CENTERIOR ENERGY

PERRY NUCLEAR POWER PLANT
10 CENTER ROAD
PERRY, OHIO 44081
(216) 259-3737

Mail Address:
P.O. BOX 97
PERRY, OHIO 44081

Robert A. Stratman
VICE PRESIDENT - NUCLEAR

February 8, 1993
PY-CEI/NRR-1605 L

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

Perry Nuclear Power Plant
Docket No. 50-440
LER 93-002

Dear Sir:

Enclosed is Licensee Event Report 93-002 for the Perry Nuclear Power Plant.

Sincerely,

Frank R Stead for

Robert A. Stratman

RAS:TSH:ss

Enclosure: LER 93-002

cc: NRC Project Manager
NRC Resident Inspector
NRC Region III

120014

JE22

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 5

TITLE (4)
Failed LLRT Surveillance Results in Technical Specification 3.0.3 Entry

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
01	09	93	93	002	00	02	08	93	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9): 3

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)		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)	X	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 (Include Area Code)
Teresa S. Hogan, Compliance Engineer, Extension 5283	(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
X	VA	ISV	P340	Yes					

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 15 single-spaced typewritten lines) (16)

On January 9, 1993 at 2242, the Secondary Containment Bypass Leakage exceeded the limitations stated in Technical Specification 3.6.1.2.d, resulting in an entry into Technical Specification 3.0.3. Manual shutdown for maintenance was in progress at the time of the event and operators continued to bring the plant towards Operational Condition 4.

The primary root cause has been attributed to improper stroke adjustment of the Containment Purge 42 inch supply inboard isolation valve (F0045) following actuator work in Refuel Outage 3. A contributing factor was the method used to correct previously identified leakage on November 5, 1992. The leakage past the 18 inch supply inboard isolation valve (F0195) has been attributed to valve travel being slightly out of adjustment, and to limited areas of unequal extrusion of the resilient seal.

The F0045 resilient seal was replaced using the updated vendor information and the valve travel was adjusted. The original seal will be further inspected for hardness. To correct the unequal extrusion of F0195, the clamping forces were relieved and the resilient seal was retightened using updated vendor information. Additionally, minor adjustments were made to the valve travel.

I. Introduction

On January 9, 1993 at 2242, the Secondary Containment Bypass Leakage exceeded the limitations stated in Technical Specification 3.6.1.2, resulting in an entry into Technical Specification 3.0.3. At the time of the event, a controlled plant shutdown for maintenance was in progress. The reactor pressure vessel [RPV] was at atmospheric conditions with the reactor water temperature at 260 degrees Fahrenheit. At 2326 on January 9, 1993, the required non-emergency one-hour notification was made to the NRC pursuant to the requirements of 10CFR50.72(b)(1)(i). This event is being reported under the requirements of 10CFR50.73(a)(2)(i).

II. Event Description

On January 9, 1993, Surveillance Instruction (SVI-M14-T9313) "Type C Local Leak Rate Test (LLRT) of 1M14 Penetration V313" was being performed to measure the as-found leakage rate of the Drywell and Containment Purge [VA] system isolation valves [ISV]. At 2242 on January 9, 1993, the leakage measured for the penetration was 22,000 standard cubic centimeters per minute (sccm) (0.22 La). As a result, the Technical Specification (TS) limits for Secondary Containment Bypass leakage specified in TS 3.6.1.2.d were exceeded and Technical Specification 3.0.3 was entered. Manual shutdown for maintenance was in progress at the time of the event and operators continued to bring the plant towards Operational Condition 4. At 2326 the one-hour ENS notification was made, as required by 10CFR50.72(b)(1)(i) At 2350 the plant entered Operational Condition 4 and exited Technical Specification 3.0.3 at 2352.

III. Cause Analysis

The initial investigation conducted on January 9, 1993, attributed approximately 20,800 sccm of the total penetration leakage past the Containment Purge [VA] 42 inch supply inboard isolation valve [ISV] (F0045) and 1200 sccm leakage past the 18 inch supply inboard isolation valve (F0195). It appeared that the resilient seal for F0045 had relaxed from the last known leak tight position. This valve had not been stroked or adjusted since its last satisfactory LLRT. Additionally, a bubble solution leakage test identified leakage at the 11 and 6 o'clock positions of F0195.

Historically, the valves associated with the Containment and Drywell Purge Ventilation System have only experienced significant leakage increases following actuator work. The F0045 actuator had been replaced during Refuel Outage 3. Satisfactory LLRT results were obtained during the retest and again in August, 1992 as a quarterly surveillance requirement. On November 5, 1992, during the quarterly LLRT surveillance, the total allowable Secondary Containment Bypass leakage rate limitation of Technical Specification 3.6.1.2.d was exceed. The

LLRT performed on January 9th, was part of the valve performance analysis required as a corrective action from LER 92022 in response to the leakage identified on November 5, 1992.

During troubleshooting of the leakage identified on November 5, 1992, an over-travel condition of F0045 was noted. The immediate corrective action for the leakage involved only minor adjustments of the resilient seal to the valve disc surface to reduce the identified leakage substantially below the Technical Specification limits. It was recognized that the valve over-travel condition would need correcting and a corrective action from LER 92022 required the valve travel to be adjusted when the plant was in an Operational Condition where the valve could be stroked. The adjustment to the valve over-travel condition was not considered critical at that time, as the noted leakage had been corrected by the seal adjustment. Additionally, the Technical Specification governing drywell and containment purge isolation valves, TS 3.6.1.8, required the inboard purge valve (F0045) to be sealed closed during Operational Condition 1, 2 and 3. Left in this condition, few forces should be acting on the valve to change its seating characteristics. To further ensure the valve complied with Technical Specifications and to allow the valve's performance following operation to be evaluated, administrative controls were established requiring LLRT and stroke time surveillances to be performed prior to handling irradiated fuel or entering Operational Conditions 1, 2 and 3 following shutdown conditions.

During the November 5, 1992 event, F0195 was determined not to be leaking and no adjustments to the valve seat or travel were made. Unlike F0045, this valve is stroked during Operational Condition 1, 2, and 3, and was operated to permit purging of the containment as required between November 5, 1992 and January 9, 1993.

The primary root cause of both the November 5, 1992 and January 9, 1993 events has been attributed to improper stroke adjustment of F045 following actuator work in Refuel Outage 3. A contributing factor to the January 9th event was the method used to tighten the F0045 resilient seal to the valve disc surface at the points of leakage on November 5, 1992. At this time seal degradation can not be ruled out as another contributing factor. The leakage past F0195 has been attributed to valve travel being slightly out of adjustment and to two limited areas of unequal extrusion of the resilient seal.

Adjustments made to F0045 in November were performed in accordance with the manufacturer's instruction manual. When contacted following the January 9th event, the manufacturer provided assistance to modify the adjustment and installation procedures associated with the seal. Instead of local adjustment in the vicinity of the leak, tightening of the entire adjustment ring which holds the resilient seal to the disc is now required. Tightening of the adjustment ring in the vicinity of the leak, combined with possible inadequate

lubrication of the clamping surfaces resulted in unequal extrusion of the resilient seal. After the local adjustments made on November 5, 1992, retest identified minimal remaining leakage; however, over time, new leakage developed at different points due to relaxation of the resilient seal.

IV. Corrective Action

Due to concerns over the seal integrity following local adjustments, the F0045 resilient seal was replaced using the updated vendor information. The valve travel was adjusted to correct the over-travel condition. Seal inspection showed some wear, but little degradation. The original seal will be further inspected for hardness. The F0195 seal was inspected and found to be acceptable for continued use. To correct the unequal extrusion, the clamping forces were relieved and the resilient seal was retightened using the updated vendor information. Additionally, minor adjustments were made to the travel of F0195. Satisfactory LLRT results were obtained on February 4, 1993 with a penetration leakage of 39.42 sccm.

In order to determine the cause for the identified over-travel on the affected valves, maintenance practice and vendor information affecting previous valve travel adjustments will be evaluated for adequacy. Additionally, the vendor manual will be updated to assure that any future work will be performed using the updated vendor information.

The two other valves in this penetration were also evaluated and it was determined that no adjustments are required at this time. No actuator work had been performed which could affect valve travel adjustments. Also numerous satisfactory LLRT results indicate that the uneven tightening of the sealing surface has not affected the sealing capability of these valves.

V. Safety Analysis

The Containment Vessel and Drywell Purge System purges potentially contaminated air from the containment vessel and drywell, via a controlled and filtered release path, maintaining airborne activity levels within acceptable values for personnel entry. 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 doses are less than the values set forth in 10CFR100 for offsite and 10CFR50, General Design Criterion 19 for the control room.

The design basis leak rate of the primary containment and its penetrations is 0.2 percent per day for the duration of the accident (La). The penetration leakage did not cause the combined leakage rate design basis of La to be exceeded.

The rate assumed in the accident analysis from Secondary Containment bypass leakage is 0.0672 La. Because the valves identified in this report are part of the Secondary Containment bypass leakage pathway, and because the leakage rate significantly exceeded the accident analysis value, this event is considered safety significant. The safety significance of this event is partially mitigated by the presence of two other operable leak tight isolation valves in the penetration, that would have closed in the event of an accident.

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