CENTERIOR ENERGY

PERRY NUCLEAR POWER PLANT
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Robert A. Stratman
VICE PRESIDENT - NUCLEAR

January 18, 1993
PY-CEI/NRR-1598 L

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

Perry Nuclear Power Plant
Docket No. 50-440
LER 92-005-01

Dear Sir:

Enclosed is Licensee Event Report 92-005-01 for the Perry Nuclear Power Plant.

Sincerely,

Robert A. Stratman

RAS:CRC:ss

Enclosure: LER 92-005-01

cc: NRC Project Manager
NRC Resident Inspector
NRC Region III

210058

Operating Companies
Cleveland Electric Illuminating
Toledo Edison

IE22 1/1

NRC FORM 366 (5-92) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES 5/31/95

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), 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.

LICENSEE EVENT REPORT (LER)

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

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Perry Nuclear Power Plant, Unit 1	05000 440	1 OF 7

TITLE (4)
Local Leak Rate Tests Result in Exceeding Technical Specification Failure Criteria

EVENT DATE (5)			LER NUMBER (6)			REPORT NUMBER (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
										05000
03	26	92	92	– 005 –	01	01	18	93	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9): 5
POWER LEVEL (10): 000

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)

	20.402(b)		20.405(c)		50.73(a)(2)(iv)		73.71(b)
	20.405(a)(1)(i)		50.36(c)(1)	X	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)		50.73(a)(2)(i)		50.73(a)(2)(viii)(A)		
	20.405(a)(1)(iv)	X	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 (Include Area Code)
Charles R. Elberfeld, Compliance Engineer, Ext. 5264	(216) 259-3737

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
C4	LD	ISV	D243	Y	C4	SJ	ISV	R340	Y
C4	SB	33	L200	Y	C2	AA	ISV	B350	Y

SUPPLEMENTAL REPORT EXPECTED (14)

YES (If yes, complete EXPECTED SUBMISSION DATE)	X NO	EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

Between March 26, 1992 and April 21, 1992, during the third refueling outage, four containment penetrations exceeded Local Leak Rate Test (LLRT) failure criteria for leakage, as defined by Technical Specification 3.6.1.2.b., 3.6.1.2.d., and 3.6.1.2.e.

The causes for three of these events were component failures. Leakage between the seat insert and the valve body caused one valve to exceed its LLRT failure criteria. A faulty torque switch prevented the second valve from closing completely. The cause of the third event was attributed to wear of the seating surfaces as part of the normal operating life of two check valves. The cause of the fourth event was attributed to the misapplication of the check valve for that function.

For the first event, engineering personnel are developing a design modification to allow the installation of a replacement valve more suited for the application or replacement of valve internals with alternate parts. Appropriate repairs and parts replacements were performed for all three events that were caused by component failures. A differently designed spring loaded lift check valve with softer seating material was installed to replace the valve in the fourth event.

I. Introduction

Between March 26, 1992 and April 21, 1992, four containment penetrations exceeded Local Leak Rate Test (LLRT) failure criteria for leakage, as defined by Technical Specification 3.6.1.2.b., 3.6.1.2.d., and 3.6.1.2.e.

All of the leakage rates were found during LLRT activities during the third refueling outage, and all appropriate notifications to the NRC Operations Center were made via the Emergency Notification System in accordance with the requirements of 10CFR50.72(b)(2)(i) and 10CFR50.72(b)(2)(iii). These events are being reported under the requirements of 10CFR50.73(a)(2)(ii) and 10CFR50.73(a)(2)(v).

II. Description of Events/Cause Analysis/Corrective Actons to Prevent Recurrence

Penetration P306

Description of Event

On March 26, 1992, at 0630, during the performance of Surveillance Instruction (SVI-P52-T9306) "Type C Local Leak Rate Test Of P52 Instrument Air Penetration P306," the Instrument Air [LD] Supply Header Containment Inboard Check Valve [ISV] (1P52-F0550) was determined to have caused the total combined Secondary Containment bypass leakage rate to exceed the Technical Specification 3.6.1.2.d. limit of 0.0504 La for Secondary Containment bypass leakage paths. This valve is manufactured by Dresser Industries, Model 5580W-1-F316. The leak rate for the valve was determined to be 3500 standard cubic centimeters per minute (sccm), which caused the total combined Secondary Containment bypass leakage to increase to 8,094.55 sccm. The 0.0504 La Technical Specification limit equates to 5,051.74 sccm for all penetrations bypassing Secondary Containment. The valve was reworked and successfully tested in accordance with SVI-P52-T9306 on March 30, 1992 with a penetration leakage rate of 476 sccm.

Cause of Event

The cause of the Instrument Air Supply Header Containment Inboard Check Valve LLRT failure is component failure. The leakage in the valve was identified to be between the seat insert and the body of the valve. The seat insert is threaded into the valve body and torqued in accordance with manufacturer's recommendations. However, leakage developed between the threaded areas of the two pieces.

Corrective Actions

The Instrument Air Supply Header Containment Inboard Check Valve (1P52-F0550) was repaired by replacing its internals (springs, disc, and valve seat insert), carefully lapping the disc and valve seat insert, and using grafoil gasket tape as a sealant between the valve seat insert and the valve body. Engineering personnel have analyzed the acceptability of the valve for the application and are developing a design modification to allow the installation of a replacement valve more suited for the application or replacement of valve internals with alternate parts. This modification is tentatively scheduled to be implemented during the fourth refueling outage following receipt of a replacement valve/component parts.

Penetration P423

Description of Event

On April 5, 1992, at 1630, during the performance of Surveillance Instruction (SVI-B21-T9423) "Type C Local Leak Rate Test Of 1B21 Penetration P423," test personnel were unable to pressurize the test volume to 11.31 psig (Pa). Because Leakage through Penetration P423 could not be quantified at that time, it was determined that the combined leakage rate prescribed by Technical Specification 3.6.1.2.b. of less than or equal to 0.60 La had been exceeded. Additionally, the Secondary Containment Bypass combined leakage rate of less than or equal to 0.0504 La was exceeded.

On April 7, 1992, at 0630, using test equipment which was capable of quantifying higher leakage rates, the leakage through the penetration was quantified as being in excess of 200 standard liters per minute (slm). The 0.60 La Technical Specification Limit equates to 60.140 slm. A faulty torque switch [33] manufactured by Limitorque Corporation, Model Number 10717, prevented the Main Steam [SB] Line Drain and Bypass Outboard Isolation Valve [ISV] (1B21-F0019) from closing completely and this valve was the source of the leakage. The torque switch was replaced and adjusted in accordance with plant procedures, and the penetration was successfully tested in accordance with SVI-B21-T9423 on April 14, 1992 with a leakage rate of 10 sccm.

Cause of Event

The cause of the Main Steam Line Drain and Bypass Outboard Isolation Valve leakage is component failure. Troubleshooting efforts revealed that the valve was not fully seated when closed electrically and the torque switch could not be adjusted to allow the valve to seat properly. The faulty operation of the torque switch caused the failure of the valve to close completely.

Corrective Actions

The closure torque switch on the Main Steam Line Drain and Bypass Isolation Valve was replaced and adjusted in accordance with plant procedures. The failure of this torque switch was considered an anomaly and considered to be an isolated incident.

Penetration P121

Description of Event

On April 8, 1992, at 1430, during the performance of Surveillance Instruction (SVI-N27-T9121), "Type C Local Leak Rate Test Of 1N27 Penetration P121," test personnel could not pressurize the penetration to 1.10 Pa (12.44 psig) as required by the SVI. The Controlled Closure Anti-waterhammer Lift Check Valve [ISV] (1B21-F0032A) was discovered to be leaking at least 7.883 gallons per minute (gpm) and the Reactor Feed [SJ] Check Valve [ISV] (1N27-F0559A) was discovered to be leaking at least 12.39 gpm. When these leakages were added to previously identified leakage, the total of 24.280 gpm exceeded the 23.0 gpm limit required by Technical Specification 3.6.1.2.e. Both valves, manufactured by Rockwell International, Model Number 7592(WCC)JNQTY, were disassembled, inspected, reworked, and cleaned as appropriate prior to reassembly.

The penetration was successfully tested in accordance with SVI-N27-T9121 on May 4, 1992. The leakage rates for the valves at that time were 0.02 gpm for 1N27-F0559A and 0.49 gpm for 1B21-F0032A.

Cause of Event

The cause for the feedwater check valve 1B21-F0032A and 1N27-F0559A failures is attributed to wear of the seating surface of the valves as part of the normal operating life of the valve. Wear on the seal areas of these valves resulted in leakage greater than allowed by the testing requirements.

Corrective Actions

Both valves were disassembled and inspected. On the Controlled Closure Anti-waterhammer Lift Check Valve (1B21-F0032A), the piston was lapped and the other parts and seal ring seating area were cleaned before the valve was reassembled. On the Reactor Feed Check Valve (1N27-F0559A), the piston and valve seats were lapped and the other parts and seal ring seating area were cleaned prior to valve reassembly. Past performance history of these valves, combined with the corrective actions taken to reduce the leakage, is expected to assure that these valves will perform as required throughout the fourth fuel cycle and beyond. Analysis of trending data and test results from previous and fourth refueling outage LLRTs on the check valves for both feedwater lines will be utilized to determine the appropriate frequency for feedwater check valve preventive maintenance.

Penetration P204

Description of Event

On April 21, 1992, during the performance of Surveillance Instruction (SVI-C11-T9204), "Type C Local Leak Rate Test Of 1C11 Penetration P204," leakage was determined to be in excess of 200 slm for the Inboard Containment Isolation Check Valve [ISV] (1C11-F0122). This resulted in the combined leakage rate prescribed by Technical Specification 3.6.1.2.b. of less than or equal to 0.60 La, as well as the Secondary Containment Bypass combined leakage rate of less than or equal to 0.0504 La as prescribed by Technical Specification 3.6.1.2.d. being exceeded. This valve is manufactured by BW/IP International, Model 82530. The 1C11-F0122 valve was replaced with a differently designed spring loaded lift check valve. The valve was successfully tested in accordance with SVI-C11-T9204 on April 30, 1992, with a leakage rate of 10 sccm.

Cause of Event

The 1C11-F0122 check valve has an LLRT failure history dating back to 1986. Following each of the previous failures, the valve was disassembled and the immediate cause was diagnosed as misalignment between the check valve seating surfaces. The failures were considered to be attributed to the misapplication of the check valve which is susceptible to leakage at low pressures due to misalignment of valve internals. This design problem had been previously identified and the valve had been scheduled for replacement with a different type in the third refueling outage.

Corrective Actions

The Inboard Containment Isolation Check Valve (1C11-F0122) was replaced with a differently designed spring loaded lift check valve, manufactured by Edward Valves Inc., Model Number B36278(F316)FT2, which has softer seating material.

III. Analysis of Events

Primary Containment integrity ensures that the release of radioactive material from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analyses. The containment design basis accident leakage rate (La) is 0.2 percent, by weight, of the contained atmosphere in 24 hours at the peak accident pressure of 11.31 psig (Pa).

Technical Specification limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses (La) at Pa. The Technical Specification combined leakage rate limit for all penetrations and all valves subject to Type B and C tests when pressurized to Pa, except for Main Steam Line Isolation valves and valves that are hydrostatically tested, is less than or equal to 0.60 La.

Secondary Containment is designed to collect the Primary Containment leakage during and following a design basis accident, delaying release to the environment until after processing through the Annulus Exhaust Gas Treatment System [VC]. This assures that the resultant doses are less than the values set forth in 10CFR100, for offsite, and less than those set forth in 10CFR50 General Design Criteria 19 for the control room. The analyzed leakage rate from Secondary Containment bypass leakage sources (sources whose leakage bypasses the Containment Annulus Exhaust Gas Treatment System) is 0.0672 La. The Technical Specification maximum allowable combined test leakage rate from potential Secondary Containment bypass leakage sources is 0.0504 La (0.75 times 0.0672 La).

The leakage rates for 1B21-F0019, 1P52-F0550, and 1C11-F0122 caused the combined leakage rate for all penetrations and all valves subject to Type B and C tests, except for Main Steam Line Isolation valves and valves that are hydrostatically tested, to exceed La and, as well, caused the combined leakage rate for all penetrations that are Secondary Containment bypass leakage paths to exceed 0.0672 La. Therefore these events are considered to be safety significant.

1B21-F0032A and 1N27-F0559A identified in this report are not part of the Secondary Containment bypass leakage pathway. These valves are on a feedwater line which is sealed post-LOCA with water from the Feedwater Leakage Control System [SJ] (FWLCS). Feedwater lines are part of the reactor coolant pressure boundary since they penetrate both the containment and drywell and connect to the reactor vessel. Each line includes three isolation valves. The isolation valve inside the drywell (1N27-F0559A) and the first isolation valve outside containment (1B21-F0032A) are both controlled closure anti-waterhammer check valves. The outermost valve (1B21-F0065A) is a motor operated gate valve. Although 1B21-F0032A and 1B21-F0065A are sealed with water post-LOCA by design, the leakage through 1B21-F0032A would have prevented the FWLCS from maintaining the water seal. Under accident conditions, containment air would flow through this pathway and would be only partially mitigated by FWLCS operation. Therefore, leakage through this penetration is considered to be potentially safety significant. Leakage through 1N27-F0559A has no effect on the FWLCS operation and this leakage by itself is not considered to be safety significant.

Previous non-MSIV LLRT failures have been documented by LERs 86-080, 88-004, 89-006-01, and 90-026-01. The 1P52-F0550 valve has previously failed LLRTs on November 10, 1986 and January 16, 1988. The 1B21-F0019, 1B21-F0032A, and

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

1N27-F0559A valves listed in this report do not have a history of LLRT failure. The 1C11-F0122 valve had failed five consecutive initial LLRTs spanning from 1986 to the present and had been previously scheduled to be replaced during the third refueling outage.

IV. Program Improvements

As part of this investigation, the LLRT effort was examined and some opportunities for improvement were noted.

All failures of LLRTs are trended and analyzed in accordance with the established Plant Administrative Procedure (PAP-1120) "Type A, B, and C Leak Rate Test and Accountability Program." Communications between component testing, system engineering, and design engineering efforts are being enhanced by the issuance of a periodic predictive maintenance report to keep affected sections apprised of testing and trending developments. Plant Administrative Procedure (PAP-0905) "Work Order Process" provides guidance for inclusion of steps to keep failed components for failure analysis in work orders. Maintenance Planners, System Engineers, and Component Test Engineers will be retrained to PAP-0905 with emphasis placed on the consideration of this option when developing work orders in response to LLRT failures.

This report is revised in its entirety. No revision bars are included to identify changes from Revision 0 of this report.

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