NRC Form 366 (9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/85

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1): Davis-Besse Unit 1

DOCKET NUMBER (2): 0 5 0 0 0 3 4 6

PAGE (3): 1 OF 0 3

TITLE (4): PORV Discharge Line Overstressed Due to Inadequate Heat Trace

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)
1 0	0 7	8 5	8 5	0 1 9	0 1	0 2	0 6	8 7		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 5

POWER LEVEL (10): 0 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)	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)	
P. Straube	4 1 9	2 4 9 - 5 0 0 0

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)

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)

The Pressurizer Power Operated Relief Valve (PORV) inlet line contained a loop seal. By design the loop seal had to be maintained at 500°F to prevent a 'cold' slug of water from being discharged through the PORV and overstressing downstream piping that connects the PORV to the Reactor Coolant System Quench Tank. A review of the June 9, 1985 event showed that the loop seal temperature was only 469°F prior to the lifting of the PORV. A new analysis of piping stresses using 469°F determined that a high system loading would result on the downstream side of the valve. Analytically, the pipe was overstressed and several supports overloaded during the lifting of the PORV. Inspection of several supports in the highly loaded region of the system and examination of two field welds showed no damage in the area immediately downstream of the PORV. However, later inspections revealed evidence of high loading near the Quench Tank.

1 The loop seal problem was resolved by adding a drain line upstream of the PORV. This will prevent the buildup of water in the loop seal.

1 Also, due to evidence of high loading near the Quench Tank, several piping welds were examined to ensure no piping damage was sustained. A complete review of the loading of the PORV lines resulted in the upgrading of two supports.

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NRC Form 366A
(9-83)

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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

Description of Occurrence:

On February 1, 1985, Toledo Edison made a submittal (Serial No. 905) for Item II.D.1 of NUREG-0737 concerning the Pressurizer Code Safety Valves and the Power Operated Relief Valve (PORV) and associated piping. The submittal included the results of a Teledyne analysis performed to evaluate a PORV blowdown with a 500°F (or higher) loop seal temperature. On June 7, 1985 (Log 1764) the NRC requested further information on this analysis. On July 19, 1985 (Serial No. 1171) Toledo Edison supplied the information to the NRC and also noted that the 500°F loop seal temperature could not be maintained. Since the original analysis done by Teledyne used 500°F, Toledo Edison decided to have it redone and reported to the NRC.

The PORV inlet line contained a loop seal prior to the 1985-86 outage. By design the loop seal must be maintained at 500°F to prevent a colder slug of water from being discharged through the PORV and overstressing downstream piping that connects the PORV to the Reactor
1 Coolant System Quench Tank. A review of the June 9, 1985 event showed that the loop seal temperature was only 469°F prior to the lifting of the PORV. A new analysis of piping stresses using 469°F determined that a high system loading would result on the downstream side of the valve.

During the June 9, 1985 event, the PORV lifted three times. Since the loop seal temperature was less than 500°F an analysis was conducted to assess the stresses in the piping as a result of the initial lifting of the PORV. The results of this PORV blowdown analysis were reported by Teledyne to Toledo Edison in a September 19, 1985 letter (No. 6388.4). The results indicated that analytically the Class 3 piping had been overstressed on the discharge side of the PORV and that several pipe supports on the discharge side had been overloaded.

On October 7, 1985 it was determined that even though this information would be reported to the NRC as a response committed to in the July 19, 1985 letter, that it should also be reported as an LER. The loop seal temperature less than 500°F and resultant overstressed discharge line is reportable under 10CFR50.73(a)(2)(ii)(B) as a condition outside the design basis.

Designation of Apparent Cause of Occurrence:

The cause of the event was the inability to maintain a 500°F loop seal with the installed heat tracing. The overstressing of the PORV discharge line is due to temperatures less than 500°F on the previous loop seal installation on the PORV inlet line. The inability to maintain the temperature is due to a heat trace application that has been very difficult to maintain.

Analysis of Occurrence:

Although analysis shows that the piping immediately downstream of the PORV was overstressed and the supports overloaded, an examination of the system in this area does

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

not reveal any damage to the piping or supports. The analytical model for the analysis conducted, like most analyses, predicted a conservative load and, thereby, a more severe condition of blowndown loading than occurred on June 9, 1985. Due to the examination results it is assumed the loading on the system in the analytically predicted high load region was not severe enough to produce damaging effects on the piping and supports based on the quality control inspections conducted. However, due to the pipe support
1 reinspection efforts two NCR's (85-920 and 85-921) on two supports close to the Quench Tank identified support deficiencies which could have been caused by high loads in that region. The analysis which had been conducted by Teledyne did not show high analytical loads in this region.

Corrective Action:

The corrective action to eliminate the high loading experienced by the system when the
1 PORV is discharged was to install a drain line upstream of the PORV. This will keep the loop seal drained and free of accumulated water which causes the high loads during PORV discharge. Also, due to the evidence of high loading near the Quench Tank, several piping welds near the Quench Tank were nondestructively examined to assure that no piping damage in this area was sustained. NCR's 85-920 and 85-921 were dispositioned and reworked. The existing heat trace was removed.

During a detailed review of the Teledyne analysis for the current configuraion and operating modes (drained loop seal with 400°F subcooled water discharge at 2450 psig), a number of modeling discrepancies were identified in the analysis. Also, several support loads were found to not be bounded by the 500°F loop seal discharge as previously stated.
1 A thorough review of all support designs for the PORV inlet and outlet piping was conducted to provide confidence that the as-built designs qualified by Teledyne were acceptable for the design loads imposed by the current piping configuration. This review resulted in identifying two support structures CCA-8-H2 and GCC-8-H16 which did not meet FSAR/USAR allowables. These supports were modified by adding stiffener plates to these supports per FCR 86-0350 and 86-0352 respectively.

Failure Data:

This is the first report of overstressing due to the inability of installed heat tracing to maintain 500°F on the loop seal.

REPORT NO: NP-33-85-25

DVR No (s): 85-158

TOLEDO EDISON

February 6, 1987

Log No. KA87-0061
NP-33-85-25 Rev. 1

Docket No. 50-346
License No. NPF-3

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

Gentlemen:

Enclosed is Revision 1 to Licensee Event Report 85-019. The revisions to the report are indicated by "1" in the left margin of each page.

Please destroy or mark superseded your previous copy of this report and replace with the attached revision.

Yours truly,

Louis F. Storz

Louis F. Storz
Plant Manager
Davis-Besse Nuclear Power Station

LFS/ed

Enclosure

cc: Mr. James G. Keppler
Regional Administrator,
USNRC Region III

Mr. Paul Byron
DB 1 NRC Resident Inspector

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