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 4

TITLE (4)
Reactor Trip Due to Closure of Main Steam Isolation Valve

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 5 0 0 0
0 3	0 2	8 4	8 4	0 0 3	0 1	0 7	2 6	8 4		0 5 0 0 0

OPERATING MODE (9): 1
POWER LEVEL (10): 0 9 9

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)	XX 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)	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
Frank Swanger	AREA CODE 4 1 9 — 2 5 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
B	S B	R V	D 2 4 3	Y	X	B A	I S V	L 2 0 0	
X	J E	O B	C 5 6 0						

1

SUPPLEMENTAL REPORT EXPECTED (14)

YES (If yes, complete EXPECTED SUBMISSION DATE) — XX NO

EXPECTED SUBMISSION DATE (15): MONTH | DAY | YEAR

ABSTRACT (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines) (16)

On March 2, 1984, Davis-Besse Nuclear Power Station was operating at 99% of rated power. At 1221 hours, the #2 Main Steam Isolation Valve went closed, isolating the steam side of Steam Generator #2. This was caused by an undetected failed relay in one safety instrumentation channel plus routine plant testing involving a second safety instrumentation channel. The closure of Main Steam Isolation Valve #2 caused feedwater and Reactor Coolant System temperature transients that led to a reactor trip on high flux. After the trip, one of the Main Steam Safety Valves did not fully close. This caused an excessive Reactor Coolant System cooldown rate and by procedure Steam Generator #2 was allowed to boil dry. After the failed Main Steam Safety Valve had been replaced, while attempting to restore level in Steam Generator #2, the auxiliary feedwater valve to Steam Generator #2 failed to open. It was opened manually to restore level in Steam Generator #2. It was discovered later that a Main Steam Safety Valve on Steam Generator #2 had failed to lift. The failed relay circuit was repaired, the safety valve that failed to close properly was replaced. The safety valve that failed to lift has been gagged and will be repaired in the future. The auxiliary feedwater valve that failed to open had its torque switch settings changed. Analyses have shown that no design parameters were exceeded on the Reactor Coolant System or the steam generator.

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

Description of Occurrence: On March 2, 1984 at 1220 hours, 103 Effective Full Power
Days, EFPD, into Cycle 4, Davis-Besse 1 was operating at approximately 99% of full
power. The plant was in full automatic control. During periodic Steam and Feed-
1 | water Rupture Control System, SFRCS, (JE), surveillance testing, a previously
undetected SFRCS channel failure resulted in closure of the Loop 2 Main Steam
1 | Isolation Valve, MSIV, (ISV). This caused an increase in feedwater to the other
steam generator (SG) which overcooled that side of the reactor. Flux increased
because of the negative moderator coefficient. The reactor tripped on high flux
approximately 13 seconds after MSIV #2 closed.

Following the reactor trip, steam pressure on SG #2 did not stabilize as would
normally be expected. It was determined through local observation that Main Steam
1 | Safety Valve, MSSV, (RV), SP17A4, with a set pressure of 1070 psig, had not fully
closed on the #2 steam line. Manual actuation of SFRCS isolated SG #2. After
1 | auxiliary feedwater, AFW, (BA), isolation, SG #2 boiled dry and depressurized to
atmospheric pressure in approximately five minutes. This depressurization caused
the Reactor Coolant System (RCS) to exceed the normal cooldown limits for a short
time. Plant cooldown was conducted with SG #1. At approximately 340°F RCS tempera-
ture, the failed MSSV was replaced. When operators attempted to restore level in
SG #2, the AFW valve (AF599) failed to open. It was opened manually, and SG #2
level was restored to operable status. The plant was then cooled down to cold
shutdown.

It was determined later that an additional MSSV (SP17A1) on the #2 steam line had failed to open when it should have.

This event is being reported under 10CFR50.73(a)(2)(iv).

Designation of Apparent Cause of Occurrence: The cause of the MSIV closure was a failed optical isolator in a relay driver card for a relay in SFRCS Channel 4. This failure, undetected and in conjunction with normal testing on another channel, resulted in a close signal to MSIV #2. During troubleshooting, a wiring anomaly was found in the circuitry for MSIV #2. This anomaly was the reason the failed relay driver card had not been detected. There was both an equipment failure and an installation/construction error associated with the MSIV closure.

The cause of the excessive cooldown rate of the RCS was equipment failure. The failed MSSV released steam from SG #2 at a rate causing excessive cooldown. Therefore, equipment failure of the MSSV was also the cause for SG #2 boiling dry.

The two MSSVs, SP17A1 and SP17A4, were both equipment failures. MSSV SP17A4 failed due to the failure of a cotter pin that secures the release nut in place at the top of the stem. The failure of the cotter pin allowed the release nut to spin down the stem while the valve was open. This nut contacted the manual lifting device and prevented the valve from closing. The cause for the failure of MSSV SP17A1 is unknown at this time. When tested subsequent to its failure to open, it lifted early and inconsistently. It has been gagged and will be repaired or replaced during the next refueling outage.

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)

The reason for the failure of the AFW valve, AF599, was attributed to the torque to the torque switch setting. After it had been opened manually, troubleshooting found no mechanical problems and the valve operated properly.

Analysis of Occurrence: There were two automatic actuations of safety systems. The Reactor Protection System (RPS) actuated to trip the reactor on high flux. The SFRCS actuated on low steam pressure in SG #2 and isolated the steam and feedwater systems to SG #2. The SFRCS was originally initiated manually on low SG level to isolate normal feedwater and align AFW to both SGs.

The availability of SG #1 was adequate for cooling down the RCS even with SG #2 dry. One SG is capable of removing the decay heat following a reactor trip.

The number of MSSVs on each steam line is more than adequate to control steam pressure following a trip. The failure of MSSV SP17A1 to lift did not pose a significant hazard.

If MSSV SP17A4 had been further open when it stuck, it would have produced an even higher cooldown rate.

There was no danger to the health or safety of the general public or station personnel as a result of any part of this event.

Corrective Action: Under Maintenance Work Order, MWO, 84-0816, MSSV SP17A4 was replaced. The cotter pins in all other MSSVs were replaced with stainless steel pins, and maintenance procedures were modified to ensure new pins are used after any maintenance or testing in the future.

MSSV SP17A1 has been declared inoperable and gagged. It will be repaired or replaced during the next refueling outage. The cause of its failure to lift is still unknown at this time. The RPS high flux trips must be set less than 99.69% of rated thermal power with this safety inoperable.

The cooldown rate exceeded Babcock & Wilcox guidelines, but was within Technical Specification limits. An evaluation of the transient has concluded that all primary pressure boundary components still meet all requirements of the ASME Boiler and Pressure Vessel Code, Section III. Plant Procedures PP 1102.10, Plant Shutdown and Cooldown, and EP 1202.24, Steam Supply System Ruptures, have been modified to incorporate lessons learned from this event to give better control of this type of cooldown.

The effects of the SG #2 boiling dry were analyzed and it was concluded that the transient was within SG design limits. The effects of high main steam flow from SG #1 when No. 2 MSIV closed was analyzed. It was concluded that some tubes (approximately 100) may have become unstable for a period of seconds. As corrective action, the suspect tubes will be eddy current tested during the next refueling outage.

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The AFW valve AF599 problem was investigated under MWO 84-0819. Under this work order, and Facility Change Request 84-039, the motor operator torque switch settings were changed from 1.5 open and close to 1.0 close and 1.5 open. This is to prevent the valve disc from being jammed into its seat. This change was also made to AF608, the AFW valve for SG #1.

The faulty relay driver board in SFRCS Channel 4 was replaced under MWO 84-0820. The wiring anomaly was corrected and verified not to exist in the circuitry for MSIV #1. This work was done under MWO 84-0827.

Failure Data: No other reactor trips and MSSV failures due to this type of event have occurred at Davis-Besse.

Report No: NP-33-84-03

DVR No(s): 84-027 thru 84-032

TOLEDO EDISON

July 26, 1984

Log No. K84-1022
File: RR-2 (NP-33-84-03)

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 84-003. The revisions to the report are indicated by a "1" in the left margin of each page.

Please destroy your previous copies of this report and replace with the attached revision.

Yours truly,

Terry D Murray

Terry D. Murray
Station Superintendent
Davis-Besse Nuclear Power Station

TDM/ljk

Enclosure

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

Mr. Walt Rogers
DB-1 NRC Resident Inspector

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