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FACIL:50-316 Donald C. Cook Nuclear Power Plant, Unit 2, Indiana & 05000316
AUTH.NAME AUTHOR AFFILIATION
DROSTE,J.B. 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 90-007-01:on 900717,Containment Type B & C leakage exceeds LCO value due to degradation of IVSS.

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TITLE: 50.73/50.9 Licensee Event Report (LER), Incident Rpt, etc.

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October 11, 1990

United States Nuclear Regulatory Commission
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Operating Licenses DPR-75
Docket No. 50-316

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:

90-007-01

Sincerely,

A. Alan Blind
A. A. Blind
Plant Manager

AAB:srb

Attachment

C: D. H. Williams, Jr.
A. B. Davis, Region III
M. P. Alexich
P. A. Barrett
J. E. Borggren
R. F. Kroeger
B. Walters - Ft. Wayne
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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)	DOCKET NUMBER (2)	PAGE (3)
D. C. COOK NUCLEAR PLANT, UNIT 2	0 5 0 0 0 3 1 6	1 OF 0 3

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

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	1 7	9 0	9 0	0 0 7	0 1	1 0	1 1	9 0		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)
J.B. DROSTE - 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
X	B D	I S V	C 4 1 8	Y					

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., approximately fifteen single-space typewritten lines)* (16)

This supplemental report is being submitted to provide additional information regarding the Type B and C Leakrate Testing initially reported on August 8, 1990.

With the Reactor Coolant System in Mode 5 (Cold Shutdown), the accumulated leakage found using the maximum pathway methodology for the Type B and C Leak Rate Tests on Containment penetrations was 0.74 La. This exceeded the L.C.O. value (0.60 La) of Technical Specification 3.6.1.2.b.

The Containment Pressure Relief Train-A Containment Isolation Valve, 2-VCR-107 (EIIS:ISV/BD), is the major contributing factor for exceeding the Technical Specification limit. 2-VCR-107 exhibited a leakage rate of 49,000 sccm. This is 60 percent of the total leakage obtained from all tested penetrations.

Other Containment Isolation Valves that exhibit leak rates in excess of Guideline acceptance criteria were repaired and retested to ensure the leak rates are within allowable limits. The final as-left Type B and C leakage rate was 0.17 La.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
D. C. COOK NUCLEAR PLANT UNIT - 2	0 5 0 0 0 3 1 6	9 0	— 0 0 7	— 0 1	0 2 OF 0 3

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

Conditions Prior to Occurrence

Unit-2 in Mode 5 (Cold Shutdown)

Description of Event

This supplemental report is being submitted to provide additional information regarding the Type B and C Leakrate Testing initially reported on August 10, 1990.

With the Reactor Coolant System in Mode 5 (Cold Shutdown), the accumulated leakage found using the maximum pathway methodology for the Type B and C Leak Rate Tests on Containment penetrations was 0.74 La. This exceeded the L.C.O. value (0.60 La) of Technical Specification 3.6.1.2.b.

The Containment Pressure Relief Train-A Containment Isolation Valve, 2-VCR-107 (EIIS:ISV/BD), is the major contributing factor for exceeding the Technical Specification limit. 2-VCR-107 exhibited a leakage rate of 49,000 sccm. This is 60 percent (or .445 La) of the total leakage obtained from all tested penetrations.

Cause

The excessive leakage rate of 2-VCR-107 was attributed to a degraded disc seal.

Corrective Action

Valve 2-VCR-107 was rebuilt, replacing the gaskets and disc seal. Other valves that exhibited leakage in excess of guideline acceptance criteria were repaired and retested.

Analysis of Event

The penetration for the Containment Pressure Relief Line, which includes 2-VCR-107, was designed to have two valves in series and is tested during the Appendix J Type C test program. A review of the test data indicates that 0.445 La was attributable to valve 2-VCR-107 (49,000 sccm). Valve 2-VCR-207 is in series with 2-VCR-107 and had a leak rate of 300 sccm. If credit is taken for 2-VCR-207, which is in series with 2-VCR-107, the total as-found leakage becomes 0.301 La, which shows that containment isolation could have been successfully achieved under the as-found conditions.

Based on the above, we believe that this event did not represent a significant hazard to public health and safety.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
D. C. COOK NUCLEAR PLANT UNIT - 2	0 5 0 0 0 3 1 6	9 0	— 0 0 7	— 0 1	0 3 OF 0 3

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

Failed Component Identification

Component Name: Containment Pressure Relief Train-A Containment Isolation Valve

Plant I.D. No: 2-VCR-107 (EIIS:ISV)

Manufacturer: Clow

Model No.: 80-9490-01

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	

[illegible]