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): Perry Nuclear Power Plant, Unit 1

DOCKET NUMBER (2): 05000 440

PAGE (3): 1 OF 4

TITLE (4): Failure to Perform Engineering Evaluation of Inoperable Snubbers Result in Technical Specification Violation

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
02	18	93	93	- 007 -	00	03	19	93	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9): 5

POWER LEVEL (10): 0

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
	20.405(a)(1)(iii)	X	50.73(a)(2)(i)		50.73(a)(2)(viii)(A)		(Specify in Abstract below and in Text, NRC Form 366A)
	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: Ron W. Gaston, Compliance Engineer Extension 5004

TELEPHONE NUMBER (Include Area Code): (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

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 February 18, 1993 while performing an evaluation in response to previously initiated nonconformance reports (NRs), it was determined that certain inoperable snubbers which were the subject of the NRs may have been inoperable since January 1991. This hypothesis was supported by a minor Reactor Pressure Vessel (RPV) level offset for the instrument channel associated with the inoperable snubbers.

The mechanical snubber failures were attributed to dried out lubricant resulting from exposure to elevated temperatures in the upper drywell area. The failed snubbers were subsequently replaced. Additional snubbers in the immediate vicinity were stroke tested and found acceptable. Due to a prior occurrence involving elevated drywell temperatures, all mechanical snubbers in the upper drywell area are being monitored to better determine their service life in this environment. All snubbers in this area will have an augmented functional test performed during the next refueling outage (RFO-4).

I. Introduction

On February 18, 1993, an evaluation performed in response to a nonconformance report (NR) involving inoperable snubbers determined that the snubbers may have been inoperable since January 1991, as evidenced by a review of anomalous readings within the affected RPV instrumentation channel. This event was considered a violation of the applicable action requirement of Technical Specification 3.7.4, which requires that with one or more snubbers inoperable, the inoperable snubber be restored or replaced within 72 hours and an engineering evaluation be performed on the attached component or the attached system declared inoperable. This event is being reported pursuant to 10 CFR 50.73(a)(2)(i)(B) due to the failure to perform the evaluation at the time the snubbers are presumed to have been inoperable.

At the time of discovery the plant was shutdown for a mid-cycle maintenance outage in Operational Condition 5 (Refueling) with the reactor pressure vessel at atmospheric pressure and reactor water temperature at 108 degrees Fahrenheit.

II. Description

On February 1, 1993, Nonconformance Report (NR) 93-S-033 was written due to a suspected locked-up snubber [SNB] on the sensing line for steam condensing pot 1B21-D004A. The suspect snubber was identified during a walkdown of drywell snubbers which was performed to visually inspect for possible damage resulting from water spray inside the drywell. The water spray originated from refueling bulkhead manway flange leakage identified while raising reactor vessel level.

The PSCO Model PSA-1/4 (1H22H0407) snubber identified in NR 93-S-033 was unpinned and stroke tested. A preliminary evaluation determined that the lubrication in the snubber was dried out. To determine the extent of the problem five additional snubbers at the same general drywell elevation were stroke tested. Two of these snubbers in the immediate vicinity (1H22H0408 and 1H22H0409) also failed the stroke test on February 16, 1993; the other three snubbers tested (1G41H1025, 1H22H0558 and 1H22H2483) had acceptable test results. Two additional NRs 93-S-048 and 93-S-049 were written to document the evaluation and disposition of snubbers 1H22H0408 and 1H22H0409 respectively. Additionally, due to unrelated concerns, seven additional snubbers were stroke tested during the 1993 mid-cycle outage and found to be acceptable.

The three failed snubbers were attached to the sensing line for steam condensing pot 1B21-D004A, which is associated with the Channel A reactor water level instrumentation. The snubbers allow for thermal growth of the level instrument line during plant heatups and cooldowns. Evaluations performed per the respective NRs concluded that the piping stresses remained within the pertinent ASME Code allowables with the snubbers locked-up in their as found position. The piping was also evaluated to be acceptable on an as-is basis for continued service following the maintenance outage.

During the evaluation of NR's 93-S-048 and 93-S-049 a probable link between snubber inoperability and past RPV level instrument reading anomalies was suspected. The trending data from the instrumentation indicated the snubbers in question may have been locked-up since as early as 1991, based on the "A" narrow range instrument reading consistently higher than the other 3 channels. This difference was within the allowable range between channels for these instruments. During plant startup in March of 1993 the previous difference between the channel reading was significantly reduced, thereby confirming the link to the degraded snubber.

III. Cause Analysis

The three affected snubbers were disassembled after failing their respective stroke tests to determine the failure mechanism. The internal lubricated surfaces within the snubber were coated with dried out lubricant. Based on the fact that there were no visual indications of mechanical defects or damage to the snubbers, the primary cause for the failures was attributed to the dried out lubricant resulting from exposure to elevated ambient temperature conditions.

IV. Similar Events

In April 1989, Licensee Event Report (LER) 89-10 was submitted to report a high temperature condition in the upper drywell area which resulted in heat damage to electrical cables and mechanical snubbers in the affected zone. The high temperature condition was caused by a lack of sufficient refueling bellows insulation. The high temperature condition was improved by installing reflective insulation on the drywell side of the refueling bellows and improving Drywell Cooling System [VB] air flow in the area surrounding the refueling bellows. Additional information regarding the elevated drywell temperature condition identified in 1989 was submitted to the NRC by letter PY-CEI/OIE-0348L and PY-CEI/OIE-0353L, dated April 19, 1989 and July 3, 1989 respectively.

V. Safety Analysis

Condensing chamber 1B21-D004A provides input to the reactor pressure vessel level Channel A instrumentation which includes wide range and narrow range level transmitters. Automatic protective functions associated with these transmitters include a Level 8 (L8) high level trip at 219.5 inches, a L3 low RPV level scram at 177.7 inches and L2 and L1 trips which initiate Emergency Core Cooling System (ECCS) functions.

The setpoints for the reactor vessel level instrument trips are established based on the cold vessel conditions. Actual calibration of the instrumentation is performed based on thermal expansion of the vessel and instrument condensing pots at plant normal operating conditions. This results in added margin incorporated in the setpoints such that protective functions will occur prior to the required setting. The specific margin built into the calibration is 1.5 inches for the Narrow Range L3 settings, 1.0 inch for the Wide Range L1 and L2 settings and 0.8 inches for the the Wide/Narrow Range L8 trip settings. The net effect of the restrained condition of 1B21-D004 was to cause the L8 trip to occur approximately 0.5 inches lower than the indicated level (approximately 219.5 inches) and the L3 and L2/L1 trips to occur approximately 0.9 inches and 0.6 respectively, lower than the corresponding indicated values which is in a non-conservative direction. However, these errors are enveloped by the margin included in the calibration setpoint values.

Since the associated piping stresses were within the pertinent ASME Code allowable values and the condition did not violate the Technical Specification acceptance criteria associated with the affected RPV level instrumentation, this event is not considered to be safety significant.

VI. Corrective Actions

The condensing chamber sensing line snubbers described previously were replaced during the 1993 mid-cycle outage with snubbers of the same type and design. Based on past experience with snubbers exposed to temperatures in the range of 200-205 degrees Fahrenheit (which envelopes the environmental conditions the snubbers experienced), a successful life of two operating cycles is expected. Therefore these snubbers are considered as acceptable for continuous service until at least RFO4.

As stated previously, 3 additional snubbers located at the same drywell elevation as the snubber which failed, successfully passed the stroke tests performed as part of the NR 93-S-033 disposition. Seven additional snubbers stroke tested for an unrelated cause during the 1993 Maintenance Outage also successfully passed their respective stroke tests. In total, 13 of the 34 snubbers in the upper drywell population were tested, with the three failures occurring in the same local area. Due to a Technical Specification functional snubber test failure which occurred during RFO3, an augmented test plan was developed, which includes functional testing of all upper drywell snubbers. These functional tests are scheduled to be performed during RFO4.

Additionally, design modifications made during RFO1 have substantially reduced upper drywell temperatures, thereby improving the service environment for the upper drywell snubbers.

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