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)	DOCKET NUMBER (2)	PAGE (3)
Davis-Besse Unit 1	0 5 0 0 0 3 4 6	1 OF 0 3

TITLE (4)
Secondary Side Pressurization of Steam Generator 1-1 Above Technical Specification Limits

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 9	0 6	8 5	8 5	0 1 7	0 0	1 0	0 4	8 5		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)	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	
James Michaelis, Operations Superintendent	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)

During auxiliary steam testing of Auxiliary Feed Pump 1-1, Steam Generator 1-1 was inadvertently pressurized to greater than 237 psig with secondary fluid temperature less than 110°F in violation of Technical Specification 3.7.2.1.

The pressurization was caused by the Auxiliary Feedwater Pump's discharge valve AF3870 being left partially open subjecting the Steam Generator to full pump discharge pressure of approximately 1000 psig. The Steam Generator was in a filled wet layup condition at the start of the test. The partially open discharge valve was found to be approximately three turns open even though the valve was verified as being in the closed position by two different operators. The cause of the event was personnel error on the part of the two operators in that their check of the valve was insufficient to verify position and the failure of Control Room operators and supervisors to monitor Steam Generator pressure.

As corrective action, all personnel involved were counseled. A procedure modification was made to the Safety Tagging Procedure AD 1803.00 to assure future valve position verifications are proper.

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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)

Description of Occurrence: Special testing of Auxiliary Feed Pumps, AFP, (BA), on auxiliary steam (SA), was scheduled for the day shift on September 6, 1985. On the previous shift, the test prerequisites were performed which included checking shut all four discharge valves on both AFPs. As an added precaution, it was decided to also tag these valves closed prior to actually starting the test. The tags were made out and placed on all four valves.

Testing was first performed on AFP 1-2 without any significant problems noted. When testing of the first phase of AFP 1-2 was complete, the Control Room was contacted to start the second phase of the test, which involved running both feed pumps to below full speed in parallel. As AFP 1-1 was brought up to speed, the Control Room received a Steam and Feedwater Rupture Control System, SFRCS, (JB), full trip alarm and a Steam Generator 1-1 low pressure alarm. Due to several other activities being performed on the SFRCS at the time, the real significance of these alarms was missed, and it was decided to try to duplicate the incident by bringing the pump speed down and then to increase the pump speed again. This attempt produced the same results in that the Control Room again received an SFRCS and Steam Generator 1-1 low pressure alarm, at which time the pump speed was again decreased.

Grounds on SFRCS in the past had caused similar type alarms. Control Room personnel along with Instrument & Control personnel investigated this possibility and repeated the pump speed change a third time, which once again produced the same alarms. At this time, a Reactor Operator arrived in the Control Room to relieve the watch. He noted a computer alarm on Steam Generator pressure which was then noted to be at approximately 800 psig and decreasing. The AFPs were shut down and the system depressurized.

A review of the data showed that on three occasions the steam generator was pressurized above 1000 psig with the highest being 1058.8 psig. Steam Generator shell temperature was about 101°F. The Technical Specifications limit pressure to 237 psig when temperatures are less than 110°F.

The event is reportable under 10CFR50.73(a)(2)(i)(B) as a condition prohibited by Technical Specification 3.7.2.1.

Designation of Apparent Cause of Occurrence: The cause of the pressurization was that the discharge valve (AF3870) from AFP 1-1 to Steam Generator 1-1 was not fully closed. It was later found to be about three turns off of its closed seat. The two operators who checked the valve position prior to starting the test did not apply sufficient force to the manual operator of this Limitorque operated valve to ensure full closure. Since the steam generator was in a wet layup condition (almost full) prior to the start of the AFP, it did not take too much additional water to fill the generator and main steam line up to the main steam isolation valve. Once it was solid, it took only a short time for the Steam Generator to see AFP discharge pressure. This valve verification error became a pressurization event because of the failure of Control Room operators and supervisors to monitor Steam Generator pressure.

Analysis of Occurrence: The secondary side pressurization of Steam Generator 1-1 was evaluated by the NSSS vendor, Babcock and Wilcox. Analysis shows no adverse effect on the Steam Generator. These results are not in conflict with the fact that the Technical Specification limit was exceeded. The limit is intended to ensure that the material toughness precludes brittle fracture. The limit is thus more of a guideline to ensure an acceptable factor of safety in lieu of specific analysis to justify other acceptable pressure and temperature conditions. Such analysis has been performed for the specific pressure and temperature conditions experienced during this incident, which were found to be acceptable.

Structural loading imposed on the main steam nozzles was considered in evaluating the water filled main steam line. Analysis ha been performed and concludes that the change in loadings as a result of the localized water loads had a negligible impact on the previously calculated stresses as documented in the Code Stress Report for Davis-Besse.

Corrective Action: The two operators involved with checking the AFP discharge valve closure, the Control Room Operator and the duty Shift Supervisor who were in charge of the test, and the on shift operators were formally counseled to more fully analyze abnormal plant events with all available indications. Specifically in this case, they should have checked Steam Generator pressure indication earlier in the event.

Modifications were made to the Safety Tagging Procedure AD 1803.00 to emphasize the need to physically check valves with sufficient force to assure the valve is in its proper position.

Meetings were conducted with all Operations personnel to discuss the event, findings, and the corrective actions taken. All operators will receive hands-on training to demonstrate the requirements for physically checking Limotorque valve operators prior to restart.

The Shift Supervisors and licensed personnel were instructed to limit future activities in the Control Room during critical initial stages of any special testing of plant equipment.

Failure Data: This is the first report of pressurization above the Technical Specification limit.

Report No: NP-33-85-23 DVR No(s): 85-143

TOLEDO EDISON

October 4, 1985

Log No. K85-1343
File: RR 2 (NP-33-85-23)

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

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

Gentlemen:

LER No. 85-017
Davis-Besse Nuclear Power Station Unit 1
Date of Occurrence: September 6, 1985

Enclosed is Licensee Event Report 85-017 which is being submitted in accordance with 10CFR50.73, to provide 30 day written notification of the subject occurrence.

Yours truly,

Louis F. Storz

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

LFS/ljk

Enclosure

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

Mr. Walt Rogers
DB-1 NRC Resident Inspector

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