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ACCESSION NBR:9212150109 DOC.DATE: 92/12/07 NOTARIZED: NO DOCKET #
FACIL:50-315 Donald C. Cook Nuclear Power Plant, Unit 1, Indiana M 05000315

AUTH.NAME	AUTHOR AFFILIATION
WEBER,G.A.	Indiana Michigan Power Co. (formerly Indiana & Michigan Ele
BLIND,A.A.	Indiana Michigan Power Co. (formerly Indiana & Michigan Ele
RECIP.NAME	RECIPIENT AFFILIATION

SUBJECT: LER 90-015-02:on 901102,discovered containment Type B & C leakage exceeded L.C.O. value due to degradation of isolation valve seating surfaces.Seat was lapped & disc was replaced.W/921207 ltr.

DISTRIBUTION CODE: IE22T COPIES RECEIVED:LTR 1 ENCL 1 SIZE: 5
TITLE: 50.73/50.9 Licensee Event Report (LER), Incident Rpt, etc.

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	DEAN,W	1	1			
INTERNAL:	ACNW	2	2	AEOD/DOA	1	1
	AEOD/DSP/TPAB	1	1	AEOD/ROAB/DSP	2	2
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	NRR/DREP/PRPB11	2	2	NRR/DST/SELB 8D	1	1
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	NRR/DST/SRXB 8E	1	1	REG FILE 02	1	1
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EXTERNAL:	EG&G BRYCE,J.H	2	2	L ST LOBBY WARD	1	1
	NRC PDR	1	1	NSIC MURPHY,G.A	1	1
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December 7, 1992

United States Nuclear Regulatory Commission
Document Control Desk
Rockville, Maryland 20852

Operating Licenses DPR-58
Docket No. 50-315

Document Control Manager:

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

90-015-02

Sincerely,

A. Alan Blind

A. A. Blind
Plant Manager

/sb

Attachment

c: D. H. Williams, Jr.
A. B. Davis, Region III
E. E. Fitzpatrick
P. A. Barrett
R. F. Kroeger
B. Walters - Ft. Wayne
NRC Resident Inspector
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B. A. Svensson

IE22
/1

NRC FORM 366 (6-89)

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104
EXPIRES: 4/30/92

ESTIMATED BURDEN PER RESPONSE TO COMPLY WTH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE RECORDS AND REPORTS MANAGEMENT BRANCH (P-530), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1): D. C. COOK NUCLEAR PLANT - UNIT 1

DOCKET NUMBER (2): 0 5 0 0 0 3 1 5

PAGE (3): 1 OF 0 4

TITLE (4): CONTAINMENT TYPE B AND C LEAKAGE EXCEEDS L.C.O. VALUE DUE TO DEGRADATION OF ISOLATION VALVE SEATING SURFACES

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)
1 1	0 2	9 0	9 0	0 1 5	0 2	1 2	0 7	9 2		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	
G. A. WEBER - PLANT 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
	B D	T K	X 9 9 9	Y	X	C B	I S V	C 6 3 1	Y
X	C B	I S V	C 6 3 1	Y					

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 updated report is being submitted to modify a Corrective Action activity and rephrase commitments as completed activities.

With the Reactor Coolant System in Mode 5 (Cold Shutdown), the measured leakage, using the maximum pathway methodology, for the Type B and C Leak Rate Tests on Containment penetrations was 6.74 La. In addition, there were two penetrations that had leak rates that could not be quantified. This exceeded the L.C.O. value (0.60 La) of Technical Specification 3.6.1.2.b.

The measured leakage for the Weld Channel Pressurization System Valve Enclosure Manway for 1-ICM-305 (EIIS:TK/BD) was 94 percent of the measured Type B and C leak rate. 1-CS-442-1 and 1-CS-442-3 (EIIS:ISV/CB) are Containment Isolation Valves for the seal water injection lines to Reactor Coolant Pump numbers 11 and 13 and had unquantifiable leak rates. The three deficiencies would not have resulted in any additional leakage from Containment, due to their function in an accident. There was no leakage into the ICM-305 valve enclosure, therefore, it would not have been a contributor. The final Type B and C leak rate was 0.112 La.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
D. C. COOK NUCLEAR PLANT - UNIT 1	0 5 0 0 0 3 1 5	9 0	— 0 1 5 —	0 2	0 2 OF 0 4

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

This updated report is being submitted to modify a Corrective Action activity and rephrase commitments as completed activities.

Conditions Prior to Occurrence:

Unit-1 in Mode 5 (Cold Shutdown)

Description of Event:

With the Reactor Coolant System in Mode 5 (Cold Shutdown), the measured leakage, using the maximum pathway methodology, for the Type B and C Leak Rate Tests on Containment penetrations was 6.74 La. In addition, there were two penetrations that had leak rates that could not be quantified. This exceeded the L.C.O. value (0.60 La) of Technical Specification 3.6.1.2.b.

The measured leak rate for the Weld Channel Pressurization System Valve Enclosure Manway for 1-ICM-305 (EIIS:TK/BD) was 94 percent of the measured Type B and C leak rate. Check valves 1-CS-442-1 and 1-CS-442-3 (EIIS:ISV/CB) are Containment Isolation Valves for the seal water injection lines to Reactor Coolant Pump numbers 11 and 13, respectively. These valves had leak rates that could not be quantified.

Cause:

The excessive leak rate of 1-ICM-305 valve enclosure manway cover was caused by a portion (approximately 12 inches) of the O-ring being out of its channel. This manway cover was last installed on June 10, 1989. On June 11, 1989 the valve enclosure was tested and had a leak rate of 1500 sccm. When the manway cover was removed, a portion of the O-ring fell off. The corresponding portion of the O-ring groove was full of grease that is used to lubricate the O-ring. It is believed that a portion of the O-ring was pulled out of its channel when the manway cover was aligned for bolting. The grease used on the O-ring, when it was installed, must have provided a sufficient seal to allow the valve enclosure to pass the June 11, 1989 Leak Rate Test, but degraded since then.

The excessive leak rates for 1-CS-442-1 and 1-CS-442-3 are attributed to pieces of neoprene found in the valves. A small piece was found in 1-CS-442-1 and a piece approximately 0.19 inches in diameter was found in 1-CS-442-3. These check valves are downstream of the seal water filters. The neoprene pieces found in 1-CS-442-1 and 1-CS-442-3 are believed to have come from the seal water filters and either broke off during filter replacement, or were trapped by the filter and fell off during removal. This is the first time we have experienced such problems.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
D. C. COOK NUCLEAR PLANT - UNIT 1	0 5 0 0 0 3 1 5	9 0	— 0 1 5 —	0 2	0 3 OF 0 4

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

Corrective Actions:

The 1-ICM-305 valve enclosure manway cover O-ring was replaced. A retest indicated no leakage (000 sccm). No further action is planned since the O-ring installation problem was an isolated event and not an indication of a generic problem or practice.

Valve 1-CS-442-1 had the small piece of neoprene removed. The seat was lapped and the disc was replaced. The as-left leak rate was 40 sccm. Valve 1-CS-442-3 had the piece of neoprene removed. No additional repair actions were necessary. The as-left leak rate was 65 sccm. The Maintenance procedure used for replacement of the Seal Water Injection Filters was reviewed ensuring that adequate steps are in place to prevent the introduction of material to the system during seal replacement. In addition, the filter seal degradation was reviewed with the manufacturer. The filter seal failure is considered to be an isolated case since the manufacturing process for the filter did not change the filter composition or configuration.

Other Containment Isolation Valves that exhibited leak rates in excess of the guideline acceptance criteria were repaired and retested to ensure the leak rates were within allowable limits. The final as-left Type B and C leak rate was 0.112 La.

Analysis:

The ICM-305 valve enclosure is outside of Containment and provides a Containment barrier for any leakage from ICM-305, the Containment Sump Recirculation Isolation Valve. This valve enclosure is not part of the Containment Integrated Leak Rate Test boundary. Therefore, an additional leak into the enclosure would have to have developed for radioactivity to escape out of Containment. No other leaks were discovered in this valve enclosure during the as-found testing or during the as-left testing. Therefore, if Containment had been pressurized and radioactivity had been released into the Containment atmosphere, the radioactivity would have been contained within Containment. The public health and safety was never threatened by this condition.

Valves 1-CS-442-1 and 1-CS-442-3 are located in the Reactor coolant Pump Seal Water Injection System. Both of these valves were found with foreign material in the seat area and were unable to fully close. The leakage from these valves could not be quantified. These check valves would not be required to isolate Containment during an accident scenario as the Reactor Coolant Pump Seal Water Injection System remains in service following a design basis accident, preventing a flow path from Containment to the outside atmosphere. If a postulated break in either of these lines occurred, Containment integrity would be maintained by the two check valves located in each line inside Containment, and downstream of 1-CS-442-1 and 1-CS-442-3.

This event has been determined to be reportable under 10 CFR 50.73 (a)(2)(i)(C). Based on the above, however, it has also been determined that this condition did not create a significant safety concern. The as-found condition of the Containment would not have put the plant in an unsafe condition.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
D. C. COOK NUCLEAR PLANT - UNIT 1	0 5 0 0 0 3 1 5	9 0	— 0 1 5	— 0 2	0 4 OF 0 4

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

Failed Component Identification:

Component name: 1-ICM-305 Valve Enclosure
Plant I.D. No.: 1-TK-84 (EIIS:TK/BD)
Manufacturer: Unknown
Model No.: Unknown

Component name: Reactor Coolant Pump No. 11 Seal Water Injection Containment Isolation Valve
Plant I.D. No.: 1-CS-442-1 (EIIS:ISV/CB)
Manufacturer: Conval, Inc.
Model No.: 12C2

Component name: Reactor Coolant Pump No. 13 Seal Water Injection Containment Isolation Valve
Plant I.D. No.: 1-CS-442-3 (EIIS:ISV/CB)
Manufacturer: Conval, Inc.
Model No.: 12C2

Previous Similar Events:

Previous Licensee Event Reports submitted for excessive type B&C Leak Rate Test results include:

050-315/79-34	050-316/79-20
050-315/81-11	050-316/79-53
050-315/81-25	050-316/81-18
050-315/82-58	050-316/83-16
050-315/83-72	050-316/84-05
050-315/85-17	050-316/86-09
050-315/87-12	050-316/89-05
050-315/89-04	050-316/90-07