SSINS No.: 6835
IN 85-50

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, D.C. 20555

July 8, 1985

IE INFORMATION NOTICE NO. 85-50: COMPLETE LOSS OF MAIN AND AUXILIARY FEEDWATER AT A PWR DESIGNED BY BABCOCK & WILCOX

ADDRESSEES:

All nuclear power facilities holding an operating license (OL) or construction permit (CP).

Purpose:

This information notice is being provided to inform licensees of a significant reactor operating event involving the loss of main and auxiliary feedwater at a pressurized water reactor. Information in this notice is preliminary and was obtained from the special NRC fact finding team which is investigating the event. A complete report of findings will form the basis for further communications or actions related to this event. The NRC expects that recipients will review this notice for applicability to their facilities. Suggestions contained in this notice do not constitute NRC requirements; therefore, no specific action or written response is required.

Description of Circumstances:

On June 9, 1985, the Davis-Besse plant was operating at 90% power with Main Feedwater Pump 2 in manual control because problems in automatic had been experienced. A control problem with Main Feedwater Pump 1 occurred, and it tripped on overspeed. Reactor runback at 50% per minute toward 55% power was automatically initiated. Nevertheless, 30 seconds later, the reactor tripped at 80% power on high pressure in the reactor coolant system.

One second after reactor/turbine trip, one channel of the Steam and Feedwater Rupture Control System (SFRCS) was automatically initiated due to a spurious signal indicating low water level in Steam Generator 2. Both Main Steam Isolation Valves (MSIVs) closed. Three seconds after the actuation, the SFRCS automatically reset. Closing of the MSIVs isolated the turbine of the operating main feedwater pump from its source of steam. The pump continued to supply feedwater to the steam generators for a few minutes as it coasted down.

Four and a half minutes after reactor trip, water level in the steam generators began to fall from the normal post-trip level which is 35 inches. After MSIV closure, steam release to atmosphere continued to remove decay heat. One minute later, Channel 1 of SFRCS actuated when the water level in Steam Generator 1 actually reached the SFRCS setpoint at 27 inches (See Figure 1). SFRCS started Auxiliary Feedwater Pump 1 and initiated alignment of it to Steam Generator 1.

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Within seconds after automatic initiation of Channel 1 of SFRCS, the operator actuated both channels of SFRCS; however, he inadvertently actuated both SFRCS channels on low steam pressure instead of low water level. When an SFRCS channel is actuated on low steam pressure, a rupture of the steam line associated with that channel is presumed to have occurred. The SFRCS closes the steam generator isolation valves, including a valve in the auxiliary feedwater line, and aligns the auxiliary feedwater pump to the other steam generator. Because both channels had been manually actuated on low steam pressure, both steam generators were isolated from both auxiliary feedwater pumps. Five seconds after the operator's inadvertent actuation of both channels on low steam pressure, SFRCS Channel 2 received an actual low water level actuation signal. Because low pressure initiation takes precedence, alignment of the auxiliary feedwater pumps remained unchanged. At six minutes into the event as both auxiliary feedwater pumps were accelerating, they tripped on overspeed.

In summary, all main feedwater had been lost, both steam generators were isolated from feedwater and were boiling dry, all auxiliary feedwater pumps were tripped, pressure of the reactor coolant system was rising, and reactor coolant system temperature was increasing.

Within one minute after the operator's inadvertent actuation of the SFRCS on low steam pressure, the mistake had been recognized and the SFRCS had been reset. If equipment had performed in accordance with system design requirements, the operator's error might not have had a significant impact on the event. The auxiliary feedwater isolation valves should have reopened automatically, but the valves did not reopen. The operator then tried to reopen the valves from the main control panel, but the valves would not reopen. Operators were dispatched to locally start the auxiliary feedwater pumps, open the auxiliary feedwater isolation valves, start the nonsafety-related motor-driven startup feedwater pump, and valve it to the system.

Pressure and temperature in the reactor coolant system continued to rise because there was not sufficient water in the steam generators to provide an adequate heat sink. At 13 minutes after reactor trip, reactor coolant system pressure reached 2425 psig, and the Pilot Operated Relief Valve (PORV) opened three times to limit the pressure rise. On the third lift, the valve remained open. The operator closed the PORV block valve and reopened it two minutes later after the PORV had closed.

Approximately 16 to 18 minutes after reactor trip, the operators had the startup and auxiliary feedwater pumps running and the valves aligned. Water levels were beginning to rise in the steam generators. Reactor coolant temperature reached a maximum of 594° F and then started to decrease to normal. Refilling of the steam generators caused the reactor coolant system to fall to 1716 psig and about 540°F before returning to normal (See Figure 2).

At 30 minutes after reactor trip, plant conditions were essentially stable.

Discussion:

For several minutes after reactor trip, the steam generators were unable to cool the reactor coolant system adequately.

The first problem contributing to this event was the loss of all main feedwater due to closure of the MSIVs. The licensee's hypothesis, based on information from Babcock & Wilcox, is that turbine trip caused a pressure transient upstream from the turbine stop valves which caused the outputs of the redundant steam generator level instrumentation channels to oscillate widely for several seconds. The licensee believes that this caused a spurious low level actuation of SFRCS which closed the MSIVs.

Three additional problems contributed to this event by affecting the availability of both trains of the auxiliary feedwater system. The first occurred when the reactor operator pressed the wrong SFRCS buttons. The second occurred when both auxiliary feedwater pumps tripped on overspeed. The third occurred when both auxiliary feedwater isolation valves did not reopen when SFRCS was reset.

Control buttons for the SFRCS are arranged in two vertical columns. Each column of buttons controls one SFRCS channel. The operator should have pressed the fourth button from the top in each column. Instead, the operator pressed the top buttons causing isolation of both steam generators.

Both auxiliary feedwater pumps are driven by Terry turbines which tripped on overspeed early in the event. When this occurred, steam was being supplied to the turbines via crossover lines, which are longer than the normal supply lines and include long horizontal runs. The licensee believes that significant condensation may have occurred in the crossover lines. Further, the licensee believes that the quality of steam arriving at the turbines may have been affected significantly by the configuration of the crossover lines and may have caused the overspeed trips.

The auxiliary feedwater system isolation valves have Limitorque motor operators. The motor operators have torque switches which prevent overtorquing of the valves by disconnecting power to the motors. When the valves are being opened, additional torque is required to overcome friction while the gates are being unseated and while a significant pressure differential may exist across the gates. During the initial part of the opening stroke, the torque switch in the motor operator is bypassed by a bypass switch so that full motor torque is developed if necessary. The licensee believes that these bypass switches went off bypass too early. The valves did not reopen until an operator unseated them by hand.

No specific action or written response is required by this information notice. If you have any questions about this matter, please contact the Regional Administrator of the appropriate NRC regional office or this office.

E L Jordan

Edward L. Jordan, Director
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and Engineering Response
Office of Inspection and Enforcement

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Attachments:
1. Figure 1 - Steam Generator 1 Level and Pressure
2. Figure 2 - RCS Temperature and Pressure
3. List of Recently Issued IE Information Notices

Attachment 1
IN 85-50
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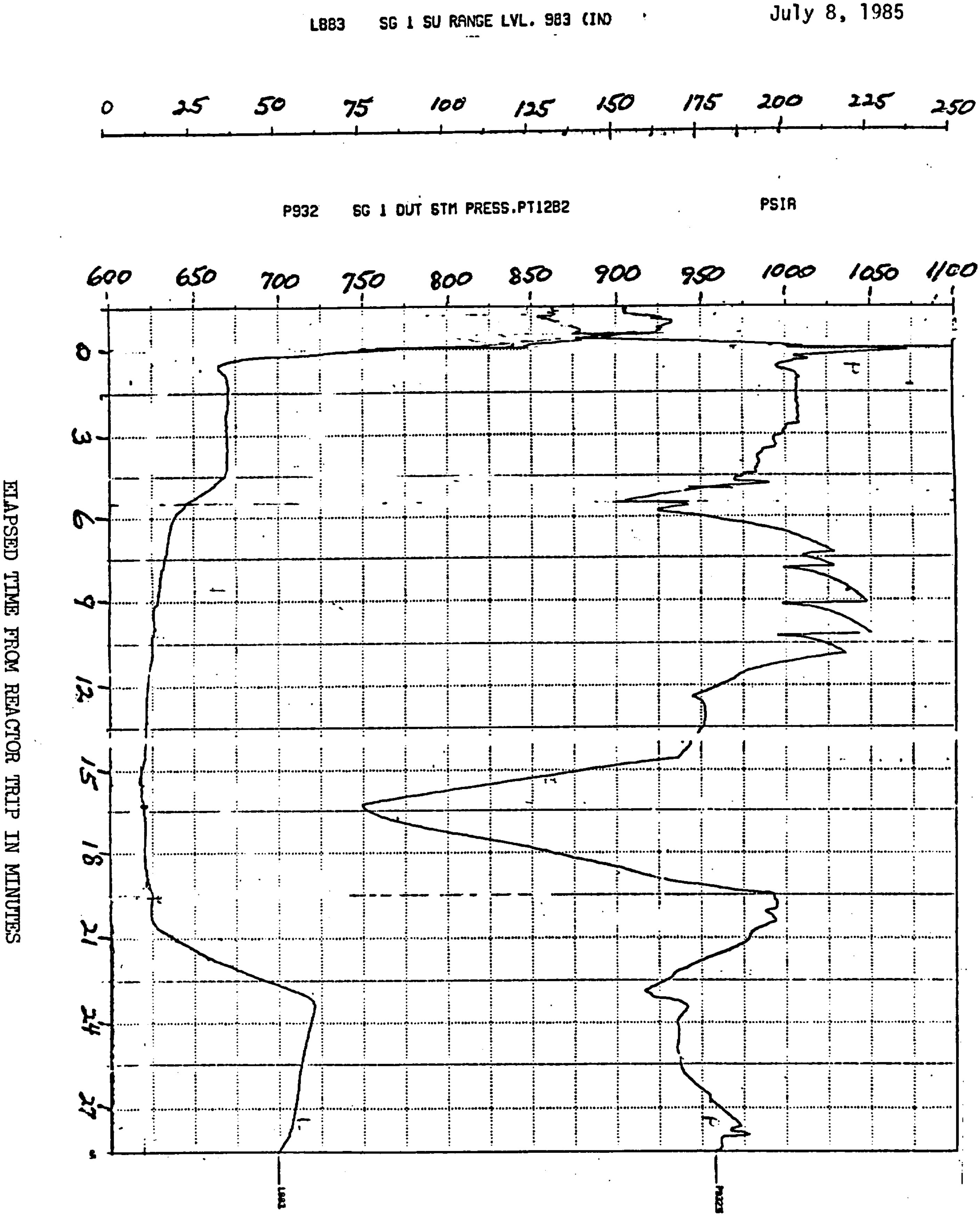

FIGURE 1: STEAM GENERATOR 1 LEVEL AND PRESSURE

Attachment 2
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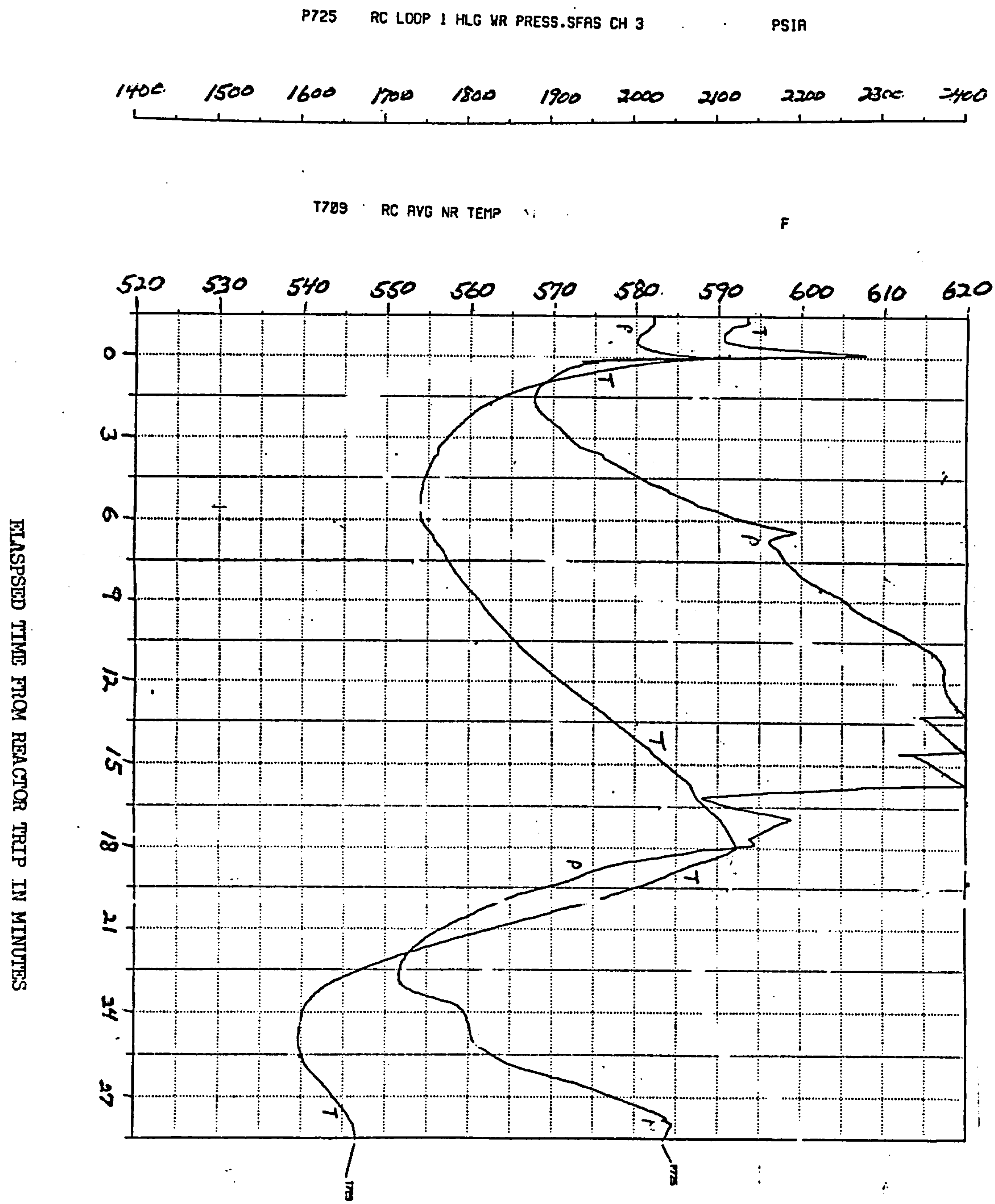

FIGURE 2: RCS TEMPERATURE AND PRESSURE

LIST OF RECENTLY ISSUED
IE INFORMATION NOTICES

Information Notice No.	Subject	Date of Issue	Issued to
85-49	Relay Calibration Problem	7/1/85	All power reactor facilities holding an OL or CP
85-48	Respirator Users Notice: Defective Self-Contained Breathing Apparatus Air Cylinders	6/19/85	All power reactor facilities holding an OL or CP, research, and test reactor, fuel cycle and Priority 1 material licensees
85-47	Potential Effect Of Line-Induced Vibration On Certain Target Rock Solenoid-Operated Valves	6/18/85	All power reactor facilities holding an OL or CP
85-46	Clarification Of Several Aspects Of Removable Radio-active Surface Contamination Limits For Transport Packages	6/10/85	All power reactor facilities holding an OL
85-45	Potential Seismic Interaction Involving The Movable In-Core Flux Mapping System Used In Westinghouse Designed Plants	6/6/85	All power reactor facilities holding an OL or CP
85-44	Emergency Communication System Monthly Test	5/30/85	All power reactor facilities holding an OL
85-43	Radiography Events At Power Reactors	5/30/85	All power reactor facilities holding an OL or CP
85-42	Loose Phosphor In Panasonic 800 Series Badge Thermo-luminescent Dosimeter (TLD) Elements	5/29/85	All power reactor facilities holding an OL or CP

OL = Operating License
CP = Construction Permit