REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR: 8705270551 DOC. DATE: 87/05/15 NOTARIZED: NO DOCKET #
FACIL: 50-331 Duane Arnold Energy Center, Iowa Electric Light & Pow 05000331
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MINECK, D. L. Iowa Electric Light & Power Co.
RECIP. NAME RECIPIENT AFFILIATION

SUBJECT: LER 87-005-00: on 870314 & 18, primary containment valves failed Type C local leak rate tests. Caused by excessive radial clearance between disc/piston assembly & valve bore. Valve restoration continuing W/870515 ltr.

DISTRIBUTION CODE: IE22D COPIES RECEIVED: LTR 1 ENCL 1 SIZE: 6
TITLE: 50.73 Licensee Event Report (LER), Incident Rpt, etc.

NOTES:

	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL
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INTERNAL:	ACRS MICHELSON	1	1	ACRS MOELLER	2	2
	AEOD/DOA	1	1	AEOD/DSP/ROAB	2	2
	AEOD/DSP/TPAB	1	1	DEDRO	1	1
	NRR/DEST/ADE	1	0	NRR/DEST/ADS	1	0
	NRR/DEST/CEB	1	1	NRR/DEST/ELB	1	1
	NRR/DEST/ICSB	1	1	NRR/DEST/MEB	1	1
	NRR/DEST/MTB	1	1	NRR/DEST/PSB	1	1
	NRR/DEST/RSB	1	1	NRR/DEST/SGB	1	1
	NRR/DLPQ/HFB	1	1	NRR/DLPQ/QAB	1	1
	NRR/DOEA/EAB	1	1	NRR/DREP/RAB	1	1
	NRR/DREP/RPB	2	2	NRR/PMAS/ILRB	1	1
	NRR/PMAS/PTSB	1	1	REG FILE 02	1	1
	RES DEPY GI	1	1	RGN3 FILE 01	1	1
EXTERNAL:	EG&G GROH, M	5	5	H ST LOBBY WARD	1	1
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NRC Form 366 (9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/88

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Duane Arnold Energy Center (DAEC)	0 5 0 0 0 3 3 1	1 OF 0 5

TITLE (4): Primary Containment Valves Fail Leak Tests Due to Various Causes

EVENT DATE (5) MONTH	DAY	YEAR	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	REPORT DATE (7) MONTH	DAY	YEAR	OTHER FACILITIES INVOLVED (8) FACILITY NAMES	DOCKET NUMBER(S)
0 3	1 4	8 7	8 7	— 0 0 5 —	0 0	0 5	1 5	8 7	None	0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): N

POWER LEVEL (10): 0 0 0

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

[ ] 20.402(b)	[ ] 20.405(c)	[ ] 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)	[ ] 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 AREA CODE	
John Reinholdt, Technical Support Engineer	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	I S V	A 3 9 1	YES	X	S J	I S V	A 3 9 1	YES
X	S J	I S V	A 3 9 1	YES	X	S J	I S V	A 3 9 1	YES

SUPPLEMENTAL REPORT EXPECTED (14)

[ ] YES *(If yes, complete EXPECTED SUBMISSION DATE)* [X] NO

EXPECTED SUBMISSION DATE (15): MONTH | DAY | YEAR

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

On March 18, 1987, the reactor was in cold shutdown for a refuel outage and Type C Local Leak Rate Testing (LLRT) was in progress. At 1518 hours, it was reported that the inboard and outboard feedwater check valves leaked excessively. Based on the leakage, it was assumed the acceptance criterion in 10 CFR 50, Appendix J, III.C.3 had been exceeded. Excessive leakage through the feedwater check valves is being reported pursuant to 10 CFR 50.73(a)(2)(v)(c).

On March 14 and 18, 1987, the reactor was in cold shutdown and it was reported that the leakage through certain Main Steam Isolation Valves (MSIVs) exceeded the limit of 11.5 scf/hr in Technical Specification 4.7.A.2.c.3.

The root causes of these failures are various. Corrective actions have been initiated to minimize the "as found" leakage through the feedwater system check valves and the MSIVs.

The valves have been or are currently being restored to an acceptable condition. The "as found" and "post-maintenance" leakage for each valve will be addressed in the LLRT Report prepared and submitted to the NRC as required by 10 CFR 50, Appendix J and Technical Specification 4.7.A.2.g.

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NRC Form 366 (9-83)

NRC Form 366B (9-83)

LICENSEE EVENT REPORT (LER) FAILURE CONTINUATION

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO 3150-0104
EXPIRES: 8/31/88

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Duane Arnold Energy Center	0 5 0 0 0 3 3 1	8 7	— 0 0 5 —	0 0	0 2 OF 0 6

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 B	I S V	R 3 4 4	Y							
X	S B	I S V	R 3 4 4	Y							
X	S B	I S V	R 3 4 4	Y							
X	S B	I S V	R 3 4 4	Y							

NRC FORM 366B (9-83)

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Duane Arnold Energy Center (DAEC)	0 5 0 0 0 3 3 1	8 7	0 0 5	0 0	0 3 OF 0 5

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

Feedwater Primary Containment Valves

On March 18, 1987, the reactor was in cold shutdown for refueling and Type C Local Leak Rate Testing (LLRT) was in progress in accordance with 10 CFR 50, Appendix J. At 1518 hours, it was reported that both inboard feedwater system check valves (EIIS System Nos. SJ-ISV-14-1 and SJ-ISV-14-3) and both outboard feedwater stop-check valves (SJ-ISV-MO-4441 and SJ-ISV-MO-4442) leaked excessively. The "as found" results were as follows:

Penetration	Valve	As Found Leakage (1) (scf/hr)
X-9B	SJ-ISV-14-1	>847
X-9A	SJ-ISV-14-3	>847
X-9A	SJ-ISV-MO-4441	108
X-9B	SJ-ISV-MO-4442	>847

(1) Leakage results reported as "greater than" represent the maximum measurable leakage obtainable with the test equipment and/or instrumentation utilized with these tests at the obtainable test pressure.

Based on the maximum measurable leakage through the feedwater penetrations, it was assumed the acceptance criterion in 10 CFR 50, Appendix J, III.C.3 had been exceeded. Therefore, the feedwater check valve failures are being reported pursuant to 10 CFR 50.73 (a)(2)(v)(c).

Various troubleshooting techniques employed after the LLRT indicate that a majority of the leakage was through the valve stem packing on SJ-ISV-MO-4442. Both of the feedwater stop-check valves are equipped with a packing leak off line which discharges to the Reactor Building Equipment Drain Sump (RBEDS).

SJ-ISV-MO-4441 and SJ-ISV-MO-4442 are 16" stop check valves, with a close-assist motor operator (powered via essential buses), manufactured by the Anchor Valve Company, Drawing No. 2817-5. To date, the packing on SJ-ISV-MO-4442 has been repaired and the valve restored to an acceptable condition. A review of the maintenance histories for these valves indicates that packing leakage is a recurring problem. Therefore, an Engineering Work Request has been initiated to investigate replacement of the current packing design with a more effective type; and evaluate the removal of the packing leak-off line so packing leaks can be identified during periodic inspections.

★U.S.GPO:1986-0-624-538/455

SJ-ISV-14-1 and SJ-ISV-14-3 are 16" tilting disc, spring-assist to close, single seat (hard seat) check valves manufactured by the Anchor/Darling Valve Company, Drawing No. 2817-5. After disassembling the two valves, it was discovered the disc was not fully seated. An accumulation of iron-oxide particles between the disc bushings and the mating shafts was found. The particles hampered free disc movement. The disc bushings and mating shafts were not corroded and it was determined the oxide particles were from an external source. Inspection of the valve internals revealed no mechanical damage. An investigation to determine the source of the iron-oxide particles and for other contributing causes is continuing.

Excessive leakage through SJ-ISV-14-1 and SJ-ISV-14-3 has occurred in the past (See LER 85-05). During the 1985 refuel outage, these valves were overhauled after discovering seat damage. The disc bushings were modified to provide a more favorable swing path of the disc into the seat. The failure mechanism discovered recently is not related to those found in 1985.

Back-leakage through the inboard feedwater check valves and the outboard valve stem packings would have been directed to the RBEDS via the packing leak-off lines. Gases venting from the RBEDS are processed through the Standby Gas Treatment System before being released to the environment. The potential leakage via the feedwater system pathway cannot be demonstrated to be within the 0.6 La criteria. However, our engineering judgement is that this leakage and associated offsite radiological consequences by this pathway in the event of a design basis accident would not have constituted a significant radiological hazard.

Main Steam Isolation Valves

On March 14, 1987, the reactor was in cold shutdown for a refuel outage and LLRTs were in progress in accordance with 10 CFR 50, Appendix J. At 0957 hours, it was reported that the leakage through Main Steam Isolation Valves (MSIVs) SB-ISV-4413 and SB-ISV-4416 was greater than the criteria of 11.5 scf/hr specified in Technical Specification 4.7.A.2.c.3. It was later determined the initial test results for SB-ISV-4416 were not accurate due to gross leakage through the inboard valve. Troubleshooting techniques used after the LLRT indicated the leakage through SB-ISV-4416 is probably acceptable. However, SB-ISV-4416 will be formally retested after the inboard is repaired. The results will be presented in the LLRT Report prepared and submitted to the NRC as required by 10 CFR 50, Appendix J and Technical Specification 4.7.A.2.g. On March 18, 1987, with the reactor still in cold shutdown, it was reported at 1518 hours that MSIVs SB-ISV-4415 and SB-ISV-4418 also failed to meet the leakage criteria.

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

The following is a summary of the "as found" MSIV leakage:

Penetration	Inboard Valve	Leakage (scf/hr)	Outboard Valve	Leakage (scf/hr)
7A	SB-ISV-4412	0	SB-ISV-4413	21.61
7B	SB-ISV-4415	457.24	SB-ISV-4416	Currently not available
7C	SB-ISV-4418	77.33	SB-ISV-4419	0
7D	SB-ISV-4420	0	SB-ISV-4421	4.66

The MSIVs are 20" Y-pattern stop valves built by the Rockwell Manufacturing Company, Figure No. 1612 JMMNY.

After SB-ISV-4413 was disassembled, it was discovered there was not complete 360° disc to seat contact. The root cause is unknown. As a corrective action, the disc and seat were machined, the valve was reassembled and an acceptable LLRT will be performed.

As reported in LER 85-05, excessive "as found" leakage through the MSIVs has been experienced in the past.

Although the "as found" leakage through SB-ISV-4413 exceeded the design basis limit of 11.5 scf/hr, the redundant inboard (SB-ISV-4412) leakage was acceptable and thus the total leakage through penetration 7A did not exceed 11.5 scf/hr. Therefore, the "as found" condition of SB-ISV-4413 would not have had an adverse affect on the potential offsite radiological consequences under design basis conditions.

The excessive leakage through SB-ISV-4415 and SB-ISV-4418 has been attributed to excessive radial clearance between the disc/piston assembly and the valve bore thereby causing excessive lateral and angular misalignment of the disc/piston assembly. The outside diameters of the disc/piston assemblies were measured and found to be at minimum acceptable dimensions.

The rib guide dimensions were also measured and no excessive wear is evident. The outside diameter of the disc/piston assembly is being built-up to reduce radial clearances and minimize lateral and angular misalignment.

Based on the acceptable leakage through redundant valve SB-ISV-4419, the excessive leakage through SB-ISV-4418 would have not affected the radiological consequences under design basis conditions. The effects of excessive leakage through SB-ISV-4415 cannot be assessed at this time due to the unavailability of "as found" leakage data through its redundant isolation valve (SB-ISV-4416).

Iowa Electric Light and Power Company
May 15, 1987
DAEC-87-0530

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

Subject: Duane Arnold Energy Center
Docket No. 50-331
Op. License DPR-49
Licensee Event Report No. 87-005

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

DLM/JPR/go

Attachment - LER 87-005

cc: Mr. A. Bert Davis
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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