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Dresden Nuclear Power Station
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October 14, 1992

CWS LTR #92-622

U.S. Nuclear Regulatory Commission
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Licensee Event Report 92-30, Docket 050237 is being submitted as required by Technical Specification 6.6, NUREG 1022 and 10 CFR 50.73(a)(2)(ii).

Signature

Charles W. Schroeder 10/16/92
Station Manager
Dresden Nuclear Power Station

CWS/jmt

Enclosure

cc: A. Bert Davis, Regional Administrator, Region III
NRC Resident Inspector's Office
File/NRC
File/Numerical

(ZDVR/768)

IE22

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 4

Title (4) Pipe Supports for the Containment Atmosphere Sampling System not connected to Structural Steel due to Design/Installation 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)
									N/A	0 5 0 0 0
0 9	1 8	9 2	9 2	0 3 0	0 0	1 0	1 4	9 2	N/A	0 5 0 0 0

OPERATING MODE (9)	N	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR (Check one or more of the following) (11)					
POWER LEVEL (10)	0 8 9	___ 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		
		___ 20.405(a)(1)(iii)	___ 50.73(a)(2)(i)	___ 50.73(a)(2)(viii)(A)	in Abstract		
		___ 20.405(a)(1)(iv)	X 50.73(a)(2)(ii)	___ 50.73(a)(2)(viii)(B)	below and in		
		___ 20.405(a)(1)(v)	___ 50.73(a)(2)(iii)	___ 50.73(a)(2)(x)	Text)		

LICENSEE CONTACT FOR THIS LER (12)

Name	TELEPHONE NUMBER
	AREA CODE
Mark J. McGivern, Technical Staff Engineer 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)

On September 18, 1992, during normal Unit 2 operation at 89% rated core thermal power, two pipe supports for the Return Line from the Containment Atmosphere Sampling System between the containment penetration X-204A and containment isolation valve FCV 2-8501-3B were found to not be connected to structural steel. The supports did not have any load bearing capacity. It was determined that this line (2-8501-1") did not meet FSAR stress limits in the existing configuration. However, it does meet operability stress limits. Engineering is in the process of designing new supports to return the system to the FSAR requirements. The system is currently in operation and installation of the design change will take place prior to the completion of the Unit 2 Refueling Outage (D2R13) currently scheduled to start January 1993. This report is being submitted to provide additional information regarding an ENS phone call made on September 18, 1992, at 1342 hours.

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:

Pipe Supports for the Containment Atmosphere Sampling System not connected to Structural Steel due to Design/Installation Deficiency

A. CONDITIONS PRIOR TO EVENT:

Unit: 2 Event Date: September 18, 1992 Event Time: 1250 Hours

Reactor Mode: N Mode Name: Run Power Level: 89%

Reactor Coolant System (RCS) Pressure: 1017 psig

B. DESCRIPTION OF EVENT:

On September 18, 1992, at 1250 hours, during normal Unit 2 operation in the Run mode at 89% rated core thermal power, two pipe supports for the 1 inch diameter return line (2-8501-1") from the Containment Atmosphere Sampling System [IK] between the containment penetration X-204A and containment isolation valve FCV 2-8501-3B were found to not be connected to structural steel. The supports did not have any load bearing capacity. This discrepancy was found during a plant walkdown conducted by ABB Impell as part of the Dresden Piping and Instrument Diagram discrepancy walkdown. It was determined that this line did not meet FSAR stress limits in the existing configuration. However, based upon ABB Impell calculations 0591-630-03 and 0591-630-04, the pipe supports met operability criteria for pipe stress. Also, containment penetration X-204A was found by Sargent & Lundy to meet operability stress limits. An ENS phone call was made on Septemer 18, 1992 at 1342 hours. Engineering is in the process of designing new supports to return the system to the FSAR requirements. The system is currently in operation and installation of the design change will take place prior to the completion of the Unit 2 Refueling Outage (D2R13) currently scheduled to start January 1993.

C. APPARENT CAUSE OF EVENT:

This system was installed prior to Unit 2 Startup in 1972. No modifications performed on this system since then would have installed extra supports. Lack of documentation rendered the root cause of the support deficiency difficult to determine. Either a design deficiency or an installation deficiency coupled with a quality control failure were the root causes for the Containment Atmosphere Sampling System piping support deficiency. Three possible sequences of events leading to the support deficiency were explored using root cause analysis. The first situation is that the designers designed the correct pipe supports but that the installers deviated from the design and installed substandard supports. This situation would have an installation deficiency as the root cause. The second situation is that the designers designed substandard pipe supports and the installers, following the design, installed these substandard supports. This second situation would have a design deficiency as the root cause. Lastly, another situation explored is one in which no supports were designed and the installers built and installed pipe supports they thought were correct. The root cause of this situation would be attributed to a design deficiency. In all three of the series of events, a quality control inspection of the pipe supports failed to notice the substandard supports.

D. SAFETY ANALYSIS OF EVENT:

Line 2-8501-1" is the Return Line from the Containment Atmosphere Sampling System [IK]. There are two safety functions associated with this line.

First, the system automatically draws atmosphere samples from various elevations inside containment, analyzes the samples for oxygen content, and displays the results in the Control Room. The system is used to establish and maintain oxygen content inside the containment less than 4%. This low concentration ensures that the hydrogen and oxygen liberated in a Loss of Coolant Accident (LOCA) are unlikely to react with each other to cause a release of energy powerful enough to damage containment. This system is not operated during accident conditions since oxygen content inside the containment is measured by the Containment Atmosphere Monitor (CAM) system.

Second, because the sample return line penetrates primary containment, it is equipped with two containment isolation valves (FCV 2-8501-3A and FCV 2-8501-3B) in series that automatically close on a high drywell pressure signal or reactor vessel low water level. The integrity of the primary containment, including the primary containment isolation valves, helps to limit the values of off-site dose to less than those of 10 CFR 100 guidelines in the event of a break in the primary system piping.

Line 2-8501-1" was found to have two supports within the safety-related boundary that were unable to be FSAR qualified. The area of concern was whether a portion of the line could fail in such a manner as to breach primary containment during a Design Basis Earthquake due to inadequate support. However, based on ABB Impell calculations 0591-630-03 and 0591-630-04, the pipe supports meet operability criteria for pipe stress. Also, containment penetration X-204A was found by Sargent & Lundy to meet operability stress limits. Therefore, both the line and the penetration would not fail during a Design Basis Earthquake.

Since the system meets operability criteria, the safety significance of this event is considered to be minimal.

E. CORRECTIVE ACTIONS:

The Unit 3 Containment Atmosphere Sampling System piping was immediately walked down. No support deficiencies were found.

Engineering is in the process of designing new supports to return the system to FSAR requirements. Installation of the design change will take place prior to the completion of the Unit 2 Refueling Outage (D2R13) currently scheduled to start January 1993 (237-180-92-23901).

Barrier Analysis for root cause determination of the piping support deficiency demonstrated that using current Dresden procedures, a piping support deficiency could not occur. The purpose of an installer's walkdown is for designers to walkdown the actual placement of system components, piping, and supports. This walkdown would have determined if a support was needed when the designer may have thought a support was unnecessary. The Nuclear Station Work Procedure design checklist, along with the review of the design, would prevent a substandard support from being designed and subsequently installed. To prevent a correctly designed support from being installed incorrectly, there are clearly written Work Packages and a Maintenance and Modification Procedures (MMP's) checklist for the installers to follow during installation. In addition, Quality Control Inspectors, during the post-installation walkdown, use the MMP checklist to verify that the correct support is installed. Periodic Technical Staff Engineer walkdowns ensure the supports remain in their standard configuration.

F. PREVIOUS OCCURENCES:

LER/DOCKET NUMBER	Title
87-003/050237	Primary Containment Structural Steel Connections Outside FSAR Design Criteria Due to Apparent Original Construction Oversight
88-003/050249	Flued Head Anchor Supports in Excess of Design Criteria Due to Design and Construction Deficiencies

G. COMPONENT FAILURE DATA:

Since there were no component failures during this event, an industry wide NPRDS data base search was not performed.