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 FACIL: 50-331 Duane Arnold Energy Center, Iowa Electric Light & Pow      05000331  
 AUTH. NAME      AUTHOR AFFILIATION  
 REINHOLDT, J.      Iowa Electric Light & Power Co.  
 MINECK, D. L.      Iowa Electric Light & Power Co.  
 RECIP. NAME      RECIPIENT AFFILIATION

SUBJECT: LER 87-005-01: on 870314, inboard & outboard feedwater check valves leaked excessively & on 870314 & 18, leakage through certain MSIVs reported, due to various causes. Valves restored to acceptable condition & leak test submitted. W/870608 ltr.

DISTRIBUTION CODE: IE22D      COPIES RECEIVED: LTR 1 ENCL 1 SIZE: 7  
 TITLE: 50.73 Licensee Event Report (LER), Incident Rpt, etc.

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	AEOD/DOA	1 1	AEOD/DSP/ROAB	2 2
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	NRR/DEST/MTB	1 1	NRR/DEST/PSB	1 1
	NRR/DEST/RSE	1 1	NRR/DEST/SGB	1 1
	NRR/DLPQ/HFB	1 1	NRR/DLPQ/QAB	1 1
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	NRR/DREP/RPB	2 2	NRR/PMAS/ILRB	1 1
	NRR/PMAS/PTSB	1 1	<u>REG FILE</u> 02	1 1
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EXTERNAL:	EG&G GROH, M	5 5	H ST LOBBY WARD	1 1
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LICENSEE EVENT REPORT (LER)

FACILITY NAME (1) <b>Duane Arnold Energy Center (DAEC)</b>	DOCKET NUMBER (2) <b>0 5 0 0 0 3 3 1 1</b>	PAGE (3) <b>1 OF 0 6</b>
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TITLE (4)  
**Primary Containment Valves Fail Leak Tests Due to Various Causes**

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																
									None																
0	3	1	4	8	7	8	7	0	0	0	5	0	1	0	6	0	8	8	7						

OPERATING MODE (9) <b>N</b>	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)									
POWER LEVEL (10) <b>0 0 0</b>	<input type="checkbox"/>	20.402(b)	<input type="checkbox"/>	20.406(c)	<input checked="" type="checkbox"/>	60.73(a)(2)(iv)	<input type="checkbox"/>	73.71(b)		
	<input type="checkbox"/>	20.406(a)(1)(i)	<input type="checkbox"/>	50.38(c)(1)	<input type="checkbox"/>	60.73(a)(2)(v)	<input type="checkbox"/>	73.71(c)		
	<input type="checkbox"/>	20.406(a)(1)(ii)	<input type="checkbox"/>	50.38(c)(2)	<input type="checkbox"/>	60.73(a)(2)(vii)	OTHER (Specify in Abstract below and in Text, NRC Form 366A)			
	<input type="checkbox"/>	20.406(a)(1)(iii)	<input type="checkbox"/>	60.73(a)(2)(i)	<input type="checkbox"/>	60.73(a)(2)(viii)(A)				
	<input type="checkbox"/>	20.406(a)(1)(iv)	<input type="checkbox"/>	60.73(a)(2)(ii)	<input type="checkbox"/>	60.73(a)(2)(viii)(B)				
<input type="checkbox"/>	20.406(a)(1)(v)	<input type="checkbox"/>	60.73(a)(2)(iii)	<input type="checkbox"/>	60.73(a)(2)(ix)					

LICENSEE CONTACT FOR THIS LER (12)						TELEPHONE NUMBER					
NAME						AREA CODE					
<b>John Reinholdt, Technical Support Engineer</b>						<b>3 1 9 8 5 1 - 1 7 3 0 6</b>					

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)											
CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NRPDS		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NRPDS	
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)						EXPECTED SUBMISSION DATE (15)		
<input type="checkbox"/> YES (If yes, complete EXPECTED SUBMISSION DATE)						<input checked="" type="checkbox"/> NO		
						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 restored to an acceptable condition. The "as found" and "post-maintenance" leakage for each valve will also 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.

This revision has been submitted to incorporate final leak test results and to present a commitment for an increased MSIV surveillance test (LLRT) frequency during Cycle 9.



LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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		8   7	-   0   0   5	-   0   1	0   3	OF	0   6

TEXT (If more space is required, use additional NRC Form 388A'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:

<u>Penetration</u>	<u>Valve</u>	<u>As Found Leakage (1) (scf/hr)</u>
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. The packings on SJ-ISV-MO-4441 and SJ-ISV-MO-4442 have been repaired and the valves 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.

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TEXT (If more space is required, use additional NRC Form 388A's) (17)

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. SB-ISV-4416 was tested again after the inboard was repaired and the results show the "as found" leakage is less than 11.5 scf/hr. 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 388A'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	3.39
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 was 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.

To demonstrate the above corrective actions have been effective, an LLRT will be performed on SB-ISV-4415 if the DAEC is shutdown anytime after six months of operation, up to four months prior to the next refuel outage. Other MSIVs may be tested as time permits. If the "as found" leakage through SB-ISV-4415 is near 11.5 scf/hr, an LLRT will be performed on SB-ISV-4418. If the "as found" leakage through SB-ISV-4415 exceeds 11.5 scf/hr, all MSIVs will be leak tested.

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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TEXT (If more space is required, use additional NRC Form 305A's) (17)

Based on the acceptable leakage through redundant valves SB-ISV-4416 and SB-ISV-4419, the leakage via this pathway is acceptable and would have not affected the radiological consequences under design basis conditions.

Revision 1 has been submitted to incorporate final leak test results and to present the commitment for an increased MSIV surveillance test (LLRT) frequency during Cycle 9.

Iowa Electric Light and Power Company

June 8, 1987  
DAEC-87-0693

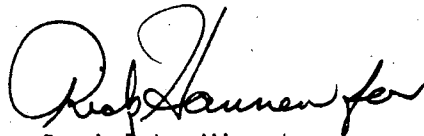
U. S. Nuclear Regulatory Commission  
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Washington, D. C. 20555

Subject: Duane Arnold Energy Center  
Docket No. 50-331  
Op. License DPR-49  
Licensee Event Report No. 87-005 Rev 1

Gentlemen:

In accordance with 10 CFR 50.73 and 10 CFR 21 please find attached a copy of the subject revised Licensee Event Report.

Very truly yours,



Daniel L. Mineck  
Plant Superintendent - Nuclear

DLM/JPR/go

Attachment - LER 87-005 Rev 1

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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