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United States Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Perry Nuclear Power Plant
Docket No. 50-440

Ladies and Gentlemen:

Enclosed is Licensee Event Report 1999-007, "Operating License Thermal Power Limits Exceeded During Previous Cycle Coastdown."

No regulatory commitments were identified in this report. If you have questions or require additional information, please contact Mr. Gregory A. Dunn, Manager - Regulatory Affairs, at (440) 280-5305.

Very truly yours,

John K. Wood

Enclosure

cc: NRC Project Manager
NRC Resident Inspector
NRC Region III

IE22 1/1

NRC FORM 366 (6-1998) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 06/30/2001
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	Estimated burden per response to comply with this mandatory information collection request: 50 hrs. Reported lessons learned are incorporated into the licensing process and fed back to industry. Forward comments regarding burden estimate to the Records Management Branch (T-6 F33), 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. If an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

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TITLE (4)
Operating License Thermal Power Limits Exceeded During Previous Cycle Coastdown

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
12	16	1999	1999	-- 007 --	00	01	18	2000	FACILITY NAME	DOCKET NUMBER

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)	X	50.73(a)(2)(i)		50.73(a)(2)(viii)
POWER LEVEL (10)	100		20.2203(a)(1)		20.2203(a)(3)(i)		50.73(a)(2)(ii)		50.73(a)(2)(x)
			20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
			20.2203(a)(2)(ii)		20.2203(a)(4)		50.73(a)(2)(iv)	X	OTHER
			20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)		Specify in Abstract below
			20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		or in NRC Form 366A

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Bruce A. Luthanen, Compliance Engineer	(440) 280-5389

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

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

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 December 16, 1999, at 1100 hours, personnel at the Perry Nuclear Power Plant discovered an error in the calculation of reactor thermal power. Investigation showed that a database within a software code that provides numeric input to thermal power calculations had been modified by setting specific numeric inputs to zero. This modification effectively removed the feedwater temperature compensation function from the Integrated Computer System. It was determined that operation in this condition, at full power and with feedwater temperature less than 420 degrees (not a normal operating state) could produce non-conservative values for reactor power (indicated power less than actual plant thermal power.)

In the previous operating cycle, there were two periods of time during coastdown in which feedwater heaters were removed from service while the plant returned to 100 % power. A review of the operating data from these periods concluded that the plant had exceeded licensed reactor thermal power limits (102.4%, maximum). This constitutes a violation of the Operating License, and is reportable under 10 CFR 50.73, as required by the Operating License section 2.F. Additionally, it was determined that the plant was operated with less than the required number of operable Average Power Range Monitors as a consequence of the software change. This is a violation of plant Technical Specifications, and is reportable under 10 CFR 50.73(a)(2)(i).

The NRC was notified via Emergency Notification System phone message at 1903 hours on December 16, 1999, (ENF #36518).

Power was reduced to 98% until Reactor Engineering could develop documentation for a software code change. Once the software code had been corrected, full power operation was resumed. No mode changes were made as a result of this event.

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

I. INTRODUCTION

Perry Nuclear Power Plant (PNPP) relies on several different controls to ensure that limits for reactor power are not exceeded. Thermal power data is determined by proprietary software. Core thermal power distribution is calculated with inputs from a database within the Integrated Computer System (ICS)[JA] in the 3D MONICORE (3DM) software code. One dataset within the 3DM code involves the Feedwater Compensation function. These inputs are also used in another thermal power calculation (titled the Nuclear Steam System (NSS) calculation), in the ICS. The NSS software is used by the Control Room staff to determine and control reactor thermal power, in accordance with the Operating License (OL).

Section 2.C of the OL states that 3579 Megawatts Thermal (MWTH) shall not be exceeded. These limits take into account power fluctuations, as described in the plant's integrated operating procedure, but that the average MWTH over an 8-hour period shall not exceed 3579 MWTH.

Reactor power is also monitored through Average Power Range Monitors (APRM's). These nuclear instruments utilize calculated thermal power as part of their calibration. If the calculated thermal power levels are erroneous, a subsequent error can be carried forward into the APRM calibration.

At the time of these events, the plant was in Mode 1 at 100 percent rated thermal power, with one or more feedwater heaters isolated. The reactor vessel was at approximately 1024 pounds per square inch gauge, with the reactor coolant at saturated conditions. There were no other inoperable systems, structures or components that contributed to these conditions.

II. EVENT DESCRIPTION

On December 16, 1999, plant personnel investigated a discrepancy in the MWTH values listed on two separate data screens displaying Integrated Computer System data. The investigation showed that a Feedwater Temperature Compensation (FWC) data table within the 3D MONICORE code from General Electric (GE) had been revised, and that the FWC function had been zero-ed out prior to commencing Cycle 7 operations. Both GE and Reactor Engineering reviewed this change at the time of implementation, and concluded that it would resolve a potentially misleading read-out of data in 3DM. However, the impacts of the code change beyond the 3DM code were not evaluated.

It was determined that disabling the FWC could possibly challenge thermal limits, if the plant were to be operated at lower feedwater temperatures (below 420 degrees) while at full power. This would mean that the actual thermal power was non-conservative (higher than the indicated thermal power calculated by the software.) Following a data review, there was no indication that there was a concern for the current Operating Cycle (8), but that Cycle 7 data should be examined. Following examination, it was identified that isolation of feedwater heaters had occurred in Cycle 7, and there was subsequent full-power operation with feedwater temperature less than 420 degrees.

On discovery, plant power was conservatively reduced in order to ensure adequate margin to the OL limits for the current operating cycle, and to allow Reactor Engineering to develop documentation for restoring the software. The plant returned to full power once the 3DM software had been corrected.

Reactor Engineering reviews concluded that thermal power limits had been exceeded during Cycle 7 coastdown in two instances during February, 1999. In one period beginning February 7,1999, totaling approximately 76 hours, thermal power exceeded 100 % of licensed power, peaking at 3623 MWTH (101.2% of rated thermal power). A second period of exceedence began on February 21, 1999, and lasted for approximately 71 hours.

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

During this second time period, there were two instances of approximately 2 1/3 hours total time in which 102% of rated thermal power was exceeded. This exceeded the Core Transient Analysis assumption for these periods. The maximum thermal power observed during these brief periods was 102.4% of rated thermal power.

The errors incurred by the loss of FWC subsequently produced improper calibration on several channels of APRM's. In some cases , less than three APRM's were properly calibrated for use in each division, which is a violation of Technical Specifications. The APRM's were in service in this condition during the final month of Operating Cycle 7, prior to the plant entering a scheduled refueling outage.

III. CAUSE OF EVENT

The cause of this event was identified as program and process weaknesses in the development of the software change. Insufficient administrative controls existed for the review of the software revision. This allowed the 3DM software change to be implemented without adequate reviews as to other potential impacts of the change on other plant systems. Additionally, the 3DM design description was insufficiently detailed.

IV. SAFETY ANALYSIS

The licensed power level of 3579 MWTH is the analysis basis for the Cycle 7 core. This power level was used for the initial conditions for the design basis accident and transient analyses performed. If reactor power is in excess of these levels at the start of a transient or accident, then the consequences of the accident may exceed the results estimated by the analysis. The Updated Safety Analysis Report(USAR) Section 15 contains bounding analysis for power levels greater than these initial assumptions. Although the thermal power exceeded limits specified in the Operating License, thermal power was never beyond analyzed limits.

It should be noted that the Perry Plant is also pursuing a License Amendment Request (LAR) in consideration of power uprate of five percent. The engineering analyses in support of this effort were reviewed by the Plant Operations Review Committee, and no significant hazards were identified, as defined in 10 CFR 50.91. The LAR was submitted to NRC in December, 1999.

Thermal power distribution limits are established within PNPP Technical Specifications, and are identified within the Core Operating Limits Report. These ensure that the reactor fuel is operated within design limits for both normal power operations as well as transient situations. Review of reactor parameters during the time period in which feedwater temperatures were reduced indicate that there was still margin to actual thermal power distribution limits. By maintaining margin to real thermal power distribution limits despite the reactor thermal power calculation error, operation of the fuel stayed within fuel design limits, and did not exceed Technical Specification limits.

Review of APRM setpoints indicated that a margin to the allowable values was maintained during the period of reduced feedwater temperature operation (March 1999, prior to a scheduled refueling outage). APRM channels were effectively maintained within trip tolerances by at least 0.5 %. Plant safety was not compromised by the mis-adjusted APRM channels.

This event had no safety significance.

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V. CORRECTIVE ACTIONS

A software change was prepared in accordance with plant procedures to restore the constants in the software database to their original values.

Additionally, the plant will take the following actions:

1) revise existing software procedures to ensure that appropriate interface reviews are provided for software changes.
2) revise the 3DM Software Design Description to clearly define system interrelations.

VI. PREVIOUS SIMILAR EVENTS

A search of Licensee Event Reports over the past five years from the Perry plant found that no similar events had been reported. A Condition Report from 1998 and a Potential Issue from 1996 documented other problems that had been encountered with 3D MONICORE that were not reportable.

Additionally, two industry events were identified which were similar, one from LaSalle in December 1999, and another from Susquehanna (LER 96-17).

Energy Industry Identification System (EIIS) Codes are identified in the text by square brackets [XX].