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ComEd

February 7, 1997

JSPLTR #97-0027

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

Enclosed is Licensee Event Report 97-002, Docket 50-237, which is being submitted pursuant to 10CFR50.73 (a)(2)(i)(B), any operation or condition prohibited by the plant's Technical Specifications.

This correspondence contains no open or incomplete commitments.

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: A. Bill Beach, Regional Administrator, Region III
NRC Resident Inspector's Office

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

APPROVED BY OMB NO. 3150-0104
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 4

TITLE (4)
Turbine First Stage Pressure Scram Bypass Switches Adjusted Outside Technical Specification Limits Due to Personnel Error

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
01	09	97	97	-- 002 --	00	02	07	97	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9)	N	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)							
POWER LEVEL (10)	100		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)		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)	X	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)
T. Lane - IMD Training Coordinator Ext. 2833	(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 January 09, 1997, Instrument Maintenance technicians performed a surveillance to calibrate the Unit 2 Turbine First Stage Pressure 45 percent Scram Bypass pressure switches. Errors in following the procedure resulted in as-found data being recorded incorrectly. Due to the nature of the errors (trip and reset data reversed), the switches were adjusted incorrectly to a setpoint which exceeded Tech Spec limits by approximately 1 to 2 percent. This condition existed for 4 days until discovered by a subsequent performance of the same surveillance due to a required Setpoint Change. Switches were immediately recalibrated to proper setpoints. During the 4 days the setpoints exceeded Tech Spec limits, a reactor scram would have occurred, but at a higher power than allowed by Tech Specs after a generator load reject or turbine trip.

During this period, reactor power level was significantly greater than the trip setpoint, and had a turbine trip occurred, the Reactor Protection System would have performed as designed. Analysis of potential events requiring the trip indicate the safety significance of this event is 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 Recommendation Practice for System Identification in Nuclear Power Plants and Related Facilities.

EVENT IDENTIFICATION:

Turbine First Stage Pressure Scram Bypass Switches Adjusted Outside Technical Specification Limits Due to Personnel Error

A. PLANT CONDITIONS PRIOR TO EVENT:

Unit: 2 Event Date: 01/09/97 Event Time: 0300

Reactor Mode: N Mode Name: Run Power Level: 100

Reactor Coolant System Pressure: 1000 psig

B. DESCRIPTION OF EVENT:

This issue is reportable pursuant to 10CFR50.73 (a)(2)(i)(B), any operation or condition prohibited by the plant's Technical Specifications.

On January 9, 1997, during performance of surveillance DIS-0500-07 Turbine First Stage Pressure 45 Percent Scram Bypass, pressure switches were calibrated incorrectly. These switches provide for automatically bypassing the turbine stop valve closure or control valve fast closure reactor scram [JC] when reactor power is <45 percent.

During normal plant operations >45 percent reactor power, these four switches are in the trip condition since turbine first stage pressure is greater than the existing trip setpoint of 387 psig. During the surveillance, test pressure was slowly lowered until the reset point of the pressure switch was reached. The Instrument Maintenance technician noted the reset point, and recorded the as-found data for the pressure switch reset incorrectly in the as-found trip column, instead of in the as-found reset column. This action proved to be significant as the technician now believed he had measured and recorded the as-found trip when he had actually recorded the reset value in the trip column.

The crew continued with the procedure and raised the pressure to check the as-found value for the trip point (Tech Spec related setpoint). However, the crew, now believing that the reset point was being checked, measured and recorded the trip setpoint as the reset value. It now appeared the switch was out of tolerance and needed calibration. The crew adjusted the switch with the result being that the as-left condition was in excess of the Tech Spec limit of 395 psig.

This mindset continued and the remaining three switches were similarly incorrectly adjusted, thus placing all four switch trip points in excess of Tech Spec limits by as much as 10 psig. Performance Improvement Forms (PIF) were generated regarding each out of tolerance condition for the four pressure switches.

A different IM technician who reviewed the data did not take into consideration the relative direction of the pressure signal, i.e., increasing or decreasing, and failed to identify the error. The IM Supervisor also failed to identify the error in their reviews.

On January 13, 1997, a different IM crew performed the same surveillance, DIS-0500-07. This performance was due to a required engineering Setpoint Change which decreased the setpoint from 387 psig to 292.1 psig. Unaware of the data errors and subsequent calibration errors in the previous procedure performance on 01/09/97, The IM crew performed the surveillance and, in accordance with the procedure, discovered the out of tolerance and out of Tech Spec conditions affecting all four switches. IM Supervisor noted magnitude of the as-found values being out of tolerance and confirmed the as found values to be in excess of previous Tech Spec limit. Operations was notified, the switches were calibrated to the new setpoints, and four PIFs were generated in response to the as-found data being out of tolerance and in excess of Tech Spec limits.

Not knowing the origin of the errors and as a precautionary measure, the test equipment used to perform the setpoint change on 01/13/97 was checked and confirmed to be in calibration. Additionally, the surveillance was re-performed on a later shift on 01/13/97 using different test equipment. The as-found data was consistent with the as-left data from the Setpoint Change on the previous shift.

On 01/14/97, the IM Supervisor obtained data from surveillance performance on 01/09/97. His investigation revealed the switches had been incorrectly adjusted at that time. These switches had remained out of tolerance and in excess of Tech Spec limits from 01/09/97 to 01/13/97. Operations was notified at this time that the switches had been out of tolerance for four days. The Setpoint Change on 01/13/97 had corrected this condition and no further immediate corrective actions were necessary.

Other work performed by the same crew on 1/9/97 was inspected and found to be satisfactory.

No other system or components were inoperable or out of service which contributed to the event. In addition, no manual or automatic engineered safety feature (ESF) actuation occurred as a result of this event.

C. CAUSE OF EVENT:

The cause of this event is attributed to personnel error. (NRC Cause Code A) During performance of the surveillance on 01/09/97, IM personnel adjusted the trip setpoints of all four turbine first stage pressure 45% SCRAM bypass pressure switches resulting in Tech Spec limits being exceeded.

Task analysis was performed and the root cause was determined to be failure to strictly follow procedure steps. Contributing to this was a combination of inattention to detail and failure to self check.

D. SAFETY ANALYSIS:

During the period that the switches were improperly calibrated, if an event had occurred which included a generator load reject or turbine trip, the RPS system would have performed as designed, and a reactor scram would have occurred as required.

The worst case situation would have been if reactor power had been just slightly greater than the power level associated with a turbine first stage pressure of 395 psig. Since the associated reactor scram would have been bypassed up to the incorrect setpoint of 405 psig this means that a reactor scram would not have occurred when required. The reactor would continue to operate at power with a tripped turbine, and steam flow slightly greater than the capacity of the bypass valves. In this situation, RPS initiation would have been provided manually or automatically as a result of high reactor pressure or hi neutron flux.

An analysis was performed which determined that the thermal limits for rated conditions (100 percent power and 100 percent flow) are bounding for transients at 60 percent power or less without crediting the scram on load rejection or turbine trip.

Since the reactor trip was not bypassed during the period of incorrect adjustment, and the analysis revealed that no fuel damage would result had an event occurred, the safety significance of this event is minimal .

E. CORRECTIVE ACTIONS:

1. Surveillance performed on 1/13/97 restored switch setpoints to their appropriate settings. No further adjustments are necessary. (Complete)

2. Instrument Maintenance personnel involved in this event were disciplined in accordance with Station Policy 60. (Complete)

3. Other work performed by the same crew over approximately three months, and previous performances of this same surveillance were inspected for discrepancies and found acceptable. (Complete)

4. The event was reviewed by the Department Head with all IM department management and technical personnel to reinforce expectations regarding surveillance performance and review. (Complete)

F. PRIOR SIMILAR OCCURRENCES:

A search was conducted for prior similar occurrences of this event relating to the incorrect recording and subsequent miscalibration of instrumentation. No previous events were found.

G. COMPONENT DATA

Not applicable.