Commonwealth Edison
Dresden Nuclear Power Station
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Telephone 815/942-2920

July 27, 1994

GFSLTR 94-0245

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

Licensee Event Report 94-017, Docket 50-249 is being submitted as required by Technical Specification 6.6, NUREG 1022 and 10CFR50.73(a)(2)(iv).

Sincerely,

Gary Spedl

Gary F. Spedl
Station Manager
Dresden Station

GFS/MB/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 3

TITLE (4)
Group I Primary Containment Isolation Due to Instrument Rack Vibration

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	DOCKET NUMBER
									None	
06	30	94	94	-- 017 --	00	07	24	94	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9)	N	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)
POWER LEVEL (10)	000		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)
Michael Baron, System Engineer Ext. 2414	(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 1408 on June 30, 1994 with Unit 3 in a shutdown mode for D3R13 refueling outage, a Primary Containment Isolation Group I alarm was received in the Control Room. The inboard Main Steam Isolation Valves (MSIV) and inboard Recirculation sample isolation valves closed. The associated outboard isolation valves were closed and tagged out-of service at the time, and did not move.

Plant parameters that could cause a Group I isolation were checked, and no abnormal conditions were observed. The alarm and isolation had occurred concurrently with a start of the 3B Containment Cooling Service Water (CCSW) Pump, so CCSW piping was walked down to determine wheather there had been a piping pressure transient (water hammer) in the system sufficient to vibrate the Group I instrument racks. No indications of pipe or restraint movement or damage were found.

While the root cause of the Group I alarm and isolation could not be positively determined, it is stongly believed that the instrument rack had been inadvertantly jarred by workers in the area.

EVENT IDENTIFICATION:

Group I Primary Containment Isolation Due to Instrument Rack Vibration.

A. PLANT CONDITIONS PRIOR TO EVENT:

Unit: 3 Event Date: June 30, 1994 Event Time: 14:08

Reactor Mode: N Mode Name: Shutdown Level: 000%

Reactor Coolant System Pressure: 0 psig

B. DESCRIPTION OF EVENT:

On June 30, 1994, at 1408 with Unit 3 in a shutdown mode during D3R13 Refueling Outage, a Group I Primary Containment isolation was received immediately after the start of 3B CCSW Pump. Recorded times were as follows, 14:08:41 3B Containment Cooling Service Water(CCSW) pump ON, at 14:08:45, Inboard Main Steam Isolation valves and the Inboard Recirculation Sample valves went closed. All Outboard Isolation valves were Out of Service (OOS) closed and did not move. (The Group I isolation alarm was the only indication received. No precursors to this event were experienced and no causal factors were apparent.) The Group I alarm reset immediately and was not captured by computer point.

An investigation team was assembled and the possibility of a CCSW piping pressure transient (water hammer) was explored, since CCSW piping is close to the Group I instrumentation racks. Engineering personnel performed walkdowns and inspections of the CCSW piping to locate any indications of movement near pipe supports or possibility of damage to axial restraints. The walkdowns included piping from the discharge of A, B, C and D CCSW pumps and downstream to the Low Pressure Coolant Injection (LPCI) heat exchangers. No indication of piping movement or damage was found.

It was determined that the timing of starting the CCSW pump and the Group I isolation was probably a coincidence. The System Engineering Department inspected all associated electrical equipment and reviewed all captured documentation, no anomalies were found. During the event, all equipment functioned as designed. All relays related to Group I isolation logic, i.e. High Radiation, High Main Steam Line Flow, Low Main Steam Line Pressure, Steam Tunnel High Temperature and Low Low Reactor Water Level were verified in their proper condition. A walkdown of instrument racks 3-2203-9B and 3-2203-10B Main Steam Line High Flow instrumentation revealed that many maintenance activities were being performed in the general vicinity, and both racks were wet with fresh paint. The Reactor Protection System (RPS) channels A and B are both present on these racks.

C. CAUSE OF EVENT:

This report is submitted in accordance with Title 10 of the Code of Federal Regulation Part 50 Section 73 (a) (2) (iv), which states "Any event or condition that resulted in manual or automatic actuation of any Engineered Safety Feature (ESF), including the RPS" must be reported.

The apparent cause of this event was a maintenance-induced vibration of the Main Steam Line High Flow instrument racks. This disturbed the vibration sensitive mercury type Barton model switches. These racks have both channels of RPS present so local vibration could result in a full Group I isolation. These racks are normally a restricted area during operation but many maintenance activities were being performed due to the refueling outage.

D. SAFETY ANALYSIS:

The safety significance of the Group I isolation is considered minimal as Unit 3 was in a Shutdown mode, Reactor pressure was at 0 psig and the reactor was vented. All Group I equipment performed as designed and experienced no damage.

If the Reactor were in the Run mode at 100% Reactor power, Upon receipt of a group I isolation signal, the Main Steam Isolation Valves (MSIV's), Main Steam Drain Valves, Isolation Condenser vent valves and Recirculation system sample valves would close. This Group I isolation would be followed by a Scram. This is discussed in Dresdens Technical Specification 3.2 Primary Containment Isolation Functions and the Revised Updated Safety Analysis Report section 7.2 Reactor Protection System.

E. CORRECTIVE ACTIONS:

Immediate verbal notification to all Departments working in these areas was conducted as inspection data was evaluated and results were finalized. In addition, this Licensee Event Report and Dresden Administrative Procedure 7-19 (control of work activity in reactor trip potential areas) will be reviewed by the Maintenance Department personnel to re-affirm the need for caution while working around vibration sensitive equipment in the plant. This item will be tracked in the Nuclear Tracking System (NTS).

Dresden Administrative procedure (DAP) 7-19 Specifically states that this procedure controls any activities on or near vibration sensitive equipment and applies to all personnel. The Shift Engineer or his designee must be contacted prior to entry into these areas. A review of all work will be presented and evaluated before permission is given to enter. Depending on the complexity of the work activity further meetings may be required such as DAP 07-26 Shift Briefing or DAP 07-37 Conduct of Heightened Level of Awareness activities and High Impact Activities. A positive trend has been noted. This was the first event of this nature since the 1987 time frame. In the future DAP 7-19 will be reviewed, if needed to maintain error free status.

F. PREVIOUS OCCURRENCES:

LER/Docket Number	Title
12-2-87-144	Reactor Scram due to spurious Main Steam Line Low Pressure signal caused by vibration.

G. COMPONENT FAILURE DATA:

N/A