Trans w/ Ltr dtd 8/21/73

"B" STEAM GENERATOR CONTROL VALVE

FAILURE ON JULY 22, 1973

ROCHESTER GAS AND ELECTRIC CORPORATION

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ROCHESTER, NEW YORK 14649

R.E. GINNA NUCLEAR POWER PLANT

UNIT NO. 1

August 21, 1973

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"B" S/G FEEDWATER CONTROL VALVE FAILURE ON JULY 22, 1973

The following is a report of the incident, investigation and summary of findings:

SECTION 1.0 - DESCRIPTION OF OCCURRENCE

The reactor trip occurred at 1306 hours on July 22, 1973. The "first out" indication on the annunciator panel in the control room indicated that the trip was actuated by a combination of low steam generator water level and a feedwater flow and steam flow mismatch in the "B" steam generator. The reactor was operating at approximately 1455 MWt and had been at this power at least 24 hours prior to this trip.

Approximately 10-15 seconds prior to the trip, the operators heard "loud rumblings" and felt the floors vibrating in the control room and in the service building. Several seconds after the noise and vibration commenced, annunciators began sounding on the main control board.

IMMEDIATE ACTION:

The head control operator immediately surveyed the control board and determined that the level in the "B" steam generator was decreasing. He then switched to manual control and attempted to regain level in the "B" steam generator. Simultaneously with this, the head control operator

directed the control operator to commence turbine load reduction. The control operator was in the process of reducing the turbine load "setter" when the trip occurred. No load reduction had actually commenced. As previously stated, the "first out" annunciator indicated the trip was activated by a combination of low steam generator level and feedwater flow-steam flow mismatch.

IMMEDIATE INVESTIGATION:

The shift foreman and auxiliary operators investigated a reported steam leak in the intermediate building and noted that a local pressure indicator near the "B" steam generator feedwater check valve #3992 had broken off. This leak was subsequently isolated.

Following the trip, the main steam and feedwater lines, including lines in the containment area, were examined for possible cause of trip and possible damage. It was apparent that the feedwater line had experienced vibration as was evident by damage to "B" steam generator feedwater control valve support. Another indication of feedwater piping displacement was damaged insulation.

The initial investigation indicated that loss of feedwater must have occurred, but the cause was not immediately evident. The "B" steam generator feedwater control valve was subsequently dismantled which revealed that the plug had become separated from the valve stem. This plug separation caused the loss of feedwater and ensuing reactor trip. Pertinent system parameters are plotted on Figure 1.1. Data for this

figure was obtained from the plant computer. It was theorized that this plug separation induced rapid feedwater flow variations resulting in the vibrations and noises witnessed by the operators on duty.

This event was discussed at a Plant Operations Review Committee Meeting on July 23, 1973.

Plans were initiated, based on visual examinations of postulated piping displacements, to determine stress levels achieved during the transient.

SECTION 2.0 - SUMMARY OF SUBSEQUENT INVESTIGATION

A thorough visual examination of the entire A and B feedwater lines, including all supports, anchors and restraints was made immediately following the transient. It was apparent that the "B" feedwater system downstream of the control valve had undergone an abnormal water hammer. A concrete support just downstream of the feedwater valve at Point 250 (See Figure 1-1 Appendix A) was cracked necessitating its replacement. Several rod hanger supports were slightly skewed indicating the feedwater line had undergone an oscillating type motion normal to the pipe axis at these points. Insulation on the "B" feedwater pipe was dented or scratched at several places and in two locations dislodged from the pipe where adjacent steel frames or sleeves had been struck. There was no evidence, however, that any permanent deformation or misalignment of the "B" feedwater line had occurred.

Examination of the "A" feedwater line revealed no indication that oscillating motion had occurred at any point on this line. There was

no evidence of any insulation marking or damage and it was concluded that the water hammer transient had been confined primarily to the "B" feedwater line. Approximate indications of pipe movement on the "B" feedwater line were recorded at locations where insulation had been marked by adjacent structures.

A time history dynamic analysis was performed to determine the magnitude of stresses resulting from the transient on the "B" feedwater line.

A description of the analysis and the resulting stresses are found in Appendix A Preliminary Report Dynamic Analysis of Feedwater Piping for Water Hammer Load Robert E. Ginna Plant.

In accordance with ANSI B31.1.0 Code for Power Piping, the allowable stress for the "B" feedwater piping is 21,000 psi and the yield stress at operating temperature is 40,000 psi. The peak water hammer stress calculated for the transient on the main "B" feedwater header was just over 31,000 psi. The header piping exceeded allowable stress but not yield at points 10, 85, and 110. It was decided that before this line would be returned for service that additional examination should be performed at these locations of high stress. This examination is more fully described in Section 3. "B" Feedwater Piping Inspection.

Subsequent investigation of branch lines indicated that the 3 inch "B" motor-driven auxiliary feedwater line had undergone some oscillating motion as the result of the water hammer transient. This line is subject to the operating pressure fluctuations of the main feedwater header. A dynamic analysis of the "B" auxiliary feedwater line indicated a peak

water hammer stress of approximately 36,000 psi at Point 115 (See Figure 1-2 Appendix A.) The analysis indicated that stress levels above allowable were exceeded at points 800, 810, 815, 820 and 825, all in the vicinity of the attachment to the main "B" feedwater header. These locations were all examined for indications of any local deformation (See Section 3).

All anchors, snubbers and restraint systems on the main header and the auxiliary feedwater line were carefully examined for indication of any damage where high loadings were indicated by the analysis. The concrete support at location Point 250 was removed and replaced with a new support. All other supports were found to be in good condition. There was no indication of any snubber failure or damage.

A summary of calculated deflections is given in Table A-7 of Appendix A. A comparison of calculated deflection with the approximate deflection indications observed after the transient show good general accord and are considered well within the tolerance of the modeling technique.

SECTION 3 - FEEDWATER PIPING INSPECTION

An examination was made of each weld on the "B" loop main feedwater piping and the auxiliary feedwater line which was shown by analysis to have been subjected to a stress greater than allowable. An examination was made of the pipe at each snubber where the pipe was shown by analysis to have been subjected to a stress greater than allowable. In addition, an examination was made of each snubber which was subjected to a load greater than design. These locations were as follows:
(See Figure 1-1, 1-2, 1-3 Appendix A)

(a)	Node Pt. 696	14" reducer to pipe circumferential weld FW-1005-19F. Upstream of steam generator inside shield wall. Shop weld.
(b)	Node Pt. 660	14" pipe at hydraulic snubber inside containment at shield wall.
(c)	Node Pt. 5	14" elbow to pipe circumferential weld FW-1005-19W. Inner containment penetration weld. Field weld.
(d)	Node Pt. 10	14" pipe to elbow circumferential weld FW-1005-19V. Outer containment penetration weld. Field weld.
(e)	Node Pt. 56	Hydraulic snubber FW-14. Outside containment within facade.
(f)	Node Pt. 65	Hydraulic snubber FW-81. Outside containment within facade.
(g)	Node Pt. 85	Hydraulic snubber FW-82 and 14" pipe at hydraulic snubber FW-82. Outside containment within facade.
(h)	Node Pt. 110	Hydraulic snubber FW-83 and 14" pipe at hydraulic snubber FW-83. Inside intermediate building in vertical piping.
(i)	Node Pt. 146	Hydraulic snubber FW-30. Inside intermediate building in horizontal piping.
(j)	Node Pt. 115	3" auxiliary feedwater nozzle weld. Inside intermediate building.
(k)	Node Pt. 800	3" pipe to valve circumferential weld. Auxiliary feedwater.

(l)	Node Pt. 810	3" valve to pipe circumferential weld. Auxiliary feedwater.
(m)	Node Pt. 815	3" elbow to pipe circumferential weld. Auxiliary feedwater.
(n)	Node Pt. 820	3" pipe to elbow circumferential weld. Auxiliary feedwater.
(o)	Node Pt. 825	3" elbow to pipe circumferential weld. Auxiliary feedwater.

Magnetic particle and ultrasonic examinations were made of the three main feedwater piping welds by Southwest Research Institute personnel. These examinations were made in accordance with Southwest Research Institute approved procedures by personnel qualified in accordance with SNT-TC-1A. There were no magnetic particle indications.

The ultrasonic examinations revealed numerous reflectors with amplitudes greater than 100% DAC. These reflectors were located in the root areas of the welds and were not readily interpretable by ultrasonics alone. As an aid to interpretation of the ultrasonic indications, radiographs of each of these welds were reviewed. The original construction radiographs were used for the field welds and new radiographs made by Buffalo X-Ray Corporation were used for the shop weld. This review indicated that areas which produced ultrasonic indications could be identified on the radiographs as root topography indications. It was the opinion of the reviewers that these welds met the code requirements

in effect at the time of construction, and that there was no apparent change in the conditions of these welds.

Visual examinations were made of the auxiliary feedwater pipe welds, the piping adjacent to the snubber clamps, and the penetration anchor welds. These examinations were made in accordance with Southwest Research Institute approved procedures. These examinations showed no visible abnormalities. In addition, it was observed that there were no circumferential pipe welds within at least 2 pipe diameters of the snubber clamps.

Visual examinations were also made of each of the hydraulic snubbers on the main feedwater line. These examinations found no evidence of damage to any of the snubbers or their support structures.

SECTION 4 - CONTROL VALVE FAILURE

The abnormal water hammer which occurred on the "B" feedwater system has been attributed to failure of the "B" control valve. Immediately following the recovery period after the incident the "B" feedwater valve was disassembled. It was discovered that the control plug had separated from the threaded stem and a small pin which prevents rotation of the plug about the stem had failed. The valve manufacturer, Copes Vulcan, was consulted and subsequent investigation revealed that the original stem on this valve had been damaged during construction. A replacement stem had been installed, however, it was determined that the recommended torqueing procedure had not been followed during reassembly of the plug on the new stem. The manufacturer indicated that had the

plug been torqued properly, the holding pin would not have failed. A new stem, plug and cage assembly was installed in the "B" feedwater valve using the manufacturer's recommended procedure.

SECTION 5 - RECOMMENDATIONS AND CONCLUSIONS

The dynamic analysis of the "B" feedwater header and auxiliary feedwater line indicated that code allowable stress values had been exceeded as the result of an abnormal water hammer caused by failure of the "B" feedwater control valve. The RGE performed an inservice type inspection of all the welds at which higher than allowable stress had occurred and visually examined the piping for indications of local deformation. No indications of pipe weld defect or failure were observed. Based on the analysis and inspections which had been performed, RGE concluded that the feedwater system could be returned to normal operation following replacement of the "B" feedwater valve internals and repair of the damaged support adjacent to this valve. A program of monitoring the stem vibrations on both the "A" and "B" feedwater valves has been instituted in order to provide additional information on the nature of throttling turbulence at this control stage.

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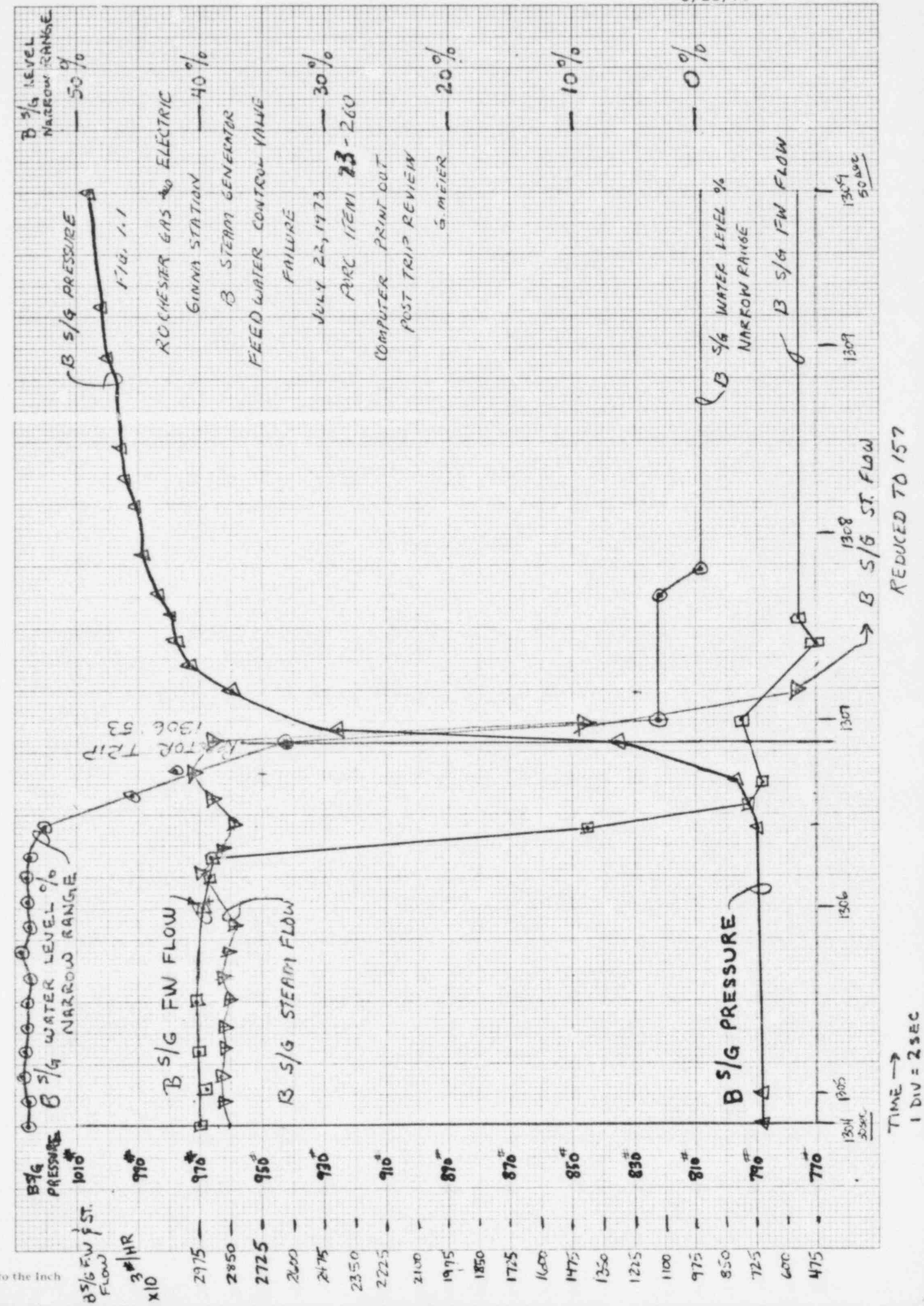

20 Squares to the Inch