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February 25 1993
ND3MNO:3417

Beaver Valley Power Station, Unit No. 2
Docket No. 50-412, Licensee No. NPD-73
LER 93-001-00

United States Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Gentlemen:

In accordance with Appendix A, Beaver Valley Technical Specifications, the following Licensee Event Report is submitted:

LER 93-001-00, 10 CFR 50.73.a.2.ii.B, "Design Stress for the Auxiliary Feedwater System Exceeded Due to Water Hammer."

L. R. Freeland
General Manager
Nuclear Operations

DJS/sl

Attachment

020056

JE22

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NRC FORM 366 (6-89)

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104
EXPIRES: 4/30/92

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE RECORDS AND REPORTS MANAGEMENT BRANCH (P-530), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Beaver Valley Power Station Unit 2	0 5 0 0 0 4 1 2	1 OF 0 4

TITLE (4)

Design Stress For The Auxiliary Feedwater System Exceeded Due To Water Hammer

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 1	2 6	9 3	9 3	— 0 0 1 —	0 0	0 2	2 5	9 3	N/A	0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 0 9 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)	[ ] 50.73(a)(2)(i)	[ ] 50.73(a)(2)(viii)(A)	
[ ] 20.405(a)(1)(iv)	[X] 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)
L. R. Freeland, General Manager Nuclear Operations	4 1 2 6 4 3 - 1 2 5 8

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
B	B A	S P T	P 0 2 9	N					

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 1/26/93 at 1720 hours, the results of extensive engineering analysis of previous plant events identified that the stress on the Auxiliary Feedwater (AFW) piping to the "C" Steam Generator, for combined water hammer and seismic events, exceeded the design stress allowables. It has been hypothesized that the "C" AFW line has been subjected to water hammer due to steam pocket formation (voiding) and steam bubble collapse. There have been two events at Beaver Valley Power Station Unit 2 since it went operational in 1987 that have been attributed to water hammer. The location of the inside containment AFW check valves (inside CV's) and piping layout have the potential to cause voiding (see Figure 1). The inside CV's are located near the main feedwater lines causing the piping upstream to be at elevated temperatures. The inside CV's are also at a higher elevation than the AFW supply tank minimum water level. If the inside CV's leak rate is less than the outside CV's, the AFW piping will depressurize causing voiding upstream of the inside CV's due to elevated temperatures. AFW automatic actuation with unthrottled flow to a voided AFW line is the main concern for water hammer. There were minimal safety implications as a result of this event. The mechanical snubbers were replaced with new hydraulic snubbers with a higher load rating.

NRC Form 366 (6-89)

Description of Event

On 1/26/93 at 1720 hours, the results of extensive engineering analysis of previous plant events identified that the stress on the Auxiliary Feedwater (AFW) piping to the "C" Steam Generator, for combined water hammer and seismic events, exceeded the design stress allowables. It has been hypothesized that the "C" AFW line has been subjected to water hammer due to steam pocket formation (voiding) and steam bubble collapse. There have been two events at Beaver Valley Power Station Unit 2 since it went operational in 1987 that could be attributed to water hammer. In event No 1, identified in April 1989, mechanical snubbers 2FWE-PSSP-009 and 2FWE-PSSP-010 on the "C" AFW line were found mechanically frozen during an ISI inspection. The cause of event no. 1 was unknown at that time and the snubbers were replaced with identical new snubbers. Event No. 2, identified in April 1992, mechanical snubber 2FWE-PSSP-009 was found mechanically frozen and snubber 2FWE-PSSP-010 was found damaged during an ISI inspection on the "C" AFW line. The cause for the snubber failure/damage was apparently due to excessive compressive or impact loading as a result of an unanticipated water hammer. The mechanical snubbers were replaced with new hydraulic snubbers with a higher load rating.

The location of the inside containment AFW check valves (inside CV's) and piping layout have the potential to cause steam voiding (see Figure 1). The inside CV's are located near the main feedwater lines causing the piping upstream to be at elevated temperatures (measured 268.3 degree F on top of pipe, 200 degrees F on bottom of pipe, approximately 2 feet upstream of the "C" AFW check valve). The inside CV's are at a higher elevation than the AFW supply tank minimum water level. If the inside CV's leak rate is less than the outside CV's, the AFW piping will depressurize causing voiding upstream of the inside CV's due to elevated temperatures. The "C" AFW line has been depressurized.

Cause of Event

The results of extensive engineering analysis of previous plant events identified that the stress on the Auxiliary Feedwater (AFW) piping to the "C" Steam Generator, for combined water hammer and seismic events, exceeded the design stress allowables. The AFW piping and pipe supports are classified as QA Category I, seismic, safety class II. The piping is designed to the requirements of the ASME Boiler and Pressure Vessel Code, Section III, 1971 Edition including addenda through Winter 1972. The supports are designed to the guidelines established in the 6th Edition of the AISC manual of Steel Construction. The AFW piping and supports have been designed for thermal, pressure, deadload, and seismic conditions. Anticipated or "designed" water hammer transients were not previously analytically considered. It was previously assumed that the piping layout and components had been designed to preclude the development of postulated water hammer events. In retrospect, there is the potential for unanticipated water hammer as a result of unthrottled AFW flow from automatic AFW pump starts to a potentially voided "C" AFW line.

Corrective Actions

The following corrective actions have been or will be taken as a result of this event:

1. Frequent monitoring of the AFW piping for evidence of elevated containment penetration temperatures, system pressures (temporary pressure gages installed), noise/water hammer.

2. A Basis for Continued Operation (BCO) has been prepared to document operation in this condition. This is based upon analysis to quantify loads resulting from a postulated water hammer event.

3. The Station is pursuing a long term solution.

4. The Station will perform ISI inspections of the AFW piping and piping supports in containment following an unplanned AFW pump start that results in unthrottled AFW flow.

5. The mechanical snubbers were replaced with new hydraulic snubbers with a higher load rating. The new hydraulic snubbers are adequately sized to withstand a water hammer event.

Reportability

This report is being submitted in accordance with 10CFR50.73.a.2.ii.B because this event resulted in operation outside the Unit 2 design basis.

Safety Implications

The safety implications are minimal. Engineering analysis has shown that in cases where the pipe stress levels under the revised load combinations (to include water hammer) occurring simultaneously with a seismic event exceeded original design code normal/upset allowables, an assessment of system operability was made to the design code faulted allowables. The pipe stress levels, for combined water hammer and seismic events, are now within the design code faulted allowables. In developing the BCO, it was determined that the use of alternate code faulted allowables (i.e. Appendix F or later revisions of the ASME code) was not required due to the rigorous analysis performed to more adequately quantify the stress.

Previous Similar Events

There were no previously reported similar events.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

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)

Figure No. 1