NRC FORM 366 (6-1998)

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104 EXPIRES 06/30/2001

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1): LaSalle County Station, Unit 1

DOCKET NUMBER (2): 05000373

TITLE (4): Average Power Range Monitor Reactor Scram and Rod Block Setpoints Found to be Non-Conservative After Calibration Due to Human Performance 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: None	DOCKET NUMBER
08	23	98	98	016	00	09	22	98	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9): 1

POWER LEVEL (10): 068

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)
☐ 20.2203(a)(1)	☐ 20.2003(a)(3)(i)	☐ 50.73(a)(2)(ii)	☐ 50.73(a)(2)(x)
☐ 20.2203(a)(2)(i)	☐ 20.2003(a)(3)(ii)	☐ 50.73(a)(2)(iii)	☐ 73.71
☐ 20.2203(a)(2)(ii)	☐ 20.2003(a)(4)	☐ 50.73(a)(2)(iv)	☐ OTHER
☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(1)	☐ 50.73(a)(2)(v)	Specify n Abstract below or
☐ 20.2203(a)(2)(iv)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(vii)	in NRC Form 366A

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Rebecca Karas, Reactor Engineer	(815) 357-6761 Extension 2251

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

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX

SUPPLEMENTAL REPORT EXPECTED (14)

☐ YES (If yes, complete EXPECTED SUBMISSION DATE) ☒ NO

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

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

On August 22, 1998, with Unit 1 at 24 percent power level, a non-conservative adjustment was made to the Average Power Range Monitor/Rod Block Monitor (APRM/RBM) setpoints when an incorrect value of 29 percent was calculated for flow converter percent flow. This value is to be less than both percent of rated Drive Flow and percent of Total Core Flow. Twenty-nine percent was less than (conservative to) the 30 percent value for total core flow, but greater than the 22.9 percent rated Drive flow value. This error resulted in the flow converters being set outside Technical Specifications 3.3.1 and 3.3.6 requirements. The condition was identified on August 23, 1998, with the reactor at 68 percent power. Immediate action was taken to restore the functional units to operable status. A reactor engineer had incorrectly calculated the value that was used during channel calibration. No specific verification of the calculation to the actual Drive flow was procedurally required before the number was used. The procedure has been revised to include independent verification of this calculation. Other surveillance procedures will be reviewed and revised, if necessary, to improve the control and verification of calculations. The safety consequences of the event are minimal. Unit 1 safety analyses take no credit for APRM/RBM thermal power scram or rod block.

PLANT AND SYSTEM IDENTIFICATION

General Electric - Boiling Water Reactor, 3323 Megawatts Thermal Rated Core Power

Energy Industry Identification System (EIIS) codes are identified in the text as [XX].

A. CONDITION PRIOR TO EVENT

Unit(s): 1	Event Date: 08/23/98	Event Time: 0800 Hours
Reactor Mode(s): 1	Power Level(s): 68%	RCS [AB] Temperature: 525 Degrees F
Mode(s) Name: Run		RCS [AB] Pressure: 969 psig

B. DESCRIPTION OF EVENT

On August 22, 1998, with Unit 1 at 24 percent power, the Average Power Range Monitor/Rod Block Monitor (APRM/RBM)[IG] flow converter to total core flow adjustment was performed. Instrument technicians and reactor engineering personnel completed channel calibrations using procedure LIS-NR-107, Unit 1 APRM/RBM Flow Converter to Total Core Flow Adjustment. The numerical value specified by a Qualified Nuclear Engineer as flow converter percent flow (required to be less than both percent of rated drive flow, WD, and percent Total Core Flow, WT) was calculated incorrectly. The 29 percent value specified was less than the current value of 30.01 percent of total core flow, but greater than the current value of 22.9 percent of rated drive flow. This resulted in a non-conservative adjustment. This placed the flow converters outside the requirements of Technical Specifications 3.3.1 and 3.3.6.

At 0800 hours on August 23, 1998, plant personnel noticed that the APRM rod block settings appeared high. They determined that the flow biased rod block and scram settings were out of specification due to the high flow unit readings. By this time, the reactor recirculation pumps had been shifted to high speed and reactor power level had been increased to 68 percent. The limiting conditions for operation in the Technical Specifications were immediately applied and the flow converters calibrated again using LIS-NR-107. The calibration was completed and the flow converter settings were corrected by 0911 hours.

The flow converters are adjusted during Unit startup prior to increasing power above 25 percent. In this manner, the APRM and RBM scram and rod block setpoints are adjusted conservatively for both jet pump total core flow and recirculation drive flow. In completing step E.1.10 of procedure LIS-NR-107 revision 8, the percent Flow Converter Flow is calculated based on an average percent rated Drive (WD) flow value (the average drive flow reading divided by the rated drive flow value x 100) and an average Total Core (WT) flow value. The average drive flow reading and the average Total Core (WT) flow value are obtained from the unit's process computer by the instrument technician. The Engineer is to provide a percent Flow Converter Flow value based on the actual flow data that is conservative with respect to the Technical Specification requirements. This value is then used to adjust the channel. In this manner, the flow converters are set conservatively to read percent (or less) Jet Pump (WT) Flow and percent (or less) rated Drive (WD) flow.

This event is being reported pursuant to 10 CFR 50.73(a)(2)(i)(B) - Any operation or condition prohibited by the plant's Technical Specifications.

C. CAUSE OF EVENT

The Qualified Nuclear Engineer incorrectly calculated the percentage of rated Drive (WD) flow, and therefore specified a value conservative with respect to WT but non-conservative with respect to WD, for use in procedure LIS-NR-107. The incorrect value met the procedure verification requirement performed by Instrument Maintenance personnel to be less than the value for average Total Core Flow percentage and was used to adjust the flow converters. There was no independent verification in the procedure to ensure the value was less than the percent rated drive flow. The cause of the event was human performance error from incorrectly entering either the average drive flow or the rated drive flow values into a calculator. The Engineer was familiar with the procedure and had performed the calculation previously. No distractions, fatigue or other adverse conditions were identified.

A contributing cause is that no specific verification of the calculated value was required. Although LIS-NR-107 required a comparison of the calculated Flow Converter value to the average Total Core Flow percentage, this would not check the value against the second half of the criteria, which required the setting to be less than percent rated drive flow as well.

D. SAFETY ANALYSIS

The safety consequences of this event are minimal. In the LaSalle Unit 1 Cycle 8 licensing analyses, only the rod withdrawal error (RWE) and the Loss of feedwater heating (LFWH) analyses could potentially credit flow dependent functions for accident/transient mitigation. The RWE analyses were performed without taking credit for the RBM rod block on rod withdrawal. Therefore, a non-conservative rod block setpoint does not invalidate the analysis results. Similarly, the LFWH analysis does not credit the thermal power scram; thus a non-conservative setpoint would not invalidate that analysis. In summary, operation with the APRM and RBM scram and rod block setpoints non-conservative did not violate any of the plant safety analysis assumptions.

E. CORRECTIVE ACTIONS

1. The Flow Converters were correctly calibrated in accordance with procedure LIS-NR-107. The value used for desired percent Flow Converter Flow was verified by a second Qualified Nuclear Engineer (QNE). (This action was completed on August 23, 1998)

2. LAP-100-35, "Reactivity Management Controls," has been temporarily changed to require independent verification of calculations provided to the operating shift that are used to set plant equipment. This change will remain in effect until any other deficient procedures are corrected per Corrective Action 5.

3. Procedure LIS-NR-107, and LIS-RR-101A, B, C and D, which also calibrate the APRM flow converters, have been revised to require independent verification of the percent rated drive flow calculation. (This action is complete.)

4. Surveillance procedures used by Station work groups that could affect nuclear safety or plant reliability, such as safety limits and/or limiting safety system settings, will be reviewed to identify those that provide calculations for meeting Technical Specification requirements. (NTS# 373-180-98-SCAQ00016.01, this action is complete)

   The identified procedures will be reviewed for compliance to the following:

   - Any calculation (formula) performed in completing the procedure shall be included and the group responsible for performing the calculation shall be identified.

   - Calculations are to be performed in accordance with a controlled document.

   - An independent verification shall be performed for calculations used in adjusting plant components.

5. All procedures identified in Corrective Action 4 above which do not meet the stated criteria will be revised such that those criteria are satisfied. (NTS# 373-180-98-SCAQ00016.02). All procedures identified in Corrective Action 4 have been temporarily or permanently changed to incorporate these criteria or have been placed on administrative hold.

F. PREVIOUS OCCURRENCES

LER NUMBER	TITLE
LER 92-01	Average Power Range Monitor Set Non-Conservative due to Communication Error

G. COMPONENT FAILURE DATA

Since no component failure occurred, this section is not applicable.