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 2 2

TITLE (4)
Reactor Trip and Loss of Feedwater Event at Davis-Besse on June 9, 1985

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 6	0 9	8 5	8 5	— 0 1 3 —	0 0	0 7	0 9	8 5		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)	X 50.73(a)(2)(iv)	73.71(b)
20.405(a)(1)(i)	50.36(c)(1)	X 50.73(a)(2)(v)	73.71(c)
20.405(a)(1)(ii)	50.36(c)(2)	X 50.73(a)(2)(vii)	X 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)	
Stan Batch	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
X	J K	S I C	G 0 8 4	Y	B	B A	I S V	L 2 0 0	Y
X	J B	L T	R 3 6 9	Y	X	A B	R V	C 7 1 1	Y

SUPPLEMENTAL REPORT EXPECTED (14)

X YES (If yes, complete EXPECTED SUBMISSION DATE) NO

EXPECTED SUBMISSION DATE (15): MONTH 0 8 DAY 3 1 YEAR 8 5

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

While operating at 90 percent of full power, the No. 1 Main Feed Pump Turbine increased in speed due to a control failure and tripped on overspeed. An automatic plant runback was initiated. A reactor trip occurred at 80 percent of full power on high Reactor Coolant System pressure. Within 8 seconds of the reactor trip, both Main Steam Isolation Valves closed from a spurious Steam and Feedwater Rupture Control System actuation isolating steam to the No. 2 Main Feed Pump Turbine. Steam generator water levels decreased to the low steam generator level trip setpoint of the Steam and Feedwater Rupture Control System at approximately 5½ minutes after the reactor trip. The Auxiliary Feedwater System was actuated, but the Auxiliary Feedwater Pumps tripped on overspeed. Reactor Coolant System temperature and pressure increased due to the loss of heat transfer. The pressurizer Power Operated Relief Valve actuated three times and did not reseat at the proper Reactor Coolant System pressure after the third actuation. Operators placed the Startup Feed Pump in operation and locally restored both Auxiliary Feed Pumps to service. Adequate subcooled margin was available throughout the transient. The quench tank contained the discharges from the Power Operated Relief Valve.

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NRC Form 366B (9-83)

U.S. NUCLEAR REGULATORY COMMISSION
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EXPIRES 8/31/85

LICENSEE EVENT REPORT (LER) FAILURE CONTINUATION

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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
X	I D	F A N	X 9 9 9	N					
X	J I	F C V	F 1 2 7	Y					

NRC FORM 366B (9-83)

Description of Occurrence: Davis-Besse Unit 1 was operating at 90 percent of full power with the No. 1 Main Feedwater Pump, MFP, operating in automatic and the No. 2 MFP in manual control. This configuration was established to limit the susceptibility of the No. 2 MFP to control problems which had previously occurred. The control problems occurred only after a reactor trip and appeared to be connected to the automatic mode of operation. This configuration, therefore, permitted automatic feedwater control during operation and offered improved availability of at least one MFP in the event of a reactor trip.

At 0135 hours, the No. 1 MFP tripped on overspeed due to an unrelated control problem. The Control Room operators increased the No. 2 MFP speed, but it did not have adequate capacity, for the existing reactor power. The reactor tripped on high Reactor Coolant System, RCS, pressure at 0135:30 hours, tripping the turbine. Reactor power was at approximately 80 percent of full power at the time of the trip.

Immediately following the trip, a spurious Steam and Feedwater Rupture Control System, SFRCS, low steam generator level full trip occurred on Channel 2, an SFRCS full trip alarm was received, and both main steam isolation valves, MSIVs, closed. An actual low steam generator level did not exist at this time. This spurious trip resulted in a partial actuation of the SFRCS components since only the MSIVs actuated. When the MSIVs closed, the main steam supply was isolated to the MFPs. The No. 2 MFP continued to supply feedwater until approximately 0140 hours at which time its discharge pressure was not high enough to supply feedwater to the steam generators. The level in the steam generators which was being maintained at the low level limit setpoint (35 inches) began to decrease. SFRCS Actuation Channel No. 1 then automatically initiated on low steam generator level, starting the No. 1 Auxiliary Feedwater Pump, AFP, to feed the No. 1 Steam Generator (see Attachment 1 for a diagram of SFRCS actuated components).

At 0141:08 hours, a Control Room operator attempted to manually initiate the SFRCS, however, he incorrectly actuated the SFRCS on low steam pressure instead of the desired low steam generator level. Therefore, each SFRCS actuation channel sensed that its respective steam generator was depressurized. SFRCS Actuation Channel No. 1 then attempted to align AFP No. 1 to feed Steam Generator No. 2. SFRCS Actuation Channel No. 2 attempted to align AFP No. 2 to feed Steam Generator No. 1. Both actuation channels closed their respective Auxiliary Feedwater Containment Isolation Valves (AF599, AF608), which prevented any auxiliary feedwater flow from reaching the steam generators. At 0141:31 hours, AFP No. 1 tripped on overspeed. At 0141:44 hours, AFP No. 2 tripped on overspeed.

At 0142:00 hours, an operator recognized the manual initiation error and reset the low pressure SFRCS buttons, and pushed the low steam generator level SFRCS manual actuation buttons. Since both SFRCS actuation channels were already tripped on low steam generator level, the SFRCS automatically began to realign the AFPs when the low pressure buttons were reset. However, the Auxiliary Feedwater Containment Isolation Valves (AF599, AF608) did not automatically open. The operators attempted to open these valves from the Control Room by operating their control switches and by reinitializing the SFRCS. These attempts failed to open the valves. Equipment

Operators were sent to open these valves locally, and when the valves were moved off their closed seats utilizing the manual handwheels, the motor operator responded and fully opened the valves. During this period, attempts were also being made to restart the AFPs and preparations were underway to start the motor operated Startup Feedwater Pump.

The RCS average temperature was increasing due to the lack of primary to secondary heat transfer. RCS pressure was increasing due to the decreasing density of the RCS water and increasing pressurizer level. RCS pressure increased to the Power Operated Relief Valve, PORV, setpoint (2425 psig). The PORV cycled a total of three times, relieving pressurizer pressure to the Quench Tank. Following the third opening, the PORV failed to reclose at the proper RCS pressure. The Control Room operator observed the primary plant conditions and closed the block valve on the PORV. RCS pressure was at approximately 2075 psig when the block valve closed. The Quench Tank contained the discharges from the PORV.

At approximately 0151 hours, the operators placed the Startup Feedwater Pump in operation to supply the steam generators. Steam Generator No. 1 pressure had decreased to approximately 750 psig. Steam Generator No. 1 repressurized to approximately 900 psig from the Startup Feedwater Pump. Steam Generator No. 2 had decreased to 920 psig. At 0152 hours, the No. 2 AFP was returned to service by the operators locally. Maximum RCS temperature had reached approximately 592 degrees Fahrenheit. At 0155 hours, the No. 1 AFP was returned to service locally by the operators. Control of the AFP turbines was maintained locally by an operator at the turbine trip throttle valve. At 0158 hours, RCS average temperature was restored to the normal post trip temperature. The cooldown of the RCS lowered RCS pressure to a minimum of approximately 1720 psig. Operators manually started the No. 1 High Pressure Injection, HPI, Pump in the piggyback mode (Decay Heat Pump No. 1 supplying the suction to the HPI Pump No. 1) in precautionary anticipation of the rapid cooldown. Only a slight amount of water (less than 50 gallons) needed to be injected.

Several other equipment malfunctions occurred which did not affect the physical plant response. One source range nuclear instrumentation, NI, channel was inoperable prior to the trip. The remaining source range NI channel failed to indicate properly when it was automatically energized after the trip. The display units for the Safety Parameter Display System, SPDS, were inoperable in the Control Room at the time of the trip. At 0158:40 hours, the suction of the No. 1 AFP automatically transferred from the Condensate Storage Tank, CST, to the Service Water System. The operator manually realigned the pump suction back to the CST. No significant amount of service water was added to the steam generator during the recovery from the transient. It was noticed that the pneumatic operator on one main turbine bypass valve was damaged, preventing the valve from being opened. This did not affect the post transient response of the plant.

Additional details of the plant transient and corrective actions will be provided in the restart report response to the Region III Confirmatory Action Letter (85-06). Attachment 2 provides a chronological listing of the event. This report is being submitted in compliance with paragraph 50.73(a)(2)(i), 50.73(a)(2)(iv), 50.73(a)(2)(v),

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

50.73(a)(2)(vi), and 50.73(a)(2)(vii). This report also satisfies the reporting requirements for a Emergency Core Cooling System Actuation Special Report, Section 6.9.2(a) of Technical Specifications. This was the fourth high pressure injection actuation cycle to date.

Designation of Apparent Cause of Occurrence: This transient was initiated when the No. 1 MFP developed control problems and tripped on overspeed. The plant tripped on high RCS pressure due to inadequate feedwater being supplied from the No. 2 MFP during the plant runback. The cause of the MFP overspeed tripping was determined to be due to a bad speed summation and valve lift reference circuit board card in the MFP control. A frequency to voltage converter chip had failed. The board is being returned to General Electric for further analysis on the root cause of the failure.

The root cause of the MSIV closure has not yet been determined. It is presently believed that the MSIVs properly responded to a momentary low level SFRCS trip. Further investigations will follow once an action plan is completed.

The cause of the SFRCS spurious trip on low steam generator level has not yet been positively determined. Troubleshooting will begin in accordance with the action plan. However, it is presently believed that the steam generator level sensing channels are sensing an extremely rapid secondary side pressure transient that occurs in the steam generator following the turbine stop valve closure on a turbine trip. These level transmitters share a common set of sensing lines with transmitters which were replaced during the 1984 Refueling Outage. Prior to the 1984 Refueling Outage, Bailey BY level transmitters were installed which have now been replaced by Rosemont Model 1153. Since these Rosemont transmitters have no significant displacement required for operation, while the Bailey BYs required a volume displacement to operate the bellows, it is postulated that the responsiveness of the sensing line and transmitter arrangement has been greatly increased by this change. This increased responsiveness allowed the SFRCS to sense the rapid secondary side pressure transients which previously were undetected. Further analysis of this condition is underway.

The cause of the incorrect manual SFRCS initiation was personnel error attributed to a poor switch layout. These SFRCS manual initiation pushbuttons had been identified in the Detailed Control Room Design Review as one of the principal items needing human engineering improvements. There are two adjacent vertical columns of buttons with five buttons in each column (see Attachment 3 for arrangement details). Each column represents one SFRCS actuation channel. To manually initiate both channels of the SFRCS for steam generator low level, the operator should have depressed the fourth button from the top in each column; instead, the two top buttons were depressed. A design change had been developed prior to this event to improve the switch layout and will be implemented during this outage.

The cause of the AFPs tripping on overspeed after initiation has not yet been positively determined. Water flashing through the nozzles of the AFP turbines is thought to be a contributor. The governor was inspected on both AFP turbines, and no contributing factors to the overspeed were seen. Further investigations and testing are planned.

The cause of the Auxiliary Feedwater Valves (AF599 and AF608) not opening by the motor operator was determined to be a combination of a high differential pressure and an improperly set torque switch bypass limit switch. With this torque switch bypass limit switch improperly set, the motor operator was allowed to torque out during the opening stroke. These valves are open during normal operation and were closed by the incorrect manual initiation of the SFRCS. If the AFPs had been operating at the time the valves attempted to open, the differential pressure across the valves would have been significantly lower, and the valves should have opened to allow the auxiliary feedwater flow to occur. These valves were stroked following the transient and ability of the valves to open (without significant differential pressure) was verified. Recent testing also verified that the valve operators torque out under high differential pressure with the improperly set torque switch bypass limit switch. Further investigations are in progress.

The cause of the control problems with the AFPs after the overspeed was reset is presently attributed to the difficulty in opening the trip throttle valves. No mechanical deficiencies were found while investigating the resetting of the overspeed trip device/linkage. Further investigations are in progress.

The cause of the PORV not properly reseating has not yet been positively identified. Operator observations at the time of the transient indicate that the electronic controls signal was calling for the valve to reclose. A visual inspection and disassembly of the PORV failed to identify the cause. Further investigations are in progress.

The two, independent SPDS display units were inoperable due to separate but similar failures in the data transmission system between the Control Room terminals and their respective processors. The failures are of an intermittent nature and the exact cause is still under investigation.

The cause of the source range NIs inoperability has not been positively identified. The failure of the source range NIs has been a repetitive problem at Davis-Besse with repeated investigations failing to determine the root cause. Since 1977, the boron trifluoride detectors, preamp, and cable in Containment have been replaced, along with the modules in the Reactor Protection System and a reworking of the grounding on the preamp and count rate amplifier module connections. No positive effect on the total elimination of the spiking, nor the erroneous/elevated count rate has occurred from these corrective actions. Further review is being performed on the possibility of ground loops, induced current or voltage from adjacent cables, or intermittent problems with the count rate amplifier module.

The cause of the Turbine Bypass Valve 2-2 damage has not been identified. The valve was disassembled and the actuator stem extension piece was found bent, four parts were missing, and the valve internals were found loose. Several valve parts were shipped to the vendor for further analysis. Further review of the turbine bypass valve failure is underway.

The cause of the inadvertent AFP No. 1 suction supply transfer from the Condensate Storage Tank to Service Water has not yet been determined. Testing and other investigations are currently being performed.

The cause of MS-106 apparently cycling in about one third of the expected stroked time is still under investigation.

Analysis of Occurrence: This event involved a temporary loss of feedwater to the steam generators. This event was bounded by the analyses previously performed (see Toledo Edison submittals to NRC Serial No. 506 dated May 22, 1979, and Serial No. 517 dated June 15, 1979), which analyzed a loss of all feedwater for 30 minutes following a reactor trip. These analyses showed that as long as either:

1) Auxiliary Feedwater is restored within 30 minutes of the loss of main feedwater,

OR

2) Within 30 minutes, at least one makeup pump and the PORV are available for primary cooling (feed and bleed) and the Startup Feedwater Pump is available to supply a steam generator,

fuel cladding temperatures would remain within a few degrees of saturated fluid temperature and no cladding rupture or metal water reaction would occur.

Operator interviews indicated that the shift was fully aware of the core status and were prepared to implement the "feed and bleed" core cooling method if the auxiliary feedwater was not restored. The Startup Feedwater Pump was available throughout the event and in fact was placed in service within ten minutes of the tripping of the AFPs. Auxiliary Feedwater was restored within 12 minutes of the loss of feedwater. These response times and equipment availability are well within the loss of feedwater analyses.

At no time during the event was the required subcooled margin (20 degrees Fahrenheit) lost. The reactor coolant pumps continued to operate throughout the event. The primary code safety valves were not challenged and at no time during the event did the RCS pressure or temperature exceed the allowable values. The maximum temperature reached was below the normal operating temperature for the hot leg temperature. There is no indication of any fuel cladding degradation based on the reactor coolant radiochemistry analysis.

An analysis has been performed by Babcock & Wilcox to determine if the transient adversely affected the steam generators. Conditions and components specifically analyzed include: (1) Main Feedwater Nozzles, (2) Auxiliary Feedwater Nozzles, (3) Steam Generator Tubes, (4) Tube to Shell Delta T's, and (5) Lower Tubesheets

The results show that the transient had no adverse structural effect on the steam generators.

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

Corrective Action: The failed circuit board will be replaced in the No. 1 MFP. As a precautionary measure, the No. 2 MFP speed control circuit will also be inspected for a similar failure.

The corrective action on the MSIV closure has not yet been determined since trouble-shooting has not yet begun.

The corrective actions for the SFRCS spurious trip on low steam generator level have not yet been determined since troubleshooting has not yet begun. The proper method of manual actuation of the SFRCS buttons will be reviewed with all licensed operators. The switch layout is being modified to add additional demarkation of the actuation buttons, and to add actuation guards over the switches (see Attachment 3).

The corrective actions to be taken to prevent the AFP trip on overspeed have not yet been determined.

The torque switch bypass limit switch will be reset on the Auxiliary Feedwater Valves AF599 and AF608. Maintenance personnel will receive additional instruction, and the procedure for setting the motor operator valve limit switches will receive additional clarification. Other nuclear safety related motor operated valves at Davis-Besse will be evaluated.

The corrective actions to correct the control problems with the AFPs after the overspeed was reset have not been identified.

Corrective actions to be taken on the PORV have not yet been identified.

Corrective actions for the repair of the data transmission systems affecting the SPDS Control Room displays have not yet been identified.

The corrective actions for repair of the source range NIs have not yet been determined.

The pneumatic actuator for Turbine Bypass Valve 2-2 will be replaced. Additional corrective actions may be necessary after further investigation to determine the root cause of the failed valve actuator.

A tabulation of the causes and corrective actions determined to date is summarized in Attachment 4.

Corrective action details for the No. 1 AFP suction supply transfer from the Condensate Storage Tank to Service Water has not yet been identified.

Corrective action details for MS-106 have not yet been identified. Further investigation is in progress.

Failure Data: This is the first occurrence at Davis-Besse of a loss of both main and auxiliary feedwater.

This is the first failure that has occurred at Davis-Besse on the MFP turbine electronic controllers which has caused an overspeed tripping of the pumps. A new electronic control system for main feedwater pumps was installed during the 1984 Refueling Outage.

Spurious closures of the MSIVs have occurred previously at Davis-Besse before time delays were added to the steam to feedwater pressure differential trip circuitry.

An SFRCS spurious half trip on low steam generator level has occurred on two previous trips since the 1984 Refueling Outage. Spurious trips on low steam generator level have not occurred prior to the 1984 Refueling Outage.

Incorrect manual initiation of the SFRCS has not previously occurred at Davis-Besse.

The AFPs tripping on overspeed after initiation has not previously occurred at Davis-Besse.

The Auxiliary Feedwater Valves AF599 and AF608 are normally open. One previous occurrence of one of these valves not opening with high differential pressure occurred after the March 2, 1984 reactor trip.

The operators do not normally attempt to control the AFP turbines locally. Problems with controlling these pumps do not appear to have been repetitive, however, some problems have been experienced previously with proper resetting of the trip throttle valve.

The PORV has not been challenged since the pressure setpoint was raised in 1979. Prior to 1979, several deficiencies were noted in the valve operation. In September 1977, the valve stuck in the open position, causing an overpressurization of the quench tank.

The diversity of the SPDS display sources (Ramtek and Chromatics display devices) has normally allowed at least one SPDS display to remain operable. The failure rate of these units is higher than is acceptable. Efforts are underway to increase the system reliability.

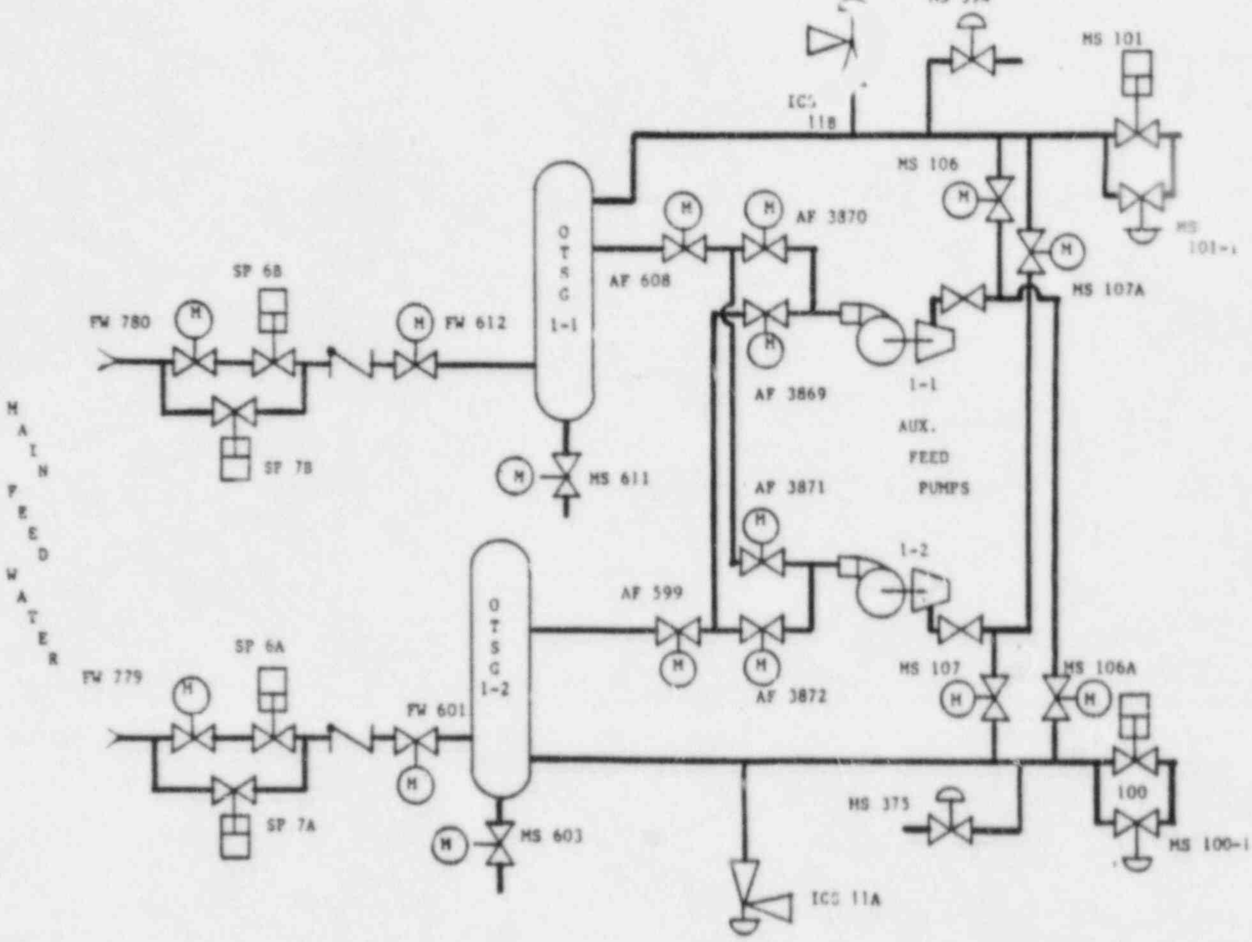
The failures of the source range NIs have been a repetitive occurrence at Davis-Besse even though exhaustive evaluations and corrective actions have been taken.

The damaged pneumatic operator on the turbine bypass valve has not previously occurred at Davis-Besse.

There have been several cases where the AFP suction inadvertently transferred from the Condensate Storage Tank to the Service Water supply.

Report No: NP-33-85-18 DVR No(s): 85-088

LER 85-013
SFRCS ACTUATED COMPONENTS



VALVE DESCRIPTIONS

FW779, FW780	MAIN FEEDWATER BLOCK VALVES
SP7B, SP7A	STARTUP FEEDWATER CONTROL VALVES
SP6B, SP6A	MAIN FEEDWATER CONTROL VALVES
FW612, FW601	MAIN FEEDWATER STOP VALVES
ICS11A, ICS11B	ATMOSPHERIC VENT VALVES
MS394, MS375	MAIN STEAM LINE DRAINS
MS101, MS100	MAIN STEAM ISOLATION VALVES
MS101-1, MS100-1	MAIN STEAM ISOLATION VALVE BYPASSES
AF608, AF599	AUXILIARY FEEDWATER STOP VALVES
AF3869, AF3870	AUXILIARY FEED PUMP 1 DISCHARGE VALVES
AF3871, AF3872	AUXILIARY FEED PUMP 2 DISCHARGE VALVES
MS106, MS106A	AUXILIARY FEED PUMP 1 STEAM INLET VALVES
MS107, MS107A	AUXILIARY FEED PUMP 2 STEAM INLET VALVES
MS603, MS611	STEAM GENERATOR DRAIN VALVES

LER 85-013
SEQUENCE OF EVENTS

* = Unexpected or off-normal response

Initial Conditions

Unit operating at 90%
No. 1 Main Feed Pump (MFP) in automatic control
No. 2 Main Feed Pump in manual
One Source Range Nuclear Instrumentation Channel inoperable
Safety Parameter Display System (SPDS) inoperable

Time	Event
*01:35:00	#1 MFP trips MFP flow increases; MFP turbine trips on overspeed
01:35:01	Unit runback toward 55% at 50% per minute initiated
01:35:21	Operator increases the speed of #2 MFP turbine. Pressurizer spray valve manually opened to 100%.
01:35:30	Reactor trip and turbine trip - RCS high pressure (2300 psig) from 80% power.
*01:35:31	Computer recorded Steam and Feedwater Rupture Control System (SFRCS), trip on Steam Generator (SG) low level, Actuation Channel 2.
*01:35:31	Both Main Steam Isolation Valves (MSIVs) start to close.
01:35:34	SFRCS actuation signal clears.
*01:35:36	MSIV #2 has closed.
*01:35:37	MSIV #1 has closed. With both MSIVs closed, the source of steam for #2 MFP turbine is isolated. Steam from main steam piping and moisture separator reheaters continued to drive #2 MFP for a while.
01:35:45	Pressurizer spray valve closed.
01:35:56	Once Through Steam Generator (OTSG) levels at normal post-trip level (35 inches).
*01:40:00	OTSG levels begin to fall from the normal post-trip level.
01:41:04	SFRCS low OTSG level (26.5 inches) Actuation Channel 1 actuates; this actuation caused Auxiliary Feedwater Pump (AFP) #1 to be aligned to feed OTSG #1.

LER 85-013
SEQUENCE OF EVENTS

Time	Event
*01:41:08	The Control Room Operator attempted to manually initiate SFRCS; however, he incorrectly actuated the SFRCS on low steam pressure instead of the desired low steam generator level. He performed the manual actuation by depressing the top switch in both strings of manual actuation switches for the respective SFRCS actuation channels. Therefore, each SFRCS actuation channel sensed that its respective steam generator was inoperable. SFRCS Actuation Channel 1 then attempted to align AFP #1 to feed SG #2 and SFRCS Actuation Channel #2 attempted to align AFP #2 to feed SG #1; both actuation channels, however, closed their respective steam generator containment isolation valves (AF599, AF608), which prevented any auxiliary feed flow from reaching the steam generators.
	Per the SFRCS design, valves positioned by the low level trip on SFRCS Channel #1 were repositioned by the higher priority pressure trip. The AFP #1 steam supply valve from OTSG #1, MS-106, had started open in response to the SFRCS Actuation Channel #1 low level trip. Following the manual initiation of the low pressure trip, the valve should have continued opening to its full open position before it cycled closed. The entire open/close stroke time should have been about 50-60 seconds. *MS-106, however, returned to its closed position in about 18 seconds. This indicates that the open command to the valve did not seal in.
01:41:13	SFRCS Actuation Channel #2 tripped on low steam generator level. Since the low pressure trip already present had priority, no change in component actuation occurred.
*01:41:31	AFP #1 tripped on overspeed.
*01:41:44	AFP #2 tripped on overspeed.
01:42:00	Manual reset of SFRCS low OTSG pressure actuation.
	*AF-599, AF-608 should reopen automatically, but did not.
	*An attempt was made to reopen AF-599 and AF-608 from the main control panel, but the valves did not respond.
*01:43:39	Upon energization, the remaining source range nuclear instrumentation channel failed. All control rods had been verified as fully inserted; emergency boration was initiated manually.

LER 85-013
SEQUENCE OF EVENTS

Time	Event
01:43:55	Assistant Shift Supervisor went to SFRCS cabinets (behind the Control Room area), opened the doors, and operated the SFRCS "Initial Reset and Bypass" function in an attempt to reset any automatic safety signals to AF-599 and AF-608. *The valves remained closed.
*01:44 - 01:52	Equipment Operators were dispatched into the plant to operate the following equipment: (1) Two Equipment Operators were sent to the AFP turbines to manually restore the AFPs to service. No. 2 AFP turbine overspeed trip was reset at 01:45:50 hours. Manual control of the turbine trip throttle valve was required to bring the turbine up to speed. No. 1 AFP turbine was reset and speed was controlled locally throughout the recovery. (2) The Assistant Shift Supervisor left the Control Room to place the Startup Feed Pump in service. This evolution required opening the pump suction valve, the pump discharge valve, and two cooling water valves. In addition, the control fuses for the pump motor circuit breaker were required to be installed. The Startup Feed Pump was started at 01:51:23 hours. (3) Two Equipment Operators were sent to open OTSG Auxiliary Feedwater Isolation Valves AF-599 and AF-608. These valves are the containment isolations for the Auxiliary Feedwater System. The operators moved the valves from the closed position, and the motor operators opened the valves. Computer printouts indicate that the #2 OTSG Valve (AF-599) was open at 01:47:48 hours, and the #1 OTSG Valve (AF-608) was open at 01:49:28 hours.
01:47:33	OTSG #1 below 960 psig and decreasing.
01:48:49	Pressurizer PORV opens (first time) at 2433 psig (2425 setpoint). Attachment 5 contains traces of RCS pressure and RCS temperature for the event.
*01:48:51	OTSG #2 below 960 psig and decreasing. (Both OTSGs now "dried out", according to criteria in plant emergency procedures.)
01:48:52	Pressurizer PORV has closed at 2377 psig (2375 setpoint).
01:50:09	Pressurizer PORV opens (second time) at 2434 psig.

LER 85-013
SEQUENCE OF EVENTS

Time	Event
01:50:12	Pressurizer PORV has closed at 2369 psig.
01:50:13	OTSG #1 Atmospheric Vent Valve opened.
01:51:17	OTSG SG #1 level falls below eight inches.
*01:51:18	Pressurizer PORV opens (third time) at 2435 psig; did not close.
01:51:30	Obtained flow from Startup Feed Pump to OTSG #1.
01:51:42	Operator started to close pressurizer PORV block valve at 2140 psig.
01:51:42	RCS Loop #1 reaches a minimum pressure of 2081 psig. Loop #1 T-hot = 588.6°F; Tave = 587.5°F.
01:51:49	Acoustic monitor indicates less than 20% flow through PORV/block valve.
01:52:33	Pressurizer spray valve closed.
01:53:00	RCS Loop #1 T-hot reaches peak value of 593.5°F.
01:53:22	AFP #2 has significant flow, with control locally via the trip throttle valve.
01:53:25	RCS Tave reaches peak value of 592.3°F.
01:53:35	OTSG #2 returns to above 960 psig.
01:53:56	PORV block valve reopened by operator.
01:54:45	OTSG #1 return to above 960 psig.
01:54:46	AFP #1 has significant flow, with control locally via the trip throttle valve.
01:56:58	OTSG #2 atmospheric vent open; SG #2 below 960 psig and decreasing.
01:57:05	SG #1 below 960 psig and decreasing.
*01:57:53	Low suction pressure developed on AFP #1; 34 seconds later, suction pressure was recovered.
01:58	Tave restored to normal post-trip temperature. The cooldown had lowered RCS pressure to about 1720 psig. Operators manually started the High Pressure Injection (HPI) Pump #1 in the piggyback mode (Low Pressure

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SEQUENCE OF EVENTS

Time	Event
	Injection Pump #1 supplying the suction to the HPI Pump #1). A slight amount of water (about 50 gallons) was injected.
01:58:08	RCS Loop #1 reaches a minimum pressure of 1716 psig. Loop #1 T-hot = 546.6°F; Tave = 546.2°F.
01:58:28	OTSG #1 atmospheric vent closed.
01:58:33	AFP #1 flow reduced to control OTSG level.
*01:58:40	AFP #1 suction automatically transferred from the Condensate Storage Tank (CST) to the Service Water System. The operator realigned to CST.
02:01	When AFP #2 was returned to service, it was initially controlled locally. The Control Room Operator later controlled the pump in manual rather than returning it to automatic.
02:01:13	AFP #2 flow reduced.
02:02:27	SG #1 returns to above 960 psig.
02:02:30	SG #2 returns to above 960 psig.
02:04	Plant conditions essentially stable.

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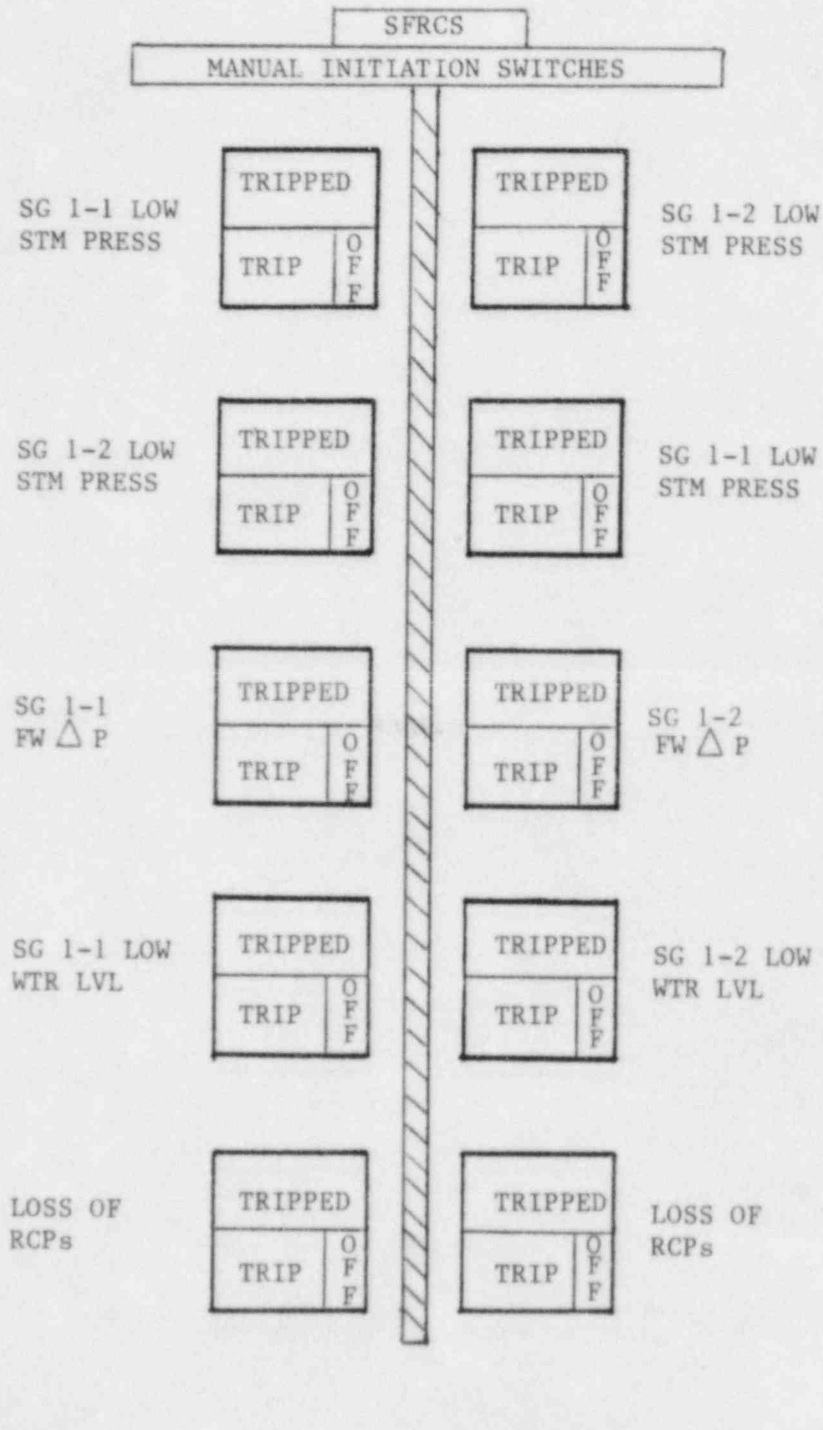
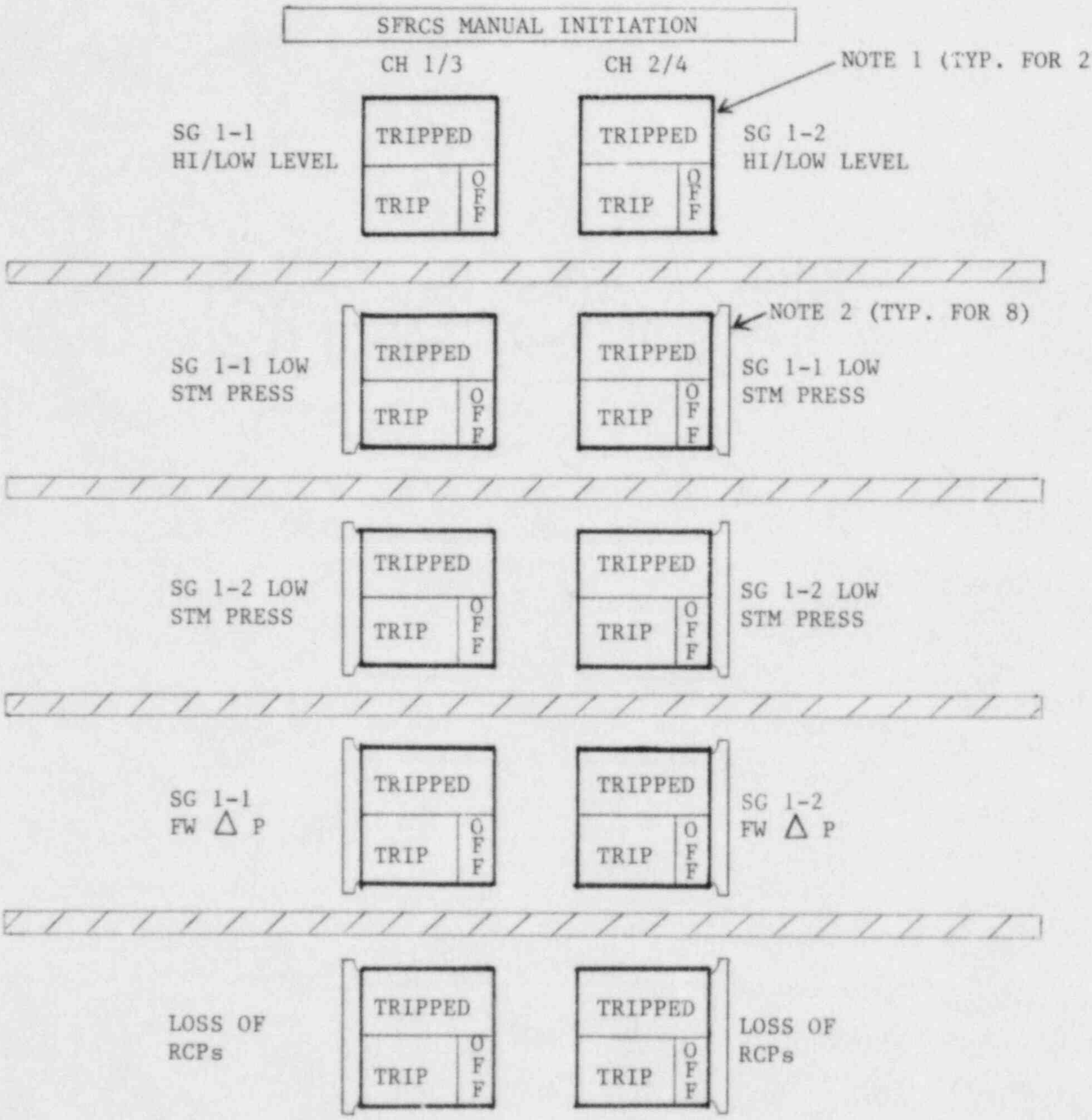
ORIGINAL LAYOUT



REVISED LAYOUT



NOTES:

1) Cutler Hammer Full Shroud (No Window) Cat #E30KT6 (Gray)
2) Cutler Hammer Guard With Clear Plastic Sliding Window, Cat #E30KR32 (Red)

TABULAR SUMMARY OF CAUSE AND CORRECTIVE ACTIONS

DEFICIENCY	CAUSE	CORRECTIVE ACTION
MFP No. 1 tripping	The MFP tripped on overspeed after a failed circuit board incorrectly increased the pump speed. The board will be sent to General Electric for further analysis.	The board will be replaced in the No. 1 MFP. As precautionary measure, the No. 2 MFP will also be inspected.
MSIV closure	The action plan has not yet been completed to allow troubleshooting to begin.	Not defined at this time.
SFRCS spurious trip on low steam generator level	The action plan has not yet been completed to allow troubleshooting to begin.	Not defined at this time.
Incorrect manual SFRCS initiation	Personnel error attributed to a poor switch layout contributed to this error.	The proper method of manual actuation will be reviewed with all licensed operators. The switch layout will be modified this outage.
AFPs tripping on overspeed after initiation	The cause of this deficiency has not yet been positively identified. The governor was inspected and no contributing factors to the overspeed were seen. Further investigations and testing will continue.	The corrective actions to be taken have not yet been determined.

TABULAR SUMMARY OF CAUSE AND CORRECTIVE ACTIONS

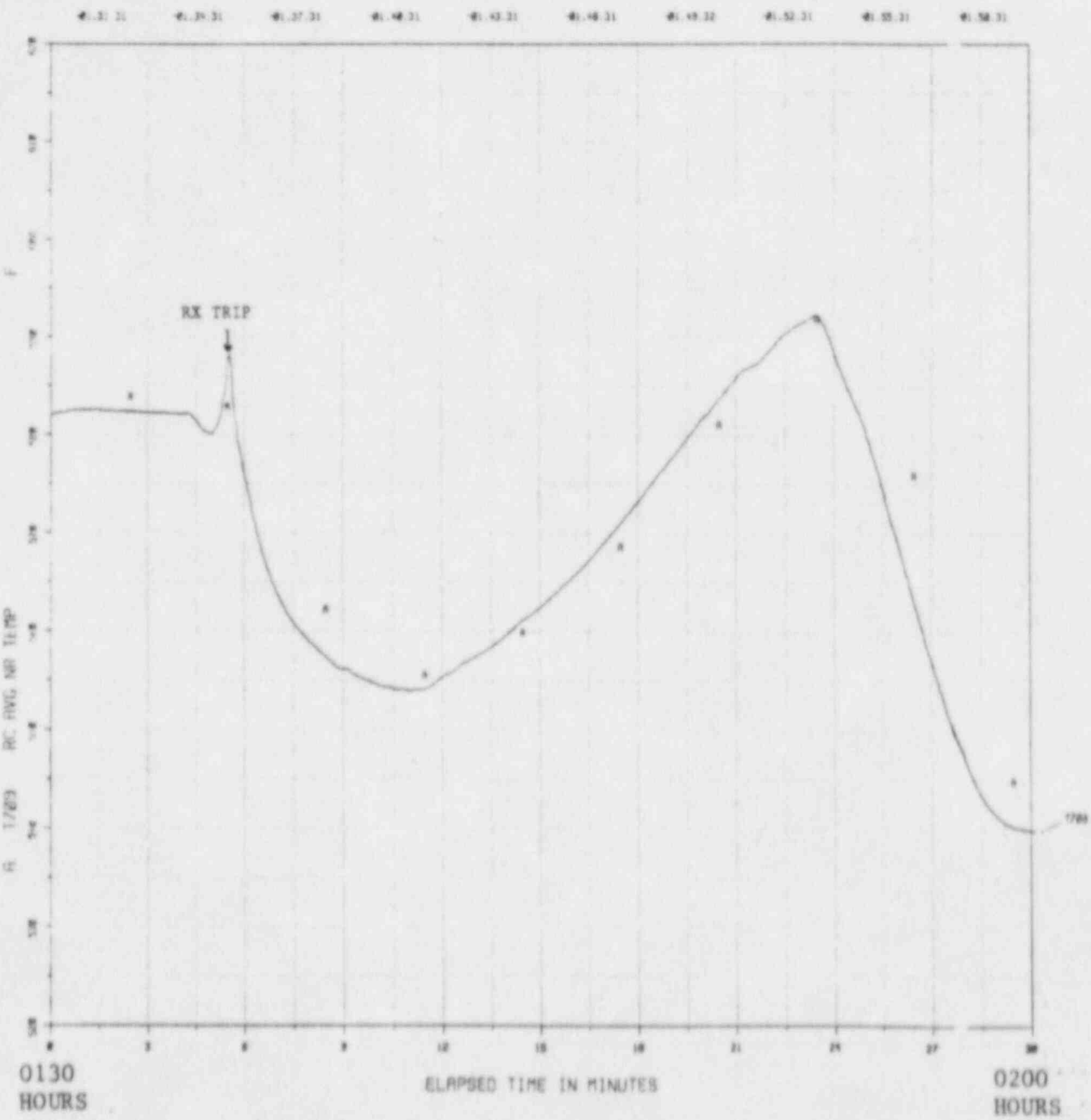
DEFICIENCY	CAUSE	CORRECTIVE ACTION
Auxiliary Feedwater Valves AF599 and AF608 not opened by motor	A combination of high differential pressure and an improperly set torque switch bypass caused the failures. Further investigations are in progress.	The torque switch bypass switch will be reset. Maintenance personnel will receive additional instruction, and the procedure for setting the motor operated valves will receive additional clarification. Other nuclear safety related motor operated valves at Davis-Besse will be evaluated.
Control problems with AFPs after overspeed reset	No mechanical deficiencies were found in the resetting of the overspeed trip device/linkage. Further investigations are still in progress.	Corrective actions to be taken are still under evaluation.
PORV not properly reseating	The PORV was disassembled and an inspection has not shown the cause of the problem. Investigative work is still in progress.	Corrective actions to be taken are still under evaluation.
SPDS inoperable prior to trip	The two, independent SPDS display units were inoperable due to separate but similar failures in the data transmission system between the Control Room terminals and their respective processors. The failures are of an intermittent nature and the exact cause is still under investigation.	Corrective actions for the repair of the data transmission systems affecting the SPDS Control Room displays have not yet been identified.

TABULAR SUMMARY OF CAUSE AND CORRECTIVE ACTIONS

DEFICIENCY	CAUSE	CORRECTIVE ACTION
Source Range NIs inoperable	Work is still proceeding on evaluating the cause of the failure of NI-1 and NI-2.	The corrective action details have not been identified for this item.
Turbine Bypass Valve 2-2 damaged	The valve was disassembled and it was found that the actuator piece was bent, four parts were missing, and the valve internals were loose. Several valve parts were shipped to the vendor for further analysis.	The corrective action details have not been identified for this item.
MS-106 cycled in about one third of expected stroke time.	Investigation is still in progress.	Corrective action has not yet been identified.
AFP No. 1 suction supply transfer from Condensate Storage Tank to Service Water.	Investigation is still in progress.	Corrective action has not yet been identified.

PLOT OF JUNE 9, 1985
TRANSIENT AT DAVIS-BESSE UNIT 1

REACTOR COOLANT AVERAGE TEMPERATURE
vs
TIME



ATTACHMENT 5
PAGE 1 OF 2

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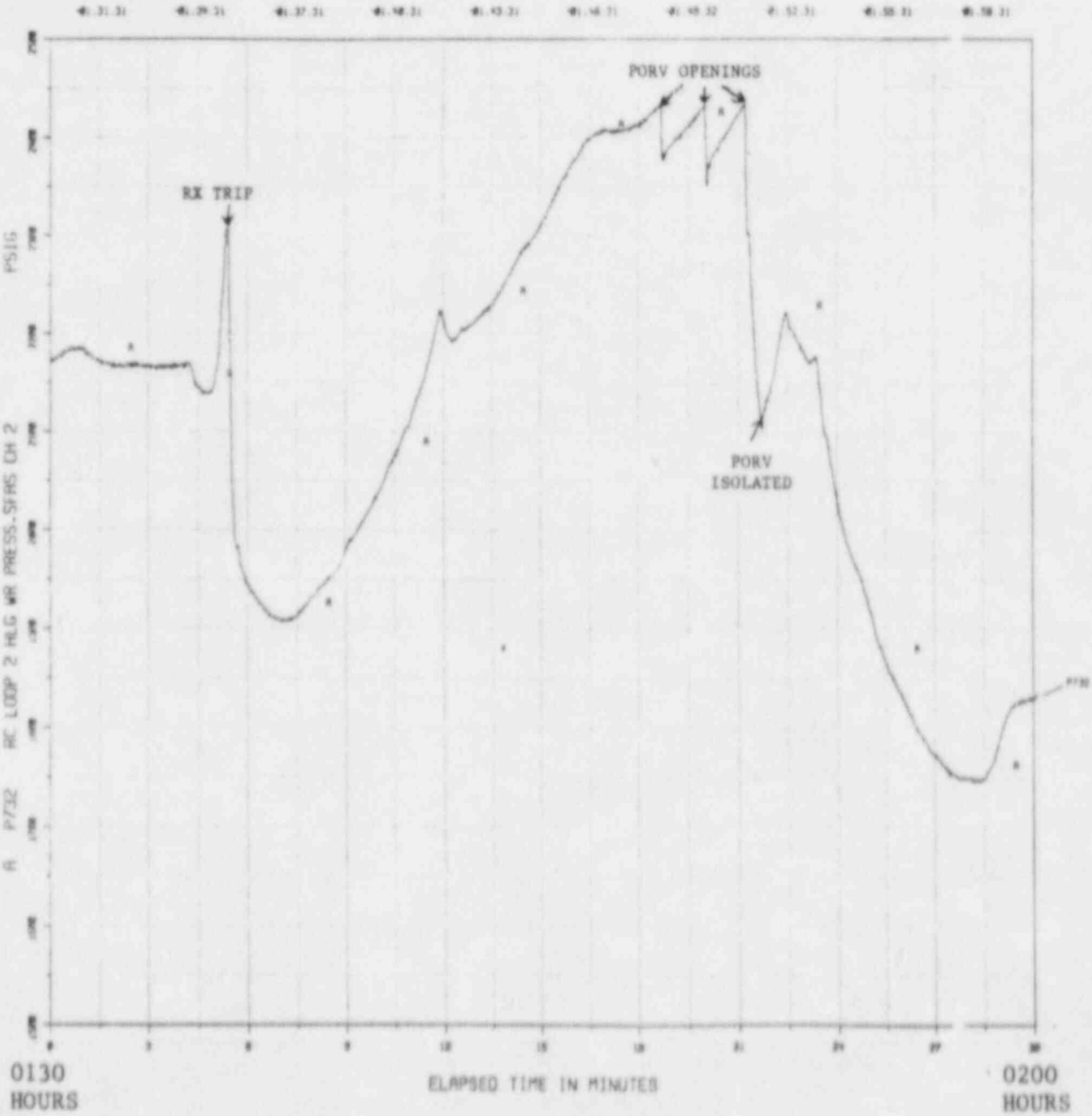
PLOT OF JUNE 9, 1985
TRANSIENT AT DAVIS-BESSE UNIT 1

REACTOR COOLANT PRESSURE
vs
TIME



TOLEDO EDISON

July 9, 1985

Log No. K85-1011
File: RR 2 (NP-33-85-18)

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

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

Gentlemen:

LER No. 85-013
Davis-Besse Nuclear Power Station Unit 1
Date of Occurrence: June 9, 1985

Enclosed is Licensee Event Report 85-013 which is being submitted in accordance with 10CFR50.73, to provide 30 day written notification of the subject occurrence.

Yours truly,

Stephen M. Quennoz

Stephen M. Quennoz
Plant Manager
Davis-Besse Nuclear Power Station

SMQ/ljk

Enclosure

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

Mr. Walt Rogers
DB-1 NRC Resident Inspector

IE22
1/1

JCS/001