NRC FORM 366 (6-89)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Perry Nuclear Power Plant, Unit 1	0 5 0 0 0 4 4 0	1 OF 0 3

TITLE (4) Failed Local Leak Rate Test Results in Exceeding Allowable 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 0	1 0	1 9	9 0		0 5 0 0 0
										0 5 0 0 0

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

OPERATING MODE (9)	5								
POWER LEVEL (10)	0 0 0	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 6855	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
C4	I P	I S V	T 0 2 0	Y					

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
X YES (If yes, complete EXPECTED SUBMISSION DATE)	NO		0 1	1 5	9 1

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

On September 19, 1990 at approximately 1115 the total combined Secondary Containment bypass leakage rate defined by Technical Specification 3.6.1.2.d was determined to have been exceeded. The major contributors to the total leakage were the Post Accident Sampling System (PASS) Instrument Sample Line containment isolation valves, 1P87-F049 and F055. One of these valves was subsequently rebuilt and the other replaced. They were satisfactorily leak tested on October 7, 1990.

The cause of this event has not been determined. A root cause analysis is being performed on the replaced valve. A supplemental report will be issued to discuss the results of this analysis and to identify corrective actions which will be taken. This supplemental report will also identify major contributors to Secondary Containment bypass leakage.

Pending completion of the root cause analysis, corrective action will be taken to limit the cycling of these valves for Position Indication Testing and PASS training to once per quarter. Flow will only be allowed through the valves semi-annually for PASS sampling and when required as a backup to the normal Reactor Water Sampling panel.

On September 19, 1990 at approximately 1115, the total combined Secondary Containment [NH] bypass leakage rate as defined by Technical Specification 3.6.1.2.d was determined to have been exceeded. At the time of discovery, the plant was in Operational Condition 5 (Refuel) with a planned refueling outage in progress. Reactor Pressure Vessel [RPV] temperature was approximately 83 degrees and reactor pressure was atmospheric.

On September 19, 1990, during the performance of Surveillance Instruction (SVI-P87-T9413) "Type C Local Leak Rate Test of P87 Penetration P413", the inboard and outboard Post Accident Sampling System (PASS) [IP] Instrument Sample Line containment isolation valves [ISV], 1P87-F049 and -F055, were determined to have exceeded the Technical Specification 3.6.1.2.d limit of $0.0504L_a$ for secondary containment bypass leakage paths. These valves are manufactured by Target Rock Corporation, model 83AU-005. The leak rate was determined to be 4735 standard cubic centimeters per minute (sccm) for the inboard isolation valve (F049) and 5872 sccm for the outboard isolation valve (F055). The $0.0504 L_a$ Technical Specification limit equates to 5,051.74 sccm for all penetrations bypassing secondary containment.

The cause of the leakage through these two valves has not been determined. The outboard isolation valve was disassembled and rebuilt. The valve seat showed signs of pitting. The inboard isolation valve was removed and replaced with a new valve. The inboard valve has been saved for root cause analysis. Both valves were satisfactorily leak tested on October 7, 1990.

Both of these valves were replaced in January, 1988 following their failure to meet Technical Specification limits for leakage. The cause of these failures was attributed to the effects of electrical arcing on the seat and disk, probably due to improper welding during installation. Small particles of foreign material were also found inside the valve bodies. Excessive operation of these valves for training was identified as a contributing factor (LER 88-004). These valves had also been replaced in November, 1986 for failure to meet Technical Specification leakage limit. The cause of the excessive leakage could not be determined at that time (LER 86-007).

On March 15, 1990, chemistry technicians discovered that these isolation valves were leaking following sampling. Calculations were performed on March 17, 1990 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 of 5051.74 sccm. An actual LLRT would have required a pressure differential of 11.31 psi across each of the two isolation valves using a maximum pathway leakage method. Due to the fact that the plant was operating, performance of an LLRT was not feasible.

The decision was made on March 17 to allow continued operation provided the valves remained closed or that additional calculations be performed if the valves were stroked with flow through them. In light of the LLRT data obtained on September 19, 1990, the calculational methodology used on March 17 to establish compliance with Technical Specifications is being reevaluated.

Secondary containment is designed to collect the fission product leakage during and following a postulated design basis accident from the primary containment and delay it until it can be released to the environment after processing through the Annulus Exhaust Gas Treatment system [VC] such that the resultant offsite doses are less than the values set forth in 10 CFR 100 and 10 CFR 50, General Design Criteria 19. These valves are part of the Secondary Containment bypass leakage pathway. The maximum permitted total leakage rate from all bypass sources is 6.72 percent of the total containment leakage. As an added conservatism, the allowable bypass test leakage is limited to less than or equal to 75 percent of this value (5.04 percent). Due to the amount of leakage, this event is considered to be of minimal safety significance.

The replaced valve will be disassembled and a root cause analysis of its failure performed. A supplemental report will be issued to discuss the results of this analysis and to identify corrective actions which will be taken. Additionally, the supplemental report will provide the results of the reevaluation of the calculational methodology. This supplemental report will also identify all LLRT failures found during this outage which contribute to Secondary Containment bypass leakage. Pending completion of the root cause analysis, cycling of these valves will be limited to once per quarter for Position Indication Testing and PASS training and as required as a backup to the normal Reactor Water Sampling panel. Flow through these valves will be limited to semi-annually for PASS sampling and when required as a backup to the normal Reactor Water Sampling panel. Prior experience with these valves being cycled an average of one to two times per week and not failing for over a two year period indicates that these valves will perform satisfactorily throughout the next cycle with the above planned restrictions.

Energy industry Identification System codes are identified in the test as [XX].