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)
Duane Arnold Energy Center	0 5 0 0 0 3 3 1	1 OF 0 4

TITLE (4)
Scram due to F. W. reduction and SRV actuation

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)
									NONE	0 5 0 0 0
0 1	0 7	8 4	8 4	— 0 0 1	— 0 0	0 2	0 6	8 4	NONE	0 5 0 0 0

OPERATING MODE (9)		THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)			
N		20.402(b)	20.405(c)	X 60.73(a)(2)(iv)	73.71(b)
POWER LEVEL (10)		20.405(a)(1)(i)	50.36(c)(1)	50.73(a)(2)(v)	73.71(c)
1 0 0		20.405(a)(1)(ii)	50.36(c)(2)	50.73(a)(2)(vii)	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)	
		20.405(a)(1)(v)	50.73(a)(2)(iii)	50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER
Michael S. Harris, Technical Support Engineer	AREA CODE 3 1 9 — 8 5 1 - 7 3 0 6

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	S J	T B G	X 9 9 9	N							
A	S B	R V	D 2 4 3	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., approximate - fifteen single-space typewritten lines) (16)

While in run mode at approximately 100% power, the "B" Feedwater Recirculation valve failed open, thereby decreasing feedwater flow to the vessel. Due to an unrelated error in the performance of an ongoing surveillance test procedure, a main steam relief valve was inadvertently opened 43 seconds later. Vessel level decreased to the low level setpoint (170" above top of active fuel) and initiated a scram as designed. The relief valve was reseated approximately 75 seconds later. Both RFP's tripped on low suction pressure. The vessel level continued to fall until the HPCI/RCIC initiation setpoint was reached (119.5" above top of active fuel). Level was restored to 194" and HPCI/RCIC were secured within 4 minutes. Prior to startup, both feedwater recirc valves were repaired or inspected and the surveillance test procedure was revised to reduce the margin for personnel error in the future.

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

At 08:08:26 (T=0), while the reactor was at 99.8% power, the "B" Feedwater Recirculation Valve (SJ-FCV-1611) failed open due to a broken air supply line fitting. As a result, a portion of "B" reactor feedwater flow was returned to the condenser and the vessel level immediately began to drop. While operators were diagnosing and compensating for the decrease in vessel level, the "B" Reactor Feedwater Pump tripped on low suction pressure and 1 of 2 safety/relief valves (SB-RV-4401) utilizing lo-lo set logic cycled open at T +43 seconds. After the relief valve opened, a momentary upswing in vessel level was experienced, followed by an rapid decrease in level to the low level trip setpoint (170") and an automatic scram initiation at T +44 seconds. At T +57 seconds, the "B" feedwater pump breaker was reset. At T +66 seconds, both feedwater pumps tripped on low suction pressure (due to condensate pumps being unable to match RFP flow with the high vessel demand and return flow to the condenser). During this time, the vessel level was continuing to fall until the lo-lo level HPCI/RCIC initiation setpoint of 119.5" was reached at T +73 seconds. At this point, HPCI and RCIC initiated per design and the level began to rise. As the vessel level fell to the lo-lo level setpoint momentarily, HPCI initiated but did not inject at any time in the event. HPCI control logic requires low-low level signal to remain present for the duration of the injection valve initiating time (approximately 15 seconds) to have HPCI actually inject. At T +112 seconds, the "B" reactor feedwater pump breaker was again closed to provide additional makeup and remained closed with no further trips. The open safety/relief valve was reseated at T +118 seconds after operator action and remained closed. As RCIC was injecting and the "B" feedwater pump was functioning, the level quickly recovered to normal operating level (Maximum level reached was 204.4" at T +4 min. 30 sec.). After the level had recovered, RCIC was secured at T +3 min. 50 sec. and HPCI was secured at T +4 min. 30 sec.. At this point, the vessel level and pressure were stable at 194" and 810 psig respectively. All systems necessary for remaining in a safe shutdown condition were operable, and no further system perturbations were experienced.

In retrospect, the following causes and determinations can be made as to the exact component failures and related events leading up to, and after, the scram.

The feedwater recirculation valve (SJ-FCV-1611) is an air actuated, fail open, control valve, Control Components model no. PDA964-96BW. Investigation revealed that a flexible air supply line fitting broke due to fatigue. When the air supply was lost, the valve cycled open. Prior to startup, the broken airline on the 'B' recirc valve was repaired. As an additional precaution, the identical airline on the 'A' recirc valve (SJ-FCV-1569) was inspected and found satisfactory. As a long term corrective action, a maintenance action request has been generated to replace the fittings subject to fatigue with higher strength fittings.

Main steam safety/relief valve (SB-RV-4401) is one of two non-ADS safety relief valves that are utilized in the lo-lo set logic. Following an event that has opened one of the six (6) safety/relief valves, the lo-lo set logic establishes a lower opening pressure (lifting pressure: 1020/1025 psig, reseating pressure: 900/905 psig) respectively. Investigation revealed that the lifting of the SRV was totally unrelated to the failure of the feedwater recirc valve. Prior to the recirc valve failure, a lo-lo set instrument functional test had commenced. While

performing the routine surveillance procedure, the instrument technician erroneously placed an ohmmeter across the hi pressure switch contacts too early in the procedure. As a result, the SRV actuation solenoid relay was falsely energized with a hi pressure test signal, which, in conjunction with the procedure step to push the test button, caused the valve to cycle open (approx 40 seconds after the recirc valve failure). Approximately 75 seconds after the SRV had opened, the operators reset the lo-lo SRV logic, thereby disarming the solenoid and cycling the valve closed. It was later theorized that due to the relatively low internal impedance of the ohmmeter, the meter acted as a jumper and bypassed the contacts. It is noteworthy that prompt identification of the cause of the SRV actuation was greatly aided by the technician volunteering that he had indeed committed an error. This theory was confirmed prior to startup after running a special test procedure. This procedure recreated the applicable sequence of test steps leading up to the opening of the SRV and revealed that the connection of an ohmmeter across the contacts in question in conjunction with pressing the test button, would in fact energize the SRV solenoid.

To preclude recurrence of similar situations, all instrument technicians were assembled and shown during a step by step reconstruction, the exact cause of the event. The importance of procedural compliance without deviating from the prescribed testing sequence was impressed upon them. To reduce the margin for personnel error, the surveillance test procedure has been revised to utilize a volt meter in lieu of an ohmmeter. This revision will eliminate the possibility of the meter acting as a jumper in a circuit under test, regardless of when it is inserted. As a long term corrective action, selected surveillance test procedures are continuing to be revised to simplify and clarify the testing sequence, thereby reducing the possibility of personnel error. In addition to this, a joint management/union task force is being formed to determine and implement methods of maintaining a heightened awareness of the importance of minimizing personnel errors.

Prior to startup, all systems initiated during the scram and subsequent evolutions were reviewed to determine whether they had functioned per design and technical specifications. At 170" reactor water level, the reactor scramed, valve groups 2, 3, 4 and 5 isolated, and both trains of the standby gas treatment system initiated per design. At 119.5" reactor water level valve group 8 isolated, LPCI logic tripped both recirc pumps, loop select logic initiated and performed its' function, and HPCI/RCIC initiated per design. During and after the transient, the operators observed a normal SRV tail pipe temperature response. As an additional precaution, a walkdown was performed on the tailpipe of the opened SRV. The piping and restraints were inspected and found satisfactory.

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

It was also observed that only one "side" of the reactor lo-lo level switches initiated HPCI/RCIC. This was later attributed to the location of the level switches' sensing lines in relation to the opened SRV. It appears that when the SRV opened, the resultant pressure profile across the top of the vessel skewed the actual level such that the low level was only seen by the switches whose sensing lines were attached to the vessel opposite the steam line on which the relief valve lifted. It appears that vessel level just reached the 119.5" setpoint and did not decrease appreciably below that level. To ensure that the low level indicating switches were functioning properly, a surveillance test procedure was performed on both sides (4 channels) of the HPCI/RCIC initiation level switches. Prior to startup, both sides were found to be in calibration and functioning per design.

With the exception of the unrelated failure of the recirc valve and safety/relief valve, all components and systems functioned per design throughout the event. At no time was the safety of the plant or public endangered. Apart from the corrective actions outlined above, no further action is warranted or planned at this time.

Iowa Electric Light and Power Company

February 6, 1984
DAEC-84- 64

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

Subject: Duane Arnold Energy Center
Docket No. 50-331
Op. License DPR-49
Licensee Event Report No. 84-001

Gentlemen:

In accordance with 10 CFR 50.73 please find attached a copy of the subject Licensee Event Report.

Very truly yours,

Daniel L. Mineck
Plant Superintendent - Nuclear
Duane Arnold Energy Center

DLM/MSH/pv

attachment

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

NRC Resident Inspector - DAEC

File A-118a

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