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Dresden Nuclear Power Station
R.R. #1
Morris, Illinois 60450
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August 6, 1991

EDE LTR #91-390

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). This revised report provides an update on corrective actions committed to in the original report.

E. D. Eenigenburg
Station Manager
Dresden Nuclear Power Station

EDE/ade

Enclosure

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

(ZDVR/247)



TE22

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 7 |

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 1 | 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 |
|---|---|
| J. Geiger, Technical Staff Engineer Ext. 2610 | AREA CODE 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 | MANUFAC-TURER | REPORTABLE TO NPRDS | CAUSE | SYSTEM | COMPONENT | MANUFAC-TURER | REPORTABLE TO NPRDS |
|---|---|---|---|---|---|---|---|---|---|
| | | | | | | | | | |
| | | | | | | | | | |

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)


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. 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 appropriate testing 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.

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

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. 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 provide adequate criteria for adequate tightness of the bolting.

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.

A study was performed to calculate the as found leakage rate past the valve flange and the off-site and control room doses that would have occurred had the leak occurred during a Design basis Loss of Coolant accident (LOCA). The measured leak rate at a containment pressure of 14.6 psig was determined to be 24.6 weight %/day. This value was then extrapolated to determine the leakage rate at design basis pressure (48 psig). Using extrapolation methods which are valid for the turbulent flow regime (as opposed to choked or laminar flow conditions), the as found leak rate at 48 psig was conservatively calculated at 31 weight %/day.

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The off-site and control room doses were determined to be within the limits specified in 10 CFR Part 100 and General Design Criteria 19 respectively. The assumptions and methodology used for the analysis were the same as the design basis LOCA analysis accepted by the NRC in 1982 for the Dresden Unit 2 SEP, except for the following three differences:

1. The treatment of the fission product holdup in the secondary containment,

2. the effect of iodine retention in the suppression pool, and

3. the efficiency of the SBGTS.

The iodine retention was calculated using the NRC Staff's recently issued Standard Review Plan 6.5.5. Fission product holdup used the more realistic, but conservative treatment accepted by the NRC in the 1983 control room habitabiity SER for Dresden. SBGTS efficiency used a conservative value of 98% which is supported by actual previous Dresden test results.

Utilizing these assumptions, the following doses were calculated for a leak rate of 31 weight %/day.

| | Boundary (See Note 1) | Low Population Zone (LPZ)(See Note 1) | Control Room (See Note 2) |
|---|---|---|---|
| Thyroid Dose: | 1.6 Rem | 39.2 Rem | 18.7 Rem |
| Whole Body Dose: | 1.3 Rem | 1.2 Rem | 1.4 Rem |
| Skin Dose: | N/A | N/A | 24.6 Rem |

Notes

1. The 10 CFR100 limits for Thyroid dose for both the 2 hour site boundary and the 30 Day LPZ is 300 Rem. The 10 CFR 100 limits for Whole Body dose for both the 2 hour site and the 30 day LPZ is 25 Rem.

2. The 10 CFR 50, Appendix A, Criterion 19 limits for Thyroid dose and Whole Body Dose are 30 Rem and 5 Rem respectively.

Although NRC calculations and assumptions differ somewhat with these results, they conclude that 10CFR100 and GDC 19 limits would not have been exceeded.

E. CORRECTIVE ACTIONS:

1. DMP 1601-2 will be revised by the Maintenance Staff to include torque valves 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).

2. The Station Work Analyst pre-job checklist has been revised to ensure adequate guidance is provided in work packages concerning bolting requirements. Existing maintenance procedures will be reviewed for adequacy of bolting instructions as they are incorporated into work packages.

3. A Nuclear Operations Directive (NOD) on bolting practices is currently being developed. This NOD is based on the Electric Power Research Institute (EPRI) manual, "Good Bolting Practices." The information in the EPRI manual has been provided to the maintenance work analysts to use as guidance in work package preparation pending issuance of the final NOD.

4. Possible modifications which would allow localized Post Maintenance LLRT of the inboard butterfly valve flanges in the post accident direction have been evaluated. The proposed configuration will make use of a custom designed, light weight, blind flange. Slip-on type flanges would be permanently welded to the outside diameter of each pipe end and then the blind flange would be attached for the LLRT. A test tap connection through the blind flange would allow pressurizing the applicable pipe segments up to the subject valves, thus challenging the valves and flanges in the proper post-accident direction.

5. An LLRT and ILRT requirements booklet is being developed by the Technical Staff which provides a comprehensive overview of all the leak rate testable boundaries (237-200-90-15203). The booklet contains color coded P&IDs which highlight the ILRT and LLRT boundaries, along with a generic description of the testing requirements for each type of boundary. The booklet, which will be implemented prior to the upcoming Unit 3 refuel outage, is intended as an easy-to-use guide in which maintenance and operating personnel can use to easily identify LLRT and ILRT boundaries.

6. Dresden Administrative Procedure (DAP) 14-5, "Leak Rate Testing Program", will be revised by the Technical Staff prior to the upcoming Unit 3 refuel outage (237-200-90-15204) to caution against inappropriate application of standard periodic LLRT lineups for PMTs.

7. The following activities were implemented to address issues of appropriate periodic testing of certain containment valves and flanges to meet Appendix J requirements:

    1. A meeting with Nuclear Reactor Regulation (NRR) was held to discuss the Commonwealth Edison and the Boiling Water Reactor Owners Group (BWROG) positions, on the testing of these valves and flanges (237-200-90-15205).

    2. Until resolution of this issue is reached, a substitute test will be performed each refuel outage to properly challenge the susject flanges to the design accident pressure (237-200-90-15206).

8. An updated response to Information Notice No. 86-16, Failures to Identify Containment Leakage Due to Inadequate Local Testing of BWR Vacuum Relief Systems Valves, has been completed addressing the plans to test the subject flanges.

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>

| FACILITY NAME (1) | DOCKET NUMBER (2) | LER NUMBER (6) Year | | Sequential Number | | Revision Number | Page (3) |
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| Dresden Nuclear Power Station | 0 5 0 0 0 2 3 7 | 9 0 | - | 0 1 8 | - | 0 1 | 0 7 OF 0 7 |

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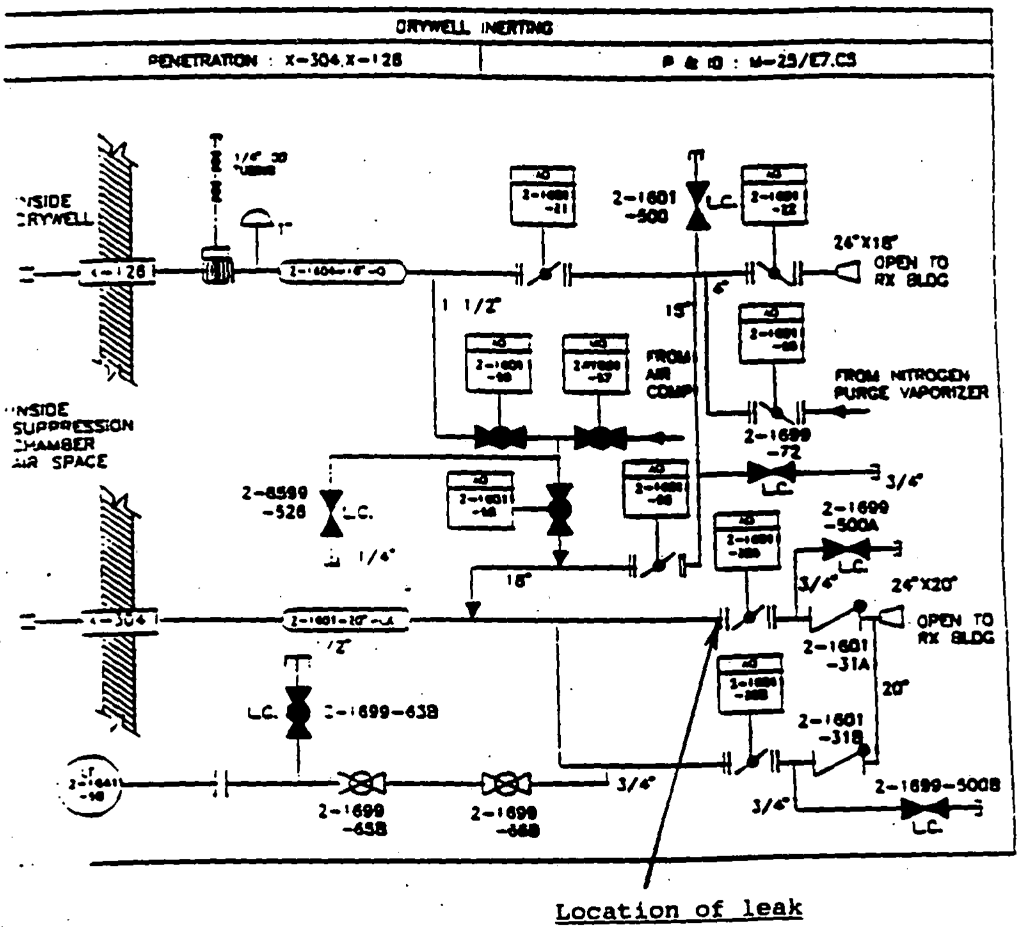


**FIGURE 1**