are processed, the satellite files will be updated to accurately reflect the current system configurations.

A complete review of the Station's drawing control program is being conducted to ensure the necessary information is available for all plant personnel and adequate controls are applied (AIR #1-85-67041).

V. PREVIOUS OCCURRENCES

A similar event of low reactor level due to loss of feedwater with HPCS injection is documented in LER 373/85-02-00.

The RCIC turbine has tripped on overspeed once before. Refer to LER 84-054-00.

VI. NAME AND TELEPHONE NUMBER OF PREPARER

R. D. Koenig, 815/357-6761, extension 575.

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March 7, 1985

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

Dear Sir:

Reportable Occurrence Report #85-017-00, Docket #050-373 is being submitted to your office in accordance with 10CFR 50.73.

for R. D. Bishop
G. J. Diederich
Superintendent
LaSalle County Station

GJD/MLD/kg

Enclosure

xc: NRC, Regional Director
INPO-Records Center
File/NRC

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