Commonwealth Edison
Dresden Nuclear Power Station
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January 14, 1991

EDE LTR #91-031

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

Licensee Event Report #90-018, Docket #050237 is being submitted as required by Technical Specification 6.6, NUREG 1022 and 10CFR 50.73(a)(2)(i)(b), 10CFR50.73(a)(2)(v).

E.D. Eenigenburg
Station Manager
Dresden Nuclear Power Station

EDE/dal

Enclosure

cc: A. Bert Davis, Regional Administrator, Region III
File/NRC
File/Numerical

ZDVR/122

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LICENSEE EVENT REPORT (LER)

Form Rev 2.0

Facility Name (1)	Docket Number (2)	Page (3)
Dresden Nuclear Power Station, Unit 2	0 5 0 0 0 2 3 7	1 of 0 5

Title (4) Leakage Path Discovered During Primary Containment ILRT Due to Management Deficiency

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 2	1 8	9 0	9 0	0 1 8	0 0	0 1	1 4	9 1	N/A	

OPERATING MODE (9): N

POWER LEVEL (10): 0 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR
(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)	X 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)
___ 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. Geiger, Technical Staff Engineer Ext. 2610	8 1 5	9 4 2 - 2 9 2 0

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

SUPPLEMENTAL REPORT EXPECTED (14)		Expected Submission Date (15)	Month	Day	Year
X Yes (If yes, complete EXPECTED SUBMISSION DATE)	___ NO		0 3	0 1	0 1

ABSTRACT (Limit to 1400 spaces, i.e, approximately fifteen single-space typewritten lines) (16)

On December 17, 1990, while performing a primary containment integrated leak rate test (ILRT) during the Unit 2 refuel outage, leakage in excess of the Technical Specification 3.7.A.2 ILRT requirement was measured due to a leaking reactor building to pressure suppression chamber vacuum breaker valve A02-1601-20A inboard flange. Further review on December 18, 1990, indicated that this vacuum breaker had been replaced during the previous refuel outage without proper testing of the inboard flange connection; 10 CFR50.72 notification was then completed. Although this degraded condition potentially existed during the previous operating cycle, the secondary containment would have mitigated release to the environs under postulated design basis accident conditions. Analysis of 10CFR 100 requirements is in progress, and a supplemental report will provide further information. The ILRT was completed satisfactorily after the flange was tightened. The underlying cause for not challenging this pathway upon earlier replacement of the vacuum breaker was attributed to management deficiency in that performance of an ILRT following this activity was not properly specified or identified. A previous Dresden Unit 2 ILRT failure due to unrelated causes is reported by LER 83-29/050237.

PLANT AND SYSTEM IDENTIFICATION:

General Electric - Boiling Water Reactor - 2527 MWt rated core thermal power

Nuclear Tracking System (NTS) tracking code numbers are identified in the text as (XXX-XXX-XX-XXXXX)

EVENT IDENTIFICATION:

Leakage Path Discovered During Primary Containment ILRT Due to Management Deficiency

A. CONDITIONS PRIOR TO EVENT:

Unit: 2	Event Date: December 18, 1990	Event Time: 0800 Hours
Reactor Mode: N	Mode Name: Shutdown	Power Level: 0%

Reactor Coolant System (RCS) Pressure: 48 psig

B. DESCRIPTION OF EVENT:

On 12/17/90 at 0615 hours, with Unit 2 in Cold Shutdown for a refuel outage, during the pressurization phase of Dresden Technical Staff Surveillance (DTS) 1600-7, Integrated Primary Containment Leak Rate Test (ILRT), a substantial leak was discovered on the inboard flanged connection of reactor building to suppression chamber vacuum breaker A02-1601-20A [BF]. The containment pressurization phase of the ILRT initially began at 0500 hours on 12/17/90. At 0530 hours, members of the Technical Staff were sent out to begin looking, listening and inspecting for any leakage from the Primary Containment [NH] per DTS 1600-7. At 0615 hours, two of the Technical Staff personnel heard a loud pop followed by a continuous loud buzzing sound emanating from the pressure suppression chamber catwalk area. A closer inspection revealed that the containment side flanged connection of valve A02-1601-20A was leaking. This air operated valve is normally closed and is inboard to check valve 2-1601-31A in the reactor building to pressure suppression chamber vacuum relief line. This valve is designed to open when pressure suppression chamber vacuum reaches 0.5 psig with respect to the reactor building atmosphere, thus preventing structural damage to the pressure suppression chamber by insuring its external pressure limits are not exceeded. The Operations Department was then notified and at 0630 hours the ILRT air compressors were secured with primary containment pressure at approximately 15 psig. Although the leakage from the flanged connection was not directly quantifiable, it was estimated that this particular leak exceeded the Technical Specification 3.7.A.2. 0.75 La ILRT criteria; consequently, the total "As Found" ILRT leak rate was also considered to be in excess of 0.75 La.

At 0800 hours with the containment pressure still at approximately 15 psig, Mechanical Maintenance and Technical Staff personnel attempted to reduce the leakage on the in-board side flanged connection of valve A02-1601-20A by tightening the flange bolts. At 0900 hours, the leakage was completely stopped. A soap bubble solution test on the suspect flange verified zero leakage. At 1010 hours, the ILRT air compressors were restarted and the containment pressurization resumed. The test continued with satisfactory results ("As Left" ILRT results within 0.75 La).

On 12/18/90, while the ILRT was still in progress, a review of the maintenance history of valve A02-1601-20A revealed that it had been installed as a new valve during the previous refueling outage. Following this replacement, a post maintenance test properly challenging the inboard flanged connection was not performed. Both Work Request 67528 and the procedure used for the replacement, Dresden

Maintenance Procedure (DMP) 1601-2, Drywell and Pressure Suppression Chamber Air Operated Butterfly Valve Maintenance, required only a Local Leak Rate Test (LLRT); the LLRT for valve A02-1601-20A was performed by pressurizing the volume of piping between valves A02-1601-20A and 2-1601-31A (see Figure 1) with air to 48 psig and monitoring the makeup flow required to maintain that pressure. With this LLRT configuration, the inboard flanged connection is not challenged. Consequently, since this valve was not removed since the last refueling outage, it was determined that this particular flanged connection was potentially in a degraded condition during the past operating cycle. At 0800 hours on 12/18/90 it was determined that this event was reportable under 10CFR 50.72 (b)(2)(i) requirements. The event was also classified as a Potentially Significant Event (PSE), initiating prompt notification of all Commonwealth Edison sites.

C. APPARENT CAUSE OF EVENT:

This report is submitted in accordance with 10CFR 50.73 (a)(2)(i)(b) (condition prohibited by the Technical Specifications) and 10CFR 50.73 (a)(2)(v) (condition preventing proper fulfillment of the primary containment function).

The cause of the leakage from the inboard flanged connection of valve A02-1601-20A was inadequate tightening at the flange bolts following the replacement activity. Review of the work package for the replacement activity shows that the flange bolts were first snugged to insure gasket contact and then several passes were made in order to tighten them. The work package allowed use of an impact wrench or full effort of a man using an ordinary spud wrench for the final tightening. A Quality Control Inspector physically observed this activity. Additional passes with a slugging wrench were performed on 12/18/90 to correct the leakage.

This condition was not discovered due to an ineffective post maintenance test performed after the valve was replaced during the previous refueling outage. The LLRT performed following the replacement never challenged the inboard flange; had a proper pressure test on the inboard flange been performed, the leakage from the flange would have been discovered, and the appropriate corrective actions taken prior to entering a mode of operation requiring primary containment integrity. It should be noted that this flanged connection configuration is not currently testable via an LLRT.

Investigation indicated that at the time the valve replacement work package and test requirements were prepared and reviewed, this concern was not identified. Therefore, the underlying root cause was attributed to management deficiency. Procedure deficiency was also a contributing factor in that DMP 1601-2 did not require testing of the inboard connection or make mandatory the use of a slugging or impact wrench for final tightening of the flange bolts.

D. SAFETY ANALYSIS OF EVENT:

The safety significance of this event is mitigated by the integrity of Secondary Containment [NG] and the function of the Standby Gas Treatment System (SGTS) [BH]. The SGTS is used to maintain a slight negative pressure in the Reactor Building during accident conditions. Filters are provided in the system to remove radioactive particulates, and charcoal adsorbers are provided to remove radioactive halogens which may be present in concentrations significant to environmental dose criteria.

Calculations are currently underway to determine the off-site dose rates that would have occurred had this flange leaked to the calculated magnitude following a Loss of Coolant Accident (LOCA). Additionally, calculations are being made to verify the Control Room habitability following an event of this type. Upon receipt of these calculations and reports, a supplemental report will be submitted.

E. <u>CORRECTIVE ACTIONS</u>:

1. A supplemental report will be submitted in order to provide the results of the 10CFR 100 analyses (237-200-90-15201).

2. DMP 1601-2 will be revised by the Maintenance Staff to include requirements for use of an impact or slugging wrench for final tightening of the flange bolts and performance of a LLRT or comparable test on the inboard flange of this valve and other similar valves (1601-20B, 21, 23, 56, and 60) whenever the integrity of these connections is disturbed (237-200-90-15202).

3. Currently, modifications are being evaluated which will allow localized testing of the subject flanges. The final design/decision will be reported in the supplement to this report (237-200-90-15203).

F. <u>PREVIOUS OCCURRENCES</u>:

<u>LER/Docket Numbers</u>	<u>Title</u>
83-29/050237	Excessive Leakage Discovered During ILRT
	During this event, excessive leakage was experienced from the shaft seals of several Pressure Suppression Chamber to Drywell Vacuum Breakers. The leakage from these penetrations was later determined to be 1178 scfh (20 scfm). Based upon calculations performed for this event, the 10 CFR 100 release limits were not violated. The cause was found to be the installation of shaft seals which provided inadequate sealing under sustained pressure conditions.

G. <u>COMPONENT FAILURE DATA</u>:

<u>This section is not applicable as the leakage was cause by inadequate tightening of the flange bolts.</u>

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

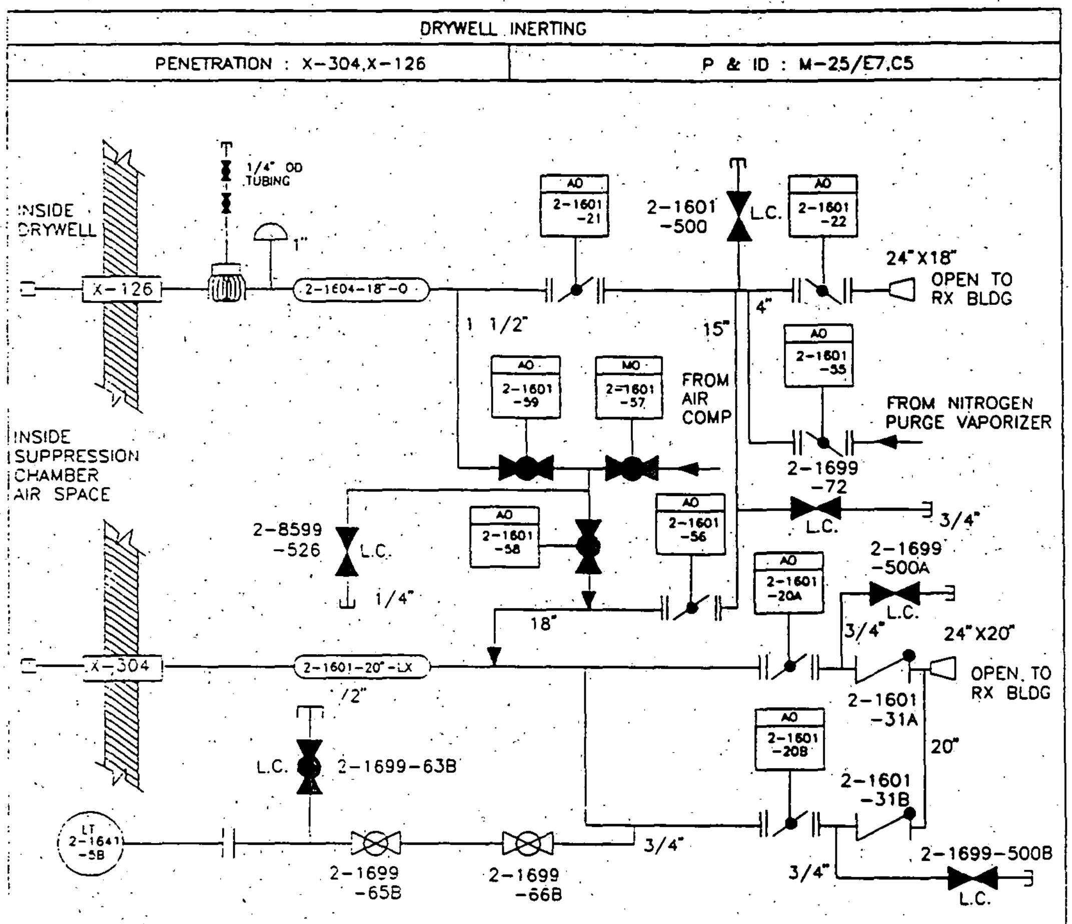

FIGURE 1