Met-Ed / GPU

Metropolitan Edison Company
Post Office Box 480
Middletown, Pennsylvania 17057
717 944-4041

April 12, 1979
GQL 0509

Mr. Denwood Ross
Assistant Director
Division of Reactor Safety
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Sir:

Three Mile Island Nuclear Station Unit 2 (TMI-2)
License No. DPR-73
Docket No. 50-320
Safety Analysis Report for Transition to Natural Circulation (C-D)

Enclosed please find the Safety Analysis Report and preliminary information for the proposed transition to long term natural circulation at TMI-2, as requested at the Commission Meeting of April 9, 1979. This report is current as of approximately April 9. The analysis is significant in that it suggests a great deal of flexibility in placing the TMI-2 reactor in a natural circulation mode. Additional analyses are continuing to be performed which will define more specifically the proposed final end point temperature and pressure conditions as well as state points in the various supporting plant systems. Detail analyses and procedures on the exact methods for achieving natural circulation are also in work. As additional information becomes available, we will supplement the attachment as necessary.

It is our conclusion, from the data in the attachment, that long term natural circulation is a viable way for placing the TMI-2 reactor into a long term stable condition, and the safest of the various options available.

Sincerely,

Original signed/
J. G. Herbein

✓ J. G. Herbein
Vice President
Generation

JGH:LWH:al
Enclosure

cc: Harley Silver (NRC)

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CORE THERMAL BEHAVIOR

CONTENTS

CONTENTS

3.0 SAFETY EVALUATION INFORMATION FOR TRANSITION TO NATURAL CIRCULATION COOLING

METROPOLITAN EDISON CO. HAS EVALUATED THE VARIOUS STATES FOR MAINTAINING THE TMI 2 REACTOR A LONG TERM COOLING MODE. WE HAVE PREPARED THE FOLLOWING EVALUATIONS WHICH DEMONSTRATE THAT THE REACTOR AND ASSOCIATED SYSTEMS CAN SAFELY UTILIZE RCS NATURAL CIRCULATION CORE COOLING WITH THE STEAM GENERATOR SECONDARY SIDE IN A SOLID FLOWING WATER CONDITIONS FOR HEAT REMOVAL.

3.1 DESCRIPTION OF COOLING MODE

ATTACHMENT 1 TO THIS REPORT ENTITLED "A SUMMARY OF NATURAL CIRCULATION ALTERNATIVES FOR LONG-TERM CORE COOLING AT TMI-2 "DESCRIBES THE PROPOSED METHOD FOR LONG TERM COOLING. THE DOCUMENT CONTAINS DETAILED INFORMATION ON THE RECOMMENDED COOLING METHOD ALONG WITH DISCUSSIONS DOCUMENTING THE SUPERIORITY OF THE RECOMMENDED METHOD OVER ALTERNATIVE CONSIDERATIONS.

3.2 BACKUP CONSIDERATIONS

METROPOLITAN EDISON CO. HAS PREPARED DETAILED OPERATING INSTRUCTIONS IN THE EVENT OF EQUIPMENT MALFUNCTION. A LIST OF THESE PROCEDURES IS PROVIDED IN ATTACHMENT 2. THE BACKUP CONTINGENCIES PROVIDED BY THESE PROCEDURES ASSURES RAPID AND ACCURATE RESPONSE TO EMERGENCY OR OFF NORMAL PLANT CONDITIONS.

3.3 SYSTEM PERFORMANCE ANALYSIS IN NATURAL CIRCULATION

B&W HAS PERFORMED DETAILED ANALYSES OF THE NATURAL CIRCULATION CONDITION FOR THE RECOMMENDED MODE OF COOLING. THESE CALCULATIONS INCLUDE BOTH HAND CALCULATIONS AND THE DEVELOPMENT OF COMPUTER CODES TO PREDICT TRANSITION SYSTEM RESPONSES. THE ANALYTICAL TECHNIQUES USED ARE DESCRIBED IN ATTACHMENT 3. THESE TECHNIQUES HAVE BEEN BENCHMARKED AGAINST NATURAL CIRCULATION DATA OBTAINED AT DAVIS-BESSE 1. IN ADDITION, TESTS WERE PERFORMED AT B&W'S ALLIANCE RESEARCH CENTER WHICH DEMONSTRATE THE EXCELLENT COOLING CAPABILITIES USING THE OTSG'S IN THE RECOMMENDED COOLING MODE. THE RESULTS OF THE ALLIANCE TESTING ARE DISCUSSED IN DETAIL IN ATTACHMENT 1.

ATTACHMENTS 1, 3 AND 4 DISCUSS THE POTENTIAL FOR CORE BLOCKAGE. AS SHOWN IN FIGURE 2 OF ATTACHMENT 1, MORE THAN ADEQUATE CORE FLOW WILL EXIST FOR THE RANGE OF ESTIMATED BLOCKAGE. IN ADDITION, ACCEPTANCE CRITERIA HAVE BEEN PREPARED WHICH WILL BE USED TO SAFELY TERMINATE THE TRANSITION TO THE NATURAL CIRCULATION MODE IF NEEDED.

IT IS THEREFORE METROPOLITAN EDISON COMPANY'S VIEW THAT DETAILED ANALYSES OF THE TRANSITION TO THE NATURAL CIRCULATION MODE OF COOLING DEMONSTRATE THAT THE RECOMMENDED COOLING MODE CAN MAINTAIN THE CORE IN A SAFE CONDITION. IN ADDITION, IN THE UNLIKELY EVENT THAT PROBLEMS DO ARISE, ACCEPTANCE CRITERIA WILL ASSURE THAT THE TRANSITION OPERATION CAN BE SAFELY TERMINATED AND THE PLANT RETURNED TO ITS ORIGINAL COOLING MODE.

3.4 CHECKPOINTS DURING THE TRANSITION OPERATION TO NATURAL CIRCULATION.

THE ACCEPTANCE CRITERIA FOR THE TRNASITION OPERATION ARE INCLUDED IN ATTACHMENT 1 AND THE THERMOCOUPLE CRITERIA ARE INCLUDED IN ATTACHMENT 4.

3.5 HYDROGEN EVALUATION

METAL-WATER REACTION DURING THE INITIAL PHASES OF THE TMI-2 INCIDENT GENERATED LARGE QUANTITIES OF HYDROGEN ON MARCH 28, 1979. THIS HYDROGEN FORMED A BUBBLE WHICH BECAME TRAPPED IN THE HEAD OF THE REACTOR VESSEL. THE PARTIAL PRESSURE OF HYDROGEN IN THIS BUBBLE CAUSED THE REACTOR COOLANT TO BECOME SATURATED WITH HYDROGEN. AFTER THE BULK OF THE BUBBLE WAS REMOVED ON APRIL 1, THE COOLANT REMAINED SATURATED WITH 1300 TO 1400 STD. CC OF HYDROGEN PER KILOGRAM OF COOLANT. EXTENSIVE DEGASSING OF THE REACTOR COOLANT DURING THE TIME PERIOD FROM APRIL 2 THROUGH APRIL 8 IS BELIEVED TO HAVE SIGNIFICANTLY REDUCED THE CONCENTRATION OF DISSOLVED HYDROGEN. HOWEVER, SOME HYDROGEN GAS WAS BELIEVED TO HAVE BEEN TRAPPED IN THE CONTROL ROD DRIVE MECHANISMS (CRDMs) AND HAS NOT READILY DISSOLVED INTO THE REACTOR COOLANT. SO, ON APRIL 9TH, THE REACTOR COOLANT SYSTEM PRESSURE WAS CYCLED TO PROGRESSIVELY LOWER PRESSURES, REACHING A MINIMUM PRESSURE OF 411 PSIG. THIS EXPANDED THE GAS TRAPPED IN THE CONTROL ROD DRIVES AND ALLOWED IT TO BE ENTRAINED IN THE RC FLOW. THE AC NOISE SIGNALS ON THE REACTOR COOLANT PRESSURE TRANSMITTER CONFIRMED THAT BUBBLES WERE RELEASED EACH TIME THE PRESSURE REACHED A NEW LOW. (BUBBLES APPARENTLY ALTERNATE THE NOISE SIGNAL AND REDUCE THE PEAK-TO-PEAK FLUCTUATION). THEREFORE, IT IS CLEAR THAT, AT PRESSURES ABOVE 411 PSIG, THE GAS WILL BE COMPRESSED FAR BACK INTO THE CRDMs AND THAT THE REACTOR COOLANT SATURATION PRESSURE IS BELOW 411 PSIG. SINCE THE SOLUBILITY OF HYDROGEN WILL DECREASE AS THE TEMPERATURE DECREASES, NATURAL CIRCULATION MUST BE PERFORMED AT A PRESSURE SUFFICIENTLY ABOVE 411 PSIG TO ASSURE THAT ANY DECREASE IN SOLUBILITY DUE TO TEMPERATURE IS OFFSET BY THE SOLUBILITY INCREASE DUE TO PRESSURE. IF THE MINIMUM TEMPERATURE EXPECTED DURING NATURAL CIRCULATION IS 140°F, AN OPERATING PRESSURE OF 600 PSIG OR GREATER WILL ASSURE THAT NO BUBBLES ARE FORMED, EVEN IF IT IS ASSUMED THAT REACTOR COOLANT IS PRESENTLY SATURATED AT 411 PSIG. ACTUALLY, THE REACTOR COOLANTS HYDROGEN SATURATION PRESSURE IS EXPECTED TO BE SIGNIFICANTLY BELOW 411 PSIG, BUT THIS WILL NOT BE ABLE TO BE

PROVEN BY PRESSURE REDUCTIONS DUE TO NPSH LIMITATIONS ON THE RC PUMPS. IN ORDER TO DETERMINE THE ACTUAL SATURATION LIMIT, PRESSURIZED REACTOR COOLANT SAMPLES WILL HAVE TO BE ANALYZED FOR DISSOLVED HYDROGEN.

THE NET PRODUCTION OF RADIOLYTIC HYDROGEN OR OXYGEN IS EXPECTED TO BE ZERO AS LONG AS THE PARTIAL PRESSURE OF HYDROGEN IN THE REACTOR COOLANT SYSTEM IS KEPT IN THE RANGE OF 5 TO 15 PSI (REF. 1).

ADDITIONAL EQUIPMENT IS NEEDED TO ASSURE ADEQUATE RCS DEGASSING CAPABILITY TO REMOVE ENOUGH GAS FROM THE SYSTEM TO ASSURE EVENTUAL DEPRESSURIZATION FROM 600 PSIG TO ATMOSPHERIC PRESSURE WITHOUT INTERRUPTING COOLANT FLOW.

<u>REFERENCES</u>

1. WATER COOLANT TECHNOLOGY OF POWER REACTORS, BY PAUL COHEN, GORDON AND BREACH SCIENCE PUBLISHERS OF NEW YORK, 1969

3.6 PRESSURE-TEMPERATURE CONSIDERATIONS

FERENCES: 1) J.H. TAYLOR TO DISTRIBUTION, SAME SUBJECT, 4/9/79, 8:53 P.M.

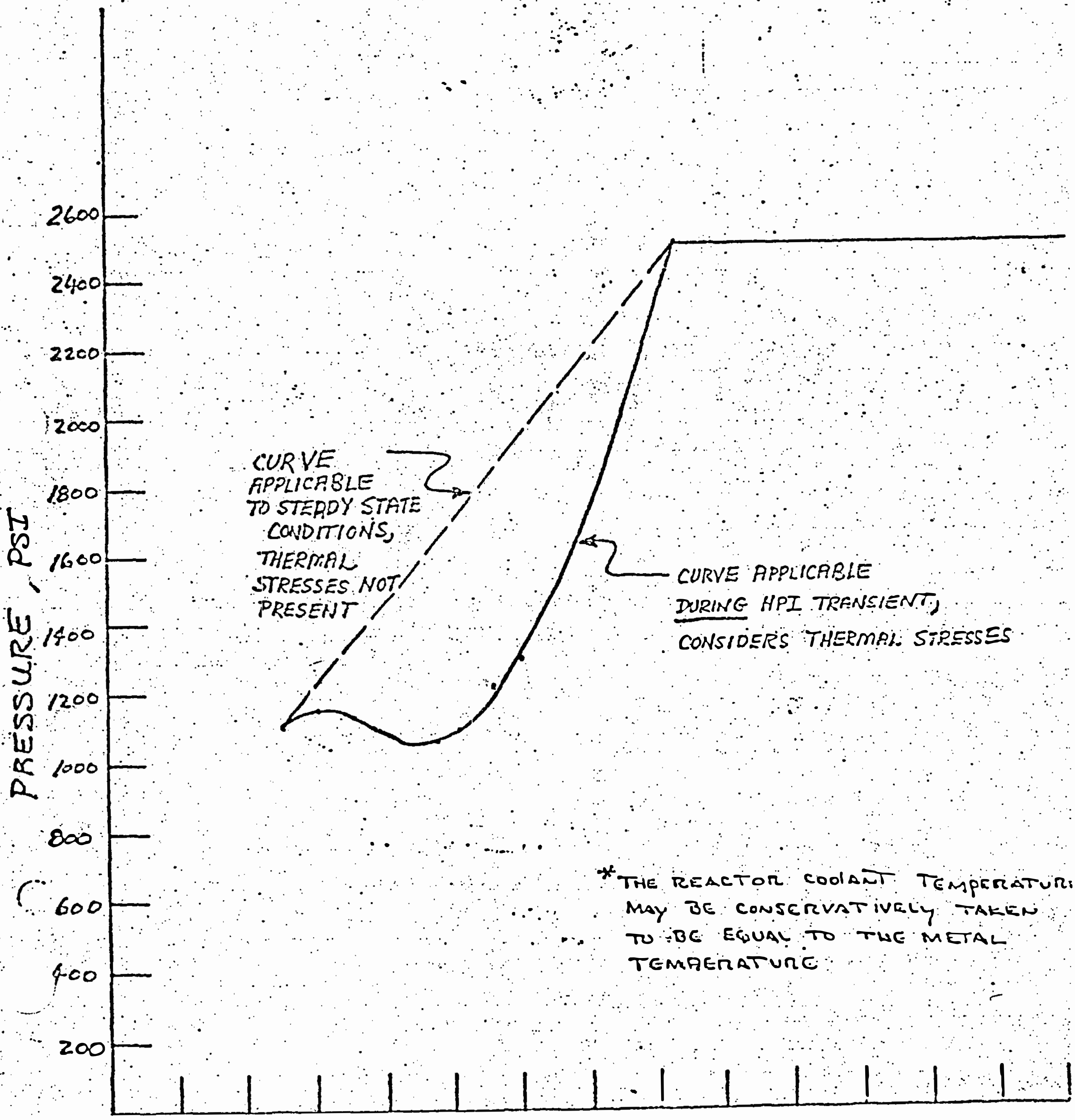
2) C.E. HARRIS TO C.W. PRYOR, "P.T. LIMITS FOR LONG TERM COOLING," 4/10/79, 5:50 P.M.

IN RESPONSE TO REFERENCE 1), THE FOLLOWING STATEMENT IS PROVIDED AS INPUT TO SECTION 3.8 OF THE SUBJECT SER. THIS INPUT IS BASED ON THE ANALYSIS RESULTS DOCUMENTED IN REFERENCE 2).

> "BASED ON FRACTURE MECHANICS ANALYSES OF THE TMI-2 REACTOR VESSEL, PRESSURE-TEMPERATURE LIMITS FOR LONG TERM COOLING OPERATION HAVE BEEN ESTABLISHED. THE ANALYSES WERE CONDUCTED IN ACCORDANCE WITH APPENDIX G TO SECTION III OF ASME CODE FOR ACCIDENT CONDITIONS. THE CALCULATIONS ARE APPLICABLE FOR FLAW DEPTHS UP TO ONE QUARTER OF THE REACTOR VESSEL THICKNESS AND SHOULD CONSERVATIVELY BOUND ANY FLAWS WHICH MIGHT EXIST IN SERVICE.
>
> THESE ANALYSES CONSIDERED A WORST CASE TRANSIENT ASSOCIATED WITH HPI SYSTEM OPERATION BY CONSERVATIVELY ASSUMING THAT NO MIXING OF HPI AND REACTOR COOLANT WATER OCCURS IN THE INLET PIPING. FOR THIS CASE, THE REACTOR VESSEL INLET NOZZLE IS THE GOVERNING WELD.
>
> A PLOT OF THE ALLOWABLE PRESSURE-TEMPERATURE ENVELOPE IS ATTACHED. THE SYSTEM WILL BE CONTROLLED DURING LONG TERM COOLING OPERATION TO ENSURE THAT THE PRESSURE-TEMPERATURE RESTRICTIONS ARE NOT VIOLATED."

FRB:nw
ATTACHMENT

PREPARED BY: C. E. Harris
4/10/79



CURVE APPLICABLE TO STEADY STATE CONDITIONS, THERMAL STRESSES NOT PRESENT

CURVE APPLICABLE DURING HPI TRANSIENT, CONSIDERS THERMAL STRESSES

*THE REACTOR COOLANT TEMPERATUR
MAY BE CONSERVATIVELY TAKEN
TO BE EQUAL TO THE METAL
TEMPERATURE

3.7 CORE MELT CONSIDERATIONS

UNDER THE CONDITIONS OF NATURAL CIRCULATION, THE CORE WILL BE SURROUNDED BY COLD WATER NEAR 100°F. THE POSSIBILITY OF CORE MELT IS CONSIDERED TO BE REMOTE UNDER THESE CONDITIONS. WITH THE CURRENT LOW DECAY HEAT RATE AND WITH APPROPRIATE MONITORING OF INCORE THERMOCOUPLES, THERE WILL BE SUFFICIENT EARLY WARNING SIGNALS TO PREVENT CORE DAMAGE.

A DETAILED DISCUSSION OF CORE MELTING POINT, ASSESSMENT OF ORIGINAL FUEL DAMAGE CONDITIONS AND EARLY WARNING SIGNALS IS PROVIDED IN ATTACHMENT 5. BASED UPON THIS ASSESSEMENT AND THE USE OF DETAILED ACCEPTANCE CRITERIA FOR THE TRANSITION TO NATURAL CIRCULATION, IT IS METROPOLITAN EDISON COMPANY'S VIEW THAT NO PROBLEMS EXIST WITH RESPECT TO THE POTENTIAL FOR CORE MELT.

3.8 CORE THERMAL BEHAVIOR

THE CORE THERMAL BEHAVIOR FOR VARIOUS POTENTIAL MODES OF OPERATION IS DISCUSSED IN ATTACHMENT 4. ATTACHMENT 4 ALSO DISCUSSES THE USE OF INCORE THERMOCOUPLES, CORE BLOCKAGE CONSIDERATIONS, THERMAL HYDRAULIC EVALUATION OF NATURAL CIRCULATION AND ANALYSIS OF VARIOUS ALTERNATIVES. BASED UPON THE INFORMATION IN ATTACHMENT 4, METROPOLITAN EDISON COMPANY CONCLUDES THAT THE PLANT CAN BE SAFELY OPERATED IN THE RECOMMENDED LONG TERM COOLING MODE.

TMI-2 CRITICALITY EVALUATION

3.9

Introduction

Evaluations of core subcriticality and potentially critical fuel configurations were begun soon after the TMI-2 incident. The analysis covered a broad spectrum of fuel configurations, ranging from the intact core to homogeneous solutions of uranium and water. Boron concentrations necessary to maintain subcriticality for the various postulated configurations were determined. The following is a description of the methods of analysis and results from the criticality evaluations.

1. Fuel in Core Region

The analysis for the various possible configurations of fuel in the core region was divided into two areas (1) fuel rods intact and (2) successive "slumping" of fuel pellets to an ultimate slab of pellets.

1.1 Fuel Rods Intact

PDQ-07 calculations were performed for the TMI-2 core at the core burnup on March 28, 1979 (88.3 EFPD) for several assumptions at cold conditions. The results are summarized in Table 1.

TABLE 1

Boron Requirements for TMI-2 Core for Cold (70°F) Shutdown

Temperature, °F	Control Rods	Keff	Boron, ppm
70	All Rods Out	.95	2155
70	All Rods Out	.99	1795
70	All Rods In	.95	1705
70	All Rods In	.99	1385

The boron concentrations listed above are based upon the following assumptions:

a. Guide tubes, spacer grids and cladding remain intact. The fuel is the basic structure as originally loaded into the core region.

b. No credit taken for Lumped Burnable Poison (LBP).

c. Xenon fully decayed.

d. No credit taken for Samarium buildup since shutdown; equilibrium Samarium at hot full power was assumed.

e. An additional 1% ΔK/K was included to provide a conservative prediction at 70°F.

The values in Table 1 are possibly non-conservative for higher temperatures because the core has a positive moderator coefficient of ∿ +.8 x 10^{-4} ΔK/K/°F at 2100 ppm boron, for an intact core.

Predicted keff values for the present temperature and boron conditions based on the above assumptions are shown in Table 2.

TABLE 2

Core Keff Values for Present Conditions, 280°F and 2100 ppm

Temperature, °F	Boron, ppm	Control Rods	Keff*
280	2100	All Rods Out	.97
280	2100	All Rods In	.92

* Predicted values from Table 1 using a +.8 x 10^{-4} ΔK/K/°F moderator coefficient at 2100 ppm boron.

1.2 Fuel Pellets Slumping

Criticality studies were performed for fuel pellets free from the cladding dropping onto the spacer grids. The grids were successively assumed to fail resulting in various slabs of "slumped" fuel atop the lower fuel segments until ultimately one large slab of all the fuel pellets rested atop the lower grid.

KENO-IV version 2 utilizing 123 g XSDRN cross-section sets were used for all calculations.

The core was modeled in seven symmetric planes with top and bottom reflector, but infinite in x,y. Thus, a slab reactor was calculated for 160°F moderator temperature, 900 psi. (An extra conservatism was the top plane having 21 inches of fuel as opposed to the 16 inch actual). The initial axial geometry assumed full (21") pellet stack heights surrounded by borated water. This condition represented a UO_2/H_2O volume ratio of .307. Subsequent calculations assumed that as the fuel slumped into each of the seven planes, defined by the spacer grid, a UO_2/H_2O volume ratio of 0.63 occurred within the fuel water mixture. The assumed volume fraction of .63 is based on measured packing fraction data at CNFP. The calculations assumed BOL isotopics, no control rods, no LBPs, no fission products, no structural material, only borated water and fuel pellets. A calculation was also made assuming that all seven planes had slumped into one slab. Fuel enrichment for all cases was assumed to be the average of the three fuel batches, ie 2.60 w/o U-235.

For those conditions which produced a critical system at the initial boron concentration of 2100 ppm (approximate level in RC system), the boron was increased until a subcritical array could be predicted. The fuel arrangement for the various cases is shown in Figure 1 and the results are summarized in Table 3.

TABLE 3

Criticality Calculations for Four "Slumped" Fuel Configurations in Core Region

Boron, ppm	Keff Case 1 (Grids Intact)	Case 2 (2 Grids Fail)	Case 3 (5 Grids Fail)	Case 4 (Total Slump)
2100	0.824 ± .004	1.016 ± .005	1.062 ± .004	1.078 ± .004
3000	---	.990 ± .004	1.035 ± .005	1.038 ± .004
4000	---	---	.992 ± .004	1.003 ± .005

An additional KENO calculation was made for the more credible situation of the upper 3 grids failing and the fuel slumping atop the fourth grid. For this situation the fuel below the fourth grid was assumed to be standing in its basic configuration. The array was assumed infinite in x-y direction. The slumped fuel pellets in the upper region was assumed to be packed in the most optimum fuel/water ratio of fuel volume fraction = 0.55. Whereas, the standing fuel pellets were assumed to be at their initial volume fraction of 0.307. Since the KENO code cannot calculate two mixtures of fuel with different cell pitch the above scenario was bounded by two KENO cases. The first case assumed normal cell pitch (1.44 cm) but had to use a larger pellet O.D. in the slumped region to produce the 0.55 volume fraction fuel. Case 2 assumed the normal pellet O.D in the slumped region with a cell pitch of 1.12 cm and a correspondingly smaller pellet O.D. in the lower intact fuel rods. Figure 2 shows the fuel configuration and Table 4 summarizes the results.

TABLE 4

Criticality Calculations for 3 Grids Failing

Case	Cell Pitch	Boron, ppm	Temp, °F	K∞
1	1.44 cm	3000	280	1.000
2	1.12 cm	3000	280	1.019

Comparing Case 2 above with the total slump Case 4 from Table 1, the single slab is slightly more reactive.

An assessment of the inherent conservatism and nonconservatism of the KENO studies (Tables 3 & 4) is presented below:

1. No fuel depletion considered. Fuel depletion will reduce K∞ by <u>∿ -2.5% ΔK/K</u>.
2. No radial leakage considered. Radial leakage will reduce K∞ by <u>∿ -3.0% ΔK/K</u>.
3. No credit taken for Ag-In-Cd Control Rods.
4. The total slump case (slab) assumed a Volume Fraction of fuel equal to 0.63, the packing fraction. However, this is <u>not</u> necessarily the optimum configuration for criticality (Section). If the most optimum fuel/water ratio is formed, the configuration may be more reactive by <u>∿ +2.5% ΔK/K</u>.

1.3 Fuel Pellets - Sphere

Calculations were performed to predict the criticality for the assumption that all the fuel pellets collapse to form an optimum spherical configuration in the reactor vessel plenum. The multiplication factor for an infinite array of non-depleted fuel pellets in the optimum water ratio for the core average enrichment 2.6 w/o U-235 is presented in Table 5. Also shown in Table 5 is the expected leakage reactivity for the hypothetical sphere.

Enrichment w/o U235	Temp, °F	Boron, ppm	Optimum VF Fuel	K∞	ρ Leakage
2.6	280	2100	0.52	1.091	-1.4% ΔK/K
2.6	280	3000	0.63	1.028	-1.6% ΔK/K

These data demonstrate that after subtracting the fuel depletion reactivity (-2.5% ΔK/K) and the expected leakage, a hypothetical sphere can be critical at 2100 ppm but will be subcritical at a boron concentration of 3000 ppm.

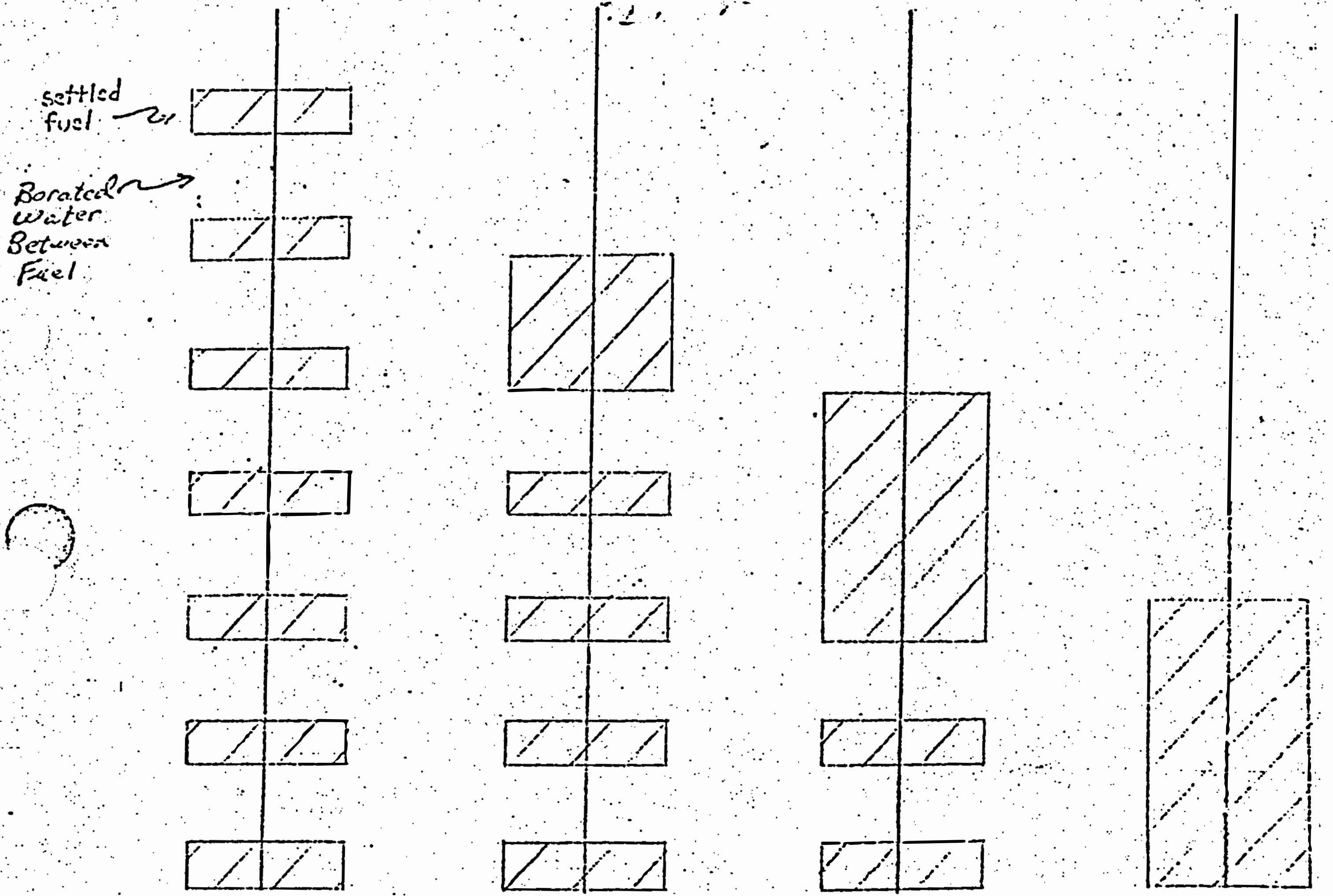
2. Parameter Study - Infinite Media

Heterogeneous mixtures of fuel and water in an infinite array were analyzed with the NULIF code. The volume fraction of fuel, fuel particle size, boron concentration, and temperature were varied. The calculations were performed for a fuel enrichment of 2.6 w/o U-235, corresponding to the average fuel enrichment in the core. No credit was taken for fuel burnup or fission product buildup.

Figure 3 shows reactivity as a function of the uranium volume fraction for several boron concentrations and 2 different temperatures. The optimum fuel/water ratio increases with increasing boron concentration. The most likely fuel volume fraction for intact fuel pellets settling in a system cavity has been experimentally determined to be .63. Figure 4 shows reactivity as a function of fuel particle size for a fuel volume fraction of .63. As can be seen by comparing Figures 3 and 4, reactivity is much more sensitive to the volume fraction of uranium in the system than to particle size. For fuel volume fractions in the range where K∞ can be greater than 1.0 with 2100 ppm boron in the system, intact fuel pellets uniformly distributed in the system were found to be more reactive than an equal amount of fuel dispersed in smaller sized particles or homogeneously mixed with the coolant. Figure 5 shows reactivity as a function of boron concentration for uranium volume fractions of .52 and .63 at 280°F. At lower boron concentrations, the lower uranium volume fractions are more reactive, at higher boron concentrations, the larger uranium volume fractions are more reactive. Figure 6 shows reactivity as a function of boron concentration for a uranium volume fraction of .63 and 3 different particle sizes.

All the above calculations were performed at 1000 psia. A drop in system pressure to 300 psia would be equivalent to a 10°F rise in the moderator temperature. For intact pellets at a volume fraction of .63 the temperature coefficient varies from $-.8 \times 10^{-4}$ Δρ/°F at 2100 ppm to $-.5 \times 10^{-4}$ Δρ/°F at 4000 ppm. Thus a drop in system pressure from 1000 psia to 300 psia would result in a slight decrease ($<$0.1% in reactivity.

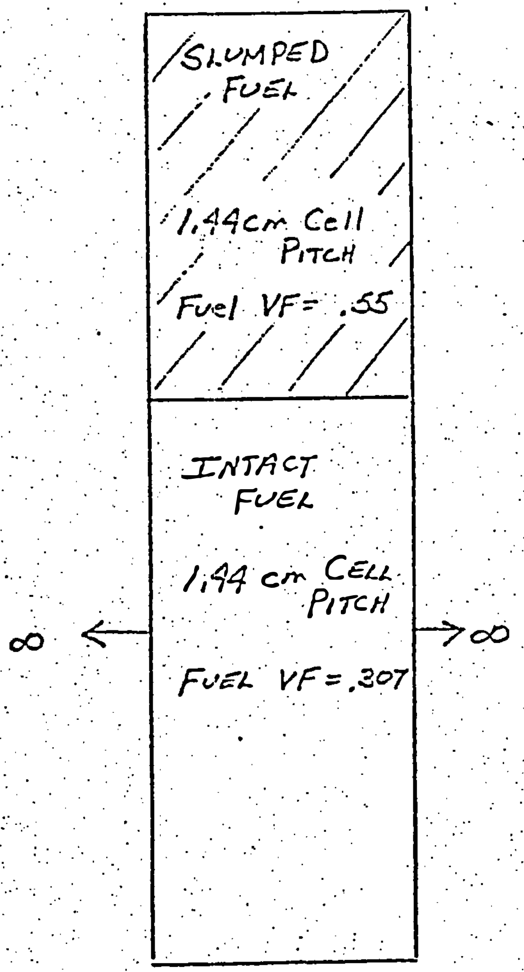
FIGURE 1 - FUEL CONFIGURATION IN KENO



	CASE I	CASE II	CASE III	CASE IV
Pitch =	1.0425	1.0435	1.0485	1.0485
PPM Boron				
2100	K_∞=0.824±.004	K_∞=1.016±.005	K_∞=1.062±.004	K_∞=1.073±.004
3000		K_∞=.990±.004	K_∞=1.035±.005	K_∞=1.033±.004
4000			K_∞=.992±.004	K_∞=1.003±.005

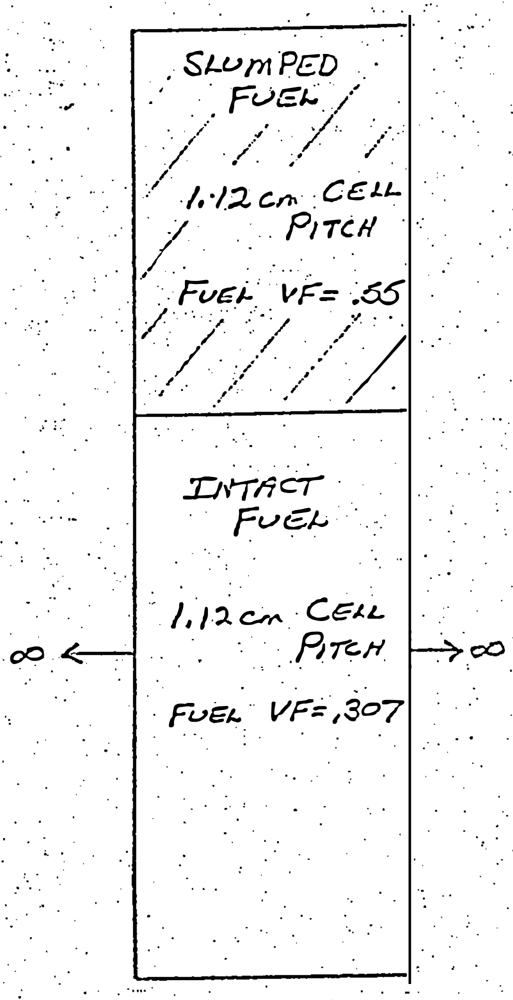
Figure 2
KENO CONFIGURATION FOR THREE (3) GRID COLLAPSE

CASE 1



K= 1.000

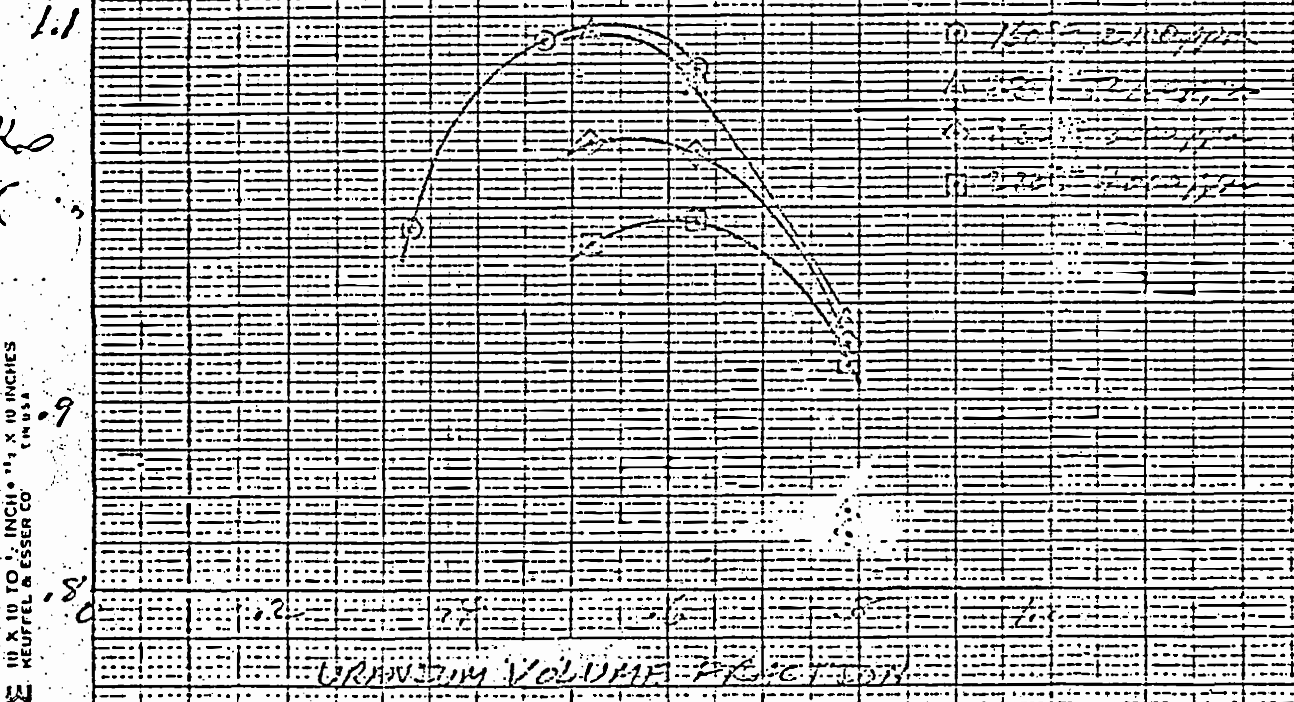
CASE 2

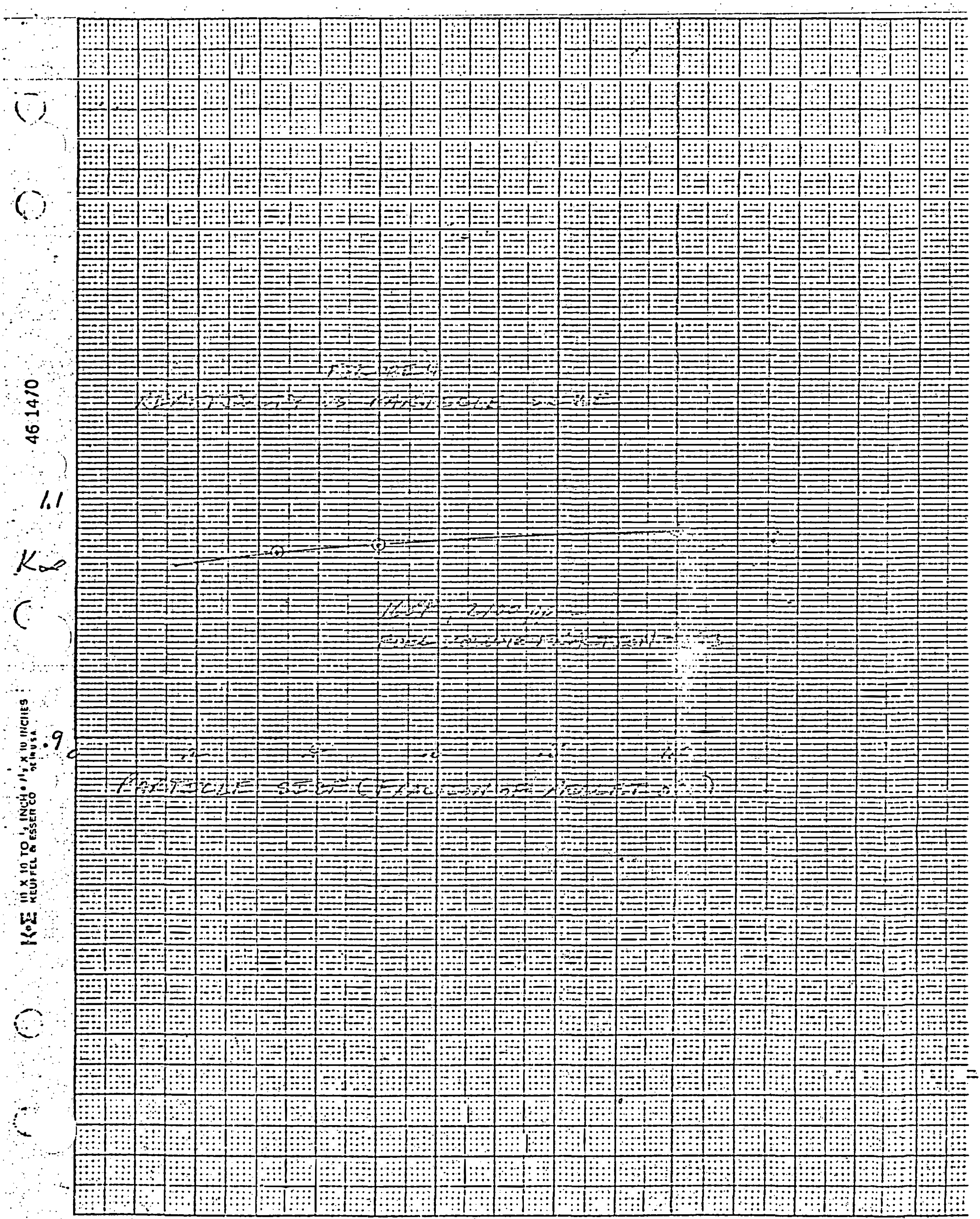


K= 1.019

FIGURE 3

REACTIVITY vs FUEL VOLUME FRACTION
FOR FUEL [illegible] [illegible] DISPERSED
IN [illegible] [illegible]




VS PARTICLE SIZE
1.1
K∞
.9
PARTICLE SIZE

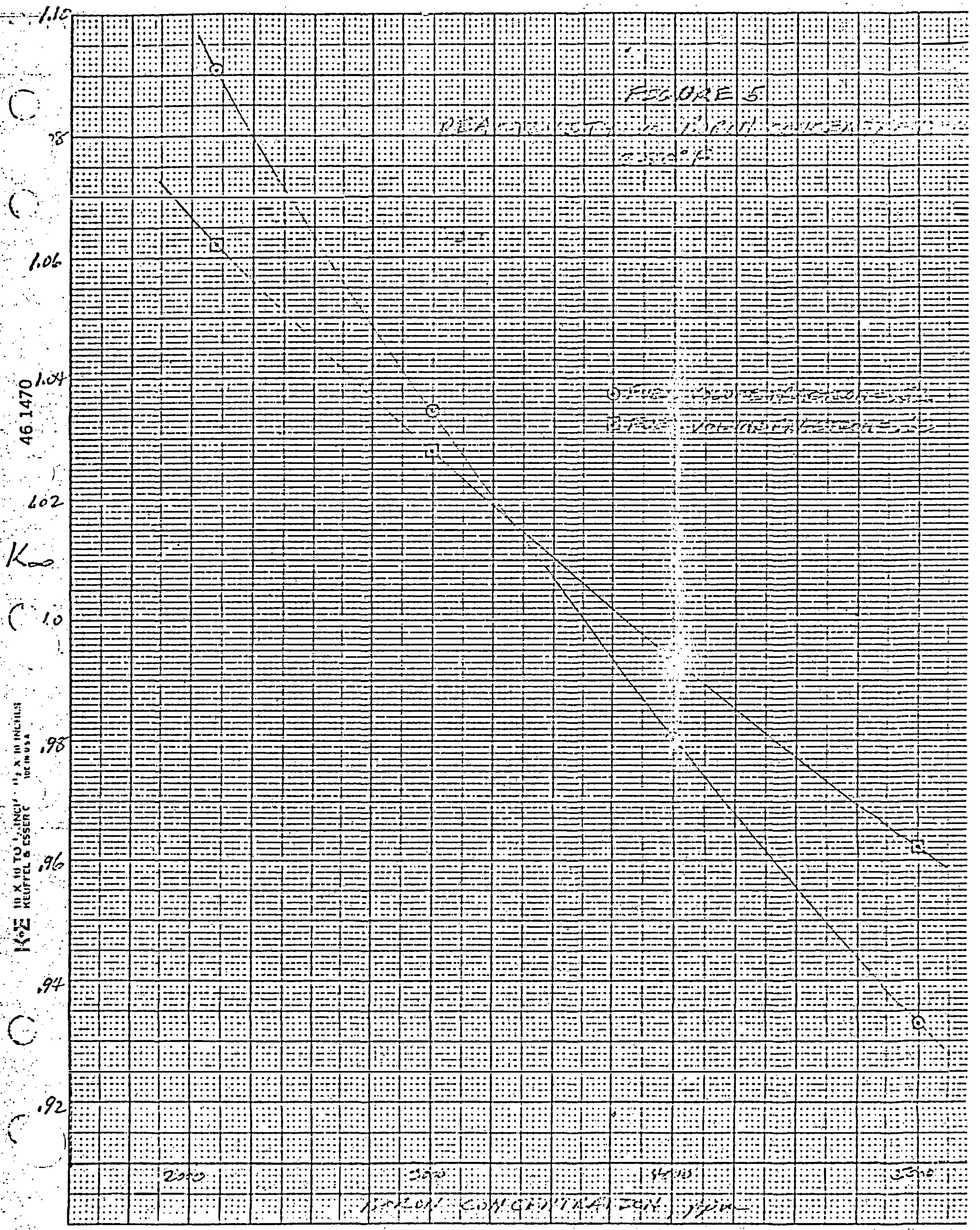
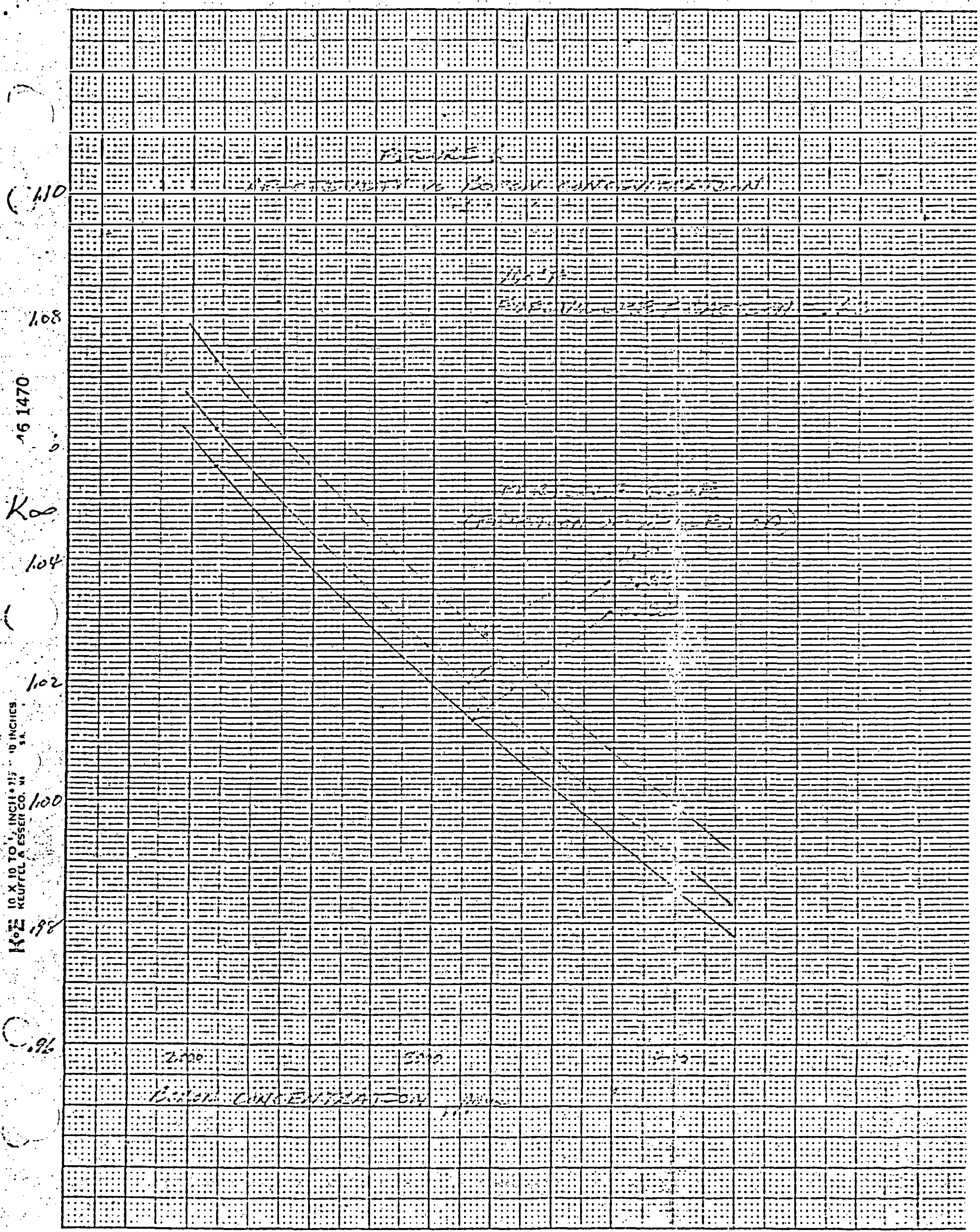


FIGURE 5
1.10
1.08
1.06
1.04
1.02
1.0
.98
.96
.94
.92
K∞
2000
3000
4000
5000
BORON CONCENTRATION, ppm
46 1470
K+E 10 X 10 TO 1/2 INCH KEUFFEL & ESSER CO.


1.10
1.08
K∞
1.04
1.02
1.00
.98
.96
2000
CONCENTRATION

Attachment 1

Babcock & Wilcox April 10, 1979

A SUMMARY OF NATURAL CIRCULATION ALTERNATIVES FOR LONG-TERM CORE COOLING AT TMI-2

prepared by

B. A. Karrasch

TABLE OF CONTENTS

INTRODUCTION:

The TMI-2 long-term cooling mode proposed by B&W utilizes RCS natural circulation core cooling with the steam generator secondary side in a solid flowing water condition for heat removal. This ultimate decay heat removal mode is a key feature of the following proposed sequence of events to achieve a stable, cold safe shutdown condition at TMI-2.

Phase I: Reduce RCS temperature to approximately 230°F by steaming the A OTSG through the turbine bypass system with one RC pump running and RC pressure controlled to a value greater than the pump NPSH, using the pressurizer in a normal mode.

Phase II: With the A OTSG steaming, the ~~B~~ A OTSG (and closed secondary system yet to be installed) will be slowly filled solid with water and a transition will be made to remove RCS decay heat with the ~~B~~ A OTSG solid. The ~~A~~ B OTSG will be isolated and the RCS temperature will be reduced to approximately 100°F with the ~~B~~ A OTSG. Reactor coolant flow and pressure conditions will remain the same as Phase I.

Phase III: The ~~A~~ B OTSG (and closed secondary system to be installed) will be filled solid with water and a transition made to remove RCS decay heat with both A and B steam generators flowing solid with 100°F feedwater. Reactor coolant flow and pressure conditions will remain the same as Phase I.

Alternate to Phases II and III: The B OTSG will remain isolated and the A OTSG will continue to remove RCS decay heat during the transition from a steaming secondary at 230°F to a solid water secondary at 100°F. This scheme would not utilize the B OTSG; however, the transition operation is more difficult with respect to steam line water hammer and maintenance of a stable RCS temperature and pressure. Change

Phase IV: With the reactor coolant system at approximately 100°F using normal RC pressure control and secondary side heat removal with a solid system (between 3000 and 5000 gpm at 100°F), the reactor coolant pump will be tripped and natural circulation core cooling will commence. Acceptance criteria for core cooling will be established and long-term cooling of the core will be maintained with natural circulation.

Phase V: With natural circulation for core cooling and a solid secondary system for OTSG heat removal, RC pressure can be reduced to a minimum value required to maintain the RCS in a sub-cooled condition. Our plan is to fill the primary system solid, including the pressurizer, and maintain pressure control with a makeup pump designed ? for such an application. To maintain a stable sub-cooled margin, we envision a long-term RCS pressure between 20 and 50 psia.

During the past week, analysis and testing has been underway at NPGD and the Alliance Research Center to define and understand the various alternatives available for core heat removal with natural circulation. The analyses were directed toward obtaining data to define an optimum long-term cooling mode. Several of the important considerations include:

1. Core natural circulation cooling, for various core ΔP configurations, with one or two OTSG's in service.

2. OTSG natural circulation cooling performance with various secondary side water flowrates to the unit through the main or auxiliary feedwater nozzles.

3. Expected transient performance during the transition from forced to natural circulation including specific acceptance criteria for the operator to determine if adequate core cooling is achieved.

SUMMARY

The preferred mode for natural circulation core cooling is to use both OTSG's solid on the secondary side, with a flowrate of 3000 gpm entering the OTSG through the main feedwater nozzles and exiting the unit through the steam outlet nozzles. This mode will provide a maximum core flowrate (> 800,000 lb/hr), a minimum core ΔT (< 30°F), and a minimum reactor coolant average temperature (< 120°F for a 100°F OTSG feedwater temperature). The secondary side OTSG cooling is a stable, forced convection mode, which transfers all the primary system energy above a tube elevation of 30 feet, thereby providing a high column of cold water for enhancing the primary side natural circulation. This mode provides a driving head similar to that obtained with the OTSG steaming with a secondary side level at 30 feet.

The solid secondary side mode of operation has a distinct advantage over a steaming mode in that a much lower reactor coolant system temperature can be achieved. The solid configuration will result in an RCS temperature very close to the OTSG feedwater temperature (approximately 100°F); the steaming mode of operation can only obtain RCS conditions equivalent to the saturation temperature at the lowest achievable steam pressure (approximately 230°F). In addition, the use of the main nozzles for OTSG feedwater addition has been shown to yield a predictable and uniform primary system heat removal suitable for natural circulation. The use of the auxiliary nozzles for OTSG feedwater addition, with water exiting the main nozzles, should also remove the primary heat at an elevated point in the unit. However, the flow distribution and uniformity of cooling is uncertain and the feedwater flowrates are limited by system design and OTSG tube crossflow velocity concerns. In addition, major secondary plant modification would be required to implement reverse flow through the OTSG main feedwater nozzles. Testing

performed on the 19-tube steam generator at the Alliance Research Center confirms that feedwater addition through the main nozzles with water exiting the steam outlet nozzles is the preferable mode for natural circulation.

The advantage of using both steam generators instead of only one is an increase in the core natural circulation core flowrate of 10 to 20 percent and a decrease in the core outlet temperature of about 5°F. Extensive analysis has been independently performed at NPGD to confirm that the difference between using one or two OTSG's is not significant from a natural circulation standpoint; one OTSG in service will provide adequate core cooling. We believe, however, that the uncertainty in local core conditions and cooling requirements, the need for heat exchanger redundancy in the long-term cooling mode, and the ease of transition and operation with a solid water secondary for two OTSG's versus one, makes operation with two loops a superior mode.

The effect of a greater core resistance on the natural circulation cooling capability has been evaluated and deemed acceptable. A core resistance of 60 times the normal value has been assumed in the calculations, and the reported results are acceptable for either one or two steam generators in operation. The difference between a normal core resistance and a core resistance 60 times normal (indicating a significant blockage) is a factor of two in core flow and ΔT. This favorable result is due to the offsetting effects of system resistance, flowrate, and temperature difference to sustain a stable natural circulation condition.

Expected transient performance during the transition from forced primary system flow to natural circulation is predictable and stable. From an initial condition with the RC pump running and primary and secondary temperature approximately 100°F, a stable natural circulation condition will be achieved within a half hour following the pump trip. The cold leg temperature will decrease slightly (due to the pump power loss) and remain stable at about

100°F. The core outlet temperature will increase by about 20-30°F within 10 minutes and be observed on the hot leg RTD in less than 20 minutes. During the first hour after the pump trip, the reactor vessel heatup with no primary system flow would only be 100°F. Acceptance criteria during the first hour of natural circulation will be provided to the operator and primary system pressure will be maintained to assure that the reactor core outlet temperature remains 100°F sub-cooled at all times. When the operator observes the increase in hot leg temperature indication, a stable natural circulation condition will be confirmed.

DISCUSSION

A. Steady State Analysis

The results of the steady state natural circulation analyses performed to date are presented in Table 1. Four different reactor configurations were evaluated to determine the sensitivity of various conditions and assumptions on the natural circulation core flowrate and core temperature drop. The configurations studied include:

1. Two loop operation with both steam generators steaming at 230°F (20 psia) at a 30-foot secondary side level (95% on operate range). This configuration is similar to that which has been tested on the Oconee Units and forms the basis for a considerable amount of analysis at NPGD. These cases have been used to provide a benchmark on the OTSG heat transfer characteristics for development of a driving head and for confirming RCS loop ΔP characteristics. Figure 1 illustrates the sensitivity of the core natural circulation flowrate with loop ΔT (the driving head gain) and loop pressure drop (the driving head loss). The flowrate will seek a stable natural circulation condition based upon the loop ΔP and the resultant core ΔT. The key to obtaining a maximum flowrate is to remove the primary system heat (i.e., change t_{hot} to T_{cold}) at as high an elevation as possible in the steam generator. Our testing and analysis confirms that the primary heat is all transferred above the liquid/steam interface (i.e., the level) on the secondary side of the OTSG. The calculational results presented conservatively assume that the primary system temperature change occurs as a step change at the height of the OTSG operate range level.

The effect of increased core ΔP has also been evaluated to determine the core flowrate and temperature drop sensitivity. The following types of analyses have been performed at NPGD to conclude that the TMI-2 core resistance in its current configuration could be as high as 60 times the nominal value, indicating a high degree of core blockage:

a. Core ΔP calculations based upon a postulated core configuration.

b. A comparison of RCS flow meter readings, with one pump running, before and after the TMI-2 incident.

c. A conservative estimate of core flowrate and pressure drop in the current TMI-2 core configuration using the actual decay heat level and the difference between the cold leg temperature and the core outlet temperature as determined by the core outlet thermocouples [i.e., core flow = $\frac{Q_{decay}}{\Delta T}$ and $\Delta P = (\text{core flow})^2$].

These analyses have provided a range of core ΔP values which have been included in the evaluations described in Table 1. The effect of increased core ΔP on the natural circulation flowrate is illustrated on Figure 2. The analyses have shown that the natural circulation flowrates are adequate with the core in its current configuration.

2. Single loop operation with OTSG A steaming at 230°F (20 psia) at a 30-foot secondary side level-OTSG B isolated. This configuration has been evaluated to provide a comparison of two loop versus single loop operation. The single loop calculations confirm that the net core flow will be 10 to 20% less in this configuration than with both steam generators in service. The resultant core ΔT will

increase about $5^{\circ}F$ (depending upon the decay heat level) and is still acceptable for core cooling. These analyses were performed to confirm an acceptable condition should an emergency situation require an immediate transition to natural circulation prior to the planned sequence to a solid steam generator secondary side.

3. Single loop operation with OTSG A in a solid secondary side mode with water addition through the main feedwater nozzles. Steam generator heat transfer analyses and testing have confirmed that a 3000 gpm feedwater flowrate to the main feedwater nozzles will provide a primary to secondary heat transfer characteristic similar to that achieved with the OTSG steaming with a 30-foot water level. The majority of the heat removal occurs above the 30-foot level in the OTSG with a 3000 gpm flowrate. If the flowrate is increased to 5000 gpm, the driving head for natural circulation is further improved to about 35 feet. The calculational results confirm that adequate natural circulation flow and core ΔT can be obtained with a single steam generator operating in a solid condition.

   Additional analyses were performed in this configuration to determine the effect of reduced core decay heat levels. These cases were run at 2 and 3 MW to provide a comparison with values of core flow and ΔT at 5 MW. As can be seen from Table 1, natural circulation core flowrate and core ΔT are both reduced for lower decay heat values, and core cooling remains acceptable.

4. Two loop operation with both steam generators in a solid secondary mode with feedwater addition through main nozzles at 3000 gpm. This is the preferred mode for long-term cooling at TMI-2 and the results are very similar to the two OTSG's steaming case. Again, the solid flowing water secondary system at 3000 gpm induces

a high heat transfer interface in the OTSG's and acceptable core natural circulation cooling is achieved.

The steady state natural circulation analysis has resulted in the following conclusions with regard to long-term cooling at TMI-2:

1. Adequate core cooling with natural circulation can be achieved with either one or two steam generators in service.

2. An increased core resistance due to blockage decreases the natural circulation flowrate and increases the core ΔT. However, it has been shown that acceptable core flow and ΔT can be maintained with significant increases in the core resistance due to blockage.

3. Adequate natural circulation flowrate can be achieved with the steam generator(s) in a steaming or solid mode if the effective heat transfer height is maintained at 30 feet or greater using a high level for steaming (30 feet) or a high flowrate for solid (3000 gpm).

4. Adequate natural circulation cooling can be maintained at reduced core decay heat levels.

B. Transient Analysis

The results of the forced flow transition to natural circulation cooling are presented in Figures 3 and 4. Transient analyses were performed at core decay heat levels of 2 and 3 megawatts to better understand the time dependent responses of core flowrate and temperature change following the loss of forced cooling. The bases for the analyses are as follows:

Core Power - 2 and 3 Megawatts
Reactor Coolant Pump Trip at Time 0
OTSG A Solid with 100°F Feedwater into the Main Nozzles at 3000 gpm
OTSG B Isolated
Core Resistance Factor - 60

The transient responses of core flow and temperature confirm that a smooth transition to natural circulation is achievable. Following the loss of forced flow, the reactor vessel heatup slowly induces a temperature gradient between the reactor vessel and upper OTSG and natural circulation occurs with no operator action. The core flowrate reaches a minimum about 1 minute into the transient and reaches a stable condition between 10 and 20 minutes. The core outlet temperature begins to increase and reaches a maximum value 4 to 5 minutes into the transient and a stable condition at about 10 minutes. There is about a four minute time delay in the response of the hot leg temperature measurement due to the approximately 1000 feet^3 in the reactor vessel upper plenum and hot leg piping. The cold leg temperature drops slowly to closely match the OTSG feedwater temperature due to the loss of the approximately 5 megawatts of pumping power.

The transient natural circulation analyses have resulted in the following conclusions with regard to long-term cooling at TMI-2:

1. A smooth transition from forced flow cooling to natural circulation can be achieved by tripping the reactor coolant pump and observing core outlet temperature. There is no reason to slowly reduce the RC pump speed for a more gradual transition to natural circulation.

2. The reactor coolant system flow and temperature will reach an equilibrium value within the first 1/2 hour of the transient; the response of the hot leg temperature measurement occurs within 5 minutes after the core outlet temperature changes.

C. OTSG Test Program

The ability to achieve and maintain a stable natural circulation flowrate is dependent upon the elevation difference between the heated core outlet temperature and the transition to cold leg temperature in the OTSG tubes. This transition point in the OTSG is in turn dependent upon the heat transfer characteristics of the unit. If the primary to secondary heat transfer can be obtained at a high elevation in the OTSG, the driving head from the density difference will be improved and natural circulation flowrate will increase.

The primary to secondary OTSG heat transfer mechanism, while in a steaming mode is boiling at or about the level of the secondary side water. Extensive analysis and testing of the OTSG in the "pot boiling" mode has confirmed that the primary system temperature transition occurs above the level of the "boiling pot." All calculations performed with the OTSG in a steaming mode conservatively assume that the primary system cold leg temperature is available for driving the natural circulation flow at the 30 foot level.

Use of the steam generator as a water to water counter flow heat exchanger is a more desirable condition to obtain during a long-term decay heat cooling mode. The primary temperatures can be maintained much nearer the temperature of the incoming feedwater to the OTSG. In order to determine OTSG characteristics in a solid mode, a test program was conducted at the Alliance Research Center on a 19-tube, full-length, steam generator. A natural circulation flowrate of 700,000 lb/hr was simulated on the primary side and forced secondary side cooling was injected into the main feedwater nozzles; flow exited the unit through the steam outlet nozzles. Feedwater flowrates were varied from a scaled value of 100 gpm up to 5000 gpm.

The results of this test program are presented on Figure 5, a plot of feedwater flowrate versus the OTSG heat transfer elevation. Heat transfer elevation is defined as that level above which all primary system heat is transferred to the secondary system fluid. That is, the height at which one can assume the primary cold leg temperature is available for driving natural circulation. The figure shows that a heat transfer elevation of 30 feet can be obtained if the feedwater flowrate is at 3000 gpm or higher. A 30 foot elevation head in the OTSG primary is adequate to achieve natural circulation as demonstrated by the calculations in the previous section.

D. Acceptance Criteria During Operation

The success or failure of natural circulation as a core cooling mode depends upon the value of the core ΔT that can be maintained. The key objective during plant operation in this mode is to maintain a primary cold leg temperature as low as possible and observe the resultant hot leg temperature. The acceptance criteria for success of the natural circulation mode is to maintain the hot leg temperature below the saturation temperature which would cause bulk boiling. Figure 6 illustrates the proposed NPGD criteria for natural circulation: to maintain a $100^{\circ}F$ sub-cooled margin to bulk boiling using the plant instrumentation in its current degraded state. The large errors which have been imposed on the pressure and temperature instrumentation make it imperative to keep the RC pressure as high as possible at the time of pump trip. This will allow a large hot leg temperature increase to occur before boiling and assure a reasonable time period to achieve a stable natural circulation. If the RC pressure is maintained at 500 psia, the hot leg temperature can reach $340^{\circ}F$ (from its initial condition of $110^{\circ}F$) before action must be taken.

An analysis of the reactor vessel was performed to determine the potential heatup rate with zero flow into the vessel. This analysis provides a bounding bulk fluid heatup rate to indicate the amount of time available to the operator to take action before a boiling condition could occur. The high probability of achieving a stable natural circulation condition indicates that such a reactor vessel heatup could never occur. The analysis is provided to show that the operator has at least one hour to confirm natural circulation before any action must be taken.

RECOMMENDATION:

Natural circulation has been shown to be an acceptable means of heat removal for long-term cooling at TMI-2 with the core in its current configuration. Use of either one or two steam generators is feasible if the proper secondary side heat transfer characteristics are established and maintained to remove the primary energy near the top of the OTSG. In addition, the expected transition process from forced cooling to natural circulation will provide a continuous and stable core cooling condition which can be monitored and controlled by the plant operator.

B&W, therefore recommends that a planned transition to natural circulation core cooling be implemented at TMI-2 as soon as the degassing process is completed. Both steam generators should be utilized in a solid flowing water condition with approximately 100°F feedwater at 3000 to 5000 gpm entering through the main feedwater nozzles. The sequence of events for this transition, as described in the Introduction of this report should be as follows:

1. Reduce RCS temperature to 230°F with a OTSG steaming.

2. Slowly fill OTSG B solid with water and begin removing primary system energy with the B OTSG by gradually increasing feedwater flow until a stable condition is reached at 200-230°F. When a stable condition has been established, isolate OTSG A.

3. Reduce the RCS temperature to approximately 100°F by increasing the feedwater flowrate to OTSG B. Fill OTSG A solid with water and prepare for operation.

4. Slowly begin feeding OTSG A with 100°F feedwater and establish a stable condition with both steam generators removing decay heat with a 3000 gpm feedwater flowrate.

5. Establish RCS natural circulation as follows:

    a. Throttle feedwater flow to both steam generators to establish approximately 25°F ΔT between feedwater temperature and OTSG secondary outlet temperature.

    b. When a stable condition has been establsihed, trip the running reactor coolant pump and increase feedwater flowrate to both OTSG's to 5000 gpm within 3 minutes. Maintain at 5000 gpm.

    c. Maintain RC pressure at the initial condition value and observe both A and B hot leg temperatures.

    d. Compare the hot leg temperatures to the acceptance criteria on Figure 6. If the temperature exceeds the limiting value, start a reactor coolant pump.

    e. When stable natural circulation conditions have been achieved, reduce RC pressure to the proposed long-term cooling value between 20 and 50 psia.

The above sequence of events will establish a stable and safe natural circulation condition for long-term cooling at TMI-2. All starting or stopping of reactor coolant pumps should be avoided until the pump is tripped to induce natural circulation. In addition, B&W recommends that the sequence of events be implemented in a planned and controlled manner, i.e., we should not wait for a complete failure of all four RCP's before establishing natural circulation. We should, however, have an alternate decay heat removal system installed and ready for operation prior to the transition to natural circulation.

APPENDIX A

INSTRUMENTATION REQUIRED TO ACHIEVE AND MAINTAIN LONG TERM COOLING

INSTRUMENTATION REQUIRED TO ACHIEVE AND MAINTAIN LONG TERM COOLING

I. REQUIRED TO CONFIRM THE INITIATION OF NATURAL CIRCULATION

Primary System

ITEM	MEASUREMENT	RANGE OF INTEREST	DESIRED ACCURACY	BACK-UP MEASUREMENT	COMMENTS
1	Reactor Core Outlet Temperature (Incore TC's)	0-700F	± 10F	Item I.2; I.3	
2	Loop A Reactor Hot Leg Temperature	0-550F	± 10F	Item I.3	
3	Loop B Reactor Hot Leg Temperature	0-550F	± 10F	Item I.2	
4	Loop A1 or A2 Cold Leg Temperature	0-350F	± 10F	Item I.5	
5	Loop B1 or B2 Cold Leg Temperature	0-350F	± 10F	Item I.4	

II. REQUIRED TO MONITOR LONG TERM NATURAL CIRCULATION (IN ADDITION TO ITEMS IN I. ABOVE)

A. Primary System

ITEM	MEASUREMENT	RANGE OF INTEREST	DESIRED ACCURACY	BACK-UP MEASUREMENT	COMMENTS
1	Pressurizer Level	0-400"	± 40"		Comp. I.D. 0387, 0388 (Not required for solid primary.)
2	Pressurizer Temperature	0-500F	± 10F	Item II.A.3	Comp. I.D. 0389 (Not required for solid primary.)
3	Loop A Reactor Coolant Pressure	0-1000 psig	± 50 psi	Item II.A.4	Comp. I.D.0398, 0399
4	Loop B Reactor Coolant Pressure	0-1000 psig	± 50 psi	Item II.A.3	Comp. I.D. 0400

INSTRUMENTATION REQUIRED TO ACHIEVE AND MAINTAIN LONG TERM COOLING

B. Secondary System

ITEM	MEASUREMENT	RANGE OF INTEREST	DESIRED ACCURACY	BACK-UP MEASUREMENT	COMMENTS
1	Steam Generator A Level	0-600"	± 30"		Comp. I.D. 0009 (Required only if steaming.)
2	Steam Generator B Level	0-600"	± 30"		Comp. I.D. 0001 (Required only if steaming.)
3	Steam Generator A Outlet Pressure	0-300 psig	± 15 psi	Item I.4	
4	Steam Generator B Outlet Pressure	0-300 psig	± 15 psi	Item I.5	
5	Steam Generator A Main Feedwater Flow	0-7000 gpm	± 50 gpm	Item II.B.1	
6	Steam Generator B Main Feedwater Flow	0-7000 gpm	± 50 gpm	Item II.B.2	
7	Steam Generator A Start-Up Feedwater Flow	0-500 gpm	± 10 gpm		(Required only if steaming.)
8	Steam Generator B Start-Up Feedwater Flow	0-500 gpm	± 10 gpm		(Required only if steaming.)
9	Steam Generator A Outlet Temperature	0-250 F	± 2 F		(Required only for solid secondary.)
10	Steam Generator B Outlet Temperature	0-250 F	± 2 F		(Required only for solid secondary.)
11	Steam Generator A Feed Temperature	0-150 F	± 2 F	Item II.B.10	Comp. I.D. 0491
12	Steam Generator B Feed Temperature	0-150 F	± 2 F	Item II.B.9	Comp. I.D. 0492
13	Steam Generator A Downcomer Temperature	0-150 F	± 2 F		Comp. I.D. 0469
14	Steam Generator B Downcomer Temperature	0-150 F	± 2 F		Comp. I.D. 0470

III. Additional Required Measurements

ITEM	MEASUREMENT	RANGE OF INTEREST	DESIRED ACCURACY	BACKUP MEASUREMENTS	COMMENTS
1	Pressurizer Heater Status (Banks 1,2,3,4,5)	0-540KW Per Bank	-	-	Not required for solid primary
2	Electromatic Relief Block Valve Position Status	Open; Closed	-	-	
3	Pressurizer Vent (EMO to Quench Tank) Valve Position Status	Open; Closed	-	-	
4	Reactor Vessel Boron Concentration	0-5000ppm	± 100ppm	Sample	
5	Makeup Flow	0-200gpm	± 5gpm	None	
6	Makeup Tank Level	0-100"	± 10"	None	
7	Makeup Boron Concentration	0-12000ppm	±200ppm	Sample	
8	Letdown Flow	0-100gpm	± 10gpm	Item III.9	Comp. I.D. 0346
9	Letdown Temperature	0-200F	± 20F	None	
10	Borated Water Storage Tank Level	0-100'	-	Redundant Sensor	
11	Borated Water Storage Tank Temperature	0-200F	± 20F	-	
12	Borated Water Storage Tank Concentration	0-5000ppm	± 100ppm	Sample	
13	Steam Generator A Activity Level			Sample	
14	Steam Generator B Activity Level			Sample	
15	Heat Sink Temperature	0-200F	± 10F		
16	Heat Sink Pressure	0-5psi 0-300psi	± 0.5psi ± 20psi		
17	Heat Sink Level (If Applicable)				

APPENDIX B

LIST OF REFERENCES

This Appendix will be included in the final report and include a complete list of all calculations and related test data and backup material for the information contained in this report.

Table 1

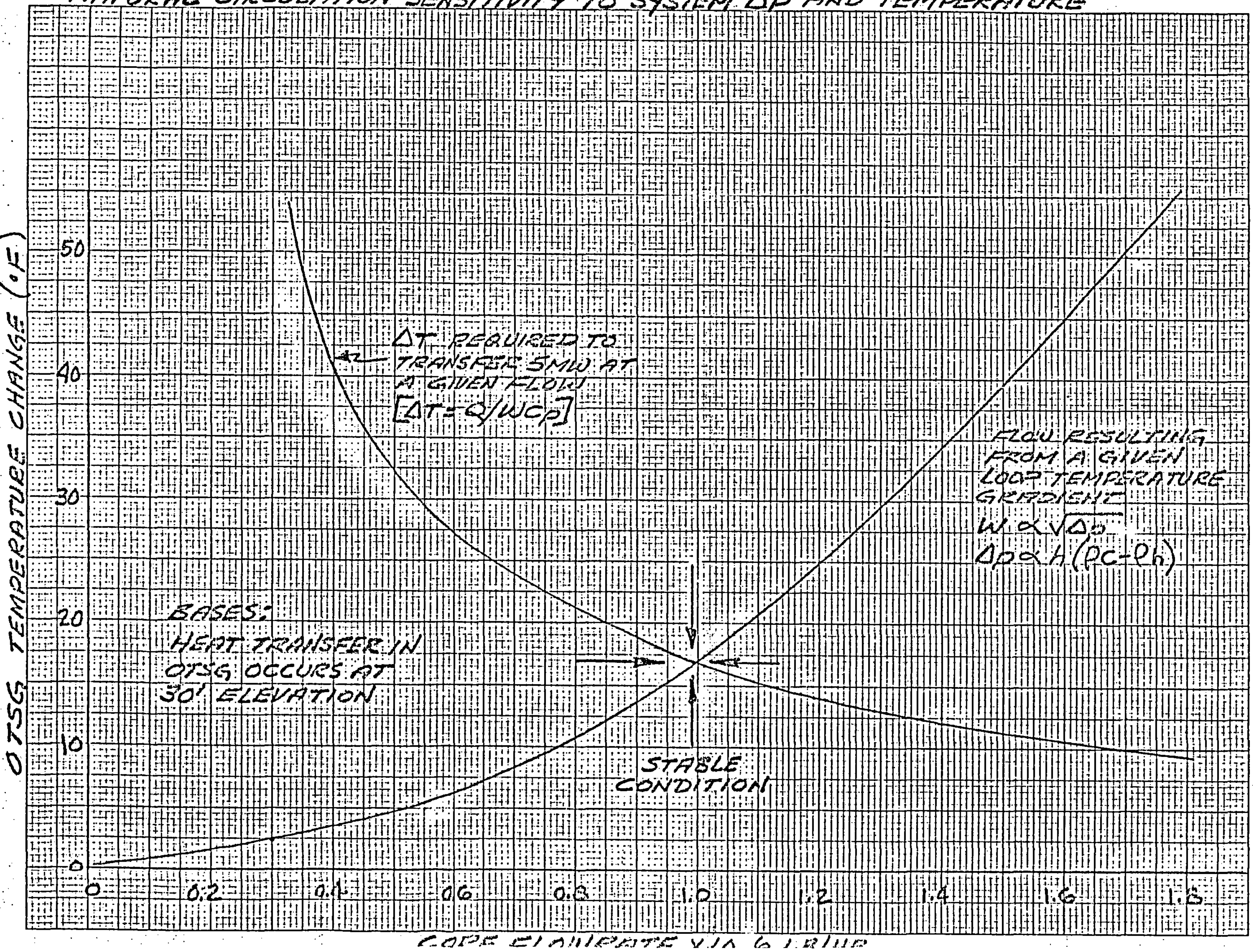
SUMMARY OF NATURAL CIRCULATION ALTERNATIVES

Reactor Configuration	Core Flow	Core ΔT	T_{cold}	T_{hot}
Two OTSG's Steaming at 230°F With a 30' Secondary Level	0.8x10^6 - 1.2x10^6 lb/hr	15-25°F	230°F	245-255°F
60 Times Normal Core Resistance	0.8 - 1.2x10^6 lb/hr	15-25°F	230°F	245-255°F
10 Times Normal Core Resistance	1.1 - 1.6x10^6 lb/hr	11-19°F	230°F	241-249°F
Normal Core Resistance	1.5 - 2.3x10^6 lb/hr	8-13°F	230°F	238-243°F
One OTSG Steaming at 230°F With a 30' Secondary Level (60 Times Normal Core Resistance)	0.7 - 1.1x10^6 lb/hr	20-30°F	230°F	250-260°F
Two OTSG's Solid With 100°F Feedwater at 3000 gpm (60 Times Normal Core Resistance)	0.8 - 1.2x10^6 lb/hr	15-25°F	105°F	120-125°F
One OTSG Solid With 100°F Feedwater at 3000 gpm (60 Times Normal Core Resistance)	0.7 - 1.1x10^6 lb/hr	20-30°F	105°F	125-135°F
3 MW Decay Heat	0.7 - 1.1x10^6 lb/hr	10-20°F	103°F	113-123°F
2 MW Decay Heat	0.6 - 1.0x10^6 lb/hr	8-15°F	101°F	109-116°F

5 MW

With reduced MW slight reduction in ΔT

FIGURE 1
NATURAL CIRCULATION SENSITIVITY TO SYSTEM ΔP AND TEMPERATURE



6 1470

FIGURE 2

NATURAL CIRCULATION SENSITIVITY TO INCREASED CORE RESISTANCE

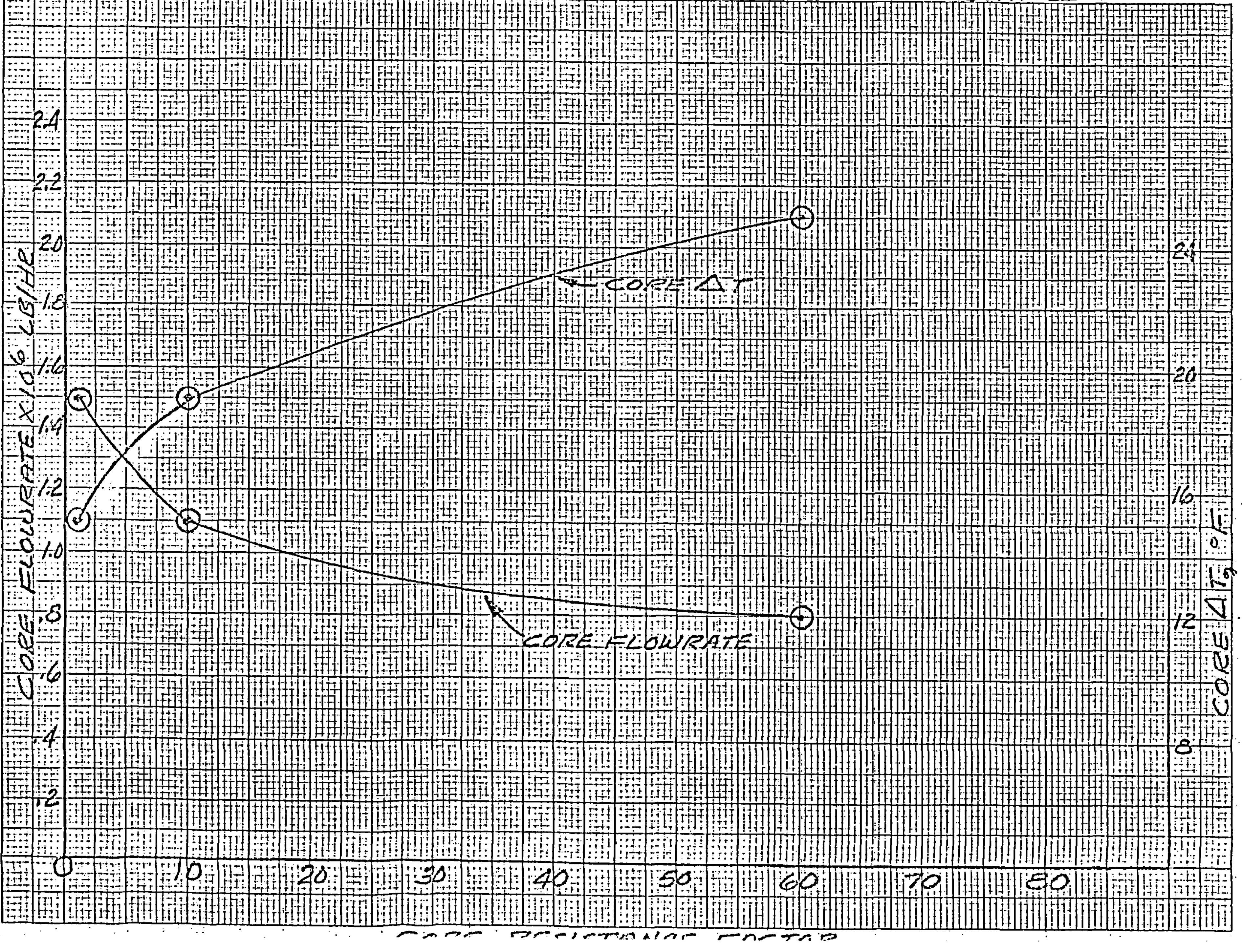
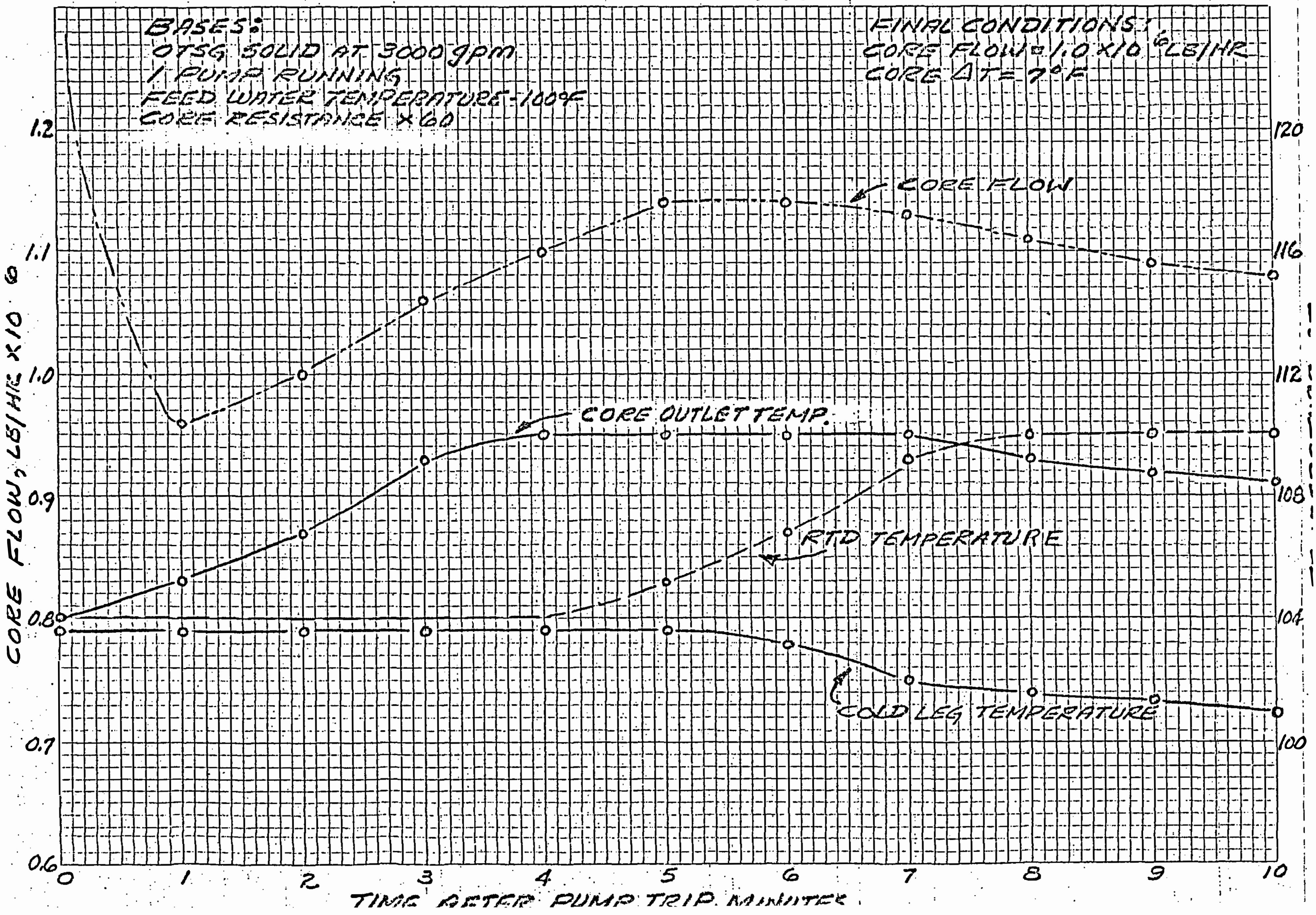


FIGURE 3

TMI-2 NATURAL CIRCULATION ANALYSIS AT 2 MW



BASES:
OTSG SOLID AT 3000 gpm
1 PUMP RUNNING
FEED WATER TEMPERATURE - 100°F
CORE RESISTANCE × 60

FINAL CONDITIONS:
CORE FLOW = 1.0 × 10^6 LB/HR
CORE ΔT = 7°F

CAP0S

FIGURE 4

TMI-2 NATURAL CIRCULATION ANALYSIS AT 3 MW

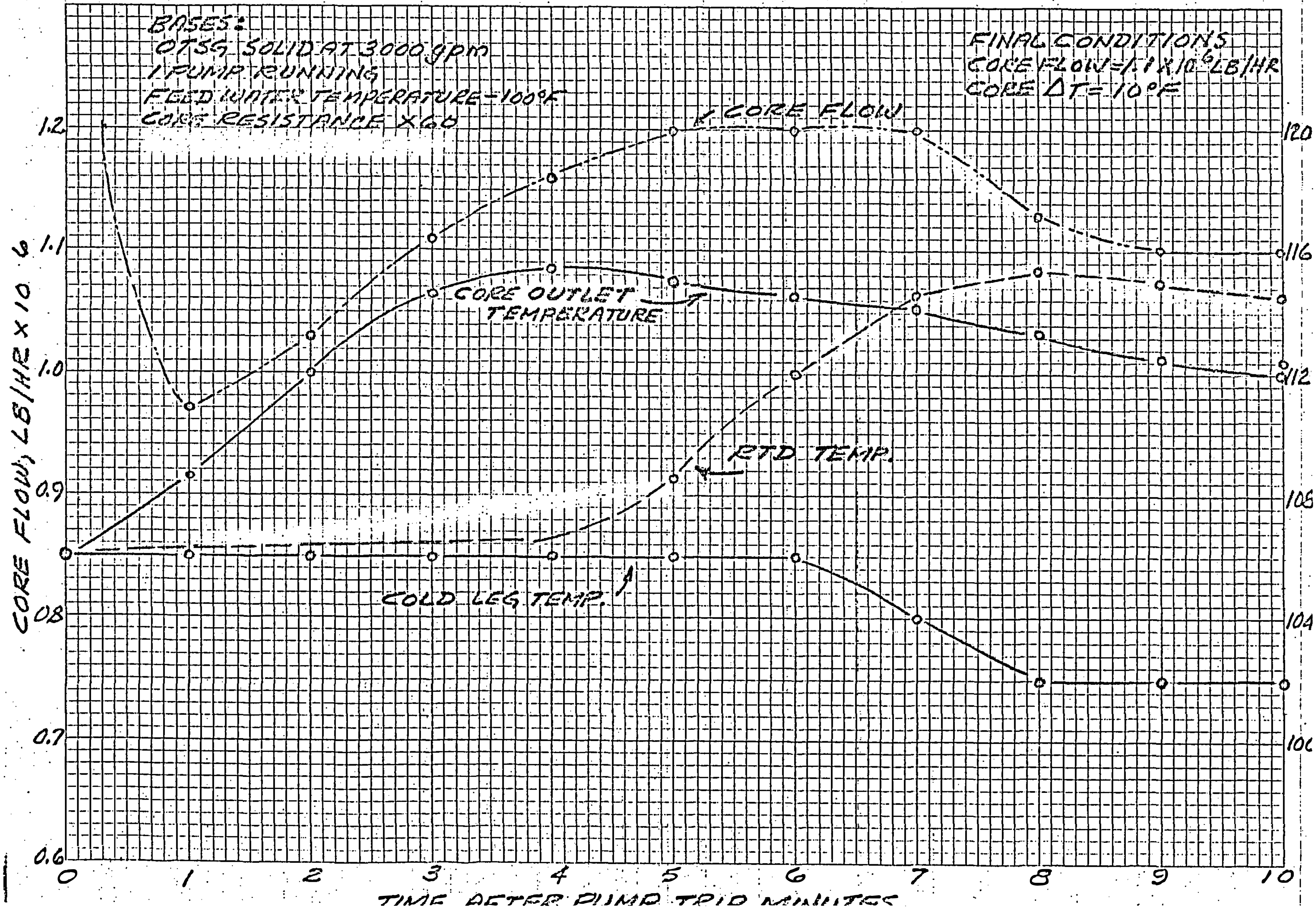


FIGURE 5

SOLID OTSG HEAT TRANSFER ANALYSIS

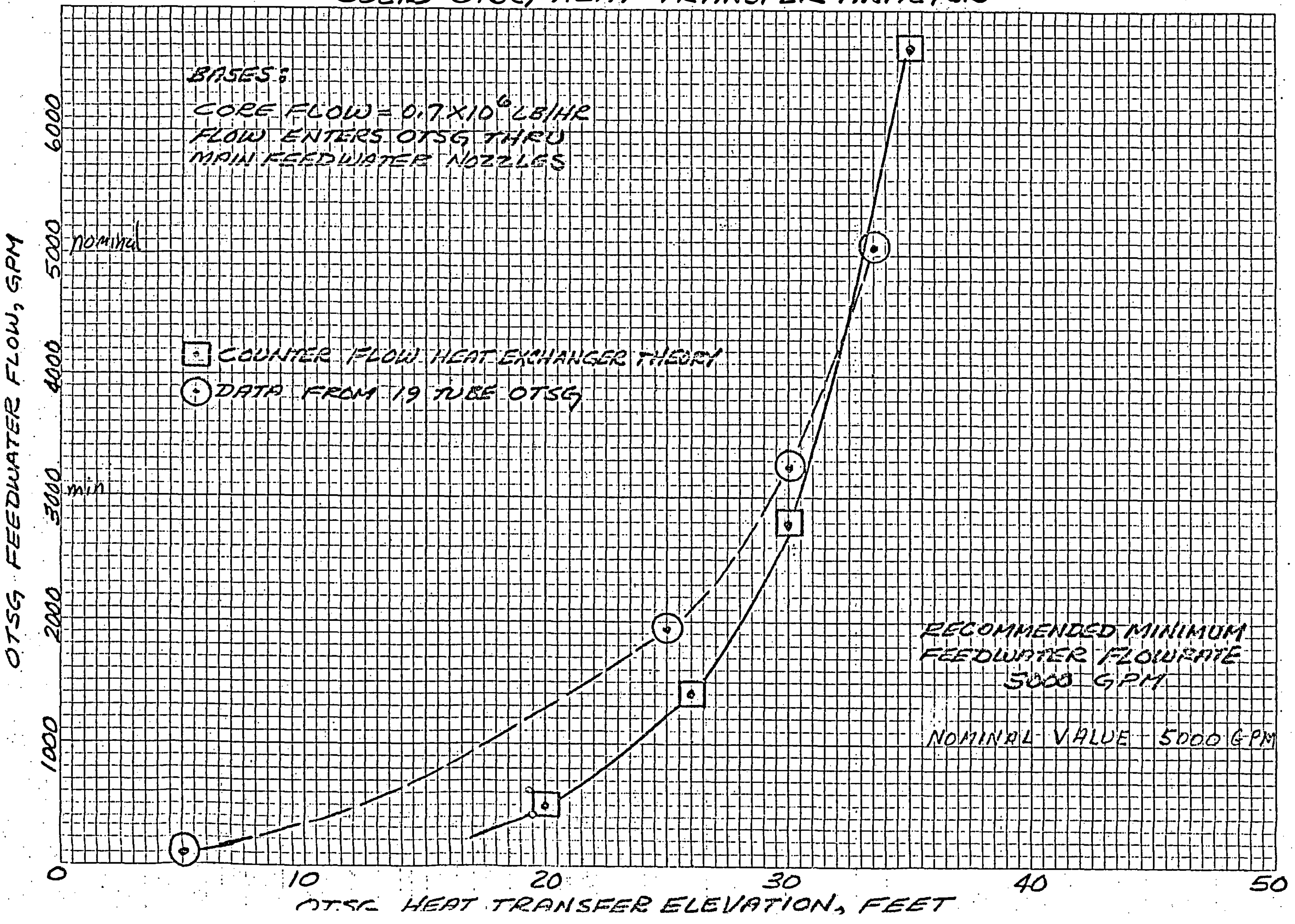


FIGURE 6

MAX. HOT LEG TEMPERATURE ALLOWED VERSUS RC PRESSURE FOR NATURAL CIRCULATION

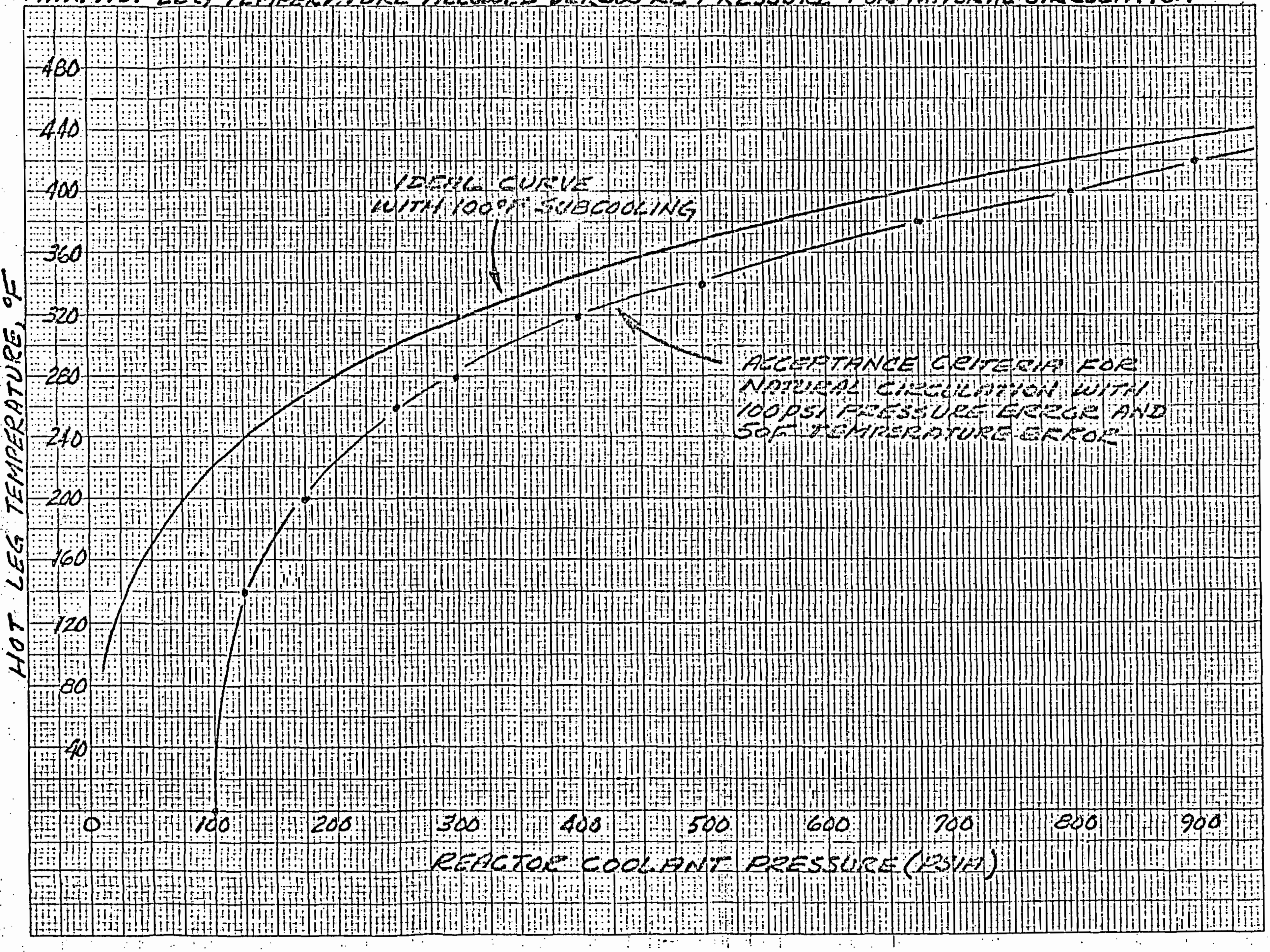
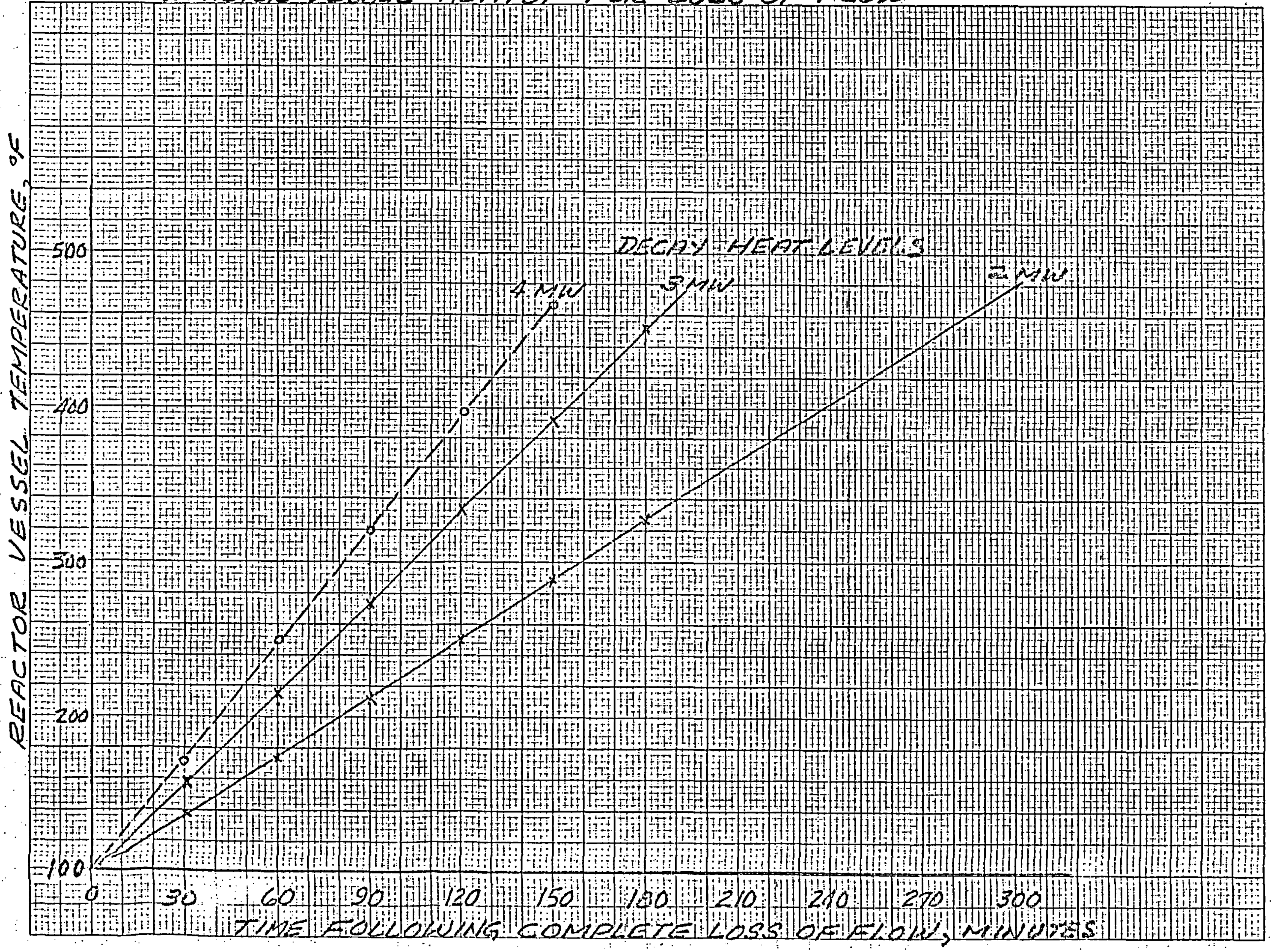


FIGURE 7
REACTOR VESSEL HEATUP FOR LOSS OF FLOW



ATTACHMENT 3

3.3. SYSTEM PERFORMANCE ON NATURAL CIRCULATION

3.3.1. DESCRIPTION OF CADDS NATURAL CIRCULATION MODEL

THE CADDS DIGITAL COMPUTER CODE[1] IS DESIGNED TO ANALYZE REACTOR TRANSIENTS, WITH OR WITHOUT SCRAM, IN A HETEROGENEOUS PRESSURIZED WATER REACTOR. IT SOLVES THE TIME DEPENDENT NEUTRON KINETICS EQUATIONS IN CONJUNCTION WITH A THERMAL HYDRAULIC SOLUTION FOR AN AVERAGE FUEL PIN DURING A REACTIVITY TRANSIENT. THE SIMULATION INCLUDES THE MAJOR FEEDBACK MECHANISMS AS WELL AS DETAILED SINGLE-PHASE NUCLEATE BOILING, TRANSITION, AND STABLE FILM BOILING CORE HEAT TRANSFER MODELS. THE ENTIRE REACTOR COOLANT LOOP, INCLUDING THE PRESSURIZER, IS SIMULATED. THE STEAM GENERATOR MODEL IS INCLUDED TO EVALUATE THE EFFECT OF FEEDWATER VARIATIONS ON THE STEAM GENERATOR AS WELL AS THE PRIMARY SYSTEM RESPONSE.

THE CADDS COMPUTER CODE DESCRIBED ABOVE NORMALLY REQUIRES THAT THE REACTOR COOLANT LOOP FLOW HISTORY BE SPECIFIED AS PART OF THE CODE INPUT. HOWEVER, A RECENTLY COMPLETED MODIFICATION TO CADD ALLOWS FLOW COASTDOWN AND NATURAL CIRCULATION TO BE CALCULATED WITHIN THE PROGRAM. THE DEVELOPMENT OF THIS MODIFICATION FOLLOWS.

A SCHEMATIC DIAGRAM OF THE PRIMARY LOOP IS SHOWN IN FIGURE 1. CONSIDERING THE FLOW AS ONE-DIMENSIONAL, THE MOMENTUM EQUATION IN THE AXIAL DIRECTION x IS

$$\frac{\partial}{\partial t}(\rho u) = -\frac{\partial}{\partial x}(\rho u^2) - \frac{\partial P}{\partial x} - \frac{\partial F}{\partial x} + F_g \qquad (1)$$

WHERE F REPRESENTS FRICTION/FORM LOSS FORCES AND F_g IS THE GRAVITY FORCE. INTEGRATING EQUATION (1) ALONG THE PRIMARY SYSTEM IN THE DIRECTION OF FLOW FROM THE PUMP DISCHARGE TO THE PUMP SUCTION YIELDS

$$\int_1^6 -\frac{\partial P}{\partial x}\,dx = \int_1^6 \frac{\partial}{\partial x}\left(\frac{\rho u^2}{g_o}\right)dx + \int_1^6 \frac{\partial}{\partial t}\left(\frac{\rho u}{g_o}\right)dx + \int_1^6 \frac{\partial F}{\partial x}\,dx + \int_1^6 F_g\,dx \,. \tag{2}$$

SUBSTITUTING FOR F AND F_g AND EQUATING THE PRESSURE INTEGRAL TO ΔP_{pump} OBTAINS

$$\Delta P_{pump} = \int_1^6 \frac{\partial}{\partial x}\left(\frac{\rho u^2}{g_o}\right)dx + \int_1^6 \frac{\partial}{\partial t}\left(\frac{\rho u}{g_o}\right)dx + \int_1^6 \rho\,\frac{g}{g_o}\frac{\partial z}{\partial x}\,dx$$

$$+ \sum_i^{cv} \left(k_i + f_i\,\frac{L_i}{D_i}\right)\frac{1}{2g_o}\,\rho_i u_i |u_i| \tag{3}$$

WHERE cv IS THE TOTAL NUMBER OF CONTROL VOLUMES. CONSIDERING THE FLOW RATE TO BE SPATIALLY INDEPENDENT, EXPRESSING VELOCITY (u) IN TERMS OF SYSTEM FLOW RATE (W), AREAS (A_i), AND DENSITIES (ρ_i), AND SUMMING OVER ALL CONTROL VOLUMES YIELDS

$$\Delta P_{pump} = \frac{W^2}{g_o}\left[\frac{1}{\rho A^2}\right]_1^6 + \frac{dw}{dt}\sum_i^{cv}\frac{L_i}{A_i g_o} + \sum_i^{cv} \rho_i\,\frac{g}{g_o}\,\Delta Z_i$$

$$+ \frac{W|W|}{2g_o}\sum_i^{cv}\frac{1}{\rho_i A_i^2}\left(K_i + F_i\,\frac{L_i}{D_i}\right) \tag{4}$$

USING A SEMI-IMPLICIT SOLUTION TECHNIQUE, EQUATION 4 CAN BE WRITTEN AS

$$\frac{W^{n+1} - W^{n}}{\delta t} \sum_{i}^{cv} \frac{L_i}{A_i g_o} = \Delta P^{n}_{pump} - \frac{W^{n}|W^{n}|}{g_o} \left[\frac{1}{\rho A^2}\right]_1^6$$

$$- \sum_{i}^{cv} \rho_i \frac{g}{g_o} \Delta Z_i - \frac{|W^{n}|W^{n+1}}{2g_o} \sum_{i}^{cv} \frac{1}{\rho_i A_i^2} (K_i + f_i \frac{L_i}{D_i}) \qquad (5)$$

REARRANGING AND NEGLECTING THE ACCELERATION TERM PRODUCES

$$W^{n+1} = \frac{\frac{W^{n}}{\delta t} \sum_{i}^{cv} \frac{L_i}{A_i g_o} + \Delta P_{pump} - \sum_{i}^{cv} \rho_i \frac{g}{g_o} \Delta Z_i}{\frac{1}{\delta t} \sum_{i}^{cv} \frac{L_i}{A_i g_o} + \frac{|W^{n}|}{2g_o} \sum_{i}^{cv} \frac{1}{\rho_i A_i^2} (K_i + f_i \frac{L_i}{D_i})} \qquad (6)$$

THEREFORE, THE SYSTEM FLOW RATE CAN BE DETERMINED BY EQUATION 6 AT ANY TIME STEP DURING THE TRANSIENT. THE MOMENTUM EQUATION IS COUPLED TO THE CADDS ENERGY EQUATION THROUGH THE FLUID PROPERTIES WHICH ARE FUNCTIONS OF THE SYSTEM PRESSURE AND THE ENTHALPY. THE CALCULATION OF FLOW RATE IS CONTAINED IN THE CADDS SUBROUTINE, NATURAL.

3.3.1.1. COMPARISON OF NATURAL CIRCULATION MODEL PREDICTIONS TO ANALYTICAL RESULTS

TWO SPECIAL CASES WERE USED TO VERIFY THE CALCULATIONS OF THE NATURAL SUBROUTINE. EACH CASE MADE USE OF ENOUGH SIMPLIFYING ASSUMPTIONS TO ALLOW THE HAND CALCULATION OF FLOW (OR FLOW VERSUS TIME) WHICH COULD THEN BE COMPARED TO NATURAL PREDICTIONS.

THE FIRST CASE SIMULATED THE FLOW COASTDOWN OF AN NSS PRIMARY LOOP. IT WAS ASSUMED THAT THE PUMP SEIZED AT THE BEGINNING OF THE TRANSIENT ($\Delta P_{pump} = 0$) AND THAT THE GRAVITY TERM COULD BE NEGLECTED SINCE INERTIA TERMS DOMINATE IN THE INITIAL STAGES OF A COASTDOWN TRANSIENT. THE NATURAL SUBROUTINE RESULTS ARE COMPARED TO ANALYTICAL VALUES IN TABLE 1.

THE SECOND CASE USED THE SAME PRIMARY LOOP AS THE FIRST CASE AND ASSUMED A LINEAR DISTRIBUTION OF DENSITY IN THE CORE AND STEAM GENERATOR. THE STEADY STATE NATURAL CIRCULATION FLOW WAS CALCULATED AND COMPARED TO TRANSIENT NATURAL PREDICTIONS. THIS COMPARISON IS SHOWN IN FIGURE 2.

BOTH OF THE ABOVE COMPARISONS SHOW EXCELLENT AGREEMENT BETWEEN THE ANALYTICAL VALUES AND NATURAL SUBROUTINE PREDICTIONS, THUS VERIFYING THE PROGRAMMING AND THE NUMERICAL SCHEME OF THE NATURAL SUBROUTINE.

3.3.1.2. COMPARISON OF CADDS WITH NATURAL TO DAVIS-BESSE DATA

TABLE 1. COMPARISON OF NATURAL PREDICTIONS TO ANALYTICAL VALUES FOR A FLOW COASTDOWN

TIME (sec)	$(\frac{W}{W_o})$ analytical	$(\frac{W}{W_o})$ NATURAL
0	1.0	1.0
0.5	0.7881	0.7882
1.0	0.6452	0.6454
2.0	0.4667	0.4669
3.0	0.3610	0.3611
4.0	0.2917	0.2919
5.0	0.2432	0.2433
10.0	0.1274	0.1275
15.0	0.08336	0.08341
20.0	0.06077	0.06081
25.0	0.04723	0.04726
30.0	0.03830	0.03832
40.0	0.02736	0.02738
50.0	0.02099	0.02101
60.0	0.01688	0.01689
70.0	0.01401	0.01402
80.0	0.01192	0.01193
90.0	0.01033	0.01034
100.0	0.009084	0.009090

FIGURE 1. PRIMARY SYSTEM SCHEMATIC

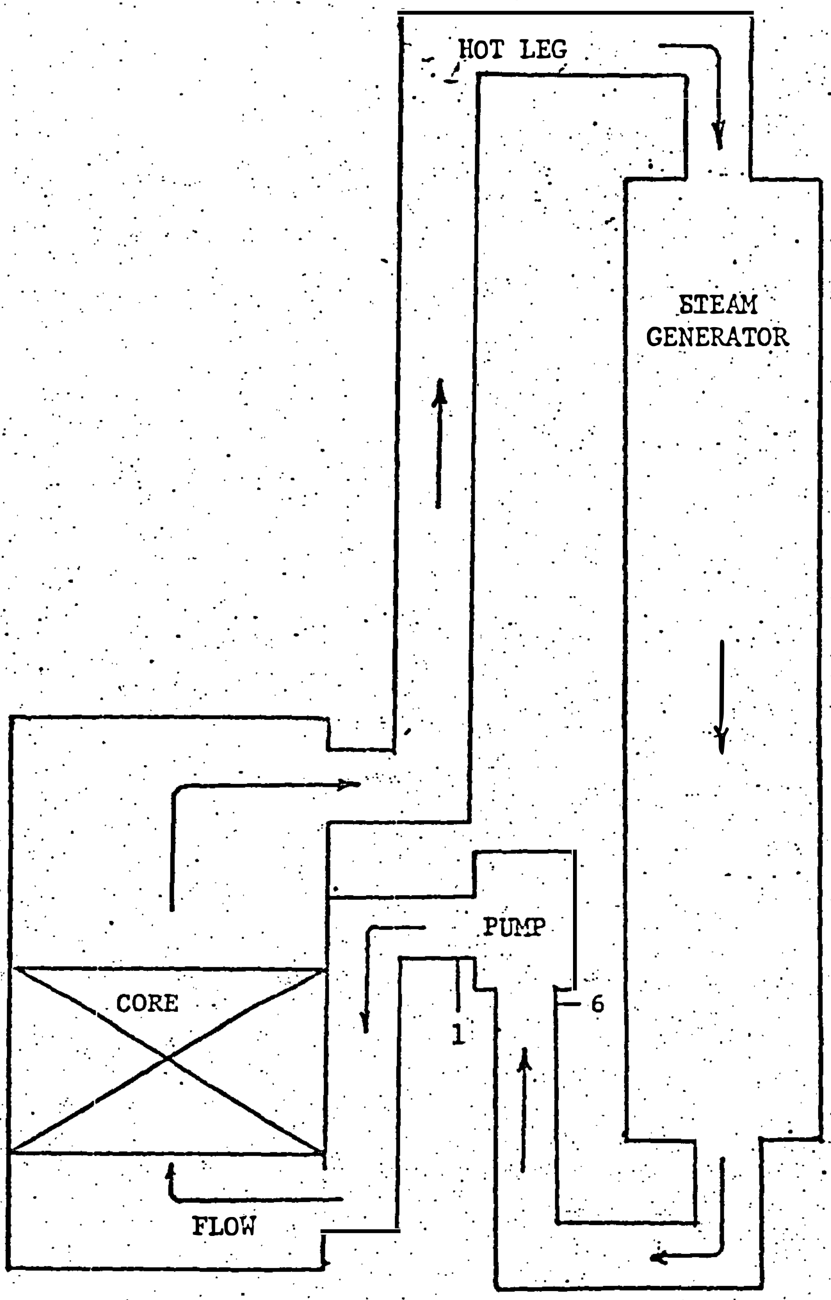
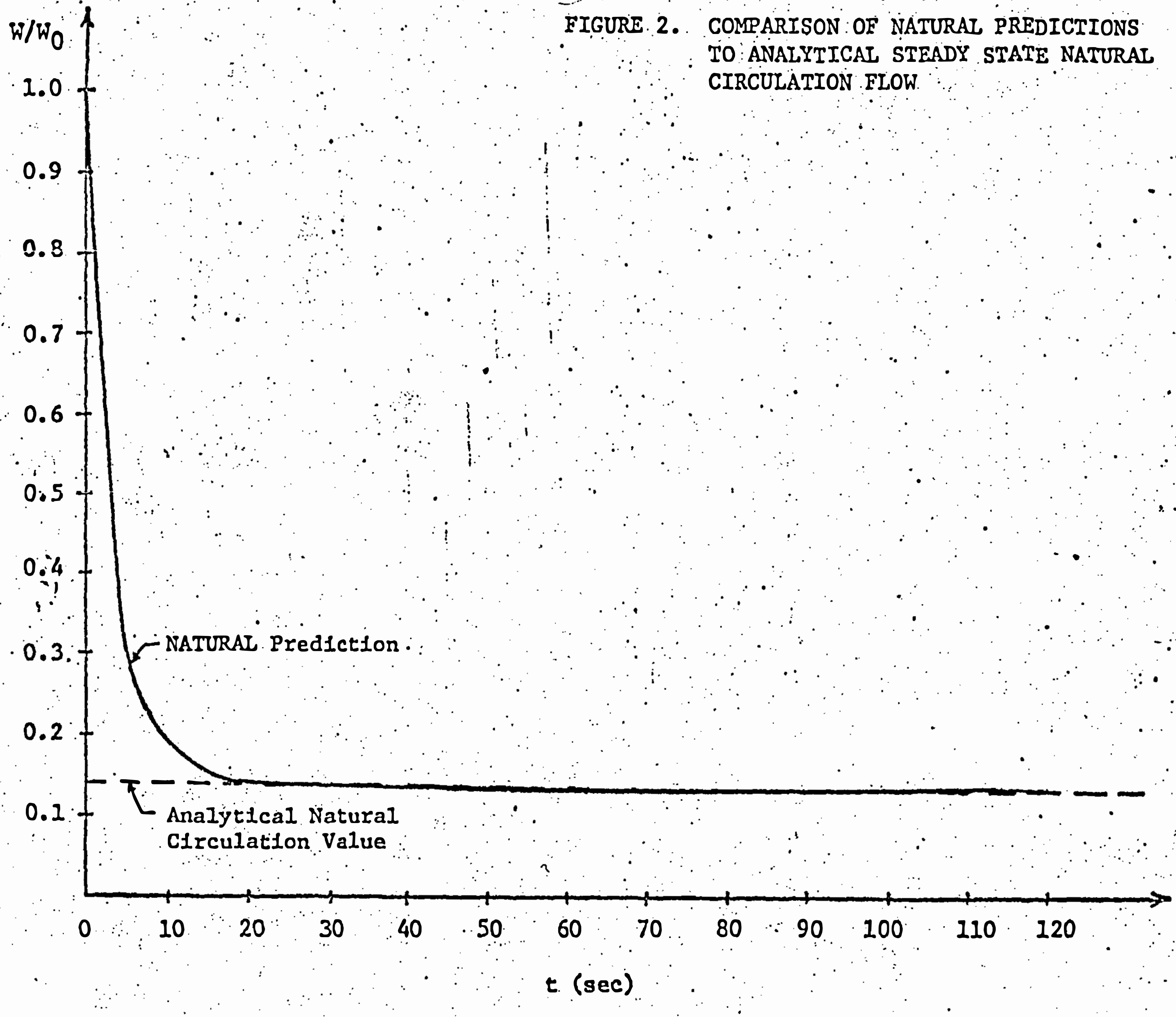


FIGURE 2. COMPARISON OF NATURAL PREDICTIONS TO ANALYTICAL STEADY STATE NATURAL CIRCULATION FLOW



3.2.2. MODELLING ASSUMPTIONS

THE NATURAL CIRCULATION RESULTS WHICH ARE SUBSEQUENTLY REPORTED WERE OBTAINED USING A ONE (1) LOOP CADDS/NATURAL MODEL. THE ASSUMPTION OF ONE LOOP MEANS THAT ALL DECAY HEAT REMOVAL OCCURS THROUGH ONE ACTIVE STEAM GENERATOR AND THAT THE STEADY STATE NATURAL CIRCULATION IN THE IDLE LOOP IS ZERO. THE CADDS MODEL ALSO ASSUMES THAT ALL FLOW FROM THE OPERATING PUMP GOES THROUGH THE CORE PRIOR TO PUMP TRIP. WHILE THIS ASSUMPTION IS NOT ENTIRELY ACCURATE UNDER FORCED FLOW CONDITIONS (BECAUSE OF REVERSE FLOW IN THE IDLE LOOP), IT DOES NOT PREVENT THE OBTAINING OF A VALID STEADY STATE NATURAL CIRCULATION SOLUTION UNDER CONDITIONS CONSIDERED HEREIN.

ANOTHER ANALYSIS ASSUMPTION IS THAT THE PUMP ROTOR LOCKS AT TIME OF TRIP AND A LOCKED ROTOR PUMP RESISTANCE IS USED FROM THEN ON. THIS RESULTS IN THE MAXIMUM FLOW COASTDOWN AND MAXIMUM TRANSIENT CORE TEMPERATURE WHILE NATURAL CIRCULATION IS BEING ESTABLISHED.

THE PRINCIPAL VALUE USED FOR CORE FORM LOSS HAS BEEN CALCULATED FROM CORE FLOW AND PRESSURE DROP INFERRED FROM TMI-2 HOT LEG FLOW MEASUREMENTS. GIVEN THAT THE CORE PRESSURE DROP IS APPROXIMATELY 18 PSI AT A CORE FLOW OF ABOUT 4500 lbm/sec AND THAT THE PRESSURE DROP IS ENTIRELY FORM LOSS, A FORM LOSS FACTOR (K) OF 1100 CAN BE CALCULATED, WHICH IS ABOUT 200 TIMES THE NORMAL CORE FORM LOSS. OTHER CORE FORM LOSS FACTORS USED IN THE NATURAL CIRCULATION ANALYSIS ARE 5.5 (NORMAL), 330 (60 TIMES NORMAL), AND 5500 (1000 TIMES NORMAL).

ADDITIONAL MODELLING ASSUMPTIONS CONCERN THE CORE BYPASS FLOW AND VENT VALVE FLOW. MOST OF THE ANALYSIS ASSUMES A CORE BYPASS FLOW OF 4.6%, BUT ONE CASE CONSIDERS A 30% BYPASS FLOW. IN ALL CASES VENT VALVE FLOWS ARE NEGLECTED.

3.3.3 RESULTS

IN ORDER TO INVESTIGATE THE FEASIBILITY OF COOLING THE CORE WITH NATURAL CIRCULATION USING WATER TO WATER HEAT TRANSFER IN THE STEAM GENERATOR, A STUDY WAS PERFORMED USING THE FOLLOWING MATRIX:

CORE POWER	5 MW
FEEDWATER TEMPERATURE	100°F
ONLY A STEAM GENERATOR OPERATING	
CORE PRESSURE DROP 60 TIMES NOMINAL	
FEEDWATER FLOW	1100, 3000, 5000 GPM
CORE BYPASS	5%

THE STEADY STATE RESULTS OF THIS STUDY ARE SHOWN IN TABLE 2.

TABLE 2

FEEDWATER FLOW GPM	CORE INLET TEMPERATURE °F	CORE OUTLET TEMPERATURE °F	PRIMARY SYSTEM FLOW LBM/SEC.
1100	123	151	210
3000	103	118	325
5000	101	116	338

FROM THE ABOVE TABLE, WE SEE THAT IF THE FEEDWATER FLOW EXCEEDS THE PRIMARY SYSTEM FLOW, THE COOLING WILL OCCUR HIGH ENOUGH IN THE OTSG TO DEVELOP REASONABLE NATURAL CIRCULATION FLOW RATES. BASED ON THESE RESULTS, A FEEDWATER FLOW RATE OF 3000 GPM AT 100°F WAS CHOSEN AS THE MODE FOR REMOVING THE DECAY HEAT AND FURTHER PARAMETER STUDIES WERE PERFORMED. TABLE 3 SHOWS THE STEADY STATE RESULTS FOR THE CASES RUN.

TABLE 3

CASE NO.	CORE POWER LEVEL MW	CORE PRESSURE DROP LOSS CHARACTERISTRICS X NOMINAL	% BYPASS FLOW	CORE INLET TEMP. °F	CORE OUTLET TEMP. °F	SYSTEM FLOW LBM/SEC.
1	3	1	5	101.9	109.7	380
2	3	60	5	101.3	111.1	305
3	3	200	5	100.8	113.7	231
4	3	1000	5	100.4	121.3	142
5	3	200	30	100.9	118.2	235
6	2	60	5	100.9	107.7	292

CASE NUMBER FIVE IS THE BEST ESTIMATE OF THE EXPECTED CORE BEHAVIOR IF NATURAL CIRCULATION IS INITIATED WHEN THE DECAY HEAT LEVEL IS 3 MW. A CORE PRESSURE DROP LOSS CHARACTERISTIC OF 200 TIMES NOMINAL IS SLIGHTLY LARGER THAN THAT REQUIRED TO PRODUCE AGREEMENT WITH CURRENT MEASUREMENTS OF REACTOR COOLANT FLOW SPLITS. (SEE P. S. BARTELLS, "CORE PRESSURE DROP FOR NATURAL CIRCULATION CALCULATION", APRIL 9, 1979). THE BEST ESTIMATE FOR CORE BYPASS FLOW IS BETWEEN 22 AND 27 PERCENT. (R. M. GRIBBLE, "CORE BYPASS FLOW FOR CORE BLOCKED AT TOP ONLY", APRIL 8, 1979.)

FIGURE 3 SHOWS A PLOT OF CORE FLOW VERSUS TIME AND FIGURE 4 SHOWS A PLOT OF TEMPERATURE AT THE HOT LEG RESISTANCE THERMOMETER VERSUS TIME FOR CASE FIVE. NOTE THE TIME DELAY BETWEEN THE START OF NATURAL CIRCULATION AND THE INCREASED HOT LEG TEMPERATURE. THIS IS DUE TO THE LOW VELOCITIES DURING NATURAL CIRCULATION.

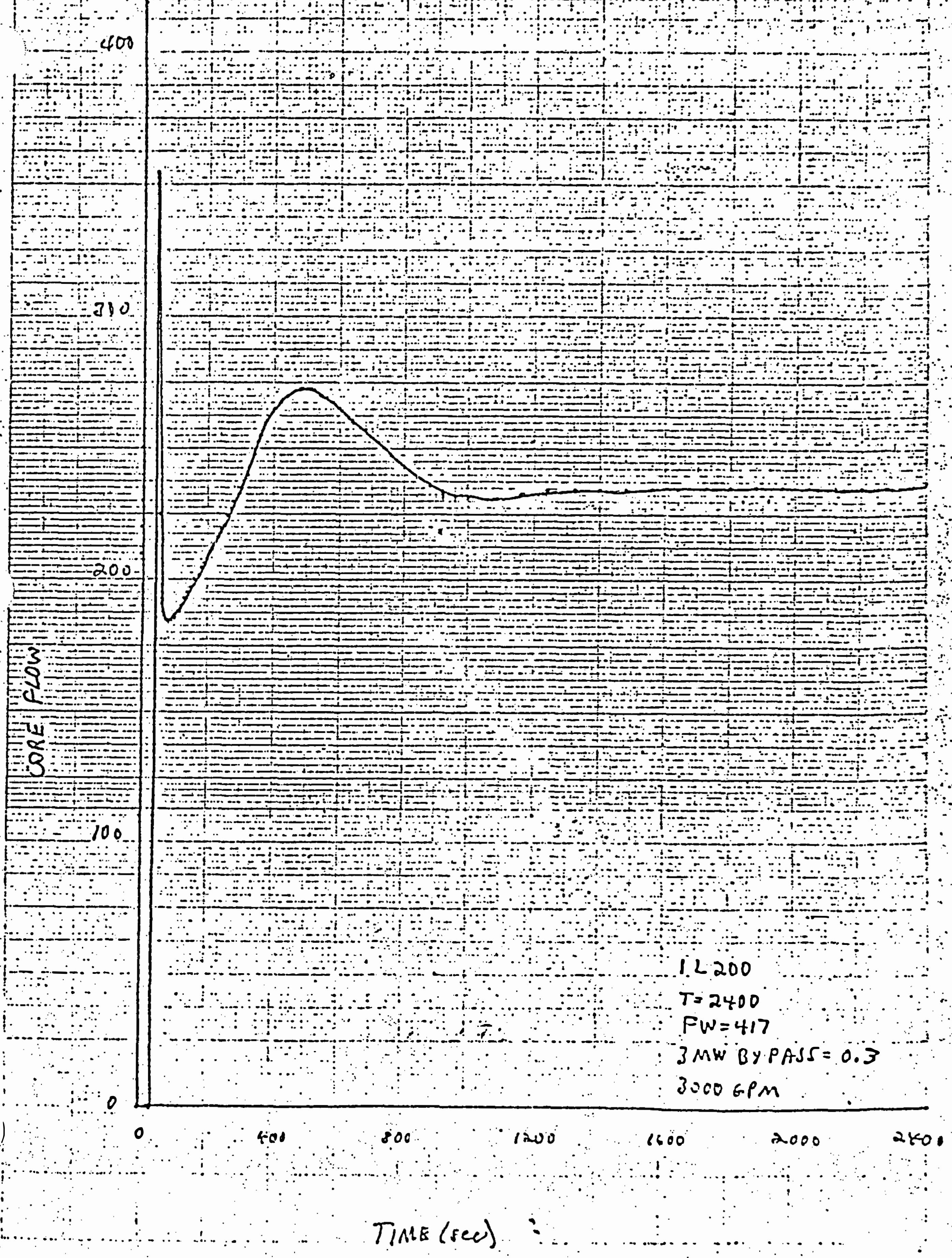
THE STEADY STATE NATURAL CIRCULATION FLOW OF 235 LBM/SEC FOR CASE FIVE PROVIDES AMPLE COOLING FOR THE CORE. IN FACT, IF THE HOT BUNDLE ONLY RECEIVED 10% OF THE NOMINAL FLOW DUE TO BLOCKAGES, A CORE FLOW of 235 LBM/SEC WOULD BE ENOUGH TO PREVENT BOILING IN THE BLOCKED BUNDLE. (D. A. FARNSWORTH, "REQUIRED FLOW VERSUS COOLING EFFICIENCY", APRIL 10, 1979).

COMPARISON OF CASES THREE AND FIVE SHOWS THE EFFECT OF CORE BYPASS FLOW ON SYSTEM FLOW AND CORE COOLANT TEMPERATURE INCREASE. AS EXPECTED, THERE IS VERY LITTLE INFLUENCE ON SYSTEM FLOW. AS THE CORE BYPASS FLOW INCREASES THE FLUID GOING THROUGH THE CORE IS HEATED MORE AND WHEN THE TWO STREAMS MEET AND ARE MIXED THE TEMPERATURE IN THE UPPER PLENUM IS NEARLY THE SAME; THUS THE DENSITY DISTRIBUTIONS ARE NEARLY THE SAME FOR BOTH RUNS AND THUS THE FLOWS ARE NEARLY EQUAL. SINCE THE CORE COOLANT TEMPERATURE RISE IS QUITE SMALL, SUBSTANTIAL BYPASS CAN OCCUR AND STILL HAVE ACCEPTABLE CORE COOLING.

COMPARISON OF CASES TWO, THREE AND FOUR SHOWS THE EFFECT OF INCREASING THE EFFECTIVE CORE RESISTANCE TO FLOW. AS EXPECTED, THE FLOW DECREASES AND THE TEMPERATURE RISE ACROSS THE CORE INCREASES; HOWEVER, EVEN FOR PRESSURE DROPS 1000 TIMES NOMINAL (ABOUT FIVE TIMES WHAT WE ESTIMATE) THE CORE FLOW IS ADEQUATE TO COOL THE CORE.

Fig 3

CORE FLOW vs. TIME



1L200
T=2400
FW=417
3MW BYPASS= 0.3
3000 GPM

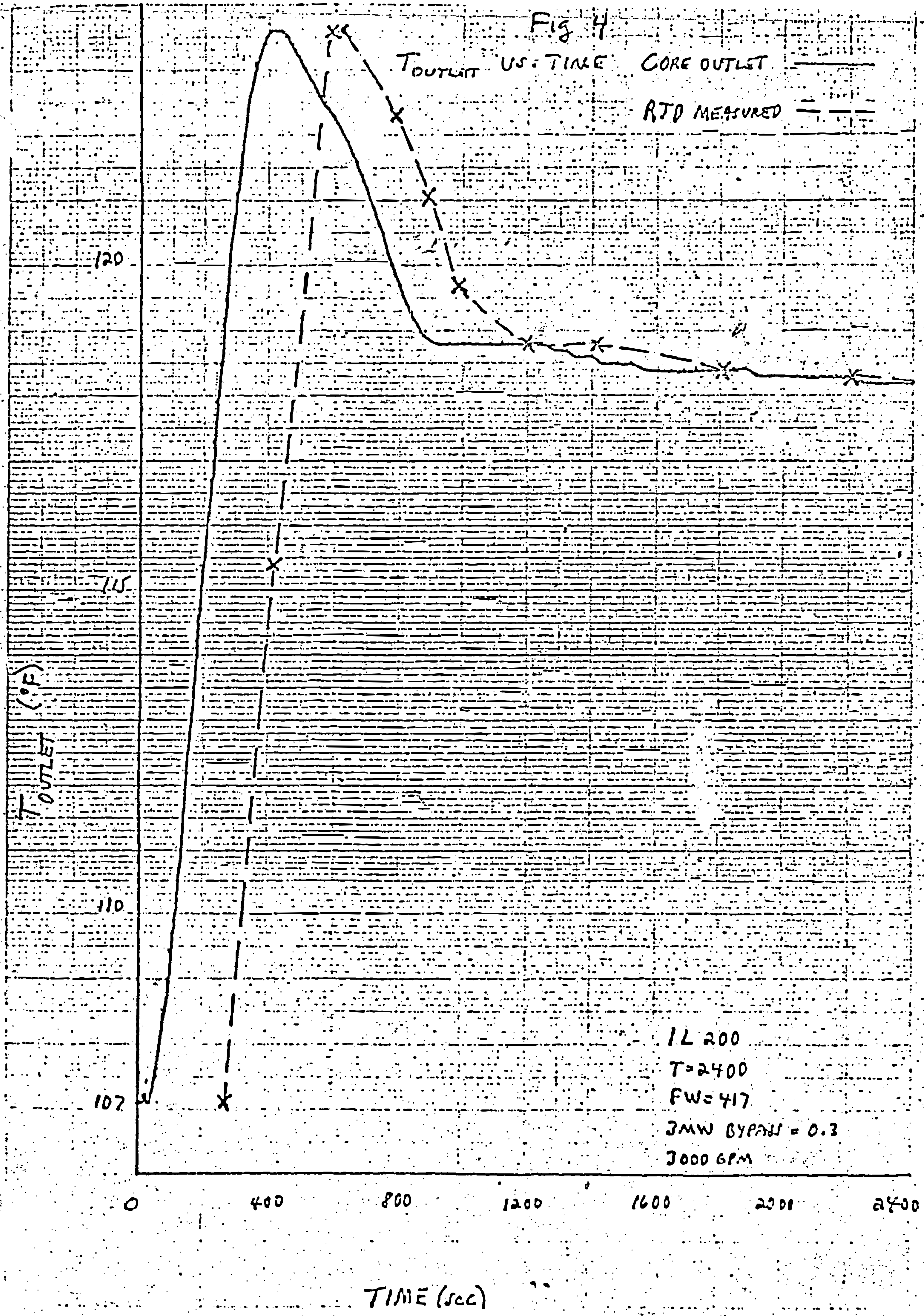


Fig 4

T_{OUTLET} vs. TIME
CORE OUTLET
RTD MEASURED
T_{OUTLET} (°F)
107
110
115
120
0
400
800
1200
1600
2001
2400
TIME (sec)
IL 200
T=2400
FW=417
3MW BYPASS = 0.3
3000 GPM

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1. Y. H. Hsii et.al., CADDS - Computer Application to Direct Digital Simulation of Transients in PWRs with or without SCRAM, Revision 1, BAW-10098P, Rev. 1, December 1977.

3.3.5 COMPARISON WITH ALTERNATE CALCULATIONS

A COMPUTER PROGRAM HAS BEEN WRITTEN TO PERFORM THE SAME TYPE OF STEADY STATE ANALYSIS AS HAS BEEN DONE BY HAND. THIS PROGRAM INCLUDES A FORCE BALANCE ON THE VENT VALVES AND WILL TREAT THE RECIRCULATION FLOW IF THE VENT VALVE OPENS. THE PROGRAM ALSO CONSIDERS THE IDLE LOOP AS A FLOW PATH. FOR A RUN WHERE THE CONDITIONS WERE SIMILAR TO CASE 1 OF TABLE 3 (POWER = 3 MW, CORE PRESSURE DROP AT ITS NOMINAL CONDITION, FEEDWATER FLOW OF 3000 GPM), THIS PROGRAM COMPUTED A CORE FLOW OF 313 LBM/SEC WHERE CADDS/NATURAL PREDICTED 380. THIS IS GOOD AGREEMENT CONSIDERING THE SMALL DIFFERENCIES IN DENSITY AND PRESSURE DROP WHICH OCCUR, AND THERE WAS A DIFFERENCE IN PRIMARY COOLANT TEMPERATURE WHICH WOULD ACCOUNT FOR PART OF THE DIFFERENCE. THE STEADY STATE CODE PREDICTED A CORE ΔT OF 9 DEGREES VERSUS CADDS-NATURAL'S PREDICTION OF 7.8. THE STEADY STATE CODE COMPUTED THAT THE PRESSURE DIFFERENCE WAS NOT SUFFICIENT TO OPEN THE VENT VALVES AND THE IDLE LOOP FLOW WAS ONLY 32 LBS; THUS, THE SIMPLIFICATIONS USED IN THE CADDS-NATURAL PROGRAM ARE REASONABLE.

3.3.1.2. COMPARISON OF CADDS WITH NATURAL CIRCULATION CALCULATION TO DAVIS-BESSE EXPERIMENT

IN ORDER TO VERIFY THE VALIDITY OF THE CADDS/NATURAL CALCULATIONAL RESULTS, A SPECIAL CASE SIMULATING THE DAVIS-BESSE PLANT WAS RUN AND THE RESULTS COMPARED WITH THE REPORTED EXPERIMENTAL MEASUREMENTS. THE FOLLOWING TEST CONDITIONS WERE USED IN THE CADDS/NATURAL INPUT.

1) 2-LOOP CONFIGURATION
2) FEEDWATER FROM THE AUXILIARY FEEDWATER SYSTEM
3) REACTOR POWER WAS CONSTANT AT 3.85% OF 2772 MWth
4) TWO DIFFERENT FEEDWATER RATES (46 AND 460 LBM/SEC) WERE USED.

THE RESULTS ARE SHOWN IN FIG. ______. AS SHOWN BY THE FIGURE, THE SYSTEM FLOW RATES DO NOT REALLY STABILIZE IN 1800 SEC AND SMALL OSCILLATIONS STILL CAN BE SEEN. HOWEVER, IT APPEARS THAT THE PRIMARY SYSTEM FLOW RATES ARE APPROACHING A STEADY STATE NATURAL CIRCULATION FLOW RATE OF 6.0% FOR THE LOW FEEDWATER RATE AND 4.4% FOR THE HIGHER FEEDWATER RATE. THE REASON FOR THIS IS THAT THE HIGHER FEEDWATER RESULTS IN LOWER PRIMARY SYSTEM TEMPERATURE, WHICH IMPLIES HIGHER DENSITY AND VISCOSITY AND CONSEQUENTLY, HIGHER FRICTION.

THE DAVIS-BESSE EXPERIMENT REPORTED THAT THE NATURAL CIRCULATION FLOW RATE RANGES FROM 4.6 TO 5.1%, DEPENDING ON THE WATER LEVEL OF THE STEAM GENERATOR. IT APPEARS THAT THE CADDS/NATURAL PROGRAM WITH THE TWO EXTREMES IN FEEDWATER FLOW PREDICTS NATURAL CIRCULATION FROM RATES THAT BRACKET THE EXPERIMENTAL RESULTS. THE DIFFICULTY IN COMPARING CADDS/NATURAL DIRECTLY TO THE DAVIS-BESSE DATA IS DUE TO THE FOLLOWING:

1) THE OPERATOR VARIED AUXILIARY FEED PUMP SPEEDS SO MUCH THAT THE FLOW RATES RANGED FROM 200 TO 1400 GPM AND THAT NO AVERAGE STEADY STATE FLOW RATE COULD BE DISCERNED FROM THE DATA.

2) DURING THE EXPERIMENT, THE MAIN FEEDWATER WAS TURNED OFF AND ONLY THE AUXILIARY FEEDWATER WAS USED. AT PRESENT, CADDS DOES NOT HAVE THE CAPABILITY OF MODELLING THE AUXILIARY FEEDWATER PROPERLY.

Attachment 4.

G. A. MEYER
APRIL 10, 1979
TIME: 1630

3.7 CORE THERMAL BEHAVIOR

A. ANALYSIS OF CURRENT OPERATING CONDITION

1) CORE BLOCKAGE STUDY

TWO METHODS HAVE BEEN USED TO ESTIMATE THE EXTENT OF THE CORE BLOCKAGE. THE FIRST METHOD INVOLVES THE USE OF THE INCORE THERMOCOUPLES TO DETERMINE THE CORE OUTLET TEMPERATURE. THIS METHOD PREDICTS A CORE FLOW OF LESS THAN 1 X 10^6 LB/HR.[1]

THE SECOND METHOD INVOLVES MATCHING THE PUMP CODE TO THE DATA FORM THE PLANT. THE PUMP CODE CALCULATES FLOW SPLITS FOR DIFFERENT PUMP CONFIGURATIONS. THE CODE WAS MODELED TO MATCH THE PRE-INCIDENT CONDITIONS. THE CORE RESISTANCE WAS INCREASED UNTIL THE CODE PREDICTED THE SAME HOT LEG FLOWS THAT ARE BEING MEASURED BY THE TWO GENTILLE DELTA P'S.

THIS CALCULATION PREDICTS A CORE FLOW OF 13 x 10^6 LBM/HR AND A CORE PRESSURE DROP OF 18 PSI. THESE CONDITIONS ARE CALCULATED WITH A CORE RESISTANCE APPROXIMATELY 200 TIMES THE NORMAL RESISTANCE[2,3,8] OR A FORMLOSS RESISTANCE OF 1650.

TO DETERMINE IF A CORE RESISTANCE OF THAT MAGNITUDE IS FEASIBLE AN ESTIMATE OF THE CORE DAMAGE WAS DEVELOPED. A CURRENT ESTIMATE OF THE MATERIAL AVAILABLE FOR CORE BLOCKAGE IS :

~~$Z_r O_2$~~	63 FT^3
UO_2	94 FT^3
OTHER	10 FT^3
	167 FT^3

THIS DEBRIS IF SPREAD EVENLY ACROSS THE CORE COULD FORM A PACKED BED OF DEBRIS THREE FEET THICK.[4] THE RESISTANCE (FORMLOSS COEFFICIENT) FOR THIS BED HAS BEEN CALCULATED AT APPROXIMATELY 1700[5] WHICH IS IN GOOD AGREEMENT WITH THE TOTAL CORE RESISTANCE CALCULATED USING THE MEASURED FLOW SPLITS. THE METHOD TWO CALCULATION OF A ONE PUMP CORE FLOW OF 13×10^6 LB/HR AND THE ASSOCIATED BLOCKAGE IS THE BEST ESTIMATE OF THE CURRENT CORE CONDITIONS FOR THE FOLLOWING REASONS.

1) CALCULATIONS HAVE SHOWN THAT SUFFICIENT UO_2 COULD BE "PACKED" AROUND THE THERMOCOUPLE TO GET A TEMPERATURE READING 35° HIGHER THAN THE FUEL ASSEMBLY BULK EXIT TEMPERATURE.[6]
2) CALCULATION INDICATE THAT ON THE AVERAGE THE THERMOCOUPLE WERE READING 5°F TOO HIGH BEFORE THE INCIDENT.[7,9]
3) IT SEEMS IMPROBABLE THAT SUFFICIENT BLOCKAGE COULD EXIST TO REDUCE THE CORE FLOW TO THE 1×10^6 LB/HR PREDICTED IN METHOD ONE.
4) THE SHIFT IN THE THERMOCOUPLE READING BEFORE, DURING, AND AFTER THE SWITCH FROM PUMP A1 TO A2 INDICATE THAT THE CHANGE IN THE FLOW DISTRIBUTION IS CAUSING

G. A. MEYER
APRIL 10, 1979
TIME: 1630

THE DEBRIS TO SHIFT WHICH IS AFFECTING THE THERMOCOUPLE READINGS. THE SHIFTING DEBRIS MAKE THE METHOD ONE ANALYSIS MORE SUSPECT AND LENDS CREDENCE TO THE HYPOTHESIZED LARGE BED OF DEBRIS CALCULATED IN METHOD TWO.

REFERENCES

1. JH Jones to GA Meyer "Estimate of Core Flow"

2. PS Bartells to Distribution, "Core Pressure Drop for Natural Circulation Calculation", 4/9/79

3. R. M. Hiatt to G. A. Meyer, "LYNXL Model for TMI-2 Blockage Study", 4/10/79

4. Core Condition Task Force, "Estimate of Loose Core Debris Volume (4/9/79 - 2000), 4/9/79

5. P. J. Henningson to G. A. Meyer, "Damage Model - Fluidized Bed", 4/10/79

6. P. J. Henningson to G. A. Meyer, "Increased T.C. Readings Due to Proximitty of Fuel Particulates"

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8) R. M. Gribble to J. D. Carlton, "Core Bypass Flow For Core Blocked at Top Only", 4/8/79

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OTHER INFORMATION

1) R. M. Hiatt to G. A. Meyer, "Core Flow Distribution for One Pump and two Pump Operation". 4/10/79

3.1 M

(2) INCORE THERMOCOUPLE

DESCRIPTION

The incore thermocouple design for TMI-2 is a grounded junction chromel-Alumel detector. The location of the detector in relation to the fuel assembly upper end fitting is shown on Figure 1. The thermocouple wire diameters are approximately 10 mils, the sheath is 62 mils OD Inconel tubing, and the insulation material is Al_2O_3. The location of the thermocouple within the instrument tube is shown in Figure 2. There are 50 thermocouples at locations in the core shown on Figure 6.

THERMOCOUPLE READINGS

The highest temperature readings of the thermocouples are plotted from 0400, 3/28 to 1300, 4/6 on Figure 3. ~~a~~At about 1300 on 4/6 a RC pump switch from 1A to 2A occurred which caused a redistribution of thermocouple readings. The change in thermocouple readings as a result of the pump switch is shown on Figure 4. The highest thermocouple readings after the pump switch are shown on Figure 5. Complete sets of thermocouple readings at selected times of each day are given on Figure 6.

The major point addressed in this section is to evaluate the thermocouple readings to determine the validity of the temperatures. This was done by examining detector accuracy and by evaluating fuel pellet debris accumulated around the thermocouple. The results of this evaluation are that the thermocouples are reasonably accurate and that the high temperatures are a result of fuel particles in the upper end fitting.

5) DETECTOR ACCURACY

PAST EXPERIENCE WITH THERMOCOUPLES INSTALLED IN B&W PLANTS SHOW TWO BASIC TRAITS. THEY TRACK FAIRLY STEADILY WHEN THE CORE IS AT STEADY STATE AND THEY RESPOND TO LOCAL CHANGES RELATIVELY ACCURATELY. A ONE TO TWO DEGREE CHANGE IN THERMOCOUPLE READING HAS OCCURRED MANY TIMES WITHOUT ANY OBVIOUS CHANGE IN CORE CONDITIONS. THERMOCOUPLE READING CHANGES GREATER THEN 2°F ARE REFLECTED IN OTHER MEASUREABLE LOCAL OR CORE CHANGES.

THE THERMOCOUPLES WERE EXPOSED TO TEMPERATURES GREATER THAN (1800°F) DURING THE INCIDENT AT TMI-2. TO DETERMINE IF THESE ELEVATED TEMPERATURES WOULD PRODUCE INACCURATE THERMOCOUPLE READINGS, TESTS WERE CONDUCTED AT THE BELFAB SUBSIDIARY OF B&W. IN THESE TESTS, 4 THERMOCOUPLES WERE RAISED TO 2000°F FOR FOUR HOURS. THE THERMOCOUPLE READINGS WERE THEN CHECKED AGAINST KNOWN READINGS OVER THE RANGE OF 200 TO 1000°F. ALL TEST THERMOCOUPLES READ WITHIN 5% OF THE CALIBRATION VALUE. IT WAS CONCLUDED THAT THE HIGH TEMPERATURE EXPOSURE WOULD NOT SIGNIFICANTL ALTER THE THERMOCOUPLE READING.

THERMOCOUPLE DATA FROM TMI-2 DURING NORMAL OPERATION BEFORE THE LOFW TRANSIENT WAS EXAMINED TO DETERMINE

IF A SYSTEMATIC BIAS EXISTED. NO DATA WAS AVAILABLE AT ZERO POWER CONDITIONS, THEREFORE DATA FROM SEVERAL POWER LEVELS WAS EXTRAPOLATED DOWN TO ZERO POWER. THE RESULTS OF THIS ANALYSIS SHOW THAT A + 5°F BIAS IS POSSIBLE ON THE THERMOCOUPLE READINGS.

THE POSSIBLILITY OF DE-CALIBRATION SINCE THE INCIDENT HAS ALSO BEEN EVALUATED. SIX THERMOCOUPLES WHICH HAD A TEMPERATURE RISE OF 7 to 33°F OVER A 5 DAY PERIOD WERE EXAMINED. NO MECHANISM COULD BE POSTULATED WHICH WOULD CAUSE DECALIBRATION TO THE EXTENT SHOWN BY THE DATA. IT WAS CONCLUDED THAT THE THERMOCOUPLE READINGS WERE ACCURATE AND THAT TRUE TEMPERATURES WERE BEING MONITORED TO ± 5°F.

THE RESULTS OF THE THERMOCOUPLE TESTS AND EVALUATIONS INDICATE THAT THE TEMPERATURE READINGS ARE GENERALLY ACCURATE WITH A POSSIBLE + 5°F BIAS.

3.7A (3) HEAT CONDUCTION FROM FUEL PELLET DEBRIS

An evaluation was performed to determine if the quasi steady state thermocouple readings of 100 to 240°F above coolant temperature could be caused by fuel pellet agglomerates located on spacer grids or in end fittings.

One evaluation was performed assuming all the pellets from one grid span were evenly distributed across the upstream grid. Enough heat is generated in this mass to produce boiling in the annulus between the instrument string and the guide tube. The steam would be vented out of the guide tube below the upper end fitting. Heat transfer to the thermocouple would be from axial conduction along the instrument string. Due to the large amounts of fuel debris required and the fact that the instrument guide tube would be cooled above the fuel mass, it is concluded that this mechanism is not the most likely reason for the high thermocouple readings.

It is more likely that fuel pellets and pellet fragments have collected in the upper end fitting and/or the mixing cup. Reactor coolant flow is sufficient to carry fuel pellet fragments above the upper end fitting where they would settle out in this stagnent region. An analysis was performed which assumed various widths of fuel assumulation in the upper end fitting around and in the mixing cup. The results of this analysis is shown on Figures 7 and 8. The case with fuel particles inside the mixing cup (T.C. Well) shows a small (10°F) increase in temperature which is insignificant. Fuel debris to a width of 3 to 4 inches in the upper end fitting itself does produce the magnitude at of ΔT's which are seen on the readings. This amount of fuel could easily fit within the upper end fitting which has an interiour width of 7 inches.

Bibliograph for 3.7A(2) and (3)

For A.B. Jackson
3217

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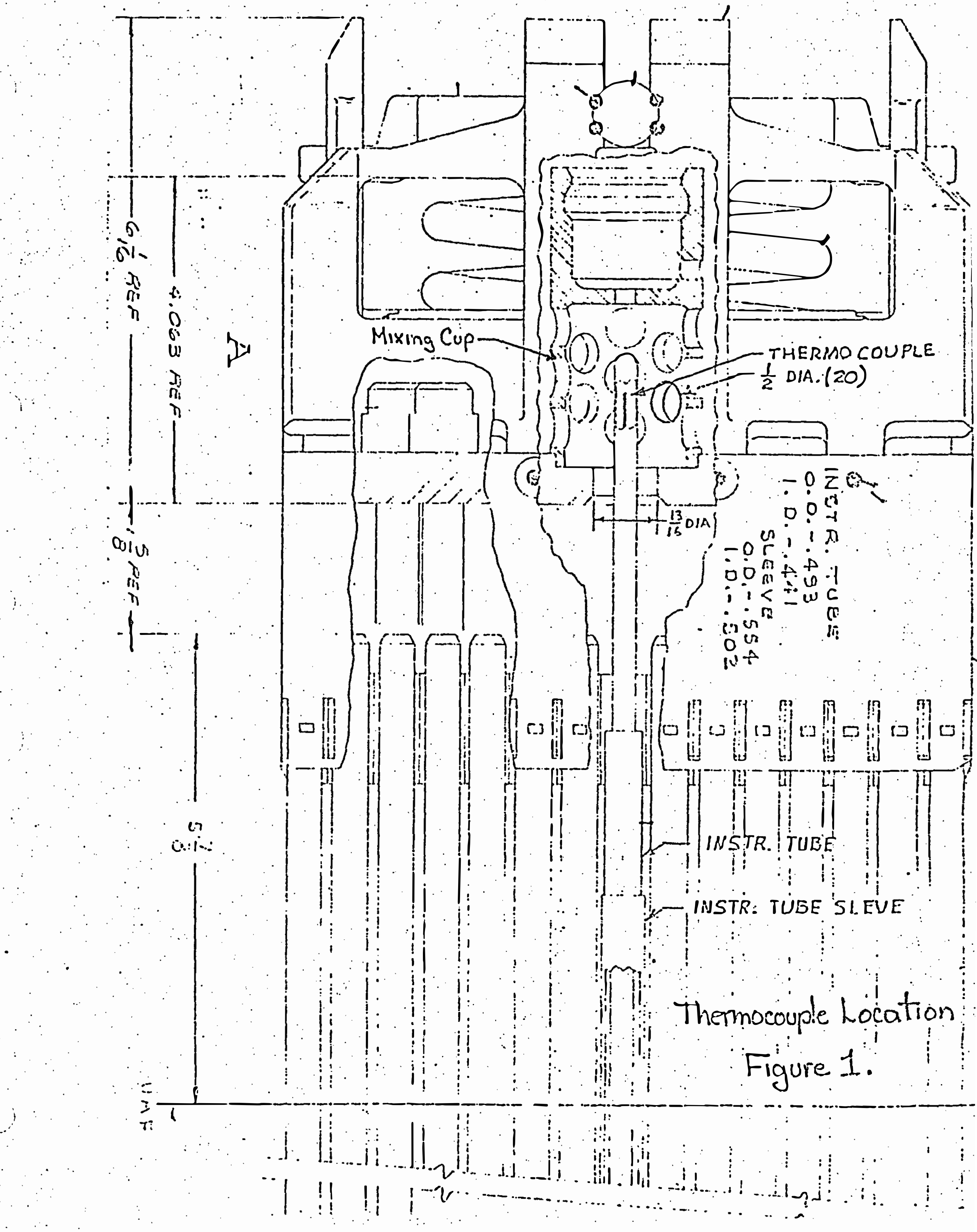
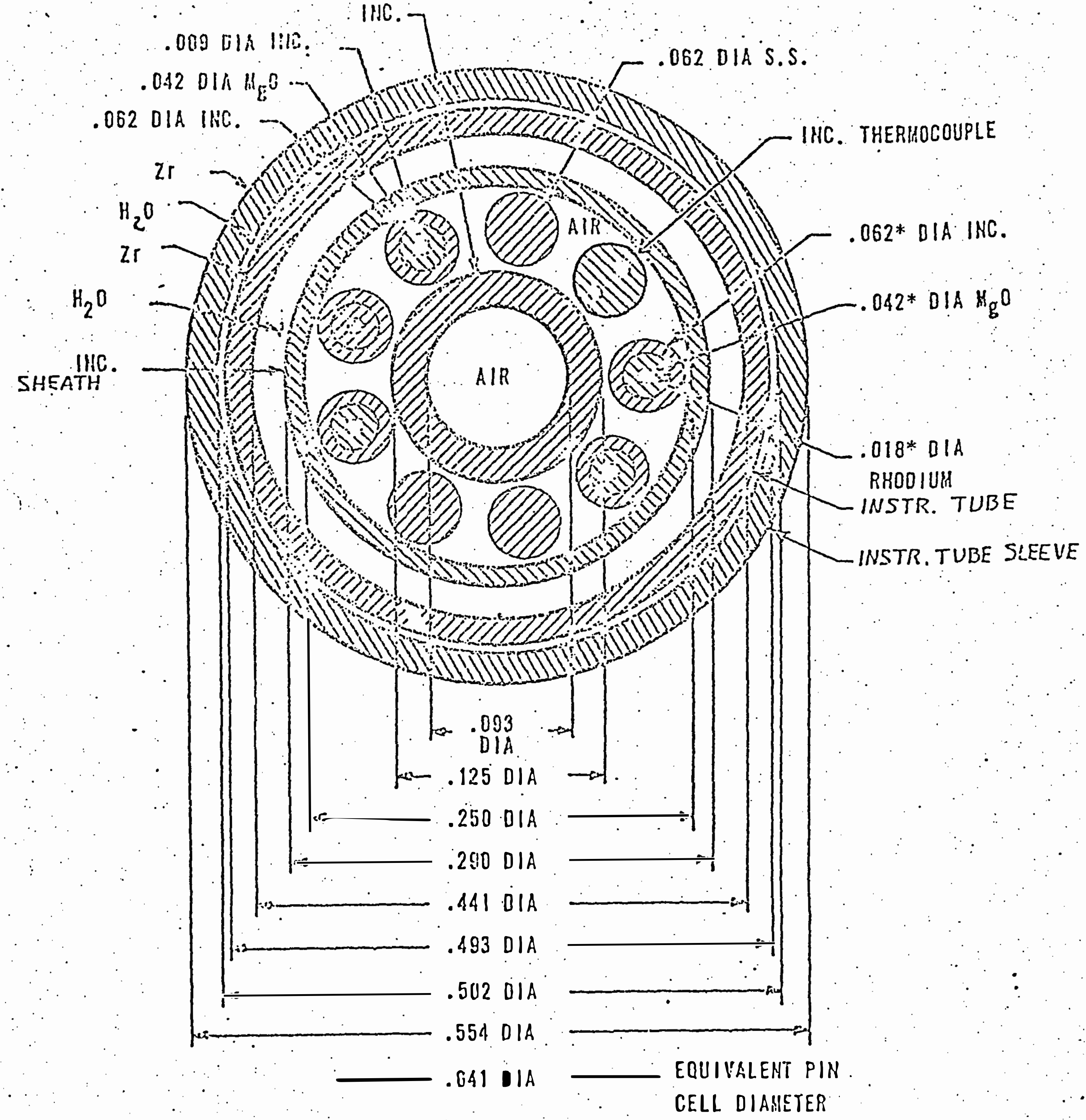
5. Memo, J.T. Willse to Engineering Operations Manager, "Action Item 143", April 6, 1979.

6. Memo, P.J. Henningson to G.A. Meyer "Increased T.C. Readings Due to Proximity of Fuel Particulates", April 10, 1979.



Thermocouple Location

Figure 1.



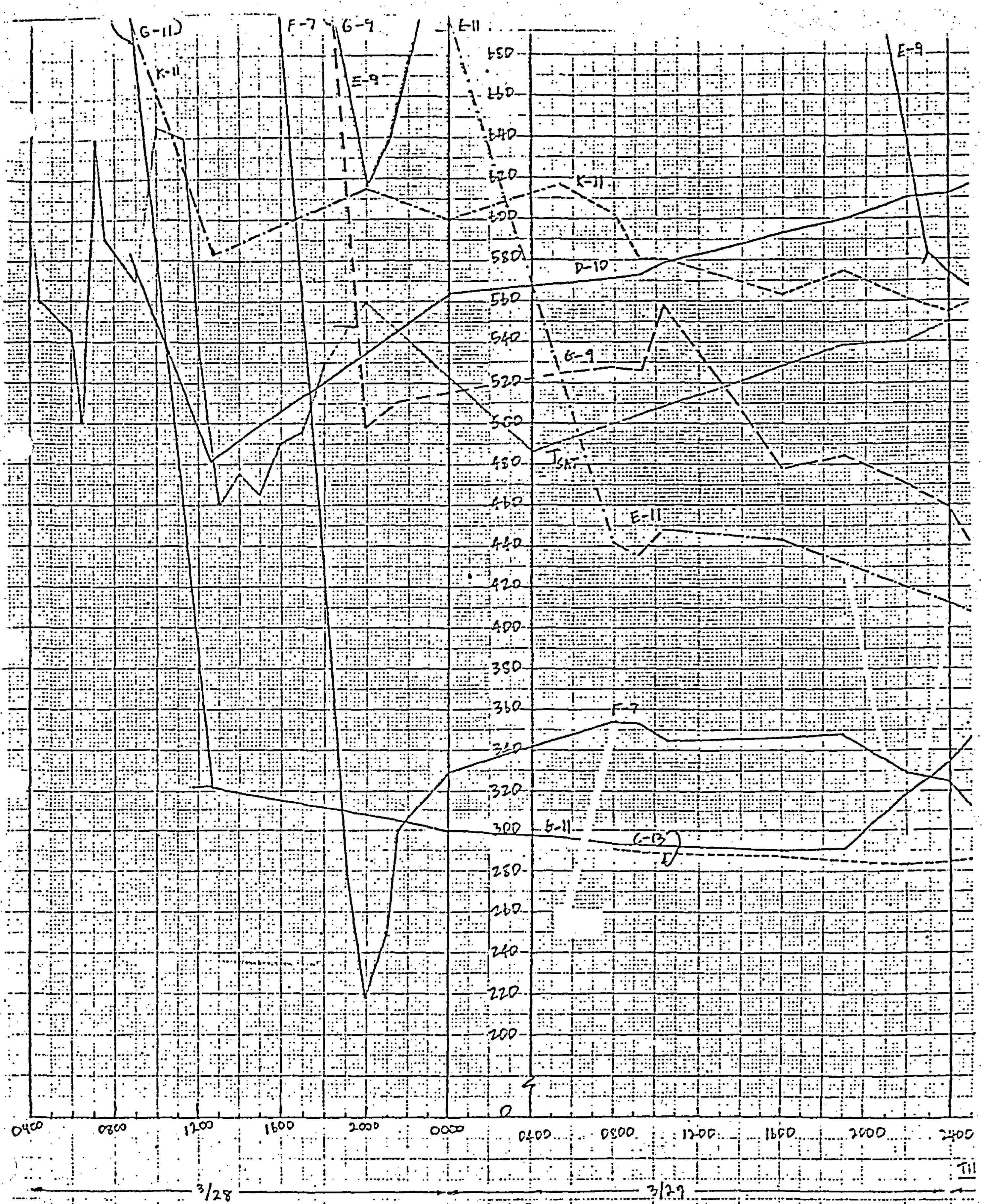
Instrument Crossection

Figure 2.

SPECIFICATION NO. 1014/1269

SPECIFICATION NO. 1134/1169

DRAWING NO. 136010E - REV. 6


G-11
F-7
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E-11
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K-11
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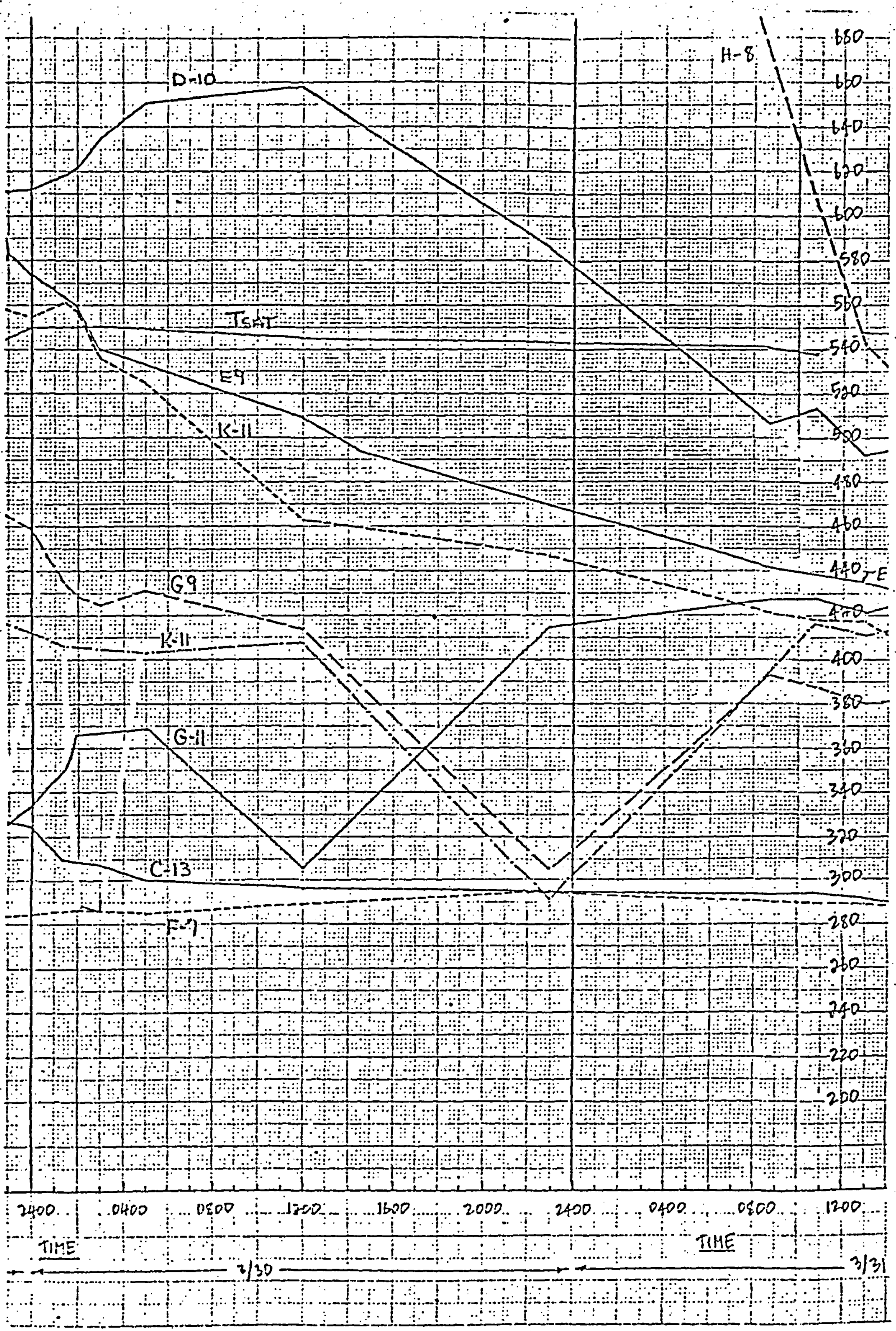
Figure 3a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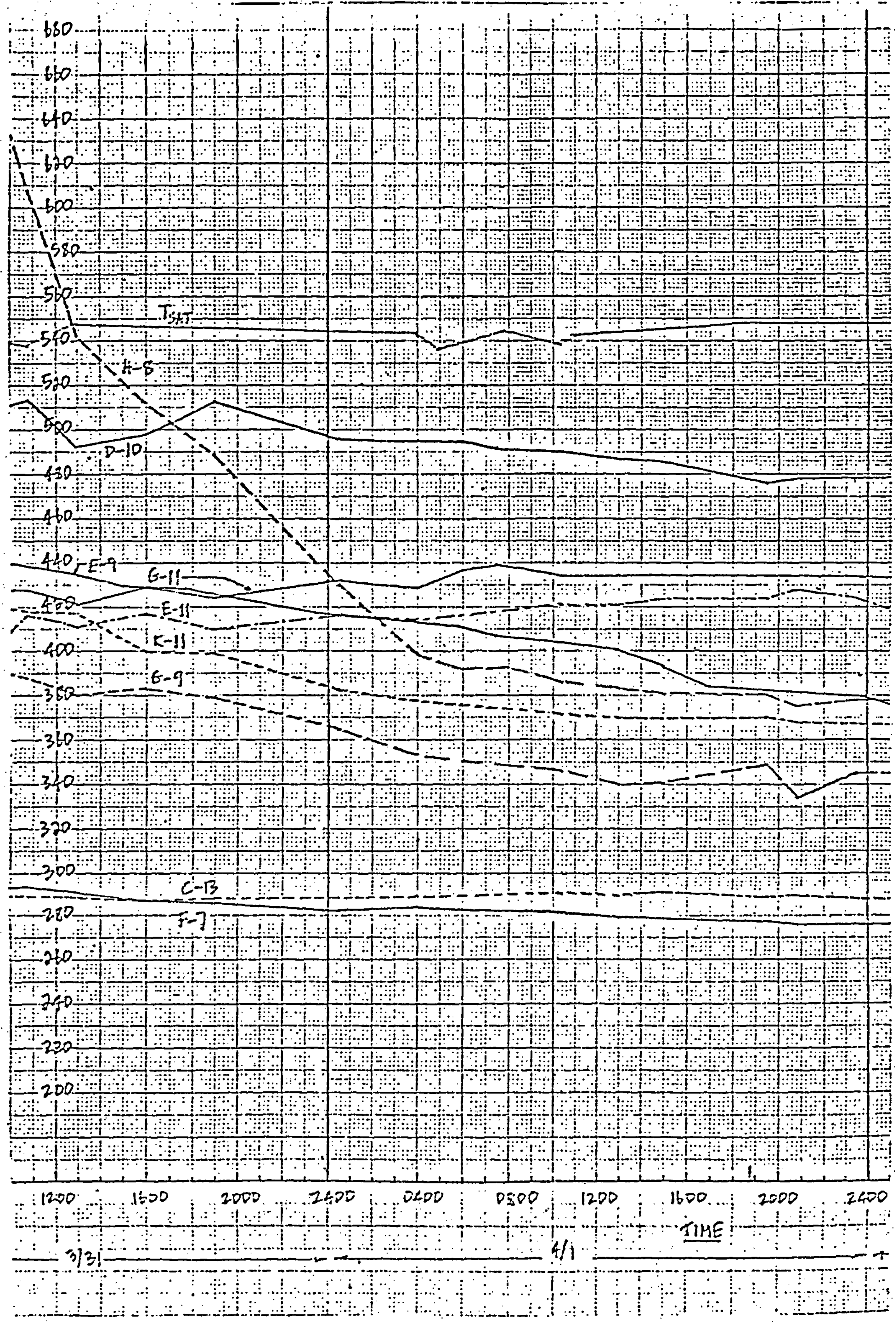
TIME
3/30
3/31

Figure 3b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C-13
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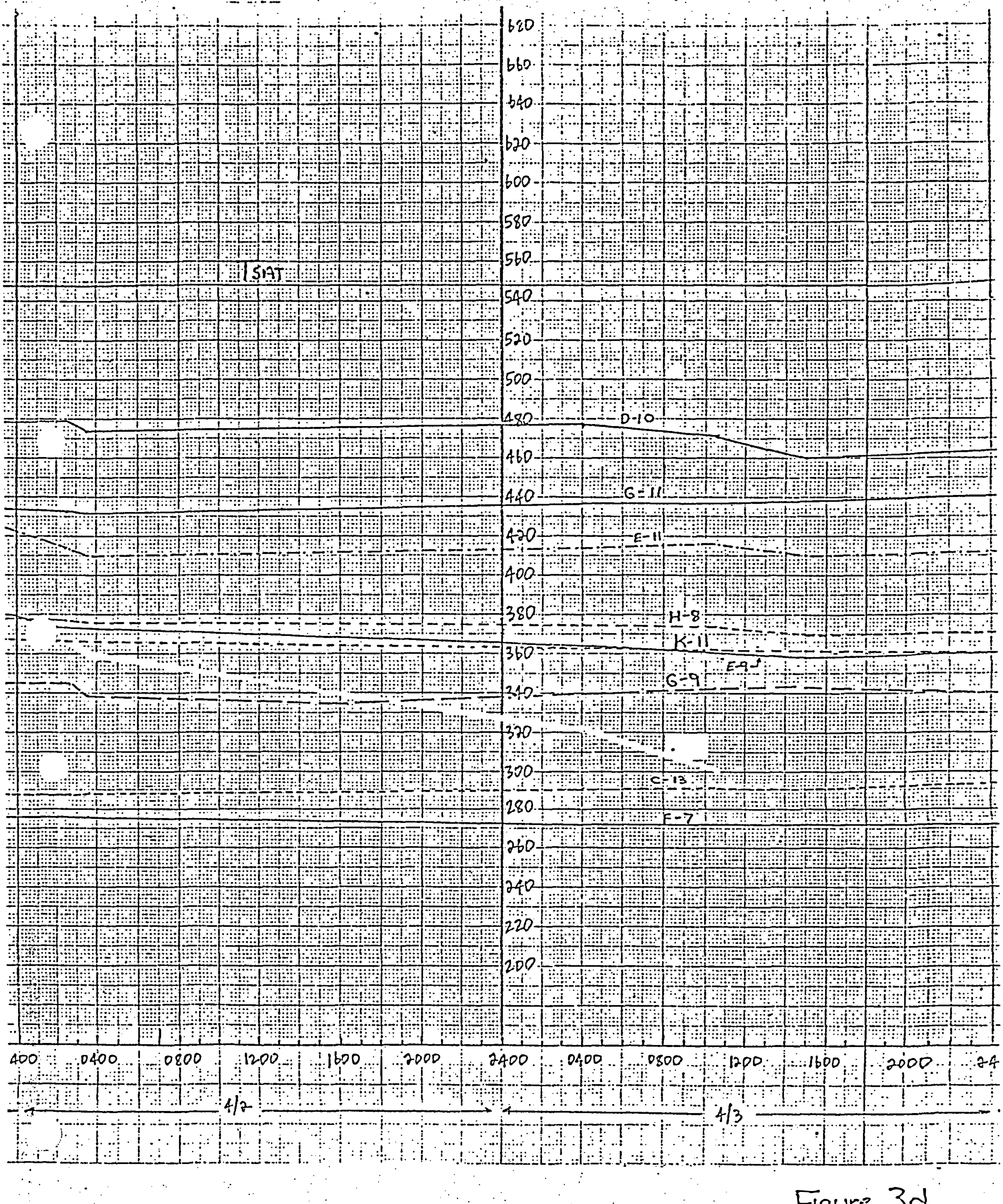
TIME
3/31
4/1

Figure 3c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G-11
E-11
H-8
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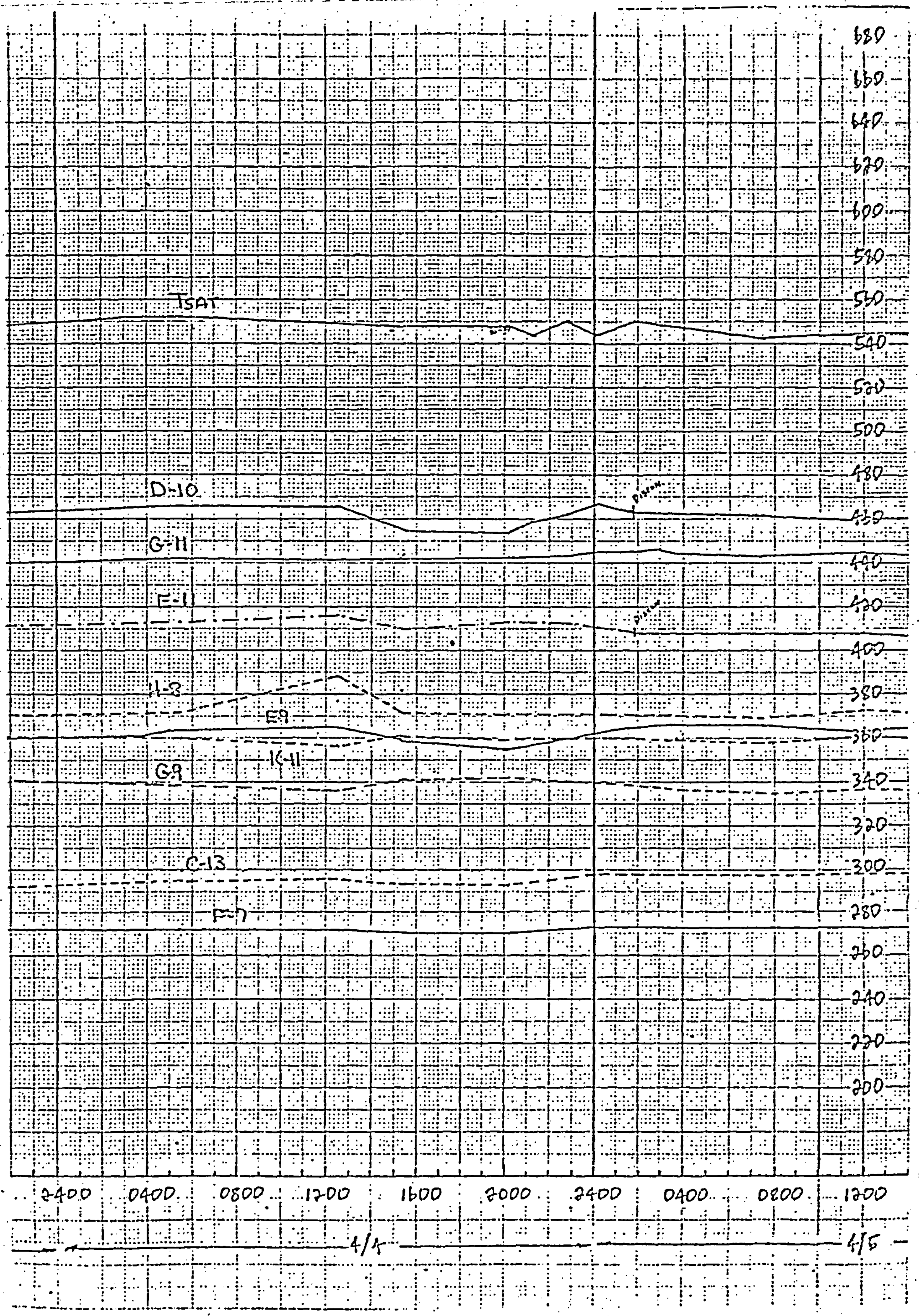
4/3

Figure 3d



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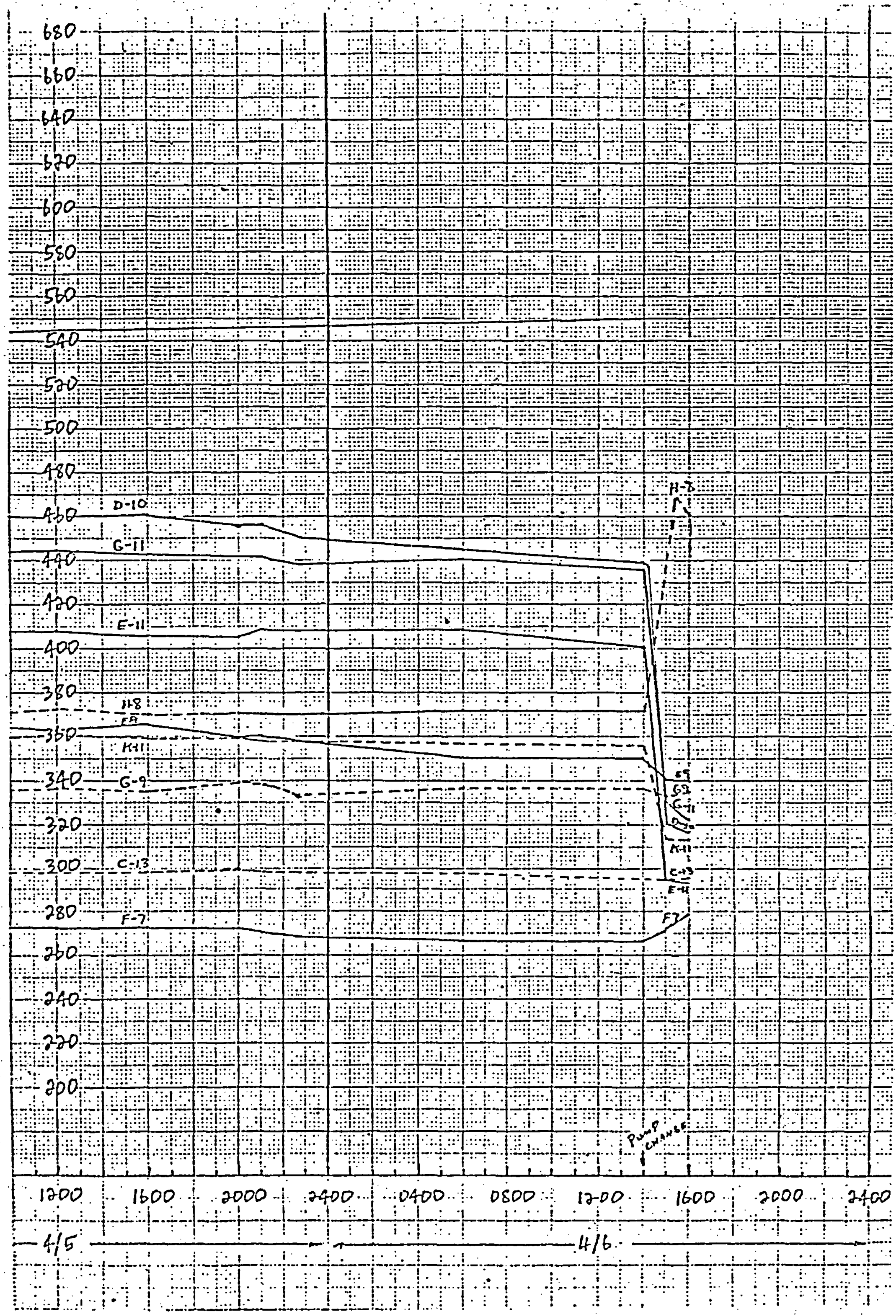
4/5

Figure 3e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K-11
C-13
E-11
F7
PUMP CHANGE
1200
1600
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2400
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0800
1200
1600
2000
2400
4/5
4/6

Figure 3f

ORM 140

THE BABCOCK & WILCOX CO.

Time ______

RC Temp. ______

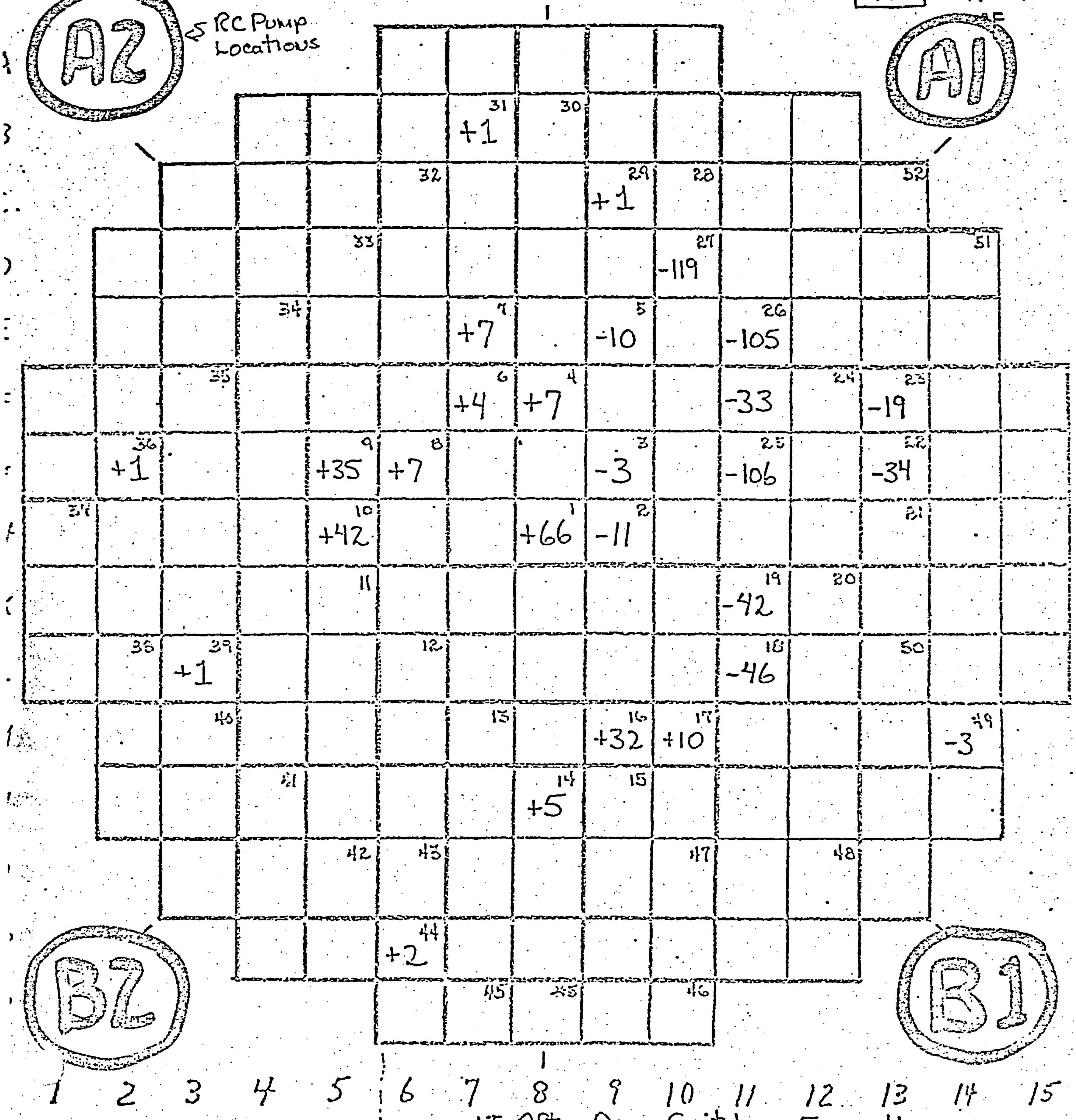
RC Press. ______

Name ______

SPND STRING NUMBERS AND LOCATIONS – 177 FA CORE

K – STRING NO.

XXX – T/C Read

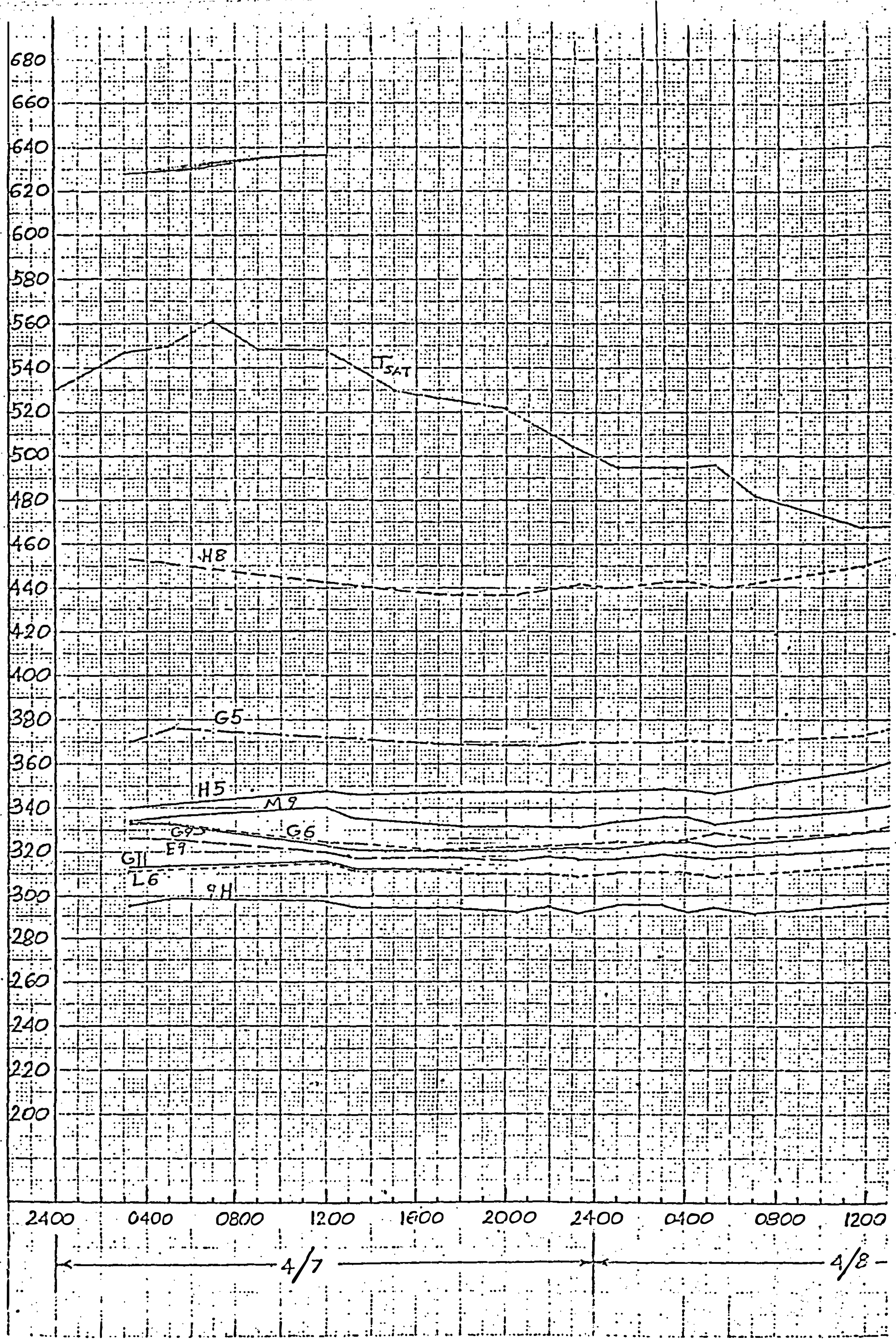


ΔT After Pump Switch – Figure 4.

USTOMER GPU

JOB NO.

UBJECT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H5
M9
G9
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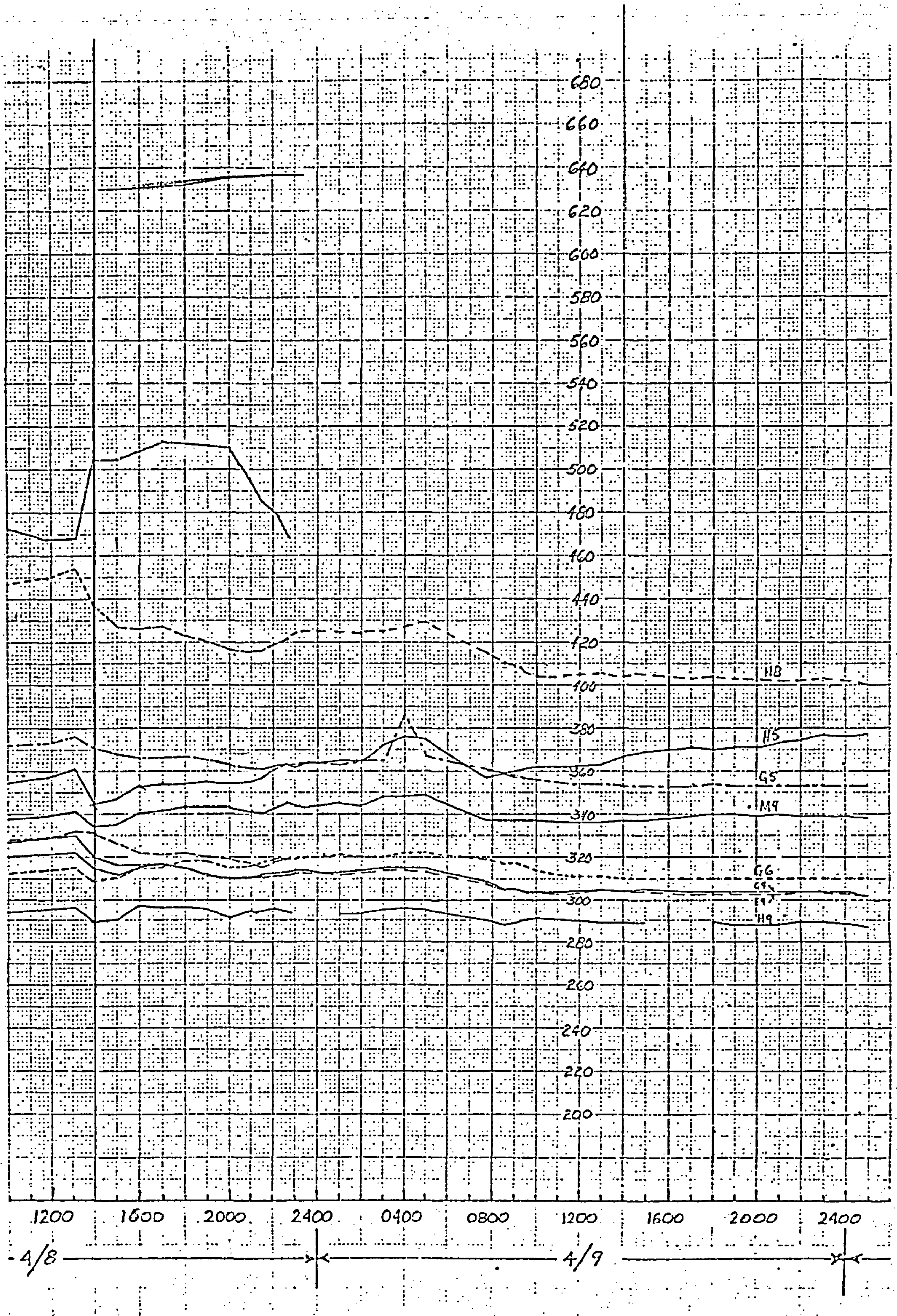
4/7
4/8

Figure 5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G5
M9
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E9
H9
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2400
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0800
1200
1600
2000
2400
4/8
4/9

Figure 5

THE BABCOCK & WILCOX CO.

TIME 1035

RC TEMP

RC PRESS

SPND ~~STRING~~ NUMBERS AND LOCATIONS — 177 FA CORE

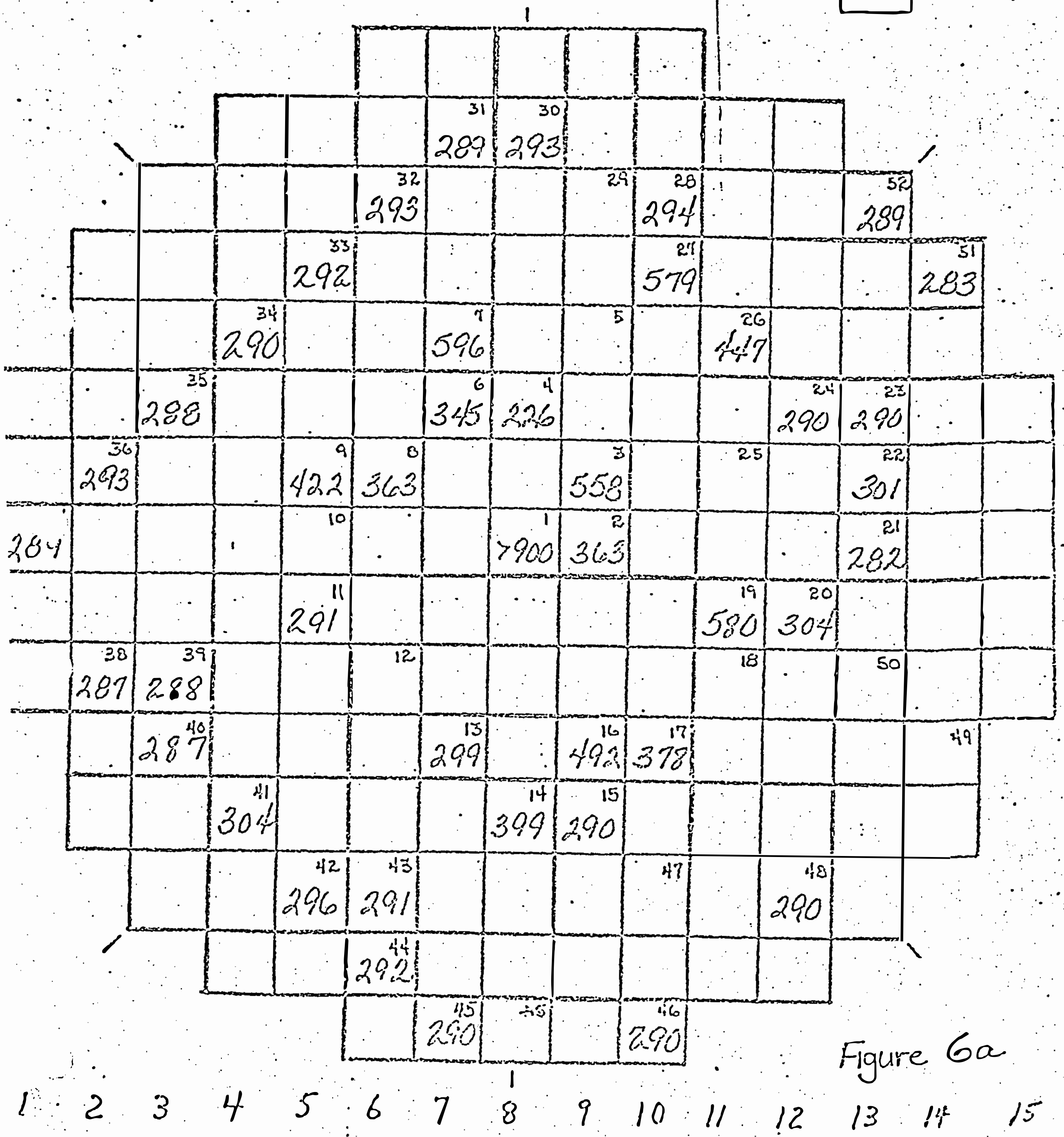


Figure 6a

THE BABCOCK & WILCOX CO.

TIME 2345
RC TEMP
RC PRESS

SPND ~~STRING~~ NUMBERS AND ~~LOCATIONS~~ - 177 FA CORE

K STRING NO.

						31 293	30 296							
					32 297			29 319	28 297			52 295		
				33 296					27 586				51 292	
			34 293			7 359		5 465		26				
	35 292					6 295	4 233				24 310	23 305		
	36 296			9 336	8 316			3 385		25 420		22 333		
293				10 299			1 895	2 322				21 307		
				11 296						19 447	20 325			
	38 294	39 293			12 296					18 302		50		
		40 298				13 299		16 350	17 326				49 300	
			41 289				14 348	15 295						
				42 296	43 294				47 297		48 294			
					44 293									
						45 283	~~45~~		46 293					
1	2	3	4	5	6	7	8	9	10	11	12	13	14	15

Figure 6b

THE BABCOCK & WILCOX CO.

TIME 0845
RC TEMP 280.5
RC PRESS 978

SPND ~~STRING~~ NUMBERS AND LOCATIONS – 177 FA CORE

K STRING NO.

1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
						31 289	30 291							
					32 292			29 311	28 291			52 290		
				33 289					27 507				51 287	
			34 288			7 356		5 442		26 417				
		35 289				6 293	4 242				24 303	23 298		
	36 290			9 330	8 319			3 392		25 427		22 330		
37 243				10 294			1 678	2 314				21 310		
				11 291						19 421	20 327			
	38 289	39 287			12 290					18 302		50		
		40 295				13 294		16 344	17 320				49 294	
			41 291				14 341	15 291						
				42 292	43 288				47 293		48 288			
					44 286									
						45 287	45		46 288					

Figure 6c

THE BABCOCK & WILCOX CO.

TIME

RC TEMP 279.7

RC PRESS 979

SPND ~~STRING~~ NUMBERS AND LOCATIONS – 177 FA CORE

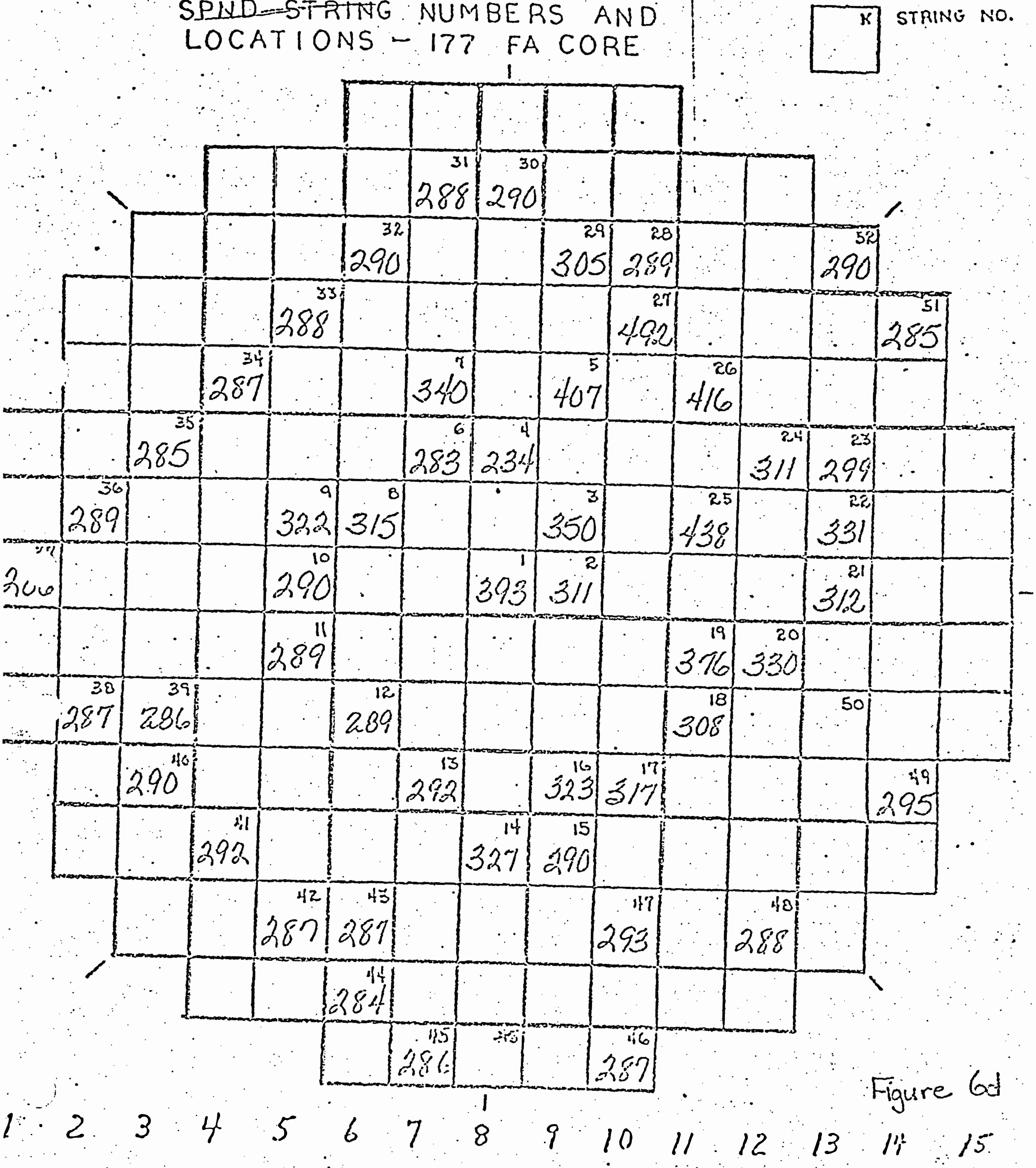


Figure 6d

THE BABCOCK & WILCOX CO.

Time 0337
RC Temp 277.9
RC Press 910

SPND STRING NUMBERS AND LOCATIONS – 177 FA CORE

☐ X STRING NO.

1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
						31 286	30 287							
					32 287			29 300	28 287			52 288		
				33 285					27 473				51 278	
			34 284			7 325		5 384		26 410				
		35 283				6 275	4 227				24 317	23 299		
	36 285			9 320	8 313			3 339		25 432		22 330		
284				10 287			1 376	2 303				21 310		
				11 286						19 360	20 326			
	38 286	39 284			12 286					18 312		50		
		40 285				13 288		16 318	17 312				49 291	
			41 288				14 306	15 288						
				42 284	43 285				47 290		48 286			
					44 279									
						45 284	45		46 283					

Figure 6e

THE BABCOCK & WILCOX CO.

TIME 054[illegible]
RC Temp 279.5
RC Press 993

SPND STRING NUMBERS AND LOCATIONS – 177 FA CORE

K STRING NO.

1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
						31 287	30 289							
					32 289			29 307	28 289			52 290		
				33 288					27 472				51 281	
			34 286			7 313		5 365		26 414				
		35 285				6 273	4				24 319	23 307		
	36 287			9 320	8 316			3 343		25 438		22 327		
[illegible] 2				10 288			1 376	2 310				21 317		
				11 288						19 363	20 333			
	38 286	39 286			12 287					18 321		50		
		40 286				13 290		16 319	17 315				49 292	
			41 290				14 302	15 290						
				42 284	43 286				47 294		48 288			
					44 281									
						45 286			46 286					

Figure 6 F

THE BABCOCK & WILCOX CO.

Time 0532
RC Temp 283.6
RC Press 1038

~~SPND~~ STRING NUMBERS AND LOCATIONS - 177 FA CORE

K STRING NO.

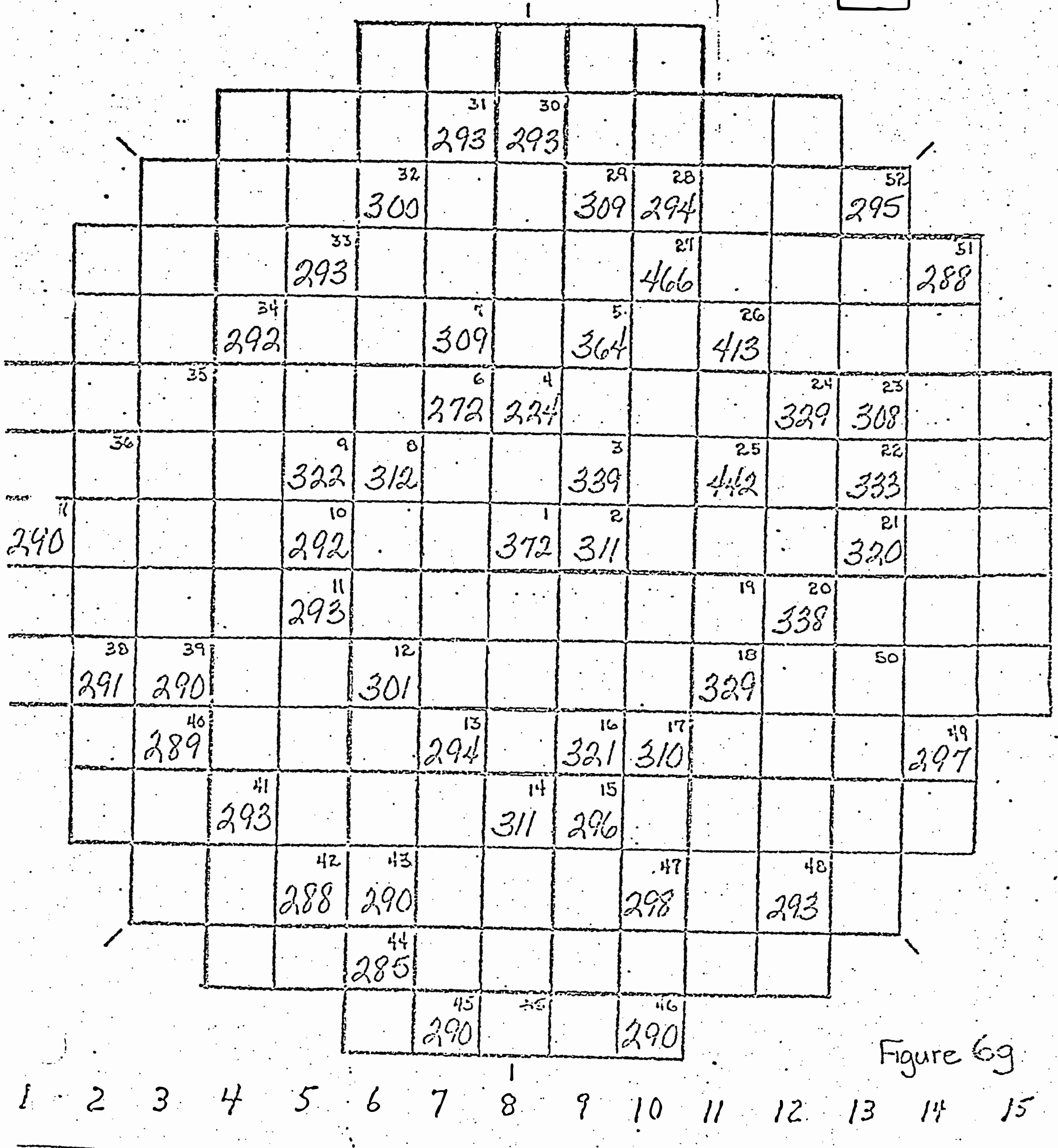


Figure 6g

THE BABCOCK & WILCOX CO.

Time 0542
RC Temp 284.9
RC Press 1005

SPND ~~STRING~~ NUMBERS AND LOCATIONS – 177 FA CORE

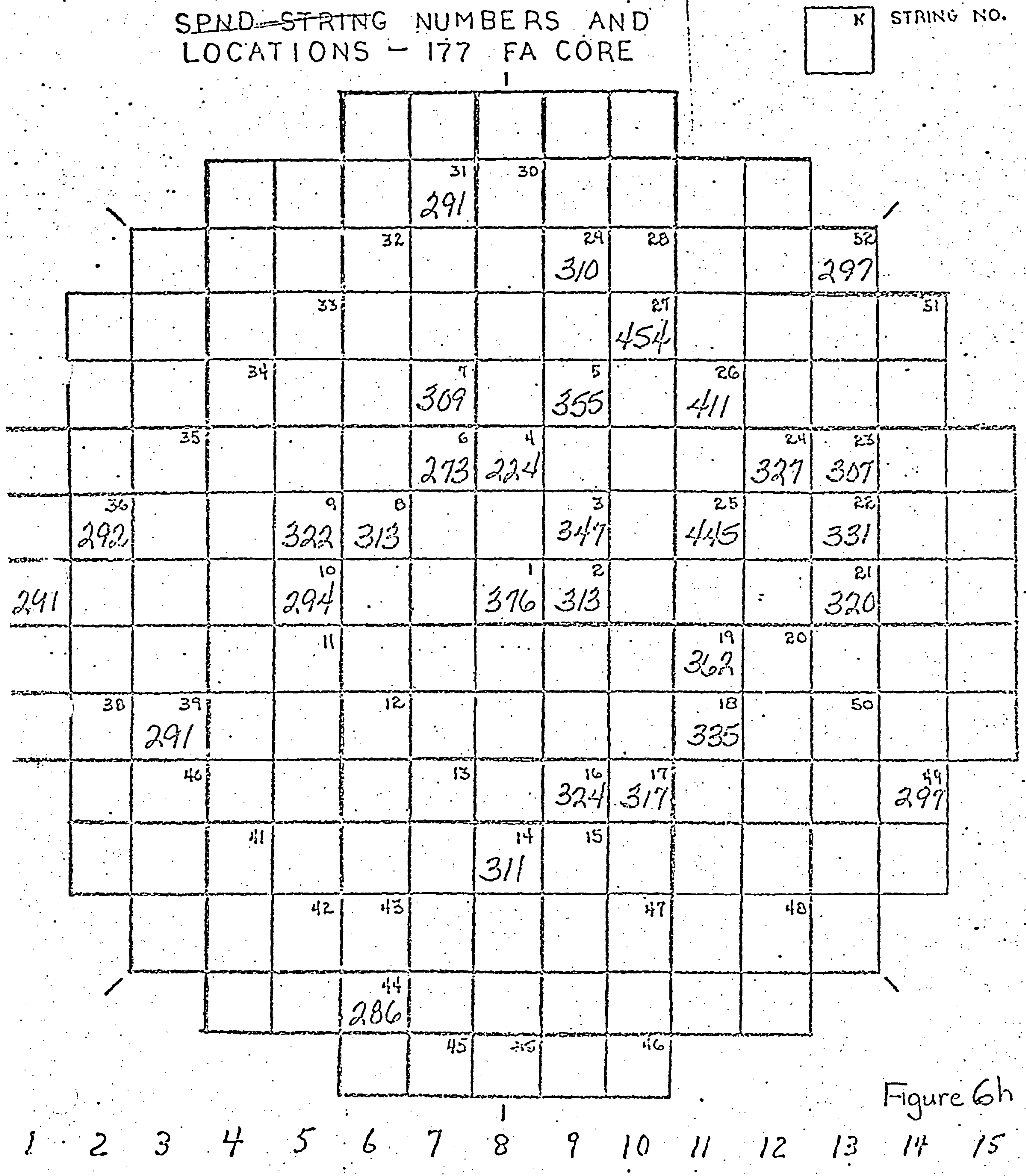


Figure 6h

RM 140

THE BABCOCK & WILCOX CO.

TIME 0541
RC TEMP 284.6
RC PRESS 1111

SPND STRING NUMBERS AND ~~LOCATIONS~~ – 177 FA CORE

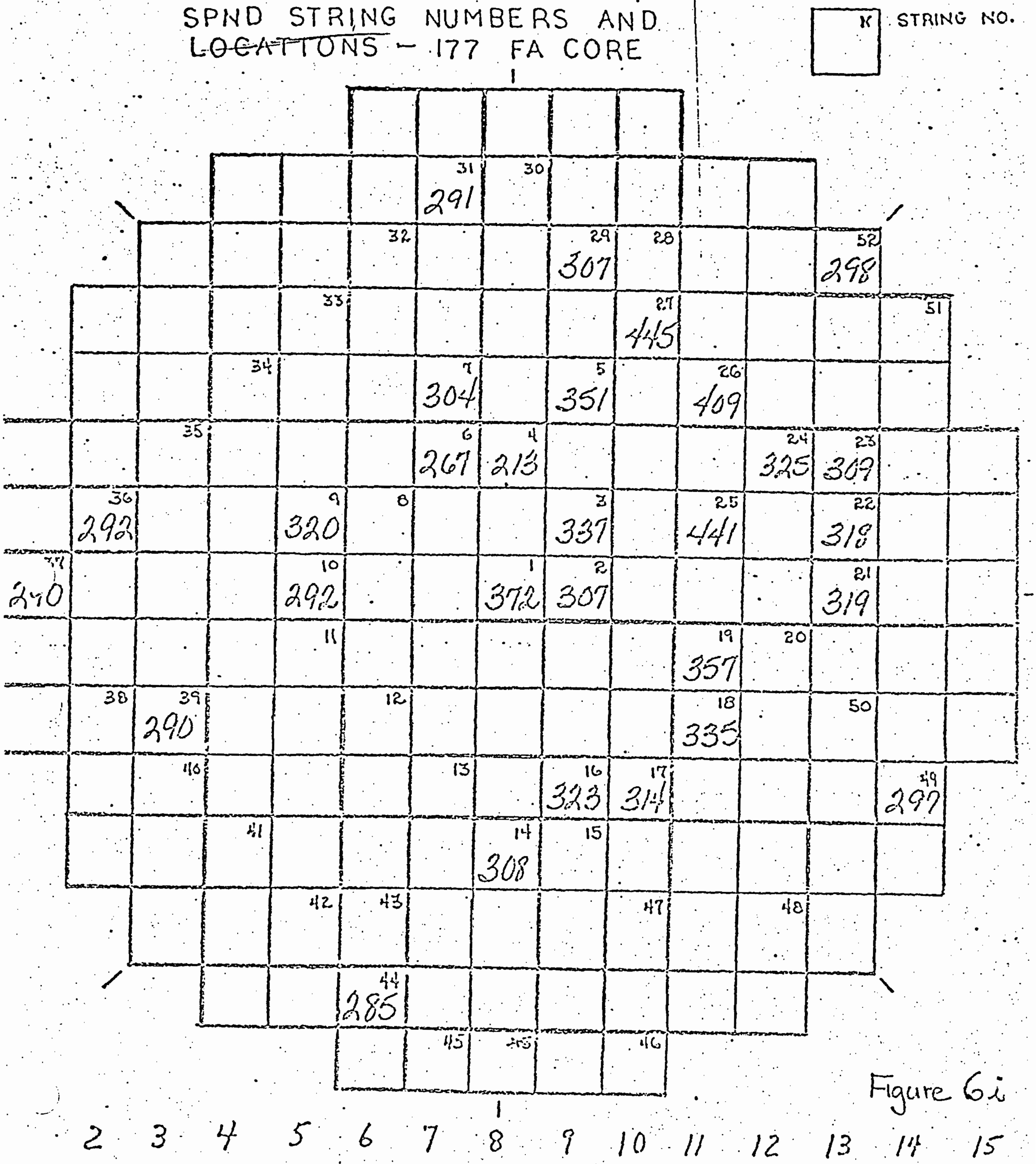


Figure 6i

THE BABCOCK & WILCOX CO.

TIME 0500
RC TEMP 284.8
RC PRESS 978

SPND ~~STRING~~ NUMBERS AND LOCATIONS - 177 FA CORE

☐ X STRING NO.

						31 292	30 293							
					32 295			29 304	28 293			52 294		
				33 294					27 294				51 291	
			34 291			7 300		5 326		26 291				
		35 298				6 272	4 213				24 291	23 290		
	36 292			9 376	8 333			3 333		25 314		22 293		
293				10 341			1 451	2 298				21 290		
				11 300						19 315	20 290			
	38 292	39 295			12 312					18 293		50		
		40 293				13 295		16 336	17 313				49 294	
			41 297				14 308	15 300						
				42 297	43 303				47 294		48 292			
					44 307									
						45 286	45		46 292					
1	2	3	4	5	6	7	8	9	10	11	12	13	14	15

Figure 6j

THE BABCOCK & WILCOX CO.

TIME 0517
RC TEMP 279.5
RC PRESS 676

SPND ~~STRING~~ NUMBERS AND LOCATIONS – 177 FA CORE

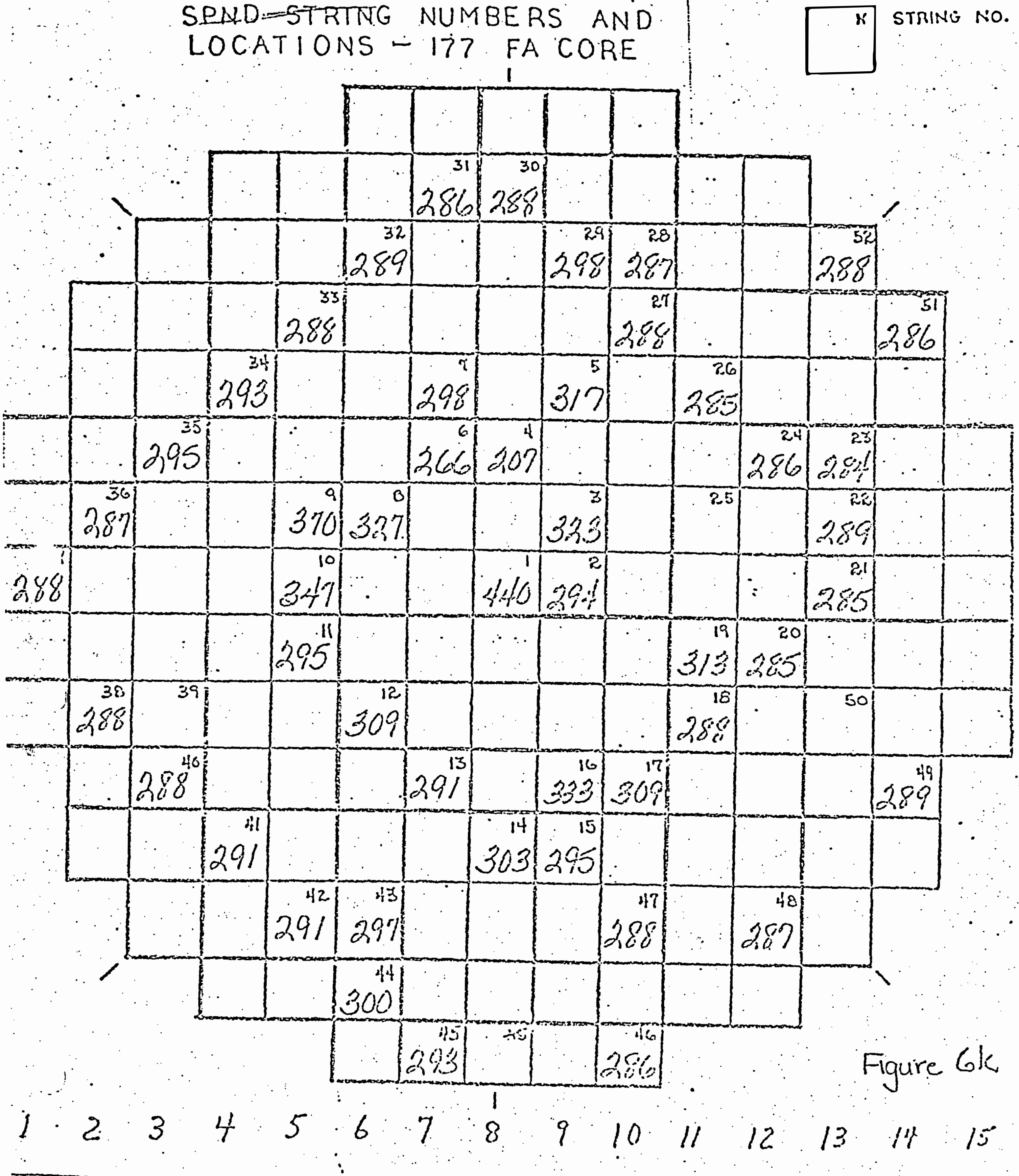


Figure 6k

THE BABCOCK & WILCOX CO.

TIME 0500
RC TEMP 281.1
RC PRESS 473

SPND STRING NUMBERS AND LOCATIONS — 177 FA CORE

K STRING NO.

1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
						31 287	30 289							
					32 290			29 299	28 288			52 291		
				33 290					27 289				51 288	
			34 291			7 296		5 313		26 287				
		35 298				6 262	4 201				24 288	23 285		
	36 289			9 367	8 322			3 315		25		22 290		
240				10 375			1 429	2 295				21 286		
				11 296						19 314	20 287			
	38 289	39			12 322					18 290		50		
		40 289				13 293		16 349	17 307				49 291	
			41 295				14 304	15 297						
				42 294	43 298				47 290		48 288			
					44 302									
						45 298	45		46 287					

Figure 6b

THE BABCOCK & WILCOX CO.

TIME 0605
RC TEMP 277.1
RC PRESS 941

SPND STRING NUMBERS AND LOCATIONS – 177 FA CORE

☐ X STRING NO.

						31 283	30 285							
					32 286			29 293	28 284			52 286		
				33 285					27 284				51 283	
			34 286			7 286		5 300		26 282				
		35 292				6 253	4 206				24 283	23 279		
	36 285			9 352	8 309			3 301		25		22 285		
7 286				10 376			1 398	2 286				21 281		
				11 294						19 302	20 282			
	38 285	39			12 313					18 284		50		
		40 286				13 287		16 335	17 299				49 286	
			41 289				14 298	15 291						
				42 290	43 294				47 286		48 284			
					44 294									
						45 291	35		46 283					
1	2	3	4	5	6	7	8	9	10	11	12	13	14	15

Figure 6m

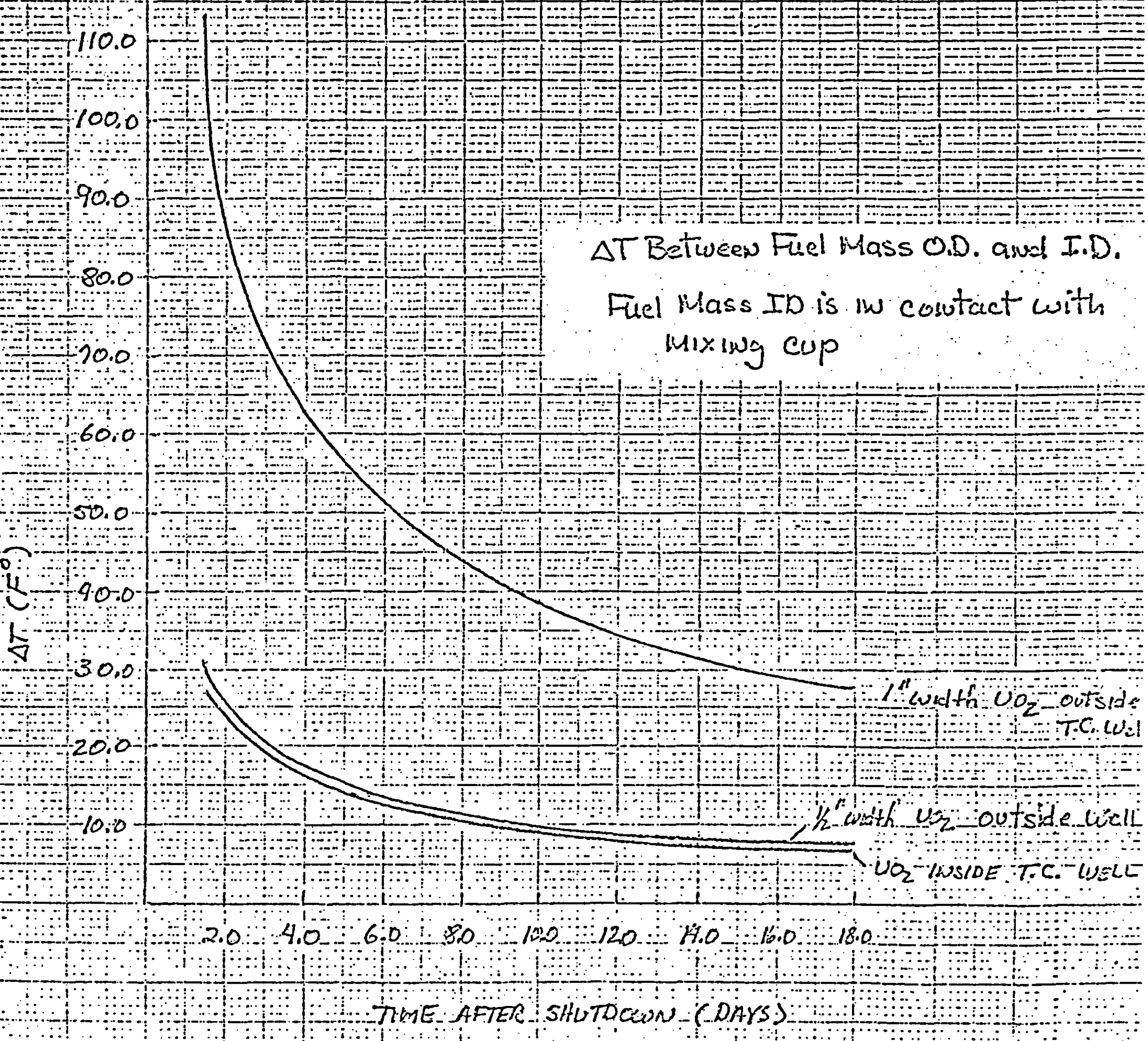

ΔT Between Fuel Mass O.D. and I.D.
Fuel Mass ID is in contact with mixing cup
110.0
100.0
90.0
80.0
70.0
60.0
50.0
40.0
30.0
20.0
10.0
ΔT (F°)
1" width UO2 outside T.C. Wel
½" width UO2 outside well
UO2 INSIDE T.C. WELL
2.0
4.0
6.0
8.0
10.0
12.0
14.0
16.0
18.0
TIME AFTER SHUTDOWN (DAYS)

Figure 7

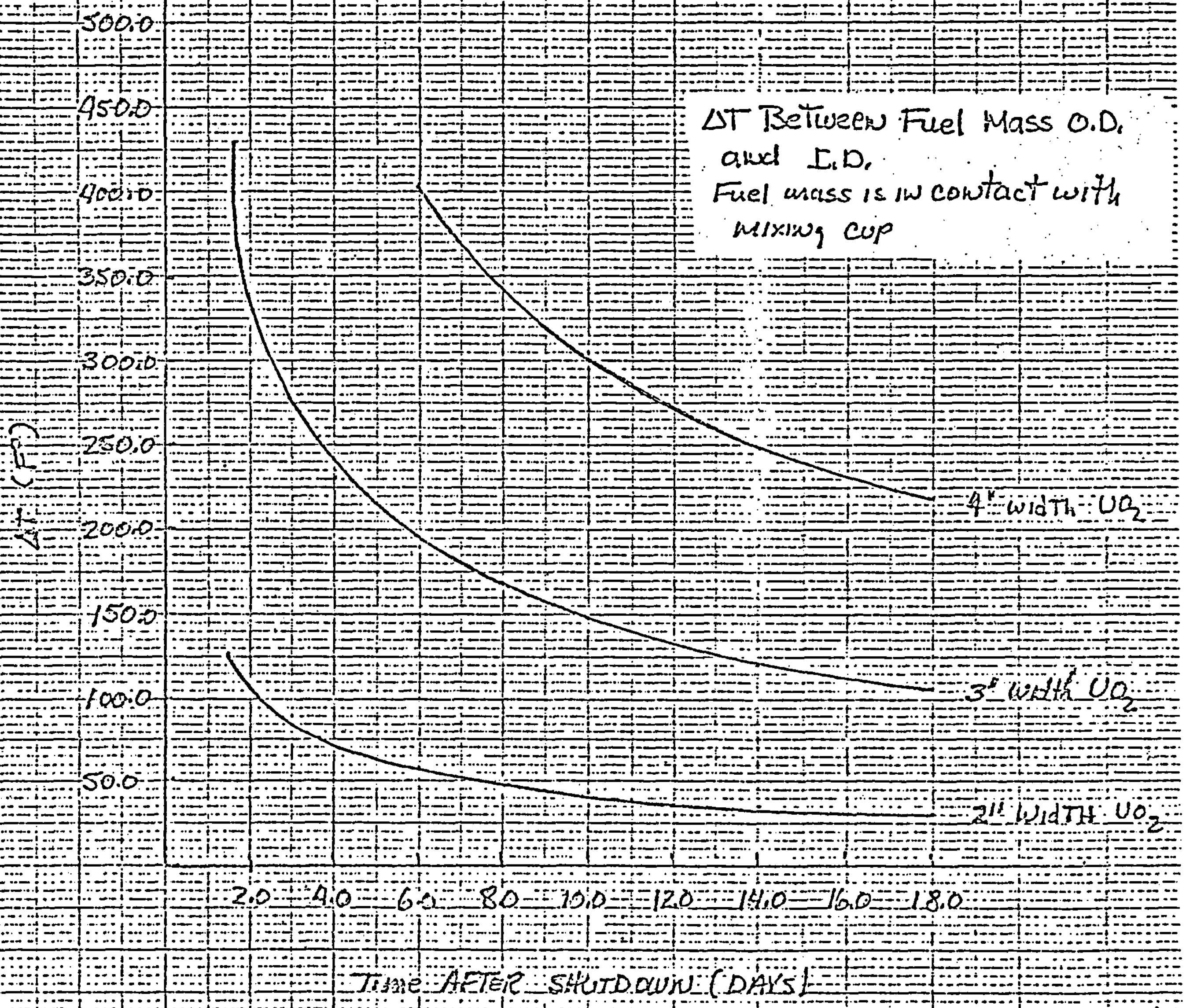

ΔT Between Fuel Mass O.D. and I.D.
Fuel mass is in contact with mixing cup
500.0
450.0
400.0
350.0
300.0
250.0
200.0
150.0
100.0
50.0
ΔT (°F)
4" width UO2
3" width UO2
2" width UO2
2.0
4.0
6.0
8.0
10.0
12.0
14.0
16.0
18.0
Time AFTER SHUTDOWN (DAYS)

Figure 8

4/10/79
7:50 pm

3.7 B

THERMAL-HYDRAULIC EVALUATION OF NATURAL CIRCULATION

OPERATION OF THE TMI-2 CORE WITH NATURAL CIRCULATION HAS BEEN EVALUATED AND JUDGED ACCEPTABLE ON A LONG TERM COOLING BASIS. CORE MINIMUM FLOW REQUIREMENTS AS A FUNCTION OF TIME AND OPERATING CONDITIONS HAVE BEEN DETERMINED. CRITERIA HAVE BEEN ESTABLISHED FOR THE ACCEPTABILITY AND ESTABLISHMENT OF NATURAL CIRCULATION. THE EFFECTS OF LOCALIZED HEATING HAVE BEEN EVALUATED AND LIMITS HAVE BEEN ESTABLISHED FOR THE INCORE THERMOCOUPLES AND HOT LEG RTD'S.

THE MINIMUM CORE FLOW REQUIREMENTS FOR LONG TERM COOLING ARE GIVEN IN REFERENCE 1. THE BASIS FOR THIS MINIMUM REQUIRED FLOW IS NO BULK COOLANT TEMPERATURE ABOVE SATURATION TEMPERATURE. THE CURVES PRESENTED IN REFERENCE 1 INCLUDE A SAFETY MARGIN OF 300% AND ARE COMPLETELY BOUNDED BY PREDICTIONS OF ACTUAL NATURAL CIRCULATION CORE FLOW ASSUMING CORE RESISTANCES 200 TO 1000 TIMES THE RESISTANCE OF A NORMAL 177 CORE.

CRITERIA FOR THE INCORE THERMOCOUPLES AND HOT LEG RTD'S ARE DESCRIBED IN REFERENCE 3. THESE CRITERIA ARE DESIGNED TO ALLOW ADEQUATE TIME FOR THE ESTABLISHMENT OF NATURAL CIRCULATION FLOW WHILE STILL ALLOWING FOR ADEQUATE SAFETY MARGIN AND TIME FOR TRANSITION TO ONE OF THE ALTERNATIVE COOLING SYSTEMS. THE DEGREE OF CONSERVATISM IN THE CRITERIA IS DEMONSTRATED BY ANALYSIS RESULTS PRESENTED IN REFERENCES 2, 3, & 4. THE REFERENCE 2 RESULTS DEMONSTRATE THAT TIMES IN ACCESS OF ONE HOUR ARE REQUIRED TO HEAT THE WATER CONTAINED ONLY IN THE CORE REGION TO SATURATION UNDER

NO-FLOW CONDITIONS AND POWER LEVELS BELOW 3 MWT. REFERENCE 3 DEMONSTRATES THAT AN ADDITIONAL HOUR WOULD BE REQUIRED (@3 MWT) TO BOIL OUT COOLANT FROM THE OUTLET NOZZLE LEVEL TO THE TOP OF THE CORE. REFERENCE 4 DEMONSTRATES THAT MORE THAN ONE HOUR IS REQUIRED (@3 MWT) FOR THE ADIABATIC HEAT-UP OF THE CORE FROM 200°F TO 2000°F ASSUMING INSTANTANEOUS CORE UNCOVERAGE.

NATURAL CIRCULATION RESPONDS TO HIGHER FLOW RESISTANCE BY GOING FURTHER INTO TWO-PHASE FLOW UNTIL THE VAPORIZED DENSITY CHANGE IS SUFFICIENT TO OVERCOME SYSTEM RESISTANCE. WITH DEBRIS BLOCKING A GRID, FLOW WILL PASS LATERALLY AROUND THE BLOCKAGE AND ADEQUATELY IMMERSE THE FUEL BEFORE AND AFTER THE BLOCKAGE. IF THE BLOCKAGE DEBRIS ALSO CONTAINS FUEL, THE MAXIMUM TEMPERATURE AT THE BLOCKAGE CENTER WILL BE CONDUCTION LIMITED AND RELATIVELY INSENSITIVE TO THE MODE OF SURFACE HEAT TRANSFER. THE FLOW PATTERNS IN NATURAL CIRCULATION ARE DIRECTLY RESPONSIVE TO RESTRICTIONS, AND MAY WELL SHOW LOCAL TWO-PHASE TRANSITIONS AND RELATIVELY HIGH VELOCITIES AT PARTIALLY-BLOCKED HOTTER REGIONS.

REDISTRIBUTION OF CORE DEBRIS IS ANTICIPATED WHEN THE PUMPED FLOW IS TERMINATED. CHANGES IN THERMOCOUPLE INCORE TEMPERATURE DISTRIBUTIONS SHOULD BE EXPECTED. NOT ALL THERMOCOUPLE READINGS CAN BE EXPLAINED BY HYDRAULIC PHENOMENA. SOME THERMOCOUPLES, PARTICULARLY NEAR THE CENTER, ARE CURRENTLY INDICATING LOCALIZED HEATING EFFECTS AND ARE NOT MEASURING BULK FLUID TEMPERATURES. HENCE, IT IS NOT REALISTIC TO REQUIRE ALL INCORE THERMOCOUPLE MEASUREMENT BE BELOW SATURATION TEMPERATURE, NOR IS IT NECESSARY.

REFERENCE 5 DISCUSSES THE ADVANTAGES AND DISADVANTAGES OF CORE COOLING AND NATURAL CIRCULATION AT VARIOUS PRESSURES. IT IS DESIRABLE ~~AND~~ PRUDENT TO ~~ACCOMPLISH THE INITIAL COOLDOWN FROM HIGH PRESSURE (UP TO 1800 PSIA) AND THEN~~ MAINTAIN LONG TERM COOLING AND NATURAL CIRCULATION AT ~~LOWER~~ PRESSURES IN THE RANGE OF 500-600 PSIA.

REFERENCES:

1) MEMO, G.A. MEYER TO J.D. CARLTON, PLANT DESIGN, "MINIMUM CORE FLOW-LONG TERM COOLING", APRIL 4, 1979.
2) MEMO, L.L. LOSH/J.F. BURROW TO G.A. MEYER, THERMAL-HYDRAULICS, "CRITERIA DURING ESTABLISHMENT OF NATURAL CIRCULATION", APRIL 10, 1979.
3) MEMO, J.A. WEIMER, R.L. HARNE TO DISTRIBUTION, "BOILING CONDITIONS IN CORE", APRIL 1, 1979.
4) MEMO, J.H. JONES TO G.A. MEYER, THERMAL-HYDRAULICS, "ADIABATIC CORE HEATUP RATES", APRIL 10, 1979.
5) MEMO, J.R. GLOUDEMANS TO G.A. MEYER, THERMAL-HYDRAULICS, "COOLDOWN PRESSURE", APRIL 10, 1979.

3.7 CORE THERMAL BEHAVIOR

C. ANALYSIS OF ALTERNATIVES

THE PLANNED MODE OF OPERATION, DISCUSSED IN SUB-SECTION B, IS TO ESTABLISH NATURAL CIRCULATION FOR THE LONG-TERM COOLDOWN. IF THE SYSTEM RESPONSE AFTER THE REACTOR COOLANT PUMP IS TRIPPED INDICATES THAT NATURAL CIRCULATION HAS NOT BEEN ESTABLISHED, THEN THE FOLLOWING ALTERNATIVE MODES OF OPERATION WOULD BE CONSIDERED:

1) RESTART ONE REACTOR COOLANT PUMP

THIS WOULD RESULT IN CORE CONDITIONS ESSENTIALLY THE SAME AS THOSE EXISTING NOW AND DISCUSSED IN SECTION A. IT IS ANTICIPATED THAT SOME SHIFTING OF DEBRIS WITHIN THE CORE WOULD OCCUR, RESULTING IN A NUMBER OF FAIRLY RAPID CHANGES IN INCORE THERMOCOUPLE READINGS, SIMILAR TO THE CHANGES WHICH OCCURRED ON APRIL 6, 1979 WHEN THE A1 PUMP TRIPPED AND THE A2 PUMP WAS STARTED.

2) DECAY HEAT SYSTEM (OR MODIFIED DECAY HEAT SYSTEM)

THE COOLING FLOW RATE AVAILABLE FROM THIS SYSTEM IS SIMILAR IN MAGNITUDE TO THAT AVAILABLE WHEN NATURAL CIRCULATION IS SUCCESSFULLY ESTABLISHED, THEREFORE THE ANALYSES OF SUBSECTION B APPLY. THE MINIMUM REQUIRED FLOWRATE, PROVIDED IN REFERENCE (C.1) INCLUDES A FACTOR OF 5.8 ON THE FLOW CALCULATED AS THAT REQUIRED TO AVOID BULK BOILING WITHIN THE CORE. LOCALIZED BOILING IS EXPECTED TO OCCUR, AND THIS COULD CAUSE SOME OF THE INCORE THERMOCOUPLES, SUCH AS THOSE LOCATED IN

CORE LOCATIONS H8 AND H5, TO INDICATE TEMPERATURES SOMEWHAT HIGHER THAN THE AVERAGE AND POTENTIALLY HIGHER THAN THE SATURATION TEMPERATURE (ASSUMING THAT THESE THERMOCOUPLES ARE IN CONTACT WITH AGGLOMERATIONS OF FUEL FRAGMENTS). THE FACTOR OF 5.8 ACCOUNTS FOR A 27% CORE BYPASS FLOW RATE WHICH, IN TURN, REFLECTS AN EXTREMELY HIGH CORE FLOW RESISTANCE RESULTING FROM THE POSTUALTED CORE BLOCKAGE.

3) <u>HIGH PRESSURE INJECTION SYSTEM</u>

THE MINIMUM FLOW RATE SPECIFIED FOR THIS MODE OF OPERATION IS BASED UPON THE SAME EVALUATION AS THAT FOR THE DECAY HEAT SYSTEM, AND REPRESENTS THAT FLOW RATE NECESSARY TO AVOID BULK BOILING. THE AVOIDANCE OF BULK BOILING, ALTHOUGH NOT CONSIDERED TO BE ABSOLUTELY NECESSARY, IS DESIRABLE TO MINIMIZE HOT SPOT TEMPERATURES IN CORE LOCATIONS SUBJECTED TO SEVERE BLOCKAGE. THE FLOW REQUIREMENT SPECIFIED IS CONSERVATIVELY BASED UPON A DECAY HEAT VALUE OF 4 MWT. FOR LOWER POWER LEVELS THE MINIMUM REQUIRED FLOW DECREASES IN DIRECT PROPORTION TO THE DECAY HEAT LEVEL.

<u>REFERENCE C.1</u>

A. B. JACKSON TO C. C. ENGLAND, "REQUIRED FLOW FOR CORE COOLING," APRIL 10, 1979.

THE BABCOCK & WILCOX COMPANY
POWER GENERATION GROUP PRELIMINARY

②

BDS 663-5

To
G. A. MEYER

om
L. L. LOSH/J. F. BURROW

Cust.
TMI-2

File No.
or Ref.

Subj.
CRITERIA DURING ESTABLISHMENT OF NATURAL CIRCULATION

Date
5:40 P.M.
APRIL 10, 1979

This letter to cover one customer and one subject only.

THE TRANSITION FROM FORCED CORE COOLING USING ONE REACTOR COOLANT PUMP TO NATURAL CIRCULATION WILL BE ACCOMPLISHED OVER A PERIOD OF TIME. HEATUP OF THE CORE COOLANT WILL GENERATE A DENSITY GRADIENT AXIALLY IN THE CORE WHICH WILL PRODUCE THE DRIVING FORCE FOR THE CIRCULATION FLOW. THE TIME REQUIRED TO ESTABLISH NATURAL CIRCULATION IS THEN DIRECTLY RELATED TO THE HEATUP RATE OF THE CORE WHICH GENERATES THE DRIVING FORCE. TO ESTIMATE THIS HEATUP TIME THE FOLLOWING ANALYSIS WAS PERFORMED: (1) IT WAS ASSUMED THAT THERE WAS NO CORE FLOW, (2) THE ENTIRE CORE COOLANT INVENTORY WAS RAISED FROM T_{IN} TO T_{SAT}, (3) NO NET STEAM QUALITY WAS GENERATED, (4) CALCULATIONS WERE BASED ON GROSS CORE AVERAGE CONDITIONS.

FIGURE 1 DEPICTS THE CORE DECAY HEAT GENERATION AS A FUNCTION OF TIME. THE RESULTS OF THIS ANALYSIS ARE SHOWN IN FIGURE 2-4 FOR VARIOUS ASSUMPTIONS ON T_{IN}, SYSTEM PRESSURE AND CORE POWER (TIME AFTER SHUTDOWN). THESE CURVES WERE USED TO ESTABLISH A MAXIMUM TIME REQUIRED TO ESTABLISH NATURAL CIRCULATION.

NATURAL CIRCULATION RESPONDS TO HIGHER FLOW RESISTANCE BY GOING FURTHER INTO TWO-PHASE FLOW UNTIL THE VAPORIZED DENSITY CHANGE IS SUFFICIENT TO OVERCOME SYSTEM RESISTANCE. WITH DEBRIS BLOCKING A GRID, FLOW WILL PASS LATERALLY AROUND THE BLOCKAGE AND ADEQUATELY IMMERSE THE FUEL BEFORE AND AFTER THE BLOCKAGE. IF THE BLOCKAGE DEBRIS ALSO CONTAINS FUEL, THE MAXIMUM TEMPERATURE AT THE BLOCKAGE CENTER WILL BE CONDUCTION LIMITED AND RELATIVELY INSENSITIVE TO THE

MODE OF SURFACE HEAT TRANSFER. THE FLOW PATTERNS IN NATURAL CIRCULATION ARE DIRECTLY RESPONSIVE TO RESTRICTIONS, AND MAY WELL SHOW LOCAL TWO-PHASE TRANSITIONS AND RELATIVELY HIGH VELOCITIES AT PARTIALLY-BLOCKED HOTTER REGIONS.

REDISTRIBUTION OF CORE DEBRIS IS ANTICIPATED WHEN THE PUMPED FLOW IS TERMINATED. CHANGES IN THERMOCOUPLE INCORE TEMPERATURE DISTRIBUTIONS SHOULD BE EXPECTED. NOT ALL THERMOCOUPLE READINGS CAN BE EXPLAINED BY HYDRAULIC PHENOMENA. SOME THERMOCOUPLES, PARTICULARLY NEAR THE CENTER, ARE CURRENTLY INDICATING LOCALIZED HEATING EFFECTS AND ARE NOT MEASURING BULK FLUID TEMPERATURES. SINCE THE PRIMARY CONCERNS DURING THE TRANSITION FROM PUMPED FLOW TO NATURAL CIRCULATION ARE ADEQUATE COVERAGE OF THE CORE AND BULK COOLANT TEMPERATURES BELOW SATURATION TEMPERATURE, IT IS REQUIRED THAT AT LEAST 10 (TEN) INCORE THERMOCOUPLES HAVE READINGS BELOW SATURATION TEMPERATURE FOR THE SYSTEM PRESSURE (FIGURE 5). ADDITIONALLY, NO TWO INCORE THERMOCOUPLES SHOULD EXCEED 800°F. ANTICIPATED CORE TRANSIENTS ARE VERY SLOW. FIGURE 4 SHOWS THAT IT WILL REQUIRE AT LEAST 45 MINUTES TO ONE HOUR TO RAISE THE WATER TEMPERATURE IN THE CORE FROM 200°F TO SATURATION TEMPERATURE FOR DATES BETWEEN 4/10/79 AND 4/17/79. ADDITIONALLY, THE CORE ADIABATIC HEATUP FROM 200°F TO 1000°F EXCEEDS ONE HOUR FOR DATES AFTER 4/12/79. HENCE, NO CORE COOLING PROBLEMS EXIST FOR AT LEAST THE FIRST HOUR OF TRANSITION TO NATURAL CIRCULATION.

THE "HOT-LEG" THERMOCOUPLE SHOULD SHOW A TEMPERATURE INCREASE AS NATURAL CIRCULATION IS ESTABLISHED. TEMPERATURE INCREASES IN THE T_H RESISTANCE TEMPERATURE DETECTOR (RTD) SHOULD BE OBSERVED WITHIN THE BOUNDS OF THE TIME TO SATURATE THE CORE AS SHOWN IN FIGURE 4 (I.E., 45 MINUTES ON 4/10/79; 2.5 HOURS ON 6/6/79).

ADDITIONALLY, THE MAXIMUM TEMPERATURE OF THE HOT LEG RTD SHOULD NOT EXCEED 250°F FOR PRESSURES ABOVE 500 PSIA AND 180°F FOR PRESSURES NEAR ATMOSPHERIC. THE CORE ΔT SHOULD BE LIMITED TO 150°F.

SUMMARY:

REQUIREMENTS FOR TRANSITION TO NATURAL CIRCULATION:

1) INCORE THERMOCOUPLES -- AT LEAST 10 (TEN) THERMOCOUPLES MUST READ BELOW THE SATURATION TEMPERATURE CORRESPONDING TO SYSTEM PRESSURE (FIGURE 5). NO TWO INCORE THERMOCOUPLES MAY EXCEED 800°F.

2) HOT LEG RTD'S -- THE HOT LEG RTD'S MUST INDICATE A TEMPERATURE RISE WITHIN THE TIME REQUIRED TO SATURATE THE CORE (FIGURE 4) AND THE MAXIMUM TEMPERATURE SHOULD NOT EXCEED 250°F FOR SYSTEM PRESSURES ABOVE 500 PSIA AND 180°F FOR PRESSURES NEAR ATMOSPHERIC. THE CORE ΔT SHOULD BE LIMITED TO 150°F.

LLL:JFB:nw

cc: J. S. TULENKO
FUEL ENG. UNIT MANAGERS
CORE HOT SPOT TASK FORCE

ATTACHMENTS (5 FIGURES)

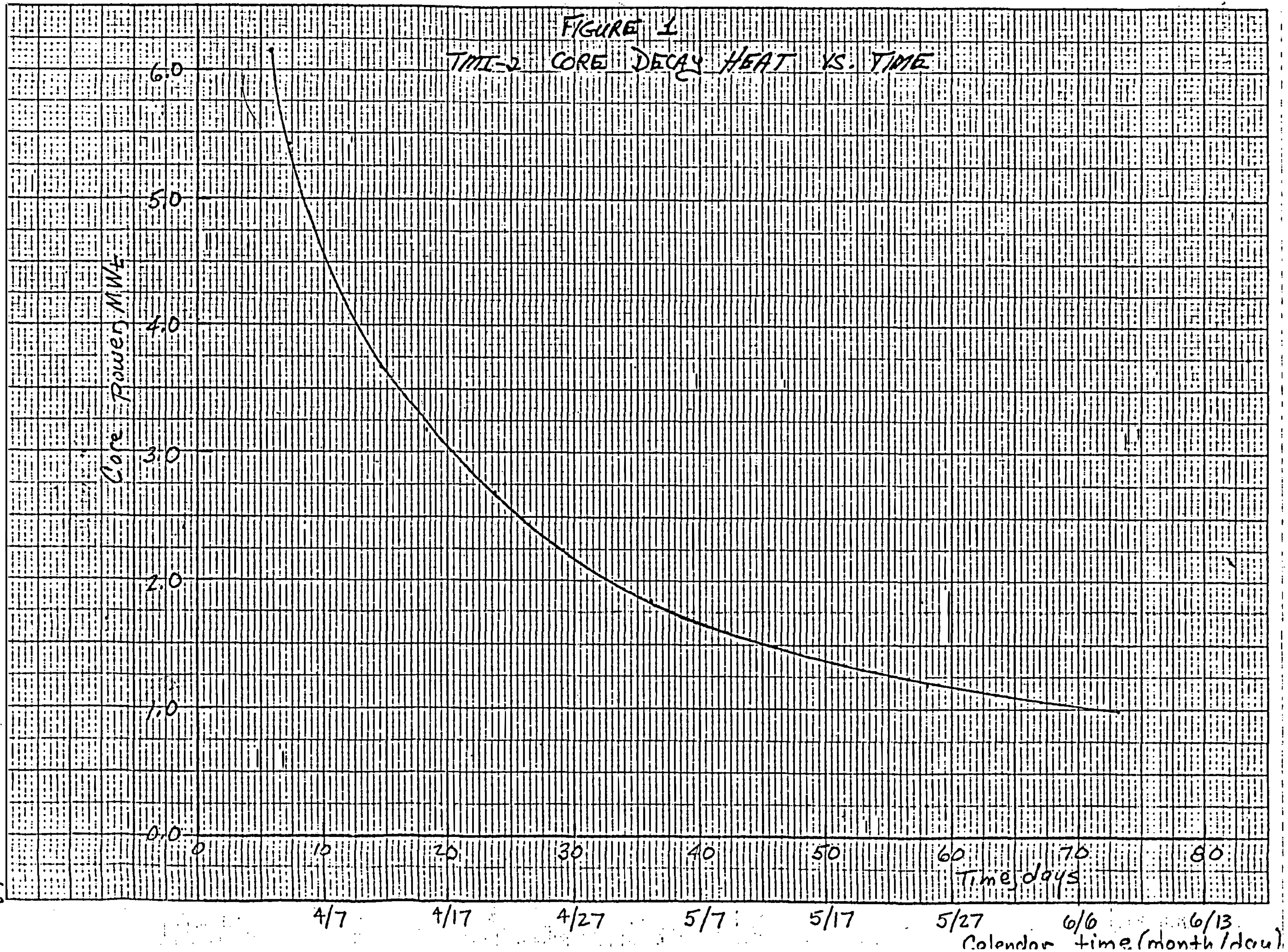


FIGURE 1

TMI-2 CORE DECAY HEAT VS. TIME

8.2.8

FIGURE 2

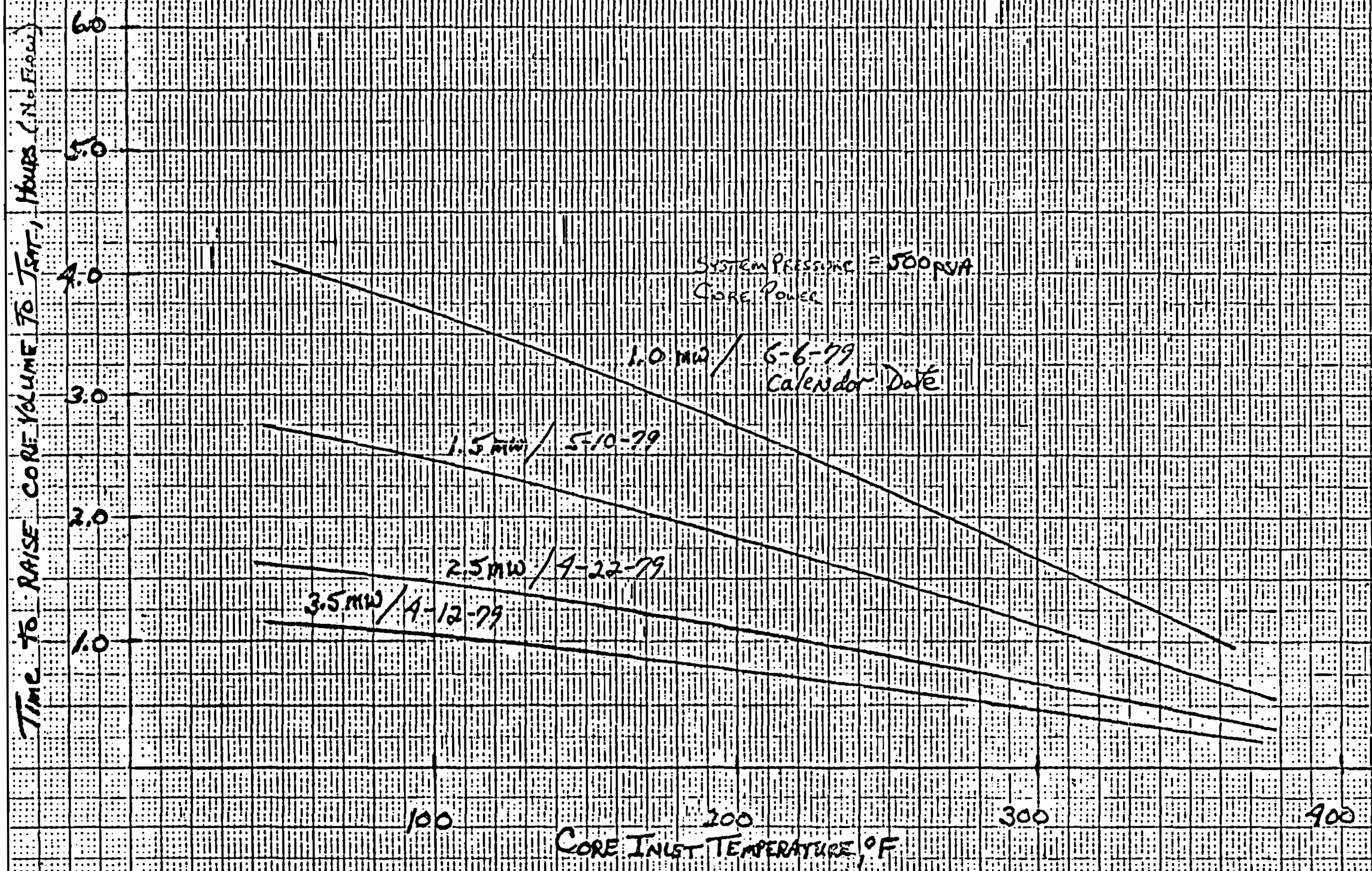

TIME TO RAISE CORE VOLUME TO TSAT-1 HOURS (NO FLOW)
6.0
5.0
4.0
3.0
2.0
1.0
100
200
300
400
CORE INLET TEMPERATURE, °F
SYSTEM PRESSURE = 500 PSIA
CORE POWER
1.0 MW / 6-6-79
Calendor Date
1.5 MW / 5-10-79
2.5 MW / 4-22-79
3.5 MW / 4-12-79

FIGURE 3

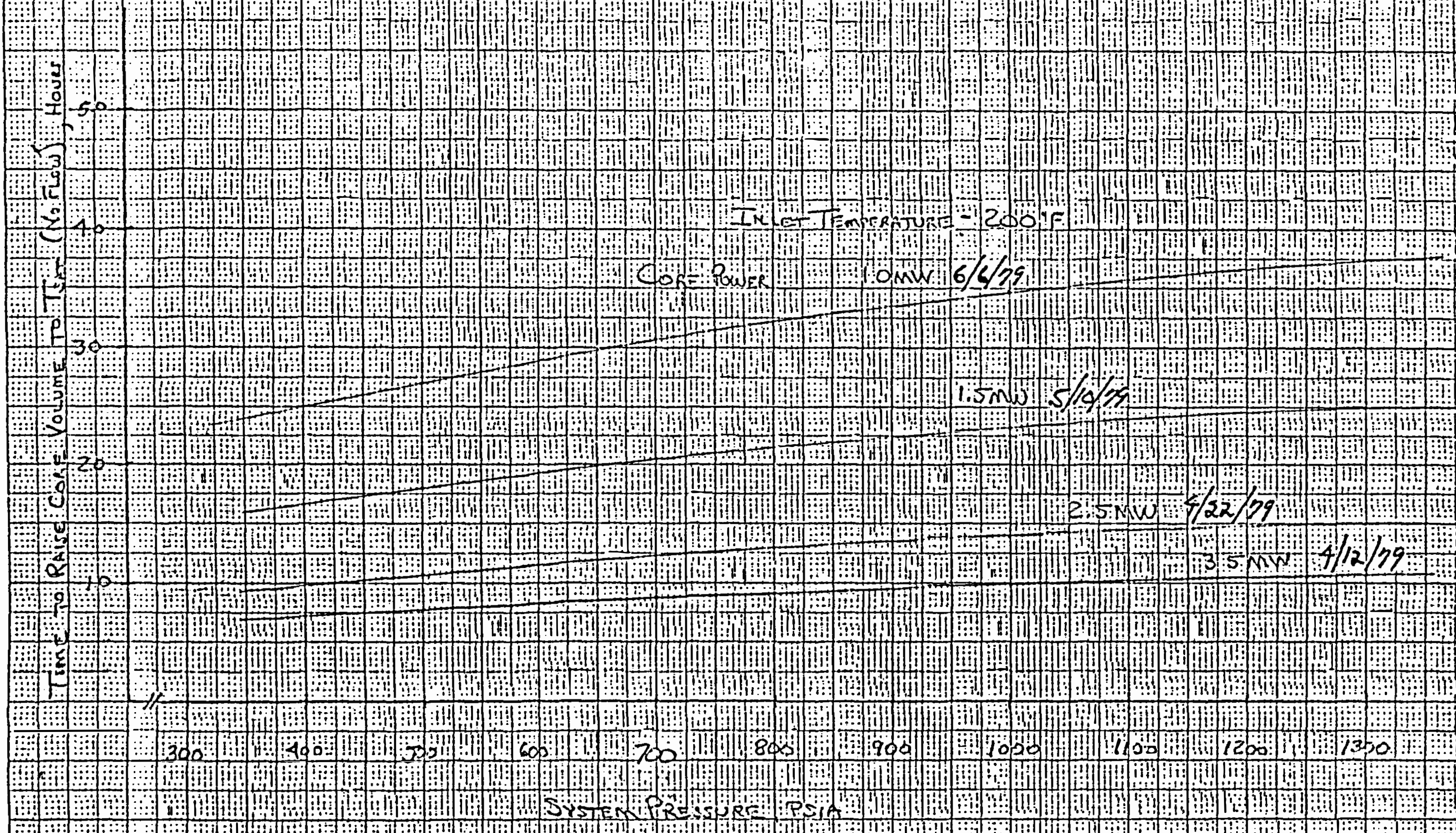

Time to Raise Core Volume to Tsat (No Flow), Hours
50
40
30
20
10
Inlet Temperature - 200°F
Core Power
1.0MW 6/6/79
1.5MW 5/10/79
2.5MW 4/22/79
3.5MW 4/12/79
300
400
500
600
700
800
900
1000
1100
1200
1300
System Pressure, PSIA

FIGURE 4

TIME TO RAISE CORE WATER VOLUME TO T_{SAT} (NO FLOW)

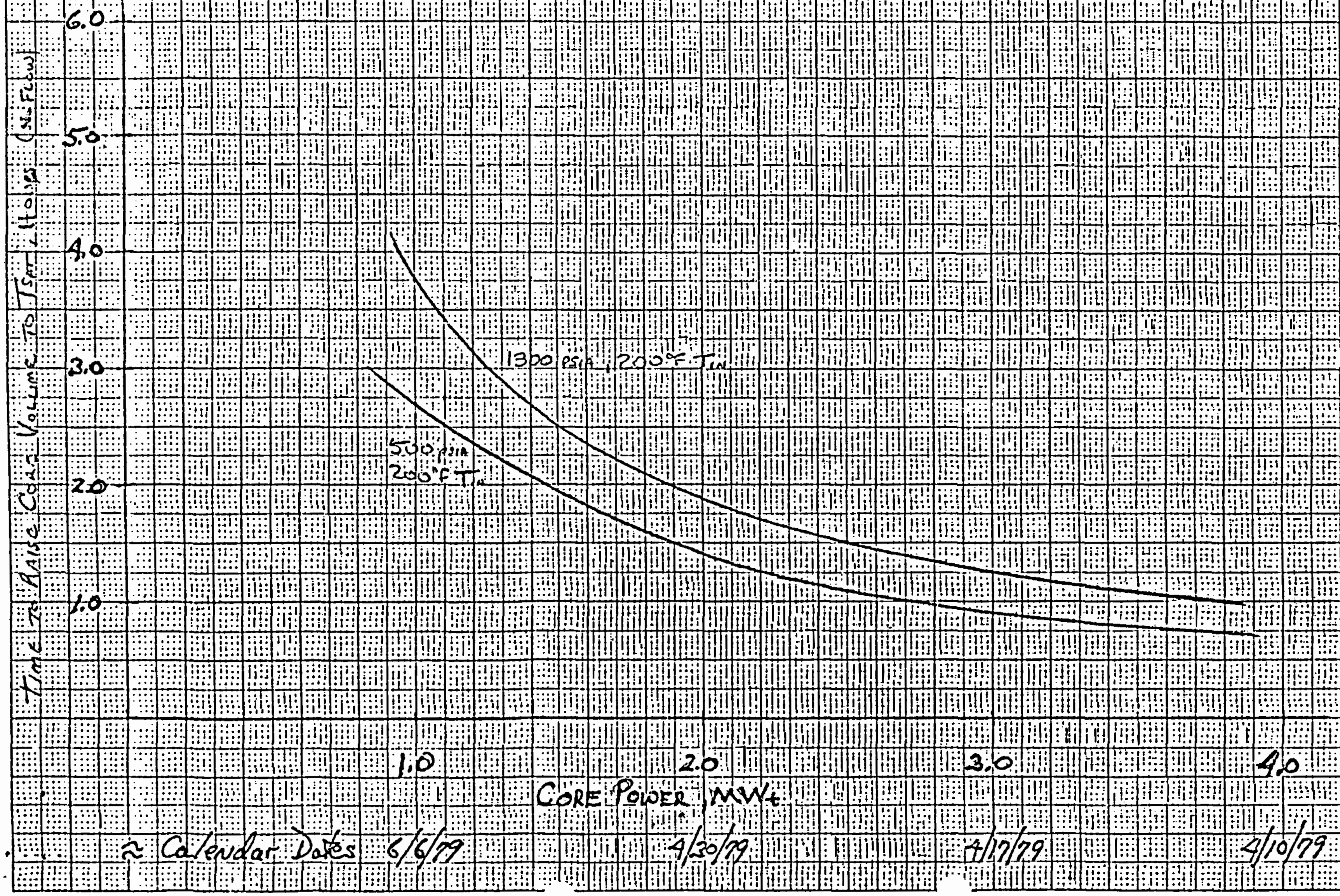
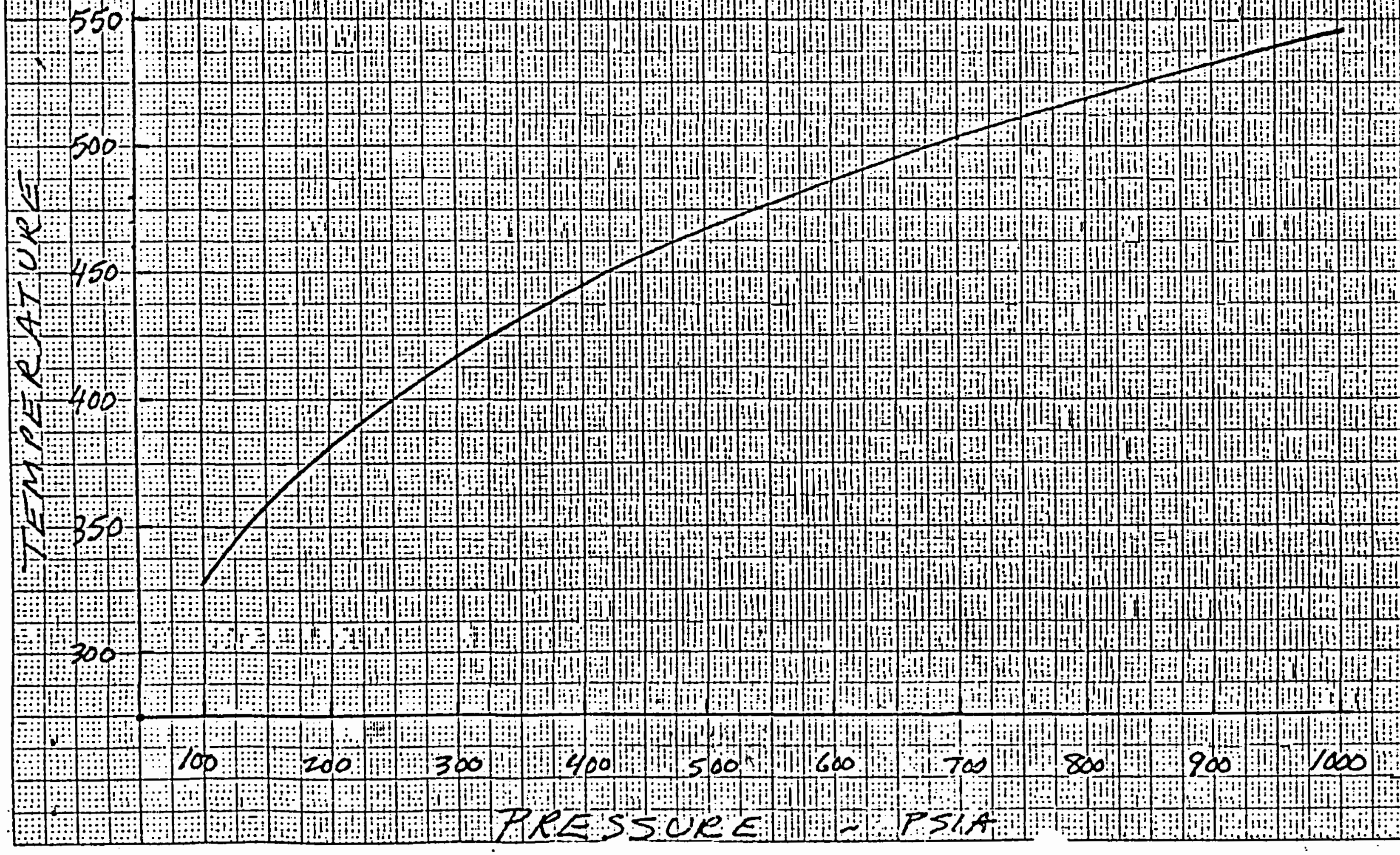


Figure 5

SATURATION TEMPERATURE VS PRESSURE



③

THE BABCOCK & WILCOX COMPANY
POWER GENERATION GROUP

To E. OELKERS, SYSTEM RELIABILITY & CONTROLS UNIT

From Thomas L Wilson 1:15 PM
T. L. WILSON, SYSTEM RELIABILITY & CONTROLS UNIT

Cust. TMI-2

Subj. RELIABILITY AND UNCERTAINTY OF THERMOCOUPLES FOLLOWING LOSS OF FEEDWATER TRANSIENT

cc: J. R. Gloudemans
G. A. Meyer
P. E. Mamola
J. A. Weimer
H. D. Warren
C. T. Rombough
R. V. DeMars

BDS 663-5

File No. or Ref.

Date APRIL 5, 1979

Acknowledgement:

I have gathered the following information through conversations with several individuals having experience with thermocouples that have been subjected to extremely high temperatures. I am indebted to P. E. Mamola, R. H. Stoudt, and Tom Kollie of ORNL for their help and cooperation.

Description of the Problem:

The incore thermocouple design at the TMI-2 core is a grounded junction Chromel-Alumel detector. The wire diameters are approximately 10 mils, the sheath is 62 mils OD Inconel tubing, and the insulation material is Al_2O_3. I have gathered information on reliability and uncertainty of this type of thermocouple after being subjected to extremely high temperatures (> 2000°F). The problem of gross failure (open circuit, sheath failure, new junction or other failure) would be indicated by no reading or extremely low, erratic reading. No detectors are giving indication of gross failure. Given the survival at present conditions, the prospect for continued operation is excellent. The primary questions are the following:

1. Are the readings accurate?
2. Are the errors in the readings consistent with the hypothesized scenario of the transient?

Decalibration Phenomenon:

A phenomenon observed by Dr. Kollie in thermocouples having experienced extremely high temperatures is a deteriorated state in which the thermocouple gives a stable but inaccurate reading. The effect is called decalibration.

The decalibration error is random but does follow some trends. The primary dependencies are the following:

1. Sheath diameter	The larger the detector, the less susceptible to decalibration.
2. Sheath material	Inconel is better than stainless. The decalibration

	error for Inconel is usually negative. Measured temperature is lower than actual.
3. Temperature	Decalibration error increases with temperature. (10°C roughly doubles the reaction rate of the mechanism.)
4. Time of exposure	Decalibration error is roughly linear with exposure time.
5. Temperature profile	Decalibration occurs along the length of the thermocouple. Hence, the error depends on the profile of the elevated temperature and on the profile of a subsequent measurement with the thermocouple.

Other dependencies which are not known specifically but are expected to be small are insulation material, length of detector, and wire diameter.

Experience with smaller diameter detectors (20 mils OD) but similar in other characteristics (grounded junction Chromel/Alumel thermocouple with Inconel sheath) show essentially zero probability of survival at 1100°C for 10 to 100 hrs. without a measurable decalibration. The decalibration error tends to be negative (reading lower than actual) for this test; however, one sample gave a positive error. The magnitude for worst case error (remembered) was -50°C. Larger decalibration errors would be expected for higher temperatures.

The primary mechanism for decalibration is migration of constituents primarily chromium in the detector through vapor phase transfer. The chromel lead loses chromium and the alumel lead gains chromium. Belfab tests indicating little decalibration for 4 hour period at 7000°F are not necessarily conclusive since the temperature gradient of the high temperature state does not simulate the transient environment nor did the test condition simulate the present environment.

Conclusions:

If it can be determined that the thermocouple is reading accurately now (for example, if it agrees with outlet RTD's), this implies that its readings are believable throughout the transient.

If thre is reason to believe a thermocouple is decalibrated, there is a high probability that actual temperature is greater than its indication. Hence, the cluster of high readings should not be disregarded.

Since a high percentage of detectors survived the transient, the maximum temperature did not attain a level for widespread failure.

Recommendations:

1. Perform testing simulating transient conditions to determine the temperature threshold for gross failure in TMI-2 design detectors.
2. Perform testing to quantify magnitude and direction of decalibration errors for a range of temperatures simulating reactor environment.

TLW:ae

THE BABCOCK & WILCOX COMPANY
POWER GENERATION GROUP

(4)

To
Engineering Operations Manager

From
J. T. Willse (Barney for JTW)

THE-79-179 BDS 663.5

Cust.

File No.
or Ref.

Subj.
Action Item 143

Date
April 6, 1979

This letter to cover one customer and one subject only.

Reference: C. T. Rombough to Engineering Operations Manager, Action Item 143, 4/5/79.

The purpose of this memo is to elaborate on conclusions no. 3 and 4 in the referenced memo. The increase in selected thermocouple readings is no cause for concern. The temperature is still 200°F below saturation temperature. The increased readings are caused by two factors. The first and smallest effect is a 4° increase in the core inlet temperature. The primary cause for the change in thermocouple readings is the change in the flow distribution caused by the shifting debris in the core. This was vividly demonstrated when the A1 pump tripped and the A2 pump was started. I would anticipate that some thermocouples would continue to change for several days until the debris redistributes into a stable configuration reflecting the change of coolant pumps.

Conclusion number 4 is inaccurate since resistance readings will show wide variations from thermocouple to thermocouple and also the readings will depend on whether the chromel or alumel wire resistance is being measured. However, if needed, we can state the following:

A test can be performed to determine whether a gross failure of a thermocouple has occurred. For a good thermocouple the resistance between one T/C lead and ground should be approximately 750 ohms while the other lead to ground should measure approximately 300 ohms to ground.

JTW:mp

cc: F. E. Unit Managers
J. S. Tulenko
Shift Technical Leader

THE BABCOCK & WILCOX COMPANY
POWER·GENERATION GROUP

To
Engineering Operations Manager

From
C.T. Rombough, Fuel Management & Development

BDS 663-5

Cust.
GPU

File No.
or Ref.
ONCP-79-038

Subj.
B&W's View of Increasing
Incore Thermocouple Readings (Instruction 143)

Date 3:30 p.m.
April 5, 1979

This letter to cover one customer and one subject only.

As requested in Action Item 143, the increasing thermocouple data has been reviewed by both NPGD and LRC personnel. These personnel have included J.B. Andrews, G.A. Meyer, J.A. Weimer, J.T. Mayer, T.L. Wilson, E.T. Chulick, P.E. Mamola, R.A. Copeland, J.W. Ewing, H.D. Warren, and J.G. Brown.

A summary of the pertinent data and their conclusions is attached for D.H. Roy's response to Bill Lowe.

CTR/dlw

CT Rombough

Incore thermocouple data from TMI-2 have been evaluated. As shown below, six of the thirty thermocouples for which we have data have shown a temperature rise of 7 - 33°F over the past 5 days (117 hours) or 1.4 - 6.8°F per day.

Location	T/C at 0845,3/31	T/C at 0542,4/5	Net Increase (°F)	Increase Per Day (°F)
13C	290	297	7	1.4
13F	298	307	9	1.8
13H	310	320	10	2.1
11G	427	445	18	3.7
12F	303	327	24	4.9
11L	302	335	33	6.8

Temperature vs time for locations 13F, 12F, and 11L are shown in the attached figure.

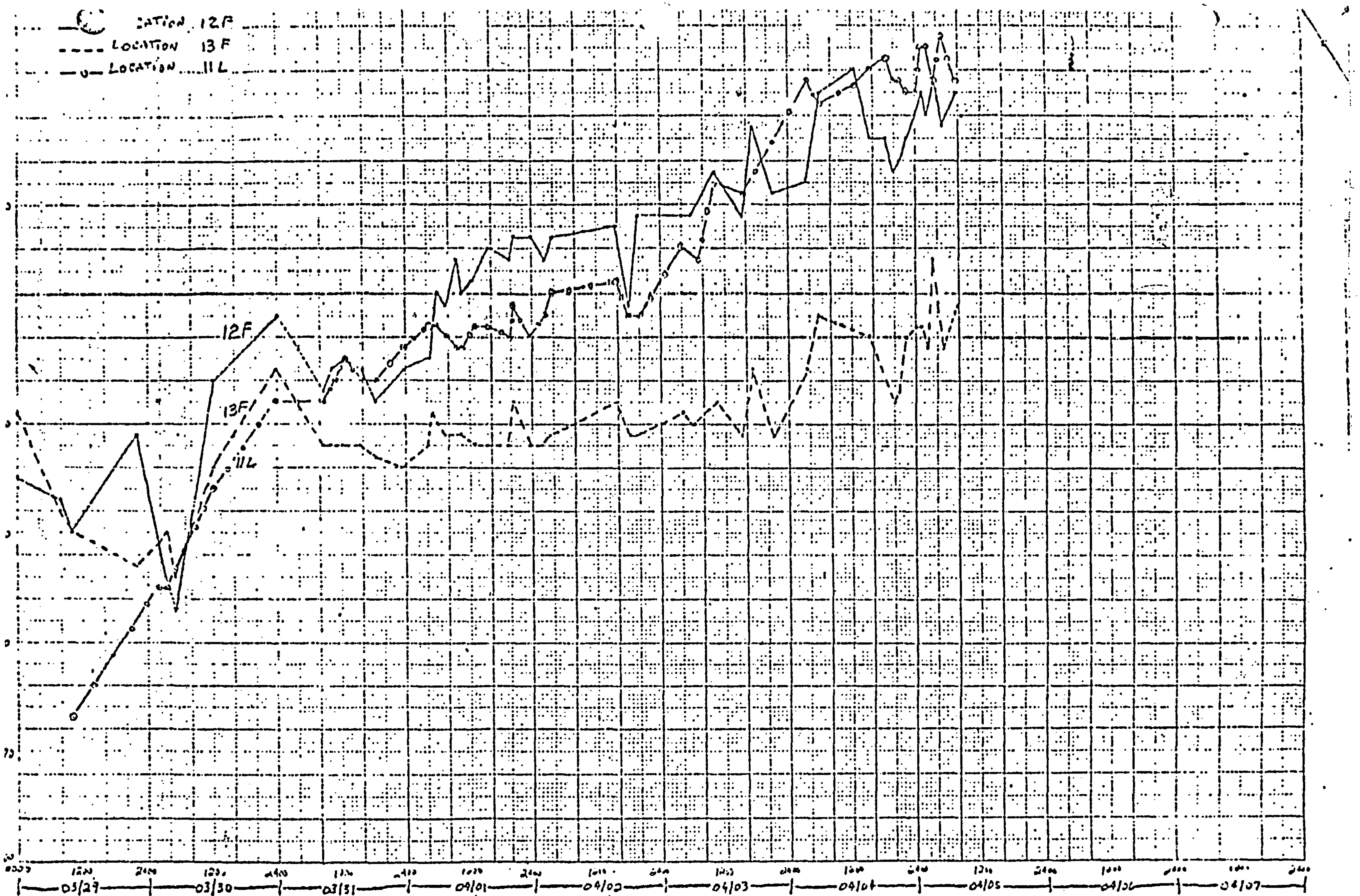
The following conclusions have been reached based on this data.

1. There is no mechanism which has been postulated that would cause decalibration to the extent shown by the data. Therefore, it is concluded that the thermocouple readings are accurate and that true temperatures are being monitored to ±5°.

2. There is nothing in the fuel, structural materials, or fission products which would cause a chemical reaction that would result in the observed rate of temperature rise.

3. As indicated in the attached map of temperature changes for 30 selected locations, the increasing readings are located preferentially in the core (in the center of the right half). This leads to the most probable explanation that a very gradual flow redistribution is occurring; either increased flow blockage in this region or decreased flow blockage in another region causing temperature increases in this region.

4. If Met Ed questions the thermocouple readings, a simple test can be performed. The resistance between the T/C lead and ground should be approximately 750 ohms.



LOCATION 12F
LOCATION 13F
LOCATION 11L
12F
13F
11L
70
0000
1200
2400
03/29
03/30
03/31
04/01
04/02
04/03
04/04
04/05
04/06
04/07

CHANGE IN T/C READING FROM 0845, 3/31 TO 0512, 4/5

280?
289

A1

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
A															
B							+2								
C									-1				+7		
D										-53					
E							-47		-87		-6				
F							-20	-18				+24	+9		
G		+2			-8	-6			-45		+18		+1		
H	-2				0			-302	-1				+10		
K											-59				
L			+4								+33				
M									-20	-3				+3	
N								-30							
O															
P						0									

THE BABCOCK & WILCOX COMPANY
POWER GENERATION GROUP

(5)

To
G. A. MEYER, MANAGER, T-H ENGINEERING UNIT

THE-79-171

From
J. A. WEIMER, T-H ENGINEERING UNIT, EXT. 3236

BDS 663.5

Cust.

File No.
or Ref.

Subj.
RESPONSE TO THERMOCOUPLE REQUEST

Date
APRIL 5, 1979

This letter to cover one customer and one subject only.

Past experience with thermocouples show two basic things. First, they track fairly steady when core condition are not changing (ie. 100% steady state conditions). Secondly, they respond to local changes relatively accurate. A one to two degree change in thermocouple reading has occurred many times without any obvious core change (or local change) conditions. Greater deltas than 2^0F usually indicate another measurable local or core change.

The two mechanisms for thermocouple changes are obviously power distribution or flow distribution. A few random local changes are usually due to local power changes. A large area group change indicate flow redistribution.

Assuming local flow blockages (in the center of the core a decrease in thermocouple reading would occur if and when the blockage was decreased and if the "hot" conditions were due to local heat sources (ie. fuel pellets) the temperature would decrease as the source burned out. Furthermore, as blockage decreases in one region of the core the outlet flow distribution would tend to flatten causing temperature increases in other portions of the core.

JAW/sgh

cc: J. S. Tulenko
FE Unit Managers
J. F. Burrow

DISTRIBUTION

Action Item: 143

ROM
C.T. ROMBOUGH (X 3748)

FILE NO. OR REF.

UBJ. REASON FOR INCREASING I/C THERMOCOUPLE READINGS

DATE 4/5/79 10:30 AM

DISTRIBUTION

J.T. MAYER G.A. Meyer
T.L. WILSON
E.T. CHULICK
R.N. KUBIK

Don Roy has requested a detailed response from B+W as to the reason for an increase in thermocouple readings seen in certain locations at TMI-2. Attached is the data that has been gathered so far regarding the problem. Please review this data as quickly as possible and provide a response to me by 1 PM today. I have additional data at my desk if you feel you need it. Thank you for your help.

CT Rombough

POWER GENERATION GROUP

To [illegible] W. PRYOR

D. H. ROY

BDS 663-

Cust.

File No.
or Ref.

Subj. REQUEST FOR B&W'S VIEW OF INCREASING
INCORE THERMOCOUPLE READINGS

Date APRIL 5, 1979

This letter to cover one customer and one subject only.

NOTES TO ANSWER TO BILL LOWE'S REQUEST TO D. H. ROY AT 0735 ON APRIL 4 REGARDING B&W'S IDEAS AND OUR CONCERNS REGARDING SOME THERMOCOUPLES SLOWLY INCREASING IN TEMPERATURE

The responses provided is inadequate for transmission to GPU. We should take the following steps:

(1) Report the increases as we see them in the data - assembly location, temperature change per unit time over some time interval.

(2) Provide the uncertainty value which we would assign for these locations.

(3) Have our radiochemists get in touch with expert fuel material people to determine if there are any break away chemical reactions or any other explanation which might be provided to explain the temperature behavior which is observed.

It is not necessary to state that we will continue to monitor thermocouple changes and report them promptly as GPU is also tracking them routinely.

D H Roy

D. H. Roy

0635

DHR:dmd

Telecon from Bill Lowe to D. H. Roy at 0735 April 4, 1979

Telecon from Bill Lowe at 0735 informed us that over the past 36 hours some trace in thermocouple readings have been observed, it varies from assembly to assembly with some going up by as much as 9°F. The absolute temperature still is o.k. but they do not like the trend. It could be associated with changes in coolant temperature and or pressurizer level. He requested that we consider various possibilities for this behavior including a break away chemical reaction. We told him that our data for thermocouple readings at the 0523 hour measurement was just coming in, we will take action to determine whether there is a chemical reaction which can account for this and try to corelate with changes in the state of the reactor.

Signed D. H. Roy

FROM

FILE NO. OR REF.

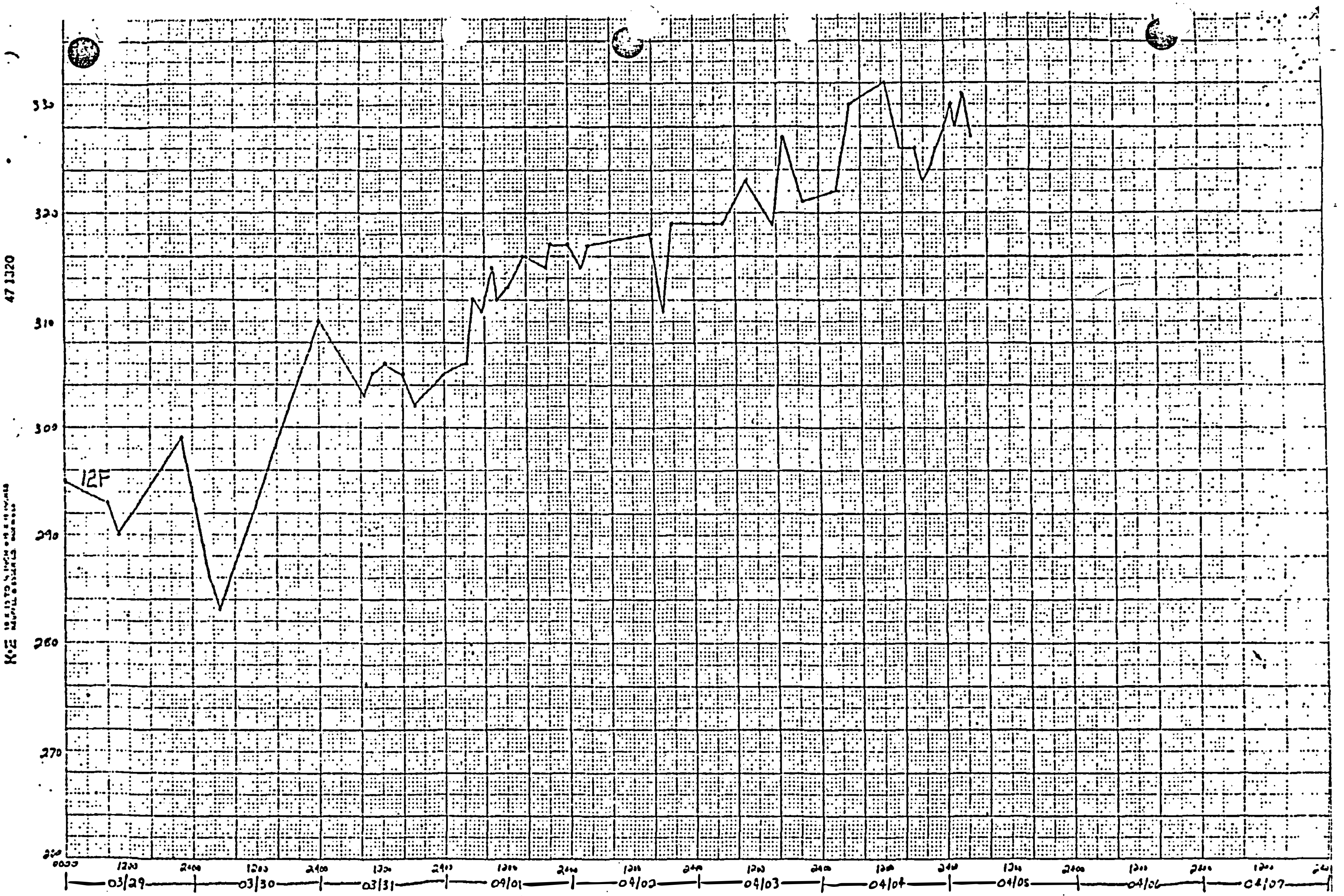
SUBJ.

DATE

The following detectors have shown a steady increase in temperature from 0900, 3/31 to 0500, 4/5 (116 hou

LOCATION	T/C at 0900, 3/31	T/C at 0500, 4/5	Net increase (°F)
12 F	303	327	24
11 G	427	445	18
13 H	310	320	10
11 L	302	335	33
13 F	298	307	9
13 C	290	297	7

Attached is a map of T/C changes for 30 locations between 0900, 3/31 + 0500, 4/5. Also attached is a graph of the behavior of location 12 F.


330
320
310
300
290
280
270
260
12F
47 1320
K+E
0000
1200
2400
1200
2400
1200
2400
1200
2400
1200
2400
1200
2400
1200
2400
1200
2400
1200
2400
1200
2400
03/29
03/30
03/31
04/01
04/02
04/03
04/04
04/05
04/06
04/07

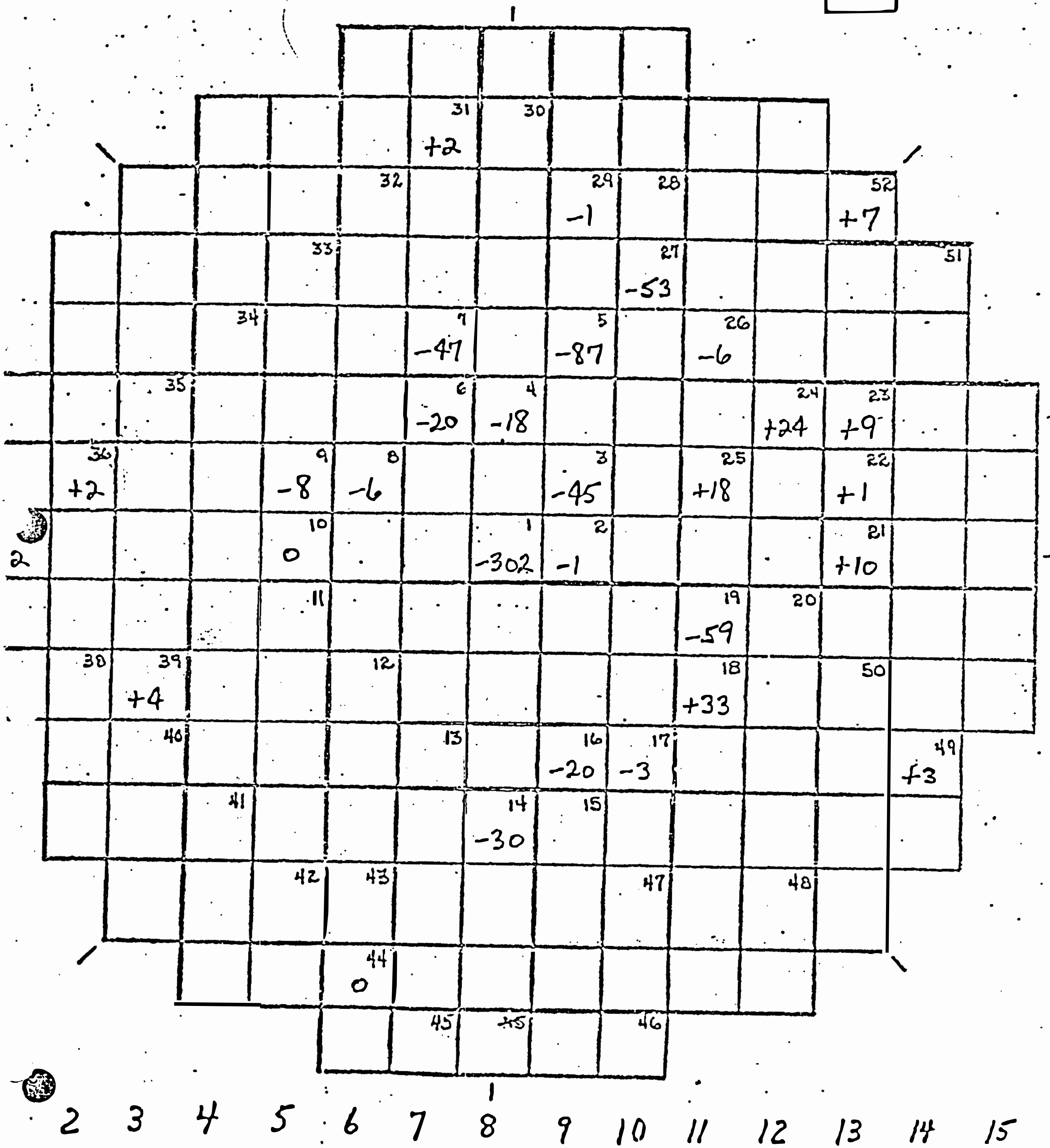
RC TEMP ____________

RC PRESS ____________

Change in T/C reading from 0900, 3/31 to 0500, 4/5

SPND STRING NUMBERS AND LOCATIONS – 177 FA CORE

☐ K STRING NO.



IER

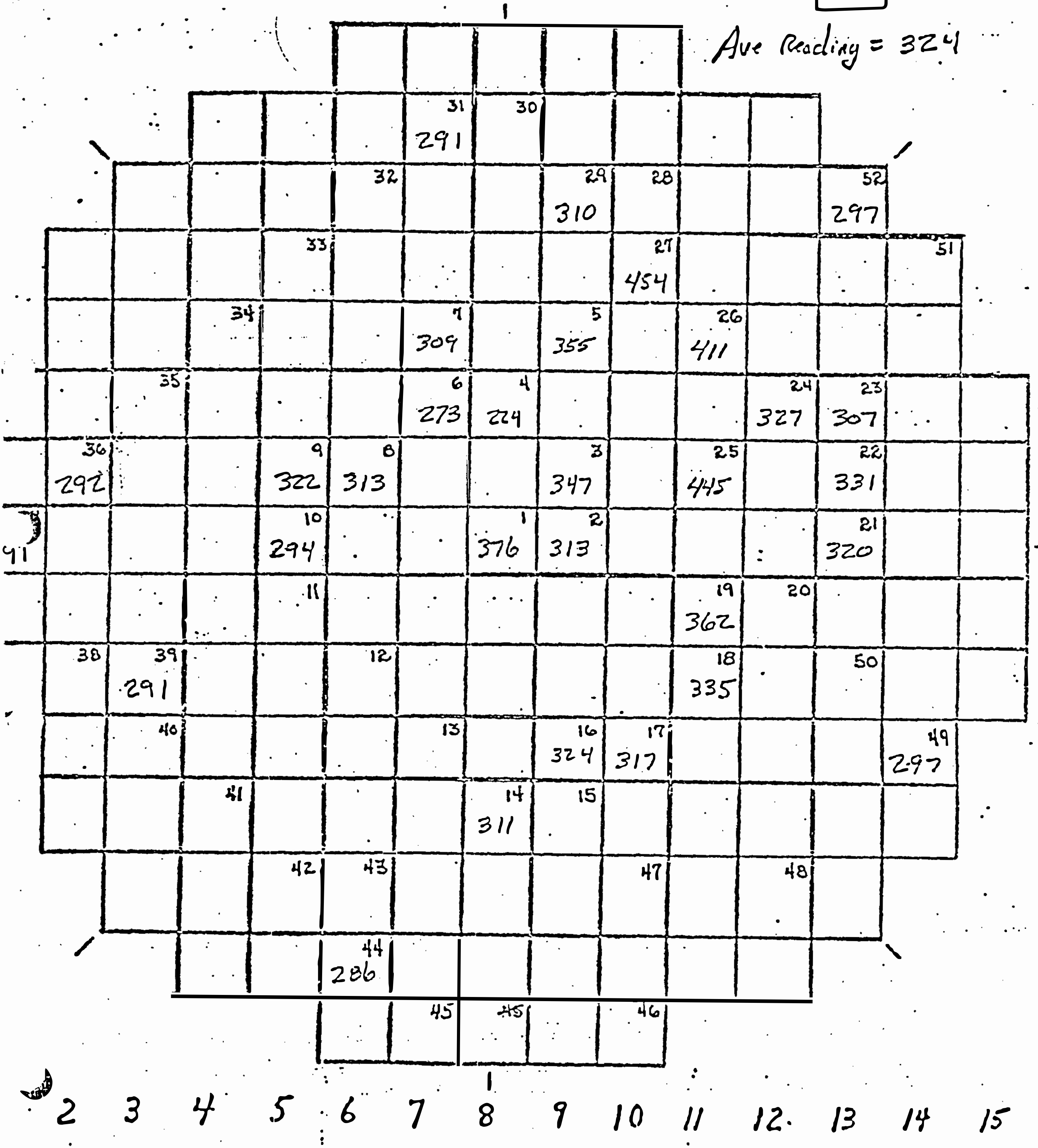
THE BABCOCK & WILCOX CO.

RC Temp Tin = 294
RC Press A = 990
B = 1031
[illegible]

SPND STRING NUMBERS AND LOCATIONS – 177 FA CORE

☐ X STRING NO.

Ave Reading = 324



ER

T

JOB NO.

Evaluate fuel pin contact with Incore thermal couples

The evaluation was performed assuming all the pellets from one grid span are evenly distributed across the upstream grid. Average temperature of the UO_2 mass is ~1500°F. Enough heat is generated to produce boiling in the annulus between the instrument string and the guide tube and possibly some superheat. Heat would be transferred to the instrument string which could be transferred axially by conduction to the thermal couple. This would produce a T-C reading higher than the surrounding coolant. This indicates that it is possible that a conglomerate of pellets with the proper size and location could produce T-C readings in the range of temperatures which are being recorded (100-600°F above coolant temperature).

C. D. Morgan
M. Montgomery
G. A. Meyer

THE BABCOCK & WILCOX COMPANY
POWER GENERATION GROUP

PRELIMINARY

⑥

BDS 66

To
C. C. ENGLAND

From
A. B. JACKSON

Cust.
TMI-2

File No.
or Ref.

Subj.
REQUIRED FLOW FOR CORE COOLING

Date
12:40 A.M.
APRIL 10, 1979

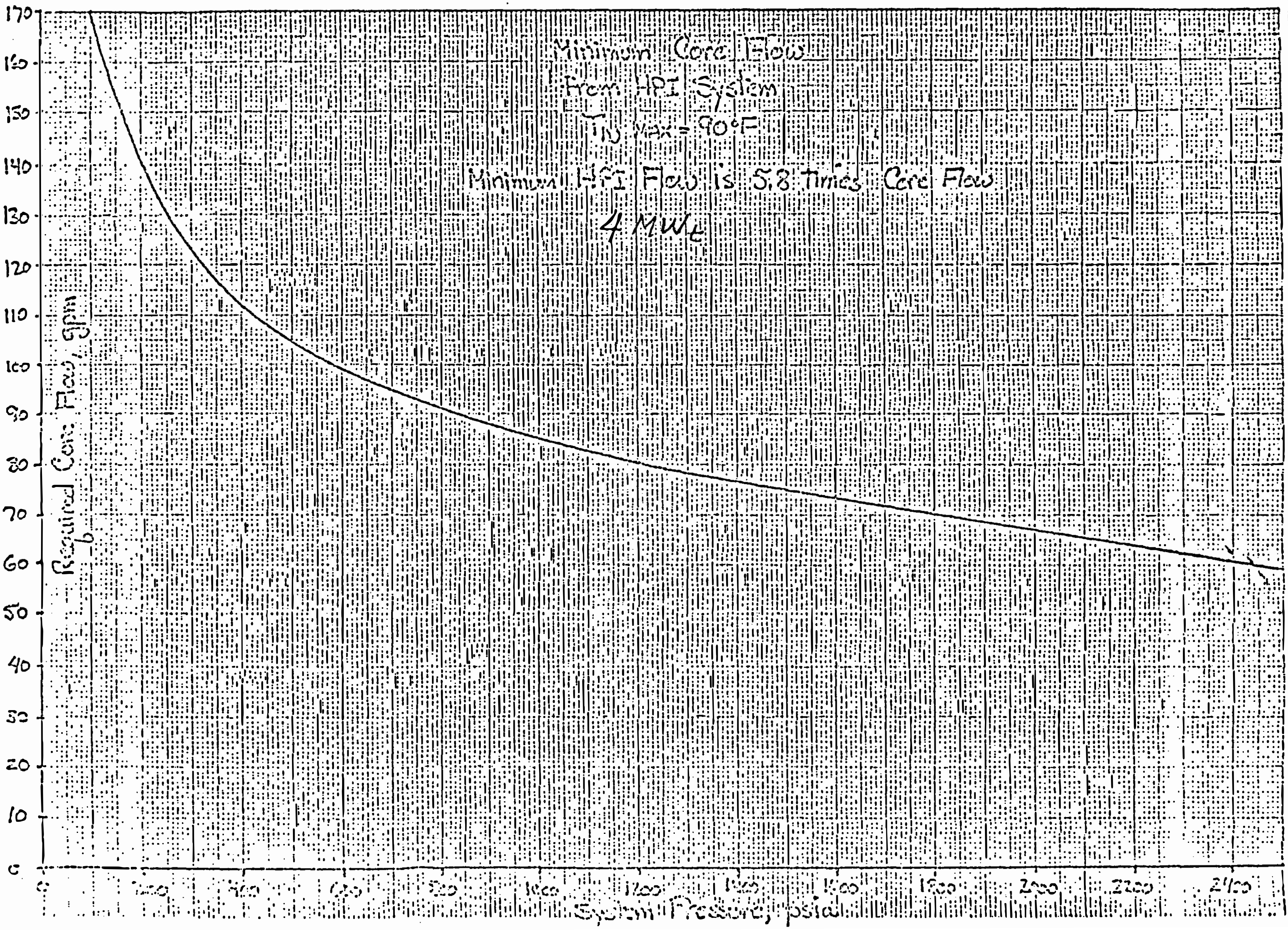
This letter to cover one customer and one subject only.

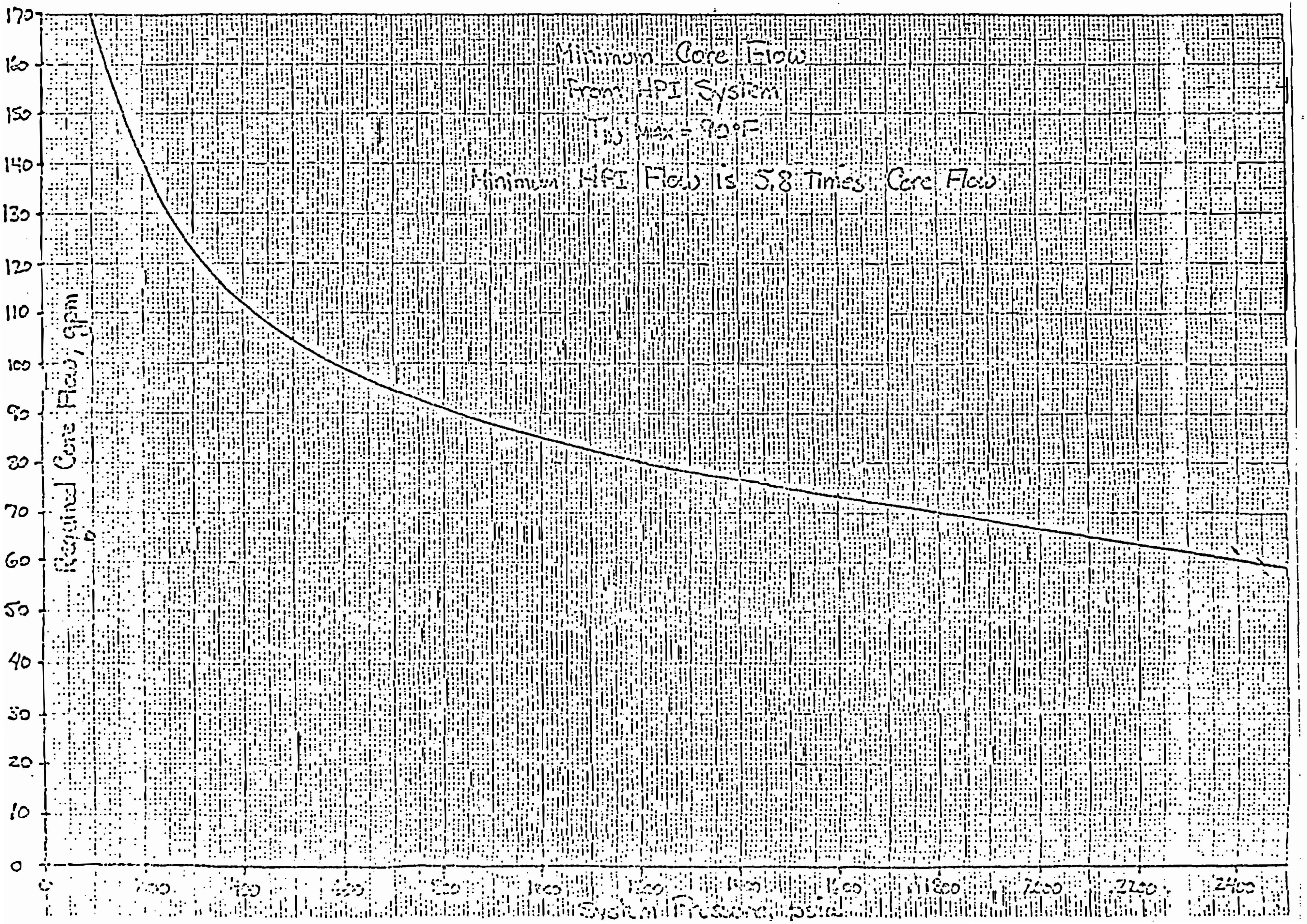
REFERENCES: (1) MEMO, C. C. ENGLAND TO R. B. DAVIS, "ALTERNATE DECAY HEAT SYSTEM REQUIREMENTS," 7:22 A.M., APRIL 9, 1979.
(2) MEMO, G. A. MEYER TO J. D. CARLTON, "MINIMUM CORE FLOW - LONG TERM COOLING," 1200, APRIL 4, 1979
(3) MEMO, A. ~~B~~. JACKSON TO C. C. ENGLAND, "CORE FLOW REQUIRED FROM HPI SYSTEM," APRIL 8, 1979

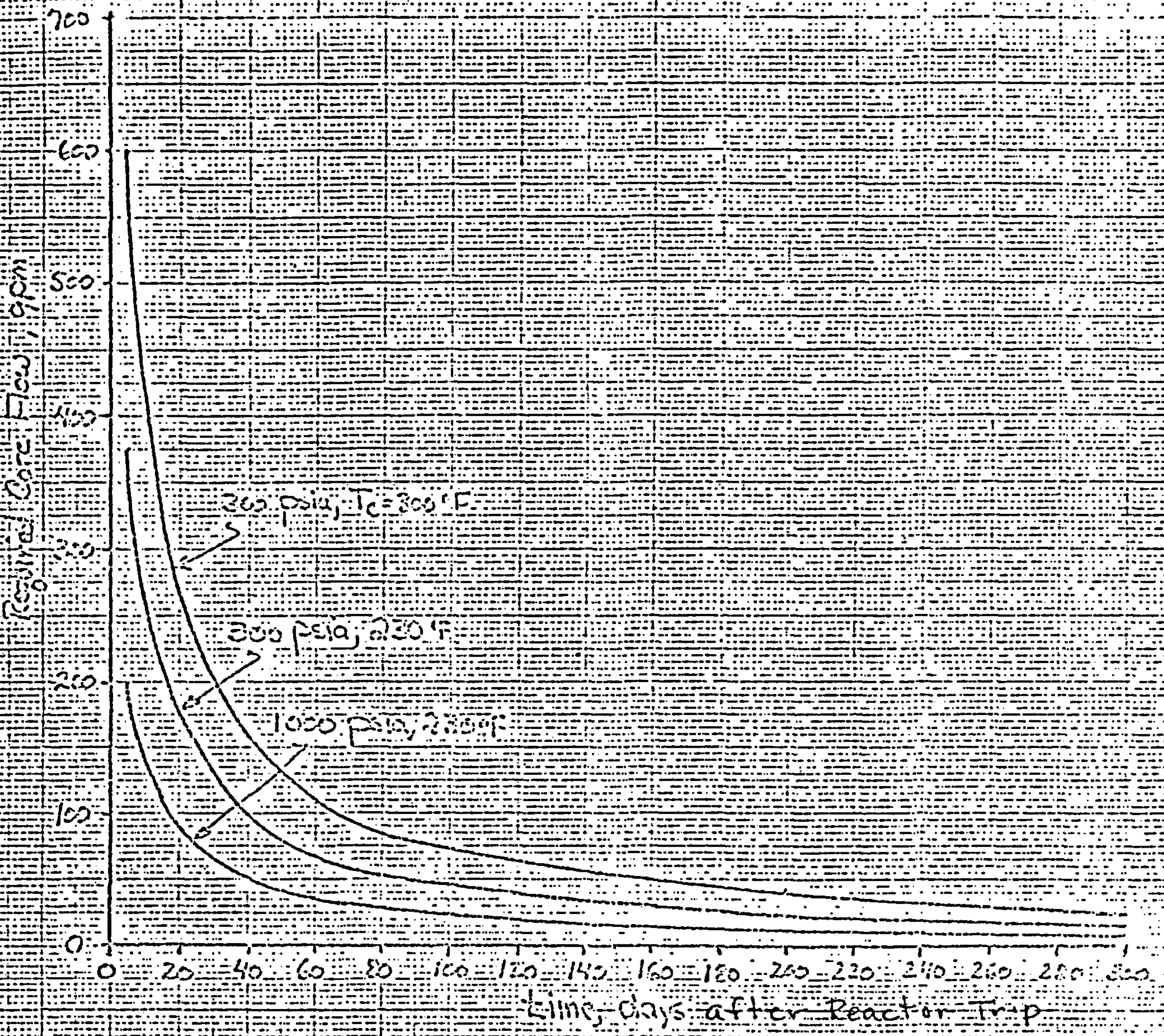
AS REQUESTED IN REFERENCE (1), THE UNCERTAINTIES INCLUDED IN THE CORE FLOW REQUIREMENTS BY T-H HAVE BEEN RE-EVALUATED BASED ON UNCERTAINTIES PROVIDED BY CENTRAL ANALYSIS. UNCERTAINTY DUE TO CORE BYPASS HAS BEEN REMOVED AND THE CORE FLOW REQUIR MENTS FROM REFERENCES (2) AND (3) ARE REPLOTTED ON FIGURES 1 AND 2. THE CONTROL ANALYSIS UNCERTAINTY OF 5.8 TIMES THE CORE FLOW REQUIRED MUST BE APPLIED TO THESE CURVES.

A. B. JACKSON

ABJ:dmd
CC: JS TULENKO
FEUM


Minimum Core Flow
From HPI System
Tin max = 90°F
Minimum HPI Flow is 5.8 times Core Flow
4 MWt
Required Core Flow, gpm
170
160
150
140
130
120
110
100
90
80
70
60
50
40
30
20
10
0
0
200
400
600
800
1000
1200
1400
1600
1800
2000
2200
2400
System Pressure, psia

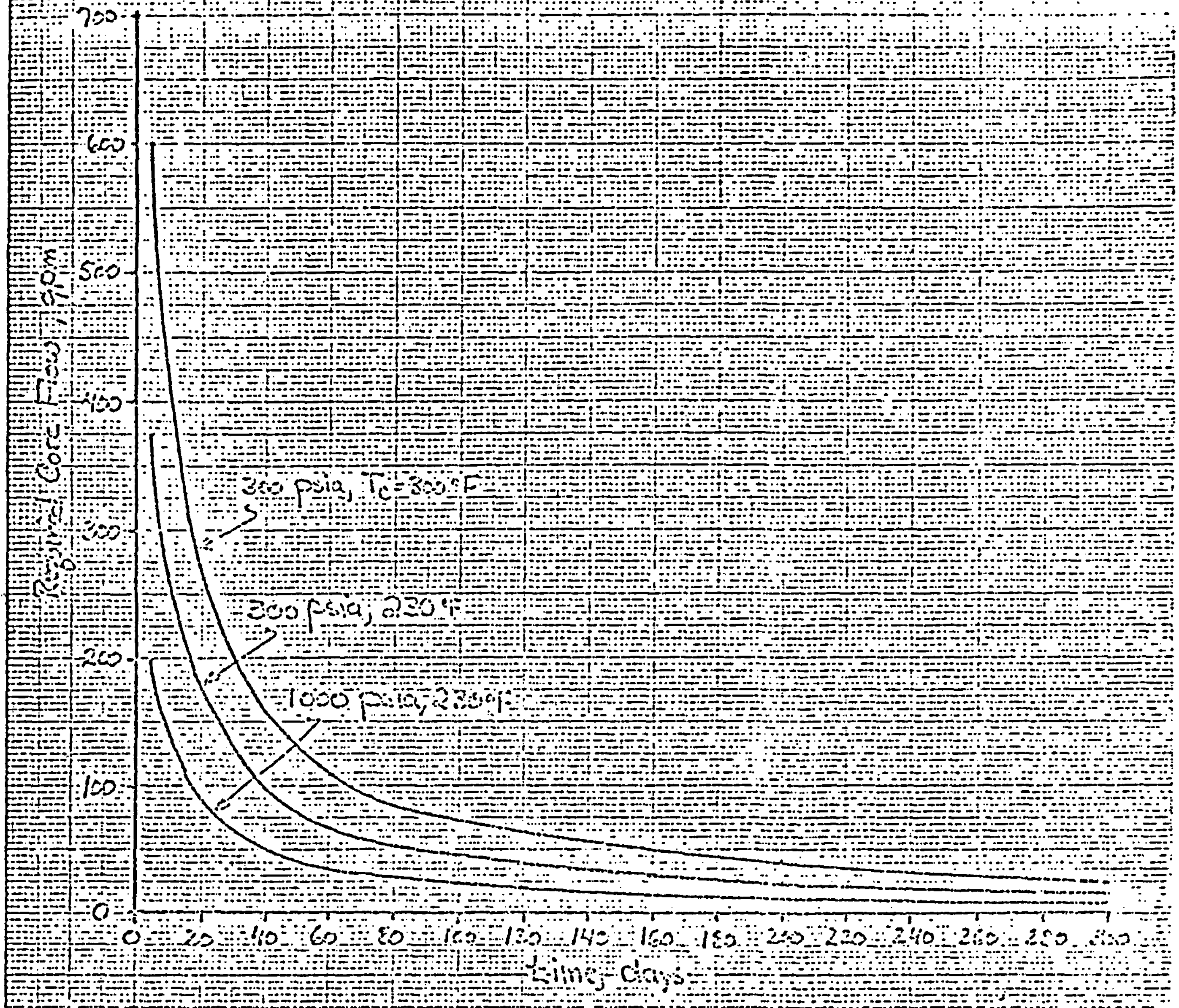

Minimum Core Flow
From HPI System
T_w max = 90°F
Minimum HPI Flow is 5.8 Times Core Flow
Required Core Flow, gpm
170
160
150
140
130
120
110
100
90
80
70
60
50
40
30
20
10
0
0
200
400
600
800
1000
1200
1400
1600
1800
2000
2200
2400
System Pressure, psig



Minimum Core Flow Required
for Long Term Cooling

Decay Heat System Flow Required = 5.8 Times Core Flow

5/79



Minimum Core Flow Required
for Long Term Cooling

Decay Heat System Flow Required = 5.8 Times Core Flow

15/79

THE BABCOCK & WILCOX COMPANY
POWER GENERATION GROUP

INSTRUCTION #310

To
R. B. DAVIS, CONTROL ANALYSIS, A. L. JACKSON, THERMAL HYDRAULICS

From
C. C. ENGLAND, LONG TERM COOLING CCEngland

BDS 663-

Cust.

File No.
or Ref.

Subj.
ALTERNATE DECAY HEAT SYSTEM REQUIREMENTS

Date
TIME: 7:22 P.M.
APRIL 9, 1979

This letter to cover one customer and one subject only.

R. B. DAVIS' MEMO OF APRIL 8, 1979 RECOMMENDS THAT DECAY HEAT SYSTEM FLOW REQUIREMENTS BE SET AT 5.8 TIMES THE CORE FLOW REQUIREMENT.

I UNDERSTAND THAT THE ATTACHED CURVE FROM THERMAL HYDRAULICS ALREADY CONTAINS SOME CONSERVATISM (ON THE ORDER OF A FACTOR OF 3) TO ACCOUNT FOR UNKNOWNS IN THE CORE.

I REQUEST THAT YOU REVIEW YOUR CALCULATIONS WITH THERMAL HYDRAULICS TO ASSURE THAT THE "CONSERVATISMS" AREN'T BEING ADDED TWICE.

A D.H. FLOW RATE MUTUALLY AGREED UP ON BETWEEN CONTROL ANALYSIS AND T.H.E. SHOULD BE ESTABLISHED.

PLEASE PROVIDE THIS BY 0800 ON 4/9/79.

CCE/CMW

ATTACHMENT

C.C. England

BABCOCK & WILCOX COMPANY
POWER GENERATION GROUP

To: CC ENGLAND

From: RB DAVIS

Cust.: TMI-2

Subj.: REQUIREMENTS FOR ALTERNATE DECAY HEAT SYSTEM

File No. or Ref.:

Date: APRIL 8, 1979 2335

This letter to cover one customer and one subject only.

Has this been factored into ADHS? x Max Core?

Yes - ADHS is 3x this 1200 gpm case

BDS 6

REFERENCE: CC ENGLAND, RB DAVIS, SAME SUBJECT, APRIL 5, 1979

IN THE REFERENCE, I RECOMMENDED THE DECAY HEAT SYSTEM FLOW REQUIREMENT BE 3 TIMES THE CORE DECAY HEAT FLOW REQUIREMENT. THIS MEMO REVISES THIS REQUIREMENT TO 5.8 TIMES THE CORE FLOW REQUIREMENT.

IN MY LATEST ANALYSIS, I IMPROVED ON MY CALCULATION TO INCLUDE (1) A BETTER HYDRAULICS MODEL, (2) A HIGHER CORE RESISTANCE (83×10^8 psi/(lbs/sec)^2) PROVIDED BY PS BARTELLS, AND (3) A CORE BYPASS FLOW OF 27%. THIS NEW DATA RAISED THE REQUIREMENT TO 5.8 TIMES THE CORE FLOW REQUIREMENT. ATTACHED IS GEORGE MEYER'S CORE FLOW REQUIREMENT ON WHICH I ADDED A NOTE ON THE DECAY HEAT SYSTEM FLOW REQUIREMENT.

THE DELTA P (FROM THE CORE FLOOD NOZZLE ACROSS THE REACTOR VESSEL AND UP TO THE DECAY HEAT DROP LINE) IS STILL NEGLIGIBLE (APPROXIMATELY .02 PSIA)

;js

JRBurris for RBD
R. B. DAVIS

Reference 5

7

THE BABCOCK & WILCOX COMPANY
POWER GENERATION GROUP

cc: D. A. Farnsworth
R. H. Stoudt

G. A. MEYER, FUEL ENGINEERING

RECEIVED
APR 10 1979
G. A. MEYER

BDS 663.5

From
J. R. GLOUDEMANS, THERMODYNAMICS UNIT, TECHNICAL STAFF

Cust.

File No.
or Ref. 845-7952-01

Subj.
COOLDOWN PRESSURE

Date
APRIL 10, 1979

This letter to cover one customer and one subject only.

COOLDOWN PRESSURES FROM ATMOSPHERIC PRESSURE TO ~ 1800 PSIA HAVE BEEN CONSIDERED. THERE ARE MANY SYSTEM RESPONSES RELATED TO PRESSURE, THE ADVANTAGES OF COOLING AT HIGHER (UP TO ~ 1800 PSIA) OR LOWER (DOWN TO ~ 15 PSIA) PRESSURES ARE:

1. <u>ADVANTAGES OF COOLING AT HIGHER PRESSURE</u>

a. GREATER MARGIN TO SATURATION TEMPERATURE.

b. LESS STRAIN ON (OPERATING) RCP's.

c. BECAUSE OF DECREASED EXPANSION OF FLUID DURING VAPORIZATION (SEE ATTACHED FIGURE), DECREASED FUEL OR FUEL-DEBRIS DISLOCATION DUE TO THE HEATING AND VAPORIZATION OF TRAPPED WATER.

2. ADVANTAGES OF COOLING AT LOWER PRESSURE

a. BECAUSE NATURAL CIRCULATION IN RESTRICTIONS AND/OR AT "HOT SPOTS" MAY REQUIRE VAPORIZATION TO ACHIEVE THE NECESSARY STEADY-STATE DRIVING HEAD, AND VAPORIZATION EXPANSION INCREASES WITH DECREASING PRESSURE (SEE ATTACHED FIGURE), LOWER PRESSURE INFERS MORE ADEQUATE COOLING OF THE MORE-RESTRICTED FLOW REGIONS IN THE CORE.

b. POSTULATING NATURAL CIRCULATION AND VOIDING AT CORE-FLOW RESTRICTIONS AS IN "2.a", AND NOTING THE DECREASE IN SATURATION TEMPERATURE WITH PRESSURE,

THE (FUEL) DEBRIS OR FUEL IN THE STARVED REGION IS AFFORDED A LOWER SURFACE TEMPERATURE (WITH LOWER PRESSURE).

c. LESS PRESSURE-INDUCED STRESS ON BOUNDARY COMPONENTS.

d. LESS BLOWDOWN (TO ATMOSPHERIC) LIKELIHOOD AND STRESS.

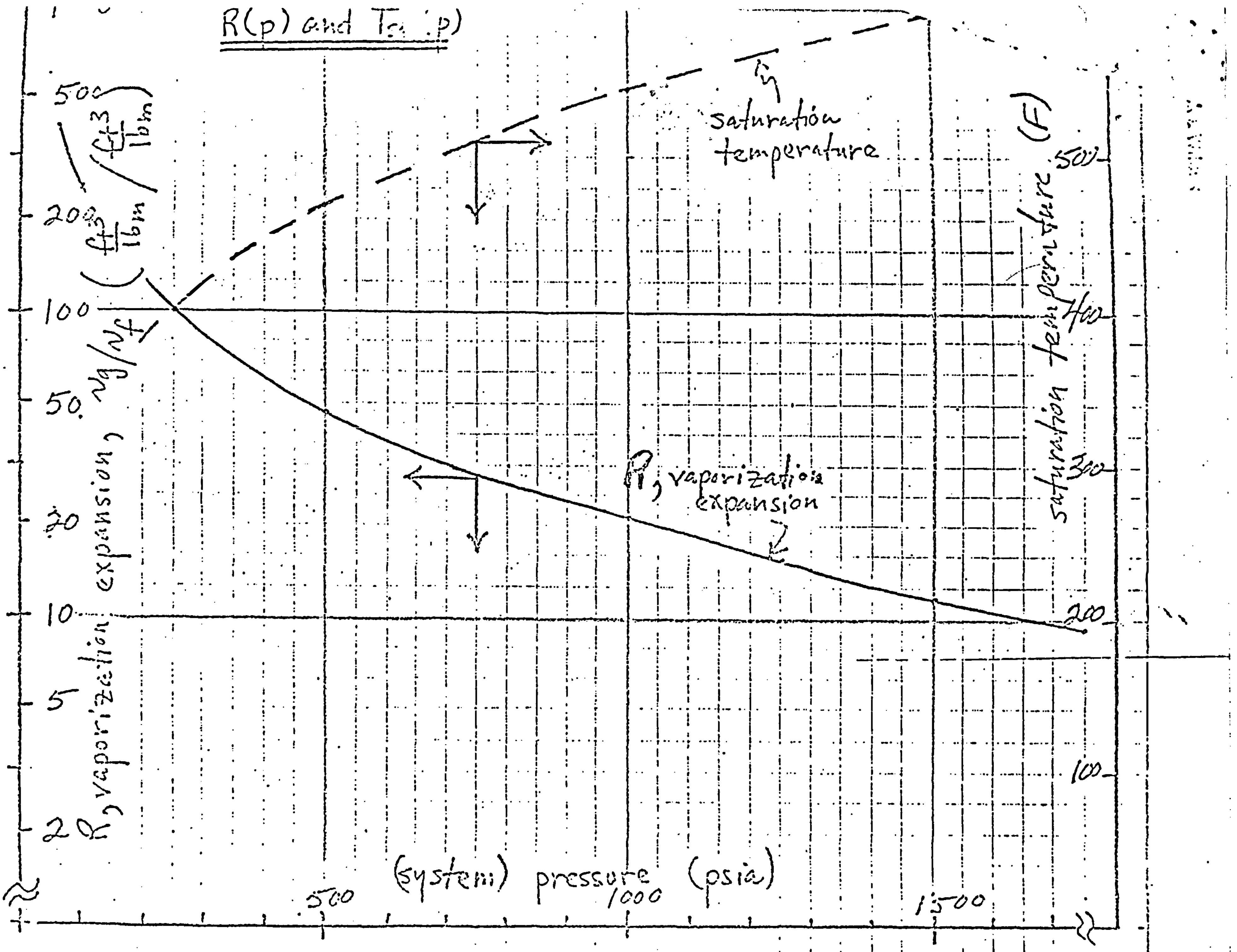
e. FACILITATED SHIFT TO THE LOWER-PRESSURE DECAY-HEAT-REMOVAL SYSTEM.

I AM CERTAIN THAT LONG-TERM COOLING SHOULD NOT BE ACCOMPLISHED AT THE HIGHER PRESSURES CONTEMPLATED, 1000 to 1800 PSIA. THE ONLY SIGNIFICANT ADVANTAGE (1a), IS OUT-WEIGHED BY ITS COUNTERPART (2b); i.e., RAISING PRESSURE RAISES T_{SAT} AND DIRECTLY RAISES THE SURFACE TEMPERATURE OF THOSE FLOW-RESTRICTED DECAY-HEAT REGIONS WHICH REQUIRE BOILING FOR THEIR HEAT TRANSFER MECHANISM.

WHILE RCP's ARE OPERATING, I RECOMMEND THE LOWER BAND OF THE CURRENT PRESSURE RANGE, i.e., ∿ 550 ± 50 PSIA. PRESSURE SWINGS SHOULD BE AVOIDED. (COOLDOWN SHOULD BE VERY GRADUAL). WHEN "COOLED DOWN" (TO APPROXIMATELY 100-200 F), AND WHEN RCP's ARE NO LONGER AVAILABLE OR ARE NOT DESIRED FOR BACKUP, SYSTEM PRESSURE SHOULD BE SLOWLY REDUCED TO APPROXIMATELY 100 PSIA (T_{SAT} ∿ 328 F). THIS LOWER PRESSURE INCREASES THE DRIVING HEAT AT VOIDING LOCATIONS AND DECREASES THE SURFACE TEMPERATURE OF BOILING-LIMITED FUEL, WHILE MAINTAINING MORE THAN 100F "MARGIN" TO SATURATION.

James R. Gloudemans
J. R. Gloudemans

VERIFIED 4/10/79 by
D. A. Farnsworth
James Gloudemans for DAF



R(p) and $T_{sat}(p)$

500
200
100
50
20
10
5
2
$\frac{ft^3}{lbm} / \frac{ft^3}{lbm}$
R, vaporization expansion, v_g/v_f ($\frac{ft^3}{lbm}$)
T_g
saturation temperature
R, vaporization expansion
(system) pressure (psia)
500
1000
1500
500
400
300
200
100
saturation temperature (F)

Reference 7

(8)

THE BABCOCK & WILCOX COMPANY
POWER GENERATION GROUP

THE-79-193

To
G. A. MEYER, MANAGER, THERMAL HYDRAULIC ENGINEERING UNIT

From
J.H.J.
J. H. JONES, THE UNIT, EXT. 3239

BDS 66

Cust.
TMI2

File No.
or Ref.

Subj.
ADIABATIC HEATUP RATES

Date
APRIL 10, 1979

This letter to cover one customer and one subject only.

FIGURE 1 SHOWS THE ADIABATIC HEATUP RATE AS A FUNCTION OF TIME IN DAYS AFTER THE LOSS OF FEEDWATER EVENT. THIS CURVE ASSUMES INSTANTANEOUS CORE UNCOVERAGE (NO HEAT REMOVAL FROM THE FUEL). THE CALCULATION WAS PERFORMED FOR A FUEL ROD ASSUMED TO BE INTACT.

FIGURE 2 SHOWS THE TIME IN HOURS TO HEAT THE FUEL FROM 200°F TO 1000°F USING THE HEATUP RATES FROM FIGURE 1.

FIGURE 3 SHOWS THE DECAY HEAT (CORE POWER) AS A FUNCTION OF TIME (SEE REFERENCE).

REFERENCE: MEMO, J. R. BURRIS TO J. D. CARLTON, "DECAY HEAT CURVE," APRIL, 2, 1979, NSS-6.

JHJ/CDB

CC: J. S. TULENKO
F.E. UNIT MANAGERS
CORE HOT SPOT/FLOW BLOCKAGE TASK FORCE

Q/A Both method and calculations have been independently checked and found to be correct.
Robert T. Chiang 4/10/79

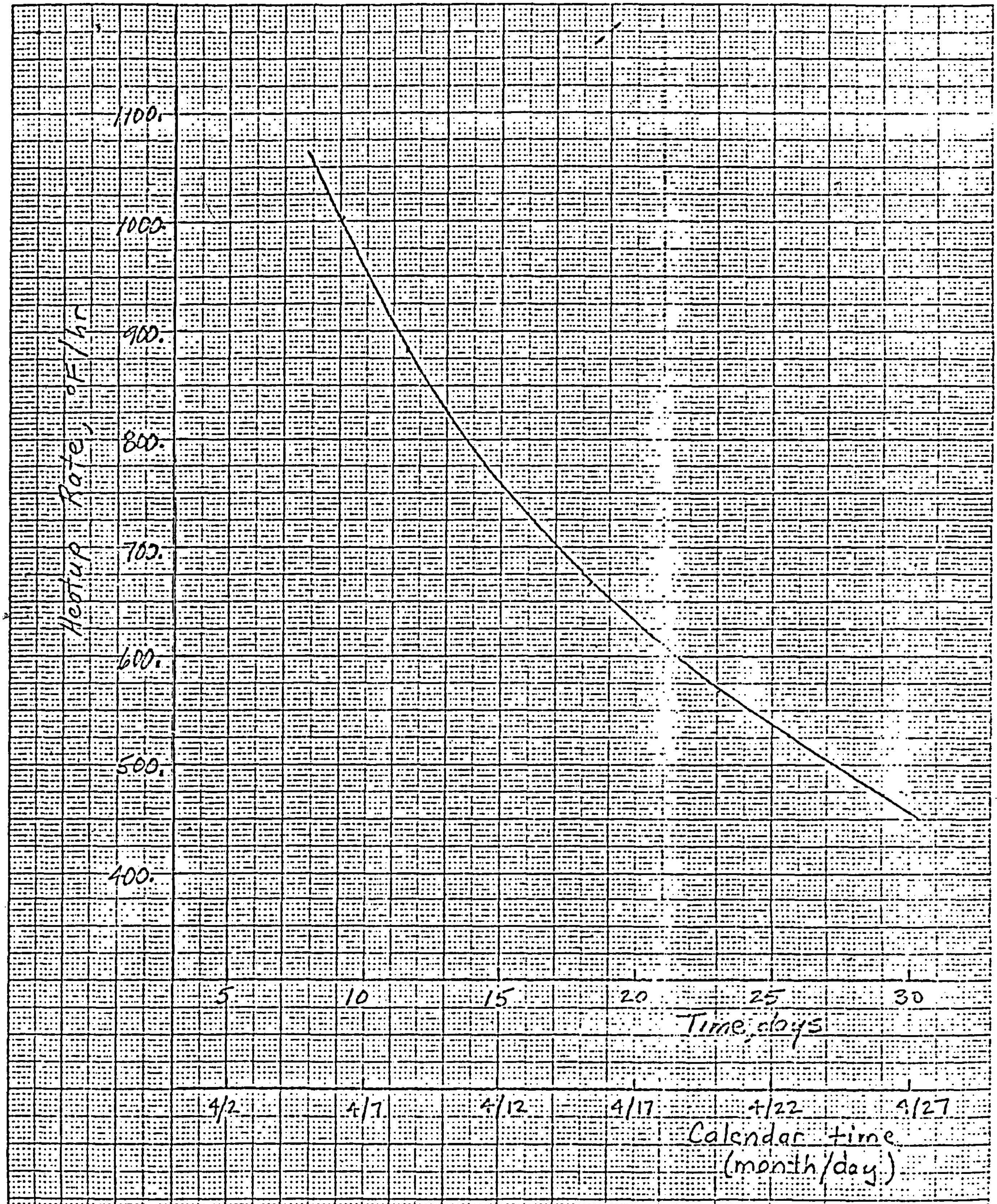


Figure 1- Heatup rate as a function of time after the loss of feedwater event (Adiabatic)

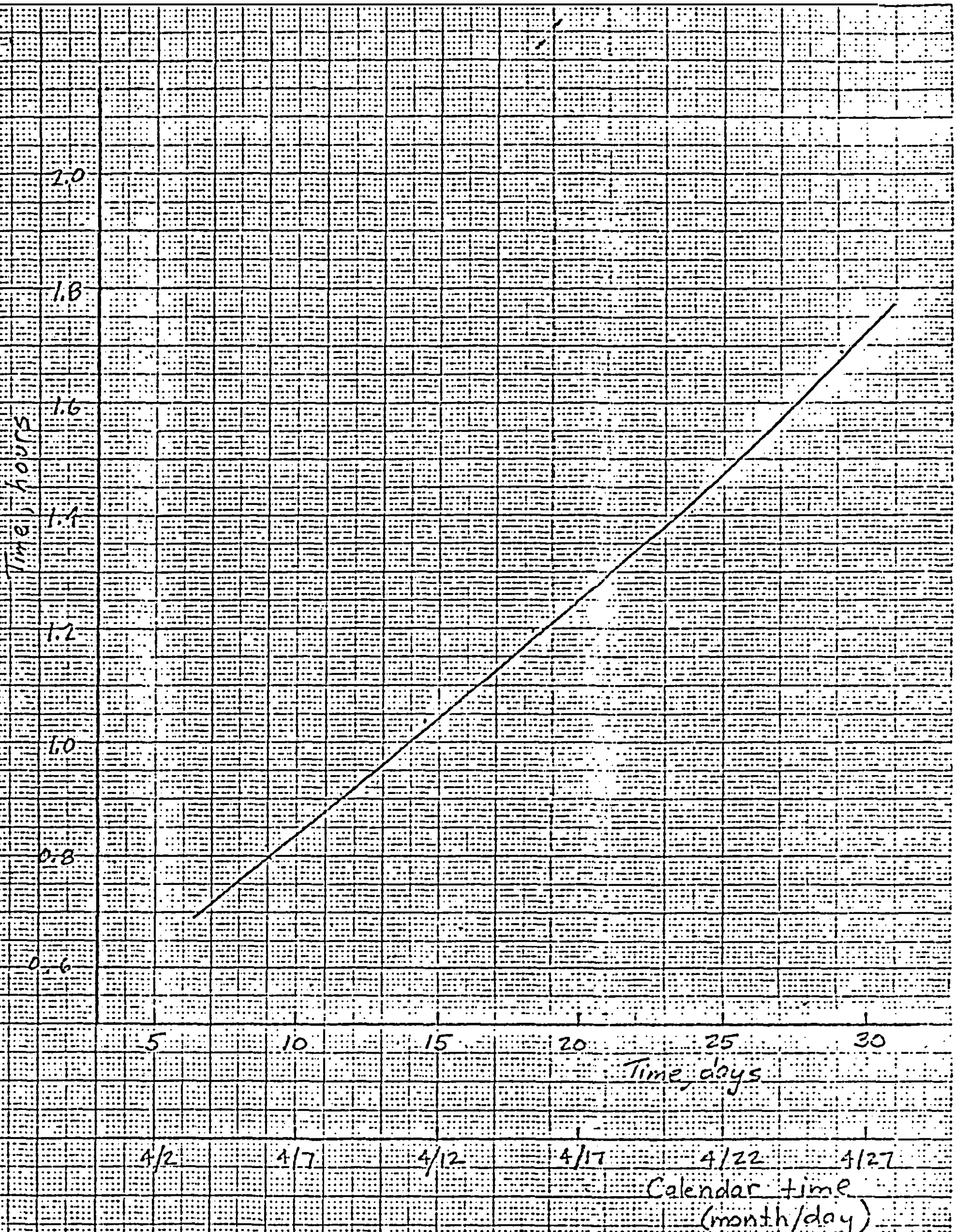


Figure 7 - Time to heat up from 200°F to 1000°F (Adiabatic)

J.W.J.
4/10/79

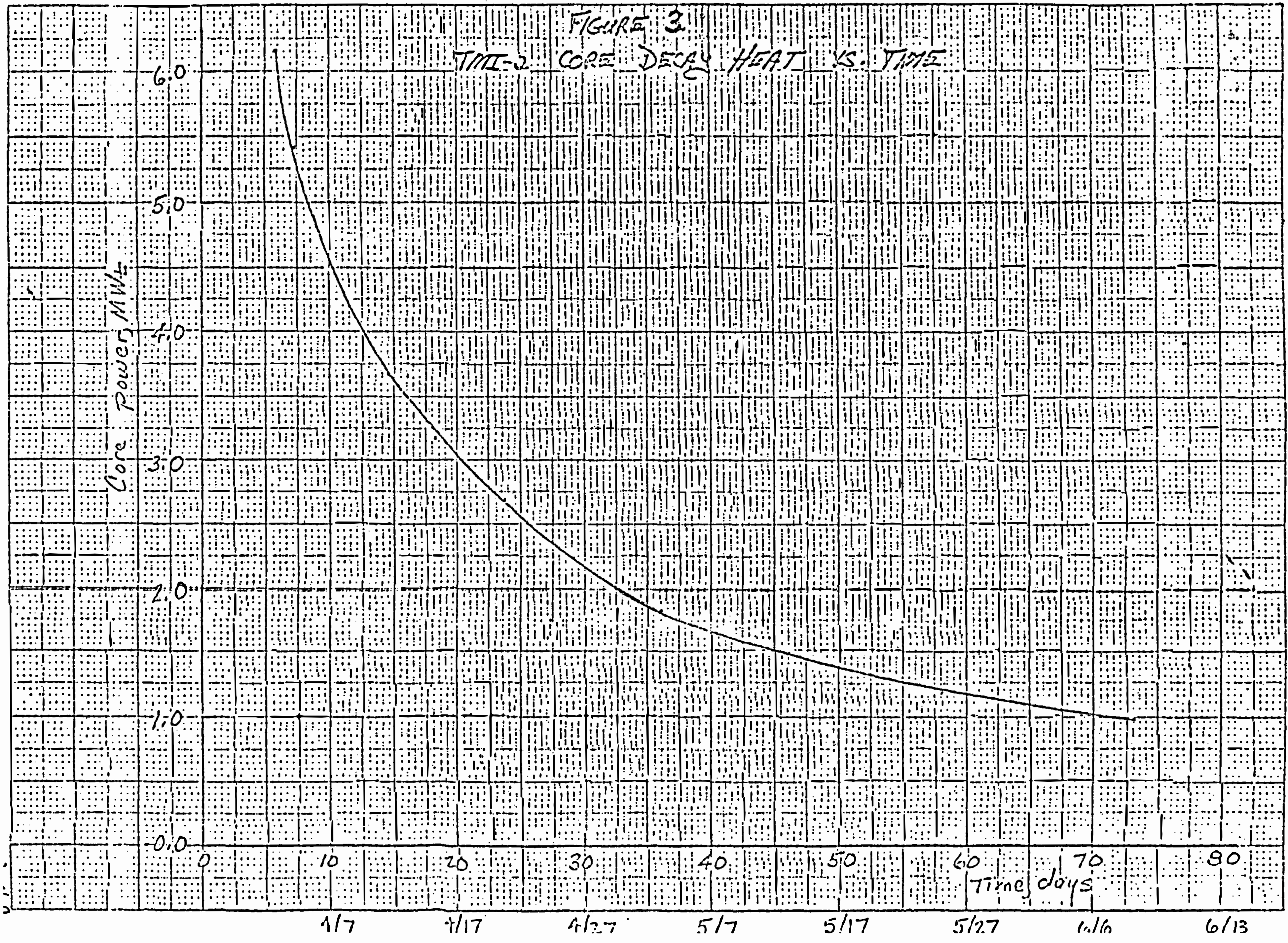


FIGURE 3

TMI-2 CORE DECAY HEAT VS. TIME

THE BABCOCK & WILCOX COMPANY
POWER GENERATION GROUP PRELIMINARY

9

To
G. A. MEYER

From
P. J. HENNINGSON Paul J Henningson

BDS 663.5

Cust.

File No.
or Ref.

Subj. INCREASED T.C. READINGS DUE TO PROXIMITY OF FUEL PARTICULATES

Date 5:30 P.M.
APRIL 10, 1979

This letter to cover one customer and one subject only.

<u>REFERENCE:</u> CORE CONDITION TASK FORCE TO J.S. TULENKO, "CURRENT ASSESSMENT OF CORE CONDITION, 4/7/79 (1800)," 4/7/79 - 7:48 P.M.

ONE POSSIBLE EXPLANATION OF THE INCREASED T.C. (LOCATED IN UEF'S) READINGS IS THE ACCUMULATION OF UO_2 FRAGMENTS IN AND AROUND THE MIXING CUP. THESE ELEVATED TEMPERATURES, RANGING FROM $\sim 100^{o}$F TO $\sim 190\ ^{o}$F ABOVE THE COOLANT TEMPERATURE, ARE IN THE CENTRAL PORTION OF THE CORE. ACCORDING TO THE REFERENCE, THIS IS THE POSITION OF THE CORE ASSUMED TO HAVE THE GREATEST DAMAGE.

THIS POSSIBILITY WAS INVESTIGATED ASSUMING UO_2 PARTICULATES WERE WITHIN THE MIXING CUP AND AT RADII OF 1/2 IN, 1 IN, 2 IN, 3 IN, AND 4 IN. AXIAL CONDUCTION AND CONVECTION WERE NEGLECTED (GROSS FAILURE).

THE RESULTS ARE SHOWN IN THE ATTACHED FIGURES. FIGURE 1 SHOWS THE TEMPERATURE DIFFERENCE (ΔT (F^{o})) EXISTING BETWEEN THE I.D. OF THE MIXING CUP AND THE O.D. OF THE INSTRUMENT STRING ASSUMING ENTRAINED UO_2. THE TEMPERATURE RISE THROUGH THE UO_2 IS SMALL FOR ALL TIMES. FIGURE 2 SHOWS THE AFFECT OF VARYING AMOUNTS OF FAILED FUEL OUTSIDE THE MIXING CUP. THE TEMPERATURE DIFFERENCE (ΔT (F^{o})) IS FROM THE SURFACE TO THE T.C. WELL SURFACE. (THE ONE INCH WIDTH OF UO_2 OUTSIDE THE T.C. WELL IS SHOWN ON FIGURE 1 FOR COMPARATIVE PURPOSES.)

IT IS POSSIBLE FOR THE ELEVATED T.C. READINGS TO BE SOLELY DUE TO LARGE AGGLOMERATES OF FUEL PARTICULATES SURROUNDING THE MIXING CUP. FURTHER CREDENCE TO THIS THEORY ARISES FROM THE CHANGE IN T.C. READINGS WHEN THE A-1 PUMP TRIPPED. THEREFORE, ANY DECISIONS UPON CHANGES IN CORE CONFIGURATION SHOULD NOT BE MADE SOLELY ON THE BASIS OF INCORE THERMOCOUPLE READINGS.

PJH:nw

CC: J. S. TULENKO
FUEL ENG. UNIT MANAGERS
CORE HOT SPOT TASK FORCE

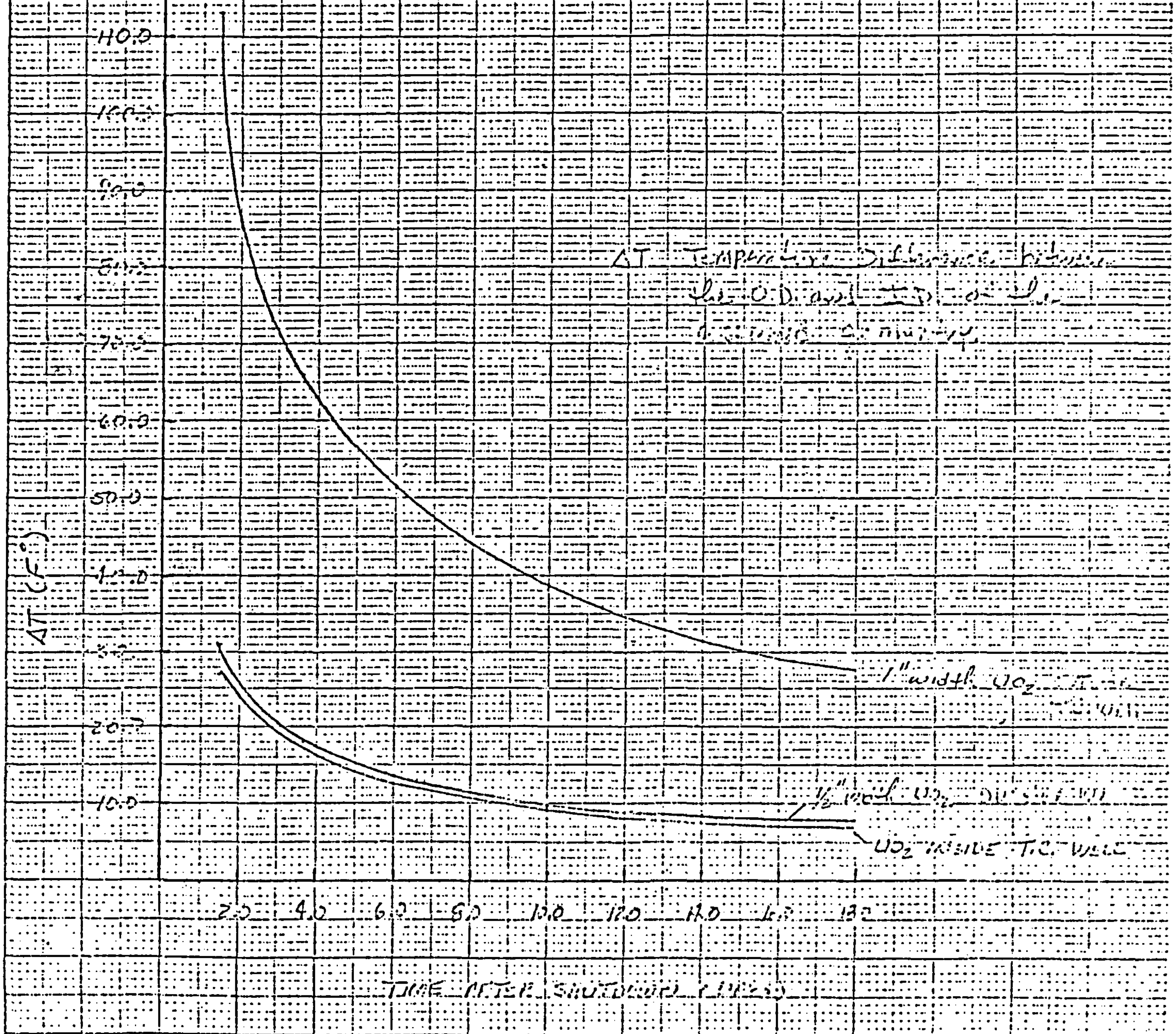

ΔT (F°)
110.0
100.0
90.0
80.0
70.0
60.0
50.0
40.0
30.0
20.0
10.0
2.0
4.0
6.0
8.0
10.0
12.0
14.0
16.0
18.0
TIME AFTER SHUTDOWN (HRS)
ΔT Temperature Difference between the O.D. and I.D. of the annulus of matrix
1" width UO2 in annulus
½" width UO2 in annulus
UO2 INSIDE T.C. WELL

FIGURE 1

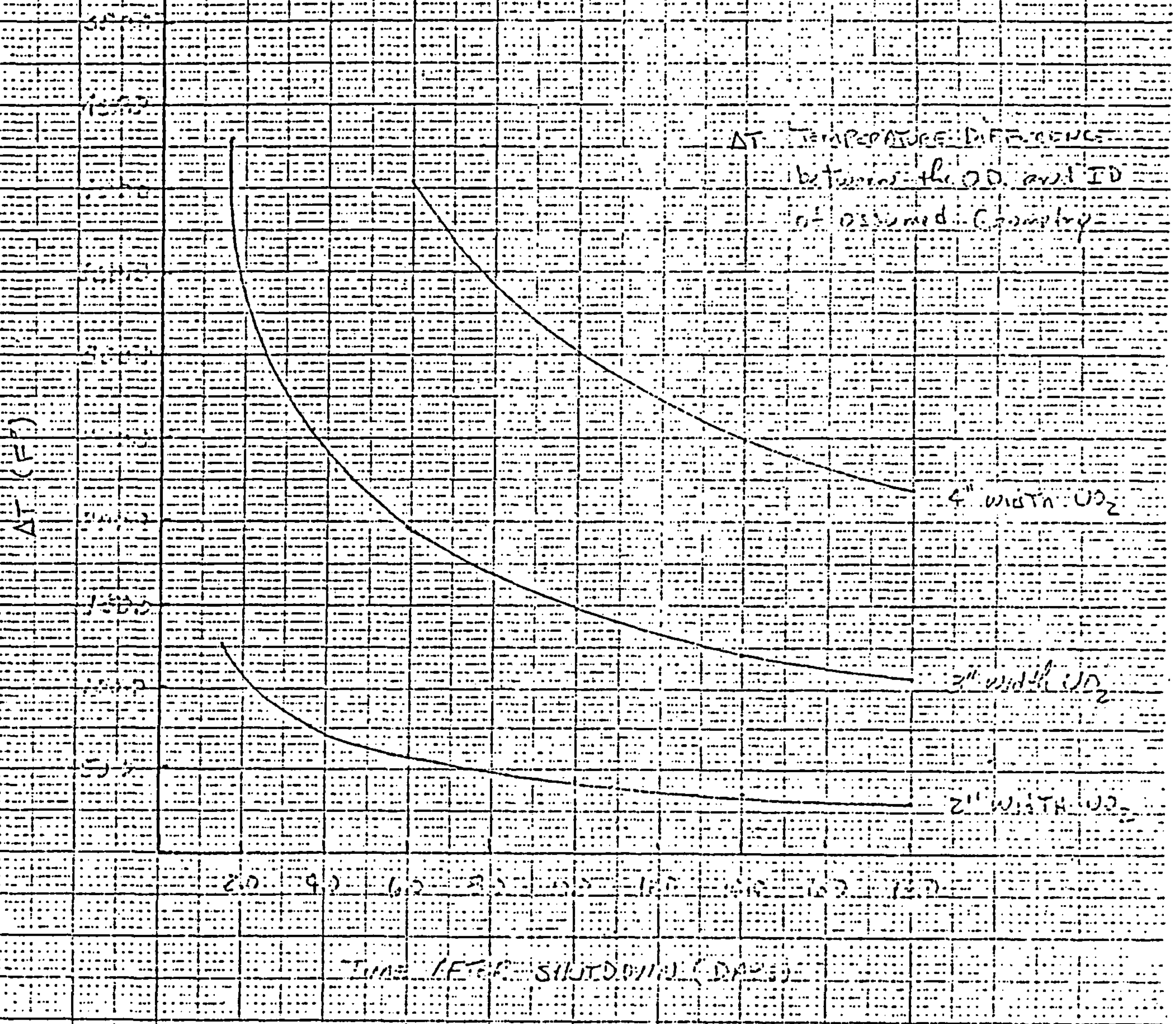
ΔT (F)
5000
4500
4000
3500
3000
2500
2000
1500
1000
500
ΔT: Temperature Difference between the OD and ID of assumed Geometry
4" width UO2
3" width UO2
2" width UO2
20
40
60
80
100
120
140
160
180
Time After Shutdown (Days)

FIGURE 2

Reference 3

THE BABCOCK & WILCOX COMPANY
POWER GENERATION GROUP

PRELIMINARY

(10)

To
DISTRIBUTION

From
J. A. WEIMER/R. L. HARNE

BDS 663-5

Cust.
TMI-2

File No.
or Ref.

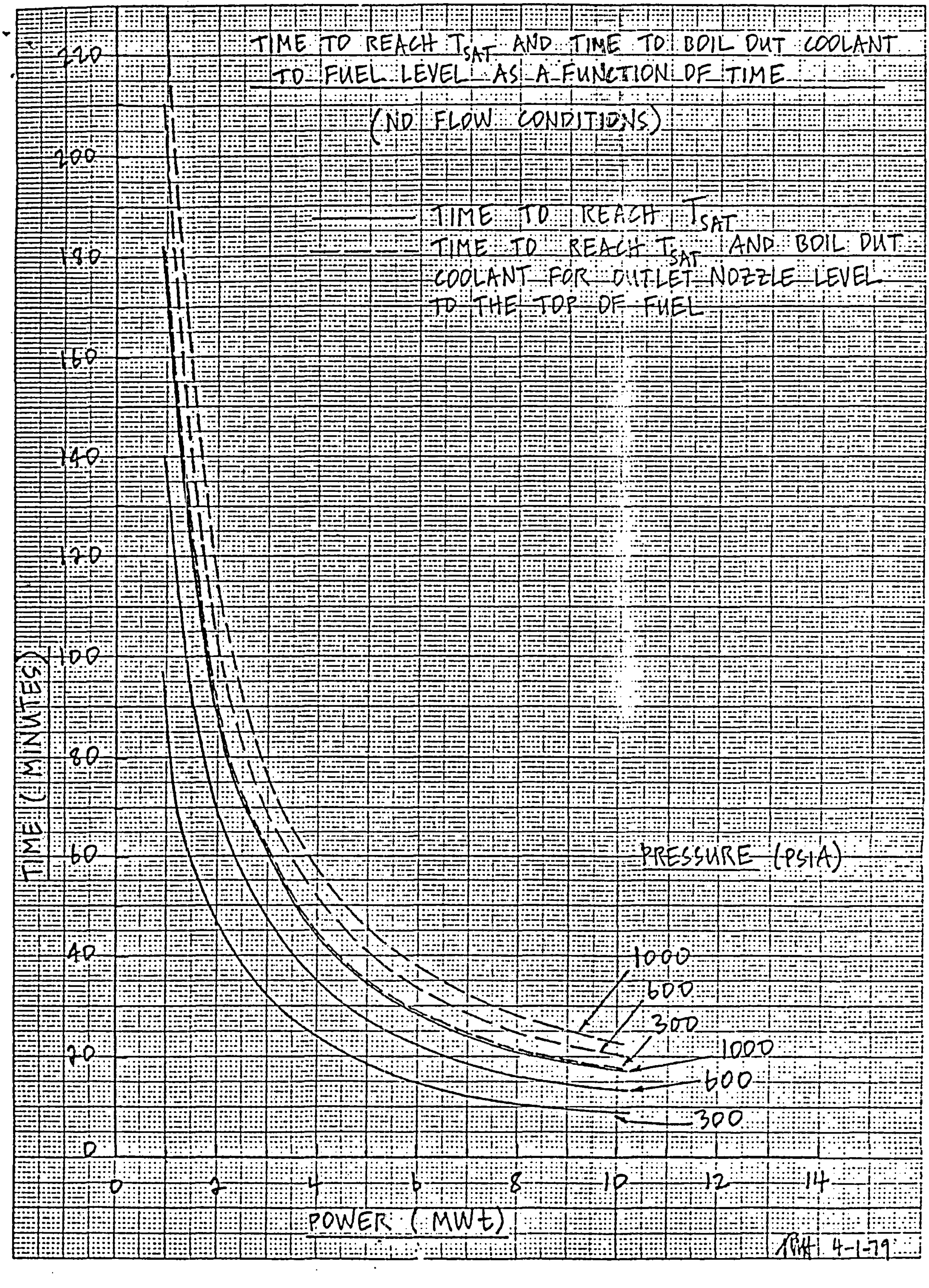
Subj.
BOILING CONDITIONS IN CORE

Date
APRIL 1, 1979

This letter to cover one customer and one subject only.

The following curve (and attached calculations) shows estimated time required to bring the water in the core to saturation temperature at 300, 600 and 1000 psia from its present 280°F energy level. This calculation assumes a no-flow condition with 161" of stagnant coolant available for heat transfer. A second calculation was made to determine additional time required to vaporize the coolant such that active fuel will be exposed to steam.

JAW/RLH:cmw
Attachment



TIME TO REACH T_{SAT} AND TIME TO BOIL OUT COOLANT TO FUEL LEVEL AS A FUNCTION OF TIME

(NO FLOW CONDITIONS)

TPH 4-1-79

Reference 2

Item #49

⑪

THE BABCOCK & WILCOX COMPANY
POWER GENERATION GROUP

To
J. D. CARLTON, PLANT DESIGN

THE-79-162

From
G. A. MEYER, MANAGER, T-H ENGINEERING UNIT (3218)

BDS 663.5

Cust.

File No.
or Ref.

Subj.
MININUM CORE FLOW - LONG TERM COOLING

Date
Time 1200
APRIL 4, 1979

This letter to cover one customer and one subject only.

Reference: 86-1100401-00, "Minimum Core Flow - Long Term," April 2, 1979, L. L. Losh.

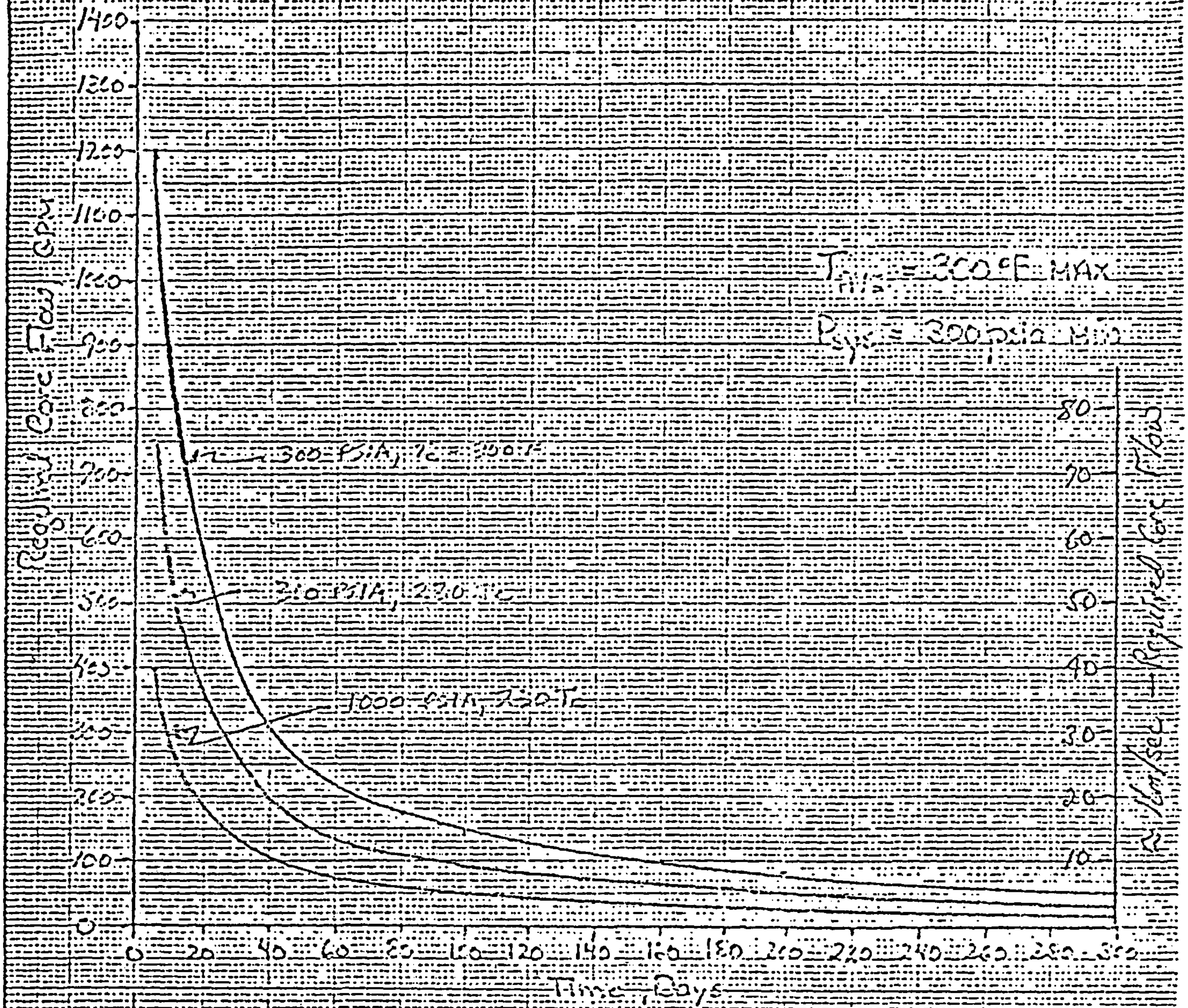
The attached figure expands upon the information provided in the reference to provide minimum long term cooling flow requirements for various proposed operating conditions. Each of the curves presented accounts for core blockage by the application of a safety factor of 3.0 to the calculated flow requirement. The curves provided are essentially the same as those determined this morning by L. L. Losh and J. R. Bohart, thus providing a QA verification of their analysis.

GA Meyer

GAM/sgh

cc: J. S. Tulenko gear
FE Unit Managers
C. Parks

Reference 1



Minimum Core Flow Required
for Long Term Cooling

Figure 1.

THE BABCOCK & WILCOX COMPANY
POWER GENERATION GROUP

PRELIMINARY

(12)

To
G.A. MEYER - MANAGER, THERMAL HYDRAULIC ENGINEERING

From
R.M. HIATT - THERMAL HYDRAULIC ENGINEERING

THE-79-195

BDS 663.5

Cust.
TMI-2

File No.
or Ref.

Subj.
CORE FLOW DISTRIBUTION FOR ONE PUMP AND TWO PUMP OPERATION

Date
APRIL 10, 1979

This letter to cover one customer and one subject only.

ONE OF THE IMPORTANT CONSIDERATIONS IN ANALYZING THE TMI-2 CORE BLOCKAGE IMPACT ON CORE COOLING IS THE FLOW DISTRIBUTION IN THE CORE. A DETERMINATION OF CORE INLET FLOW DISTRIBUTION FOR ONE PUMP OPERATION WITHOUT BLOCKAGE WAS BASED ON A REVIEW OF THE VESSEL MODEL FLOW TEST (VMFT) DATA AND ENGINEERING JUDGEMENT AS FOLLOWS:

THE TRANSFER OF MASS CAN BE MODELED SIMILAR TO ELECTRIC CIRCUITRY. THE FLOW PATHS CAN BE REPRESENTED BY A SYSTEM OF RESISTANCES AND THE FLOW WILL SELECT THE FLOW PATH IN SUCH A WAY TO EQUALIZE THE PRESSURE DROP ACROSS THE SYSTEM. THUS, FLOW HAS A "LOOK AHEAD" CAPABILITY THAT TENDS TO EQUALIZE THE POTENTIAL (ΔP) ACROSS A SYSTEM OF RESISTANCE. THE FLOW CHANNELS THROUGH THE CORE CAN BE VIEWED AS A SYSTEM OF RESISTANCES. BASED ON THIS PRINCIPLE, ASSUMING THAT RATE OF CHANGE OF MOMENTUM FROM COLD LEG INLET TO HOT LEG OUTLET IS THE SAME, AND AN INSPECTION OF THE VMFT INLET FLOW FACTORS CAN BE USED TO IDENTIFY THE FLOW DISTRIBUTION AT THE CORE INLET FOR ONE PUMP OPERATION.

FIGURE 1 ILLUSTRATES THE CORE INLET FLOW FACTORS FOR 4 PUMP OPERATION FOR THE 177 FA PLANT. SUMMARIZED ON FIGURE 1 ARE THE AVERAGE FLOW FACTORS FOR EACH QUADRANT. THE OUTLET PIPE IS LOCATED BETWEEN QUADRANTS A1 AND A2

AND BETWEEN B1 AND B2. THUS, WITH BOTH LOOPS OPERATING THE RESISTANCES ACROSS THE CORE FOR EACH QUADRANT WOULD BE EXPECTED TO BE ABOUT THE SAME. FIGURE 1 ILLUSTRATES A SLIGHT BIAS TOWARDS QUADRANTS A1; A2; AND B1. THIS BIAS IS PROBABLY DUE TO FABRICATION TOLERANCES.

FIGURE 2 ILLUSTRATES THE CORE INLET FLOW FACTORS FOR TWO PUMP OPERATION. NOTE THAT THE OPERATING PUMPS A1 AND B2 ARE ARRANGED IN OPPOSITE QUADRANTS. THE SUMMARY FLOW FACTOR FOR EACH QUADRANT AGAIN ILLUSTRATES A RELATIVELY UNIFORM FLOW FOR THE QUADRANTS WITH A SLIGHT BIAS TOWARDS THE QUADRANTS CONTAINING THE OPERATING PUMPS. NOTE THAT THE OPERATING PUMP COMBINATION IS A TWO LOOP OPERATION. THEREFORE, THE RESISTANCE ACROSS EACH QUADRANT SHOULD BE ABOUT THE SAME WITH THE RESISTANCES HIGHER FOR THOSE CHANNELS FARTHEST FROM THE PUMP AND FARTHEST FROM THE OUTLET PIPING. HOWEVER, THE MOMENTUM OF THE FLOW DISCHARGED INTO THE CORE MAY BE SUFFICIENT TO OVERCOME LATERAL CORE RESISTANCES IN THE LOWER PLENUM. THIS IS ILLUSTRATED IN FIGURE 3.

FIGURE 3 SHOWS THE CORE INLET FLOW DISTRIBUTION FOR A TWO PUMP OPERATION A1, AND B1 IN THIS INSTANCE THE TWO PUMPS ARE NOT OPPOSING. NOTE THAT BOTH LOOPS ARE OPERATING. THUS, THE RESISTANCE ACROSS EACH QUADRANT ARE ABOUT THE SAME. HOWEVER, AS MENTIONED PREVIOUSLY, THE LATERAL MOMENTUM OF THE FLOW DISCHARGED INTO THE LOWER PLENUM FORCES FLOW TO THE ADJACENT QUADRANT WITH A2 AND B2 HIGHER IN INLET FLOW THAN A1 AND B1 CONTAINING THE PUMPS.

FIGURE 4 ILLUSTRATES THE CORE INLET FLOW DISTRIBUTION FOR A 1 LOOP 2 PUMP OPERATION (A1 AND A2). THE INLET FLOW FACTORS SHOW A BIAS WITH QUADRANTS A1 AND A2 HIGHER IN FLOW. THIS SUGGESTS THAT THE RESISTANCE OF THE CORE QUADRANTS FOR THE CLOSED LOOP IS HIGH OVERCOMING THE LATERAL MOMENTUM OF THE

DISCHARGED FLUID AND RECEIVES LESS FLOW. FIGURE 4 SHOWS THAT ABOUT 4% MORE FLOW ENTERS LOOP A CORE QUADRANTS THAN ENTERS LOOP B QUADRANTS. WHEN VIEWED WITH THE INFORMATION OF FIGURE 3, IT IS BELIEVED THAT THE QUADRANTS B1 AND B2 HAVE A HIGHER RESISTANCE OVER THE COMPLETE LENGTH OF THE CORE. THIS OCCURS DUE TO THE "LOOK AHEAD" CAPABILITY OF THE COOLANT WHICH SEES THE CLOSED LOOP AND THE LATERAL RESISTANCE BETWEEN QUADRANTS B1 AND B2 AND THE OUTLET PIPING OF LOOP A. THEREFORE, IT IS CONCLUDED THAT A COMBINATION OF FIGURE 3 AND FIGURE 4 IS THE MOST REPRESENTATIVE OF ONE PUMP OPERATION. FROM FIGURE 3, THE QUADRANT ADJACENT AND IN THE SAME LOOP WILL BE BIASED ABOUT (1.5 - 2.5%) HIGH IN CORE INLET FLOW. THE FLOW FACTORS SHOWN IN FIGURE 5 ARE RECOMMENDED FOR ANALYZING TMI-2 ONE PUMP CORE INLET FLOW CONDITION.

Reid M. Hiatt

RMH/FFA

CC: F.E. UNIT MGRS.
J.S. TULENKO
CORE HOT SPOT TASK FORCE

Note:

Fig 1

TC FRESS

SPND STRING NUMBERS AND LOCATIONS - 177 FA CORE

☐ X STRING NC

A_2 Ave = 1.0103

A_1 Ave = 1.01165

					.937	.890	.943	.967	.970					
			.963	1.05	.983	31 .967	30 .983	.977	.993	.960	.960			
		.953	1.017	1.010	32 .977	1.04	.983	29 .997	28 1.073	1.000	1.010	52 .987		
	.980	1.020	.990	33 1.120	1.027	1.017	1.040	1.037	27 1.030	1.000	1.080	1.013	51 .987	
	.987	1.010	34 1.070	1.097	1.080	7 1.040	1.027	5 1.070	1.037	26 1.070	.987	.993	1.043	
.410	.940	35 1.033	1.067	1.140	1.043	6 1.000	4 1.023	.970	1.140	.990	24 1.043	23 1.000	1.023	.95
.950	36 .990	.957	1.083	9 .973	8 1.037	1.017	1.007	3 1.017	1.093	25 1.117	1.010	22 .993	.973	.98
37 893	.963	1.120	.930	10 1.057	1.077	1.083	1 1.107	2 1.020	1.053	.920	.987	21 .980	.970	.96
91.7	.963	1.000	1.120	11 1.143	1.000	1.097	.997	1.033	1.037	19 1.083	20 1.033	.923	1.017	1.01
883	38 .940	39 .980	1.063	1.040	12 1.070	1.027	1.070	.977	.957	18 1.000	.983	50 .967	1.007	.88
	.940	40 .977	1.110	1.170	1.057	13 1.073	1.010	16 .950	17 1.037	.930	1.000	1.000	49 .960	
	.900	.957	41 .993	1.117	1.110	1.033	14 .947	15 .973	1.000	.970	.970	.933	.900	
		.977	.930	42 .980	43 .913	1.050	1.070	1.017	47 .923	.933	48 .893	.947		
			.900	.990	44 .940	.957	.960	.970	.940	.937	.907			
					.910	45 .957	46 .910	.920	46 .897					
	2	3	4	5	6	7	8	9	10	11	12	13	14	15

B_2 Ave = 1.00597

B_1 Ave = .97244

DMER

JOB NO.

RC PRESS ______

Fig. 2

SPND STRING NUMBERS AND LOCATIONS – 177 FA CORE

K
STRING NO.

Az
Ave=.9962

Az
Ave=1.01

					.920	.907	.970	1.060	1.070					
			.933	1.020	.973	31 .907	30 .987	.993	1.007	1.020	1.037			
		.960	.987	.980	32 .943	1.040	1.003	29 1.030	28 1.033	1.043	1.090	52 1.050		
	.940	.997	.930	33 1.097	1.023	.997	1.047	1.037	27 1.047	.957	1.170	1.093	51 .987	
	.947	.983	34 1.043	1.063	1.067	7 1.067	1.040	5 1.083	1.017	26 1.107	.990	1.023	1.037	
23	.927	35 1.003	1.050	1.097	1.067	6 .983	4 .973	.967	1.020	1.000	24 1.033	23 .980	.990	.923
80	36 .973	.967	1.053	9 .970	8 .987	1.047	1.040	3 1.017	1.087	25 1.137	1.007	22 .940	.967	.933
37 7	.930	1.067	.893	10 1.070	1.103	1.113	1 1.097	2 1.017	1.023	.910	.970	21 .933	.950	.89
97	.953	.960	1.123	11 1.083	.993	1.127	1.077	1.053	1.057	19 1.103	20 1.023	.953	.960	.910
43	38 .930	39 .950	1.057	1.037	12 1.137	1.033	1.090	1.043	.993	18 1.037	1.040	50 .923	.950	.837
	.923	40 1.013	1.097	1.153	1.040	13 1.087	1.023	16 .970	17 1.057	.943	1.063	1.003	49 .920	
	.910	.963	41 1.010	1.073	1.090	1.037	14 .963	15 .990	1.043	.997	.993	.890	.863	
		.973	.937	42 .983	43 .927	1.033	1.083	1.023	47 .910	.970	48 .890	.933		
			.920	1.017	44 .960	.943	.960	.963	.927	.920	.897			
					.953	45 .980	45 .933	.930	46 .883					
1	2	3	4	5	6	7	8	9	10	11	12	13	14	15

Bz
Ave=1.006333

Bz
Ave=.97277

RC PRESS ____

Fig. 3

SPND STRING NUMBERS AND LOCATIONS – 177 FA CORE

K	STRING NO.

Ave=1.04098

Ave=.983

A1

A2

					1.040	.977	1.000	1.077	1.063					
			1.023	1.140	1.050	31 1.017	30 1.047	.980	.930	.960	.967			
		1.023	1.103	1.060	32 1.020	1.040	1.000	29 1.057	28 1.043	1.003	.977	52 .943		
	1.013	1.097	.980	33 1.163	1.033	1.000	1.080	1.010	27 1.000	.953	1.077	1.017	51 .910	
	1.010	1.027	34 1.133	1.080	1.077	7 1.070	1.060	5 1.040	.960	26 1.033	.933	.960	.970	
963	.990	35 1.033	1.057	1.103	1.130	6 1.010	4 .950	.953	1.123	.953	24 .957	23 .967	.980	.893
000	36 1.007	1.030	1.047	9 1.000	8 1.003	1.053	.993	3 .947	1.037	25 1.073	.960	22 .913	.910	.897
37 7	.947	1.097	.993	10 1.087	1.107	1.120	1 1.070	2 .973	.960	.890	.940	21 .913	.950	.863
910	.953	.977	1.143	11 1.120	.983	1.153	1.047	1.030	1.047	19 1.070	20 1.013	.937	.940	.930
913	38 .930	39 .950	1.093	1.060	12 1.153	1.003	1.080	.980	.983	18 1.033	1.013	50 .920	.937	.823
	.913	40 .997	1.060	1.153	1.023	13 1.087	1.027	16 .967	17 1.073	.943	1.043	.990	49 .917	
	.897	.950	41 .983	1.127	1.103	1.057	14 .963	15 1.003	1.057	1.003	1.000	.883	.857	
		.927	.930	42 .963	43 .940	1.063	1.103	1.037	47 .913	.977	48 .887	.943		
			.910	1.017	44 .953	.977	.983	.980	.930	.907	.907			
					.910	45 .960	45 .920	.923	46 .850					
1	2	3	4	5	6	7	8	9	10	11	12	13	14	15

B1

B2

Ave=1.00856

Ave=.915

Fig. 4

RC PRESS ______

SPND STRING NUMBERS AND LOCATIONS — 177 FA CORE

K STRING N

Ave=1.0489 A2

Ave =1.0 A1

					.950	.953	.963	1.010	1.020					
			.977	1.040	1.023	31 .963	30 1.020	.997	.993	1.007	1.027			
		1.010	1.053	1.040	32 1.013	1.050	1.020	29 1.050	28 1.040	1.030	1.050	52 1.033		
	1.033	1.060	1.023	33 1.170	1.077	1.027	1.070	1.057	27 1.040	1.003	1.157	1.100	51 1.027	
	1.010	1.063	34 1.090	1.100	1.097	7 1.090	1.063	5 1.063	1.030	26 1.123	1.017	1.040	1.030	
027	1.037	35 1.013	1.080	1.120	1.147	6 1.060	4 1.043	1.073	1.087	1.063	24 1.050	23 1.080	1.120	1.09
.167	36 1.090	1.077	1.093	9 1.030	8 1.053	1.077	1.037	3 1.003	1.043	25 1.093	1.017	22 1.047	1.047	1.04
37 3	1.013	1.040	.983	10 .997	1.040	1.027	1 1.060	2 1.013	.867	.877	.817	21 .910	.953	.953
790	1.053	1.000	1.140	11 1.057	1.030	1.070	1.033	1.033	1.040	19 1.100	20 1.037	1.050	1.033	1.07
.003	38 1.013	39 1.013	1.073	1.073	12 1.117	1.083	1.077	1.027	1.007	18 1.043	1.087	50 1.043	1.057	1.00
	1.013	40 1.060	1.067	1.067	1.037	13 1.043	1.043	16 1.013	17 1.040	.977	1.010	1.013	49 .997	
	.997	1.020	41 1.003	1.050	1.053	.950	14 .877	15 .863	.950	.853	.877	.833	.830	
		.813	.827	42 .817	45 .770	.840	.873	.857	47 .757	.683	48 .717	.787		
			.730	.793	44 .777	.770	.810	.863	.827	.853	.867			
					.853	43 .900	.837	.837	46 .797					
1	2	3	4	5	6	7	8	9	10	11	12	13	14	15

B2 Ave=.97314

B1 Ave =.9385

MER

JOB No.

RC PRESS ____________

Fig. 5.

Best estimate of core inlet flow factors
for 1 pump operation 177 FA plants

K STRING NO.

Ave =1.[illegible]95

Ave = 1.05

A2

A1

					.936	.939	.963	1.025	1.035					
			.963	1.025	1.008	31 .949	30 1.020	1.012	1.008	1.022	1.042			
		.9	1.038	1.025	32 .998	1.035	1.020	29 1.065	28 1.055	1.045	1.065	52 1.048		
	1.018	1.	1.008	33 1.153	1.061	1.012	1.070	1.073	27 1.055	1.018	1.174	1.116	51 1.042	
	.995	1.	34 1.074	1.084	1.081	7 1.074	1.063	5 1.079	1.045	26 1.140	1.032	1.055	1.045	
.012	1.022	.	1.064	1.104	1.130	6 1.044	4 1.043	1.089	1.103	1.079	24 1.065	23 1.096	1.137	1.109
150	36 1.074	1.	1.077	9 1.015	8 1.038	1.061	1.037	3 1.018	1.058	25 1.109	1.032	22 1.062	1.062	1.055
37 3	1.013	1.	.983	10 .997	1.040	1.027	1 1.060	2 1.013	.967	.877	.817	21 .910	.953	.953
990	1.053	1.	1.140	11 1.057	1.030	1.070	1.033	1.033	1.040	19 1.100	20 1.037	1.050	1.033	1.077
.003	38 1.013	1.	.073	1.073	12 1.117	1.083	1.077	1.027	1.007	18 1.043	1.087	50 1.043	1.057	1.007
	1.013	1.	.067	1.067	1.037	13 1.043	1.043	16 1.013	17 1.040	.977	1.010	1.013	49 .997	
	.997	1.	41 .003	1.050	1.053	.950	14 .877	15 .863	.950	.853	.877	.833	.830	
		8	.827	42 .817	43 .770	.840	.873	.857	47 .757	.683	.717	.787		
			730	.793	44 .777	.770	.810	.863	.827	.853	.867			
					.853	45 .900	45 .837	.837	46 .777					
1	2			5	6	7	8	9	10	11	12	13	14	15

B2

B1

Ave = .97

Ave = .9385

THE BABCOCK & WILCOX COMPANY
POWER GENERATION GROUP

13

To	J. D. CARLTON
From	B Davis for RM Gribble R. M. GRIBBLE (EXT. 2684)
Cust.	TMI-2
Subj.	CORE BYPASS FLOW FOR CORE BLOCKED AT TOP ONLY

P. S. BARTELLS
J. R. BURRIS
J. M. KNOLL
P. A. TREVENTI
J. R. GLOUDEMAS
C. E. PARKS
E M DUNN
D W LEETLE BDS 65

File No.
or Ref.

Date APRIL 8, 1979

This letter to cover one customer and one subject only.

REFERENCE: MEMO, R. M. GRIBBLE TO J. D. CARLTON, "CORE BYPASS FLOW THROUGH CORE BASKET (UNIFORM BLOCKAGE)," 4/8/79.

THE REFERENCED MEMO REPORTS CORE BYPASS FLOW FOR A UNIFORMLY BLOCKED CORE. FOR THIS CONDITION, 22% OF THE VESSEL FLOW BYPASSES THE CORE THROUGH THE CORE BASKET. ANOTHER CASE HAS BEEN CONSIDERED AND IS REPORTED HEREIN.

CORE BLOCKAGE AT ONLY THE TOP OF THE CORE HAS BEEN ANALYZED TO DETERMINE CORE BYPASS FLOW FOR THIS MORE LIMITING SITUATION. RESULTS OF THIS ANALYSIS INDICATED THAT CORE BYPASS FLOW FOR THE CORE BLOCKED ABOVE THE UPPERMOST INTERMEDIATE SPACER GRID WILL BE APPROXIMATELY 27% LEAVING 73% AVAILABLE FOR CORE HEAT REMOVAL.

MAJOR ANALYSIS ASSUMPTIONS FOLLOW:

1. MAXIMUM RESISTANCE OF THE CORE AND CORE BASKET
   = 83 X 10^{-8} PSI/$(LB/SEC)^2$, 17.7 PSI AT 4600 LB/SEC
2. NOMINAL CORE BASKET RESISTANCES
   - LOCA HOLES (CROSSFLOW) R = 7.64 X 10^{-6} PSI/$(LB/SEC)^2$
   - UPPER BASKET R = 5.43 X 10^{-6} PSI/$(LB/SEC)^2$
3. CORE GEOMETRY IS NOMINAL BELOW BLOCKAGE

4. CORE BASKET GEOMETRY IS UNDISTORTED.
5. RELATIVE RESISTANCES OF THE CORE AND CORE BASKET REMAIN UNCHANGED DURING NATURAL CIRCULATION CONDITIONS COMPARED TO THEIR VALUES DURING 1/0 PUMP OPERATION.

This analysis has been reviewed/approved for inputs, method of analysis, and applicability of results

HE BABCOCK & WILCOX COMPANY
OWER GENERATION GROUP

(14)

o G. A. MEYER, MANAGER, T-H ENGINEERING UNIT

THE-79-194

rom Garn for JAW

J. A. WEIMER, T-H ENGINEERING UNIT, EXT. 3236

BOS 663.5

ust. TMI-2

File No.
or Ref.

ubj. INCORE THERMOCOUPLE ERROR EVALUATION

Date
APRIL 10, 1979

This letter to cover one customer and one subject only.

AN ANALYSIS WAS DONE TO DETERMINE THE MAGNITUDE OF INCORE THERMOCOUPLE ERRORS FOR TMI-2 PRIOR TO MARCH 28, 1979. THIS ANALYSIS WAS BASED ON A TEMPERATURE AND POWER DISTRIBUTION AT 98% AND 15% FULL POWER. THIS WORK ASSUMES THAT THE INLET AND OUTLET RTD (RESISTANCE TEMPERATURE DETECTOR) TEMPERATURES AND POWER DISTRIBUTIONS WERE CORRECT, AND INADDITION, ASSUMED A CONSERVATIVE + 3^{0}F DIFFERENCE BETWEEN THE CORE OUTLET AND VESSEL OUTLET TEMPERATURE AT 98% POWER. THIS RESULTS IN A 0.5^{0}F DIFFERENCE AT 16% POWER. MORE REALISTIC TEMPERATURE DIFFERENCES (IE. 2^{0}F AT 98% FP AND $.2^{0}$F AT 16% FP) WOULD INCREASE THE PREDICTED T-C ERRORS SLIGHTLY.

THE METHOD USED FOR THIS ANALYSIS WAS BASED ON A KNOWN BUNDLE DELTA ENTHALPY, AND FLOW RATES (FROM ONLINE COMPUTER (OLC)) FOR AN AVERAGE POWER BUNDLE (RELATIVE POWER = 1.0). THE EQUATION USED FOR THIS ANALYSIS IS:

$$H_{OUT_2} = \frac{Q_2}{Q_1} \times \frac{W_1}{W_2} \times (H_{OUT_1} - H_{IN_1}) + H_{IN_1}$$

WHERE

Q_2 = RELATIVE POWER OF BUNDLE FOR EACH CALCULATION (FROM OLC)

Q_1 = RELATIVE POWER OF BUNDLE FOR AN RPD OF 1.0

Q_1 = 1.0

W_1 = BUNDLE FLOW FOR AN RPD OF 1.0 (FROM OLC)

W_2 = BUNDLE FLOW OF BUNDLE FOR EACH CALCULATION (FROM OLC)

H_{OUT_1} H_{IN_1} = DELTA ENTHALPY FOR AN RPD OF 1.0

RPD = RELATIVE POWER DIFFERENCE (NORMALIZED TO AVERAGE ASSEMBLY POWER)

H_{OUT_2} = CALCULATED BUNDLE OUTLET ENTHALPY FOR EACH BUNDLE.

H_{OUT_2} IS THEN CONVERTED TO T_{OUT} AND COMPARED TO THE MEASURED T_{OUT} (T-C READING). THIS ANALYSIS (AT 98% AND 16% FP) WAS EXTRAPOLATED TO 1% FP.

ANY INHERENT ERRORS ON THE OLC FLOW AND RPD CALCULATIONS ARE ELIMINATED BY THIS RATIOING METHOD. THEREFORE, THE ONLY REAL UNCERTAINTY IS IN THE H_{OUT_1} AND H_{IN_1} MEASUREMENTS. THESE WERE ASSUMED CORRECT FOR THIS ANALYSIS.

THE RESULTS OF THIS ANALYSIS INDICATE AN AVERAGE + 7.94°F ERROR AT 98%, AND A + 5.59°F ERROR AT 16% POWER. THIS EXTRAPOLATES TO A + 5.16°F ERROR AT 1% POWER.

ASSUMING NO DAMAGE OCCURRED TO THE T-C'S DURING THE TRANSIENT OF MARCH 28, 1979 AT TMI-2 THESE RESULTS WOULD APPLY TO THE PRESENT T-C READINGS, THUS IT IS POSSIBLE THAT THE INCORE THERMOCOUPLE READINGS PRESENTLY BEING OBTAINED ARE HIGH BY AN AVERAGE OF 5°F.

FINALLY, THE AVERAGE T-C ERRORS WERE CALCULATED AS A FUNCTION OF DIFFERENT POSITIONS IN THE CORE. THE RESULTS SHOW NO INHERENT CORE REGION DEPENDENCY.

JAW/SGH

CC: FE UNIT MANAGERS
J. S. TULENKO
CORE HOT SPOT TASK FORCE

QA: THE METHOD AND CALCULATIONS WERE REVIEWED AND FOUND TO BE CORRECT AND CONSISTENT WITH THE STATED ASSUMPTIONS.

DATE 4/10/79

J. T. Willse
T. L. Wilson

⑮

W Bojdey
LL Losh
PE Mainola
CT Rombough

TMI-2

PRELIMINARY

Discrepancy between Thermocouples and Outlet RTD Temperature Measurements.

4/9/79

Following the LOFW transient, the average of the thermocouples at the core outlet has been higher than hotleg RTD's. This memo provides some data on the thermocouple / RTD comparison prior to the LOFW transient.

Bill Bojdey and I reviewed the thermocouple data from PDO's prior to the transient to determine if a systematic bias existed. No isothermal, zero-power readings were found. However, we did locate several sets as a function of power. These were average two ways

1) weighting the 52 readings equally
2) weighting each reading according to the number of assemblies it represents.

These results and corresponding hotleg RTD readings are given in the attached table.

Note that the thermocouples give higher readings than the RTD's in all cases.

It is reasonable to suscept this difference may be due to normal bypass flow. Assuming this is correct, we can state the following approximate formula:

$$P = (\dot{m}_T - \dot{m}_{BP}) K (T_{T/C} - T_{IN})$$

and

$$P = (\dot{M}_T) K (T_{RTD} - T_{IN})$$

where

P = Power

$\dot{m}_T$ = Total flow

$\dot{m}_{BP}$ = Bypass flow

K = Average power per flow rate per degree ΔT

$T_{T/C}$ = Average Thermocouple temperature

T_{RTD} = Hotleg RTD

T_{IN} = Inlet temperature

Setting power equal

$$(\dot{m}_T - \dot{m}_{BP})\,K\,(T_{T/c} - T_{IN}) = \dot{m}_T\,K\,(T_{RTD} - T_{IN})$$

~~Try~~ ~~Solve for~~ ~~Simplify~~ Simplifying we get

$$(1 - f)(T_{T/c} - T_{IN}) = T_{RTD} - B\,T_{IN}$$

where $f = \frac{\dot{m}_{BP}}{\dot{m}_T}$

~~a~~ and

~~$T_{out} - T_{RTD} = f\ T_{T/c} - T$~~

$$T_{T/c} - T_{RTD} = f(T_{T/c} - T_{IN})$$

Clearly if the system were isothermal, (as at zero power) then $T_{T/c} = T_{RTD} = T_{IN}$. Moreover, since the ~~[illegible]~~ ΔT is proportional to power, the following ~~instance~~ should be true:

$$T_{T/c} - T_{RTD} = \frac{f \; \text{[illegible]} \; P}{K(\dot{m}_T - \dot{m}_{BP})}$$

The plot of $\Delta T = T_{T/c} - T_{RTD}$ in the attached figure illustrates that, [illegible] extrapolating to $P=0$, the ΔT is not zero. This suggests a possible bias, prior to the transient, of + 5°F in the thermocouple readings. No explanation for the cause of the bias was found

Table: Thermocouple / RTD Comparison

				OUTLET			
	Date	T*	T**	T(A)	T(B)	Power	ΔT
1	09/19/78	592.9	593.2	585.97	587.32	16%	6.26
2	9/30/78	599.14	600.44	591.96	592.55	91	7.64
30	3/13/79	612.02	619.76	606.10	606.37	98%	10.78
31	03/19/79	612.2	630.1	606.54	606.66	98%	10.70

* weighted average

** straight average

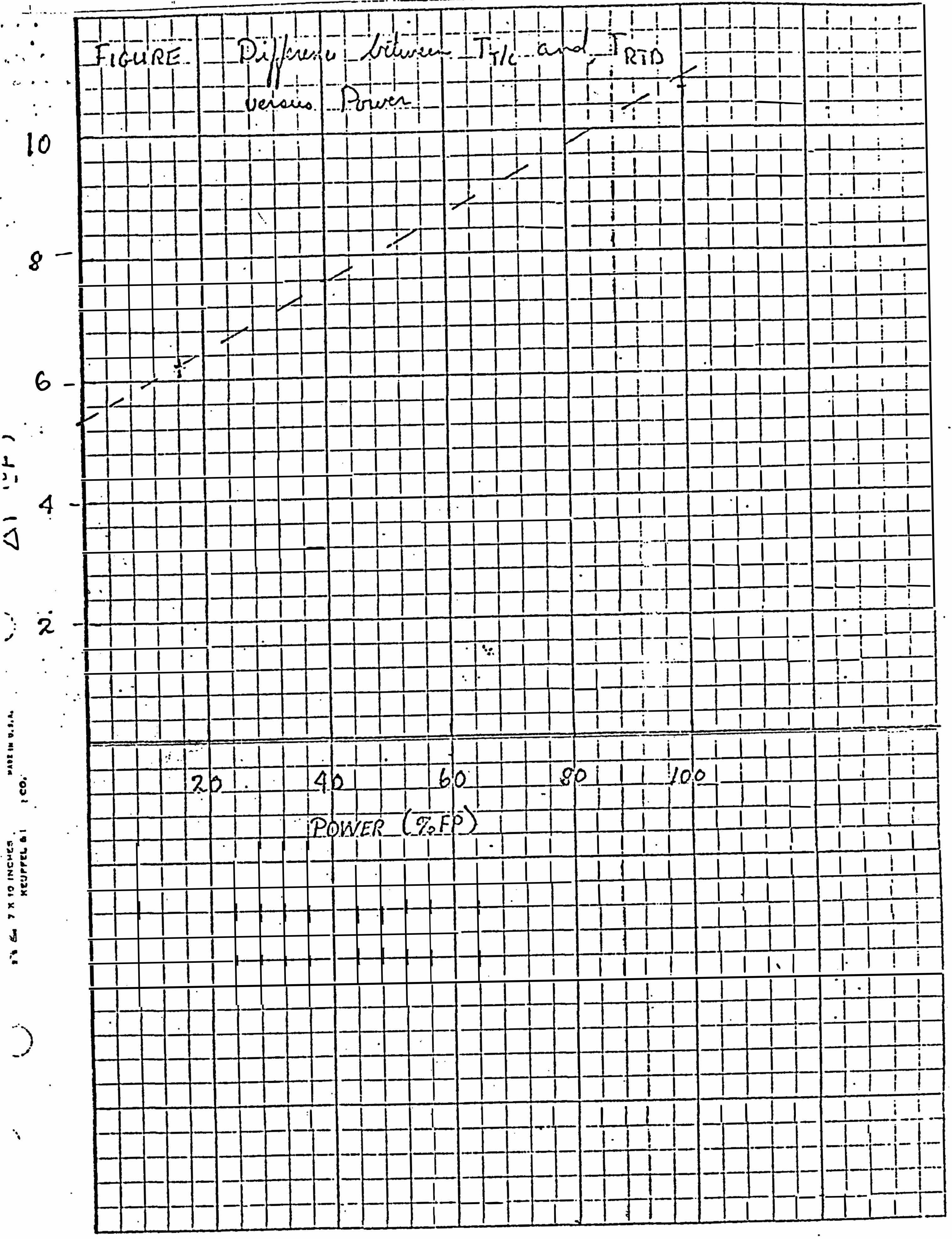


FIGURE Difference between $T_{T/C}$ and T_{RTD} versus Power

THE BABCOCK & WILCOX COMPANY
POWER GENERATION GROUP PRELIMINARY

16

To
G.A. MEYER - MANAGER, THERMAL HYDRAULIC ENGINEERING

From
R.M. HIATT - THERMAL HYDRAULIC ENGINEERING

THE-79-191

BDS 663-5

Cust.
TMI-2

File No.
or Ref.

Subj.
LYNX1 MODEL FOR TMI-2 BLOCKAGE STUDY

Date
APRIL 10, 1979

This letter to cover one customer and one subject only.

THE OBJECTIVE OF THIS WORK WAS TO DEVELOP A METHOD FOR ANALYZING LOCAL COOLANT CONDITIONS FOR TMI-2 DURING SELECTED TIMES OF THE RECENT ACCIDENT. AN EQUALLY IMPORTANT CONSIDERATION WAS THE DETERMINATION OF LOCAL COOLING CAPABILITY FOR THE BLOCKED CORE UNDER NATURAL CIRCULATION.

FIGURE 1 ILLUSTRATES THE NODING SCHEME FOR THE SIMPLIFIED CORE MODEL. THE TWELVE CHANNELS WERE DESIGNED TO SEGMENT THE CORE INTO AREAS THAT IT WAS BELIEVED EXPERIENCED MAJOR DAMAGE, SOME DAMAGE AND POSSIBLY NO DAMAGE.

LYNX1 PERFORMED WELL WITH THE MODEL FOR A CLEAN CORE. HOWEVER, WE WERE NOT ABLE TO ACHIEVE THE HIGH PRESSURE DROP (18 PSI) ACROSS THE CORE WITH THE FLOW RATE FOR ONE PUMP OPERATION AND A BLOCKED CORE CONDITION PREDICTED BY CONTROL ANALYSIS. THEY PREDICTED A FLOW OF 4500 LB/SEC FOR CORE PLUS CORE BARREL-CORE BAFFLE ANNULUS FOR A BLOCKED CORE CONFIGURATION. SINCE LYNX1 STRUGGLED FOR CONVERGENCE DUE TO A SENSITIVITY OF THE CODE TO THE BLOCKAGE MODEL, A NUMBER OF MODELING SCHEMES WERE TRIED WITH VARYING DEGREES OF SUCCESS. SEVERE BLOCKAGES AT EACH SPACER GRID (K=30-35), AN INCREASE IN WETTED PERIMETER FOR ALL CHANNELS, AND SLIGHT VARIATION IN RESISTANCE FROM CHANNEL TO CHANNEL WERE ITERATING PARAMETERS. THE BEST ESTIMATE OF LOCAL FLOW BEHAVIOR OBTAINED TO DATE ACHIEVED AN UNRECOVERABLE

PRESSURE DROP OF ABOUT 7 PSI FOR 4722 LB/SEC CORE FLOW. THE BEST ESTIMATE FROM CONTROL ANALYSIS AT PRESENT IS 18 PSI FOR 4500 LB/SEC FLOW THROUGH THE CORE PLUS CORE BARREL-CORE BAFFLE ANNULUS.

A CHATA CASE WAS MODELED TO GIVE THE FLOW SPLIT BETWEEN THE CORE AND THE CORE BARREL-CORE BAFFLE ANNULUS. IT WAS ESTIMATED THAT 78% OF THE 4500 LB/SEC PREDICTED BY CONTROL ANALYSIS FLOWS THROUGH THE CORE WHICH AGREES WELL WITH CONTROL ANALYSIS ESTIMATES. THUS, BEST ESTIMATES TO DATE SHOW THAT 3510 LB/SEC IS FLOWING THROUGH THE CORE WITH 1 PUMP OPERATION. ONE IMPORTANT FACT THAT WAS EVIDENT FROM AN ENERGY BALANCE ON THIS FLOW RATE IS THAT THE INDICATED THERMOCOUPLE ΔT IS NOT POSSIBLE CONSIDERING A 4-5 MW_T DECAY HEAT RATE UNLESS THE THERMOCOUPLES ARE MEASURING LOCAL EFFECTS, SUCH AS AGGLOMERATIONS OF PELLETS NEAR THE THERMOCOUPLES. THIS APPEARS TO DISCREDIT THE THERMOCOUPLES, THEREFORE, SOME DISCRETION IS NECESSARY IN THE INTERPRETATION OF THIS DATA.

IN CONCLUSION, ALTHOUGH LYNX1 MODELING HAS NOT BEEN SUCCESSFUL IN MATCHING FLOW AND EXIT PRESSURE AT THIS TIME FOR A BLOCKED CORE WITH ONE PUMP OPERATION, IT IS BELIEVED THAT AN ACCEPTABLE MODE CAN BE DEVELOPED. THE ADVISABILITY OF ADDITIONAL WORK IN THIS AREA DEPENDS ON THE WORK SCOPE OF FUTURE WORK ON THE TMI-2 ACCIDENT. FROM PAST EXPERIENCE, THE MODEL DEVELOPMENT WILL NOT BE QUICK BUT COULD REQUIRE A MONTH'S EFFORT.

RMH/FFA

CC: F.E. UNIT MGRS.
J.S. TULENKO
CORE HOT SPOT TASK FORCE

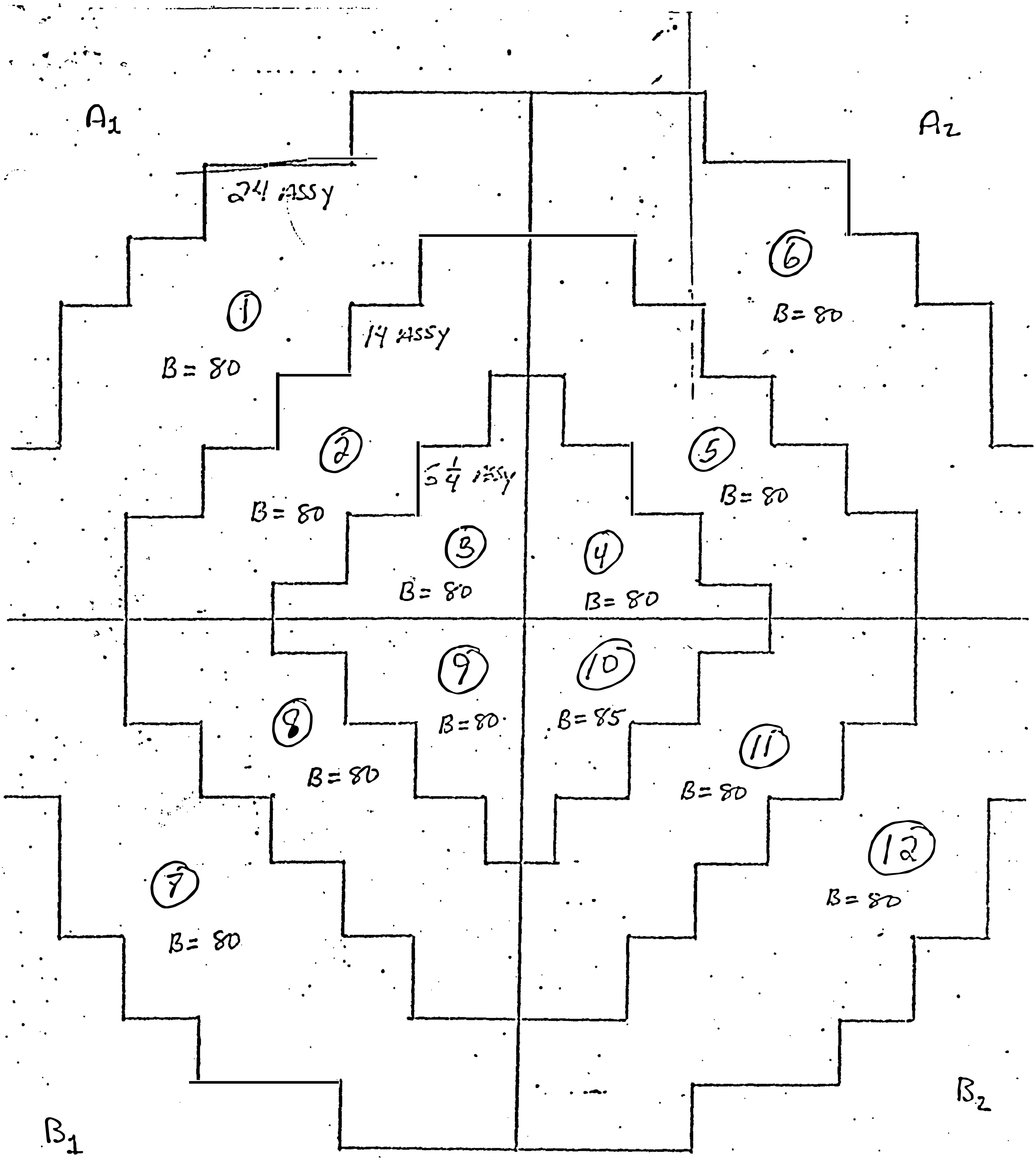


Figure 1 LYNX1 model (B= approximate blockage-%)

THE BABCOCK & WILCOX COMPANY
POWER GENERATION GROUP

17

To
G. A. MEYER, MANAGER, T-H ENGINEERING UNIT

THE-79-190

om
Paul J Henningson
P. J. HENNINGSON, T-H ENGINEERING UNIT, EXT. 3515

12E12.13

BDS 663.5

Cust.
TMI-2

File No. 86-1100502-00
or Ref. 660-021A

Subj.
DAMAGE MODEL - FLUIDIZED BED

Date
APRIL 10, 1979

This letter to cover one customer and one subject only.

REFERENCES: 1) MEMO, CORE CONDITION TASK FORCE TO J. S. TULENKO, "CURRENT ASSESSMENT OF CORE CONDITION, APRIL 7, 1979 (1800)," APRIL 7, 1979 (7:48 PM)

2) PERRY'S CHEMICAL ENGINEERS HANDBOOK, FOURTH EDISON, PP. 549 - 551.

3) MEMO, P. J. HENNINGSON TO G. A. MEYER, "POSSIBLE MODE OF INCREASED T.C. READINGS," APRIL 7, 1979.

THE DAMAGED TMI-2 CORE WAS HYDRAULICALLY MODELED AS A PACKED BED. THE MECHANISM OF FUEL FAILURE WOULD RESULT IN APPROXIMATELY THIS GEOMETRY AND BE LOCATED IN THE UPPER REGION OF THE CORE. BRIEFLY THE CORE WOULD BE CONFIGURED AS UNDAMAGED FUEL UP TO A HEIGHT WITH DAMAGED FUEL (FUEL PARTICULATES AND CLADDING) ABOVE THIS RESEMBLING A POROUS MASS.

THE BASIC CONFIGURATION OF THE CORE WAS OBTAINED FROM REFERENCE 1. THE CORE WAS ASSUMED UNDAMAGED AT THE PERIPHERY WITH INCREASING FAILURE TOWARDS THE CENTER. PARTICLES OF FAILED FUEL WHICH COMPRISED THE FLUIDIZED BED WERE ASSUMED TO EVOLVE FROM THE FOLLOWING FAILURE MECHANISM:

THE FUEL CRACKED ALONG TWO PERPENDICULAR AXES LENGTHWISE AND ALONG THREE PLANES PERPENDICULAR TO ITS AXIS.

THE MASS OF FUEL WOULD THEN CONSTITUTE THE MAJORITY OF THE CONGLOMERATE WITH CLADDING FRAGMENTS ASSUMED TO HAVE A SIMILAR GEOMETRY.

A SUITABLE CORRELATION FOR PRESSURE DROP THROUGH A PACKED BED WAS OBTAINED FROM REFERENCE 2. THIS CORRELATION (ATTRIBUTED TO LEVA) WAS APPLICABLE IN THE HIGH REYNOLDS NUMBER RANGE EXISTING IN THE DAMAGED CORE (RE ∿ 10,000). IT IS IMPORTANT THAT THE RANGE OF REYNOLD'S NUMBER APPLICABILITY BE ASCERTAINED FOR A GIVEN CORRELATION. THE SENSITIVITY OF THE FRICTION FACTOR TO CHANGES IN FLOW FROM VISCOUS TO TURBULENT IN THE PACKED BED CANNOT BE NEGLECTED.

AN ATTEMPT TO MODEL THE CORE AS DEFINED IN REFERENCE 1 WAS MADE. THE LOW RESISTANCE IN THE PERIPHERAL BUNDLE CAUSED THIS METHOD TO FAIL. IT WAS THEN ASSUMED THAT FAILED FUEL (OR A CONGLOMERATE OF PARTICLES) EXISTED AT THE PERIPHERY. THE CORE TOOK ON THE FOLLOWING SHAPE:

CENTRAL BUNDLES (116)	4 FEET OF FAILED FUEL BELOW THE UPPER END FITTING (PACKED BED)
REMAINING BUNDLES	2 FEET OF FAILED FUEL BELOW THE UPPER END FITTING

THE GENERAL SHAPE AND RECOMMENDATION OF A FOUR FOOT HEIGHT WAS OBTAINED FROM REFERENCE 1.

A TRIAL AND ERROR APPROACH WAS USED. THE VOID FRACTION OF THE PACKED BED WAS VARIED AND A CORE ΔP CALCULATED. THE FINAL RESULT WAS THAT FOR THE ABOVE CONFIGURATION A CORE $\Delta P = 14$ PSI WAS OBTAINED FOR A CORE FLOW OF 13.1×10^6 LBM/HR. THE FLOW IN THE CENTRAL BUNDLES (61) WAS $.058 \times 10^6$ LBM/HR AND IN THE OUTER BUNDLES $.082 \times 10^6$LBM/HR. THIS WAS FOR A PACKED BED HEIGHT OF FOUR FEET AT THE CENTER 61 BUNDLES AND TWO FEET ON THE REMAINDER OF THE CORE. A VOID OF 60% WAS USED WHICH COMPARED WELL WITH THE 50% RECOMMENDED IN REFERENCE 1.

NO FURTHER ATTEMPTS WERE MADE TO MATCH PRESENTLY ASSUMED CORE CONDITIONS - $\Delta P \sim 16.0$ PSI, CORE FLOW $\sim 14.10^6$ LBM/HR. VARIOUS ASSUMPTIONS CAN BE MADE CONCERNING THE GEOMETRY AND MAKEUP OF THE FAILED FUEL WHICH IS ASSUMED TO RESEMBLE A PACKED BED. WHAT IS IMPORTANT IS THAT:

1) CORE CONDITIONS COULD BE APPROXIMATED WITH THE PACKED BED ASSUMPTION,
2) FAILED FUEL (OR A HIGH RESISTANCE EXISTS ACROSS THE CORE). THE FUEL AT THE PERIPHERY COULD BE UNDAMAGED WITH A LAYER OF PARTICULATES BENEATH THE CORE SUPPORT PLATE ALTHOUGH IT SEEMS UNLIKELY THAT THE MATERIAL WOULD BE THAT NON-HOMOGENEOUS.

FURTHERMORE, IF THE FAILURE MODE OF THE CORE DESCRIBED IN REFERENCE 1 IS ASSUMED, THEN IT APPEARS THAT THE THERMOCOUPLES COULD BE SURROUNDED BY UO_2. THIS WOULD EXPLAIN THEIR HIGH READINGS. THE EFFECT OF UO_2 SURROUNDING THE THERMOCOUPLE WELL WAS DESCRIBED IN REFERENCE 3.

PJH/sgh

QA: THE METHODS PRESENTED HAVE BEEN REVIEWED FOR APPLICABILITY AND THE CALCULATIONS SPOT-CHECKED FOR ACCURACY AND CONSISTENCY. THE METHOD IS DEEMED APPROPRIATE FOR THIS PARTICULAR APPLICATION.

JCMoxley by [illegible] DATE 4/10/79

CC: J. H. JONES
A. B. JACKSON
J. C. MOXLEY
D. V. DEMARS
B. J. BUESCHER
H. W. WILSON
R. A. KING
D. C. SCHLUDERBERG
G. S. CLEVINGER

THE BABCOCK & WILCOX COMPANY
POWER GENERATION GROUP

18

BDS 663-5

To J. B. ANDREWS

From CORE CONDITION TASK FORCE

Cust.

File No. or Ref.

Subj. ESTIMATE OF LOOSE CORE DEBRIS VOLUME (4/9/79 - 2000)

Date TIME:
APRIL 9, 1979

This letter to cover one customer and one subject only.

ATTACHED IS AN ESTIMATE OF THE AMOUNT OF DEBRIS AVAILABLE FOR CORE BLOCKAGE AND ITS POTENTIAL DISTRIBUTION. THIS IS TO AID IN THE OVERALL ASSESSMENT OF CORE BLOCKAGE.

MOST OF THE INFORMATION IN FORMING THIS ASSESSMENT IS STILL PRELIMINARY AND IS BASED ON OUR BEST ESTIMATE OF CORE DAMAGE.

CC: P. HENNINGSON
CORE CONDITION TASK FORCE
J. S. TULENKO

ATTACHMENT

CORE BLOCKAGE ESTIMATE

BELOW IS AN ESTIMATE OF THE LOOSE MATERIAL AVAILABLE TO CONTRIBUTE TO CORE BLOCKAGE.

1. ASSUME 30% OF TOTAL Zr CORE INVENTORY IS OXIDIZED PRODUCING ZrO_2

45,000 LB	TOTAL FUEL CLAD INVENTORY
7,900 LB	OTHER Zr INVENTORY
52,900 LB	TOTAL

- 15870 LB OF Zr IS OXIDIZED WITH A 1.6 BULK VOLUME INCREASE, FORMING 63 FT^3 OF ZrO_2.

2. THE TOP 30% (46 IN.) OF THE CLAD OXIDIZES EXPOSING 41" OF UO_2 TO THE COOLANT.

- TOTAL UO_2 EXPOSED IS 57,400 LBS OR 94 FT^3.
- ASSUME AN ADDITIONAL 10 FT^3 OF MATERIAL IS EXPOSED FROM FUEL ROD (SPRINGS, END PLUGS, ETC.).

3. BASED ON THE ABOVE THE EXPOSED MATERIAL AVAILABLE FOR CORE BLOCKAGE IS:

ZrO_2	63 FT^3
UO_2	94 FT^3
OTHER	10 FT^3
	167 FT^3

4. OF THIS, SOME IS CAPABLE OF BEING MOVED BY FLOW

	VOLUME FT^3		
	AVAILABLE	MOBILE	IMMOBILE
ZrO_2	63	45 (FLAKES & DUST)	18 LARGER FLAKES OR ON RODS
UO_2	94	14 (<1/16" SIZE)	80 (>1/16")
OTHER	10	----	10
	167	59	108

THE SMALLER PARTICLES MAY EXIT AND MOVE WITH THE FLOW AND MAY RE-DEPOSIT IN THE CORE OR SETTLE OUT ELSEWHERE IN THE SYSTEM.

EQUIVALENT FLOW BLOCKAGE

- ASSUME THE EQUIVALENT CORE FLOW AREA (10.6 FT DIA), IS 88 FT^2
- TOTAL EQUIVALENT DEPTH OF BLOCKAGE IS

  $\frac{167}{88}$ = 1.90 FT ASSUMING SOLID MATERIAL
- ASSUME 1.5 VOLUME INCREASE FOR PACKING, THEN THE EQUIVALENT TOTAL DEPTH IS 1.90 × 1.5 = 2.85 FT
- .IT IS EXPECTED THAT THE DAMAGE WILL BE GREATER AT THE CENTER THAN AT THE CORE PERIPHERY (SEE NEXT SECTION).

CORE DAMAGE DISTRIBUTION

THE CORE DAMAGE WILL BE MORE SEVERE IN THE CENTER OF THE CORE THAN ON THE PERIPHERY. THIS RESULTS FROM THE CORE DECAY HEAT POWER DISTRIBUTION WHICH CLOSELY FOLLOWS THE CORE POWER DISTRIBUTION PRIOR TO SHUTDOWN (SEE FIG. 1). THIS WILL RESULT IN CORE DAMAGE DISTRIBUTION AS SHOWN IN FIGURE 2. THE FUEL RODS IN PERIPHERAL ASSEMBLIES MAY BE RELATIVELY INTACT WHILE THE CENTER ASSEMBLY IS PROBABLY SEVERELY DAMAGED, POSSIBLY TO THE CENTER OF THE CORE. THE CENTER ASSEMBLIES MAY HAVE VIRTUALLY NO RECOGNIZABLE ARRAY IN THE UPPERMOST GRID SPANS.

Fig. 1

P.O.A 3/19/79

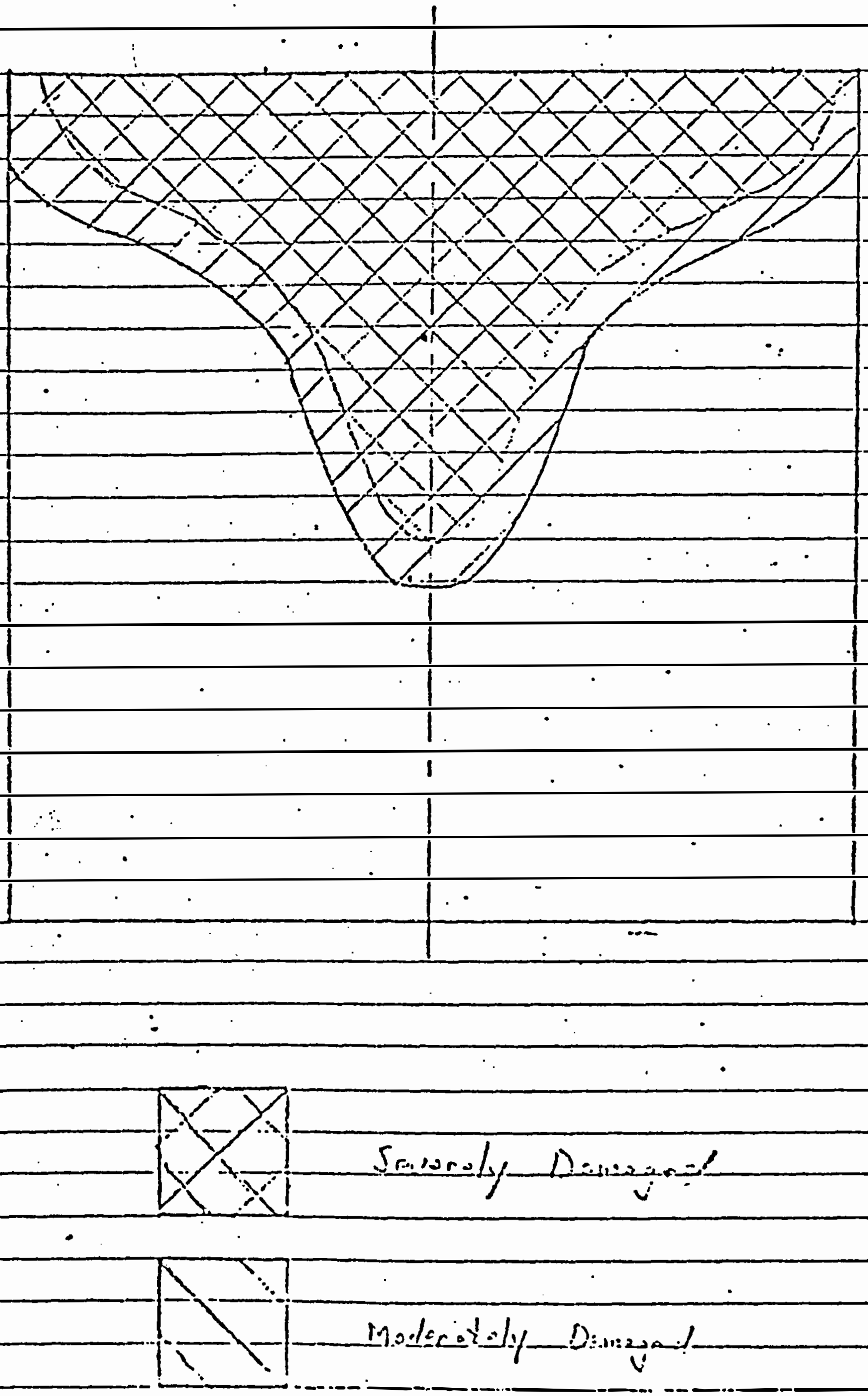
```
****  FULL CERTIFICATION  ****      ****  FULL CERTIFICATION  ****      ****  FULL CERTIFICATION  ****
(FLAMS V2.1 CREATED 12/26/78) - TMI-LGO -  03/29/79  CASE 11.4 (6,8,32)     TMI-2 FLAN.         79.6EFPD  225.5MWD AF
                              RELATIVE POWER DENSITY
 I/J      8          9          10         11         12         13         14         15

 H    ①1.4535     1.2727  *  1.2033  *  1.1757     1.0776  *  1.1615   ②1.2703     .8796
 K     1.2727   ②1.2314  *  1.2050     1.1464  *  1.1052  *  1.0530  *  1.0779   ④.8196
 L     1.2050  *  1.2050     1.1567  *  1.1272  *   .9974  *  1.0430  *  1.1037   ⑤.6124
 M     1.1794     1.1474  *  1.1257  ③1.0555  *  1.0165      .9222  *   .8357
 N     1.0976  *  1.1052  *   .9594  *  1.0135      .9155  ④   .8112   ⑤.5319
 O     1.1474  *  1.0570  *  1.0442      .9272  *   .8102   ⑤.5157
 P   ⑤1.2703     1.0769  *  1.1037      .8331   ⑤.5319
 R      .8796 ④   .8196   ⑤.6196
```

group	RPD	# Assy's
1	1.45	1
2	1.23	28
3	1.09	88
4	.86	40
5	.57	20

Fig. 2

Core Damage Distribution



THE BABCOCK & WILCOX COMPANY
POWER GENERATION GROUP

PRELIMINARY

19

To
DISTRIBUTION

From
P. S. BARTELLS, TECHNICAL STAFF

BDS

Cust.
TMI-2 Part of

File No.
or Ref.

Subj.
CORE PRESSURE DROP FOR NATURAL CIRCULATION CALCULATION

Date
APRIL 9, 1979 6:

This letter to cover one customer and one subject only.

DISTRIBUTION

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C.E. Parks
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R.H. Stoudt
J.D. Carlton
J.A. Castanes
E.F. Dowling
J.M. Knoll
P.A. Treventi

REFERENCE: MEMO, SAME SUBJECT, APRIL 6, 1979.

ADDITIONAL INFORMATION HAS BEEN OBTAINED FROM TMI WHICH FURTHER SUBSTANTIATES THE CONCLUSIONS PRESENTED IN THE REFERENCED MEMO. BASED ON THIS INFORMATION, AND THE FLOW SPLITS PREDICTED BY THE PUMP CODE, AN 18 PSIA DROP ACROSS VESSEL DOWNCOMER AND CORE (AT PRESENT CONDITIONS AND CORE FLOW OF 4500 lbm/sec) IS INDICATED.

less 27% Core [illegible]

ATTACHMENT 1 IS THE TABULATION OF RC LOOP FLOW TRANSMITTER DIFFERENTIA PRESSURE SIGNALS OBTAINED AT 4:00 a.m., APRIL 9. ATTACHMENT 2 IS THE CONVERSION OF THE TRANSMITTER VOLTAGE MEASUREMENTS TO LOOP FLOWS.

AS A FURTHER CHECK ON THE RESULTS, I HAVE ASKED MIKE KNOLL OF CONTROL ANALYSIS TO ANALYZE THE SAME CASE USING THE SPLIT CODE. IT IS HOPED THIS INFORMATION WILL BE AVAILABLE BY LATER TOMORROW AFTERNOON.

AT THIS POINT I WOULD LIKE TO ACKNOWLEDGE THE EFFORTS OF MIKE KNOLL AND PHIL TREVENTI IN PERFORMING THIS ANALYSIS (WITHOUT WHICH I WOULD STILL BE SETTING UP THE INPUT).

IF ANY ADDITIONAL INFORMATION ON PLANT STATUS (WHICH COULD PROVIDE A FURTHER CHECK ON THESE PREDICTIONS) IS KNOWN TO BE AVAILABLE, PLEASE CONTACT ME IMMEDIATELY.

PSB/DH

Attachment 2

Conversion of transmitter voltage measurements to flow –

Assumptions:

1. 10 volts = 80 x 10⁶ lbm/hr.
2. 10 volts = 895.8" ΔP

Method 1:

$$W_a = W_d \left(\frac{v_d}{v_a}\right)^{.5} \left(\frac{V_m}{10}\right)^{.5}$$

W_a = actual flow, lb/sec.

W_d = flow at voltage = 10 volts.

v_d = specific volume at W_d conditions

v_a = specific volume at actual conditions

V_m = measured transmitter voltage, volts.

now:

v_a(460 psia, 280°F) = .01724 ft^3/lb

v_d(2200 psia, 600°F) = .0232 ft^3/lb

W_d = 80 x 10⁶ lbm/hr = 22222 lbm/sec

Therefore:

$$W_a = 25778 \left(\frac{V_m}{10}\right)^{.5} \text{ lbm/sec}$$

Method 2:

If: $\Delta P_d = 895.8'' = 10$ volts.

$$\Delta P_a = \left(\frac{\Delta P_d}{10}\right) V_m$$

then:

$$W_a = W_d \left(\frac{\rho_a \, \Delta P_a}{\rho_m . \Delta P_m}\right)^{.5}$$

for:

$W_d = 22222$ lbm/sec

$\rho_a = 1/.01724 = 58.005$ lbm/ft^3

$\rho_d = 1/.0232 = 43.103$ lbm/ft^3

and:

$$W_a = 25778 \left(\frac{\Delta P_a}{\Delta P_m}\right)^{.5}$$

Results:

Following are the results of the voltage measurements;

Hot Leg A

RPS CHANNEL	V_m (Volts)	ΔP (inches)	W (lbm/sec)
A	1.9998	179.14	11528
C	2.0982	187.96	11808
D	1.7411	155.97	10756

Hot Leg B

RPS CHANNEL	V_m (Volts)	ΔP (inches)	W (lbm/sec)
A	-.7061	63.25	-6850
C	-.7141	63.97	-6889
D	-.6947	62.23	-6794

Loop A Average = 11364 lbm/sec

Loop B Average = -6844 lbm/sec

This calculation has been reviewed and is correct ____________

THE BABCOCK & WILCOX COMPANY
POWER GENERATION GROUP

To
DISTRIBUTION

From
PS BARTELLS, TECHNICAL STAFF

BDS 653

Cust.
TMI-2 Part of

File No.
or Ref.

Subj.
CORE PRESSURE DROP FOR NATURAL CIRCULATION CALCULATIONS

Date
APRIL 6, 1979 10:10 PM

This letter to cover one customer and one subject only.

DISTRIBUTION

EA WOMACK
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GF MALAN
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DH ROY
BE BINGHAM
JD CARLTON

An analysis (using the PUMP code) was performed earlier this week to estimate core flow blockage. Vessel (i.e. core and bypass) flow resistance was varied over a wide range and the change in loop flow rates, core flow rates and vessel delta P were calculated. The results are tabulated below: (* means unblocked core)

Rv	ΔP_V	W_V	W_{HLA}	W_{HLB}
1.712*	1.6*	10170*	12810*	-2643*
3.5	2.98	9450	12690	-3244
7.0	5.05	8596	12500	-3900
10.0	6.45	8095	12380	-4281
15.0	8.22	7443	12200	-4752
30.0	11.89	6291	11870	-5580
60.0	15.79	5136	11510	-6376

Where:

Rv = downcomer + core + bypass flow resistance, $(psia)/(lbm/sec)^2 \times 10^8$
ΔPv = pressure drop across core + bypass, psia
Wv = core + bypass flow rate, lbm/sec
W_{HLA} = hot leg flow rate, active loop, lbm/sec
W_{HLB} = hot leg flow rate, idle loop, lbm/sec

an be seen from the tabulated results, the active loop flow is not a ong function of the vessel resistance. This is due to the high reverse ows through the idle pumps. However, the reverse flow through the idle loop is a strong function of the vessel resistance. Prior to this afternoon, I had been under the impression that no method existed for calculation of reverse flow in the idle loop. Recent information from the I&C group shows this not the case. As early as last weekend they estimated the reverse flow to be -14.5% which translates to -6444 lb/sec. A further check today results in an estimate of -6797 lb/sec. Based on the tabulated data, the vessel pressure drop is <u>at least</u> 16 psia.

Additional evidence to back up this is the indicated flow in the active loop which is consistently indicating 49-50% of nominal which translates to approx. 11,000 lbm/sec.

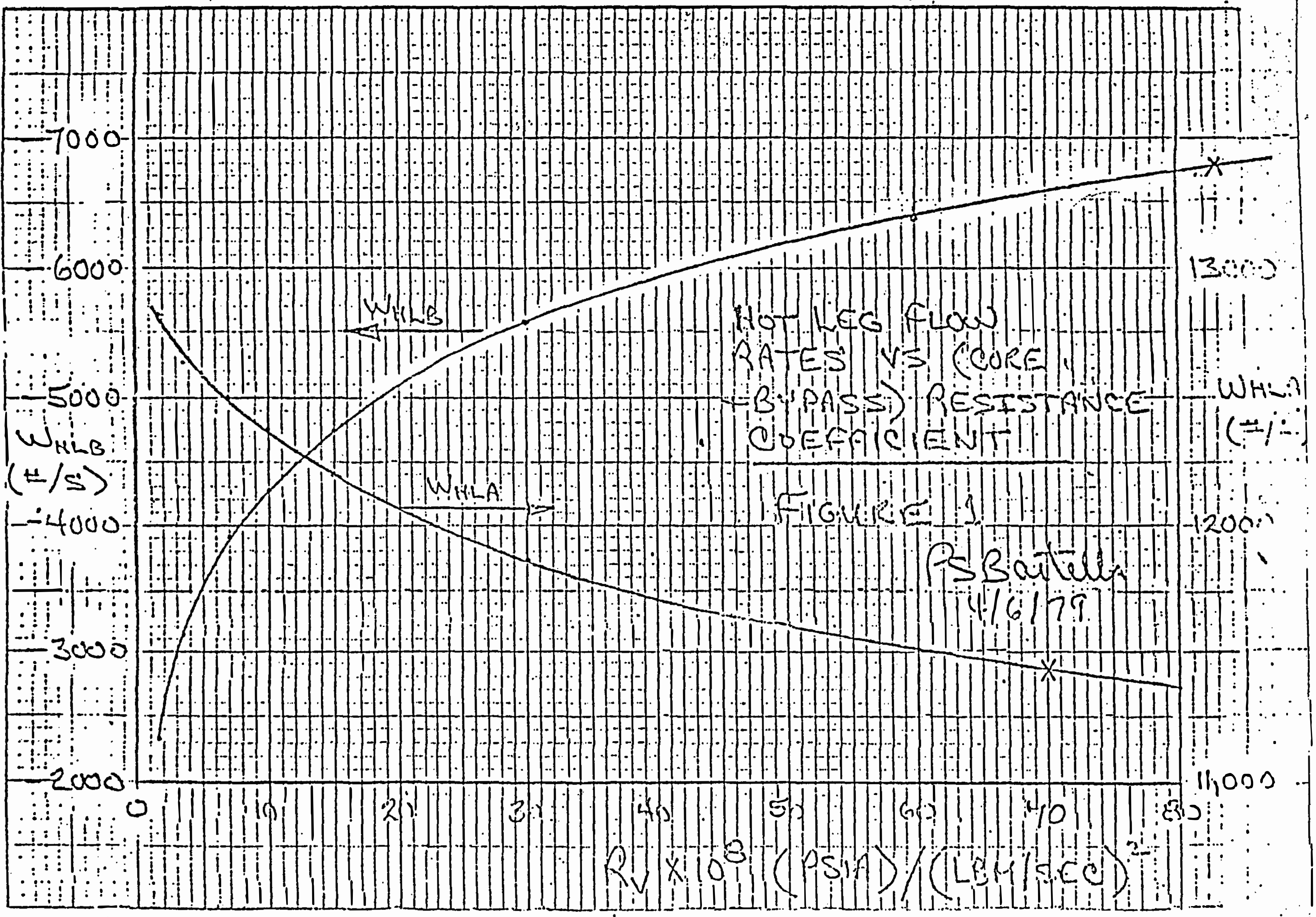
Separate calculations by Jim Veenstra and Larry Losh (see attachments) on 4/4/79 place measured flow in the active loop at 88,350 GPM (based on Gentille delta P = 173"), which is a flow rate of 11,445 lbm/sec.

The attached figures indicate that the core and bypass pressure drop is between 16.7 and 17.7 psia. Allowing for conservatism, I would recommend the use of <u>18 psia for natural circulation calcuations</u>. Additionally, I would estimate available core + bypass flow at present conditions to be 4600 to 4800 lbm/sec.

I have asked John Castanes to obtain up-to-date readings on Gentille delta P's as a further check on this analysis. He has been in contact with BMCo and they feel that the transmitter accuracy is very good.

ATTACHMENT
PSB:jws

B Bartells



HOT LEG FLOW RATES VS (CORE + BYPASS) RESISTANCE COEFFICIENT

FIGURE 1

PS Bartell
4/6/79

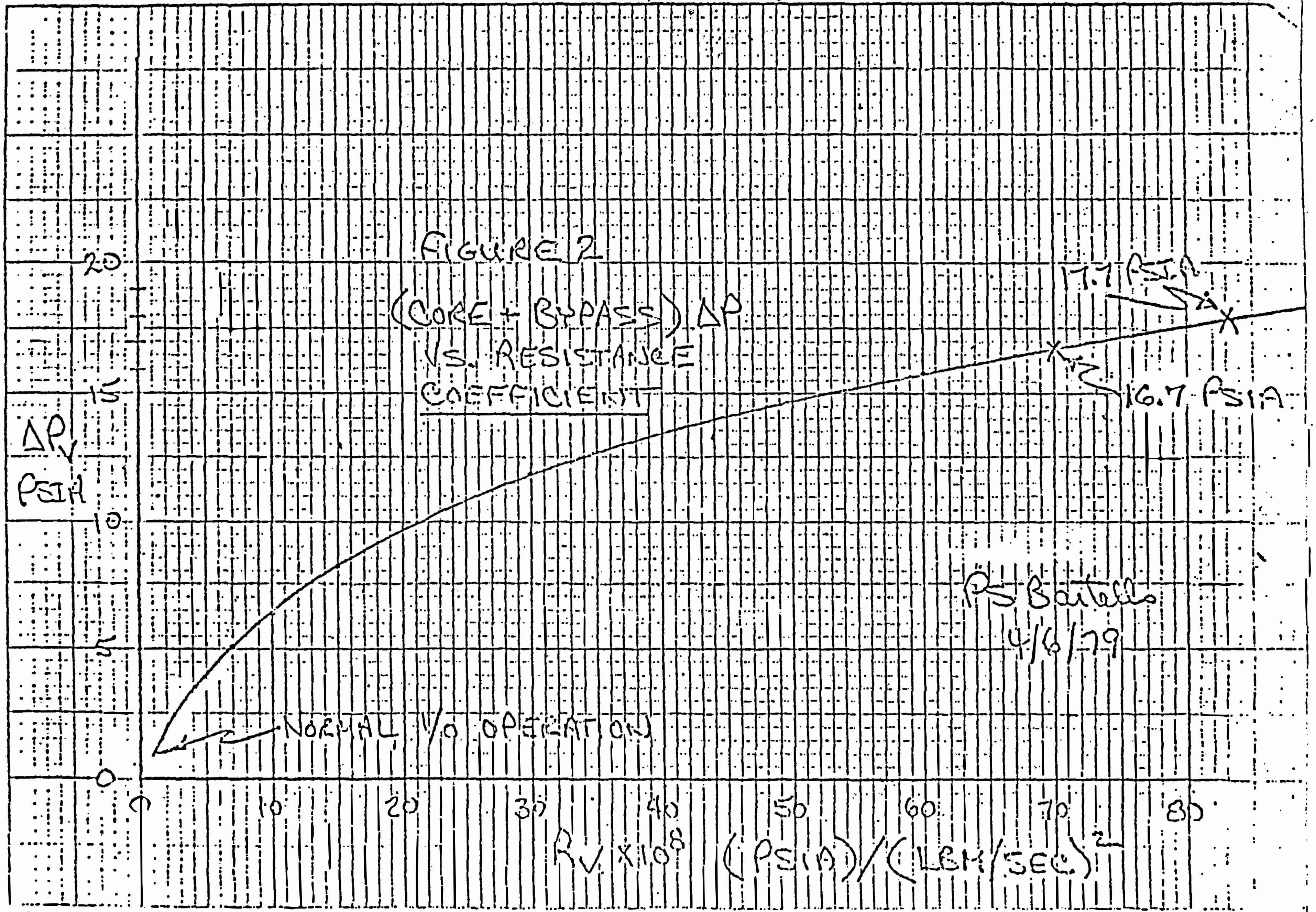
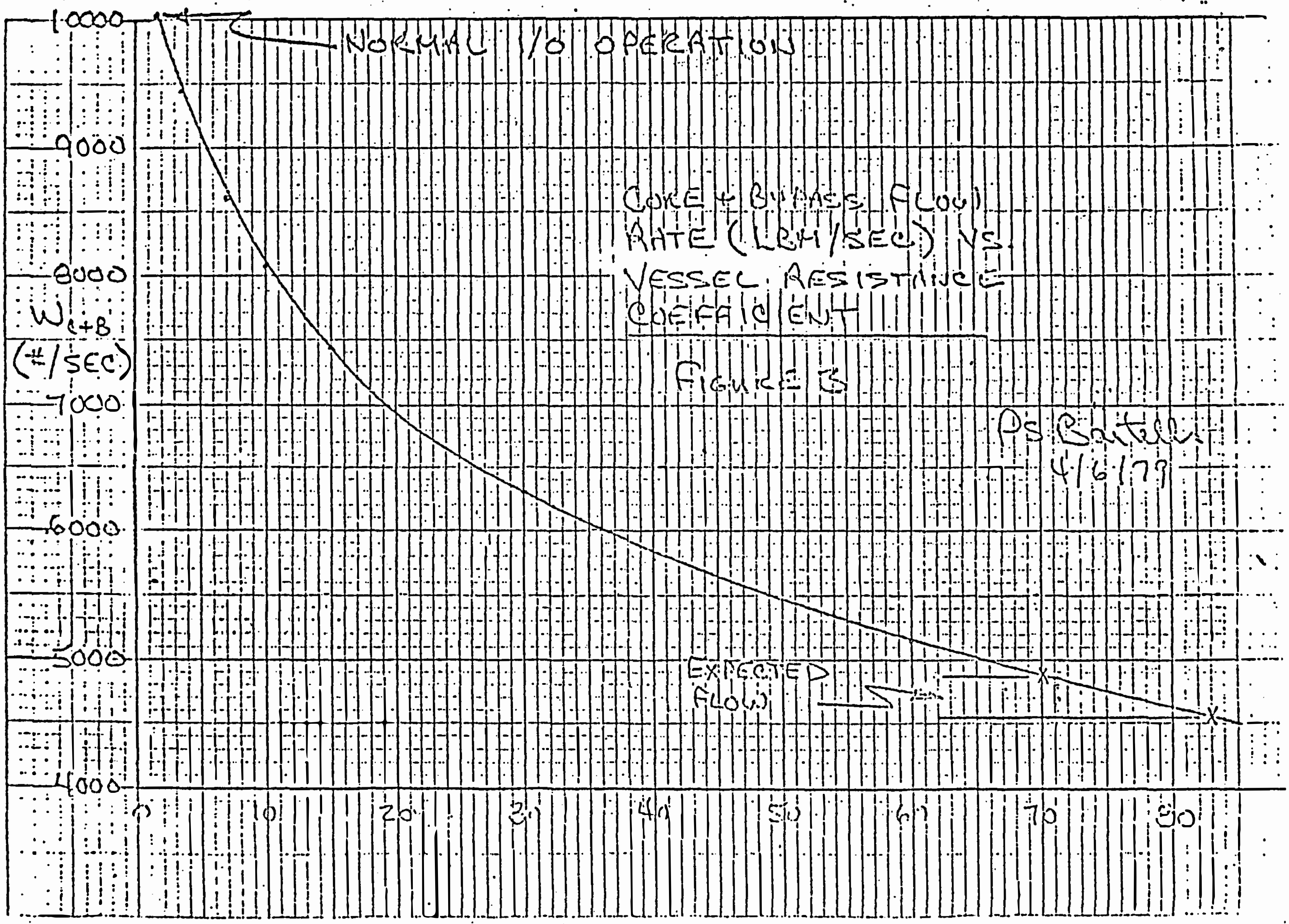


FIGURE 2
(CORE + BYPASS) ΔP
VS. RESISTANCE
COEFFICIENT

PS Baitells
4/6/79



CORE + BYPASS FLOW RATE (LBM/SEC) VS. VESSEL RESISTANCE COEFFICIENT

FIGURE 5

PS Baitell
4/6/79

THE BABCOCK & WILCOX COMPANY
POWER GENERATION GROUP

20

BDS 663.5

To J. S. TULENKO - MANAGER, FUEL ENGINEERING

From R.V. DE MARS - CCTF LEADER R.V. DeMars
CORE CONDITION TASK FORCE

Cust.

File No. or Ref.

Subj. CURRENT ASSESSMENT OF CORE CONDITION 4/7/79 (1800)

Date 4-7-79 - 7:48 p.m.

This letter to cover one customer and one subject only.

ATTACHED IS THE CURRENT ASSESSMENT OF THE CORE CONDITION BASED ON INFORMATION AVAILABLE AS OF 4/7/79. MOST OF THE INFORMATION USED IN FORMING THIS ASSESSMENT, IS STILL PRELIMINARY AND REQUIRES VERIFICATION AND DOCUMENTATION. THE MOST SIGNIFICANT UNCERTAINTY IS THE TIME AND TEMPERATURE CONDITIONS PRESENT DURING THE CORE UNCOVERY.

THE CORE CONDITION TASK FORCE CONSIDERS THIS A PRELIMINARY BUT REALISTIC ESTIMATE BASED ON VARIOUS SOURCES OF INFORMATION INCLUDING INPUT FROM THE EPRI TASK FORCE ON FUEL DAMAGE ASSESSMENT.

AS FURTHER INFORMATION BECOMES AVAILABLE THE ASSESSMENT WILL BE UPDATED ACCORDINGLY.

RKK:dww

ATTACHMENT
CC: D.H. ROY
E.A. WOMACK
C.D. MORGAN
XC: CORE CONDITION TASK FORCE

INTRODUCTION

THE PHYSICAL CONDITION OF THE CORE IS BASED ON THE FOLLOWING POSTULATED SEQUENCE OF EVENTS. THE CORE WAS UNCOVERED ~11 FT DURING THE FIRST 15 MINUTES FOLLOWING THE SECOND PUMP TRIP. (SEE FIG. 1) THE CORE WAS THEN UNCOVERED 8 FT FOR 105 MINUTES. FOR THE REMAINDER OF THE TRANSIENT, THE CORE WAS ASSUMED TO BE QUASI-COVERED TO THE POINT THAT NO SIGNIFICANT OXIDATION OCCURRED. DURING THE INITIAL UNCOVERING, THE CLADDING WOULD FAIL NEAR THE TOP OF THE ROD DUE TO STRESS RUPTURE. DEPENDING ON THE HEATING RATES, THESE FAILURES WOULD HAVE OCCURRED BETWEEN ~1200-1650°F. THIS MAY PRECLUDE INITIAL FAILURE BY EUTECTIC FORMATION BETWEEN INCONEL GRID AND ZIRCALOY RODS. CLADDING STRAINS DUE TO HIGH-TEMPERATURE DEFORMATION PRIOR TO RUPTURE COULD APPROACH 35%. DURING THE HOLD TIME SUBSEQUENT TO THE RUPTURE, THE CLADDING OXIDIZED SEVERELY, FORMING ZIRCONIUM OXIDE AND RELEASING HYDROGEN GAS. THE DEGREE OF OXIDATION WILL VARY WITH THE POWER, HAVING BOTH AXIAL AND RADIAL DISTRIBUTION. THE DEGREE OF OXIDATION ALONG THE LENGTH OF A ROD COULD VARY FROM NEGLIGIBLE AT THE BOTTOM TO 100% AT THE HOTTEST REGION NEAR THE TOP OF THE ROD.

BASED ON EVALUATION AND INTREPRETATION OF AVAILABLE INFORMATION AS OF (4/7/79) IT IS POSTULATED THAT THE CURRENT CORE CONDITION IS:

1. <u>FUEL ROD PRESSURE BOUNDARY</u>

   APPROXIMATELY 90% OF THE FUEL RODS MAY HAVE PERFORATED CLADDING, ALLOWING RELEASE OF HELIUM AND VOLATILE FISSION PRODUCTS.

2. <u>FUEL ROD STRUCTURAL INTEGRITY</u>

   MANY OF THE INTERIOR FUEL ASSEMBLIES MAY VIRTUALLY HAVE NO RECOGNIZABLE FUEL ROD ARRAY BETWEEN THE UPPER END FITTING AND FIRST (TOP) INTERMEDIATE SPACER GRIDS. IN SOME ASSEMBLIES THIS CONDITION MAY EXIST TO A LESSER EXTENT AS FAR DOWN AS THE SECOND OR THIRD INTERMEDIATE GRIDS. MOST OF THE PERIPHERAL RODS AND THE LOWER PORTION OF MOST RODS WILL BE OXIDIZED BUT NOT TO AN EXTENT TO SIGNIFICANTLY AFFECT STRUCTURAL INTEGRITY.

3. FUEL ASSEMBLY STRUCTURE

THE INTERMEDIATE INCONEL SPACER GRIDS SHOULD BE CLOSE TO THEIR ORIGINAL AXIAL POSITION. THE UPPER END GRID AND END FITTING IN MANY OF THE INTERIOR ASSEMBLIES MAY HAVE LITTLE STRUCTURAL SUPPORT. THE FIRST AND SECOND INTERMEDIATE SPACER GRIDS IN THESE INTERIOR ASSEMBLIES ARE LIKELY TO BE SUPPORTED AXIALLY FROM BELOW BY BADLY OXIDIZED GUIDE TUBES AND POSSIBLY FUEL RODS. THE REMAINING LOWER GRIDS ARE EXPECTED TO HAVE STRUCTURAL SUPPORT FROM THE DEGRADED BUT REMAINING GUIDE TUBES AND FUEL RODS.

4. ZIRCALOY COMPONENT MATERIAL CONDITION

THE ZIRCONIUM OXIDE (ZrO_2) PRODUCED BY THE OXIDATION OF THE ZIRCALOY COMPONENTS HAS RELATIVELY LOW DENSITY AND CAN RANGE IN FORM FROM SMALL PARTICLES OF A FEW MILS IN SIZE, TO IRREGULAR SHAPED FLAKES OF A FEW MILS IN THICKNESS AND UP TO A QUARTER INCH ON A SIDE, TO VIRTUALLY INTACT TUBULAR BUT FRAGILE SEGMENTS OF CLADDING. THE PARTICLES AND FLAKES ARE LIKELY TO BE MOBILE IN MOVING WATER. THESE PARTICLES CAN BE EXPECTED TO LODGE IN THE UPSTREAM SIDE OF ANY FLOW RESTRICTION SUCH AS SPACER GRIDS. GRAVITY MAY BE SUFFICIENT TO CAUSE THE LARGER ZIRCALOY AND ZrO_2 FRAGMENTS TO SETTLE OUT ON THE DOWNSTREAM OR UPPER SIDE OF SPACER GRIDS. THE QUANTITY OF ZrO_2 AND FRAGMENTED ZIRCALOY PRODUCED DURING THE PARTIAL CORE UNCOVERY IS LARGE. EXCEPT FOR SOME RODS IN PERIPHERAL ASSEMBLIES AND THE LOWER PORTION OF MOST RODS IN ALL ASSEMBLIES, THE TEMPERATURES PROJECTED FOR THE ZIRCALOY FUEL RODS WAS SUFFICIENT TO CAUSE SIGNIFICANT OXIDATION. THUS, THE MOBILITY, QUANTITY AND ORIGIN OF ZrO_2 IS SUCH THAT LOCAL FLOW BLOCKAGE COULD BE EXPECTED TO OCCUR IN ALMOST ANY LOCATION IN THE CORE. HOWEVER, THE MOST EXTENSIVE FLOW BLOCKAGE COULD BE EXPECTED IN THE UPPER CENTRAL PART OF THE CORE, WHERE THE ZrO_2 PARTICLES COULD FURTHER RESTRICT THE GENERAL FLOW RESTRICTION CAUSED BY THE HEAVIER FUEL PARTICLES AND FUEL ROD FRAGMENTS.

5. FUEL (UO_2) CONDITION

THE FUEL RELEASED FROM THE DETERIORATED CLADDING IS VERY DENSE. THE ORIGINAL SIZE OF PELLETS IS APPROXIMATELY 3/8 INCH IN DIAMETER BY 5/8 INCH LONG. UNDER IRRADIATION, THERMAL STRESSES CAUSE THE PELLETS TO BREAK UP INTO FRAGMENTS GENERALLY RANGING IN SIZE FROM 1/16 INCH TO 1/4 INCH ON A SIDE. DURING A TRANSIENT AND THE PERIOD FOLLOWING, THE FLOWING WATER AND STEAM CAN BE EXPECTED TO CAUSE SOME FUEL EROSION, WHICH WILL PRODUCE VERY

SMALL PARTICLES WHICH CAN BE SUSPENDED IN MOVING WATER.

IN THE CENTER ASSEMBLIES, IT IS LIKELY THAT MOST OF THE FUEL HAS BEEN RELEASED FROM THE RODS BETWEEN THE END FITTING AND THE SECOND OR THIRD INTERMEDIATE GRIDS. BASED ON THE UNDERSTANDING THAT THE FLOW IN THE CORE IS SEVERELY BLOCKED, THE FUEL FRAGMENTS HAVE SETTLED ON TO THE INTERMEDIATE GRIDS. LOCAL FLOW PERTURBATIONS CAN MOVE PELLET FRAGMENTS THROUGHOUT THE SYSTEM. FUEL FROM THE UPPER LEVEL MAY HAVE SETTLED DOWN THROUGH THE TOP INTERMEDIATE SPACER GRID TO THE SECOND LEVEL AND LOWER LEVELS TO A LESSER EXTENT. THERE IS SOME REMOTE POSSIBILITY THAT THE STRUCTURE SUPPORTING THE FIRST TWO INTERMEDIATE GRIDS IN THE CENTER FEW ASSEMBLIES MAY COLLAPSE, CAUSING THE TOP 5 FT OF FUEL TO SETTLE ON THE THIRD INTERMEDIATE GRID. THE FUEL FRAGMENTS WOULD LIKELY BE MIXED IN WITH SOME REMAINING ZIRCALOY ROD FRAGMENTS. THE SPACE BETWEEN FRAGMENTS COULD BE FILLED WITH WATER, STEAM, ZrO_2, OR SOME COMBINATION THEREOF. THE LARGE QUANTITY OF SMALL ZrO_2 PARTICLES COULD CAUSE SOME LOCALIZED FLOW BLOCKAGE TO PREVENT FULL COVERAGE WITH WATER. THE PRESENCE OF SOME TUBULAR SEGMENTS COULD ALLOW LOCAL FLOW CHANNELING AND ATTENDANT "JETTING".

6. PROJECTED STABILITY OF CORE CONDITION

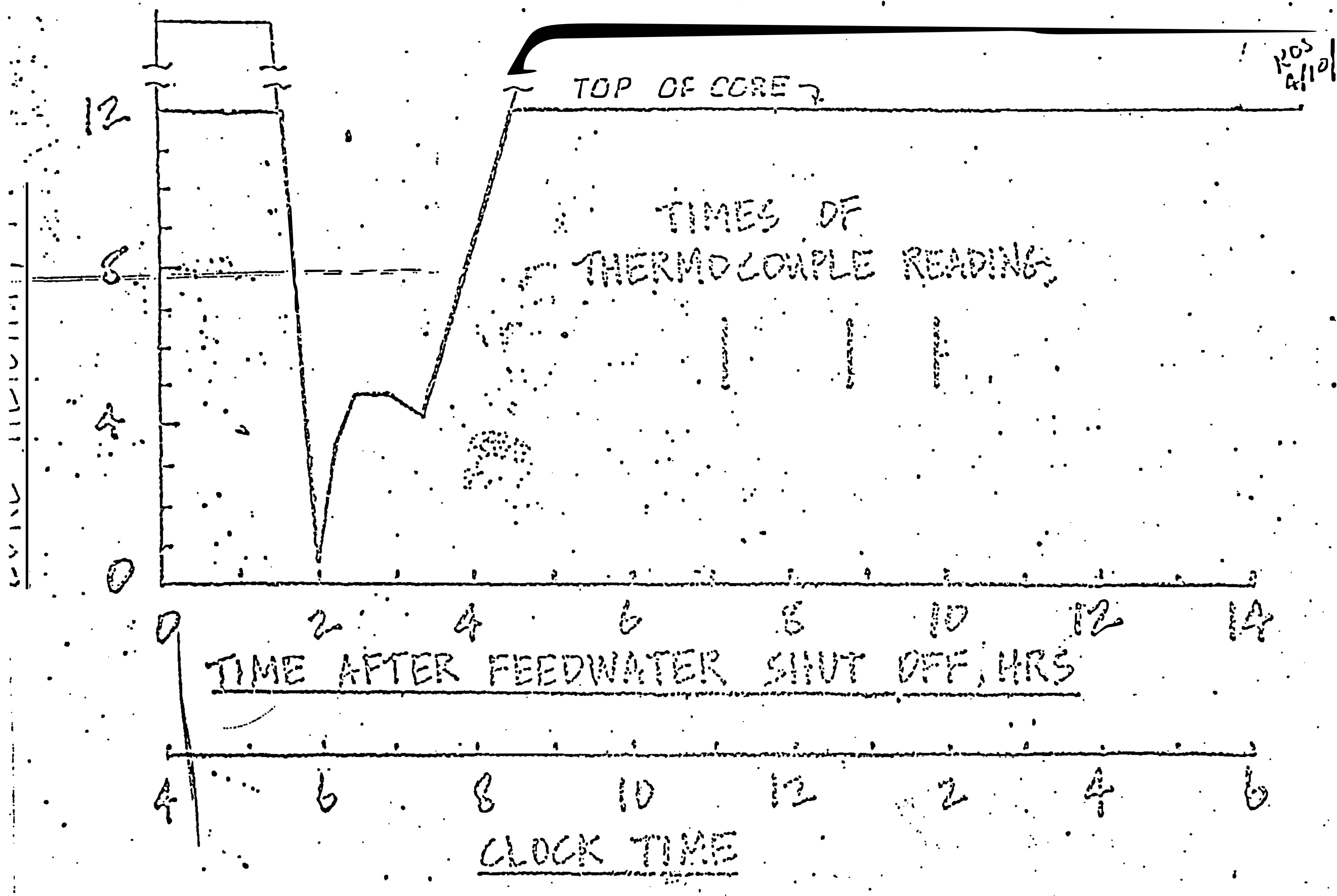
THE POSSIBILITY OF CONTINUED STRUCTURAL DEGRADATION REQUIRES FURTHER EVALUATION. FLOW BLOCKAGE IS LIKELY WHICH CAN CAUSE LOCALIZED BOILING. WHEN LOCALIZED BOILING EXISTS, A FURTHER REDUCTION IN SYSTEM PRESSURE WILL INCREASE THE AREA OF BOILING AND RAISE THE TEMPERATURE OF CLADDING IN THE AFFECTED AREA. IF THE TEMPERATURE OF ANY ZIRCALOY COMPONENT EXCEEDS $1000^{\circ}F$., ACCELERATED OXIDATION WILL ADD TO THE GENERATION OF HYDROGEN AND CAUSE FURTHER DEGRADATION OF THE CORE STRUCTURE.

7. DISTRIBUTION OF FUEL AND ZrO_2 IN SYSTEM

IT IS VERY LIKELY THAT PARTICLES OF ZrO_2 AND UO_2 ARE CIRCULATING THROUGHOUT THE PRIMARY SYSTEM AND MAY SETTLE OUT IN STAGNANT AREAS.

FIGURE [illegible]

ESTIMATED CORE COOLANT LEVEL



NOS 4/10/79

[illegible] [illegible] Fl. = 90 GPM

SUBJECT NO. 3

ATTACHMENT 5

CORE MELT SCENARIOS

.1 GENERAL

UNDER THE CONDITIONS OF NATURAL CIRCULATION, THE CORE WILL BE SURROUNDED BY COLD WATER NEAR 100°F. THE POSSIBILITY OF CORE MELT IS CONSIDERED TO BE REMOTE UNDER THESE CONDITIONS. WITH THE CURRENT LOW DECAY HEAT RATE AND WITH APPROPRIATE MONITORING OF INCORE THERMOCOUPLES, THERE WILL BE SUFFICIENT EARLY WARNING SIGNALS TO PREVENT A CORE MELT SITUATION.

.2 CORE MATERIAL MELTING POINT

THE MELTING TEMPERATURES FOR THE CRITICAL FUEL ASSEMBLY MATERIALS ARE SUMMARIZED ON THE ATTACHED TABLE. THE MATERIALS INCLUDE THE FUEL ASSEMBLY STRUCTURAL MATERIALS (END FITTINGS, GRIDS, HOLDDOWN SPRING, AND GUIDE TUBES) AND PRIMARY FUEL ROD MATERIALS (PELLETS, CLADDING, AND END CAPS). IN ADDITION, THE MELTING TEMPERATURE OF THE CHROMEL ALUMEL THERMOCOUPLES IS ALSO INCLUDED.

.3 ASSESSMENT OF ORIGINAL FUEL DAMAGE CONDITIONS

DURING THE INITIAL ACCIDENT, WHEN THE CORE WAS PARTIALLY UNCOVERED, THE THERMAL CONDITIONS WERE VERY SEVERE. HOWEVER, THERE ARE INDICATIONS THAT THE CORE DID NOT UNDERGO MELTING.

THE DECAY HEAT RATE WAS IN EXCESS OF 25 MW_T. ANY WATER NEAR THE CORE WAS NEAR SATURATION TEMPERATURE OR ~500-650°F. UNDER THESE CONDITIONS THE FUEL ROD CLADDING REACHED 2000F OR HIGHER AND OXIDIZED SEVERELY TO PRODUCE HYDROGEN. HOWEVER, CONTINUED OPERATION OF THE INCORE CHROMEL ALUMEL THERMOCOUPLES, WHICH HAVE A MELTING POINT NEAR 2500F, INDICATE THAT THE STEAM TEMPERATURE

INSIDE OF THE INSTRUMENT TUBE WAS LESS THAN 2500F. THIS IS APPROXIMATELY 2500F FROM THE MELTING POINT OF UO_2. THE CENTER-LINE TEMPERATURE OF UO_2 PELLET FRAGMENTS IS ESTIMATED TO BE NO MORE THAN 100F HIGHER THAN THE STEAM; THUS, SHOWING A LARGE MARGIN TO UO_2 MELTING. LOCAL HOTSPOTS FUEL MAY HAVE BEEN HIGHER FOR SHORT PERIODS BUT THE ESTIMATED 2500F MARGIN TO MELTING WAS SUFFICIENT TO PRECLUDE MELTING.

THE RADIOCHEMISTRY ANALYSIS OF B_A 140 AND OTHER ISOTOPES IN COOLANT SAMPLE TAKEN A DAY AFTER THE ACCIDENT DID NOT INDICATE THAT UO_2 MELTING HAD OCCURRED.

[illegible].4 EARLY WARNING SIGNALS

DURING THE TRANSITION TO NATURAL CIRCULATION, THE INCORE THERMOCOUPLES WILL BE MONITORED. THE TEMPERATURES ON THESE THERMOCOUPLES HAVE A NORMAL READOUT RANGE OF UP TO 900F. SINCE SOME LOCALIZED BOILING IS EXPECTED, A FEW OF THE THERMOCOUPLES CAN BE EXPECTED TO READ HIGHER THAN SATURATION TEMPERATURE. HOWEVER, BECAUSE OF THE SLOW HEAT UP OF THE OVERALL SYSTEM, THE MAJORITY OF THERMOCOUPLES, AS A GROUP, CAN BE USED TO MONITOR THE BULK COOLANT TEMPERATURE AT THE TOP OF THE CORE. BULK COOLANT TEMPERATURES APPROACHING SATURATION TEMPERATURE WOULD BE AN EARLY INDICATOR THAT LOCALIZED BOILING WAS SPREADING AND THAT CORRECTIVE ACTION SHOULD BE TAKEN. LARGE AREAS OF BOILING ARE UNDESIRABLE SINCE THEY LEAD TO HIGH STEAM TEMPERATURES. WHEN THE STEAM EXCEEDS 1000F, THE ZIRCALOY

COMPONENT WILL BEGIN TO OXIDE AND PRODUCE HYDROGEN. AT HIGHER TEMPERATURES THE RATE OF HYDROGEN PRODUCTION WILL INCREASE. HOWEVER, AS INDICATED IN [illegible].3, EVEN WITH THE HIGH STEAM TEMPERATURES PRODUCED DURING THE INITIAL CORE UNCOVERY, THE UO_2 DID NOT MELT. THUS, THE INCORE THERMOCOUPLES CAN PROVIDE EARLY WARNING SIGNALS SUCH THAT CORRECTIVE ACTION CAN BE TAKEN TO PREVENT A CORE MELT SITUATION.

TABLE [illegible]

Melting Points of Core Materials

Material	Melting Point ^oF
UO_2	5081 (1)
Zr-4	3353 (2)
Inconel X-750	2570 (3)
Inconel 718	2323 (4)
CF3MSS (and fitting)	2550 (5)
Zr O_2	5010 (6)
Chromel Alumel	2500 (7)

Oxidation of Zircaloy is assumed to initiate at 1000 F

1. Hausner, H., "Determination of the Melting Point of Uranium Dioxide", Journal of Nuclear Materials, Vol. 15,1965.

2. Hammer, R. R., In-1093, Sept., 1967.

3. Huntington Alloys Technical Bulletin, Inconel Alloy X-750, (1970).

4. Huntington Alloys Technical Bulletin, Inconel Alloy 718, (1968).

5. Materials Engineering, Materials Selector 77.

6. Lynch, J. F., et al, <u>Engineering Properties of Selected Ceramic Materials</u>, (1966).

7. Weast, R. C., <u>CRC Handbook of Chemistry and Physics</u>, 48th Ed, (1968).