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ComEd

November 21, 1994

JSPLTR 94-0022

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

Licensee Event Report 94-019, Docket 50-249 is being submitted as required by Technical Specification 6.6, NUREG 1022 and 10CFR50.73(a)(2)(iv).

Sincerely,

For Richard J Stef

J. Stephen Perry
Vice President
BWR Operations

JSP/:cfq

Enclosure

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

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NRC FORM 366 (5-92) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 5/31/95
LICENSEE EVENT REPORT (LER)	ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Dresden Nuclear Power Station, Unit 3	05000249	1 OF 4

TITLE (4)
Reactor Scram Due to Instrument Valving 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 NAME None	DOCKET NUMBER
10	26	94	94	-- 019 --	00	11	23	94	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9): N
POWER LEVEL (10): 000

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)

	20.2201(b)		20.2203(a)(3)(i)		50.73(a)(2)(iii)		73.71(b)
	20.2203(a)(1)		20.2203(a)(3)(ii)	X	50.73(a)(2)(iv)		73.71(c)
	20.2203(a)(2)(i)		20.2203(a)(4)		50.73(a)(2)(v)		OTHER
	20.2203(a)(2)(ii)		50.36(c)(1)		50.73(a)(2)(vii)		(Specify in Abstract below and in Text, NRC Form 366A)
	20.2203(a)(2)(iii)		50.36(c)(2)		50.73(a)(2)(viii)(A)		
	20.2203(a)(2)(iv)		50.73(a)(2)(i)		50.73(a)(2)(viii)(B)		
	20.2203(a)(2)(v)		50.73(a)(2)(ii)		50.73(a)(2)(x)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Richard O. Ralph, System Engineering Department Ext. 2670	(815) 942-2920

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
	YES (If yes, complete EXPECTED SUBMISSION DATE).	X	NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

At 0126 hours on October 26, 1994, Unit 3 was in the Refuel mode, with Dresden Instrument Surveillance (DIS) 500-01, Reactor Vessel High Pressure Scram Pressure Switch Calibration, in progress. As pressure switch 2-263-55C was being returned to service, a full RPS scram, Group II isolation and Group III isolation occurred on a reactor vessel low water level signal. All control rods were fully inserted before the scram, therefore no control rod motion occurred. All isolation valves and the Standby Gas Treatment System responded as expected. There was no measured change in reactor coolant temperature while the Shutdown Cooling System was isolated. Pressure switch 2-263-55C has a common sensing line to reactor vessel water level transmitters 2-263-57C and 2-263-57D which provide signals to the reactor protection system and primary containment isolation system. The event was caused by a pressure transient produced while valving in pressure switch 2-263-55C with a differential pressure across the isolation valve. There have been two previous events involving instrument valving.

NRC FORM 366A (5-92)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 5/31/95
LICENSEE EVENT REPORT (LER) TEXT CONTINUATION		ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Dresden Nuclear Power Station, Unit 3	05000249	94	-- 019 --	00	2 OF 4

TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

EVENT IDENTIFICATION:

Reactor Scram Due to Instrument Valving Procedure Deficiency

A. PLANT CONDITIONS PRIOR TO EVENT:

Unit: 3 Event Date: 10/26/94 Event Time: 0126

Reactor Mode: Mode Name: Refuel Power Level: 0%

Reactor Coolant System Pressure: 0 psig

B. DESCRIPTION OF EVENT:

At 0126 hours on October 26, 1994, Unit 3 was in the Refuel mode, with Dresden Instrument Surveillance (DIS) 500-01, Reactor Vessel High Pressure Scram Pressure Switch Calibration, in progress. As pressure switch [63] 3-263-55C was being returned to service, a full RPS [JC] scram, Group II isolation and Group III isolation occurred on a reactor vessel low level signal. All control rods were fully inserted before the scram, therefore no control rod motion occurred. All isolation valves and the Standby Gas Treatment System [BH] responded as expected. There was no measured change in reactor coolant temperature while the Shutdown Cooling System [BO] was isolated. Other occurrences during this event included: Reactor water level quickly increased to approximately 48 inches; The Shutdown Cooling Pumps did not trip; and Annunciator 903-5 G-7, Feedwater Cont. Signal Failure, alarmed.

Pressure switch 3-263-55C is on a common sensing line that includes reactor vessel level transmitters 3-263-57C and 3-263-57D which provide signals to the reactor protection system and the primary containment isolation system. These instruments are part of the vessel water level medium range instruments.

DIS 500-01, Reactor Vessel High Pressure Scram Pressure Switch Calibration, contained a step to apply pressure to the pressure switch equal to reactor pressure. The reactor was at zero pressure, therefore the switch was not pressurized. Although the reactor was not pressurized, the static pressure from the height of the instrument sensing line was approximately 13 psi. Therefore, even at zero pressure in the reactor, 13 psid existed across the instrument isolation valve.

C. CAUSE OF EVENT:

This report is submitted in accordance with 10 CFR 50.73(a)(2)(iv), which requires the reporting of any unplanned Engineered Safety Feature (ESF) actuation.

Our investigation revealed that the reactor low water level signal was generated from a pressure transient in the reference leg of the level instruments as a result of valving in PS 3-263-55C. Interviews with the IM technician performing the surveillance indicate that the technician's actions were in accordance with the procedure and department expectations.

A special test, SP 94-10-126, was performed on October 29, 1994 which duplicated the conditions which were present on October 26. The test confirmed that the static head of the reference leg (approximately 13 psi) will cause pressure oscillations when the instrument isolation valve is slowly opened. The pressure oscillations were measured and were large enough to cause the scram and group II and III isolations. The special test also confirmed that DIS 500-01 can be successfully performed if the instrument to be valved in is pressurized to include the static head pressure to equalize the pressure upstream of the isolation valve.

During the refueling outage D3R13, which the unit was in at the time of the scram, the reactor vessel instrumentation sensing line backfill modification was installed to address the non-condensible gas issue. Part of this modification removed the pressure compensation of the narrow range instruments and caused the vessel medium range instruments to read the same as medium range at zero reactor pressure. The medium range instruments used to read 16 to 18 inches higher than narrow range. The size of the pressure transient generated while valving in the pressure switch with 13 psid was small enough that this additional 16 inches of water level prevented the scram in the past.

The apparent cause of this event has been found to be procedural deficiency. DIS 500-01, Reactor Vessel High Pressure Scram Pressure Switch Calibration, did not account for the static head of the instrument sensing line. Pressure switch 3-263-55C has a common sensing line to reactor vessel level transmitters 3-263-57C and 3-263-57D which provide signals to the reactor protection system and primary containment isolation system. The event was caused by a pressure transient, produced while valving in pressure switch 3-263-55C while a differential pressure existed across the isolation valve, that was sensed by the other two transmitters.

The rapid reactor vessel level increase is due to a redistribution of mass within the reactor vessel. The Shutdown Cooling System provides forced circulation of coolant through the reactor. This causes the water level within the shroud and moisture separator to be higher than the water level in the annulus. When the Shutdown Cooling System isolated, Shutdown Cooling flow stopped and the water level within the shroud equalized with the annulus. The change in level constitutes an approximate 2,300 gallon change. The volume of water in the moisture separator amounts to approximately 1600 gallons. CRD flow contributed slightly as well as a momentary opening of the low flow feedwater regulating valve.

The Shutdown Cooling Pumps continued operation has been observed before and is believed to be caused by the ability of the pump minimum flow line to provide adequate NPSH. The pump suction pressure switches need to be checked and the line hydraulics need to be reviewed. These actions can be completed after unit startup because the system demonstrated that it can operate isolated without degradation for a short time.

The reason for the Feedwater Level Control System trouble alarm (903-5, G-7) is believed to have been spurious which had no adverse effect on the Feedwater Level Controller (Bailey Net 90), Feedwater Level Control Valves or the events experienced after the scram. This alarm is actuated by loss of demand signal for the 3A or 3B Feedwater Regulating Valves (FWRVs). The FWRVs were in manual with a hard close demand applied. If a valid loss of demand signal had occurred, one or both of the FWRVs would have locked up. This did not happen. Additionally, the alarm typer did not register receiving the alarm. The Bailey Net 90 was inspected the following day. No abnormalities within the system or its power supplies were found.

D. SAFETY ANALYSIS:

At the time of the event, Unit 2 was shutdown in the Refuel mode with all rods in. The Scram, Group II, and Group III automatic actions functioned as expected. Shutdown Cooling system was restarted at approximately 0140 and there was no observed increase in Reactor Coolant Temperature while the Shutdown Cooling system was isolated. Reactor Water Cleanup system was restarted at 0150, Reactor Building Ventilation system was restarted and Standby Gas Treatment system secured at approximately 0140. For these reasons, this event had minimal safety significance.

E. CORRECTIVE ACTIONS:

The corrective action to prevent reoccurrence is to revise DIS 500-01 and similar procedures to include the static head of the sensing line when pre-pressurizing the instrument to equalize pressure before the instrument isolation valve is opened. (249-180-94-01901) Procedures which were to be performed in the near term have been revised. They are:

DIS 0201-01, Reactor vessel 600 PSI Scram Bypass Pressure Switch Calibration;
DIS 1300-01, Sustained Reactor High Pressure Calibration;
DIS 1500-01, Reactor Low Pressure (350 PSIG) ECCS Permissive.

The following procedures are in the final approval stage:
DIS 0250-03, Electromatic Relief Valve/Target Rock Valve Pressure Switches Calibration; and
DIS 0600-02, Narrow Range Reactor Pressure Calibration.

The following procedures will be revised before they are performed:
DIS 0263-04, Unit 2 ATWS Transmitter and Master Trip Unit Calibration and Logic System Functional Test;
DIS 0263-06, Unit 3 ATWS Transmitter and Master Trip Unit Calibration and Logic System Functional Test; and
DIS 0260-03, Reactor Level Sensing Line Backfill Procedure.

The Unit 3 Shutdown Cooling Pumps' trip capability will be tested to verify proper system operation by May 31, 1994. (249-180-94-01902)

F. PREVIOUS OCCURRENCES:

LER 12-2-91-142, Inadvertent Closure of Core Spray Minimum Flow Valve 2-1402-38A Due to Procedural Deficiency;

LER 12-2-93-015, Unit 2 Reactor Scram, Group 2 and Group 3 Isolation, While Shutdown Due to Instrumentation Valving Error.

G. COMPONENT FAILURE DATA:

None.