NRC FORM 366 (4-95) U.S. NUCLEAR REGULATORY COMMISSION

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LICENSEE EVENT REPORT (LER)

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Dresden Nuclear Power Station, Unit 2	05000237	1 of 6

TITLE (4)
Spurious Local Power Range Monitor Spike Results In Full Reactor Scram Caused By Design/ Manufacturing Deficiency And Management 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: N/A	DOCKET NUMBER
12	23	97	97	019	00	01	22	98	FACILITY NAME: N/A	DOCKET NUMBER

OPERATING MODE (9)	1	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR § (Check one or more) (11)							
			20.2201(b)		20.2203(a)(2)(v)		50.73(a)(2)(i)		50.73(a)(2)(viii)
POWER LEVEL (10)	099		20.2203(a)(2)(i)		20.2203(a)(3)(i)		50.73(a)(2)(ii)		50.73(a)(2)(x)
			20.405(a)(1)(ii)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
			20.2203(a)(2)(ii)		20.2203(a)(4)	X	50.73(a)(2)(iv)		OTHER
			20.2203(a)(2)(iii)		50.36(c)(1)	X	50.73(a)(2)(v)		Specify in Abstract below or in NRC Form 366A
			20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Rob Whalen, Plant Engineering	(815) 942-2920 Ext. 2462

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 December 23, 1997, at approximately 1141, during performance of Dresden Instrumentation Surveillance (DIS) 0500-01, Reactor Vessel High Pressure Scram Pressure Switch Calibration, an unexpected Reactor protection System (RPS) Channel A half scram was received from a spiking Local Power Range Monitor (LPRM). This in conjunction with the expected RPS Channel B half scram already received during DIS 0500-01, resulted in an unexpected full reactor scram. Following the reactor scram, a feedwater transient occurred which could have resulted in water entering the High Pressure Coolant Injection (HPCI) steam supply. However, the HPCI system was isolated at the time of occurrence. The root cause of the LPRM spike was a design/manufacturing deficiency of the LPRM detector and an inadequate review of a GE SIL which provided actions to reduce the probability of spurious LPRM spikes. The root cause of the feedwater transient was design deficiency of the feedwater control system [SJ]. The corrective actions include: possible expediting of the schedule to replace the LPRM detectors with the improved type, reevaluating the GE SIL, and modification of the feedwater control system. The overall safety significance of the is event was minimal.

PLANT AND SYSTEM IDENTIFICATION:

General Electric - Boiling Water Reactor - 2527 MWt rated core thermal power

Energy Industry Identification System (EIIS) Codes are identified in the text as [XX] and are obtained from IEEE Standard 805-1984, IEEE Recommended Practice for System Identification in Nuclear Power Plants and Related Facilities.

EVENT IDENTIFICATION:

Spurious LPRM Spike Results In Full Reactor Scram Caused By Design/ Manufacturing Deficiency And Management Deficiency

A. PLANT CONDITIONS PRIOR TO EVENT:

Unit: 2 Event Date: 12/23/97 Event Time: 1141 CST

Reactor Mode: 1 Mode Name: Run Power Level: 99%

Reactor Coolant System Pressure: 1002 psig

The High Pressure Coolant Injection system was out of service with its steam supply isolated prior to this event.

B. DESCRIPTION OF EVENT:

This LER is being submitted in accordance with 10 CFR 50.73(a)(2)(iv) which requires the reporting of any unexpected actuation of the Reactor Protection System [JC](RPS). Additionally, this submittal is reporting an event or condition that alone could have prevented the fulfillment of the safety function of a system needed to mitigate the consequences of an accident in accordance with 10 CFR 50.73(a)(2)(v)(D).

On December 23, 1997, at approximately 1141, during performance of Dresden Instrumentation Surveillance (DIS) 0500-01, Reactor Vessel High Pressure Scram Pressure Switch Calibration, which had RPS Channel B in the trip condition, an unexpected RPS Channel A half scram was received. The channel A half scram signal in conjunction with the RPS Channel B half scram signal resulted in an unexpected full reactor scram.

On December 23, 1997, at approximately 0743, Instrument Maintenance (IMD) received Operations authorization to begin DIS 0500-01 in accordance with the station work schedule. This surveillance was started at 0748 and progressed in accordance with the surveillance procedure. At approximately 1116, calibration was started on pressure switch (PS) 02-0263-55D, which inputs an expected RPS Channel B half scram. At approximately 1128, the as-found setpoint value was found outside of tolerance. In accordance with DIS 0500-01, the IMD Control System Technician (CST) adjusted the setpoint within tolerance. At approximately 1140, adjustments had been completed and IMD was preparing to request Operations to reset the RPS Channel B half scram prior to returning the pressure switch to service. At approximately 1141 and prior to resetting the Channel B half scram, Local Power Range Monitor [IG](LPRM) 2D-24-41 spiked causing an Average Power Range Monitor [IG](APRM) 2 High-High, generating an RPS Channel A half scram. This resulted in a full reactor scram.

Following the scram, a feedwater level control system [SJ] transient was experienced. Reactor water level lowered to -13 inches below instrument zero (instrument zero is 144 inches above top of active fuel) and then recovered to above +60 inches. The appropriate Dresden Emergency Operating (DEOP) procedure was entered, DEOP 100, Reactor Control. At approximately 1235, the reactor scram was reset. DEOP 100 was exited after maintaining normal reactor level and clearance of all entry conditions at approximately 1241. An ENS notification was made at approximately 1338.

Interviews with personnel on watch at the time of the event, review of alarm typer data, Transient Analysis Recorder (TARS) data, Control Room charts, logs, procedures, and other such data indicates that the plant systems responded generally as would be expected for a scram from full power, with the exception of the feedwater control system.

LICENSEE EVENT REPORT (LER)

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

LPRM Spiking

Troubleshooting of LPRM 2D-24-41 was completed and it was determined that Power Supply voltage levels and cable/connectors performance were satisfactory. However, it was indicated by a detector breakdown test that a significant 'whisker' was present in the detector. The presence of a 'whisker' in the detector provides for a current path within the detector resulting in a spurious high signal if the current is of sufficient magnitude as it was in this case.

Several Unit 2 & 3 LPRMs have a history of spiking and monthly Current Voltage (I-V) checks are performed to reduce the probability of a spurious high signal resulting from a 'whisker'. However, LPRM 2D-24-41 is not among the previously identified LPRMs with spurious high signal spiking problems. LPRM 2D-24-41 was installed during cycle 10 (January 1985), and is a Reuter Stokes type RS212 and has not exhibited evidence of spiking in the past 5 years.

In 1989, GE issued Service Information Letter (SIL) 500 which identified design and manufacturing problems with some LPRM detectors, including the RS200 series. As a result of GE SIL 500, Dresden implemented actions to address problem detectors. One of the GE recommendations specified performance of a quarterly I-V curve on all LPRMs, regardless of spiking problems, to reduce the frequency of spikes. This GE recommendation was reviewed in 1990 and determined to be unnecessary. The I-V curve would only be performed when a given LPRM demonstrated a spiking problem.

Feedwater Level Control

The reactor water level decreased from +30" to -13" in 12 seconds based on TARS medium range level indicator A and B. A post scram level shrink of this magnitude is expected due to void collapse. The feedwater control system properly responded to the low level by opening both Feedwater Regulation Valves (FRV) and providing additional coolant flow to the vessel. However, as level recovered, the slow response of the feedwater level control system to the level recovery caused a significant increase in reactor level. The increase in level resulted in the trip of the operating Reactor Feed Pumps. Reactor level reached at least +60 inches above instrument zero (limit of span of the medium range instrumentation) and would have entered the High Pressure Coolant Injection [BJ](HPCI) steam supply system had not the HPCI system been isolated during the event. The HPCI system was isolated (inoperable) during this event due to troubleshooting of an unrelated condition (reference LER 97-018, docket number 05000237).

Review of system configuration records indicates that during the last Unit 2 refuel outage (D2R14) a modified single-element control configuration was implemented. This modification in part changed the setpoint setdown logic. The setpoint setdown logic delayed the initiation of the setpoint setdown following a scram and uses lagged vessel level input to the control system. In this event, this resulted in a delayed feedwater control system response to the increasing vessel level and thus, exceeding +60 inches.

No other system or component inoperabilities have been identified which contributed to this event.

C. CAUSE OF EVENT:

The root cause of the Unit 2 scram was Management/Quality Assurance Deficiency, NRC cause code E. During the review of GE SIL 500, an appropriate balance between LPRM spike avoidance and manpower/risks associated with performing the I-V curves was not implemented. Performance of the I-V curves for all the detectors on a more aggressive frequency would have reduced the probability of spurious detector spikes and may have prevented the this event.

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An additional root cause of the Unit 2 scram is manufacturing and/or design deficiency of the LPRM detector, NRC cause code B. As identified in GE SIL 500, the LPRM detectors experienced manufacturing and design deficiencies which resulted in spurious spiking in some detectors. The spurious LPRM spike generated an RPS Channel A trip from the associated APRM while half scram testing on RPS Channel B was in progress.

The root cause of the feedwater transient was design deficiency of the feedwater control logic, NRC cause code B. The time delay (lag) on the medium range vessel level signal input into the Bailey Control System results in a delay of the setpoint setdown, and can result in a significant level increase following a scram from high power. The decision to use the lagged response scenario was based on the design inputs available at the time of implementation.

D. SAFETY ANALYSIS

As discussed above, the reactor scram occurred due to a coincident half scram situation. The safety significance of the scram was limited to the fact that it involved an automatic scram and initiation of systems as expected for a scram from full power. Plant response to the event was as expected with the exception of feedwater level control. The safety significance of the feedwater level control issue revolves around the potential for flooding the HPCI and/or the Isolation Condenser steam lines. In this particular event, it is clear that reactor water level following the scram increased such that water was introduced into the HPCI and Isolation Condenser [BL] steam lines. No water was introduced to the HPCI turbine because of the fact that HPCI was isolated at the time. Introduction of water into the Isolation Condenser system is not believed to be a threat to its operation due to the geometry of its steam line.

The issue of significance is that this scram has demonstrated that the existing configuration of feedwater level control on Unit 2 can result in the introduction of water into the HPCI steam line. This is believed to be a historical issue at Dresden; discussion with senior Operators indicates that in early years of operation it was not unusual for Operators to trip reactor feed pumps in order to prevent this type of overshoot following a scram (in fact, the current transient level response procedure referred to taking such action if necessary but did not specifically reference water in the HPCI and / or Isolation Condenser steam lines).

The UFSAR (section 6.3.3.1.3.2.1) does speak to this issue, although it does not state that flooding the line via feedwater post-scram is specifically analyzed. HPCI's design function is to provide reactor inventory post-LOCA for small breaks (breaks of small enough size whereby the reactor is not promptly depressurized via the break). The possibility of HPCI becoming unavailable due to ingestion of feedwater is not outside the design basis of the plant. The feedwater system is not safety-related, and as such this problem is similar in principle to failure of the feed pump trip switches or other portions of feedwater logic.

In addition, the Automatic Depressurization System [SB](ADS) was available to reduce the vessel pressure to allow the Core Spray [BM] or Low Pressure Coolant Injection [BO](LPCI) to provide the needed make-up inventory during this event if required.

LOCA analyses were performed for various break sizes and locations ("LOCA Break Spectrum Analysis for Dresden Units 2 and 3," EMF-97-025(P), Rev. 0, May 1997). The analysis of record for Units 2 and 3 is the double-ended guillotine break of a recirculation suction line with failure of the LPCI injection valve. The limiting break characteristics were based on the assumption that for large breaks, the HPCI system is inoperable; for small breaks, the HPCI system is operable. HPCI injection valve is the limiting single failure for small breaks. Additional small break analyses were performed to assess the impact of operation with HPCI out of service and an additional ECCS failure.

Results from small break analyses without credit for HPCI demonstrate that the ECCS Peak Clad Temperature (PCT) acceptance criterion is satisfied even if HPCI is inoperable; however, if HPCI is inoperable the limiting break will be a small break case (Single Failure - ADS). The overall safety significance of this event, based on the above, was minimal.

E. CORRECTIVE ACTIONS:

Immediately following the scram, Dresden Senior Management assembled an event response team. The team performed a detailed review of the event, evaluation of plant response, root cause evaluation, and readiness of restart of Unit 2. The findings of the team were On Site Reviewed and presented to the Plant Operating Review Committee (PORC) to assure readiness for restart. (complete)

LPRM Actions

1. All LPRM detectors, on both Units 2 & 3, which input into an APRM (scram function) have had an I-V plot performed on them. (complete)
2. The schedule for replacement of the LPRMs with upgraded types will be reviewed to determine if the schedule can be expedited. (2371809701901)
3. GE SIL-500 will be re-evaluated to determine appropriate frequency for performing I-V curves for all LPRM detectors that are identified by the SIL. (2371809701902)
4. Predefines will be created for periodic performance of I-V curves on all LPRM detectors, as determined by action 3 above. (2371809701903)

Feedwater Control System Actions

Note: Several short term modifications to improve the performance of the feedwater control system following a scram were reviewed. It was determined that the appropriate conservative decision was to avoid short term modifications to the system and to install the full feedwater control system modification at the soonest opportunity. This decision was predicated on providing Operator training on manual feedwater control following a scram and revision of the operating procedure for transient level control, both of which were promptly completed as part of the restart plan.

1. Operations personnel were trained on controlling Reactor water level following a scram on the Dresden simulator. (Complete)
2. Operations procedure on transient level control was revised to recommend the securing of at least 1 Feedwater Pump following a scram, after verification of increasing reactor level. (Complete)
3. The Unit 2 feedwater control system will be modified to improve scram recovery performance of the reactor water level control. (Including three element control.) (2371809701904)
4. Unit 3's feedwater control system was reviewed. The Unit 3 feedwater control system has prompt setpoint setdown and does not use lagged level signals for level control, making the Unit 3 feedwater control system less susceptible to post scram vessel level increase to the HPCI steam line. (complete)

F. PREVIOUS OCCURRENCES:

No previous events were identified that were attributed to LPRM spiking caused by inadequate OPEX review or Design/ Manufacturing deficiencies.

For feedwater in HPCI steam supply system

LER/ Docket Number	Description
95-008/ 05000249	Upon an automatic scram from high turbine vibration and a load reject, the Unit 3 reactor vessel level increased above the HPCI steam supply system nozzle, resulting in water entering the HPCI steam lines. The root cause of the event was attributed to a design deficiency with the feedwater control system. The problem centered around the initiation setpoint for the flow control mode of the system. The corrective action focused on the specific problem with the flow control initiation setpoint which included lowering the Unit 3 flow control initiation setpoint. Unit 3 scrams from power following this changed demonstrated that the change was successful. However, the corrective actions from this event would not have prevented the December 1997 event. The subject of the deficiency in the 1995 event was the flow control mode setpoint. The design implemented for Unit 2 during D2R14 used a new control system which did not include a flow control mode. Thus, lessons learned concerning flow control mode setpoint would not have been applicable to the design implemented in Unit 2 during D2R14 (1996).

G. COMPONENT FAILURE DATA:

None