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ACCESSION NBR: 8710210039 DOC. DATE: 87/10/16 NOTARIZED: NO DOCKET #
FACIL: 50-315 Donald C. Cook Nuclear Power Plant, Unit 1, Indiana & 05000315

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RECIP. NAME RECIPIENT AFFILIATION

SUBJECT: LER 87-012-01: on 870702, Type B & C containment leak rate tests failed on two valves. Caused by ripped diaphragm on one valve. Cause unknown on second valve. Valves repaired. W/871016 ltr.

DISTRIBUTION CODE: IE22D COPIES RECEIVED: LTR 1 ENCL 1 SIZE: 5
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
	PD3-3 LA	1	1	PD3-3 PD	1	1
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INTERNAL:	ACRS MICHELSON	1	1	ACRS MOELLER	2	2
	AEOD/DOA	1	1	AEOD/DSP/NAS	1	1
	AEOD/DSP/ROAB	2	2	AEOD/DSP/TPAB	1	1
	ARM/DCTS/DAB	1	1	DEDRO	1	1
	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
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	NRR/DREP/RAB	1	1	NRR/DREP/RPB	2	2
	NRR/DRIS/SIB	1	1	NRR/PMAS/ILRB	1	1
	REG FILE 02	1	1	RES DEPY GI	1	1
	RES TELFORD, J	1	1	RES/DE/EIB	1	1
	RGN3 FILE 01	1	1			
EXTERNAL:	EG&G GROH, M	5	5	H ST LOBBY WARD	1	1
	LPDR	1	1	NRC PDR	1	1
	NSIC HARRIS, J	1	1	NSIC MAYS, G	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)
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TITLE (4) Type B & C Containment Leak Rate Test Failure Due to a Ripped Diaphragm and an Undetermined Cause

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 7	0 2	8 7	8 7	0 1 2	0 1	1 0	1 6	8 7		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 5
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)	[ ] 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)	[X] 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	
T. K. Postlewait- Technical Engineering Superintendent	6 1 6	4 6 5 - 5 9 0 1

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
C	C A	I S V	G 2 5 7	N					
C	C A	I S V	G 2 5 7	N					

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)

This is a supplemental report submitted to provide additional information regarding the subject violation reported on July 24, 1987.

On July 2, 1987, the as-found results of Type C testing performed on two manual containment isolation diaphragm valves between the Reactor Coolant Drain Tank and Refueling Water Purification Pumps reflected an excessive leak rate. This resulted in the accumulated leakage for all penetrations and valves subject to Type B&C tests exceeding the allowed limit of 0.60 La of Technical Specification (T/S) 3.6.1.2b.

The cause of one of the valve failures was determined to be a ripped diaphragm. No definitive cause for the other valve failure can be determined. Subsequent repairs to both valves corrected the leakage problem. These valves will be tested on an increased frequency in order to monitor their performance. The results will be used to determine future repair/replacement actions.

The as-left leakage rate for the valves was below the applicable ISI guideline leakage limits and the combined as-left leakage for all penetrations and valves subject to Type B and C testing was calculated to be 0.046 La, which is well below the T/S limit of 0.60 La.

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PDR ADOCK 05000315
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TEXT (If more space is required, use additional NRC Form 366A's) (17)

Conditions Prior to Occurrence

Unit 1 in Mode 5 (Cold Shutdown)

Description of Event

This is a supplemental report submitted to provide additional information regarding the subject violation reported on July 24, 1987.

At 1400 hours on July 2, 1987, the accumulated leakage found while performing Type B and C Leak Rate Testing on containment penetrations exceeded the limiting condition for operation (LCO) of 0.60 La required by Technical Specification (T/S) 3.6.1.2b. A four hour report was made to the NRC via the ENS at 1610 hours.

The excessive leakage was discovered during testing of two manual containment isolation valves (EIIS/ISV) located between the Reactor Coolant Drain Tank (EIIS/CA-TK) and Refueling Water Purification Pumps (EIIS/CA-P). These diaphragm valves (SF-159 and SF-160, in series), failed to achieve an adequate seal under test conditions resulting in a leak rate that was not quantifable with the test equipment being used. A second attempt to quantify the leakage was not possible because the repair of the valves had already inadvertently been initiated.

The combined as-found leakage for all other penetrations subject to Type B and C tests, excluding valves SF-159 and SF-160, was determined to be 0.12 La. Therefore, the failure of valves SF-159 and SF-160 presented the only significant concern.

With the exception of the subject valves (SF-159 and SF-160), there were no inoperable structures, components or systems that contributed to the consequences of this event.

Cause of Event

The dismantling of valve SF-159 indicated that the excessive leakage was the result of a ripped diaphragm. The dismanting of valve SF-160 indicated that the diaphragm appeared to be in good condition. No definitive cause can be determined concerning SF-160. A review of the test procedure and maintenance repair/adjustment procedure did not indicate any problems with either procedure. Subsequently, the diaphragms for both valves were replaced which corrected the leakage problem. Both valves had been left within In-Service Inspection (ISI) guideline leakage limits after the last required Type C Leak Rate Test.

★U.S.GPO:1986-0-624-538/455

NRC Form 366A
(9-83)

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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

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

Analysis of Event

This event is considered reportable under the criteria of 10CFR50.73(a)(2)(i)(B) in that contrary to the requirements of T/S 3.6.1.2b, the reactor coolant system temperature was in excess of 200 degrees F while the combined leakage for all penetrations and valves subject to Type B and C tests was > 0.60 La.

Valves SF-159 and SF-160 are part of the Refueling Cavity Drain Line. The valves are situated between the Reactor Coolant Drain Tank and the Refueling Water Purification Pumps. These valves are open only during the draining of the Refueling Cavity. At all other times these valves are sealed closed. The piping upstream of the valves is Class 1, which must remain in-service after a postulated accident. The piping downstream of the valves is Class 3, which is assumed not to remain in-service after a postulated accident.

If a postulated accident were to have occurred, then the possibility of leakage through SF-159 and SF-160 to the atmosphere would have to be assumed, since the downstream piping is Class 3. Therefore, containment isolation would not have been maintained.

Regarding actual conditions, however, a review of the system configuration indicates that there was a water barrier present downstream of the valves at all times which would have prevented any release to the atmosphere. There were no obstructions between the Refueling Water Purification Pumps and the Refueling Water Storage Tank (RWST) and only one normally open valve between the Purification Pumps and SF-160. Based on the minimum RWST level (governed by T/S 3.1.2.8.b.1) it can be shown that adequate head was present to ensure that the system was "water solid". Therefore, since the last Type B and C Leak Rate Testing, it can be concluded that containment isolation was maintained as the water barrier was present and the piping remained intact.

Based on the above we believe the condition did not involve a significant safety problem.

Corrective Actions

The diaphragms of both valves were subsequently replaced. Retests of the valves on July 23, 1987, indicated that the combined as-left leakage for SF-159 and SF-160 was less than 60 standard cubic centimeters per minute (sccm). This was well within the applicable ISI guideline limit of 900 sccm. The total as-left leakage for all penetrations and valves subject to Type B and C Leak Rate Testing was calculated to be 0.046 La, which is well below the 0.60 La limit required by T/S 3.6.1.2b.

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

Corrective Actions (cont'd)

To prevent recurrence, all manual containment isolation diaphragm valves in both Unit 1 and Unit 2 will be Type C Leak Rate tested on an increased frequency (concurrent with scheduled refueling and mid-cycle surveillance outages) in order to monitor their performance. The results will be used to attempt to determine any contributing factors that affect their performance and any future repair/replacement actions. There are only three manual containment isolation diaphragm valves per Unit subject to T/S 3.6.1.2b Type C Leak Rate Tests.

Failed Component Identification

NOTE: The failure of the subject valves was the result of a ripped diaphragm on valve SF-159 and an undetermined cause on valve SF-160. No design, manufacturing, construction or installation irregularities were identified.

Component Name: Reactor Coolant Drain Tank to Refueling Water Purification Pumps Containment Isolation Valves

Plant I.D. No.: SF-159
SF-160

Manufacturer: ITT Grinnell

Model No.: C15DB035

EIIS Code: CS-ISV

Previous Similar Events

There have been no previous similar events in which failure of Type B and C Testing was caused by the failure of either of these valves.

*U.S.GPO:1986-0-624-538/455

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INDIANA
MICHIGAN
POWER

October 16, 1987

United States Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

Operating License DPR-58
Docket No. 50-315

Document Control Manager:

In accordance with the criteria established by 10 CFR 50.73 entitled Licensee Event Reporting System, the following report is being submitted:

87-012-01

Sincerely,

W. G. Smith, Jr.
Plant Manager

/afh

Attachment

cc: John E. Dolan
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