NRC FORM 366 (5-92) U.S. NUCLEAR REGULATORY COMMISSION

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EXPIRES 5/31/95

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.

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Dresden Nuclear Power Station, Unit 2	05000237	1 OF 6

TITLE (4)
The Control Rod Drive Scram Discharge Volume's Reactor Protection System Control Logic Fails To Meet the Single Failure Criteria Due to 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
									None	
10	20	95	95	-- 019 --	01	02	28	96	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)		50.73(a)(2)(iv)		73.71(c)
	20.2203(a)(2)(i)		20.2203(a)(4)	X	50.73(a)(2)(v)	X	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)
Paul K. Garrett, Plant Engineering Ext. 2713	(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 October 20, 1995, at approximately 1100, while Unit 2 was in refuel with all fuel removed from the Reactor Vessel, it was determined that the Control Rod Drive Scram Discharge Volume's (SDV) control logic did not meet the single failure criterion. The SDV was declared inoperable and an ENS phone call was made. The root cause of the failure was due to an inadequate design review process and inattention to detail during the SDV modification development. Corrective actions included a review of the new modification practices to assure changes made in 1986 are still in place, and a sampling review of other modifications developed and reviewed by the cognizant engineers who developed the SDV modification containing the error.

EVENT IDENTIFICATION:

The Control Rod Drive Scram Discharge Volume's Reactor Protection System Control Logic Fails To Meet the Single Failure Criteria Due to Design Deficiency

A. PLANT CONDITIONS PRIOR TO EVENT:

Unit: 2 Event Date: 10/20/95 Event Time: 1100

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

Reactor Coolant System Pressure: 0 psig

B. DESCRIPTION OF EVENT:

This report is being submitted in accordance with 10CFR50.73(a)(1), reportable events, and 50.73(a)(2)(v)(A), any event or condition that alone could have prevented the fulfillment of the safety function of structures or systems that are needed to shutdown the reactor and maintain it in a safe shutdown condition.

On October 19, 1995, at approximately 1700, while, performing Dresden Instrument Surveillance (DIS) 0500-05, Unit 2 Scram Discharge Volume (SDV) Level Sensor Calibration and Functional Test, a discrepancy in the procedure was identified by the Technicians performing the surveillance. The Process Computer Data points' nomenclature did not match the corresponding section on the surveillance's test summary sheet. The surveillance was stopped. The Process Computer nomenclature was revised in 1988 and no corresponding revision was made to DIS 0500-05. Thus, when the surveillance was being performed, the discrepancy became apparent to the Technicians and a review initiated.

It was determined that the series level switches which actuate the Unit 2 Reactor Protection System (RPS) [JC] Channels A1, A2 and B1, B2 should have been wired differently. Each of the 4 RPS channels should have one actuation signal from the West SDV level switch and one from the East SDV level switch. However, the configuration installed by a plant modification, completed in 1985, had both actuation signals to the individual SDV RPS channels from the same side, i.e., both East or both West. A specific example is that the RPS A1 logic relay (590-100A) is actuated by two SDV level switches in series, which are both located on the West SDV. The result is that if relay 590-100A failed to actuate (single failure), with a high level in the West SDV, a RPS Channel A half scram would NOT occur, resulting in a failure of the full scram. A Performance Improvement Form (PIF) was written to document the discrepancy.

On October 20, 1995, at approximately 1100, while Unit 2 was in refuel with all fuel removed from the Reactor Vessel, a final determination was made that the Control Rod Drive Scram Discharge Volume's (SDV) RPS control logic did not meet the single failure criterion. The SDV was declared inoperable. At 1500 (ET) an ENS phone call was made pursuant to 10CFR50.72(b)(2)(iii)(A).

On October 22, 1995, during the design review for the modification to correct the SDV RPS control logic, four RPS cables were identified with a lack of sub-channel separation. These RPS cables were grouped with other sub-channel cables, while one Balance of Plant cable was routed in a common conduit with other RPS cables.

A review of the Unit 3 SDV RPS control logic indicates that Unit 3 does not have the same RPS logic configuration and is not susceptible to a single failure.

On October 30, 1995, a joint Corporate and Station (Dresden and Quad Cities - Dockets 50-254 and 50-265) root cause team was assembled for event investigation.

C. CAUSE OF EVENT:

A modification installed on the Unit 2 SDV, completed in 1985, to meet the requirements of NRC IE Bulletin 80-17, Failure Of 76 Control Rods To Fully Insert During a Scram At a BWR, provided an incorrect design for the revised SDV control logic and cable routing. The single active failure requirement of the project plan and GE (RPS cable) separation criteria were not met.

The Architect-Engineer (A/E) design reviews failed to identify that a single failure analysis was not performed and the GE separation criteria were not met. In addition, the project plan and design specification did not contain reference to the GE RPS system criteria. Also, the ComEd procedural and management expectations for the review process of modifications designed by A/Es was inadequate.

The focus of the modification work was directed at the new type of level switches used in the modification. A single failure analysis was performed on the new switches but not on the already existing RPS control relays. The already existing RPS control relays were only rewired by the modification making it appear that they were not affected by the modification scope.

The root causes of this event was an inadequate design review process and inattention to detail, NRC cause code B.

D. SAFETY ANALYSIS:

The SDVs are to receive and contain the water exhausted from all of the CRDs during a Reactor scram. The SDV RPS control logic would have provided the necessary actuation and scram signals if the SDV level reached an unacceptable level as verified by the previous surveillance testing. The maintenance history for the initiation relays (590-100A, B, C and D) show that there have been no relay failures since the SDV RPS control logic was modified.

However, if a single failure of the initiation relay would have occurred along with a high level in one of the SDVs, a scram signal would not have been initiated. If, while in this condition, a RPS scram signal had occurred, only the CRDs controlled by the bank of HCUs unaffected by the single failure scenario would have fully inserted into the reactor core. During this scenario, upon reaching a predetermined reactor pressure or level set points, the recirculation pumps are automatically tripped and negative reactivity would be added. In addition, Manual Operator action would complete the reactor shutdown, which could include the use of the Stand By Liquid Control (SBLC) system or draining the SDV.

The ComEd Probabilistic Risk Assessment (PRA) Group performed a bounding quantitative analysis using the current Dresden Unit 2 PRA model with guidance on failure-to-scram quantification from the industry BWR IPE methodology.

Further qualitative evaluation shows that the undetected filling of one of the SDVs without the expected full scram is highly unlikely because of the Dresden Unit 2 SDV modifications to meet the requirements of NRC IE Bulletin 80-17. The evaluation also shows that, during an ATWS scenario involving the single failure of concern, a power reduction would be expected from scramming the CRDs controlled by the bank of HCUs unaffected by the single failure scenario.

Based on the bounding quantitative analysis and the qualitative evaluation, the ComEd PRA Group concluded that the impact of the Dresden Unit 2 SDV level switch RPS logic failing to meet the single failure criterion is Non-Risk-Significant.

The Balance of Plant control cable routed in the RPS conduit does not have sufficient voltage to effect the other RPS control cables in the conduit.

The RPS sub-channel leads routed in the same conduit would not have prevented the SDV RPS control logic from initiating a scram. Any impact trauma to the conduit that could cause the wire to be damaged would result in a half scram signal.

Dresden considers this event to be a significant design problem, but the events' safety significance is minimal. There was no challenge to the system, no single failure occurred nor was there a history of single failure for the initiation relays, and there is a low probability of occurrence involving the single failure of concern.

E. CORRECTIVE ACTIONS:

Senior Station Management was notified of the incorrect logic.

Quad Cities and LaSalle Nuclear Power Stations were notified of the incorrect SDV RPS logic.

The SDV RPS logic and cable routing was corrected prior to the re-loading of Unit 2's fuel. Final testing and modification close out of the correction will be performed prior to starting up from the current refueling outage. (237-180-95-01901)

The Unit 3 SDV RPS cable routing was reviewed and no lack of RPS cable separation issues were identified.

The significant modification process improvements undertaken in 1986 in response to the NRC Safety System Outage Modification Inspection which address the root cause identified include: requirement for detailed conceptual design and review meeting, training of personnel on detailed responsibilities, and the issuance of ENC-QE-06.3, Engineering Evaluation Of Designs Provided by Outside Organizations. ENC-QE-06.3 provided detailed requirements to ComEd engineers reviewing designs made by A/Es and included a detailed checklist to address the requirements of the review process. This procedure challenges the design inputs for the modification and their considerations in the modification design.

A review of the new modification process will be performed to assure changes made in 1986 are still in place. (237-180-95-01901S1)

A sampling of modifications designed by the same A/E and/or reviewed by the same cognizant ComEd engineers (as well as others) prior to the 1986 modification process improvements will be reviewed, with emphasis on single failure and system interactions. (237-180-95-01902S1)

F. PREVIOUS OCCURRENCES:

LER/Docket Number	Title
94-010/0500237	HPCI Room Cooler Fan Minimum Starting Voltage Above Second Level Undervoltage Relay Setpoint The root cause of the event was a deficiency in the review process for the 120 VAC motor starter for the HPCI Room Cooler Fans. Corrective actions included review of the specific modification and improved process since the original deficiency. A sampling of previous modifications for similar errors was not performed.
94-013/0500237	Auto Initiation of Diesel Generator 2/3 During Modification Due to Design Error The cause of the event was a wiring error during the installation of the Bus 33-1 and Bus 23-1 Tie modification due to a design error in the installation drawings (see Attachment A). Corrective actions included review of the specific modification and improved process since the original deficiency. A sampling of previous modifications for similar errors was not performed.
94-13/0500249	Missed Tech Spec 4.6.I.1.b. Surveillance (Snubber Visual Inspection) The modification process did not adequately inform the site personnel of performed changes to the site. Also, an adequate review of the Unit 3 Technical Specifications would have indicated that the snubber was to be inspected on a periodic interval and would have directed the modification engineer to ensure this surveillance is performed. Corrective actions included review of the specific modification and improved process since the original deficiency. A review of all Unit 2 and 3 modifications that affect snubbers since 1980 will be performed to determine any additional snubbers that were added.
95-001/0500237	Inoperable Control Room HVAC Booster Fans, due to improperly sized thermal overload heater devices The root cause of the fan trip was determined to be inadequately sized thermal overload heater to account for all conceivable operating conditions. Corrective actions included review of the specific modification and

improved process since the original deficiency. A sampling of previous modifications for similar errors was not performed.

95-011/0500237 — Unit 2 and Unit 3 Nitrogen Make-up Flow Found Not to Meet Technical Specifications Due to Not Clearly Establishing the Design of the Nitrogen Make-up System.

The design of the normal nitrogen make-up paths was not clearly established. The design was not maintained or sufficiently verified as a succession of corrective maintenance items and modifications were completed which affected the system. Corrective actions included review of the specific modification and improved process since the original deficiency. A sampling of previous modifications for similar errors was not performed.

G. COMPONENT FAILURE DATA:

None.