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Perry Nuclear Power Plant
Docket No. 50-440
LER 96-007-00

Gentlemen:

Enclosed is Licensee Event Report 96-007-00, Design Modification Program Weakness Results in Missed Surveillance Requirements and Technical Specification Violations.

If you have questions or require additional information, please contact Mr. James D. Kloosterman, Manager - Regulatory Affairs at (216) 280-5833.

Very truly yours,

R. Brandt for L. W. Myers

L. W. Myers
Vice President - Nuclear

CRE:sc

Enclosure: LER 96-007-00

cc: NRC Region III Administrator
NRC Resident Inspector
NRC Project Manager

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NRC FORM 366 (4-95)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

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APPROVED BY OMB NO. 3150-0104
EXPIRES 04/30/98

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TITLE (4)
Design Modification Program Weakness Results in Missed Surveillance Requirements and Technical Specification Violations

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
										05000
08	21	96	96	-- 007 --	00	09	19	96	FACILITY NAME	DOCKET NUMBER 05000

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 or in NRC Form 366A
			20.2203(a)(2)(iv)	50.36(c)(2)		50.73(a)(2)(vii)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Keith E. Jury, Supervisor-Compliance	(216) 280-5594

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

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)				EXPECTED SUBMISSION	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 August 21, 1996, at approximately 1500 hours, it was determined that the omission of isolation logic verification within two Technical Specification (TS) surveillance instructions (SVIs) resulted in a condition prohibited by the plant TS. A complex modification to Leak Detection system instrumentation was installed during the last refueling outage; however, associated SVIs were not properly revised to account for the complete logic system functional and channel calibration TS surveillance requirements (SRs). As a result, the two SVIs were inadequate to demonstrate that the applicable TS SRs were met. Performance of applicable sections of revised SVIs to verify the relay contacts in question was successfully completed in accordance with the 24 hour time period allowed by TS SR 3.0.3. This event is being reported in accordance with 10CFR50.73(a)(2)(i)(B) and the Perry Nuclear Power Plant, Unit No. 1 Facility Operating License, Section 2.F.

The cause of this event is a weakness in the design modification process. The process does not include sufficient detail to ensure critical functional testing and performance requirements are identified for post-modification testing.

The design modification process is being revised to provide detailed requirements for prescribing functional testing and performance requirements.

I. Introduction

On August 21, 1996, at approximately 1500 hours, it was determined that the omission of isolation logic verification within two Technical Specification (TS) surveillance instructions (SVIs) resulted in a condition prohibited by the plant TS. On August 22, at 1305 hours, a 24-hour non-emergency notification No. 30911 was made to the NRC as required by the Perry Nuclear Power Plant, Unit No. 1 Facility Operating License, Section 2.F. This event is being reported in accordance with 10CFR50.73(a)(2)(i)(B) and the Facility Operating License, Section 2.F.

At the time of the event, the plant was in Mode 1 at 100 percent of rated thermal power. The reactor pressure vessel pressure was at approximately 1024 psig with reactor coolant at saturated conditions.

II. Event Description

In February and March 1996, during Refueling Outage 5, Design Change Package (DCP) 87-725 was installed in the plant to replace selected Riley Temperature Modules with Nuclear Measurement Analysis and Control (NUMAC) Leak Detection Monitors (LDMs) in the Leak Detection system (E31). As part of the design change process, the associated TS SVIs were revised to accommodate testing of the modified instrument channels. SVI-E31-T0086-A, "NUMAC LDM Calibration for 1E31-N700A," and SVI-E31-T0086-B, "NUMAC LDM Calibration for 1E31-N700B," were utilized for the post-modification testing of the work orders that installed the NUMAC LDMs.

Performance of these SVIs is intended to fully meet the channel calibration TS Surveillance Requirements (SR) 3.3.6.1.4, and partially meet the logic system functional test TS SR 3.3.6.1.5, for the Primary Containment and Drywell Isolation Instrumentation channels listed in TS SR Table 3.3.6.1-1. The specific items affected are 1.e., 3.e., 3.f., 3.g., 3.h., 4.c., 4.d., 4.e., 4.f., 4.g., 4.h., 4.i., and 5.a. SVI-E31-T0086-A verifies calibration for LDM 1E31-N700A and verifies operability for relay output unit 1E31-N702A. SVI-E31-T0086-B verifies calibration for LDM 1E31-N700B and verifies operability for relay output unit 1E31-N702B. SVI-E31-T0086-A was completed February 25, 1996, and SVI-E31-T0086-B was completed March 10, 1996. Both SVIs are required to be performed on an 18-month frequency.

On August 20, 1996, at 1500 hours, during surveillance/drawing review activities for SVI-E31-T0086-A and B, it was discovered that the following relay contacts had been omitted from testing for the Reactor Core Isolation Cooling (RCIC)[BN], Reactor Water Cleanup (RWCU)[CE], and Residual Heat Removal (RHR)[BO] systems' respective isolation logic. The specific contacts and their system function(s) are delineated below.

Relay 1E31-N702A-K5 Contacts 5 and 9	Division 1	RCIC Isolation Logic
Relay 1E31-N702B-K5 Contacts 5 and 9	Division 2	RCIC Isolation Logic

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

Relay 1E31-N702A-K6 Contacts 5 and 9	Division 1	RWCU Isolation Logic
Relay 1E31-N702B-K6 Contacts 5 and 9	Division 2	RWCU Isolation Logic
Relay 1E31-N702A-K7 Contacts 5 and 9	Division 1	RHR Isolation Logic
Relay 1E31-N702B-K7 Contacts 5 and 9	Division 2	RHR Isolation Logic

As a result of the deficiencies identified with the surveillance tests, at 1630 hours on August 20, 1996, the provisions of TS SR 3.0.3 were invoked for TS Limiting Condition for Operations (LCOs) 3.3.6.1 Primary Containment and Drywell Isolation Instrumentation, Table 3.3.6.1-1 Items 3.f., 4.i., and 5.a. SR 3.0.3 specifies that if it is discovered that a surveillance was not performed within its specified frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified frequency, whichever is less. This delay period is permitted to allow performance of the surveillance. For the case of this situation, the surveillance requirements not met for the instrument channels were SR 3.3.6.1.4 (Channel Calibration) and SR 3.3.6.1.5 (Logic System Functional Test), which both have an 18 month frequency. The 24 hour time provision was invoked due to it being less than the frequency of the missed SRs.

SVIs-E31-T0086-A and B were revised to include the checking of the omitted contacts. Performance of applicable sections of the SVIs to verify the relay contacts in question was successfully completed on August 21, 1996, at 1241 hours. At 1500 hours, review of the issue determined that, due to the length of time between installation of the NUMAC modification and the discovery of the omission of TS SR verifications, the required actions were not taken in accordance with TS LCO 3.3.6.1. completion times, which is a condition prohibited by the plant's technical specifications and reportable as required by 10CFR50.73(a)(2)(i)(B) and the Facility Operating License, Section 2.F.

III. Cause of Event

The cause of this event is a weakness in the design modification process, in that, the modification procedure does not clarify post-modification testing requirements. Nuclear Engineering Instruction (NEI)-0373, "Initiating, Developing, and Processing Design Modifications," describes the initiation and review of design modifications; however, it does not include sufficient detail to ensure critical function, design, functional testing, and performance requirements are identified in the modification packages. There is no specific direction to confirm existing testing requirements for components or functions presently tested. Additionally, there is no direction or requirement to describe, in detail, how a new design is to be tested, nor is there any required comparison from the existing design to the new design, to ensure that such new designs are verified to meet their specified (e.g., Technical Specification, In-Service Inspection, Motor-Operated Valve program, etc.) requirements.

DCP 87-725 replaced 52 Riley Temperature Modules (which include actuation logic relays) with two NUMAC LDMs (each with an associated relay output unit). This modification resulted in reconfiguration of the isolation actuation logic and in the reduction in the number of channel functional and channel calibration instructions from 48 to 4. The technical preparation of SVI-E31-T0086-A and B was incomplete and inadequate; the magnitude of the changes adversely impacted the thorough preparation and self-checking of the preparer. During the development of the SVIs in response to DCP 87-725, a structured process to verify testing methodology and completeness was not in place. Without such a structured process, the in-depth technical review was also ineffective in detecting the omissions. The omissions were not discovered until a non-proceduralized review process was performed to identify which respective SVI verified proper operation of the contacts shown on the electrical elementary drawings.

A contributing factor is that the non-proceduralized method utilized to ensure the adequacy of surveillance testing was not performed in a timely manner. This method prescribes review of the test methodology, by the Instrumentation and Controls (I&C) procedure writers, against the electrical elementary/logic drawings to ensure the appropriate circuits and components are tested; however, there is not a time requirement in which the review must be performed. Had this review been performed prior to approval of the SVIs, the omissions could have been detected and corrected prior to the post-modification performance of the SVIs.

IV. Safety Analysis

Primary containment isolation instrumentation automatically initiates closure of primary containment isolation valves (PCIVs). The function of PCIVs, in combination with other accident mitigation systems, is to limit fission product release during and following postulated Design Basis Accidents (DBAs). Primary containment isolation within the time limits specified for those isolation valves designed to close automatically, ensures that the release of radioactive material to the environment is consistent with the assumptions used in the analyses for a DBA.

The isolation instrumentation includes the sensors, relays, and switches that are necessary to cause initiation of primary containment and reactor coolant pressure boundary isolation. The channels include electronic equipment (e.g., trip units) that compares the measured input signals with pre-established setpoints. When the setpoint is exceeded, the channel output relay actuates, which then outputs a primary containment isolation signal to the isolation logic.

The RCIC System Isolation, Main Steam Line Pipe Tunnel Temperature-High Instrumentation (TS Table 3.3.6.1-1 Item 3.f.) provides containment isolation signals to Division 1 (i.e., outboard) and Division 2 (i.e., inboard) RCIC System isolation logic when ambient temperature in the Main Steam Line Pipe Tunnel exceeds approximately 152 degrees Fahrenheit (F). Additionally, the late verification of the associated relay contacts resulted in not meeting the

Operational Requirements Manual (ORM) channel calibration and logic system functional test requirements 6.2.16.3 and 6.2.16.4 for Isolation Actuation Instrumentation Operational Requirement Table 6.2.16-1.3 RCIC System Isolation, Item a. Main Steam Line Tunnel Differential Temperature - High. These instrumentation channels utilize the same relay contacts to provide containment isolation signals when differential temperature between two locations in the Main Steam Line Pipe Tunnel exceeds approximately 103 degrees F.

The RWCU System Isolation, Main Steam Line Pipe Tunnel Temperature-High Instrumentation (TS Table 3.3.6.1-1 Item 4.i.) provides containment isolation signals to Division 1 and Division 2 RWCU System isolation logic when ambient temperature in the Main Steam Line Pipe Tunnel exceeds approximately 152 degrees F. Additionally, the late verification of the associated relay contacts resulted in not meeting ORM channel calibration and logic system functional test requirements 6.2.16.3 and 6.2.16.4 for Isolation Actuation Instrumentation Operational Requirement Table 6.2.16-1.2 RWCU System Isolation, Item b. Main Steam Line Tunnel Differential Temperature - High. These instrumentation channels utilize the same relay contacts to provide containment isolation signals when differential temperature between two locations in the Main Steam Line Pipe Tunnel exceeds approximately 103 degrees F.

The RHR System Isolation, RHR Equipment Area Ambient Temperature-High Instrumentation (TS Table 3.3.6.1-1 Item 5.a.) provides isolation signals to Division 1 and Division 2 RHR system isolation logic when ambient temperature in RHR equipment areas exceeds approximately 153 degrees F. Additionally, the late verification of the associated relay contacts resulted in not meeting ORM channel calibration and logic system functional test requirements 6.2.16.3 and 6.2.16.4 for Isolation Actuation Instrumentation Operational Requirement Table 6.2.16-1.4 RHR System Isolation, Item a. RHR Equipment Room Differential Temperature - High. These instrumentation channels utilize the same relay contacts to provide containment isolation signals when differential temperature between two locations in RHR equipment areas exceeds approximately 49 degrees F.

Although the specific relay contacts were not checked before the NUMAC LDMs were placed into service, other contacts on the relays were verified as part of the associated post-modification testing. The relay contacts in question were verified to successfully perform their intended function when tested on August 21, 1996, and it can be assumed that these contacts would have performed successfully, if required, between the time the equipment was placed into service and the time the contacts were verified. Therefore, this event is not considered to be safety significant.

V. Similar Events

LERs 93-016-01, 93-017, 94-016, 94-022, and 96-003 address events in which TS SRs were not performed within time requirements. LERs 93-016-01 and 94-022 documented events in which programmatic controls for scheduling performance of SVIs were not effective in meeting time requirements for ventilation system charcoal sampling, and for testing diesel generators utilizing a staggered test basis, respectively. LER 93-017 documented an event in which containment penetrations were not properly tested in accordance with 10CFR50 Appendix J requirements, which resulted in not meeting containment leak rate TS SRs. LERs 94-016 and 96-003 documented events in which TS SRs were not performed within time requirements due to personnel errors. None of the corrective actions associated with these five LERs could reasonably be expected to have prevented the issue documented by this LER from occurring, since the root cause for this issue primarily involved weaknesses in the modification process.

LER 94-011-01 documented an event in which a combination of design program weaknesses and failure to follow procedure resulted in an inadvertent Engineered Safety Feature actuation when the modified equipment was returned to service. Corrective actions included extensive revision to the design change program and development of a post-maintenance test manual to provide a consistent set of requirements for post-maintenance/modification testing of equipment and components. However, these actions did not prevent the occurrence of the event documented by LER 96-007. The modification discussed in LER 94-011-01 involved replacing a non-regulating transformer with a regulating type. Part of the problem perceived at the time was that test requirements were not consistent. Even though there were other regulating type transformers in the plant, the test requirements were not reflected in a consistent manner. Efforts were focused on development of a post-maintenance test manual to provide consistent test requirements for like components. The revision to the design change program at that time, provided additional guidance for functional testing and performance requirements; however,the guidance did not provide sufficient detail to ensure that specified requirements adequately tested the functional operation and performance intended by the design.

VI. Corrective Actions

The following corrective actions have been taken or are in progress:

1. SVIs-E31-T0086-A and B were revised to include the proper contact verifications; the appropriate portions of the SVIs were subsequently performed to meet TS surveillance requirements.

2. An extent of condition review was performed for other recent design changes to the plant to verify the completeness of design change implementation with regard to TS surveillance requirements and testing. No other conditions similar to the ones reported under this LER were identified.

3. An engineering review determined that no complex modifications similar to DCP 87-725 are scheduled for installation prior to the next refueling outage. This action provides additional confidence that the situation should not recur prior to implementation of future corrective actions.

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4. The I&C surveillance writers have been counseled on this event and are aware of the importance of timely verification of surveillance testing adequacy against the applicable elementary drawings to ensure the proper fulfillment of SRs.

5. The Design Engineering section will discuss expectations for post-modification test requirements at a section meeting scheduled to be held by October 4, 1996.

6. Procedure NEI-0373 is being revised to provide additional detailed requirements for prescribing functional testing and performance requirements for design modifications. These detailed requirements will include at a minimum, a detailed explanation of:

   - current testing and performance requirements,

   - components tested and by what existing documents present testing is performed,

   - the new design and identification of any new tests or required changes to tests.

7. A Maintenance Administrative Instruction is being developed to formalize the non-proceduralized method of verifying surveillance test methodology against the electrical elementary/logic drawings to ensure the appropriate circuits and components are tested. This instruction will require that the verification be performed prior to incorporating any SVI changes created due to design modifications.

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

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The following table identifies those actions which are considered to be regulatory commitments. Any other actions discussed in this document represent intended or planned actions, are described for the NRC's information, and are not regulatory commitments. Please notify the Manager-Regulatory Affairs at the Perry Nuclear Power Plant of any questions regarding this document or any associated regulatory commitments.

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Commitments

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1. The Design Engineering section will discuss expectations for post-modification test requirements at a section meeting scheduled to be held by October 4, 1996.

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2. NEI-0373 is being revised to provide additional detailed requirements for prescribing functional testing and performance requirements for design modifications. The procedure revision is scheduled to be completed by January 15, 1997.

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3. A Maintenance Administrative Instruction is being developed to formalize the non-proceduralized method of verifying surveillance test methodology against the electrical elementary/logic drawings to ensure the appropriate circuits and components are tested. This instruction will require that the verification be performed prior to incorporating any SVI changes created due to design modifications. The instruction is scheduled to be developed by January 15, 1997.

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