REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR: 8708050048 DOC. DATE: 87/07/31 NOTARIZED: NO DOCKET #
FACIL: 50-400 Shearon Harris Nuclear Power Plant, Unit 1, Carolina 05000400

AUTH. NAME	AUTHOR AFFILIATION
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RECIP. NAME	RECIPIENT AFFILIATION

SUBJECT: LER 87-029-01: on 870428, engineer determined that snubber found damaged on 870422 required action per Tech Spec 3.7.8. Snubber damaged due to water hammer occurring when valve isolated. Snubber replaced. W/870731 ltr.

DISTRIBUTION CODE: IE22D COPIES RECEIVED: LTR 1 ENCL 1 SIZE: 4
TITLE: 50.73 Licensee Event Report (LER), Incident Rpt, etc.

NOTES: Application for permit renewal filed. 05000400

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	NRR/DREP/RAB	1	1	NRR/DREP/RPB	2	2
	NRR/PMAS/ILRB	1	1	NRR/PMAS/PTSB	1	1
	REG FILE 02	1	1	RES DEPY GI	1	1
	RES TELFORD, J	1	1	RES/DE/EIB	1	1
	RGN2 FILE 01	1	1			
EXTERNAL:	EG&G GROH, M	5	5	H ST LOBBY WARD	1	1
	LPDR	1	1	NRC PDR	1	1
	NSIC HARRIS, J	1	1	NSIC MAYS, G	1	1

TOTAL NUMBER OF COPIES REQUIRED: LTTR 45 ENCL 43

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)
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TITLE (4)
Steam Generator Blowdown Piping Snubber - Inoperable

EVENT DATE (5) MONTH	DAY	YEAR	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	REPORT DATE (7) MONTH	DAY	YEAR	OTHER FACILITIES INVOLVED (8) FACILITY NAMES	DOCKET NUMBER(S)
0 4	2 8	8 7	8 7	0 2 9	0 1	0 7	3 1	8 7		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1
POWER LEVEL (10): 1 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)	[ ] 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 AREA CODE	
Dean L. Tibbitts - Director Regulatory Compliance	9 1 9	3 6 2 - 2 7 1 8

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS

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 fifteen single-space typewritten lines)* (16)

On April 22, 1987, with the plant in Mode 1 at 100% power level a snubber was discovered damaged on a steam generator blowdown system pipe in the Reactor Auxiliary Building. Action was initiated to repair the snubber, and Technical Specification 3.7.8, SNUBBERS, was reviewed for required actions. A decision was made by shift operations personnel and subsequently confirmed by plant management that Specification 3.7.8 was not applicable to this snubber based on the non-safety significance of the attached system and the location of the snubber and the respective piping relative to safety related equipment.

Upon further review, it was determined on April 28 that the snubber in question was covered by Specification 3.7.8 and that action under this specification was required. The snubber was replaced on April 28 and an engineering evaluation was conducted to verify that the out of service snubber did not result in unacceptable damage to the piping.

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

DESCRIPTION

On April 22, 1987, while the plant was operating in Mode 1 at 100% power level a snubber was found damaged on the Steam Generator Blowdown System piping from Steam Generator 1A. The snubber was located inside the Reactor Auxiliary Building. A work request was initiated to repair the snubber. The on-shift operations personnel reviewed Technical Specification 3.7.8, SNUBBERS, and determined that this specification was not applicable because the attached system had no safety function, and was not covered by Technical Specifications.

Work was begun on April 23 to correct the situation. In addition to the damaged snubber, two broken pipe supports and apparent pipe displacement were discovered. The supports were replaced in the evening of April 23. The snubber could not be replaced due to inadequate clearances resulting from the pipe displacement.

On April 27, engineers from the In-Service Inspection group, who had become involved in the repair, informed the Manager Operations that the snubber did require action under Technical Specification 3.7.8, based on their understanding of the specification and its bases. Operations and Regulatory Compliance reviewed the situation and determined that Specification 3.7.8 did not apply based on the non-safety significance of the attached system, on the location of the affected piping relative to safety-related systems, and on the location of the affected piping relative to the containment isolation valve for that particular blowdown line.

The Quality Assurance department became aware of the situation and issued a Non-Conformance Report on April 28. This led to a more detailed review by Regulatory Compliance of the particular snubber and the Technical Specification bases for Specification 3.7.8., as well as the design bases for the steam generator blowdown system. It was determined that Specification 3.7.8 did in fact apply to the snubber and therefore the plant had not met the requirements of the Technical Specifications to restore the snubber within 72 hours and to evaluate the effect on the attached system, or declare the affected system inoperable. However, since the Steam Generator Blowdown System is not covered by a specific specification, declaration of inoperability would require no specific action be taken.

The snubber was replaced on April 28. An engineering evaluation concluded that the "as-left" condition of the piping was acceptable; this was completed on May 7, 1987.

During the time period in question, steam generator blowdown was in service. The plant was in the applicable modes for which Specification 3.7.8 applies throughout the period. However, during an unrelated outage from May 2 to May 9, the plant entered Mode 4 and steam generator blowdown was secured.

★U.S.GPO.1986-0-624-538/455

NRC Form 366A
(9-83)

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
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TEXT (If more space is required, use additional NRC Form 366A's) (17)

CAUSE

The design of the Steam Generator Blowdown piping is safety-related because of FSAR commitments made to support the analysis for high energy piping breaks outside the reactor containment building. Having a Seismic Category I design limits the number of postulated piping breaks to consider as well as relaxing the requirements for pipe whip restraints. Due to the safety-related design of the pipe and the consequences of blowdown line ruptures in the Reactor Auxiliary Building, snubbers in the Steam Generator Blowdown System are subject to Specification 3.7.8. This information is not stated in the Technical Specification nor in the bases section, nor is it clear based on the action statement of the specification.

Personnel and managers involved in the assessment of the snubber were also not aware of its significance to pipe break analyses. Because a Technical Specification Limiting Condition for Operation was not entered, immediate management attention was not given to the situation and the snubber repair did not proceed within the time allowed by the specification.

The cause of the blowdown piping displacement and snubber damage was determined to be water hammer occurring when the isolation valve for the line from containment is opened.

ANALYSIS

This event is being reported per 10CFR50.73 (a) (2) (i) (B) as a violation of the Technical Specifications. The plant design depends on the integrity of steam generator blowdown piping to satisfy high energy line break analyses requirements. Technical Specifications limit the allowed time during which the condition of the piping can be indeterminate prior to requiring action to be taken. This time limit, 72 hours, was exceeded.

There were no safety consequences as a result of this event since the blowdown piping was not damaged to the point where leakage occurred.

CORRECTIVE ACTION

The probability of water hammer in the steam generator blowdown piping will be reduced by increasing the open stroke time of the isolation valve. This change will be completed in conjunction with the next scheduled outage, which is currently planned for the last quarter of 1987.

The requirements for action under Specification 3.7.8 have been addressed in a Technical Specification Interpretation document.

★U.S.GPO:1986-0-624-538/455

CP&L

Carolina Power & Light Company

HARRIS NUCLEAR PROJECT
P.O. Box 165
New Hill, NC 27562

JUL 31 1987

File Number: SHF/10-13510C
Letter Number: HO-870478 (0)

U.S. Nuclear Regulatory Commission
ATTN: NRC Document Control Desk
Washington, DC 20555

SHEARON HARRIS NUCLEAR POWER PLANT UNIT 1
DOCKET NO. 50-400
LICENSE NO. NPF-63
LICENSEE EVENT REPORT 87-029-01

Gentlemen:

In accordance with Title 10 to the Code of Federal Regulations, the enclosed Licensee Event Report is submitted. This report fulfills the requirement for a written report within thirty (30) days of a reportable occurrence and is in accordance with the format set forth in NUREG-1022, September, 1983.

Revision 1 to LER 87-029-00 is being submitted to change the completion date for the corrective action. LER 87-029-00 stated that the change to the open stroke time of the isolation valve will be completed by July 31, 1987. Needle valves have been installed to accomplish this; but due to operational considerations, the steam generator blowdown system cannot currently be isolated as needed to adjust the stroke times of the valves. Therefore, the corrective action will now be completed in conjunction with the next scheduled outage.

Very truly yours,

R. A. Watson
Vice President
Harris Nuclear Project

ONH:RAW:ddl

Enclosure

cc: Mr. B. Buckley (NRR)
Dr. J. Nelson Grace (NRC - RII)
Mr. G. Maxwell (NRC - SHNPP)

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