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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 1999-005, "Control Rod Moved without Re-establishing a More Restrictive Rod Withdrawal Limit during Testing." This is a voluntary submittal provided for your information.

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,

for John Wood

Enclosure

cc: NRC Project Manager
NRC Resident Inspector
NRC Region III

IE22

PDR ADOCK 05000440

NRC FORM 366 (6-1998) U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

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

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TITLE (4)

Control Rod Moved Without Re-establishing a More Restrictive Rod Withdrawal Limit During Testing

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
11	4	1999	1999 --	005 --	000	12	22	1999	FACILITY NAME	DOCKET NUMBER

		THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)							
OPERATING MODE (9)	1		20.2201(b)		20.2203(a)(2)(v)		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 November 5, 1999, following initial review of post-maintenance test data from routine control rod speed and drift calibration, an operability determination concluded that that rod movement had been made with the rod withdrawal limiter function of the Rod Control and Information system apparently inoperable. Plant procedures require that the deselect/reselect button be depressed following any rod insertion, in order to re-initialize the rod withdrawal limit. This would have produced an alarm and a rod block (preventing further withdrawal) at two notches withdrawn. The deselect/reselect button was not depressed initially, and so the rod withdrawal limit was not re-established as required procedurally. When the operator recognized that the control rod was withdrawn two notches without an expected alarm or rod block function, the operator immediately stopped withdrawal. The rod settled at three notches withdrawn, one notch beyond the desired limit, but within bounded, analyzed limits.

The event was initially reported to the NRC via telephone per ENF # 36397 on November 4, 1999. Subsequent analysis determined that the rod withdrawal limiter was operable at the time of the rod withdrawal, and that no condition prohibited by plant Technical Specifications existed. A retraction was issued on November 30, 1999. This event is submitted as a voluntary report in accordance with guidance in NUREG 1022, Revision 1 as an event having generic interest to the industry.

The root cause of this event was determined to be man-machine interface difficulties in operating the rod withdrawal limiter function. Re-initializing the rod withdrawal limiter following any rod insertion was determined to be a conservative operating practice, as it is currently controlled in plant instructions. Roles and responsibilities during rod movement were re-established, and the balance of rod testing was completed successfully. Procedural controls for the rod withdrawal limiter will be enhanced to minimize future errors.

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

I. INTRODUCTION

The Rod Control and Instrumentation System(RC&IS) [JD] provides critical information to the Control Room staff for the safe operation of the reactor. One of these functions is the Rod Withdrawal Limiter (RWL). The RWL serves to prevent excessive Control Rod (CR) withdrawal. Excessive CR withdrawal can lead to an insertion of excessive positive reactivity to the core, which could challenge the Minimum Critical Power Ratio (MCPR) Safety Limit, and could also challenge fuel integrity.

For each CR selected, depressing the deselect/reselect button on the main control board after CR insertion reinitializes the RWL. This permits only limited CR withdrawal, generating an alarm in the Control Room, and also a rod block signal. Per the testing instruction, the CR selected was to be driven in completely, and then the deselect/reselect button was to be depressed to re-set the RWL. This operation is noted in the introductory portion (Precautions and Limitations section) of the operating instruction, and is also placarded on the reactor controls panel as an approved operator aid.

Two different RWL alarms are part of the system operation. Depending on reactor power level, either two-notch or four-notch withdrawal limits are enforced by the RWL. As a conservative administrative control, depressing the deselect/reselect button is required following any rod insertion.

At the time of the event, PNPP was in Mode 1 at approximately 100 percent rated thermal power. The reactor vessel was at approximately 1024 pounds per square inch gauge, with the reactor coolant at saturated conditions. There were no inoperable systems, structures or components that contributed to this condition.

II. EVENT DESCRIPTION

At approximately 0100 hours on October 31, 1999, operations personnel were involved in routine CR speed and drift calibration testing following the completion of replacement of the Directional Control Valves(DCV's) in the Hydraulic Control Units (HCU's) controlling CR movement. A total of four CR's were to be timed, with CR 38-35 being the first. The CR speed time testing involves a full-length insertion and withdrawal, so that the stroke time may be calculated. In accordance with procedures, CR movement is conducted with an independent verifier assisting the operator actually performing the CR movement. Additionally, any CR movement is accomplished with supervisory oversight provided by Senior Reactor Operators and a qualified Reactor Engineer present in the Control Room.

The control room staff reviewed the appropriate procedures and controls prior to beginning the evolution. All CR movement was to be conducted in accordance with plant operating instructions, and oversight was to be provided by the Unit Supervisor, a licensed Senior Reactor Operator. Plant procedures control how CR movement is to be conducted, including the re-initialization of the RWL after all rod insertions.

As CR stroke time testing progressed, CR 38-35 had been fully inserted at one point. There was a brief discussion between Control Room staff and Reactor Engineering personnel which confirmed that testing for this CR was satisfactory, and that no further time testing was required. The deselect/reselect button was not depressed, and the operator began to withdraw the CR from full insertion. When the expected alarm and rod block were not received at 2 notches withdrawal (position 04), the operator immediately removed his finger from the withdrawal button to stop CR withdrawal. The CR settled at the three-notches-withdrawn (06) position, which was one notch beyond the expected position, but within bounded, analyzed limits.

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

The Control Room staff immediately suspended further testing and reviewed the incident. Prior to recommencing CR stroke time testing, roles and responsibilities were clarified. A Condition Report from the system engineer was submitted after review of the data several days later, when it was suspected that the RWL had been inoperable. The event was reported via telephone to the NRC (ENF#36397) on November 4, 1999. An operability determination was subsequently submitted on November 5, 1999, which stated that the RWL was inoperable during the withdrawal in question, since the deselect/reselect function had not been re-initialized. This determination was based on the RWL function as it was understood at the time. The investigation of this event discovered that there was a general misunderstanding of the RWL function and the role it played in the rod withdrawal error (RWE) accident analysis.

Further review of the RWL function by Reactor Engineering personnel indicated that the RWL was operable at the time of the CR withdrawal, and that a condition prohibited by Technical Specifications had not existed. A revised operability determination was submitted to the Control Room staff on November 29,1999. The revised operability determination incorporated information provided by General Electric(GE) via both teleconferences and reviews of historical documents from initial licensing and plant start-up.

Further review indicated that there was no reportability under 10 CFR 50.72, and a retraction was issued on November 30, 1999. This event is submitted as a voluntary report in accordance with guidance in NUREG 1022, Revision 1 as an event having generic interest to the industry.

III. CAUSE OF EVENT

The root cause of the event was identified as man-machine interface difficulties. In a 1988 design review from GE (DTS-8806, dated August 8, 1988) to the site, GE acknowledged that the RWL operated as designed, but that it could be "compromised by multiple errors." The GE correspondence went on to state that "the effectiveness of the Rod Withdrawal Limiter is dependent to a degree upon the adequacy of Utility administrative procedures associated with Core Management." Administrative controls were put into place to address this item. The design of the RWL was not identified as deficient by GE, and was not reportable under Part 21.

Insufficient written procedural guidance and a lack of clarity in the scope of responsibilities for the independent verifier were also contributing causes.

The cautions for deselecting/reselecting a rod were maintained in the precautions and limitations section of the speed time testing instruction, rather than in the procedural text, and so procedural deficiency was also a contributor.

IV. SAFETY ANALYSIS

The RWL is designed is to prevent exceeding the Minimum Critical Power Ratio (MCPR) Safety Limit (SL) and the cladding one-percent plastic strain fuel design limit that could result from a single CR withdrawal error (RWE) event. The RWL imposes a two-notch-limit at greater than 70 percent power, and a 4-notch-limit at power greater than 20 percent and less than 70 percent.

The initial operability determination had concluded that the RWL was inoperable without the action of depressing the deselect/reselect button to enable it. The revised operability determination considers additional analysis on RWL operating issues that were previously addressed with GE, the Nuclear Steam Supply System (NSSS) vendor. The GE correspondence states that a CR may be moved to any position within the existing RWE analysis by continuous withdrawal, as long as core conditions otherwise remain unchanged.

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

Prior to beginning the maintenance activities on CR 38-35, the position for this rod was fully withdrawn. As part of the maintenance, it was fully inserted until it could be tested to ensure that all CR criteria for stroke time testing were satisfied. During testing, the CR was to be fully withdrawn (position 48) to measure stroke time. Without depressing the deselect/reselect button, the rod withdrawal limit was not re-set from position 48, and would have theoretically only alarmed at 52 notches withdrawn, which is not attainable. Per the guidance above, the CR could be inserted or withdrawn to any position without compromising RWL function, as long as core conditions were constant.

Core conditions were maintained unchanged throughout the CR testing evolution through administrative means. There was no change in the core flow, and there were no movements of the flow control valves. Core temperature remained unchanged. There was no change in feedwater temperature through the addition or removal of feed-water heaters during this time.

The CR in question could not have been withdrawn to a position that was not bounded under current core operating limits. There were no parameters exceeded in the rod withdrawal error accident analysis.

Therefore, there was no safety significance associated with this event.

V. CORRECTIVE ACTIONS

The following corrective actions were instituted by the site :

1) Revisions are proposed for existing procedures to more clearly direct the operation of the RWL.
2) This event will be reviewed as a lesson learned in operator requalification to ensure that RWL operation is understood.
3) Expectations for Control room staff were re-emphasized for all shift crews.
4) Timeliness of submittals for Condition Reports was addressed under a separate Condition Report, which was initiated and completed under senior plant management direction.
5) The roles and responsibilities of the Independent Verifier will be clarified for CR movement

VI. PREVIOUS SIMILAR EVENTS

A review of Licensee Event Reports from the past five years at PNPP did not discover any similar events at the site. One similar event was discovered in a 1988 Condition Report, CR 88-0143, which did not result in any regulatory actions taken. Previously mentioned correspondence from General Electric from this same time period (see Section III) acknowledged that the RWL was not configured optimally, as described previously, but that a condition reportable under Part 21 did not exist for the RWL.

No regulatory commitments were identified in this report.

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