NRC FORM 366 (7-77)

U. S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT

CONTROL BLOCK: (1) (PLEASE PRINT OR TYPE ALL REQUIRED INFORMATION)

0 1 | O H D B S 1 (2) LICENSEE CODE | Ø Ø - Ø Ø Ø Ø Ø - Ø Ø (3) LICENSE NUMBER | 4 1 1 1 1 (4) LICENSE TYPE | (5) CAT

CON'T 0 1 | REPORT SOURCE L (6) | Ø 5 Ø Ø Ø 3 4 6 (7) DOCKET NUMBER | Ø 8 2 Ø 8 2 (8) EVENT DATE | 0 1 3 1 8 6 (9) REPORT DATE

EVENT DESCRIPTION AND PROBABLE CONSEQUENCES (10)

0 2 (NP-33-82-43) On August 20, 1982 at 1655 hours, the containment to shield building
0 3 differential pressure reached a maximum of 25.2 inches of water, exceeding the
0 4 Technical Specification limit (T.S. 3.6.1.4) of 25.0 inches water. The action
0 5 statement requirements were being met since the station was in Mode 3 at the time
0 6 of the occurrence. There was no danger to the health and safety of the public or
0 7 to station personnel. The differential pressure of 25.2 inches of water (0.91 psig)
0 8 represents only 2.5% of the Containment Vessel design pressure (36 psig).

	SYSTEM CODE	CAUSE CODE	CAUSE SUBCODE	COMPONENT CODE	COMP SUBCODE	VALVE SUBCODE
0 9	S A (11)	B (12)	A (13)	Z Z Z Z Z Z (14)	Z (15)	Z (16)

(17) LER/RO REPORT NUMBER	EVENT YEAR		SEQUENTIAL REPORT NO.		OCCURRENCE CODE	REPORT TYPE		REVISION NO.
	8 2	—	Ø 3 9	/	Ø 3	X	—	1

ACTION TAKEN	FUTURE ACTION	EFFECT ON PLANT	SHUTDOWN METHOD	HOURS (22)	ATTACHMENT SUBMITTED	NPRD-4 FORM SUB.	PRIME COMP. SUPPLIER	COMPONENT MANUFACTURER
X (18)	X (19)	Z (20)	Z (21)	Ø Ø Ø Ø	Y (23)	N (24)	Z (25)	Z 9 9 9 (26)

CAUSE DESCRIPTION AND CORRECTIVE ACTIONS (27)

1 0 The cause is a design error in the Containment Purge System valves. The station
1 1 has been committed to keeping these valves closed while in Modes 1-4, due to design
1 2 deficiencies concerning their ability to close under accident conditions and the
1 3 SFAS logic. The containment pressure was restored within limits by 1843 hours
1 4 using four containment vacuum relief valves.

1 5 | FACILITY STATUS C (28) | % POWER Ø Ø Ø (29) | OTHER STATUS (30) NA | METHOD OF DISCOVERY A (31) | DISCOVERY DESCRIPTION (32) During RCS heatup process.

1 6 | ACTIVITY RELEASED Z (33) | CONTENT OF RELEASE Z (34) | AMOUNT OF ACTIVITY (35) NA | LOCATION OF RELEASE (36) NA

PERSONNEL EXPOSURES
1 7 | NUMBER Ø Ø Ø (37) | TYPE Z (38) | DESCRIPTION (39) NA

PERSONNEL INJURIES
1 8 | NUMBER Ø Ø Ø (40) | DESCRIPTION (41) NA

LOSS OF OR DAMAGE TO FACILITY (43)
1 9 | TYPE Z (42) | DESCRIPTION NA

PUBLICITY
2 0 | ISSUED N (44) | DESCRIPTION (45) NA | NRC USE ONLY

DVR #82-093

NAME OF PREPARER Ted Lang

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TOLEDO EDISON COMPANY
DAVIS-BESSE NUCLEAR POWER STATION UNIT ONE
SUPPLEMENTAL INFORMATION FOR LER NP-33-82-43

DATE OF EVENT: August 20, 1982

FACILITY: Davis-Besse Unit 1

IDENTIFICATION OF OCCURRENCE: Containment to shield building differential pressure high.

Conditions Prior to Occurrence: The unit was in Mode 3 with Power (MWt) = 0 and Load (Gross MWe) = 0.

Description of Occurrence: On Friday, August 20, 1982, the containment to shield building differential pressure gradually increased during the process of heating the Reactor Coolant System (RCS). By 1655 hours, control room gauge PDI645 had reached a maximum of 25.2 inches of water, exceeding the Technical Specification limit (T.S. 3.6.1.4) of 25.0 inches of water. The action statement of Technical Specification 3.6.1.4 required the station to be within limits in one hour or to be in hot standby within six hours and in cold shutdown within the following 30 hours. The plant was in hot standby at the time of the occurrence. At 1842 hours on the same day, check valves on each of four containment vacuum breakers had been isolated and secured open. The motor operated valves on each of these vacuum breakers were reopened and the containment pressure was within specifications by 1843 hours. At 1930 hours, the vacuum breakers were restored to normal operation. Therefore, the action statement of Technical Specification 3.6.1.4 was followed.

Designation of Apparent Cause of Occurrence: The cause of the occurrence was a design error in the containment purge system valves. Normally, the containment purge system would be used during plant heatup to vent the pressure due to expanding air inside the containment vessel. Due to design deficiencies in the containment purge isolation valves concerning their ability to close under accident conditions and concerning the Safety Features Actuation System (SFAS) logic, the station has been committed to keeping the valves closed while in Modes 1 through 4. The increase in containment pressure during previous startups had been handled by venting containment pressure through the hydrogen dilution system exhaust, but
1 during this startup, the venting flow was not started soon enough. RCS pressure and temperature were slightly elevated during this startup for scheduled piping inspections and some mirror type insulation was removed. Such conditions may have slightly contributed to the containment pressure rise due to additional heating of the containment air.

Analysis of Occurrence: There was no danger to the health and safety of the public or to station personnel. The containment vessel design pressure is 36 psig and the containment vessel is pneumatically tested to 1.25 times design pressure (45 psig). The "high" differential pressure of 25.2 inches of water (0.91 psig) represents only 2.5% of design pressure. In the unlikely event of the worst loss of coolant accident for containment internal pressure, a 14.14 ft^2 hot leg split, the containment internal pressure could rise by 34.7 psig (updated Safety Analysis Report, Revision 0, 7/82). This pressure increase is based on the reactor being at 100% power, therefore, the pressure increases during a LOCA at the above conditions would probably be lower. Accordingly, adding the pressure which was reached on August 20, 1982, the maximum containment pressure would have been no more than 35.6 psig. Therefore, the structural integrity of the containment vessel would not have been challenged by any circumstances related to the subject of this report.

Corrective Action: The Plant Startup Procedure PP 1102.02 was modified
by T-6874 to include steps to open the hydrogen dilution system exhaust.
The modification was later incorporated as revision 15 to the procedure.
The efforts to design a modification to the Containment Purge and Exhuast
1 Valve operators and the SFAS isolation logic were judged prohibitive in
light of the limited possibility of regulatory approval for utilization
of the valves in modes 1 through 4. Prior to entering Mode 4, the control
power fuses for these valves are removed. This causes the valves to close,
and prevents their operation until the fuses are once again installed in
Mode 5.

Failure Data: Previous occurrences of the containment to shield building differential pressure exceeding the Technical Specification limit were reported in Licensee Event Reports NP-33-77-81 and NP-33-78-100 (78-083).

LER #82-039

TOLEDO EDISON

January 30, 1986

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File: RR 2 (NP-33-82-43)

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 82-039. The revisions to the report are indicated by a "1" in the left margin of each page.

Please destroy or mark superseded your previous copies of this report.

Yours truly,

Louis F. Storz

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

LFS/syc

Enclosure

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

Mr. Walt Rogers
DB-1 NRC Resident Inspector

IE22
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