NRC Form 366
(9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/88

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Perry Nuclear Power Plant, Unit 1	0 5 0 0 0 4 4 0	1 OF 0 3

TITLE (4) Misadjustment of Average Power Range Monitor Readings Due to an Error In Heat Balance Calculation Results in Technical Specification Violation

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 NAMES	DOCKET NUMBER(S)
0 2	2 7	8 8	8 8	— 0 0 9 —	0 0	0 3	2 5	8 8		0 5 0 0 0
										0 5 0 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)

OPERATING MODE (9): 1

POWER LEVEL (10): 0 3 5

[ ] 20.402(b)	[ ] 20.405(c)	[ ] 50.73(a)(2)(iv)	[ ] 73.71(b)
[ ] 20.405(a)(1)(i)	[ ] 50.36(c)(1)	[ ] 50.73(a)(2)(v)	[ ] 73.71(c)
[ ] 20.405(a)(1)(ii)	[ ] 50.36(c)(2)	[ ] 50.73(a)(2)(vii)	[ ] OTHER (Specify in Abstract below and in Text, NRC Form 366A)
[ ] 20.405(a)(1)(iii)	[X] 50.73(a)(2)(i)	[ ] 50.73(a)(2)(viii)(A)	
[ ] 20.405(a)(1)(iv)	[ ] 50.73(a)(2)(ii)	[ ] 50.73(a)(2)(viii)(B)	
[ ] 20.405(a)(1)(v)	[ ] 50.73(a)(2)(iii)	[ ] 50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (AREA CODE)
Gregory A. Dunn, Compliance Engineer, Extension 6484	2 1 6 2 5 9 - 3 7 3 7

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)

[ ] YES (If yes, complete EXPECTED SUBMISSION DATE) [X] NO

EXPECTED SUBMISSION DATE (15): MONTH | DAY | YEAR

ABSTRACT (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines) (16)

On February 27, 1988 at 04:58, the Average Power Range Monitor (APRM) readings were misadjusted during a channel calibration surveillance instruction which could have resulted in the Reactor Protection System (RPS) Flow Biased Simulated Thermal Power-High and Neutron Flux-High trips occurring outside of the setpoint allowable values. After this error was promptly discovered at 05:45, the APRMs were recalibrated by 06:09. Evaluation determined that the duration with less than the required minimum operable channels per trip system exceeded the one hour time period allowed by Technical Specification 3.3.1.

The cause of the APRM misadjustment was an error in the heat balance calculation by the process computer which used an incorrect input value for average feedwater flow. The cause of the incorrect value for the average feedwater flow has not been determined, and this problem has not recurred since this event. Contributing to this event was inadequate knowledge by the personnel involved in the surveillance instruction to identify the problem with the heat balance calculation.

As a result of this event, the surveillance instruction has been revised to include a check to ensure the validity of the heat balance calculation prior to adjusting the APRMs. Additionally, a review of other applicable surveillance requirements resulted in a revision of the Plant Round Instruction for Technical Specification Rounds to add this additional check to the existing validity checks for the process computer thermal limit calculation.

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

On February 27, 1988 at 04:58, the Average Power Range Monitor (APRM) [IG] readings were misadjusted during a channel calibration surveillance instruction which could have resulted in the Reactor Protection System (RPS) [JC] Flow Biased Simulated Thermal Power-High and Neutron Flux-High trips occurring outside of the setpoint allowable values. After this error was promptly discovered at 05:45, the APRMs were recalibrated by 06:09. Evaluation determined that the time period with less than the required minimum operable channels per trip system exceeded the one hour limit allowed by Technical Specification 3.3.1. At the time of the event, the plant was in Operational Condition 1 (Power Operation) during a plant startup with reactor power approximately 35 percent power. Reactor vessel [RPV] pressure was approximately 950 psig.

On February 27 after attaining 25 percent rated power during a plant startup, surveillance instruction (SVI)-C51-T0024, "APRM Channel Calibration Evaluation/Adjustment," was completed in accordance with Technical Specification 4.3.1.1. This SVI adjusts the APRM channel if the absolute difference from a calculated heat balance is greater than 2 percent of rated thermal power when thermal power is greater than or equal to 25 percent of rated thermal power. All APRMs were calibrated by 04:58. After performance of the SVI, the reactor power increase was restarted at approximately 05:35. At 05:45, plant operators noticed a discrepancy in the relative values of the calculated thermal power, the APRM readings, and the electrical output. Plant electrical efficiency is initially low at less than 25 percent rated power and increases as rated power is attained. Investigation of the discrepancy revealed that the heat balance calculation by the plant process computer was incorrectly determining thermal power due to an incorrect average feedwater flow input value. After resetting this value, the heat balance calculated the thermal power as 36 percent of rated power. At this time the APRM readings ranged from approximately 28 to 35 percent. SVI-C51-T0024 was reperformed utilizing the correct thermal power and all APRMs were recalibrated by 06:09.

It was determined that the first performance of SVI-C51-T0024 had adjusted the APRMs nonconservatively. With the APRMs adjusted nonconservatively, the Reactor Protection System instrumentation setpoints for APRM Flow Biased Simulated Thermal Power - High and Neutron Flux - High, and the Control Rod Block instrumentation setpoint for Flow Biased Neutron Flux - Upscale were outside the allowable value. The time period with less than the required minimum operable channels per RPS trip system and per control rod block trip function exceeded the one hour allowed by Technical Specifications 3.3.1 and 3.3.6. RPS trip system A had less than 3 operable APRMs for 74 minutes and RPS Trip System B had less than 3 operable APRMs for 72 minutes.

The cause of the APRM misadjustment was an error in the heat balance calculation by the process computer which used an incorrect input value for average feedwater flow. After resetting the computer scan for average feedwater flow, the process computer correctly calculated thermal power. The cause of the incorrect value for the average feedwater flow has not been determined, and this problem has not recurred since this event. Contributing

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

to this event was inadequate knowledge by the personnel involved in the surveillance instruction to identify the problem with the heat balance calculation.

The APRM system provides signals representative of thermal power production of the reactor core. These signals are used by various plant protection systems to prevent damage to reactor fuel in the event of unanticipated power transients. The APRMs consist of eight channels, each receiving inputs from 20 or 21 of the Local Power Range Monitors (LPRMs). Each APRM channel averages the inputs from its assigned LPRMs and provides an output signal proportional to the average of the LPRM flux signals. The APRMs are divided into two RPS Trip Systems. A single trip signal from any APRM will trip its respective RPS Trip System. A trip signal from at least one APRM in each trip system will result in a full RPS actuation and a reactor scram. As a result of the incorrect adjustment, the APRM power readings were nonconservatively set for all 8 of the APRMs. However, APRM A was found to be within 2% rated power as allowed by Technical Specifications during the second performance of SVI-C51-T0024, and therefore is considered to have been operable during this event.

A review of the conditions of this event has concluded that the plant response would have been bounded by existing analyses if a transient event had occurred. At power levels less than 40 percent, no credit is taken for the APRM scram signals for the limiting events contained in the Final Safety Analysis Report, Chapter 15. Due to the numerous plant indications available (including thermal power calculation, electrical load, APRM readings, and steam/feedwater flow), this condition would not go undetected long enough for high power levels to be reached. In fact, this condition was detected soon after restarting the reactor power increase. Plant operators promptly discovered the source of the discrepancy and then immediately contacted I&C to recalibrate the APRMs to within Technical Specifications limits. Additionally, a large margin existed between the core thermal limits and their respective Technical Specification limits during this event. For these reasons, this event is considered to have no safety significance, and to have prompt operator corrective action upon discovery of the discrepancy. No previous similar events were identified.

As a result of this event, several corrective actions have been performed. Troubleshooting of the process computer has been conducted to determine the cause of the incorrect average feedwater flow value. This troubleshooting has not revealed the cause of the problem, and this problem has not been experienced again since this event. SVI-C51-T0024 has been revised to include a check to ensure the validity of the heat balance calculation prior to adjusting the APRMs. Additionally, a review of other applicable surveillance requirements resulted in a revision of the Plant Round Instruction for Technical Specification Rounds to add this additional check to the existing validity checks for the process computer thermal limit calculation.

Energy Industry Identification System Codes are identified in the text as [XX].

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY
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Al Kaplan
VICE PRESIDENT
NUCLEAR GROUP

March 25, 1988
PY-CEI/NRR-0833 L

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

Perry Nuclear Power Plant
Docket No. 50-440
LER 88-009-00

Dear Sir:

Enclosed is Licensee Event Report 88-009-00 for the Perry Nuclear Power Plant.

Very truly yours,

Cyril M. Shuster for

Al Kaplan
Vice President
Nuclear Group

AK:cab

Enclosure: LER 88-009-00

cc: T. Colburn
K. Connaughton

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