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United States Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

Perry Nuclear Power Plant
Docket No. 50-440
LER 93-021-01

Ladies and Gentlemen:

Enclosed is Revision 1 to Licensee Event Report (LER) 93-021, Loss of Safety Function For Emergency Closed Cooling System A. This revision provides a more detailed assessment of the safety consequences and implications of the event as required by 10CFR50.73 and satisfies a commitment made in a reply to a Notice of Violation contained in NRC Inspection Report 50-440/96-08. This LER is revised in its entirety.

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

Very truly yours,

John P. Stetz
Senior Vice President - Nuclear

Enclosure: LER 93-021-01

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

NRC FORM 366 (4-95) U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

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

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Perry Nuclear Power Plant, Unit 1	05000440	1 OF 7

TITLE (4)

Improper Setting of Motor-Operated Valve Results in Loss of Emergency Closed Cooling System Safety Function and Condition Prohibited by Technical Specifications

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
12	23	93	93	-- 021 --	01	12	20	96	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9): 1

POWER LEVEL (10): 100

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)
	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)		OTHER
	20.2203(a)(2)(iii)		50.36(c)(1)	X	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 R. 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 December 23, 1993, it was identified that a condition existed from March 19 to July 2, 1993, which resulted in operation prohibited by Technical Specifications and which could have prevented the fulfillment of the safety function of both loops of the Emergency Closed Cooling (ECC) system under postulated accident conditions. This situation occurred due to excessive leakage through the ECC to Nuclear Closed Cooling (NCC) system isolation valve 0P42-F295A. This valve leakage, coincident with an inoperability of the Division 2 diesel generator during Loss Of Offsite Power (LOOP) conditions, could have caused loss of the ECC loop A cooling water and the availability of ECC loop B, thus preventing the system from fulfilling its safety function. It was also identified that a similar condition existed in 1989, which also could have resulted in the ECC system not performing its safety function.

The excessive leakage through 0P42-F295A was caused by a combination of personnel error and inadequate procedural direction for setting motor-operated valve limit switches and mechanical stops. The limit switches and mechanical stops for valve 0P42-F295A were subsequently readjusted on July 2, 1993. The procedure for adjusting Limitorque limit/torque switches was revised to provide the requisite level of direction for setting limit and torque switches, and adjusting mechanical stops. Training on this issue was conducted for appropriate maintenance and engineering personnel with emphasis on the necessity of verifying proper butterfly valve closure.

These conditions are being reported in accordance with 10CFR50.73(a)(2)(i)(B) and 10CFR50.73(a)(2)(v)(D).

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

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

I. Introduction

On December 23, 1993, it was identified that a condition existed from March 19 to July 2, 1993, which resulted in operation prohibited by Technical Specifications (TS) and which could have prevented the fulfillment of the safety function of both loops of the Emergency Closed Cooling (ECC) system [CC] under postulated accident conditions. This situation resulted from excessive leakage through the ECC to Nuclear Closed Cooling (NCC) system isolation valve [ISV] 0P42-F295A. This valve leakage, coincident with an inoperability of the Division 2 diesel generator during Loss Of Offsite Power (LOOP) conditions, could have caused loss of the ECC loop A cooling water and the availability of ECC loop B, thus preventing the system from fulfilling its safety function. It was also identified that a similar condition existed in 1989, which also could have resulted in the ECC system not performing its safety function.

The 1993 condition was documented on a Condition Report on December 30, 1993, and an event notification was made (Event Notification Number 26551) to the NRC as required by 10CFR50.72(b)(2)(iii)(D). At the time of this operability review, the plant was in Mode 1 at 100 percent of rated thermal power. The reactor pressure vessel pressure was at approximately 1015 psig with reactor coolant at saturated conditions. With respect to the 1989 scenario, the impact on ECC system operability was not recognized and as such, was not reported. These events are being reported under the requirements of 10CFR50.73(a)(2)(i)(B) and 10CFR50.73(a)(2)(v)(D).

II. Event Description

On July 1, 1993, during testing to verify ECC loop A heat exchanger performance, the ECC to NCC system isolation valve 0P42-F295A was identified as potentially leaking; a special test instruction was written to determine the leakage rate and path. Leakage past 0P42-F295A was determined to be approximately 250 gallons per minute. The valve limit switches and mechanical stops were adjusted in accordance with the applicable maintenance procedure to rectify the cause of the leak. The special test instruction was utilized to verify that 0P42-F295A was not leaking after the completion of these adjustments. A Condition Report was written to investigate the root cause of the leakage and to determine past operability.

During the Condition Report investigation, it was determined that limit switch and mechanical stop adjustment had been performed on 0P42-F295A on March 19, 1993, in accordance with the appropriate maintenance procedure applicable at that time. Valve retest consisted of only valve stroking and position verification. A post-maintenance leakage test was not programmatically required nor performed at that time since the valve was classified as an American Society of Mechanical Engineers (ASME) Code, Section XI, Category "B" valve and as such, did not have specific leakage criteria assigned. No subsequent work was performed on 0P42-F295A until leakage identification on July 1, 1993.

The operability review determined that allowable ECC loop A leakage was 3.3 gallons per minute with the actual leakage past 0P42-F295A exceeding that amount. As a result, a condition existed

from March 19 to July 2, 1993, such that ECC loop A could not perform its safety functions without reliance upon Division 2 components/systems being operable. Additionally, under specific postulated accident conditions (i.e., a LOOP concurrent with the Division 2 diesel generator inoperability), both ECC loops could not perform their safety functions. It was also identified that from June 14, 1993, at 0313 hours, until June 15, 1993, at 2305 hours, the Division 2 diesel generator was inoperable for maintenance. During this time period, which lasted approximately 44 hours, a LOOP alone would have resulted in both ECC loops not being able to perform their safety functions. Control room personnel were informed of this determination on December 30, 1993; the required NRC notifications were initiated at that time.

On January 17, 1989, at 1950 hours, the ECC loop B surge tank overflowed while attempting to shift the cooling water supply going to Control Complex chiller "B" from the NCC system to ECC loop B. The cause of the event was the incorrect setting of the limit switches for ECC to NCC system isolation valves 0P42-F295B and 0P42-F325B, which is believed to have occurred during preventive maintenance activities performed on December 12, 1988. These valves were incorrectly set from December 12, 1988 to January 27, 1989, and December 12, 1988 to January 28, 1989, for 0P42-F295B and 0P42-F325B, respectively. During that time, it was not recognized that the incorrect limit switch settings could adversely affect system operability. A leakage rate was not measured for the valves. As a result, during this time period, ECC loop B could not perform its safety function during a LOOP with the concurrent inoperability of the Division 1 diesel generator. From January 10, 1989, to January 12, 1989, the Division 1 diesel generator was taken out of service for scheduled maintenance. During this period of time, which lasted approximately two days, a LOOP alone would have resulted in both ECC loops not being able to perform their safety functions.

III. Cause of Event

The 1993 ECC loop A inoperability resulted from excessive leakage through valve 0P42-F295A. This leakage was caused by a combination of personnel error and inadequate procedural direction for setting motor-operated valve limit switches and mechanical stops. Contributing to this event was the lack of positive means to visually verify that the valve was fully seated in the closed position once the limit switches and mechanical stops had been set. The applicable maintenance procedure, Generic Electrical Instruction (GEI)-0014, "Limitorque Limit/Torque Switch Adjustment," discussed the need to manually shut the valve and determine increased resistance when the valve disk reaches the seat; however, there was no requirement to verify that leakage did not exist nor did the procedure contain specific leakage acceptance criteria. As the limit switches and mechanical stops normally do not exhibit drift once they are set, 0P42-F295A was considered to be leaking excessively from March 19 to July 2, 1993.

The causes for the 1989 issue (ECC loop B) parallel those identified during the evaluation of the 1993 scenario (e.g., limit switches were inappropriately set).

IV. Safety Analysis

The ECC system provides a heat sink for the removal of process and operating heat from safety related components during accident or transient conditions. During normal operation, the ECC system is normally maintained in a standby condition and operated as needed to support surveillance testing of safety related components. The ECC system is also required to operate when the plant is in hot standby condition, and during a normal shutdown period to supply cooling water to the Residual Heat Removal (RHR) pump seals and various room coolers. The ECC system serves as a barrier to the release of radioactive byproducts between potentially radioactive systems and the Emergency Service Water system, and thus to the environment.

The ECC system consists of two independent and redundant loops, that provide cooling water to safety related equipment. Each ECC loop consists of a pump, surge tank, heat exchanger, piping, valves, instrumentation, and controls. An open surge tank in the system provides alarm functions to ensure sufficient net positive suction head is available. The pump in each loop is automatically started on receipt of an actuation signal.

The principal safety related function of the ECC system is to supply cooling water to the RHR, Low Pressure Core Spray (LPCS), and Reactor Core Isolation Cooling (RCIC) room coolers, RHR pump seals, and cooling water to the hydrogen analyzers. The ECC system also supplies the emergency source of cooling water to the Control Complex chillers.

ECC loop A supplies cooling water to Division 1 safety related systems and components. During accident conditions cooling water supply to the control complex chillers is transferred from the NCC system to the ECC system. Isolation between the NCC system and ECC loop A is provided, in part, by in-series valves 0P42-F295A (powered electrically by Division 1) and 0P42-F290 (powered electrically by Division 2). These valves are open when NCC is supplying cooling water and automatically shut during a transfer to ECC under accident conditions. If the interface between NCC and ECC is not properly isolated when the cooling water supply to the control complex chillers is transferred from the NCC system to the ECC system, a flow path is provided to pump the cooling water of ECC loop A into the NCC system, eventually depleting the water inventory from ECC loop A surge tank and disabling the system, if no operator action is taken to restore the water inventory in the surge tank.

In the event of a Design Basis Accident (DBA), one ECC loop is required to provide heat removal capability assumed in the safety analysis for the systems to which it supplies cooling water. To ensure this requirement is met, two ECC loops must be operable. At least one ECC loop will operate assuming the worst single active failure occurs coincident with the loss of offsite power. An ECC loop is considered operable when the associated pump and surge tank are operable; and the associated piping, valves, heat exchanger, and instrumentation and controls required to perform the safety related function are operable. Because valve 0P42-F295A was considered to be inoperable

from March 19 to July 2, 1993, ECC loop A was inoperable for that period of time. ECC loop A inoperability was not recognized during that period; therefore the following TS Limiting Conditions for Operation (LCO) Actions were not taken as required:

TS LCO 3.7.1.2 Emergency Closed Cooling Water System

TS LCO 3.7.2 Control Room Emergency Recirculation System

TS LCO 3.7.3 Reactor Core Isolation Cooling System

TS LCO 3.3.7.5-1 Item 9 Accident Monitoring Instrumentation, Primary Containment and Drywell Hydrogen Concentration Analyzer and Monitor

TS LCO 3.5.1 ECCS - Operating

TS LCO 3.6.3.2 Containment Spray

TS LCO 3.6.3.3 Suppression Pool Cooling

TS LCO 3.8.1.1 A.C. Sources - Operating

TS LCO 3.0.3 Applicability

TS LCO 3.0.4 Applicability

The design basis of the ECC system is to support the Emergency Core Cooling systems and other safety related equipment following an accident. The ECC system is designed to perform its function with a single failure of any active component, assuming a loss of offsite power. During most of the period in which ECC loop A was considered to be inoperable, the Division 2 diesel generator was operable to supply electrical power for ECC loop B in the event of a loss of offsite power. As long as Division 2 electrical power was available, valve 0P42-F290 would have closed, isolating the flow path that would have allowed ECC loop A to pump its cooling water into the NCC system. However, during a LOOP with a loss of Division 2 diesel generator, this would not have been the case.

From June 14, 1993, at 0313 hours, until June 15, 1993, at 2305 hours (a period of approximately 44 hours), the Division 2 diesel generator was taken out of service as part of scheduled maintenance activities. During this period of Division 2 diesel generator inoperability, coincident with valve 0P42-F295A inoperability, 0P42-F290 did not have a source of emergency electrical power. This condition would have prevented ECC loop A from fulfilling its safety function during DBA conditions. 0P42-

F290 would fail to close due to loss of power providing a flow path from ECC loop A into the NCC system. Within 1 minute ECC A could lose its water inventory to a point where the pump would cavitate on low net positive suction head and eventually fail. Additionally, since emergency electrical power was not available for ECC loop B operation, loop B could not perform its safety function. Therefore, the safety function for the ECC system was lost for the approximately 44 hour period this configuration existed. Based on the loss of ECC system safety function, this event is considered to have safety significance.

With respect to the 1989 scenario, the safety analysis conclusions and applicable TS LCO Actions that were not taken closely parallel those discussed above for the 1993 event.

V. Similar Events

A Condition Report was written in 1991 to address the setting of limit switches and mechanical stops for all safety related butterfly valves. Evaluation determined a root cause that the limit switches and mechanical stops were not properly set. It was determined that for valves that have containment isolation function the periodic local leak rate tests were sufficient to confirm valve/system operability. Corrective actions were to review the work history of each valve and properly set the limit switches and mechanical stops. This corrective action was intended to be addressed for 0P42-F295A by the work completed on March 19, 1993.

LER 96-008, "Degraded Breaker Results In Loss Of Safety Function And Exceeding Technical Specification Action Statements," documented an event in which an electrical circuit breaker, improperly assembled by the manufacturer would trip prematurely under certain plant conditions and resulted in loss of safety functions for associated equipment and failure to meet TS LCO Action requirements for the time the breaker was installed in the plant. The subject breaker was replaced and other applicable breakers were inspected for the same deficiency. This event occurred after the event documented by LER 93-021-01 and no connection could be reasonably made between the causes and corrective actions for the two events.

VI. Corrective Actions

For the 1993 event, the limit switches and mechanical stops for valve 0P42-F295A were readjusted on July 2, 1993. An allowable leakage criteria was determined for the valve and a post-maintenance leak test was performed the same day. The valve successfully met its leakage acceptance criteria and passed the post-maintenance test. The procedure for adjusting Limitorque limit/torque switches, GEI-0014, was revised to provide the requisite level of direction for setting limit and torque switches, and adjusting mechanical stops. The procedure revision also added a post-maintenance test requirement for butterfly valves that have an established seat leakage limit. Training was conducted for appropriate maintenance and engineering personnel on this issue and on the necessity of verifying proper butterfly valve closure.

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An engineering evaluation was performed to determine if other motor operated butterfly valves were affected by the causes that resulted in the inoperability of valve 0P42-F295A; other than the other valves in the ECC system which perform a similar isolation function (which were not leaking but had no leakage criteria established); no other concerns were identified. These valves were re-categorized as ASME Code, Section XI, Category "A" valves on October 8, 1996. As such, they will be periodically leak tested as part of the In-Service Testing Program (ISTP) against specific leakage acceptance criteria. Operator action during a DBA to support the specific leakage acceptance criteria has also been proceduralized. Additionally, the Engineering Department conducted a review of other ASME Code, Section XI, Category "B" valves in the ISTP to determine if their respective categorization needed to be changed; no additional valves requiring re-categorization were identified.

With respect to the 1989 event, the valve limit switches were adjusted and the repetitive tasks for preventive maintenance on selected valves were revised to include checking limit switch settings after preventive maintenance was performed.

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

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.

Commitments
None