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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,C.E. Indiana Michigan Power Co. (formerly Indiana & Michigan Ele
RECIP.NAME RECIPIENT AFFILIATION

SUBJECT: LER 92-007-00:on 920705,accumulated leakage for Type B & C leakrate tests on containment penetrations exceeded TS LCO, due to integrity of redundant isolation valves.Plant procedures modified.W/920128 ltr.

DISTRIBUTION CODE: IE22T COPIES RECEIVED:LTR 1 ENCL 1 SIZE: 9
TITLE: 50.73/50.9 Licensee Event Report (LER), Incident Rpt, etc.

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AO-4 dbp

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January 28, 1993

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:

92-007-01

Sincerely,

for A. A. Blind
Plant Manager

/sb

Attachment

c: 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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J. G. Keppler
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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 1	0 5 0 0 0 3 1 5	1 OF 0 9

TITLE (4): CONTAINMENT TYPE B AND C LEAKAGE EXCEEDS L.C.O. VALUE

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 5	9 2	9 2	0 0 7	0 1	0 1	2 8	9 3		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	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
X	K G	I S V	I 2 0 7	Y	B	B C	I S V	K 0 8 5	Y
X	I L	I S V	A 2 0 0	Y	D	B D	I S V	C 4 1 8	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 supplemental report is being submitted to provide additional information regarding the Type B and C Leakrate Testing results reported on July 17, 1992.

On July 5, 1992, with Unit 1 in Mode 5 (Cold Shutdown) the accumulated leakage for the Type B and C Leakrate Tests on Containment penetrations exceeded the Technical Specification L.C.O. value of 0.60 La. A four hour phone report was made to the NRC as required by 50.72(b)(2), at 0829 hours on July 5, 1992. The As-Found Type B and C measured leakrate was 0.728 La. In addition to the measured leakrate, three test penetrations had leakrates that exceeded the maximum flow of the test instrument, which is greater than 0.6 La, before the correct test pressure test could be obtained. In the case of two other valves, repair activities were performed prior to obtaining As-Found leakrates. As a result, the actual As-Found leakage rate was not determined, but is estimated as exceeding test acceptance criteria (0.6 La) by a factor of 4 to 5. Following repair of the defective valves, acceptable Leakrate Tests were obtained. The As-Left Type B and C Leakrate was 0.078 La. These deficiencies would not have resulted in any additional leakage from Containment and would not have put the Plant in an unsafe condition due to: the integrity of redundant isolation valves; the design function of the systems during a DBA and in some cases the system design configuration. Based on the Cook Nuclear Plant Probabilistic Risk Assessment, the increased risk resulting from this event was minimal. This condition did not have a significant impact on the health and safety of the public.

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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D. C. COOK NUCLEAR PLANT - UNIT 1	0 5 0 0 0 3 1 5	9 2	— 0 0 7 —	0 1	0 2 OF 0 9

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

This supplemental report is being submitted to provide additional information regarding the Type B and C Leakrate Testing reported on July 17, 1992.

Conditions Prior to Occurrence:

Unit 1 in Mode 5 (Cold Shutdown)

Event Description:

The total measured leakage for the Type B and C Leakrate Tests on Containment penetrations exceeded the L.C.O. value (0.60 La) of Technical Specification 3.6.1.2.b. The As-Found leakrate was 0.728 La. The As-Found leakrate does not include the leakrates of three valves that exceeded the maximum flow of the test instrument, which is greater than 0.6 La, before the correct test pressure could be obtained. Leakrates of up to 67,000 sccm are measurable. A leakrate of 66,000 sccm correlates to a Containment leakrate of 0.6 La. As a result, the actual As-Found leakage rate was not determined, but is estimated as exceeding test acceptance criteria (0.6 La) by a factor of 4 to 5. In addition, repair activities were performed on two valves prior to obtaining the as-found leakrates. The failure to meet the Technical Specification Limit is attributed to the following deficient valves:

- 1-WCR-955 - Non-Essential Cooling Water Supply and Return for No. 1 Reactor Coolant Pump Motor Cooler (EIIS:ISV/KG), was the major contributor for the high measured leakrate and exhibited a leakrate that accounted for 82 percent of the measured Type B and C leakrate.

- 1-SM-1 - Containment Lower Compartment Radiation Detectors ERS-1300 and ERS-1400 Return Header Check Valve. (EIIS:ISV/IL) The leakrate for 1-SM-1 could not be quantified, but was determined to be in excess of 67,000 sccm. The correct test pressure could not be achieved to determine the actual leakrate.

- 1-R-157 - Ice Condenser Refrigeration Glycol Return Header Containment Isolation Valves Pressure Relief Header Check Valve. (EIIS:ISV/BC) The leakrate for 1-R-157 could not be quantified, but was determined to be in excess of 67,000 sccm. The correct test pressure could not be achieved to determine the actual leakrate.

- 1-VCR-101 - Containment Instrumentation Room Purge Supply Train-A Containment Isolation Valve. (EIIS:ISV/BD) The leakrate for 1-VCR-101 could not be quantified, but was determined to be in excess of 67,000 sccm. The correct test pressure could not be achieved to determine the actual leakrate.

- 1-CS-442-2 and 1-CS-442-3 - Reactor Coolant Pump Seal Water Injection to Reactor Coolant Pumps Nos. 2 and 3 Containment Isolation Check Valves (EIIS:ISV/AB). Repairs were performed on these valves prior to obtaining an As-Found leakrate.

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 2	— 0 0 7	— 0 1	0 3 OF 0 9

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

Cause of Event:

The cause for the individual valve failures are described below:

- 1-WCR-955 - was not fully closed during the initial Leakrate Test. The valve internals were found to be in good working order, however, the adjustment bushing and associated lock-nut were found out of position, preventing the valve from fully closing. We could not determine when or why the adjustment bushing and lock-nut were incorrectly positioned.

- 1-SM-1 - The reason for the failure could not be determined. The swing check valve internals are connected to the valve bonnet. If there was any internal binding of the flapper it may have been relieved when the bonnet was removed. Upon inspection, all parts were determined to be serviceable and a blue check verified a 100 percent contact between the disc and seat.

  The previous Leakrate Test performed in 1990 indicates that the valve functioned correctly at that time. A review of the maintenance history sine the 1990 test date indicates that no work had been performed on this valve since the last Refueling Outage.

- 1-R-157 - The failure was caused by scale on valve seat and a small crack in the surface of the neoprene seat.

- 1-VCR-101 - The valve actuator stop screw prevented the valve from fully closing. There was no debris or dirt found on the valve seat that would have prevented proper closing of the valve. The stop screw limits the closing motion of the actuator and prevented full closure of 1-VCR-101. The previous Leakrate Test on this valve was performed in December of 1990, and was acceptable. The valve actuator was rebuilt prior to the December 1990 test and adjustments to the stop screw were made at that time. We believe that the last adjustment to the stop screw resulted in the valve barely seating and did not ensure a positive closing force on the valve seat. This allowed the valve to meet the leakrate acceptance criteria, but did not ensure continued operability of 1-VCR-101. An Engineering Review of the procedures used for overhauling of the air actuators revealed that there was insufficient guidance for setting of the stop screw.

- 1-CS-442-2 and 1-CS-442-3 - Work activities on these valves preceded the testing due to a lack of notification in changes to the schedule. The valve repair activities were originally scheduled to be performed in the correct sequence. The testing activity was subsequently rescheduled during the outage and since the two activities were not linked together, the testing was not performed until after the valve repairs.

  The testing activity referenced the Containment Penetration and not the Containment Isolation Valves being tested. Consequently the rescheduling of the Leakrate Test for the penetration was not tied to scheduled repair activities for these valves and was not realized by the person doing the scheduling.

A review of the past performance of these valves indicate that no adverse failure trends exist.

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 2	— 0 0 7	— 0 1	0 4 OF 0 9

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

Analysis of Event:

The Type B and C Leakrate exceeded the Technical Specification Limit of 0.6 La on July 5, 1992. A four hour phone report was made to the NRC as required by 50.72(b)(2), at 0829 hours on July 5, 1992. In addition, the total as-found leakrate is in excess of Technical Specification 3.6.1.2 and therefore, is reportable per 10CFR50.73(a)(2)(i)(B).

A technical evaluation has determined that these deficiencies would not have resulted in any additional leakage from Containment. The analysis is limited to the identified valve failure, and does not analyze for an additional failure on the affected penetrations. This condition did not have a significant impact on the health and safety of the public.

A review of the individual valves is described below:

- 1-WCR-955 - This is an NESW from RCP #1 Motor Cooler loop which is designated as a Class A line requiring dual isolation valves. Type C test results of the second series connected isolation valve 1-WCR-945 was found to have no leakage. Therefore, it can be concluded that Containment isolation was assured, and that the leakage from the subject valve would not have contributed significantly to the total leakage.

- 1-SM-1 - The As-Found leakage of this valve was unquantifiable. This is an air particulate radiation gas monitor line which is designated as a Class A line requiring dual isolation valves. Type C test results of the second connected series isolation valve, 1-ECR-36, was found to have no leakage. Therefore, it can be concluded that Containment isolation was assured, and that the leakage from the subject valves would not have contributed significantly to the total leakage.

- 1-R-157 - The As-Found leakage of this valve was unquantifiable. This is a glycol return expansion line which is designated as a Class A line requiring dual isolation valves. Type C test results of the second in-series isolation valve (1-VCR-20) was found to have an acceptable leak rate (70 sccm). Therefore, it can be concluded that Containment isolation was assured and that the leakage from the subject valves would not have contributed significantly to the total leakage.

- 1-VCR-101 - The As-Found leakage of this valve was unquantifiable. This is a Containment Ventilation Instrument Room supply line which is designated as a Class A line requiring dual isolation valves. Type C test results of the second connected series isolation valve (1-VCR-201), with 1-VCR-101 blanked off, had an acceptable leak rate (25 sccm). Therefore, it can be concluded that Containment isolation was assured and that the leakage from the subject valves would not have contributed significantly to the total leakage.

- 1-CS-442-2 - The As-Found leakage of this valve was not measured. This is an RCP #2 seal water line which is designated as a Class D line requiring a single isolation valve since the system is in operation

Analysis of Event continued:

following an accident. This check valve would not be required to isolate the Containment during a design basis accident since the Reactor Coolant Pump Seal Water Injection System remains in service, providing a high pressure positive leakage control, which would inject water into the Containment at pressure higher than the DBA value. Therefore, the previous (1990 value) recorded As-Left leakage was acceptable (50 sccm) and provides a reasonable As-Found value for assessing functionability of the valve. A review of the work package was performed to determine the as-found condition of the valve. The repair activity was initiated to repair a small body-to-bonnet leak. When the valve was disassembled, a blue check was performed and indicated that a good seating surface was available. Therefore, the work performed should not have influenced the ability of the valve to perform its design function. Based on this assessment, it can be concluded that the functionability of the valve is acceptable.

- 1-CS-442-3 - The As-Found leakage of this valve was not measured. This is an RCP #3 seal water line which is designated as a Class D line requiring a single isolation valve since the system is in operation following an accident. This check valve would not be required to isolate the Containment during a design basis accident since the Reactor Coolant Pump Seal Water Injection System remains in service, providing a high pressure positive leakage control which would inject water into the Containment at pressure higher than the DBA value. Therefore, the previous (1990 value) recorded As-Left leakage was acceptable (70 sccm) and provides a reasonable As-Found value for assessing functionability of the valve. A review of the work package was performed to determine the as-found condition of the valve. The repair activity was initiated to repair a small body-to-bonnet leak. When the valve was disassembled, a blue check was performed and indicated that a good seating surface was available. Therefore, the work performed should not have influenced the ability of the valve to perform its design function. Based on this assessment, it can be concluded that the functionability of the valve is acceptable.

Based upon the above review, it can be concluded that the observed conditions did not create a significant safety concern. That is, the As-Found condition of the Containment Isolation Valves would not have put the Plant in an unsafe condition.

Additionally, the leakages were reviewed for increased risk potential, and for impact on the Cook Nuclear Plant Probabilistic Risk Assessment (PRA). Based on this review, the leakages were not found to have any impact on the calculated core damage frequency of 6.26 E-05/year. However, failure probability to isolate Containment Post-Accident did increase from 3.031E-04 to 3.191E-04 per year. Multiplying the latter value by 6.26E-05/year yields 2.0E-08/year, which is an insignificant value representing chances of core damage followed by failure to isolate Containment. Thus, it is concluded that the impact on off-site consequences and hazards to the general public are insignificant from the PRA standpoint.

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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

Corrective Actions:

The corrective actions taken for the individual valves are listed below:

- 1-WCR-955 - The stop nuts were adjusted to allow the valve to close. Following reassembly a subsequent Leakrate Test indicated an acceptable leak rate.
- 1-SM-1 - The valve internals were cleaned and reassembled. A subsequent leakrate test indicated an acceptable leak rate.
- 1-R-157 - The valve internal were cleaned and the neoprene seat was reversed. A subsequent leakrate test indicated an acceptable leak rate.
- 1-VCR-101 - The valve seat was cleaned and the stop screw was repositioned all the way out. The position of the stop screw was checked on several other similar valves. In each case the stop screw was adjusted all the way out. A subsequent leakrate test indicated an acceptable leakrate. Based on an Engineering Review, the Plant Procedures for refurbishment of the air actuators have been modified to require the air side stop be positioned all the way out to ensure proper seating of the valve.
- 1-CS-442-2 and 1-CS-442-3 - A Scheduling Software interface is under revision to enhance identification of preceding and succeeding requirements for the work order activities. The change will allow for the scheduling person to monitor the prerequisite items to satisfy the conditions required for the work as well as schedule those actions required at the completion of the primary activity. The upgrade is scheduled for implementation by June 30, 1993.

Failed Component Identification:

Component Name: Containment Lower Compartment Radiation Detectors ERS-1300 and ERS-1400 Sample Return Header Check Valve (1-SM-1)

Manufacturer: ALOYCO

Model: 374

EIIS Code: ISV/IL

Component Name: Ice Condenser Refrigeration Glycol Return Header Containment Isolation Valves Pressure Relief Header Check Valve (1-R-157)

Manufacturer: KEROTEST

Model: 9645

EIIS Code: ISV/BC

NRC FORM 366A (6-89)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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

Failed Component Identification continued:

Component Name: Containment Instrumentation Room Purge Supply Train A Containment Isolation Valve (1-VCR-101)

Manufacturer: CLOW

Model: Style 7610 - 14"

EIIS Code: ISV/BD

Component Name: Non-Essential Cooling Water Return for No. 1 Reactor Coolant Pump Motor Cooler (1-WCR-955)

Manufacturer: ITT Grinnell

Model: 32101

EIIS Code: ISV/KG

Component Name: Reactor Coolant Pump Seal Water Injection to Reactor Coolant Pump-2 Containment Isolation Check Valve (1-CS-442-2)

Manufacturer: Conval, Inc.

Model: 12C2

EIIS Code: ISV/AB

Component Name: Reactor Coolant Pump Seal Water Injection to Reactor Coolant Pump-3 Containment Isolation Check Valve (1-CS-442-3)

Manufacturer: Conval, Inc.

Model: 12C2

EIIS Code: ISV/AB

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
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TEXT (If more space is required, use additional NRC Form 366A's) (17)

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

NRC FORM 366A
(6-89)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)
FAILURE CONTINUATION

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

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH 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.

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 2	— 0 0 7 —	0 1	0 9 OF 0 9

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	A B	I S V	C 6 3 1								
X	A B	I S V	C 6 3 1								