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

LICENSEE EVENT REPORT (LER)

APPROVED BY OMB NO. 3150-0104
EXPIRES 04/30/98

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

TITLE (4)
Unit 2 Reactor Scram From A Main Turbine Trip Due To Inadequate Design Review Performed During Modification

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
									N/A	05000
01	13	98	98	002	00	02	12	98	FACILITY NAME N/A	DOCKET NUMBER 05000

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)		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)
R. Jackson, Maintenance Staff	(815) 942-2920 ext 2483

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 January 13, 1998 at 1904 hours with Unit 2 operating at 100 percent power a Main Turbine [TA] trip, 2B and 2C Reactor Feedwater pumps (RFP)[SJ] trip resulted in a subsequent reactor scram, as well as, a Group II and Group III isolation. At the time of the event, an Instrument Maintenance(IM) technician was valving in local level indicator 2-263-59A following calibration of the indicator per Dresden Instrument Surveillance (DIS) 0263-14, Local Reactor Level Indicator(Safe Shutdown) Yarways LI 2(3)-263-59A and LI 2(3)-263-59B Calibration. The level indicator is installed on a common sensing line with Rosemount level transmitters 2-263-23A and 23C. LT 2 -263-23A and 23C initiate a turbine trip and feedwater pump trip on high reactor water level. The root cause of the Unit 2 scram was found to be inadequate design review during the modification of Yarway level switch replacement. During development of the modification, the impact that the modification would have on plant maintenance activities was not identified by the design team (Design Engineering, Plant Engineering, Maintenance, etc.). The safety significance of the event was minimal since all plant systems operated as designed. Corrective actions included revision to DIS 263-14 to be performed off-line only and emphasizing to design engineers the necessity of thorough design reviews, detail industry event database searches and the need for detailed questioning of maintenance and operations personnel during design scope meetings. There have been no previous occurrences of this type of event.

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:

Unit 2 Reactor Scram From A Main Turbine Trip Due To Inadequate Design Review Performed During Modification

A. PLANT CONDITIONS PRIOR TO EVENT:

Unit: 2 Event Date: 01-13-98 Event Time: 1904 CST
Reactor Mode: 1 Mode Name: Run Power Level: 100
Reactor Coolant System Pressure: 1000 psig

No other equipment was out of service that contributed to this event.

B. DESCRIPTION OF EVENT:

This LER is being submitted pursuant to 10 CFR 50.73(a)(2)(iv) which requires the reporting of any event or condition that results in a manual or automatic actuation of any Engineered Safety Feature, including the Reactor Protection System [JC].

On January 13, 1998 at 1904 hours with Unit 2 operating at 100 percent power a Main Turbine [TA] trip, 2B and 2C Reactor Feedwater pumps (RFP)[SJ] trip resulted in a subsequent reactor scram. At the time of the event, an Instrument Maintenance (IM) technician was valving in local reactor level indicator 2-263-59A following calibration of the indicator per Dresden Instrument Surveillance (DIS) 0263-14, Local Reactor Level Indicator(Safe Shutdown) Yarways LI 2(3)-263-59A and LI 2(3)-263-59B Calibration. The level indicator is installed on a common sensing line with Rosemount level transmitters 2-263-23A and 23C. LT 2 -263-23A and 23C initiate a turbine trip and feedwater pump trip on high reactor water level.

The IM technician was returning the indicator to service by valving in the reference leg which created a pressure spike on the common sensing line. Consequently, an invalid high water level signal was sensed by level transmitters 2-263-23A &23C which initiated a trip of the 2B & 2C reactor feedwater pumps and main turbine. This resulted in the a reactor scram and Group II and III isolations.

Due to the loss of feedwater flow and the subsequent reactor scram, reactor water level decreased from +30 inches to -22 inches, which is typical of previous full power scrams. As designed, when reactor water level decreased to +8 inches, Reactor Water Cleanup [CE] (Group I), Containment Ventilation and Auxiliaries [VA] (Group II) isolated and the Standby Gas Treatment system [BH] started. The 2A RFP automatically started, and feedwater flow automatically increased to the maximum capacity for one RFP to restore reactor water level.

Once water level was recovered to +20 inches, the Nuclear Station Operator (NSO) placed the feedwater control system in Manual, and reduced feedwater flow in accordance with plant procedures. Prior to reestablishing feedwater flow, vessel level decreased to -30 inches. Reactor water level was restored and stabilized at approximately +30 inches.

Interviews with Operations personnel on watch at the time of the event, review of alarm typer data, TARs data, control room charts, logs, procedures and other data indicated that plant systems responded as would be expected for a scram from full power. A prompt investigation team was formed to investigate the scram event.

C. CAUSE OF EVENT:

The root cause of the Unit 2 reactor scram was found to be inadequate design review, as a result of oversight, during the modification of Yarway level switch replacement. Modification M12-2-94-002 that was completed on March 20, 1996 replaced Yarway level switches with the existing Rosemount transmitters. This modification left the local Yarway indicators LI 2-263-59A & 59B located on racks 2202-5 & 6 in-place due to Safe Shutdown Analysis requirements. During development of the modification, the impact of the modification on plant maintenance activities was not identified. The design team (Design Engineering, Plant Engineering, Maintenance, etc.) did not consider the characteristic differences between the Yarway and Rosemount transmitters. [NRC Cause Code: A]

Discussions with IM personnel indicated that the 263-59 instruments as well as other instruments manufactured by Yarway are more prone to inducing sudden changes in pressure than are other instruments used in the plant. Yarway instruments have large internal volumes relative to other instruments (Rosemounts, etc.) on the racks and thus Yarways have large displacements when compared to displacements for Rosemounts. Therefore, pressure spikes experienced during valving operations of Yarway instruments may be larger. The pressure spike produced on the common sensing line when the IM technician opened the indicator's isolation valve was sensed by Rosemount level transmitters 2-263-23A & 23C. These transmitters provided trip signals to the main turbine and 2B & 2C reactor feedwater pumps.

Prior to the modification, the level transmitters were Yarway devices and were replaced with existing Rosemount 1153 model transmitters. Due to the design differences of the two transmitters, there are substantial response time variations. The Yarway is a mechanical device while the Rosemount is an electronic device. Therefore, if a pressure spike is induced into the common sensing line, the Rosemount would provide a shorter response time trip function than the Yarway. Consequently, the modification replaced the Yarway transmitters with higher sensitivity devices.

There have been industry events documenting Rosemount 1153 transmitters causing ECCS initiations during valving operations on common sensing lines. License Event Report (LER) document 87-030 at Hope Creek station and LER 92-013 at LaSalle station discusses spurious initiations of ECCS systems caused by quick response time of Rosemount transmitters during instrument valving operations.

Pressure spikes were generated by valving operations on common sensing lines that were sensed by Rosemount transmitters. LER document 87-030 at Hope Creek indicated that Rosemount model 1153 transmitters have such a quick response time that even a minor instantaneous pressure spike will cause the transmitters to trip. The difference in the Yarway and Rosemount transmitter's design and their impact on maintenance activities were not identified during the development of modification M12-2-94-02. Consequently, DIS 263-14 did not have the needed prerequisites to only perform this activity with the unit in a shutdown mode. This would have eliminated the risk of a turbine trip and subsequent reactor scram.

The investigation also concluded that there was two contributing causes to the scram event.

One contributing cause was identified as IM personnel developing a high comfort zone for high risk evolutions. This was caused by the day-to-day nature of IMs performing work activities which can potentially trip the unit. [NRC Cause Code: E]. This conclusion was based upon interviews with IM personnel who prepared, reviewed and approved the surveillance procedure during the modification process, the modification design engineer, IM personnel who performed the surveillance in the field at the time of this event, and IM personnel who reviewed the procedure during scope deletion of the surveillance activity from the refuel outage to an on-line activity. Most of the individuals indicated that the activity of calibrating Yarways on-line had been performed successfully for many years prior to the modification. This was the first time in performing calibration of a Yarway instrument with the current procedure DIS 263-14 while the plant was operating.

An additional contributing cause was the removal of this work activity from the outage without appropriate technical review. DIS 263-14 was performed once on each unit. Those two occasions occurred while each unit was shutdown as a part of the post modification testing. The surveillance activity was scheduled to be performed during the upcoming D2R15 refueling outage but was removed from the outage schedule and placed on the unit's daily schedule for on-line activities. This task was removed from the outage schedule per an outage scope deletion form in accordance with Dresden Administrative Procedure (DAP) 18-04, Management Of Planned Outages. Interviews conducted with the IM personnel who originated the scope deletion form and with management personnel who approved the scope deletion indicated that it was done in accordance with our current procedural guidelines. The originator of the form was the only IM personnel that approved the form. The logic used to justify performing the task on-line was that it only provided indication and historically, Yarway calibrations were successfully performed on-line prior to the modification. The scope deletion form was not intended to perform technical reviews. However, the lack of a thorough technical review in order to remove this item from outage activities was a contributing cause to this event. [NRC Cause Code: E]

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

D. SAFETY ANALYSIS

During this event, the 2B & 2C RFPs tripped coincident with the turbine trip on the momentary invalid high reactor water level signal. Due to the loss of feedwater flow and the subsequent reactor scram, reactor water level dropped from +30 inches to -22 inches, which is typical of previous full power scrams. Per design, the 2A RFP automatically started, and feedwater flow automatically increased to the maximum capacity for one RFP to restore reactor water level.

Once water level recovered to +20 inches, the NSO placed the feedwater control system in Manual, and reduced feedwater flow in accordance with plant procedures. Prior to reestablishing feedwater flow, vessel level decreased to -30 inches. Reactor water level was restored and stabilized at approximately +30 inches. The combination of automatic feedwater operation and operator actions restored of normal reactor water level within 80 seconds of the scram.

The feedwater system operation during this event was reviewed against the previous Unit 2 scram event taking into account changes made to operator training for post scram feedwater system operation. Equipment performance and operator actions in response to this event were found to be acceptable in preventing a recurrence of the vessel overfill from the previous Unit 2 scram.

Based on the review of this event, it is concluded that the safety significance of this event was minimal. All system operated as expected to limit the consequences of this occurrence. The health and safety of the public were not compromised as a result of this event.

E. CORRECTIVE ACTIONS:

Immediate corrective actions included the following:

1. The prompt investigation was reviewed by Operations shift personnel. (Complete)

2. An administrative hold was placed on all activities which have been removed from D2R15 outage and re-scheduled as on-line maintenance. (Complete)

Corrective actions to prevent recurrence are as follows:

1. DIS 0263-14 has been revised to be performed as an off-line activity. (Complete)

2. DIS 0263-14 will be revised to ensure that the reactor vessel is flooded up prior to performing the calibration procedure. (NTS #237-180-98-00201)

3. IMD will perform Just-In-Time training to address conservative decision making and the importance of stopping to question risks considered excessive. (Complete)

4. A multi-discipline team consisting of Maintenance, Operations and Engineering will be formed to review all activities that have been removed from D2R15 refueling outage and rescheduled as on-line maintenance. (NTS 237-180-98-00202).

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

5. A preliminary review was conducted to identify items removed from the D3R15 outage. (Complete)

6. A formal scope and review process will be implemented for removing PM activities and surveillance procedures from refueling outage frequency to performance on-line. (NTS # 237-180-98-00204)

7. Dresden Administrative Procedure (DAP) 21-03, Processing Plant Design Changes, has been revised, since this modification, to include instructions for all involved design team members (Operations, Engineering, Maintenance etc.) to identify the impact of the design on their respective departments and procedures. (Complete)

8. A training request (TR # 98-186) has been submitted to the Maintenance Training Advisory Committee to evaluate the need for conservative decision making for the maintenance departments. (Complete)

Additional corrective actions:

1. The Design Engineering Manager will discuss this event with all design engineers reminding them of the necessity of all team members following DAP 21-03 to ensure thorough design reviews are conducted. In addition, emphasis will be placed on conducting detail industry event data base searches and the need of thorough questioning of maintenance and operations personnel during design scope meetings. (NTS 237-180-98-00203)

F. PREVIOUS OCCURRENCES:

A search was conducted of the PIF data base for the last two years concerning LERs in the area of inadequate design review for plant impact. No previous LER events were found.

An OPEX review was also conducted and two similar events were found. These are discussed in the Cause Of Event section of this report.

G. COMPONENT FAILURE DATA:

None.