Commonwealth Edison
Dresden Nuclear Power Station
R.R. #1
Morris, Illinois 60450
Telephone 815/942-2920

February 1, 1990

EDE LTR #90-116

U.S. Nuclear Regulatory Commission
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Licensee Event Report #90-001-0, Docket #050237 is being submitted as required by Technical Specification 6.6, NUREG 1022 and 10 CFR 50.73(a)(2)(iv).

E.D. Eenigenburg
Station Manager
Dresden Nuclear Power Station

EDE/jt

Enclosure

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

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LICENSEE EVENT REPORT (LER)

Form Rev 2.0

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Title (4)

Primary Containment Group I Isolation and Reactor Scram Due to Procedure 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 1	0 5	9 0	9 0	0 0 1	0 0	0 2	0 1	9 0		0 5 0 0 0

OPERATING MODE (9)	N
POWER LEVEL (10)	0 9 9

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR
(Check one or more of the following) (11)

	20.402(b)		20.405(c)	X	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)		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	
Joseph Welch, Technical Staff System Engineer Ext. 2666	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 2241 hours on January 5, 1990, with Unit 2 operating at 99.3% power, a Primary Containment Group I isolation and subsequent reactor scram occurred while an Instrument Mechanic (IM) was performing Dresden Instrument Surveillance (DIS) 250-1, Main Steam Line High Flow Isolation Switch Calibration. The root cause of the unplanned Group I isolation has been attributed to a procedural deficiency within DIS 250-1 such that inadequate controls were provided to prevent a potential pressure transient large enough to cause a trip on instruments in both trip channels, located on a common instrument sensing line header. Corrective action included a special test of the tripped high flow switch and isolation manifold, possible hardware changes to assist the IM in performing the surveillance, and a study to investigate feasibility of replacing the present switches with an analog trip. The safety significance of this event is minimal since the high flow switches responded as designed and tripped in response to pressure transient induced to the common header, thus initiating the automatic Group I isolation and reactor scram. A previous event was reported by LER 87-016/050249.

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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 (BWR) - 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:

Unit 2 Reactor Scram and Primary Containment Group I Isolation Due to a Pressure Transient on the Main Steam Line High Flow Switch Common Header During a Calibration Surveillance Due to a Procedure Deficiency.

A. CONDITIONS PRIOR TO EVENT:

Unit: 2 Event Date: January 5, 1990 Event Time: 2241 hours

Reactor Mode: N Mode Name: Run Power Level: 99.3%

Reactor Coolant System (RCS) Pressure: 1003 psig

B. DESCRIPTION OF EVENT:

At 2241 hours on January 5, 1990, with Unit 2 operating at 99.3% of rated core thermal power, an unplanned Primary Containment Group I Isolation [JM] occurred while an Instrument Mechanic (IM) was performing Dresden Instrument Surveillance (DIS) 250-1, Main Steam [SB] Line High Flow Isolation Switch Calibration and Functional Test. The Group I isolation signal initiated automatic closure of the Main Steam Isolation Valves (MSIVs) and other Group I isolation valves, and an automatic reactor scram. Subsequently, a Group II and Group III isolation and automatic start of the Standby Gas Treatment System (SBGTS) [BH] occurred as expected on a low reactor water level signal following the scram. The main generator [EL] Output Circuit Breakers (OCBs) were opened via Operator action approximately 3-1/2 minutes following the scram due to concern that the generator did not appear to trip within the automatic reverse power trip setpoints.

At 2246 hours, the Isolation Condenser [BL] was manually initiated for pressure control at 1000 psig increasing vessel pressure using clean demineralized water [KC] for make up of the shell side of the Isolation Condenser. The Reactor Water Clean-Up (RWCU) [CE] System was unisolated and restarted at 2250 hours. At 0016 hours on January 6, 1990, the Isolation Condenser was secured and the RWCU system flow was increased to 1000 GPM to assist in post scram decay heat removal. Subsequently, the MSIVs were unisolated and the reactor scram signal was reset. Prior to the Group I isolation, there were no systems or components inoperable that contributed to this event.

C. APPARENT CAUSE OF EVENT:

This event is being reported in accordance with Title 10 of the Code of Federal Regulations Part 50 Section 73(a)(2)(iv), which states that any event that results in manual or automatic actuation of any Engineering Safety Feature, including the Reactor Protection System (RPS), must be reported.

Based on a review of the event and interviews of the personnel involved, the root cause of the event has been determined to be a procedural deficiency within DIS 250-1. This procedure provided inadequate precautionary controls to prevent a potential pressure transient large enough to cause a trip on instruments in both trip channels, located on a common instrument sensing line header. It is believed that a pressure spike was induced into the common sensing line header when the instrument isolation valve was opened. The common instrument header supplies the process pressure which supplies four high flow switches (2-261-2E, -2F, -2G, and -2H). Two of the switches are located on the A Channel and two on the B Channel. (See Figure 1).

DIS 250-1 was being performed on Unit 2 and had progressed normally through 7 switches (2-261-2A, -2B, -2C, -2D, -2E, -2F, -2G). The procedure then directed the IM to close the isolation valves for the 2-261-2H high flow switch. The equalizing valve was then opened to equalize the high and low side instrument legs. With the switch pressure equalized, the vent cap on the low side leg was opened to bleed down the instrument line. The test equipment was then installed on the high side leg for loading of the differential pressure switch. The switch was then loaded with a high differential pressure, initiating a half Group I isolation and the associated Control Room alarm at 2239 hours as designed. The switch was then depressurized and the alarm and half Group I isolation reset. The IM then proceeded to valve the instrument back into the system by pressurizing the instrument to approximately reactor pressure with the test equipment, closing the test valve and then opening the isolation valves in accordance with the procedure. When valving the instrument back into the system, the IM did not have any specific comparison of the instrument system pressure versus header pressure. It is believed that there was a large enough differential pressure between the pressure in the instrument and the pressure in the header prior to valving in the instrument which caused a pressure spike resulting in the unplanned Group I Isolation and subsequent reactor scram.

A system history review indicates that a previous event involving an unplanned Group I isolation during performance of DIS 250-1 occurred on Dresden Unit 3 in 1987 (see Section F). In this previous case, the cause was attributed to an apparent valving error, although inspection/testing of a sensing line valve manifold was also performed.

The main generator is equipped with two circuits designed to automatically trip the main generator on reverse power conditions. The first, associated with a primary GGP relay (32G3), is initiated by way of a turbine trip signal and is designed to trip the generator at a real reverse power value of approximately -2.18 MWe after a 5-second time delay. The primary GGP circuit also prohibits the Operator from opening the OCBs via main Control Room panel 902(3)-8, until the reserve power and time delay setpoints are satisfied. This open-inhibit interlock does not apply to the OCB control switches on the 345 KV switchyard control panel located in the common area of the Control Room. The second reverse power trip circuit is a secondary GGP relay (92G3) which is designed to trip the main generator at a real reverse power value of approximately -2.18 MWe with a 15-second time delay. Thus, the secondary GGP relay normally trips the main generator unless a turbine trip signal is present.

Approximately 3-1/2 minutes after the reactor scram, an Operator opened the OCBs from the 345 KV switchyard panel (which generated an automatic trip) in order to separate the generator from the 345KV switchyard because the generator appeared not to trip within the expected setpoints. This action was taken in accordance with reactor scram procedure DGP 2-3. Review of this concern with on-site Operational Analysis Department (OAD) personnel resulted in preparation of a special functional test of the reverse power circuitry during startup. During performance of the special test, the reverse power circuitry was concluded to operate satisfactorily. However, following a subsequent Unit 2 scram on January 16, 1990, the OCBs were again manually opened due to concern that the generator appeared to motor in excess of the reverse power trip setpoints. Further investigation and testing then concluded that biasing of the reverse power relay setpoints by significant reactive loads may have been the underlying root cause behind the reverse power trip concerns noted during both the January 5, 1990, and January 16, 1990 Unit 2 events.

D. <u>SAFETY ANALYSIS OF EVENT</u>:

In accordance with Technical Specification Table 3.2.1., two Main Steam Line High Flow switches must be operable per steam line (with a trip setpoint of less than or equal to 120% of rated steam flow) or an orderly load reduction must be initiated and the reactor must be in a hot standby condition within eight hours.

The Main Steam Line high flow Group I Isolation logic is arranged in a 1 out of 2 twice logic. There are two safety system channels with each one consisting of two subchannels which contain four differential pressure switches. See Figure 1 for the logic arrangement. In order to initiate a Group I isolation, at least one of the pressure switches must trip in each of the safety system channels. The pressure switches actuate on high differential pressure. Since pressure switch DPIS 2-261-2H was the only switch being calibrated at the time of the scram, a half Group I Isolation occurred as required as part of the surveillance after loading the switch with a high differential pressure. However, because of a high differential pressure between the instrument pressure and the header pressure, a pressure transient in the header occurred while valving the switch back into service. The 16 differential pressure indicating switches utilized in this trip system are divided such that four sets of four switches use a common sensing header. The sensing header for DPIS 2-261-2H is also used by DPISs 2-261-2E, -2F and -2G. Each of these four switches input into a separate subchannel of the logic system as shown on Figure 1. Therefore, the pressure transient that occurred on this sensing header resulted in the actuations of DPIS 2-261-2E and/or DPIS 2-261-2G on the A channel and DPIS 2-261-2F and/or DPIS 2-261-2H on the B channel, which in turn caused a full Group I Isolation. The safety significance of this event was minimal since the pressure switches operated as designed and actuated on high differential pressure, thus initiating the primary containment Group I Isolation and subsequent reactor scram.

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

As immediate corrective action, Work Request 89711 was written to perform a special test on the Main Steam Line high flow switches and their valve manifolds to determine the cause of the Group I isolation. Various tests were performed prior to Unit startup involving valving the 2-261-2H switch in and out of service while monitoring the header and switch pressures. No problems could be identified with the switch or the valve manifold which could have resulted in the isolation. However, the valve manifold for the 2-261-2H high flow switch was replaced as a conservative measure.

This event was reviewed with the IM Department in order to insure awareness of this concern (237-200-90-00201). A revision to DIS 250-1 was also initiated by the IM Department (237-200-90-00202). Installation of pressure gauges on the common headers for each switch was also initiated in order to help prevent recurrence of this type of event (237-200-900-00203). The Technical Staff is also investigating physical hardware changes such as installation of a head chamber, a different method of venting, and installation of analog transmitters and associated digital components to replace the high flow switches. In addition, the Technical Staff is evaluating the performance of the surveillance at a lower reactor power level and the technical feasibility of increasing the Main Steam Line High Flow trip setpoint (237-200-90-00204).

Following the subsequent January 16, 1990 Unit 2 scram, the reverse power relays were replaced. Operator response procedures were also reviewed and revised as appropriate in order to insure Operator awareness of this concern. Formal training was also initiated for all Licensed Operations personnel, including the generator reverse power relay issue. Further study by OAD of the reverse power relay design is also continuing (237-200-90-00205).

F. PREVIOUS EVENTS:

LER Number	Title
87-016-0 (Docket #050249)	Primary Containment Group I Isolation and Reactor Scram Due to Apparent Personnel Error

On September 28, 1987 at 0231 hours with the Unit 3 reactor operating at 78% power, a primary containment Group I isolation and subsequent reactor scram occurred while an IM was performing a calibration procedure for the Main Steam Line high flow switches. The root cause of the event was attributed to apparent personnel error by the IM not properly isolating Main Steam Line high flow switch DPIS 3-261-2N during the surveillance.

G. COMPONENT FAILURE DATA:

As no component failures occurred during this event, this section is not applicable.

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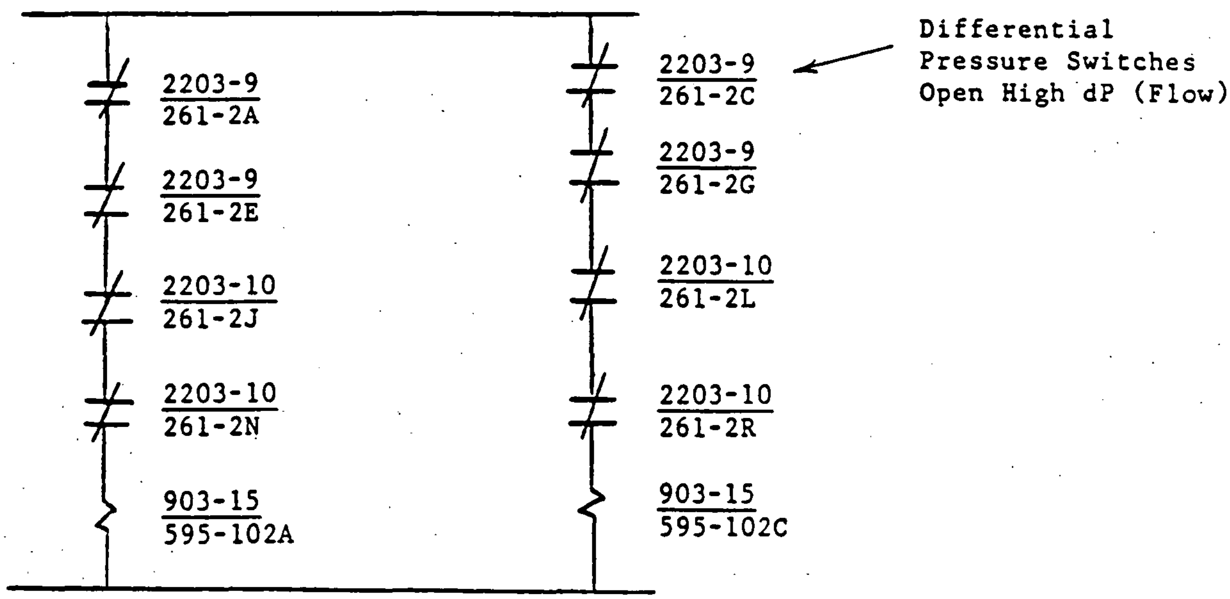
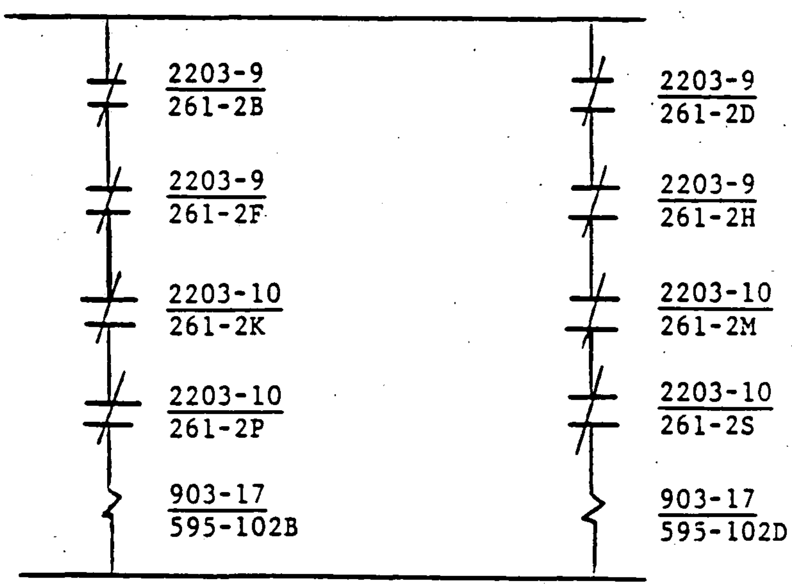
TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]



Main Steam Line High Flow
Channel A



Main Steam Line High Flow
Channel B

Figure 1

Main Steam Line High Flow Switch Logic Diagram