SMUD
SACRAMENTO MUNICIPAL UTILITY DISTRICT ☐ 6201 S Street, Box 15830, Sacramento, California 95813; (916) 452-3211

May 14, 1980

Director of Nuclear Reactor Regulation
Attention: Mr. Robert W. Reid, Chief
Operating Reactors, Branch 4
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

Docket No. 50-312
Rancho Seco Nuclear Generating
Station, Unit 1

Dear Mr. Reid:

Your letter of February 26, 1980, requested information concerning the Rancho Seco, Unit 1, Auxiliary Feedwater System. The Sacramento Municipal Utility District responded on March 18, 1980 and April 14, 1980, with a portion of the information requested. The attachment to this letter provides additional information listed by Part and Item numbers corresponding to those in your letter.

Sincerely,

John J. Mattimoe

John J. Mattimoe
Assistant General Manager
and Chief Engineer

Attachment

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SACRAMENTO MUNICIPAL UTILITY DISTRICT

Response to February 26, 1980, NRC Requirements for the Continued Upgrade of the Rancho Seco Auxiliary Feedwater (AFW) System

Part A Item 1 AFW Reliability Study Mission Success Criteria

The attainment of flow to at least one steam generator, for decay heat removal, is the primary function of the AFW System and provides a complete definition for mission success. The success criterion does not include a requirement to deliver AFW flow before steam generator dry out, because this is a function of secondary importance. The AFW System was designed for automatic actuation partly to preclude steam generator dry out; however, even if actuation is delayed for twenty minutes, the AFW System can still accomplish the primary function of decay heat removal.

It is correct that no high pressure injection cooling is available in Case 3, a total loss of all AC power; however, the success criteria on page 2 of the AFW Reliability Report is in error in stating that AFW delay is acceptable only if an HPI pump is operating. Supporting analyses on loss of feedwater events indicate that recovery of AFW flow after a delay of twenty minutes is sufficient to assure adequate core cooling with no dependence on HPI availability. An AFW flow delay of thirty minutes has not been analyzed. It should be noted that a loss of all AC power is not a design basis event. With adequate core cooling as a criterion, the five and fifteen minute data is meaningful for all cases examined including the loss of all AC power case.

In summary the success criteria selected for the B&W AFW Reliability Study was based on the concern of ultimate importance - maintaining adequate core cooling - and not on preventing steam generator dry out; and therefore, the success criterion will not be revised. However, the "criterion for mission success" on page 2 will be changed to delete the stated dependence on high pressure injection.

Part A Item 2 AFW Pump Suction and Discharge Instrumentation Modifications to AFW Reliability Study

The absence of AFW pump suction and discharge pressure in the control room does not significantly affect the unavailability of the Rancho Seco AFWS. The Rancho Seco fault tree has been requantified to account for the lack of AFW pump suction and discharge pressure indication in the control room. The numerical difference between this and the original quantification is negligibly small (a maximum difference of 7%).

The above noted difference was kept small by conservative assumptions in the original analysis related to the causes and locations of valve closure, and the small credit for operator action based on suction indication. If less conservative assumptions had been used, the benefit of pump suction indication would have been more apparent.

Because suction pressure indication was originally assumed available and because it is the preferred indication for assuring pump suction, no credit was taken in the original analysis for pump discharge pressure indication. Consequently, absence of pump discharge pressure indication has no affect on the calculated AFWS unavailability.

Part A Item 3 ICS/NNI Failure Impact on AFW Reliability Study

The ICS/NNI power supply was not identified as a potential single failure source for the AFWS because this battery-backed 120 VAC power source was assumed to be available for all cases (as stated on page 3 of the report). This simplifying assumption was required by the NRC for conformance with NRC analyses on other plants; its use ruled out the investigation and identification of this and similar power-related failure mechanisms.

A detailed analysis of AFWS actuation and control systems was outside the scope of the study. In recognition of this fact, the NRC provided an unavailability value (7×10^{-3} per demand) to be assigned to each separate actuation and control train. B&W's analysis was performed only to the extent necessary to determine if separate, redundant actuation trains were available. If not, a single, common control devise was assumed. For Rancho Seco, it was determined that commonality existed for circuits which open control valves in both AFWS trains Thus, as stated in assumption (8) on page 3 of the report it was assumed that the ICS consisted of a single, common control device with inputs to both trains and an unavailability of 7×10^{-3} was assigned to this device. During quantification for Case 1, this unavailability was reduced by a factor of .2 to account for an estimated 80% of ICS failures which result in a fail-safe mode;i.e., inadvertent opening of the valves. It was implicitly assumed that this number included all related actuation and control failures including power supply failures.

There is no discrepancy between the assumption of the ICS as a single control device and the fault tree shown on Pages A-7 and A-8. While ICS failure is shown on separate branches of the fault tree, this was done only to illustrate that ICS failure has a direct effect on the operation of control valves in both trains. The equivalency of the SMUD fault tree to a tree emphasizing the ICS commonality is shown in Figures A.3-1 and A.3-2.

In summary, the ICS is treated as a single control device with signals to both AFW control valves, and ICS/NNI power supplies were not identified because of simplifying assumptions dictated by the NRC.

Part B Item 3 <u>System Modification for Full Flow AFW Pump Testing</u>

The District does commit to provide position indication in the control room for the motor operated valve that will replace FWS-055. In addition, we commit to modify the test procedures for Full Flow AFW Testing when the modification is completed and will provide a P&ID of the final design for Staff's information prior to the implementation of the modification.

Part B Item 5 <u>AFW System Flow Path Verification</u>

The District proposed a revised technical specification related to full flow testing to the steam generator on April 30, 1980.

Part B Item 7 <u>AFW Pump Endurance Test</u>

a. Test Method

The Auxiliary Feedwater Pumps, P-318 and P-319, were individually run for a total time of 40 hours each. The first 32 hours for each pump was run with the pump in a "minimum flow" configuration. The pumps ran at an outlet pressure of approximately 1300 psig while discharging through a mini-flow orifice to the main condenser (see attached flow schematic, Figure B.7-1). The remaining eight hours of testing on each pump was performed at rated flowrate approximately 780 gpm minimum, by throttling the pump dischare (see attached flow schematic, Figure B.7-2). This same test procedure for flowrate verification is used in the Quarterly Auxiliary Feedwater System Surveillance Test.

Pump suction and discharge pressures were measured using 1/2% accuracy test quality gauges.

Pump motor and turbine bearing temperatures were measured using a hand-held pyrometer.

b. Test Conditions

Auxiliary Feedwater Pump design flowrate is 780 gpm, plus 60 gpm mini-flow, with 1113 psid pump differential.

The design conditions for the Turbine/Motor Driven Auxiliary Feedwater Pump, P-318, assume saturated steam at 1050 psig Steam Generator pressure. This is a reasonable assumption since the Main Steam Valves are set to begin relieving at 1050 psig. This is considerably above the normal operating Steam Generator pressure of approximately 905 psig.

During the test of P-318, the Main Steam System was used to supply motive steam to its turbine drive, K-308. The steam temperature and pressure were approximately 590°F and 925 psig, respectively.

Pump flow during the "minimum flow" portion of the test was approximately 60 gpm. The pump differential under these conditions is approximately 1300 psid. The Full Flow portion of the testing done on P-319 was done at 1113 psid pump differential. This pump will deliver at least 840 gpm total flow (including 60 gpm through its mini-flow orifice) with 1113 psid pump differential. This fact is verified by a Surveillance Procedure, SP 210.01B, each calendar quarter. The Full Flow portion of the testing done on P-318 was done at 1000 psid pump differential. This pump will deliver at least 900 gpm total flow (including 60 gpm through its mini-flow orifice) with 1000 psid pump differential. This fact is verified by a Surveillance Procedure, SP 210.01A, each calendar quarter.

c. Data Plots

Plots of bearing temperature versus time are shown in Figures B.7-3 through B.7-10. At no time did the measured bearing temperatures exceed the manufacturer's rated maximum of 240°F. During the Full Flow portion of the test, bearing temperature increased due to the following three factors:

1. Condensate Storage Tank temperature increased from its initial value of 60-70°F up to approximately the temperature of the condenser hotwell due to make-up. Since the pumps and turbine use water extracted from the first pump stage for bearing oil cooling, any increase in pump suction temperature would be reflected in higher bearing temperatures.

2. Ambient temperature at the Site varied from 58°F at 0300 on July 7, 1979 to 72°F at 1100.

3. Increasing gland seal leakage from the turbine driver caused the measured bearing temperature at the West (governor end) Turbine Bearing to increase at the end of the test of P-318. This measured temperature is higher than the bearing temperature

due to heating of the pyrometer probe by leaking steam. This steam leakage, while not excessive, tends to raise the measured bearing temperature when a contact pyrometer is used.

d. Pump Enclosure

The Auxiliary Feedwater Pumps at Rancho Seco are enclosed in a missile shield of open steel mesh. This missile shield provides negligible restriction to ambient air flow and weather. The missile shield is located outdoors in the Reactor Yard area. Because of this, the pumps and drivers are at all times exposed to the weather and temperature at the Site and no artificial environment is assumed.

e. Pump Vibration and Subsequent Pump Performance

Pump vibration was within limits during the entire test.

Both Auxiliary Feedwater Pumps successfully passed their respective Quarterly Surveillance Tests following the Endurance Test. Approximately two weeks elapsed between the performance of the Endurance Test and the Surveillance Tests.

Both Auxiliary Feedwater Pumps have continued to perform quite well during their scheduled Surveillance Tests and required no major maintenance as a result of the Endurance Run.

f. Follow-Up Testing

The turbine bearing temperature data for Pump P-318 was rather erratic. This is most probably due to poor technique in obtaining the data. Since it cannot be conclusively shown that the bearing temperatures stabilized at an acceptable value during the test, this pump will be retested using the turbine driver. The turbine will be tested with the pump at full flow for a minimum of eight hours and data will be taken hourly. If the turbine bearing temperatures have not stabilized at the end of eight hours, the test will be continued until each turbine bearing temperature has been stable for at least four hours. A temperature will be assumed to be stable if in a four hour period the temperature does not vary more than 20°F and the final temperature is not more than 5°F higher than the temperature at the previous hour's data point. This test will be performed during power operation following the refueling/maintenance outage currently in progress.

Part B Item 10 New Technical Specification Related AFW System Outages

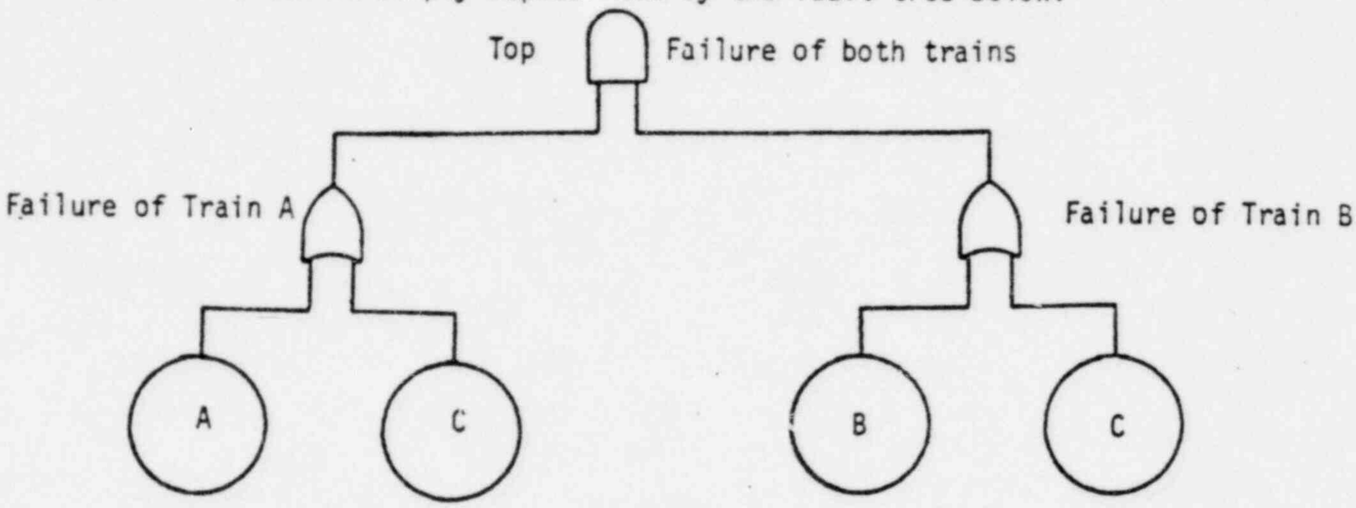
The District submitted a proposed revised technical specification concerning AFW System outages on April 30, 1980.

Part D Design Basis for AFW System Flow Requirements

The analysis required by this request is being completed by the Babcock and Wilcox Company. As soon as this information is reviewed by the District it will be submitted to the NRC. This submittal should be no later than June 1, 1980.

FIGURE A.3-1

In the Auxiliary Feedwater Reliability Analysis for the Rancho Seco Nuclear Generating Station, ICS failures are shown on separate branches of the fault tree. This can be simply represented by the fault tree below:



A = Failure of AFWS control valve in train A

B = Failure of AFWS control valve in train B

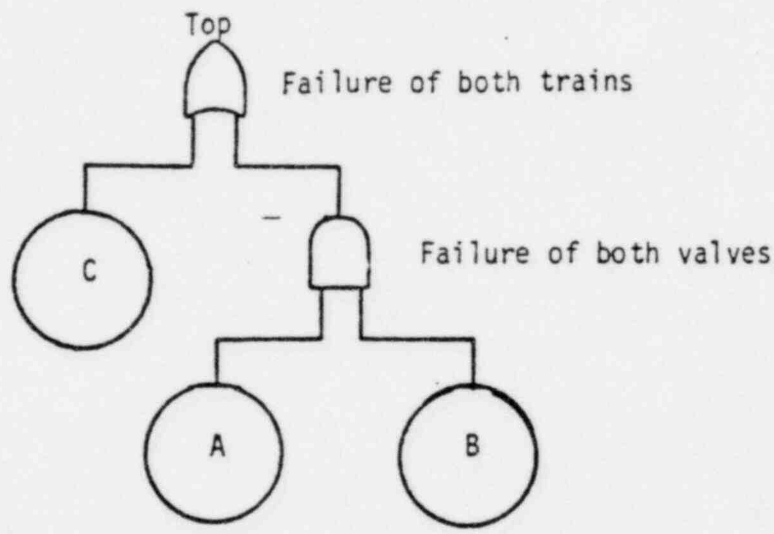
C = Failure of the ICS.

The probability of the top event in this tree is $P\{(A \cup C) \cap (B \cup C)\} =$

$$= P(A \cup C) + P(B \cup C) - P((A \cup C) \cup (B \cup C))$$
$$= P(A \cup C) + P(B \cup C) - P(A \cup B \cup C)$$
$$= (P(A) + P(C) - P(AC)) + (P(B) + P(C) - P(BC))$$
$$- \{P(A) + P(B) + P(C) - P(AB) - P(AC) - P(BC) + P(ABC)\}$$
$$= P(AB) + P(C) - P(ABC)$$

FIGURE A.3-2

An optional fault tree that emphasizes the independence of the ICS is:



Here the probability of the top event is:

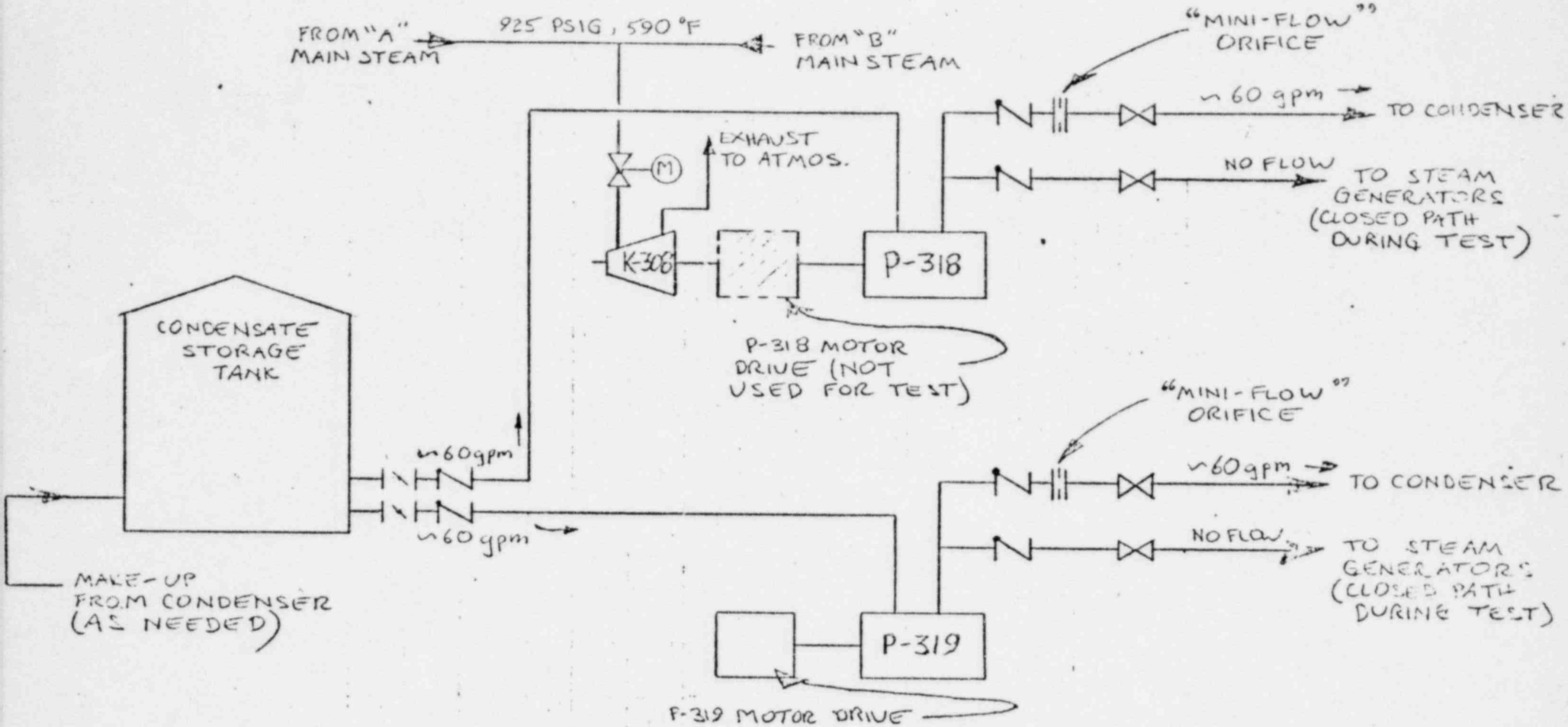
$$P\ \{(A \cap B) \cup C\} = P(A \cap B) + P(C) - P\{(A \cap B) \cap C\}$$

$$= P(AB) + P(C) - P(ABC)$$

which is the same as the probability of the previous tree.

FIGURE B.7-1

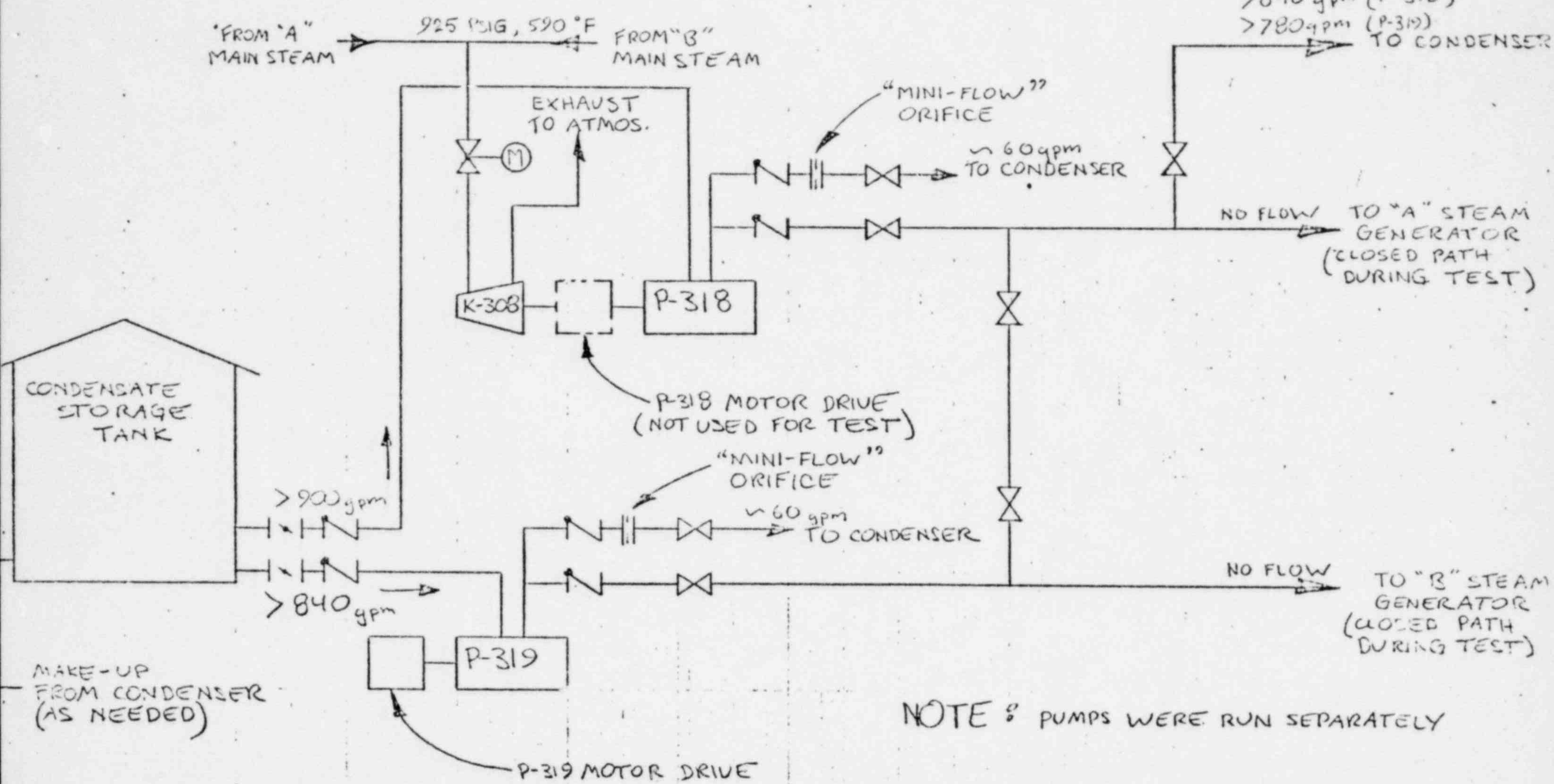
MINI-FLOW CONFIGURATION FLOW SCHEMATIC



NOTE: PUMPS WERE RUN SEPARATELY

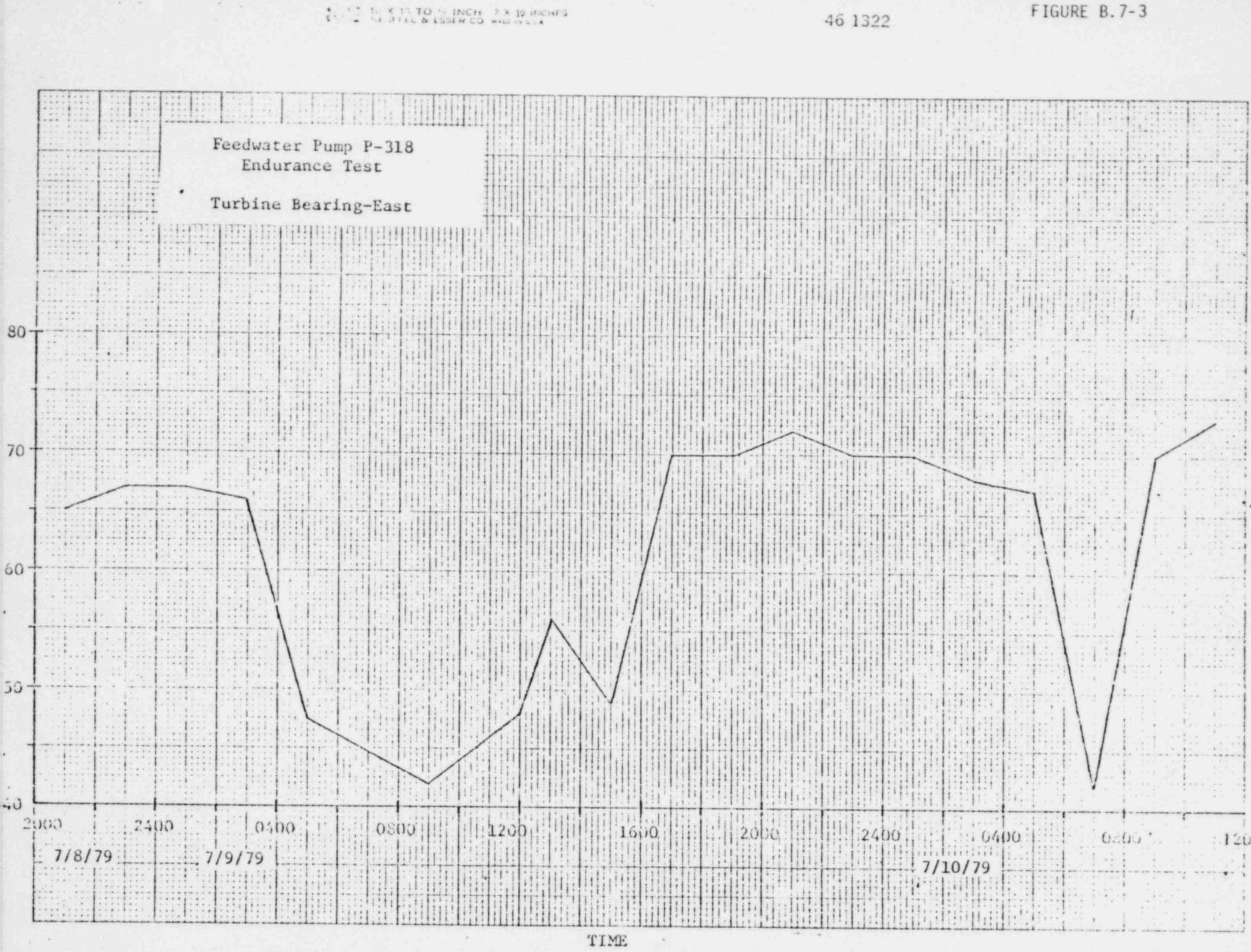
FIGURE B.7-2

FULL FLOW TEST FLOW SCHEMATIC



NOTE: PUMPS WERE RUN SEPARATELY

FIGURE B.7-3



46 1322
Feedwater Pump P-318
Endurance Test
Turbine Bearing-East
80
70
60
50
40
2000
2400
0400
0800
1200
1600
2000
2400
0400
0800
1200
7/8/79
7/9/79
7/10/79
TIME

46 1322

FIGURE R.7-4

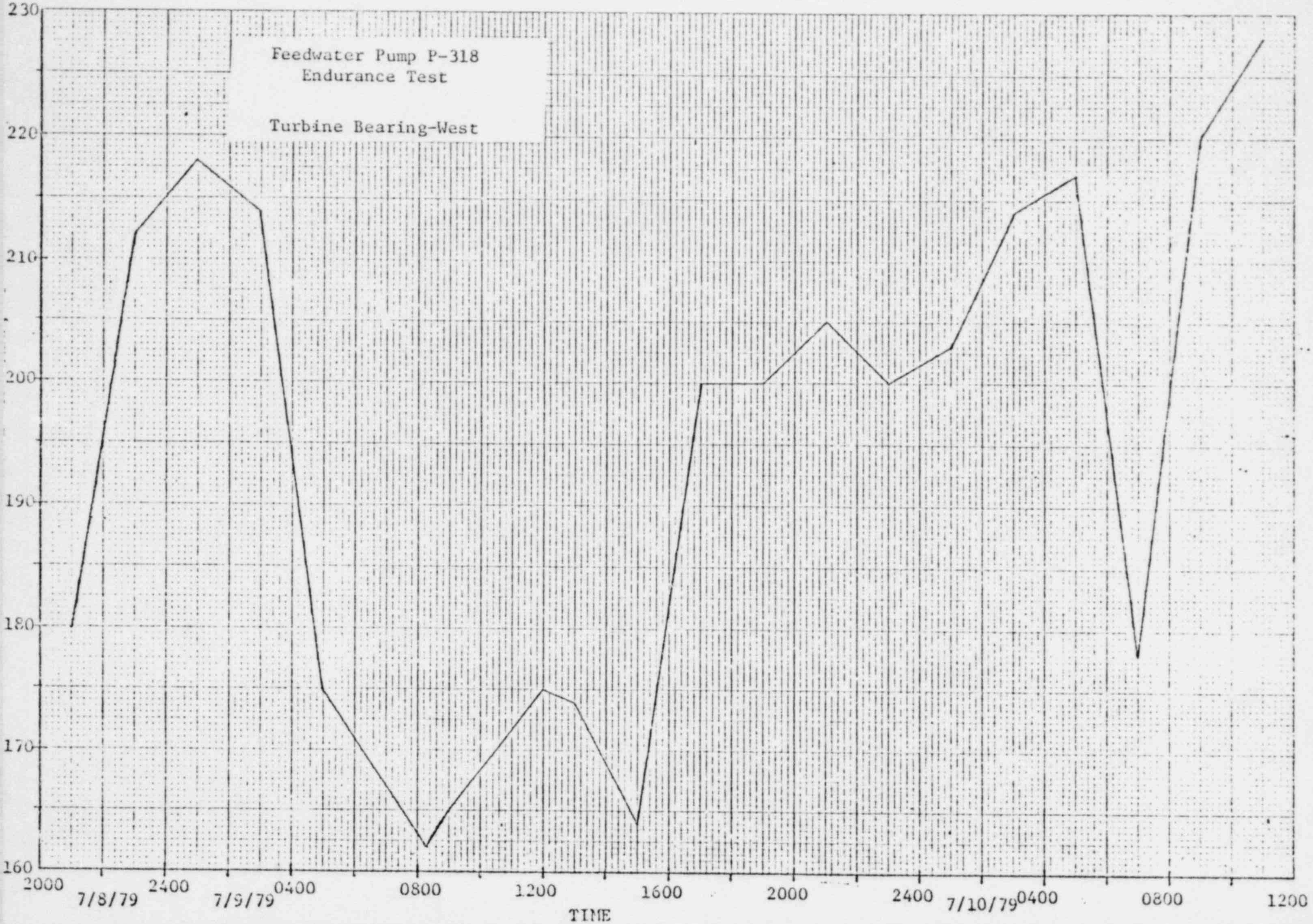
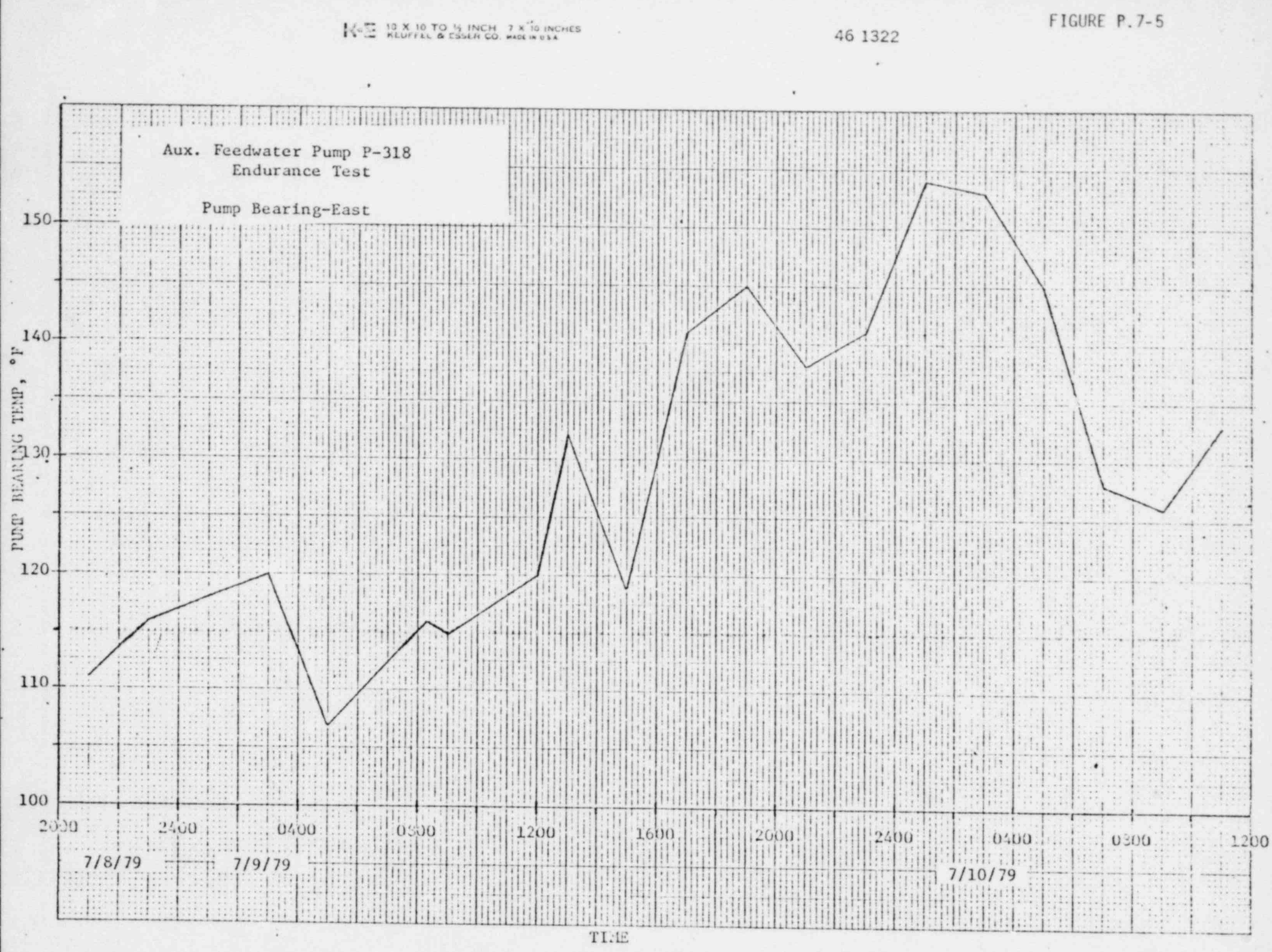


FIGURE P.7-5



K-E 10 X 10 TO ½ INCH 7 X 10 INCHES KEUFFEL & ESSER CO. MADE IN U.S.A.
46 1322
Aux. Feedwater Pump P-318
Endurance Test
Pump Bearing-East
PUMP BEARING TEMP, °F
150
140
130
120
110
100
2000
2400
0400
0800
1200
1600
2000
2400
0400
0800
1200
7/8/79
7/9/79
7/10/79
TIME

46 1322

FIGURE B.7-6

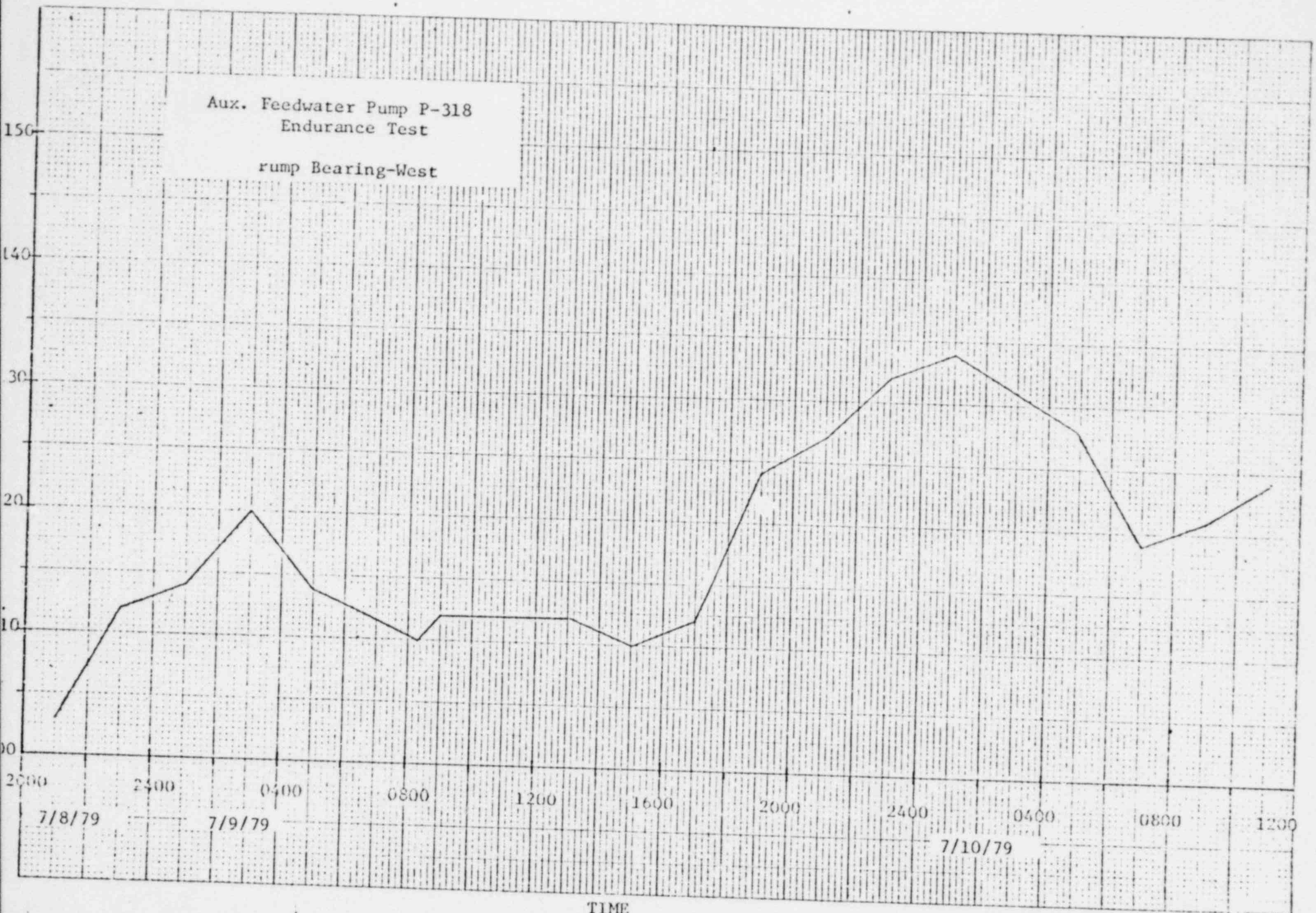


FIGURE B.7-7

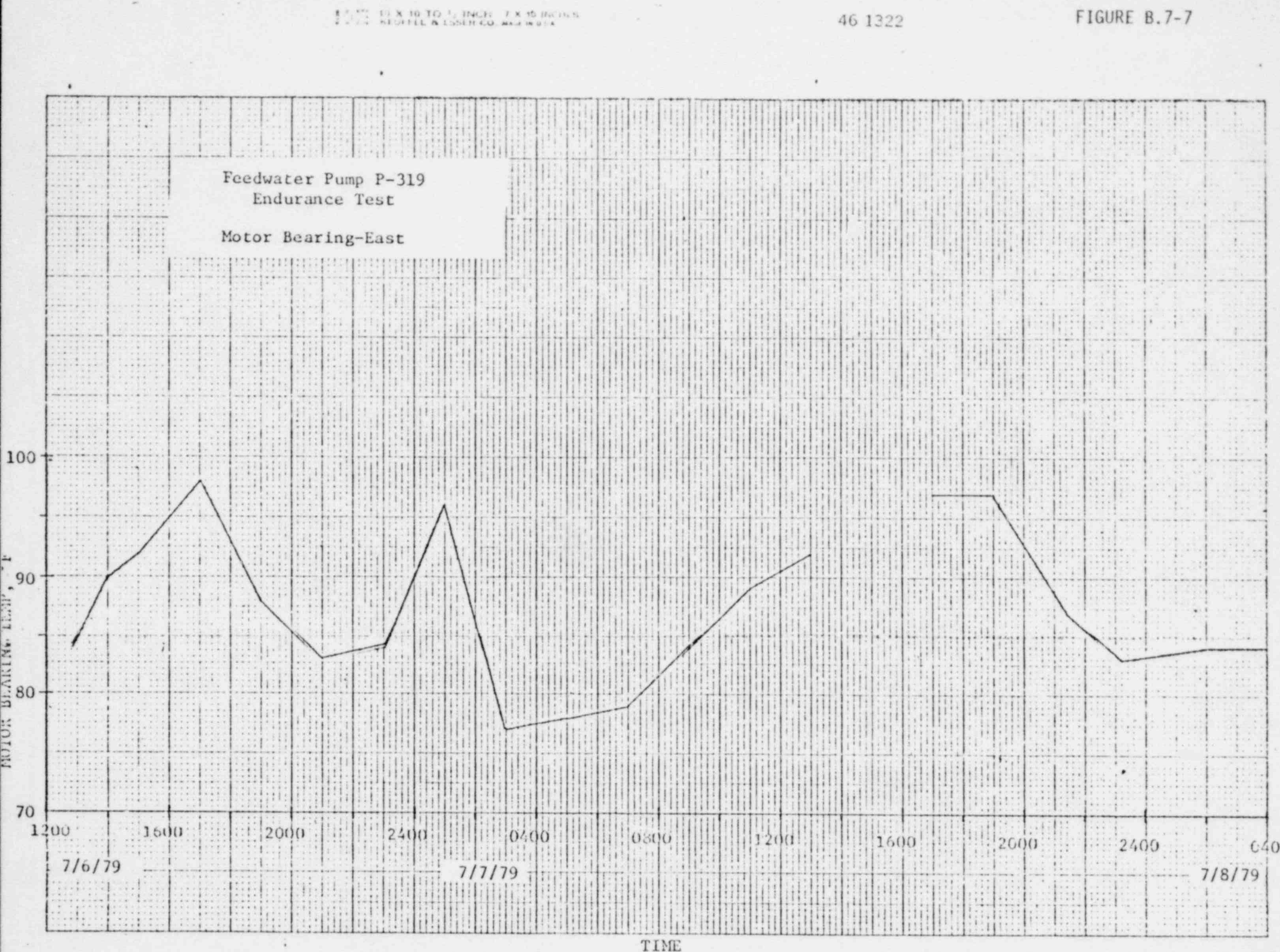
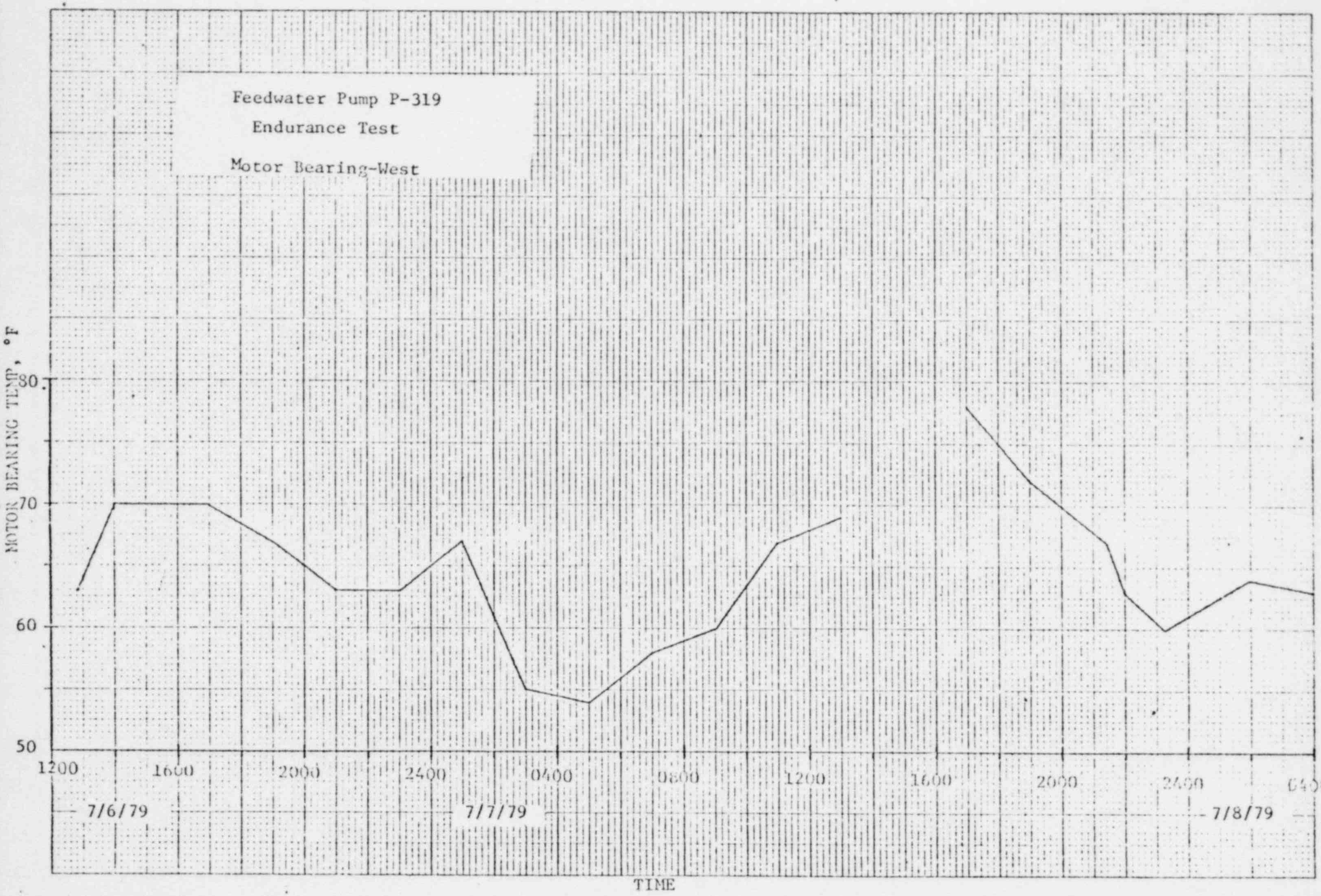
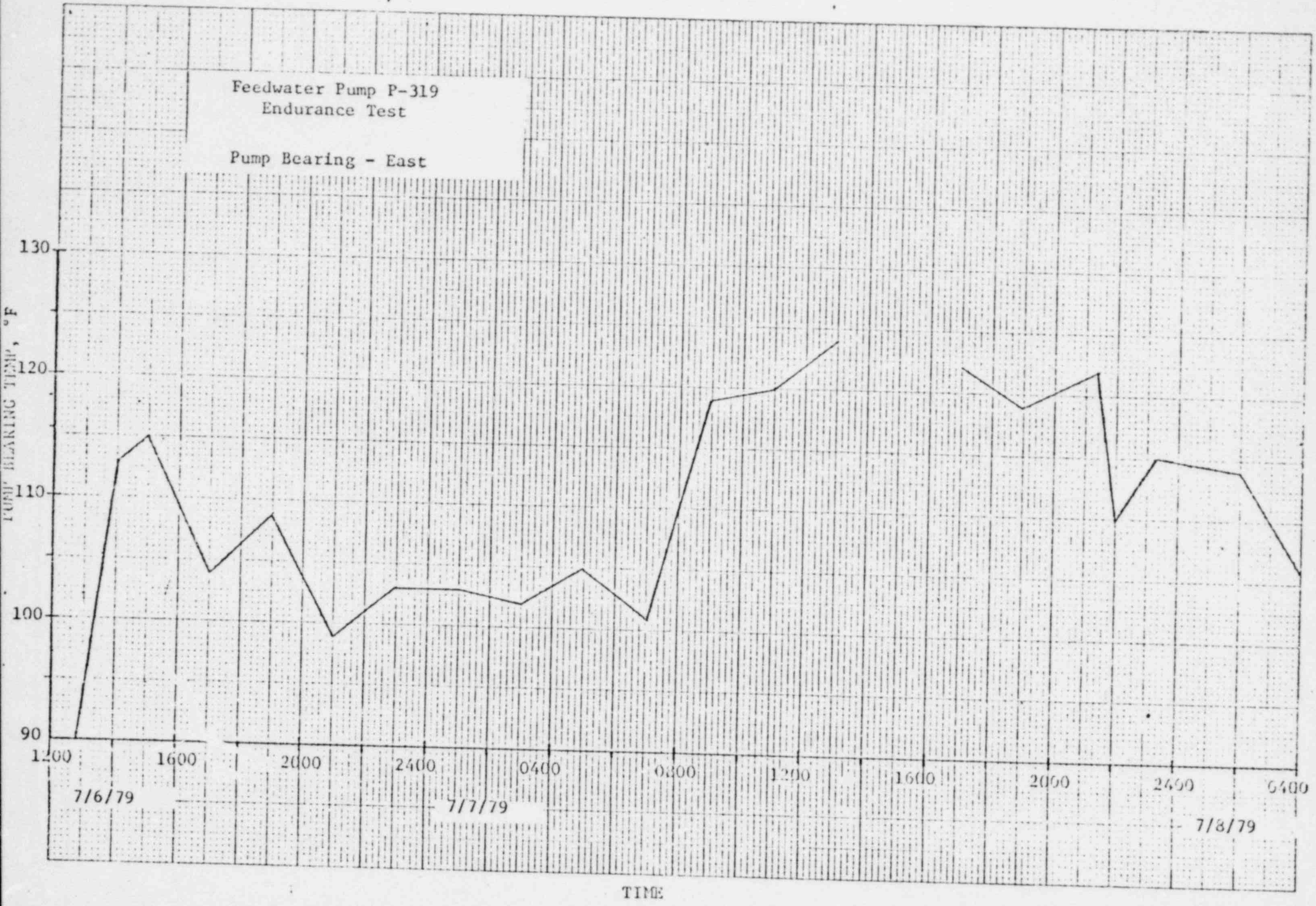

46 1322
Feedwater Pump P-319
Endurance Test
Motor Bearing-East
100
90
80
70
MOTOR BEARING TEMP, °F
1200
1600
2000
2400
0400
0800
1200
1600
2000
2400
0400
7/6/79
7/7/79
7/8/79
TIME

FIGURE B.7-8


Feedwater Pump P-319
Endurance Test
Motor Bearing-West
MOTOR BEARING TEMP, °F
80
70
60
50
1200
1600
2000
2400
0400
0800
1200
1600
2000
2400
0400
7/6/79
7/7/79
7/8/79
TIME

46 1322

FIGURE B.7-9



K+E 10 X 10 TO ½ INCH 7 X 10 INCHES KEUFFEL & ESSER CO. MADE IN U.S.A.

46 1322

FIGURE B.7-10