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) Failure to Recalibrate Level Instruments Following Design Change 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 5	0 5	8 8	8 8	— 0 1 8 —	0 0	0 6	0 3	8 8		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9)	1	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)			
POWER LEVEL (10)	1 0 0	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)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE)	X NO				

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

On May 5, 1988, a detailed investigation determined that elevation changes to the reactor vessel level instrument condensing chambers, performed in April 1987, failed to result in corrections to test instructions or instrument recalibrations. This resulted in the setpoint for the Channel D high level scram to be outside the Allowable Value as discovered during the scheduled surveillance (SVI) on January 15, 1988. A similar situation was discovered with the Channel B Scram Discharge Volume level transmitter during the scheduled calibration SVI, on February 3, 1988. The rod block setpoint was outside the Allowable Value. Detailed investigation following these events discovered that in August 1986, a one inch error was identified in the original head correction factor. The SVI was revised, however the instrument was not recalibrated at that time. Since these deficiencies were not identified until the next scheduled surveillance, no Technical Specification Actions were completed.

The cause of these events was personnel error. Changes to the reference data caused changes to the instrument calibration, yet in neither case was the instrument identified as requiring recalib[illegible]. Review of calibration data for other instruments affected has shown [illegible] values were within the Allowable Values. All applicable SVI's [illegible]sed to correct the instrument calibrations. Other instrume[illegible]ll be reviewed for similar conditions. A program change w[illegible]o ensure required recalibrations are completed following [illegible]ation changes. Additionally, the engineering staff and s[illegible]ill receive training concerning the conditions which led to these [illegible]

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

On May 5, 1988 a review of instrument calibration data identified setpoints for two instruments outside Technical Specification Allowable Values from August 1986 to February 1988 for one and April of 1987 until January 1988 for the other without completion of required actions. During these periods of time, the plant had been operated up to 100 percent of rated reactor power.

During a plant outage in April 1987, nozzle inserts were installed in the Reactor Vessel Level Instruments [JM] reference legs to prevent level indication errors when Reactor Core Isolation Cooling (RCIC) is injecting through the reactor head nozzle. The reference leg condensing chambers and piping had to be cut off and reinstalled on the reactor vessel nozzles to provide access to install the nozzle inserts. During the modification work, the D leg condensing chamber was installed higher than previously. Subsequent to performing the work, elevation measurements indicated a change of +7/16 inches over the original values. These values are used in developing the calibration surveillance instructions (SVI). A Non-conformance Report (NR) was generated and an engineering evaluation was performed to determine the impact on the plant due to the elevation change. The design documents for the wide range level instruments supplied off this reference leg provide 2.2 inches allowable drift. The engineering evaluation therefore determined that the 7/16 inches change was acceptable and the NR was dispositioned to "use-as-is" until the next scheduled calibration of the affected instruments. However, the narrow range level instruments have only 0.6 inches allowable drift. This fact was overlooked during the NR disposition. The reactor high level scram signal is generated by the narrow range level instruments. On January 15, 1988 the setpoint for Channel D reactor high level scram was found +0.06 inches outside the Technical Specification 3.3.1 Allowable Value (+220.1 inches above top of active fuel) during performance of the calibration SVI. The D narrow range level instrument was recalibrated and returned to service.

Similar changes were made to the A, B and C reference leg nozzles with the elevation change of the condensing chambers less than the design allowable limit. Therefore, the design drawings were not changed and the SVI's were not revised. A review of previous SVI data has shown that applying the elevation change to the As Left data for all instruments associated with the A, B and C reference legs, still leaves the instruments within Technical Specification Allowable Value.

On February 3, 1988 the Channel B Scram Discharge Volume (SDV) [AA] high level rod block was found outside of Technical Specification 3.3.6 Allowable Value. The instrument was recalibrated and returned the service. Subsequent investigations determined that on August 4, 1986, the head correction factor used for the calibration SVI was found to be incorrect. The SVI was revised two days later but the responsible system engineer failed to initiate recalibration of the instrument. The instrument remained in this condition until its next scheduled performance (February 1988).

NRC Form 366A
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LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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

The cause of these events was personnel error. Due to engineering's use of incorrect data for the narrow range instruments and overlooking the elevation changes in the A, B and C reference legs, the required instruction changes and instrument calibrations were not completed. The system engineer failed to recognize the need for recalibration of the Channel B SDV level instrument when the SVI was revised.

The reactor scram signal from high reactor vessel level is intended to offset the addition of reactivity effect associated with the introduction of a significant amount of relatively cold feedwater during an overfeed condition as described in USAR Chapter 15.1.2. The setpoint methodology provides a safety margin for loop drift, calibration errors and condensing chamber movement. The Channel D reactor vessel high level scram setpoint was found to be 0.06 inches outside the Technical Specification Allowable Value which is well within the safety margin. A reactor scram on reactor vessel high level occurred September 9, 1987 as described in LER 87-064 and a Reactor Protection System [JC] actuation occurred April 28, 1988 due to reactor vessel high level as described in LER 88-014. Both of these events were evaluated to have been within the analysis envelope of USAR Chapter 15. The SDV high level rod block is designed to prevent control rod movement if control rod drive hydraulic system leakage starts filling the SDV. Should SDV level continue to rise, separate instruments provide a reactor scram signal. No system leakage great enough to fill the SDV existed and the redundant rod block instrument was operable. Therefore, this event is not considered to be safety significant. No previous similar events have been identified.

In order to prevent recurrence, the following corrective actions have been or will be taken:

1. Review of calibration data for other instruments affected by these changes has shown as left values within the Technical Specification Allowable Values.
2. Other instruments utilizing condensing chambers were reviewed for similar changes that would have affected the setpoints, none were identified.
3. Changes to the design drawings to incorporate the as-built elevations will be made.
4. All applicable SVI's for instruments associated with the A, B and C reference leg, will be revised to correct the instrument calibrations.
5. In January 1987 a change to the surveillance program was implemented which would ensure any SVI change was evaluated for impact upon plant instrumentation. Consequently, all SVIs which were last performed prior to January 1987 will be reviewed to identify any revisions since the last performance of the SVI and determine the impact of those revisions on present plant conditions.
6. A program change will be generated to ensure any future changes to the physical configuration of instrumentation will be evaluated for impact on the plant and applicable recalibrations performed.
7. The engineering staff and system engineers will be trained on the need to identify changes which affect instrument calibrations and to ensure required recalibrations are completed.

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