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Dresden Nuclear Power Station
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February 3, 1994

GFSLTR 94-0038

U.S. Nuclear Regulatory Commission
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Washington, D. C. 20555

Licensee Event Report 94-001, Docket 50/237 is being submitted as required by Technical Specification 6.6, NUREG 1022 and 10 CFR 50.73(a)(2)(ii).

for 2-3-94

Gary F. Spedl
Station Manager
Dresden Station

GFS/cfq

Enclosure

cc: J. Martin, Regional Administrator, Region III
NRC Resident Inspector's Office
File/NRC
File/Numerical

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LICENSEE EVENT REPORT (LER)

Form Rev 2.0

Facility Name (1)	Docket Number (2)	Page (3)
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Title (4)

Process Line Primary Containment Isolation Valves Never Subjected to Type C Local Leak Rate Test 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)
0 1	0 5	9 4	9 4	—	0 0 1	—	0 0	0 2	0 4	9 4	N/A N/A	

OPERATING MODE (9)	N	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIRMENTS OF 10CFR (Check one or more of the following) (11)

POWER LEVEL (10)						
1 0 0		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)
		20.405(a)(1)(iii)		50.73(a)(2)(i)		50.73(a)(2)(viii) (A)		
		20.405(a)(1)(iv)	X	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)		

LICENSE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER	
	AREA CODE	
M. McGivern, Local Leak Rate Test Coordinator Ext. 2526	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)				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)

At approximately 1100, on January 5, 1994 with Unit 2 operating at 100% power, as a result of a walkdown of primary containment penetration X-146, it was determined that a process line, which taps off an instrument line upstream of the excess flow check valve, had primary containment isolation valves which had never been given a proper 10 CFR 50, Appendix J Type C Local Leak Rate Test (LLRT). Upon identification of this problem, the instrument line was isolated at the containment penetration rendering the 2A Core Spray loop line break detection instrumentation inoperable. Subsequently, the 2A Core Spray loop was declared inoperable which placed Unit 2 in a 7 day Limiting Condition for Operation (LCO). The process line, which was used to obtain Reactor Coolant Samples, had rarely been used since Unit 2 Start Up Testing. This sample line, which did not show up on a Piping and Instrumentation Diagram, was determined to be an undocumented plant modification. The safety significance of the leakage past the sample line isolation valves was minimal, since the isolation valves leaked 0 scfh when given a Type C LLRT. The leakage out of containment, as determined on a minimum pathway basis, would not cause the maximum off-site dose rates established in 10 CFR 100 to be exceeded in the event of a LOCA. The sample line was subsequently cut and capped and the instrument line placed back in service.

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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:

Process Line Primary Containment Isolation Valves Never Subjected to Type C Local Leak Rate Test due to Management Deficiency

A. CONDITIONS PRIOR TO EVENT:

Unit: 2 Event Date: 01/05/94 Event Time: 1100 hrs.

Reactor Mode: N Mode Name: Run Power Level: 100%

Reactor Coolant System (RCS) Pressure: 990 psig

B. DESCRIPTION OF EVENT:

At approximately 1100, on January 5, 1994 with Unit 2 operating at 100% power, as a result of a walkdown of primary containment penetration X-146, it was determined that a process line, which taps off an instrument line upstream of the excess flow check valve, had primary containment isolation valves which had never been given a proper 10 CFR 50, Appendix J Type C test. See Figure 1 on page 5. The discovery was the result of a self-imposed walkdown of Unit 2 containment penetrations by the LLRT Coordinator in response to a previous discovery of abandoned instrument tubing through penetration X-135E.

The Shift Engineer was notified and a Problem Identification Form (PIF) was initiated per Dresden Administrative Procedure (DAP) 02-27, Integrated Reporting Process. An ENS phone notification was then made at 1514 Eastern Standard Time on Wednesday January 5, 1994 to report a condition outside the design basis of the plant. FSAR Section 6.2.6 states that Local Leak Rate Testing (LLRT) will be performed in accordance with 10 CFR 50, Appendix J. The instrument line was isolated at the containment penetration rendering the 2A Core Spray loop line break detection instrumentation inoperable. Subsequently, the 2A Core Spray loop was declared inoperable which placed Unit 2 in a 7 day Limiting Condition for Operation (LCO). The process line, which was used to obtain Reactor Coolant Samples, had rarely been used since Unit 2 Start Up Testing. This sample line, which did not show up on a Piping and Instrumentation Diagram, is believed to be an undocumented plant modification dating from original plant startup. After isolation of the instrument line at containment penetration X-146, the instrument and sample lines were drained and the primary containment isolation valves were given a 10 CFR 50, Appendix J Type C LLRT which resulted in no leakage from containment.

Under Work Request D23612, the sample line was cut between the first and second isolation valves and a cap welded in place. The welded cap was subjected to a simulated 10 CFR 50, Appendix J Type A test and an inservice leak test; both of which demonstrated no leakage. The instrument line isolation valve was reopened, 2A Core Spray loop was declared operable on January 9, 1994 and the LCO was terminated.

C. APPARENT CAUSE OF EVENT:

This report is being submitted in accordance with 10 CFR 50.73(a)(2)(ii) which requires the reporting of any condition that was outside the design basis of the plant.

The root cause of not performing an LLRT of the primary containment isolation valves on the sample line was due to a management deficiency of not having documented a plant modification. This deficiency is believed to be from original plant startup.

In addition, 10 CFR 50, Appendix J states:

> "Type C Tests" means tests intended to measure containment isolation valve leakage rates. The containment isolation valves included are those that:
>
> 1. Provide a direct connection between the inside and outside atmospheres of the primary reactor containment under normal operation, such as purge and ventilation, vacuum relief, and instrument valves. etc.

Safety Guide 11, now known as Reg. Guide 1.11, recommended that instrument lines have a self-actuated excess flow check valve installed to minimize containment leakage. Using the aforementioned references, an exemption from performing Type C tests on normally open manual isolation valves upstream of instrument line excess flow check valves was requested and subsequently granted by the NRC.

Based upon this testing exemption, an interpretation that all instrument lines are exempt from 10 CFR 50, Appendix J Type C testing was incorrectly made. This is a contributing factor that resulted in a walkdown of primary containment instrument penetrations being excluded from a containment overview performed during 1989.

D. SAFETY ANALYSIS OF EVENT:

The safety significance of the leakage past the Reactor Coolant Sample Isolation Valves was minimal since the isolation valves leaked 0 scfh. Since the current as-left leakage (Type A test) of .6706 wt%/day is still less than the Technical Specification limit of $0.75L_a$ (1.2 wt%/day), the maximum off-site dose rates established in 10 CFR 100 would not be exceeded in case of a LOCA.

E. CORRECTIVE ACTIONS:

The current modification process is controlled by Dresden Administrative Procedure (DAP) 05-01, Plant Modification Program. With this program in place, plant changes and design documentation would have prevented occurrence.

Under Work Request D23612, the sample line was cut between the first and second isolation valves and a cap welded in place. The welded cap was subjected to a simulated 10 CFR 50, Appendix J Type A test and an inservice leak test; both of which demonstrated no leakage. The 2A Core Spray loop was declared operable on January 9, 1994 and the LCO was terminated.

In order to ensure no other similar conditions exist, every Unit 2 primary containment penetration will be walked down. All Unit 2 accessible containment penetrations have been walked down with no similar conditions identified. All inaccessible penetrations will be walked down prior to the end of the next Unit 2 refuel outage. (237-201-93-43201)

Unit 3 was walked down and a similar problem was discovered. This is described in LER/Docket Number 94-002/0500249.

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

<u>LER/Docket Numbers</u>	<u>Title</u>
92-016/0500237	Unchallenged Primary Containment Due to Management Deficiency

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

No Component Failure.

LICENSEE EVENT REPORT (LER) TEXT CONTINUAT

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Figure 1