CENTERIOR ENERGY

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VICE PRESIDENT - NUCLEAR

July 12, 1991
PY-CEI/NRR-1371 L

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

Perry Nuclear Power Plant
Docket No. 50-440
LER 90-026-01

Dear Sir:

Enclosed is Licensee Event Report 90-026-01 for the Perry Nuclear Power Plant.

Sincerely,

Michael D. Lyster

MDL:TSH:njc

Enclosure: LER 90-026-01

cc: NRC Project Manager
NRC Sr. Resident Inspector
NRC Region III

IE22

NRC Form 366 (9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/88

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Perry Nuclear Power Plant, Unit 1	0 5 0 0 0	1 OF 0 6

TITLE (4): Failed Local Leak Rate Results in Exceeding Allowable Primary and Secondary Containment Bypass Leakage

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 NAMES	DOCKET NUMBER(S)
0 9	1 9	9 0	9 0	0 2 6	0 1	0 7	0 3	9 1		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 5

POWER LEVEL (10): 0 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (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)		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	
	AREA CODE	
Henry L. Hegrat, Compliance Engineer, Extension 5185	2 1 6	2 5 9 - 3 7 3 7

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
C2	I P	I S V	T 0 2 0	Y	C4	L F	I S V	F 1 3 0	Y
C2	A A	I S V	B 3 5 0	Y	C 4	C C	I S V	C 6 3 0	Y

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 fifteen single-space typewritten lines) (16)

Between September 19 and November 19, during the second refuel outage, four containment penetrations exceeded Local Leak Rate Test (LLRT) failure criteria for total combined Secondary Containment [NH] bypass leakage rate, as defined by Technical Specification 3.6.1.2. One penetration also exceeded the total combined Primary Containment bypass leakage rate criteria.

The causes of these events were component failures, and design inadequacy. The design deficiencies of a Control Rod Drive Hydraulic system check valve was attributed to susceptibility to leakage due to misalignment problems. Two Post Accident Sample System (PASS) valves were considered to have a design deficiency in that in that the original system design does not provide flow restriction which would eliminate cavitation at the isolation valves, or flow indication which could be used by the panel operator in order to control sampling flow.

Appropriate corrective maintenance resulted in satisfactory LLRT results for all of the penetrations involved. The PASS valves will be administratively controlled to limit the cycling of these valves to once per quarter for PASS training and as required as an alternate sample point for reactor water sampling. The CRDH valve is scheduled to be replaced in the third refuel outage with a piston style, soft seated check valve.

I. Introduction

Between September 19 and November 19, during the second refuel outage, four containment penetrations exceeded Local Leak Rate Test (LLRT) failure criteria for total combined Secondary Containment [NH] bypass leakage rate as defined by Technical Specification 3.6.1.2. One penetration also exceeded the total combined Primary Containment bypass leakage rate criteria. At the time of these events, the plant was in Operational Condition 5 (refuel). These events are detailed in the following report.

II. Description of Events

Post Accident Sampling Valves

On March 15, 1990, chemistry technicians discovered that the Post Accident Samplin System (PASS) [IP] sample line containment isolation valves [ISV] were leaking following sampling operation. At the time of this event the plant was in Operational Condition 1 (Power Operation) at approximately 100 percent power. A Local Leak Rate Test (LLRT) to confirm that the secondary containment bypass leakage was not in excess of Technical Specification requirements would have required separate measurements of air leakage across each of the two isolation valves at a pressure differential of 11.31 psi. Due to the fact that the plant was operating, performance of an LLRT was not feasible. Calculations were performed on March 17 to determine the extent of penetration leakage by relating water leakage through the penetration with both valves shut, to air leakage at a pressure differential of 11.31 psi. The leak rate calculated using this methodology was then added to the known secondary containment bypass leakage rate to verify that the total secondary containment bypass leakage was less than the Technical Specification limit. The decision was made on March 17 to continue operation provided the valves remained closed to avoid any increase leakage possibility. Additional calculations would be required if the valves were stroked with flow through them. The plant continued in Operational Condition 1 (Power Operation) until the second refuel outage began on September 7, 1990.

On September 19, 1990 during the performance of Surveillance Instruction (SVI-P87-T9413) "Type C Local Leak Rate Test of P87 Penetration P413", penetration P413 was determined to have exceeded the Technical Specification 3.6.1.2.d limit of 0.0504 La (5052 standard cubic centimeters per minute (sccm)) for secondary containment bypass leakage paths. The leakage rate was determined to be 4735 sccm for the inboard isolation valve, 1P87F049, and 5870 sccm for the outboard isolation valve, 1P87F055. The inboard isolation valve was replaced with a new valve while the outboard isolation valve was disassembled and rebuilt. The penetration was satisfactorily leak tested on October 7, 1990, with leakage rates of both valves of 10 sccm.

After reviewing the the 1990 LLRT results, the March 17 calculation methodology was re-examined. Because of the different valve configuration during each

evaluation of leakage performance, it was determined that the LLRT results should not be correlated to the calculation method used during the March 17 event. The calculation performed in March was determined to be a valid engineering approach, demonstrating that the operational leakage experienced was less than Technical Specification limitations. It is engineering's evaluation that the calculation method utilized remains appropriate and adequately demonstrated compliance with the Technical Specification.

Other Identified Leakage

On October 19, 1990 during the performance of Surveillance Instruction (SVI-P51-T9308) "Type C Local Leak Rate Test of P51 Penetration P308", an outboard Service Air [LF] containment isolation valve, 1P51F150, was also determined to have exceeded the Technical Specification limit for secondary containment bypass leakage paths. The leakage rate was determined to be approximately 5500 sccm. On November 11, 1990 the inboard Service Air containment isolation valve in the same penetration, 1P51F530, also experienced excessive leakage of 4720 sccm. The valve stroke for 1P51F150 was adjusted and its limit switch reset. The bonnet and disc for 1P51F530 was replaced. On November 23 both valves were successfully retested with leakage rates of 138 sccm for 1P51F150 and 314 sccm for 1P51F530.

On November 15, 1990 during performance of Surveillance Instruction (SVI-C11-T9204) "Type C Local Leak Rate Test of 1C11 Penetration P204", an inboard Control Rod Drive Hydraulic (CRDH) System [AA] containment isolation check valve, 1C11F122, was determined to have exceeded the Technical Specification limits for primary and secondary containment bypass leakage paths. The 3.6.1.2.b Technical Specification limit of 0.6 La equates to 60,140 sccm for all primary containment penetration that are required to be Type B and C tested. The leakage rate for 1C11F122 was determined to be approximately 127,000 sccm. The valve was disassembled, lapped and reassembled and was successfully retested on November 22 with a leakage rate of 56 sccm.

On November 19, 1990 during performance of Surveillance Instruction (SVI-P43-T9311) "Type C Local Leak Rate Test of 1P43 Penetration P311", an outboard Nuclear Closed Cooling (NCC) [CC] containment isolation valve, 1P43F140, was suspected of having exceeded the Technical Specification limit for secondary containment bypass leakage path. The test equipment could not be pressurized; therefore, an accurate leakage rate could not be determined. No repairs to the valve were made, however, after larger test equipment was connected to quantify the leakage and the valve was stroked to clear the test line of water, a leakage rate of 117.5 sccm was obtained. The value was acceptable and no further testing was performed.

III. Cause Analysis

The causes of these events were component failure, and design inadequacy. Details of this analysis follow.

Post Accident Sampling Valves

Both PASS isolation valves were replaced in November, 1986 for failure to meet Technical Specification leakage requirements during LLRT activities. The cause of the excessive leakage could not be determined at the time (LER 86-007). Both PASS valves were also replaced in January 1988 (LER 88-004) following an additional failure to meet Technical Spe .fication requirements. These failures were attributed to the effects of electrical arcing on the seat and disk, due to improper welding during installation as evidenced by blue discoloration on the valve disc surface. Small particles of foreign material were also found inside the valve bodies. Excessive operation of these valves for training was also identified as a contributing factor. The valves were successfully Leak Tested upon replacement in May of 1988 and were also successfully leak tested in the first refuel outage in March of 1989.

The PASS valves, manufactured by Target Rock Corporation, model 83AU-005, are solenoid operated valves and are not used for throttling sample flow. A throttle valve is provided at the sample panel. A light indicates when the minimum flow necessary to obtain a representative sample has been reached; however, indication of flow rate is not available. If the throttling valve is opened excessively, high flow rates through the PASS isolation valves results in cavitation at the valve seating surfaces. During the second refuel outage, inspection of both PASS isolation valves revealed that the seating surfaces were pitted, characteristic of such cavitation. No blue discoloration was noted as was found during the 1988 inspection. Therefore, the cause of the leakage of the PASS valves is considered to be inadequate design, in that the original system design does not provide flow restriction which would eliminate cavitation at the isolation valves, or flow indication which could be used by the panel operator in order to control sampling flow.

Other Containment Isolation Valves

The CRDH isolation check valve has a LLRT failure history dating back to 1986. Following each of four failures, the valve was disassembled and the immediate failure cause was diagnosed as misalignment between the check valve seating surfaces. In each case, after valve maintenance using the vendor manual and successful retest, the system was returned to service. The failures are considered to be attributed to the design of the check valve (stainless steel seating surface, swing check; BW/IP International, model 82530) which is susceptible to leakage due to misalignment problems.

The outboard Service Air isolation valve, 1P51F150, failed its LLRT due to the valve stroke being out of adjustment. A review of the work history since the last successful LLRT did not indicate any work which could have affected the valve travel adjustment. This valve is manufactured by Fisher Controls, model 667-ES. The inboard Service Air isolation valve, 1P51F530, is considered to have failed its LLRT due to rust and debris on the valve seat. This valve is manufactured by BW/IP International Inc., model 81440.

The initial LLRT failure for the NCC isolation valve, 1P43F140, is believed to have been caused by debris on the valve seat, which was removed when the valve was opened to blow dry the test volume. Because the leakage rate from the 1989 LLRT was 107.5 sccm compared to 117.5 sccm during this test, any seat damage was minimal. The test rig was determined not to be the cause of the initial LLRT failure. The 117.5 value was acceptable and the valve was not disassembled to inspect the seat. This valve is manufactured by Contromatics, model 2498-00-26.

VI. Corrective Actions

During this fuel cycle the corrective action is to limit the cycling of these valves to once a quarter for PASS training and as required as an alternate sample point for reactor water sampling. These evolutions are controlled by a management directive and information tags on the valves. Based on the valve performance and the limited valve cycling, the valves currently installed are expected to remain satisfactory for the current fuel cycle. Acceptability of valve design and any long term changes to the PASS panel and piping will be determined following LLRT testing during the third refuel outage.

The CRDH valve seat was lapped and the valve bonnet, disc, clevis and arm were replaced. The valve bonnet was hydrostatic tested and adjusted prior to local leak rate testing. An acoustic expert has analyzed this valve for flow induced chattering and found no evidence of chattering in steady state operation. This valve will be acoustically analyzed on a quarterly basis to monitor its performance. To prevent recurrence, a design change is being implemented to replace the valve with a piston style, soft seated check valve. This design change is scheduled for implementation in the third refuel outage.

The outboard Service Air isolation valve was adjusted and the limit reset. As this valve had not failed an LLRT since 1985, its leakage history does not indicate a design deficiency. The bonnet and disc were replaced in 1P51F530. This valve was also cleaned and the gasket replaced following a higher than acceptable leakage rate in 1989. This was the only previous LLRT failure of this valve. After 1P43F140 was stroked to clear the test line of water, it was satisfactorily leak tested. No repair activities were taken with respect to this valve. This valve does not have a history of leakage problems.

Failure of 1P51F150, 1P51F530 and 1P43F140 resulted in an evaluation of the LLRT tracking method. The current program was deemed to be adequate, with an onsite computer program used to track the LLRT results. The LLRT history is reviewed with regard to repeat failures and those valves which may be susceptible to failure. To enhance this review, the program is being changed to formalize communications to design engineering concerning repeat failures. Contingency maintenance is planned prior to a refuel outage for valves that are considered to have a high failure risk. The LLRT history did not indicate these three valves would experience leakage problems.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Perry Nuclear Power Plant, Unit 1	0 5 0 0 0	9 0	0 2 6	0 1	0 6 OF 0 6

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

V. Safety Analysis

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. Technical Specification limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure of 11.31 psig, Pa. Secondary containment is designed to collect the primary containment leakage during and following a postulated design basis accident, delaying release to the environment until after processing through the Annulus Exhaust Gas Treatment system [VC]. This assures that the resultant offsite doses are less than the values set forth in 10 CFR 100 and 10 CFR 50, General Design Criterion 19. The valves identified in this report are part of the Secondary Containment bypass leakage pathway. Because leakage rates identified during the performance of these LLRT's was in excess of that assumed in the accident analysis, this event is considered to be safety significant.

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