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ComEd

June 28, 1996

JSPLTR #96-0101

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

Licensee Event Report 96-009, Docket 50-237 is attached and is being submitted pursuant to 10CFR50.73(a)(2)(iv), which requires reporting of any event that results in unplanned manual or automatic actuation of any engineered safety feature (RPS).

This correspondence contains the following commitments:

1. The System and I&C engineers, Mod Engineer, Design and Plant Engineering Superintendents, Unit Supervisor and Special Procedure technical reviewers will be counseled and coached as appropriate. (2371809600901)

2. The Operations Training Advisory Committee (TAC) will review operator response to the event and include appropriate practice sessions in Continuing Operator Simulator training. (2371809600902)

3. A better method to control process control logic design changes will be developed and implemented. (2371809600903)

4. Training will be developed and provided for the Engineering Department on their role in establishing and maintaining a conservative safety culture. (2371809600904)

050037

If you have any questions, please contact Pete Holland, Dresden Regulatory Assurance Supervisor at (815) 942-2920 extension, 2714.

Sincerely,

J. Stephen Perry
Site Vice President
Dresden Station

Enclosure

cc: H. Miller, Regional Administrator, Region III
NRC Resident Inspector's Office
Illinois Department of Nuclear Safety

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 2	0500237	1 OF 9

TITLE (4)
Manual Reactor Scram Due To Lowering Reactor Water Level Due to Automatic Feedwater Level Control System Design 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	DOCKET NUMBER
05	31	96	96	-- 009 --	00	06	28	96	None	
									FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9)	N
POWER LEVEL (10)	45

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)
K. Robbins, Plant Engineering Ext. 2314	(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)

On May 31, 1996, at approximately 2037, with Unit 2 in the run mode at approximately 45% power while performing an on-line logic configuration change to the Bailey Network 90 Feedwater Control System (FWCS) a loss of reactor feedwater control occurred. 2B Feedwater Regulating Valve moved fully shut during a logic reconfiguration of the Bailey Network 90 FWCS. This was the result of a logic execution sequence error, which was a design characteristic of the Bailey system. The Unit 2 Nuclear Station Operator initiated a manual scram when reactor water level reached a pre-established limit established due to special procedural controls associated with the feedwater testing. The lowest water level in the reactor reached was approximately -13.6 inches after the scram, which is more than 10 feet above the top of active fuel.

The overall safety significance was determined to be minimal because all safety systems performed as required and there was no danger to health and safety at any time.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Dresden Nuclear Power Station, Unit 2	0500237	96	-- 009 --	00	2 OF 9

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

EVENT IDENTIFICATION:

Manual reactor scram due to lowering reactor water level due to automatic feedwater level control system design deficiency.

A. PLANT CONDITIONS PRIOR TO EVENT:

On May 31, 1996, at 2037, Dresden Unit 2 was in the "Run" (N) Mode at 940 psig reactor pressure. The reactor was operating at approximately 45 percent core thermal power, 340 Mwe. Feedwater Control System (FWCS) modification testing was in progress by System Engineering [non-licensed] Test Engineers. The 2A Reactor Feed Pump [SJ] was operating with 2B Feedwater Regulating Valve (FWRV) controlling reactor water level in single element automatic control. The 2A FWRV was closed in manual mode. The Low Flow Control Valve (LFCV) was closed in automatic mode. The Feedwater Control System was maintaining normal reactor water level at approximately +30 inches. Normal offsite electrical power was available. Unit 2 had been on line since May 27, 1996.

Dresden Unit 2 and 3 share a common control room. Unit 3 was in cold shutdown.

B. DESCRIPTION OF EVENT:

This LER is being submitted pursuant to 10 CFR 50.73(a)(2)(iv), any event or condition that resulted in a manual or automatic actuation of any Engineered Safety Feature (ESF) including the Reactor Protection System (RPS).

In February 1994, the station approved a Feedwater System modification which included the replacement of the 2A FWRV and hydraulic actuators. The final Feedwater System modification scope was changed to include air operators for the FWRVs and a Bailey Network 90 Feedwater Control System (FWCS). The final modification package was issued in June 1995. During July and August 1995, the logic was tested using the Dresden simulator. As a result of this testing, several logic improvements were made. The logic drawings were approved in February 1996.

In March 1996, pre-operational post modification testing of the FWCS was conducted off-line prior to startup from the D2R14 refueling outage. Testing was controlled through a Special Procedure. During this testing, changes to the control system logic were made as necessary, by the Test Director (either the System Engineer or the Instrumentation & Control (I&C) Engineer) [non-licensed]. The review of these changes was performed by the second Test Director. These changes were then reviewed and concurred with by the Original Equipment Manufacturer (OEM). This approval process was not documented per station procedure.

On March 30, 1996, startup testing for the Bailey Net 90 modification began using startup modification testing procedure, SPI-96-01-01 (SP). From March 30 to May 25, 1996, FWCS testing continued as permitted by operational conditions.

On May 25, 1996, a planned shutdown of Unit 2 occurred for unrelated reasons. On May 26, 1996, the System Engineer and I&C Engineer made off-line configuration logic changes that were identified during previous testing between May 10 and 25, 1996. These changes were reviewed and concurred with by the Original Equipment Manufacturer (OEM). This approval process was not documented per station procedure.

On May 31, 1996, at approximately 1130, SP testing of the automatic transition from the 2B FWRV to the 2A FWRV resulted in a level transient, a decrease of approximately 7 inches. The System Engineer made a determination that the level transient was unacceptable.

At approximately 1200, the System Engineer notified the Plant Engineering Superintendent of the problems with 2A FWRV logic. The System Engineer and the OEM representatives later determined that a minor logic change was required to correct the problem. This logic change was not expected to produce any observable change in plant operation. It was decided to wait until afternoon shift arrived to confirm that a logic change was required. FWCS testing continued with a portion of the SP not impacted by the identified logic changes.

At approximately 1300, testing began with the FWCS in 3-element control. At approximately 1526, the Heightened Level of Awareness Briefing (HLA) for the oncoming shift was performed. At 1700, testing of the 2A FWRV was secured and testing began on the 2B FWRV.

At 1830, the afternoon test team arrived on site and discussed the day team's conclusion. The discussion included a concern that Unit 2 would have to be shutdown in order to perform the logic change identified by the day shift test team. The OEM representative suggested making an on-line configuration change. The team reviewed the OEM Technical Manual process for on-line configuration changes and found that the proposed logic changes met the proscribed rules for re-configuration of the software with the control system on-line.

The OEM representative communicated that he had performed this type of on-line configuration numerous times before (at non-nuclear sites) with no problem. The OEM representative indicated that the system would "check" the logic before going into the control mode, and that there would be no impact on Unit 2 operation. The logic change was reviewed by the test teams and they all concurred on the change. This approval process was not documented per station procedure.

The Test Directors did not recognize that the on-line configuration (logic) change feature of the Bailey Network 90 system was potentially untested at Dresden and that no 10CFR50.59 or independent review of the change was performed. Both Engineering Test Teams agreed that the delay in feedwater testing would provide an opportunity to install the logic change into the Unit 2 FWCS, since it did not effect reactivity.

At approximately 2000, the I&C Engineer discussed the logic change to correct the 7 inch level error with the Unit 2 Supervisor (licensed Senior Reactor Operator). During the decision making process, the Unit 2 Supervisor asked about possible problems and what would be seen on the front panel. The I&C Engineer and OEM personnel repeatedly responded that the change would have no plant impact and that the transfer would be "bumpless", meaning that the NSO would not see any appreciable change in reactor water level. However, they did not adequately review the SP.

The Unit 2 Supervisor told the Test Team not to begin until he briefed the Nuclear Station Operator (NSO) [Licensed Reactor Operator]. The Unit 2 Supervisor held a prejob brief with the Unit 2 NSO and outlined the contingencies made during the HLA for the feedwater testing and dedicated him to the feedwater panel. The NSO was to manually scram the reactor at plus 20 inches. To ensure no effect to the operating unit, the logic change was to be made with the 2A FWRV closed in manual, the 2B FWRV was to remain open in automatic and the LFCV was closed in automatic.

At 2032, the NSO received an unexpected alarm when the backup control module was placed in the "configure" mode. The Test Team determined that the alarm was expected and the evolution was allowed to continue. The new logic configuration was then inserted on the backup control module. Automatic diagnostic checks indicated a successful load and no signs of trouble, unexpected alarms or indications were received.

At approximately 2037, the backup control module was placed in the "execute" mode. This would place it as the primary control module and it would now control feedwater flow.

At 2037:16, the 2B FWRV began to close after the backup FWCS control mode was placed in execute, resulting in a sudden drop in feedwater flow and reactor water level.

At 2037:18, the Condensate Booster Pump Suction Pressure Hi and Condensate Booster Discharge High Pressure alarms were received.

At 2037:23, Reactor Low Level alarm was received at approximately plus 27.0 inches. Reactor level was approximately plus 27 inches (Narrow Range (NR) Channel A). The NSO took manual control of 2B FWRV and opened the valve per the contingency plans briefed for SP-96-01-01.

At 2037:25, the LFCV, which was in AUTO mode, began to open as designed due to the removal of the close bias when the 2B FWRV was placed in manual. This produced a higher indicated flow than expected by the NSO.

At 2037:47, the NSO manually started to close the 2B FWRV to decrease feedwater flow. The Reactor FW Flow A High alarm actuated.

At 2037:52, the LFCV began to close automatically due to the high flow condition.

At 2037:55, reactor level was observed to be approximately plus 21 inches (NR Channel A) and decreasing.

At 2037:57, the NSO manually began to open the 2B FWRV, and the LFCV opened in response to the low level.

At 2038:13, RPS Channel A half-scram, Reactor Low Level is received at approximately plus 15 inches. Upon receiving this alarm, the Unit 2 Supervisor ordered a manual scram of the reactor. The Unit 2 NSO initiated the manual reactor scram at 2038:17 and placed the mode switch to SHUTDOWN at 20:38:18.

All control rods fully inserted. As the voids in the reactor core collapsed, reactor water level dropped to approximately -13.6 inches. The reactor low water level signal initiated a Group II and Group III primary containment isolation signal at 2038:18. The appropriate isolation actions were verified by the Operators and the 'B' Standby Gas Treatment train started.

The reactor recirc pumps were verified to run back to their minimum speed when the feedwater flow dropped below 20%.

During a subsequent design review of the Bailey Network 90 FWCS, a logic execution sequence error was found in the original logic design. This error caused the 2B FWRV to move fully shut when the backup module attempted to take over process control from the primary. It was also determined that a similar problem would have occurred if the primary Multifunction Processor had failed and automatically transferred to the backup Multifunction Processor. However, the on-line configuration change would have been successful if the proper execution sequence had been installed at the factory.

C. CAUSE OF EVENT:

C.1 Event Root Cause

C.1.1. An internal logic sequencing error caused the single element Proportional-Integral-Derivative (PID) controller function to execute with default values prior to being set to values correct for the current plant condition. The cause code for this event is NRC Cause Code B, a Design, Manufacturing, construction/Installation cause, specifically the logic execution sequencing error was internal to the Bailey system. This caused the Bailey Network 90 to close the FWRV when the on-line configuration change was performed with the master station in automatic.

This sequence error is a firmware design characteristic of the Bailey Network 90 system and would have resulted in the same process control failure anytime the backup control module attempted to take control from the primary control module with the control system in automatic.

C.1.2. The decisions made during the FWCS testing should have been more conservative and the software logic changes should not have been made online. The changes to the logic should have been made using approved design change procedures. The review and assessment of the risks and consequences associated with change could have been more thorough. Indications of the need to strengthen the conservative safety culture are as follows:

a. The Test Director believed the OEM Technical Representative when told that the on-line configuration could safely be done.

b. It was not recognized by the testing team that the on-line configuration function of the Bailey Network 90 was a potentially untested function and that it should have been tested prior to relying on it for the logic configuration change.

c. The Test Director did not consult with others outside of the test team on the decision to conduct an evolution he was personally unfamiliar with.

d. A review with independent personnel outside the test team was not performed and may have identified the importance of placing the Master Station into manual prior to performing the evolution.

e. The Unit 2 Supervisor did not challenge the Test Team on where the Special Procedure allowed on-line configurations to complete minor logic changes.

f. The Plant Operations Review Committee approved a Special Procedure which required more guidance concerning allowed tuning changes.

Note: The 10CFR50.59 for the SP did state, in an attachment, that there would be "no logic changes".

g. A precursor event on May 5, 1996, was not aggressively acted upon.

D. SAFETY ANALYSIS:

A review of selected safety systems and parameters was performed to verify proper response following this event. Plant systems operated as expected and followed the sequence of events as outlined in the FSAR. The plant response was bounded by the Dresden Rebaselined Updated final Safety Analysis Report (RUFSAR) design basis. The event described in the RUFSAR assumes a feedwater controller malfunction demanding closure of the FWRVs. The incident on May 31, 1996, was bounded by the analyzed RUFSAR event since the event initial power level was lower and operator actions were taken to mitigate the water level transient consequences. The actual event was additionally bounded by the conservative analysis as evident by the fact that the RUFSAR case assures water level never drops below 5 feet above the fuel. During the event the water level in the reactor reached a low value of approximately -13.6 inches, which is more than 10 feet above the fuel.

The overall safety significance was determined to be minimal because all safety systems performed as required and there was no danger to health and safety at any time.

E. CORRECTIVE ACTIONS:

E.1. The Engineering Test Directors were relieved of Test Director duties until further notice.

A multi-discipline Root Cause investigation team was formed.

The Unit 2 Bailey Network 90 was quarantined until its stored data was printed out and a SP was developed to diagnose the cause of the problems observed.

The Special Procedure was performed on the Unit 2 Bailey Network 90 in order to:

1. Restore the dual processors to the pre-incident configuration.
2. Configure the FWRVs to 2A in manual/shut and 2B in automatic/open and blocked.
3. Attempt to reconstruct the problem symptoms and diagnose them.

Unit 3 startup was placed on hold pending execution of the Special Procedure on Unit 2 and review of its results by PORC and the Station Manager. The hold on Unit 3 startup was released on June 6, 1996. The release was based on the results of the SP which indicated that the failure was limited to the on line configuration mode and the significant differences between the Unit 2's and Unit 3's hardware and software.

E.2 Short Term Corrective Actions

1. A team of personnel from General Electric (GE), ComEd, an Architect and Engineering Firm, and the OEM performed extensive testing and troubleshooting of the FWCS. The team identified the cause of the problem, a logic sequencing error. The logic sequencing error was corrected.

2. The complete logic design was re-reviewed and second reviewed by the OEM to assure any further errors or improvements were identified. The changes/ improvements recommended by the OEM were incorporated into the design. The revised design was also independently reviewed by GE and ComEd Corporate Engineering.

3. The test procedure (SP) was improved and made stronger based on the recommendations of a GE expert's review of the SP.

4. The following were performed prior to additional modification testing or special tests:

   a. Independently review of the 50.59 by qualified team members.

   b. Senior Management to assign Test Engineers as appropriate to provide management oversight.

   c. Provide appropriate training and qualification for individuals assigned as Test Coordinators and Test Engineers.

5. Provided formal training to Engineering Department personnel on management expectations for procedural adherence and communications with Operations Department personnel. Engineering personnel received the training prior to performing plant duties.

6. Provided briefings to operating shifts to include:

   a. A discussion of the Unit 2 event.

   b. Methods to control work activities (cue card and checklist).

7. Developed a Plant Impact Statement for use with all plant activities.

8. Internal Tech Alert Number TA 96-16, entitled "Process Software Logic Changes, dated June 10, 1996, provided other ComEd stations notification of the event.

E.3 Long Term Corrective Actions:

1. The System and I&C engineers, Mod Engineer, Design and Plant Engineering Superintendents, Unit Supervisor and Special Procedure technical reviewers will be counseled and coached as appropriate. (2371809600901)

2. The Operations Training Advisory Committee (TAC) will review operator response to the event and include appropriate practice sessions in Continuing Operator Simulator training. (2371809600902)

3. A better method to control process control logic design changes will be developed and implemented. (23718096000903)

4. Training will be developed and provided for the Engineering Department on their role in establishing and maintaining a conservative safety culture. (2371809600904)

F. PREVIOUS OCCURRENCES:

LER/Docket Number	Title
96-004/50-249	Reactor Vessel Level Transient Resulting In Reactor Scram and Emergency Core Cooling System Actuation Caused By Feedwater Regulating Valve (FWRV)[SJ] Stem Separation

This LER documents a feedwater transient which resulted in the initiation of an automatic reactor Scram due to low reactor water level and Emergency Core Cooling System (ECCS) actuation on low low reactor water level. The plug from the 3B Feedwater Regulating Valve (FWRV) separated from the stem resulting in an isolation of feedwater flow to the reactor. The cause of the stem separation was fatigue cracking. In addition, 2 Group I isolation valves improperly re-opened upon resetting the isolation signal. The cause for isolation valves re-opening is a manufacturing defect and is potentially reportable under 10CFR21. Corrective actions include; replacing the 3B valve stem to existing specifications, operational limits and vibration monitoring of the valve upon resumption of power operations, and future reconfiguration of the 3B FWRV, and replacing the relays. Corrective actions from this materiel condition LER would not have prevented LER 96-009/50237.

87-024/50-249 Unit 2 Reactor Scram On Low Level Due to 2A Feedwater Regulating Valve Failure

On August 21, 1987, a failure occurred on the 2A FWRV. The 2A was a Copes Vulcan D100 valve, with the original double ported internals and a different style stem than the 3B FWRV. The valve stem separated at approximately the same location as the 3B. At the time of failure the 2A was in automatic and the 2B was 25% open in manual, Unit 2 was at approximately 93% power and the valve was in service for approximately 30 months prior to failure. The root cause was attributed to fatigue and no further review was performed to identify the fatigue initiator. The corrective action was to install a new stem and plug in the 2A valve and weld the stem/ plug together. Corrective action to prevent recurrence was to review the modification planned for Unit 3 (from the 1987 Unit 3 feedwater transient, LER 87-013, docket number 050-0249) and determine if a similar modification for Unit 2 would be appropriate to reduce valve vibration and fatigue. The Hush trim was installed on the 3B valve in 1988 (M12-3-87-45B). Later modifications installed CCI Drag trim on the 2A, 2B, and 3A FWRVs. The CCI drag trim modification on the 3B FWRV was scheduled for the upcoming Unit 3 D3R14 refuel outage.

G. COMPONENT FAILURE DATA:

None.