NRC FORM 195
(2-76)

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DESCRIPTION

Ltr re our 3-29-79 ltr...trans the following:

ENCLOSURE

Addl info concerning Cycle 2 reload..........

(17 cys encl rec'd)

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PLANT NAME: Three Mile Island #1

SAFETY		FOR ACTION/INFORMATION		ENVIRO [illegible]
	ASSIGNED AD :			ASSIGNED AD :
✓	BRANCH CHIEF :	Reid		BRANCH CHIEF :
✓	PROJECT MANAGER:	Bridges		PROJECT MANAGER :
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	MIPC	MACCARY			SITE TECH
	CASE	KNIGHT		OPERATING REACTORS	GAMMILL
	HANAUER	SIHWEIL		STELLO	STEPP
	HARLESS	PAWLICKI			HULMAN
				OPERATING TECH	
	PROJECT MANAGEMENT	REACTOR SAFETY	✓	EISENHUT	SITE ANALYSIS
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	P. COLLINS	NOVAK	✓	BAER	BUNCH
	HOUSTON	ROSZTOCZY	✓	SCHWENCER	J. COLLINS
	PETERSON	CHECK	✓	GRIMES	KREGER
	MELTZ				
	HELTEMES	AT & I		SITE SAFETY & ENVIRO	
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		RUTBERG		DENTON & MULLER	

EXTERNAL DISTRIBUTION

✓	LPDR: Harrisburg, Pa	NATL LAB	BROOKHAVEN NATL LAB	CONTROL NUMBER
✓	TIC	REG. V-IE	ULRIKSON (ORNL)	38
✓	NSIC	LA PDR		
	ASLB	CONSULTANTS		
✓	ACRS 16 ~~HOLDING~~/SENT	TO LA Ingram		

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METROPOLITAN EDISON COMPANY SUBSIDIARY OF GENERAL PUBLIC UTILITIES CORPORATION

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April 8, 1976
GQL 0506

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U.S. NUCLEAR REGULATORY COMMISSION
Mail Section

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

Dear Sir:

THREE MILE ISLAND NUCLEAR STATION UNIT 1 (TMI-1)
OPERATING LICENSE NO. DPR-50
DOCKET NO. 50-289

In response to your letter of March 29, 1976 requesting additional information pertaining to our Cycle 2 Reload Report, please find enclosed three signed originals (37 conformed copies sent separately) of our responses.

Sincerely,

R. C. Arnold
Vice President

RCA:CWS:ilm

Enclosure

RESPONSES TO NRC QUESTIONS CYCLE 2 RELOAD REPORT

NRC QUESTION (1)

. . . . . provide the following:

a. A description of the flow measurement technique used along with the data measured and an error analysis for the measurements.

b. A discussion of the bases for the power to flow into reactor trip setting and the overpower trip setting (if it is based on measured flow).

c. A proposed surveillance and/or test program to confirm that the value of the core flow rate has not decreased below the value used as the basis for reactor power/flow trip (and/or the overpower trip), including appropriate uncertainties.

RESPONSE TO QUESTION (1)

Attachment (1) provides the response to parts a and c above and attachment 2 provides the response to part b.

NRC QUESTION (2)

Rod bowing is not considered in the report. A discussion should be included on the effect of rod bowing and either justification given for not including the effects or revised technical specifications to account for rod bowing should be proposed.

RESPONSE TO QUESTION (2)

Proposed Technical Specifications to account for the effects of possible fuel rod bowing have been forwarded by our letter of April 2, 1976.

NRC QUESTION (3)

Provide justification for the moderator and doppler coefficients used in the accident and transient analysis (Table 7.1-1 of Reload Report) and explain how these values relate to those listed in Table 5.1-1.

RESPONSE TO QUESTION (3)

The moderator and doppler coefficients given in Table 7.1-1 are taken from the FSAR. When the FSAR accident analyses were performed these values were used to provide conservative worst case values for these coefficients. On line measurements ($\sim$ BOC) of these coefficients are reported in the TMI-1 Initial Startup Report (SUR). The reported values are: moderator -0.222×10^{-4} $\Delta k/k/$ F and doppler -1.07×10^{-5} $\Delta k/k/$ F. (See Initial Startup Report Table 5.6-2.) Also note that doppler coefficient is extracted from power doppler coefficient measurements by calculation (See Startup Report Section 5.6). EOC on line measurements for these coefficients have not been evaluated however, initial indications are that they are very close to the predicted Cycle 1 values given in Table 5.1-1 of the Cycle 2 Reload Report.

The conclusion to be drawn from a comparison of the moderator and doppler coefficients given in Table 7.1-1 and Table 5.1-1 is that the values used for the purposes of the Accident Analyses are indeed conservative.

NRC QUESTION (4)

Indicate which of the values listed in Table 5.1-1 were observed in actual Cycle 1 operation and compare these observed values with calculations. Parameters of specific interest include critical boron concentrations, control rod worths, and core temperature coefficients.

RESPONSE TO QUESTION (4)

All values given in Table 5.1-1 are taken from the TMI-1 Cycle 1 Physics Test Manual (PTM) except as noted. Attached is a copy of Table 5.1-1 with item numbers assigned to facilitate discussion. The following table lists the table or figure number from which each item was taken.

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Table 5-1 TMI-1, Cycle 2 Physics Parameters

	Cycle 2	Cycle 1
Cycle length, EFPD	296	466
Cycle burnup, MWd/mtU	9144	14,396
Average core burnup - EOC, MWd/mtU	18,612	14,396
Initial core loading, mtU	82.1	82.1
1 Critical boron - BOC, ppm (No Xe.)		
HZP - all rods out	1350	1634
HZP - groups 7 and 8 inserted	1187	1494
HFP - groups 7 and 8 inserted	1004	1382
2 Critical boron - EOC, ppm (Eq. Xe)		
HZP - all rods out	390	480
HFP - group 8 (37.5% withdrawn, equil. Xe)	46	180
3 Control rod worths - HFP, BOC, %Δk/k		
Group 6	1.17	1.58
Group 7	.97	0.99
Group 8 (37.5% wd)	.54	0.44
4 Control rod worths - HFP, EOC, %Δk/k		
Group 7	1.32	1.37
Group 8 (37.5% wd)	.50	0.26
5 Ejected rod worth - HZP, %Δk/k		
BOC	.57+	0.48++
EOC	.54+	0.72++
6 Stuck rod worth - HZP, %Δk/k		
BOC	2.15	4.27
EOC	2.21	2.69
7 Power deficit, HZP to HFP, %Δk/k		
BOC	-1.64	-1.32
EOC	-2.48	-2.10
8 Doppler coeff - BOC, 10^{-5} ($\Delta k/k/^{o}F$) 100% power (0 Xe)	-1.49	-1.51
9 Doppler coeff - EOC, 10^{-5} ($\Delta k/k/^{o}F$) 100% power (equil Xe)	-1.53	-1.67
10 Moderator coeff - HFP, 10^{-4} ($\Delta k/k/^{o}F$)		
BOC (0 Xe, 1000 ppm. groups 7 and 8 inserted)	-1.06	-0.23
EOC (equil Xe, 17 ppm, group 8 inserted)	-2.63	-2.70

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	Cycle 2	Cycle 1
11 Boron worth - HFP, ppm/%Δk/k		
BOC (1000 ppm)	108	98
EOC (17 ppm)	100	95
12 Xenon worth - HFP, %Δk/k		
BOC (4 days)	2.61	2.71
EOC (equilibrium)	2.67	2.65
13 Effective delayed neutron fraction (HFP)		
BOC	.00577	.00690
EOC	.00516	.00514

+Ejected rod value for group 5, 6, 7 and 8 inserted
++Ejected rod value for Group 6, 7, and 8 inserted

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TABLE I

Item No.	Reference For Value
(1)	PTM Table 2-4
(2)	PTM Figure 2-61 & 2-49
(3)	PTM Table 2-3
(4)	PTM Table 2-3
(5)	Recent B&W Calculations
(6)	PTM Figure 2-39 (full length APSR)*
(7)	PTM Figure 2-55*
(8)	Recent B&W Calculations (TM-159)
(9)	Recent B&W Calculations
(10)	PTM Figure 2-52 & 2-57**
(11)	PTM Figure 2-47 & 2-48
(12)	PTM (12/10/73)
(13)	PTM Page 2.1-3

*EOC Value From Recent B&W Calculations

**With Operating Control Rod Alignment

The values given in Table 5.1-1 are calculated and represent best estimates of on line values under the stated conditions. The techniques used by B&W to calculate these values are considered to be state-of-the-art techniques, and are refined, where possible and appropriate, based on experimental and operational data. The same or similar techniques were used to predict Cycle 2 values.

Table 2 compares the calculated and measured (where available) physics parameters. It can be seen that where observed values are available they compare very favorably with calculated values and support the validity of the calculational techniques used for Cycle 1 and Cycle 2 physics parameters. References for the measured parameters are given in the notes for Table 2.

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. NRC QUESTION (5)

It is our understanding that the check value proposed in your correspondence of April 19, 1975 to address the problem of long-term cooling following a loss of coolant accident (LOCA) will not be available for installation prior to start-up of Cycle 2. If this is the case it will be necessary for you to provide specific information on how power would be restored to the necessary valves following a failure of the 1C Engineered Safeguard Valve 480 V control center prior to utilization of that aspect of the cooling system.

RESPONSE TO QUESTION (5)

The 2-inch stop check valve which originally was to be installed in the Decay Heat Pressurizer Auxilliary Spray line, is not available due to delivery delays. However, a 1½ inch stop check valve is available and will be installed prior to Cycle 2 operations. The subject valve will be installed between the existing inside containment isolation valve, DH-V63, and penetration 320S. This new valve will function as the inside containment isolation valve and will permit manual valves DH-V62 and DH-V63 to be locked open during power operations. In this manner, a flow rate of about 70 gpm could be established through the auxilliary spray line for boron control following a LOCA. This would be accomplished by opening motor operated valve RC-V4 and the manual outside containment isolation valve DH-V64. Please note that the primary flow path for boron control following a LOCA is through the Decay Heat Drop Line (i.e. through motor operated valves DH-V1, DH-V2 and DH-V3) to either the reactor building sump or to the Decay Heat pumps. The auxilliary spray line flow path is, therefore, only a back-up flow path to be utilized should a single failure, such as failure of DH-V1 or DH-V2, preclude establishment of flow through the Decay Heat Drop LIne.

As noted in our April 19, 1975 letter, the worst single failure which could occur is the failure of the 1C Engineered Safeguards Valve 480 V control center. In such case power to open valves DH-V1, DH-V2 and DH-V3 in the primary flow path or RC-V4 in the alternate flow path would not be immediately available. In this case the following procedure would be used to open RC-V4 to establish long-term flow for post LOCA boron control:

1. Open breaker for RC-V4 at the 1C ES Valve Control Center.

2. Verify MU-V2A is in its closed position and open the breaker for MU-V2A at the 1B ES Valve Control Center .

3. At penetration 315E, lift the power and control cables from the MU-V2A motor controller.

4. At penetration 317E, which is located about 10 feet from penetration 315E, lift the power and control cables from RC-V4 motor control center.

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5. Using jumpers, connect the MU-V2A motor controller power and control cables removed from penetration 315E to penetration 317E to penetration 317E connections for RC-V4.

6. Utilizing MU-V2A motor controller open RC-V4.

As noted in our April 19, 1975 letter, 30 days would be available to accomplish the above emergency action and, therefore, no plant changes are considered necessary.

TABLE 2 [illegible]CLE 1 PHYSICS PARAMETERS

	Calculated	Measured	Note
Critical boron - BOC, ppm (No Xe.)			
HZP - all rods out	1634	1617	(1)
HZP - groups 7 and 8 inserted	1494	1545	(2)
HFP - groups 7 and 8 inserted	1382		
Critical boron - EOC, ppm (Eq. Xe)			
HZP - all rods out	480		
HFP - group 8 (37.5% withdrawn, equil. Xe)	180	199	(3)
Control rod worths - HFP, BOC, %Δk/k			
Group 6	1.58	1.52	(4)
Group 7	0.99	0.98	(4)
Group 8 (37.5% wd)	0.44	0.46	(4)
Control rod worths - HFP, EOC, %Δk/k			
Group 7	1.37		
Group 8 (37.5% wd)	0.26		
Ejected rod worth - HZP, %Δk/k			
BOC	0.48	0.688	(5) (6)
EOC	0.72		
Stuck rod worth - HZP, %Δk/k			
BOC	4.27		
EOC	2.6[illegible]		
Power deficit, HZP to HFP, %Δk/k			
BOC	-1.32	-0.90	(7)
EOC	-2.10		
Doppler coeff - BOC, 10^{-5} ($\Delta k/k/^{o}F$)			
100% power (0 Xe)	-1.51	-1.07	(8)
Doppler coeff - EOC, 10^{-5} ($\Delta k/k/^{o}F$)			
100% power (equil Xe)	-1.67		
Moderator coeff - HFP, 10^{-4} ($\Delta k/k/^{o}F$)			
BOC (0 Xe, 1000 ppm. groups 7 and 8 inserted)	-0.23	-0.222	(9)
EOC (equil Xe, 17 ppm, group 8 inserted)	-2.70		
Boron worth - HFP, ppm/%Δk/k			
BOC (1000 ppm)	98		
EOC (17 ppm)	95		
Xenon worth - HFP, %Δk/k			
BOC (4 days)	2.71		
EOC (equilibrium)	2.65		
Effective delayed neutron fraction (HFP)			
BOC	.00690		
EOC	.00514		

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TABLE 2 (continued)

NOTES:

(1) SUR Page 4.4-1
(2) SUR 4.1-2 with group 7 at 26.5% withdrawn
(3) Hot Full Power and Operating Control Rod Alignment Last day of operation approximately 466 EFPD.
(4) SUR Table 4.7-3 Hot Zero Power
(5) SUR Table 4.8-1
(6) 0.706 at approximately 250 EFPD following rod swap usingboron swap method
(7) SUR Page 5.6-2
(8) SUR Table 5.6-2 with equil. xenon
(9) SUR Table 5.6-2

R. C. FLOW MEASURING TECHNIQUE & ASSOCIATED ERRORS

The RC flowrate at TMI-1 is determined by two methods: a direct method using Gentille flow meters in the Loop A and B reactor outlet piping and an indirect method using plant heat balance data and measured feedwater flowrates for Loop A and B. Historically, the primary side flowmeters have been indicating approximately 2% lower flows than the actual flow. After TMI-1 achieved rated power, an attempt was made to re-calibrate the primary side flowmeter instrumentation to eliminate the -2% offset but gradually over a years time, the primary side flowmeter indication degraded again 2%. Additional effort spent trying to arrest this degradation was successful, however, the policy has evolved that the heat balance combined with the feedwater flowmeter measurement is the best method for determining actual reactor coolant flowrate. This procedure is not unique to the TMI-1 plant but is utilized at all operating B&W plants. For power levels near or equal to rated power, the net core power calculated from secondary side measurements is the value used to periodically re-calibrate the nuclear instrumentation for power level. Thus, to indirectly determine the actual total reactor coolant flowrate with plant heat balance data and measured feedwater flowrates at full power is both accurate and consistent with plant calibration procedures.

The specific measurements required to determine total reactor coolant flowrate are presented in the following table:

Table 1

B&W NSS Heat Balance Measurements

1. Loop A reactor Outlet Temperature
2. Average of 2 Loop A Reactor Inlet Temperatures
3. Loop A RC pressure

4. Loop A feedwater temperature
5. Loop A feedwater pressure (at OTSG inlet)
6. Loop A steam temperature (at OTSG discharge)
7. Loop A steam pressure (at OTSG discharge)

8. Loop A feedwater flowrate

9. Loop B Reactor Outlet Temperature
10. Average of 2 Loop B reactor Inlet Temperatures
11. Loop B RC pressure

12. Loop B feedwater temperature
13. Loop B feedwater pressure
14. Loop B steam temperature
15. Loop B steam pressure

16. Loop B feedwater flowrate

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17. Input power to all four RC pumps
18. Letdown flowrate
19. Makeup flowrate
20. Letdown temperature
21. Makeup temperature

Measurements 1, 2, and 3 yield a net Loop A enthalpy change and 9, 10, and 11 similarly yield a net Loop B enthalpy change. Measurements 4, 5, 6, and 7 are used to determine a Loop A secondary side enthalpy increase and items 12, 13, 14 and 15 provide the Loop B secondary side enthalpy increase.

For a normal secondary side heat balance calculation of net reactor power, the Loop A and B feedwater flowrates and the Loop A and B enthalpy increase values are combined with pump input power values and energy losses and gains via the letdown and makeup flows to calculate the power level of the reactor. To determine the total reactor coolant flow from the plant heat balance measurements, a heat balance equation was derived for the steam generators as shown in Appendix A.

Table II below displays typical heat balance data recorded and printed out by the TMI-1 plant computer on February 16, 1976.

TABLE II

TMI-1 PLANT HEAT BALANCE DATA

Parameter	Run #1	Run #2
T hot-A-F	602.2	602.1
T cold, A-1-F	557.6	557.4
T cold, A-2-F	556.2	555.9
Avg T cold-A-F	556.9	556.65
RC Pressure-A-psia	2170	2169
* ΔH primary-Loop A	60.87	61.04
T fdw-A-F	458.1	458.3
P fdw-A-psia (assumed)	970	970
T steam-A-F	593.1	593.0
P steam-A-Psia	917	916
* ΔH secondary-Loop A	813.31	813.11
T hot-B-F	601.3	601.2
T cold, B-1-F	556.8	556.7
T cold, B-2-F	556.9	556.8
Avg T cold-B-F	556.85	556.75
RC Pressure-B-psia	2170	2169
* ΔH primary - Loop B	59.62	59.60

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T fdw-B-F	458.1	458.3
P fdw-B-psia(assumed)	970	970
T steam-B-F	592.8	592.7
P steam -B-psia	917	916
* ΔH secondary - Loop B	812.88	812.86
Loop A Feedwater Flow lbs/Hour	5.389×10^6	5.396×10^6
Loop B Feedwater Flow lbs/Hour	5.233×10^6	5.259×10^6
Total RC Flowrate(calculated) lbs/hour	143.35×10^6	143.61×10^6
Ratio to Design RC Flowrate	109.5%	109.7%

To determine the influence of the measurement errors on the calculated value of the total RC flow, Met-Ed performed an error analysis identical to that carried out by C. L. Howard et al for the USAEC (Contract # AT(04-3)-189). The theory, assumptions and a list of the carefully evaluated measurement errors are included in Appendix A. After considering all the errors and their functional dependence, it was possible to determine the deviations in Loop A and B RC flowrates and in the total RC flowrate. It was found that the maximum probable error on the total RC flowrate will not exceed 1.49% for 95% of the data

As to reactor coolant flowrate surveillance program, Metropolitan Edison Company will verify the total RC flowrate within three months after refueling and after that periodically every six months (plus/minus 30 days) using the heat balance technique described above.

APPENDIX A

Error In Core Flow Determination At TMI-1

An error analysis has been conducted to determine the effect of instrumentation and calculational uncertainties in the determination of reactor core flow. The analysis has been made on a statistical basis using the method of C. L. Howard's at all (USAEC Contract AT (04-3)-189).

As mentioned earlier in this report the practice has evolved at TMI-1 that the actual reactor core flow rate is determined from heat balance based on feedwater flow measurement. The primary loop flows (A/B) are determined from heat balance over the steam generators (SG) from the following relation

$$W = W_F \frac{(h_s - h_F)}{(h_h - h_c)} + \frac{L}{(h_h - h_c)} \qquad (1)$$

Where

W = primary loop flow (A/B) (lb/hr)

W_F= feedwater flow (lb/hr)

h_s= steam enthalpy (Btu/lb)

h_f= feedwater enthalpy (Btu/lb)

h_h= hot leg enthalpy (Btu/lb)

h_e= cold leg enthalpy (Btu/lb)

L = heat loss from the system surface bounded by the temperature sensors (Btu/hr)

The core flow for this analysis is defined to be the sum of the loop flows. The flows obtained from relation (1) are termed actual flows (expected values) and are based on actual instrument readings. The uncertainity in loop flows because of instrument errors was determined based on the following assumptions:

1) instrument errors follow approximately a normal probability distribution.
2) the tails of the distribution are truncated because those instruments that are not within specifications are rejected, and our carefully established surveillance and test program is the assurance that specification limits will not be reached.
3) the manufacturer guaranteed maximum error is considered to be 2 σ confidence limit (Supplement to ASME Power Test Codes, Part 5).

4) the instrument string errors were determined from the random maximum component errors, manufacturer guaranteed maximum drift included.
5) since our main interest is decreased reactor safety the confidence level is based on use of only one "tail" of the distribution curve.

The core flow error will be the statistical combination of the loop flow errors.

The variance of W, being a function of the uncorrelated variables W_f; h_s; h_f . . . etc. is calculated from the relationship

$$\sigma_W^2 = \left(\frac{\partial W}{\partial W_F}\sigma_{W_F}\right)^2 + \left(\frac{\partial W}{\partial h_s}\sigma_s\right)^2 + \left(\frac{\partial W}{\partial h_F}\sigma_F\right)^2 + + \left(\frac{\partial W}{\partial W_h}\sigma_h\right)^2 + \left(\frac{\partial W}{\partial h_c}\sigma_c\right)^2 + \left(\frac{\partial W}{\partial L}\sigma_L\right)^2 \qquad (2)$$

Following partial differentiations

$$\sigma_W^2 = \left(\frac{\Delta h_F}{\Delta h_{RC}}\sigma_{W_F}\right)^2 + \left(\frac{W_F}{\Delta h_{RC}}\sigma_s\right)^2 + \left(\frac{W_F}{\Delta h_{RC}}\sigma_F\right)^2 + + \left(W_F\frac{\Delta h_F}{(\Delta h_{RC})^2}\sigma_h\right)^2 + \left(W_F\frac{\Delta h_F}{(\Delta h_{RC})^2}\sigma_c\right)^2 + \left(\frac{1}{\Delta h_{RC}}\sigma_L\right)^2 : (*) \qquad (3)$$

The standard deviations of the enthalpies are determired from

$$\sigma_h^2 = \left(\frac{\partial h}{\partial T}\sigma_T\right)^2 + \left(\frac{\partial h}{\partial p}\sigma_p\right)^2 \qquad (4)$$

For the calculations the numerical values of RUN-2 were used (See TABLE II of the report) in conjunction with the instrument string errors shown in TABLE I:

(*) $\Delta h_F = h_s - h_F$

$\Delta h_{RC} = h_h - h_c$

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TABLE I

Instrument String Errors Used for Loop Flow Error Analysis

Feedwater flow	± 0.63%
Feedwater temp.	± 1.94°F
Steam Pressure	± 3.0 psi
Steam temp.	± 2.12°F
RC hot leg temp.	± 0.934°F
RC cold leg temp.	± 0.52°F
RC pressure	± 30.0 psi
Heat loss to ambient	± 50%

The feedwater flow error shown in TABLE I was calculated as follows: The feedwater flow in both loops is measured by flow elements supplied by BAILEY METER CO. The flow is determined from the relation

$$W_F = (3.1460 \times 10^4)\, c \sqrt{\rho h_w} = K c \sqrt{\rho h_w} \qquad (5)$$

Where c; ρ and h_w are; the mean discharge coefficient, feedwater density in flow element (lb/ft^3) and differential pressure (in. water at 68 F) respectively. The mean discharge coefficients for both flow elements were determined by ALDEN RESEARCH LABORATORIES, Holden, Mass..

The error in feedwater flow measurement was determined by the method discussed above. The variance of W_F for either loop is obtained from the expression

$$\sigma_{W_F}^2 = K^2 \left[\left(\sqrt{\rho h_w}\right)^2 \sigma_c^2 + \left(\frac{c}{2}\sqrt{\frac{h_w}{\rho}}\right)^2 \sigma_\rho^2 + \left(\frac{c}{2}\sqrt{\frac{\rho}{h_w}}\right)^2 \sigma_{h_w}^2 \right] \qquad (6)$$

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were the σ -s are standard deviations defined by their subscripts. The errors used for the calculations are shown in TABLE II

TABLE II

Errors used for Feedwater Flow Error Analysis

Nozzle discharge coefficient	$\pm$ 0.5%
Feedwater temp.	$\pm$ 1.94°F
Nozzle pressure differential	$\pm$ 5.45 in water

The feedwater flow errors (2 σ) obtained for loop A and B were .612% and .628% respectively. A conservative value of .63% was entered in TABLE I.

Based on the method, assumptions and numerical values discussed above the maximum probable error obtained for the primary loop A RC flow (1.645 σ) was 1.512 $*$ 10^6 lbs/hr (2.1%). Since the system conditions in each loop are nearly the same, the errors are nearly identical. Therefore the maximum probable error in core flow due to the independent random loop errors will not exceed 1.49% for 95% of the data.

The 2 σ loop flow error is 1.8384 $*$ 10^6 lbs/hr (2.56%) leading to a 2 σ core flow error of 2.60 $*$ 10^6 lbs/hr (1.82%).

The results of this analysis indicate that the 106.5% design flow for Cycle 2 is acceptable relative to the 108% generic "nominal" flow (See Cycle 2 Reload Report p. 6.1) based on the maximum probable error, and is highly conservative relative to the actual RC flow at TMI-1 of 109.3%. This value is based on numerous heat balance calculations carried out in 1975 and 1976, and is considered a time averaged stable flowrate without drift.

BFH:ilm

Attachment (2)

Bases For Power/Flow Trip

I. BACKGROUND

This enclosure provides a discussion of the bases for the power to flow into the reactor trip setting. The bases for the overpower trip setting is not included since it is not based on measured flow.

It is the purpose of this enclosure to demonstrate that the value of the flux/flow trip ratio is still conservative and adequate in the light of certain parameters different from those values assumed in the FSAR. These parameters are:

A. Core Flow Rate; measured flow rate (with 1.5% margin) is used rather than design flow rate.

B. Response Time Flux/Flow Tr ,, Field Change FC-137 increased the time constant of the reactor coolant flow sensing string to 1.4 seconds from the 0.65 seconds assumed in the FSAR.

The effect of rod bowing has also been addressed. Consequences of rod bowing other than those affecting DNB are covered in our Technical Specifications Change Request 30 Amendment 2, submitted April 2, 1976.

II. THERMAL-HYDRAULIC METHODS

To determine the flux/flow trip setpoint that is necessary to meet the hot-channel DNB ratio criteria, several calculational steps are required. These steps involve such things as the determination of steady-state operating conditions, fuel densification effects and transient calculations.

A. Thermal-Hydraulic Conditions During Normal Operation

The hot channel thermal hydraulic conditions are calculated for design conditions at 108% of rated power. The power level of 108% includes operation at 102% of rated power plus a maximum power level measurement error of 6% (4% neutron flux error and 2% heat balance error). This serves as the benchmark calculation from which the densification penalty and the transient effects can be determined. The steady state analysis is performed using the TEMP computer code (BAW-10021) (1) with the appropriate hot channel factors, coolant inlet temperature and system pressure errors, and a 5% hot assembly flow maldistribution factor applied. These conservatisms are consistent with the calculational techniques employed in the FSAR analyses. The design flow rate of 131.32×10^6 #/hr. (88,000 GPM/Pump) was used for first cycle analysis. For second cycle analysis, the reanalysis used 106.5% of the design flow rate, based on system flow measurements made during the first cycle. For both cycles, the hot assembly power distribution consisted of a 1.78 radial-local nuclear peaking factor ($F \Delta H$) with a 1.5 cosine axial flux shape. Incorpor ion of the increased flow rate into the analysis was accompanied by a corresponding increase in the reactor coolant inlet temperature, from 554F to 555.6F for the nominal, rated power

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condition (assumed to be 2,568MWt for this analysis.) Neither the increase in system flow nor the increase in inlet temperature represents a change in the operation of the plant. The integrated control system maintains a constant average reactor coolant temperature, an increase in the steady-state system flow rate results in an increase in core inlet temperature and a corresponding decrease in reactor vessel outlet temperature.

As a result of pump and reactor coolant system tests, a majority of the orifice plugs were removed from peripheral fuel assemblies prior to startup of TMI-1 and other similar B&W 177 FA plants. This was done to preclude operation with excessive coolant flow through the reactor core. The result was an increase in the maximum core bypass (or leakage) flow conservatively estimated to be 2.3% (from 6.04% to 8.34% of total RCS flow). This increased leakage was not accounted for in those analyses based on design flow (Cycle 1) because it was a direct result of the higher system flow. For those analyses based upon the increased system flow rate, the increased leakage was taken into account, thus for an increase of 6.5% in system flow, the corresponding core flow increase was 4.2%.

Batch 4 fuel assemblies, which will be loaded primarily in peripheral locations for cycle 2 operations, have a slightly lower resistance to flow than do the batch 2 and 3 assemblies. Since the batch 3 fuel is located in the hottest core locations (hot assembly is in batch 3), the result is that the coolant flow through the hot assembly is slightly less than if all assemblies were identical. This difference is conservatively accounted for in the thermal-hydraulic analysis by assuming that the cycle 2 core consists of two batches (116 assemblies) of the less restrictive Mark B4 assemblies and one batch (60 assemblies) of the more restrictive Mark B3 assemblies, as discussed in section 6.1 of the TMI-1 cycle 2 Reload Report. The resulting predicted minimum DNBR is 2.05 (BAW-2) for the 108% overpower, maximum design condition (undensified, including temperature and pressure errors.)

B. Densification Effects

The fuel densification penalty applied to the hot channel for cycle 1 operation was determined by the methods discussed in the Oconee II Fuel Densification Report, BAW-1395, June 197?, page A-5. A conservative slumped and spiked 1.83 outlet peaked axial power shape was used in conjunction with a 1.49 radial-local factor to determine the maximum fuel densification effect on DNB ratio. This reduced hot channel DNB ratio is the basis for establishing the initial conditions for the transient calculations.

The power spike model and densification penalty analyses for cycle 2 are discussed in subsections 4.3 and 6.2, respectively, of the cycle 2 Reload Report. A conservative DNBR penalty for densification was assured by the use of a penalty analysis based upon the most limiting (batch 3) fuel, without consideration of the effects of burnup. Application of the densification penalty results in a reduction in the predicted minimum DNBR for the 108% overpower, maximum design, case from 2.05 to 2.00.

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C. Effect of Open Vent Valve Assumption

For the flux-flow trip setpoint analysis, it has been conservatively assumed that one core barrel vent valve is stuck open. This assumption reduces the effective core flow rate by 4.6% and results in a corresponding reduction in minimum DNB. For second cycle analysis, the effect of this assumption is a reduction in predicted minimum DNBR for the 108% overpower, maximum design case from 2.00 to 1.85. This valve represents the initial MDNBR for the transient analysis described below.

The RADAR (3) computer code is utilized to analyze two isolated channels, representing an average subchannel and the hot subchannel. Input to the first channel includes nominal subchannel and fuel rod geometry. Initial condition inputs to this channel include the nominal input power for a fuel rod, the reactor overpower (1.08%) inlet enthalpy and outlet pressure representative of the 108% oprating condition, including errors, and average channel flow rate. Transient inputs include normalized reactor power and flow versus time. Heat input to the coolant is calculated by a transient fuel pin model which includes 15 radial nodes in the fuel, a calculated fuel-clad gap coefficient, and 2 radial nodes in the cladding. The axial heat input distribution within the fuel is represented by a symmetrical 1.5 peak-to-average cosine axial flux shape distributed over 60 axial nodes. Primary output from this calculation is the pressure drop versus time for the average subchannel.

The second channel analyzed by RADAR represents the hottest subchannel in the core. This channel, and its associated fuel rod, is modeled in the same manner as the first channel with appropriate hot channel factors added. Input power to this channel is higher than that of channel 1 by the maximum design radial x local power factor of 1.783 plus an added factor to account for the densification penalty. The flow rate in this subchannel is calculated for both initial and transient conditions so that the hot channel pressure drop always matches that of the average channel. By analyzing the reactor core in this fashion, the first channel represents the average core response during the transient while the second channel represents the response of the hottest subchannel. The result is a more severe hot channel transient than would be indicated if the transient core flow function were applied directly to the hot channel.

D. Transient Hot Channel Conditions During a Loss of Flow

The flux/flow trip setpoint is derived to protect the core during a one pump coastdown. A one pump coastdown is analyzed because redundant pump monitors are provided which will provide DNB protection for all other pump coastdowns including coastdowns while the plant is in partial pump operation. The pump monitor logic will not cause a reactor trip for the loss of one pump from four pump operation.

The thermal-hydraulic response of the hot channel is calculated by RADAR computer code (BAW-10069) (3). The initial hot channel DNB ratio is set equal to the steady state value with densification and open vent valve effects included. The RADAR output in the form of Hot Channel DNB ratio versus time is the basis for establishing the flux/flow ratio trip setpoint.

E. Rod Bowing Effects

Analysis was performed with the COBRA III-C code to determine the effect of a fuel rod bowing into the hot channel and reducing the flow area of that channel. The results demonstrate that rod bow of the magnitude predicted is adequately compensated for by the flow area reduction factor. Rod bow away from the hot channel was also analyzed. In this analysis, the effect of a power spike was added to the hot rod in the area of the minimum DNBR. This analysis also demonstrated that the current TMI-1, Cycle 2 DNBR results conservatively account for the effects of fuel rod bowing.

III. PROCEDURE FOR DETERMINING FLUX/FLOW SETPOINT

The determination of the flux/flow setpoint is accomplished in four basic steps. The result of these steps is designed to yield a value of the flux/flow ratio that will prevent the minimum hot channel DNBR from going below the limiting design DNBR for the coastdown for which protection is required. These steps are as follows:

A. Total Time Determination

From a plot of minimum DNBR versus time find the time that yields a DNBR of 1.3 for the maximum power level (108%) for the maximum number of pumps lost for which the flux/flow trip must provide protection (one pump for TMI-1, although the original Technical Specifications were based on a two pump coastdown since redundant pump monitors were installed subsequent to the original calculations).

B. Coasting Time Determination

The total time to reach a DNBR of 1.3 minus a conservative value of the total trip delay time gives the maximum allowable coasting time prior to trip initiation.

C. Minimum Flow Determination

From a plot of flow versus time for the coastdown of interest, the percent flow for the maximum allowable coasting time is found. This yields the flow at which trip must be initiated.

D. Flux/Flow Ratio Calculation

The maximum allowable flux/flow ratio is the maximum real power level of interest (108%) minus the power level measurement error (6%) divided by the minimum flow.

IV. CALCULATIONAL RESULTS

Figure 1 shows the flow versus time that is the design basis for the determination of the flux/flow ratio.

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Figure 2 shows the calculated DNBR versus time with the effects of densification included. From this figure it is seen that a DNBR of 1.3 (W-3) is reached at about 3.35 seconds. Using figure 1 and the technique explained previously, this yields a flux/flow ratio of 1.08. This is the value presented in the FSAR Technical Specifications for densified fuel. Figure 3 shows DNBR versus time for a TMI-1 one pump coastdown using the BAW-2 (2) correlation.

Curves are shown for both the cycle 2 analysis and the most recent cycle 1 analysis. Differences in the initial minimum DNBR occur because of the increased flow rate accounted for in cycle 2 (106.5% of design flow) and because of core configuration and fuel assembly modeling differences discussed in the above paragraphs and in section 6.1 of the Reload Report. From figure 3, the limiting design DNBR (1.32 for cycle 1, 1.30 for cycle 2) is reached at 5.45 seconds for both cases. Using the Method defined in III. above, with a trip delay of 1.3 seconds, the maximum allowable flux/flow ratio is then 1.12.

In recent flux/flow setpoint analysis, such as the Oconee Unit 1 third cycle analysis (4), the method defined in III. above, has been refined slightly to include the effect of "DNBR Turnaround". This effect results from the fact that some finite time is required after control rod motion starts before the minimum DNBR is reached. For the TMI-1 setpoint analysis, this effect can be conservatively accounted for by adding 0.5 seconds to the "trip delay" time. Using a value of 1.9 seconds for trip delay, the maximum flux/flow setpoint would then be reduced from 1.12 to 1.11.

It should be emphasized that the above described analyses are based on the assumption that one vent valve is stuck open. This assumption reduces the effective core flow by 4.6%. Elimination of this conservative assumption would have the effect of increasing the calculated allowable flux/flow setpoint by approximately 0.04. This conservatism of the Tech Spec value (1.08) is assured.

V. CONCLUSIONS

Cycle 2 analyses for TMI-1 have been based upon reactor coolant flow measurements taken during the first cycle operation and have incorporated revisions to the standard B&W analysis techniques. The analysis described in this report has demonstrated that the technical specification value of the flux/flow trip setpoint (1.08) is conservative.

REFERENCES

(1) B&W Topical Report BAW-10021, "TEMP-Thermal Enthalpy Mixing Program" April 1970.

(2) B&W Topical Report BAW-10000, "Correlation of Critical Heat Flux in a Bundle Cooled by Pressurized Water," March 1970.

(3) B&W Topical Report BAW-10069, "RADAR-Reactor Thermal and Hydraulic Analysis During Reactor Flow Coastdown," July 1973.

(4) TMI-1, Cycle 2 Reload Report

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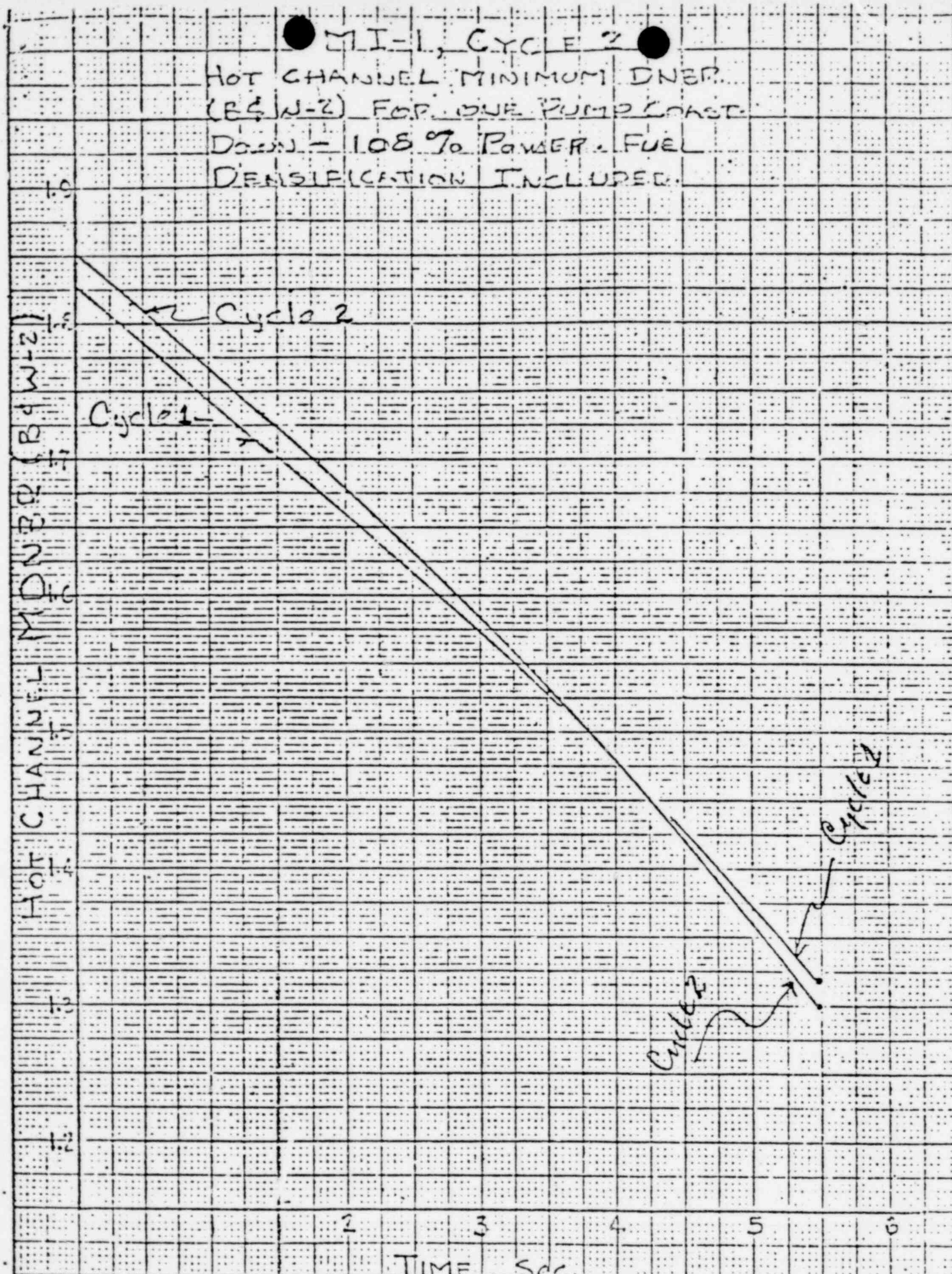

FIGURE 3

POOR ORIGINAL

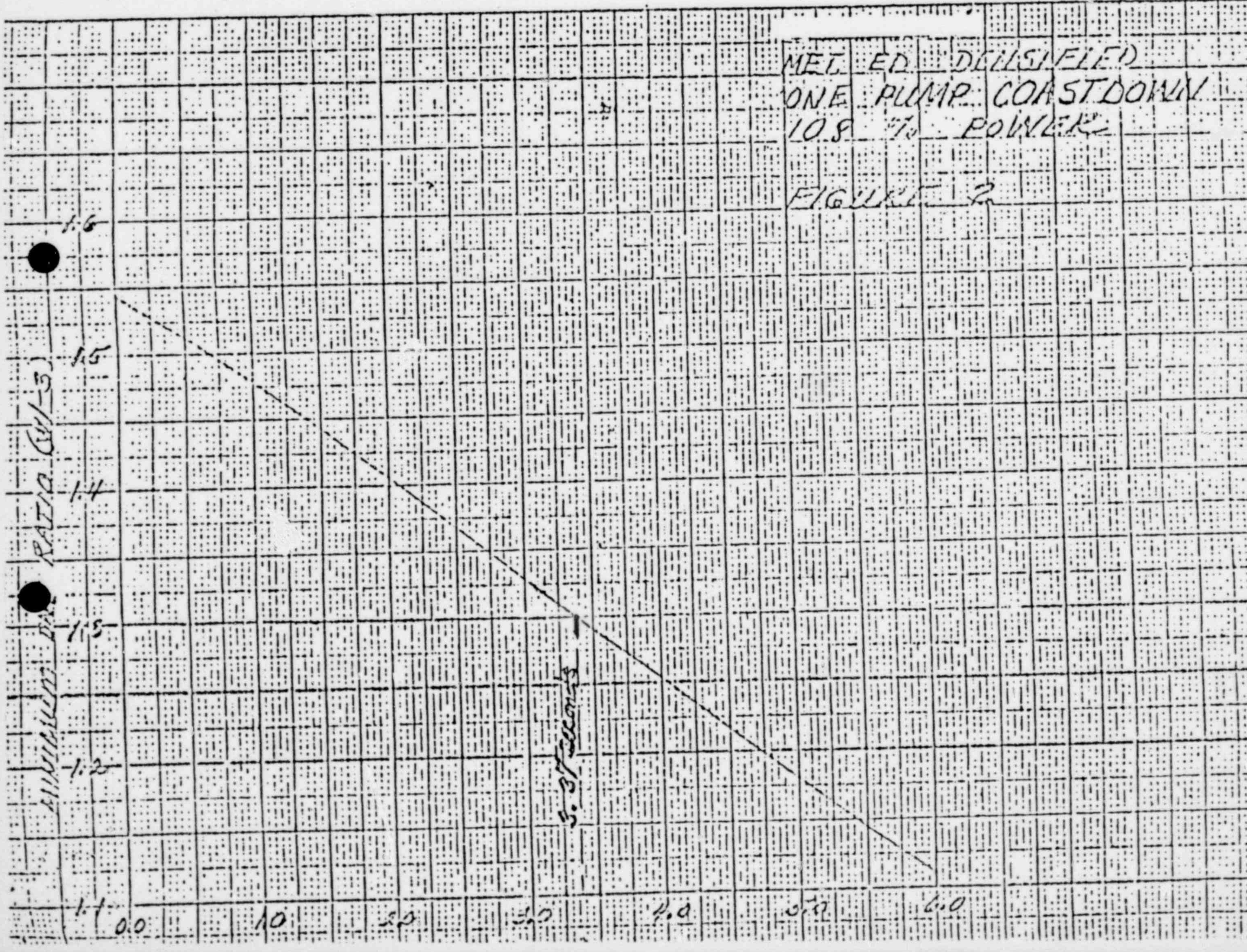
MET ED DEILSFIELD
ONE PUMP COASTDOWN
108 % POWER
FIGURE 2
MINIMUM DNB RATIO (W-3)
1.1
1.2
1.3
1.4
1.5
1.6
0.0
1.0
2.0
3.0
4.0
5.0
6.0
3.37 seconds

FIG 2

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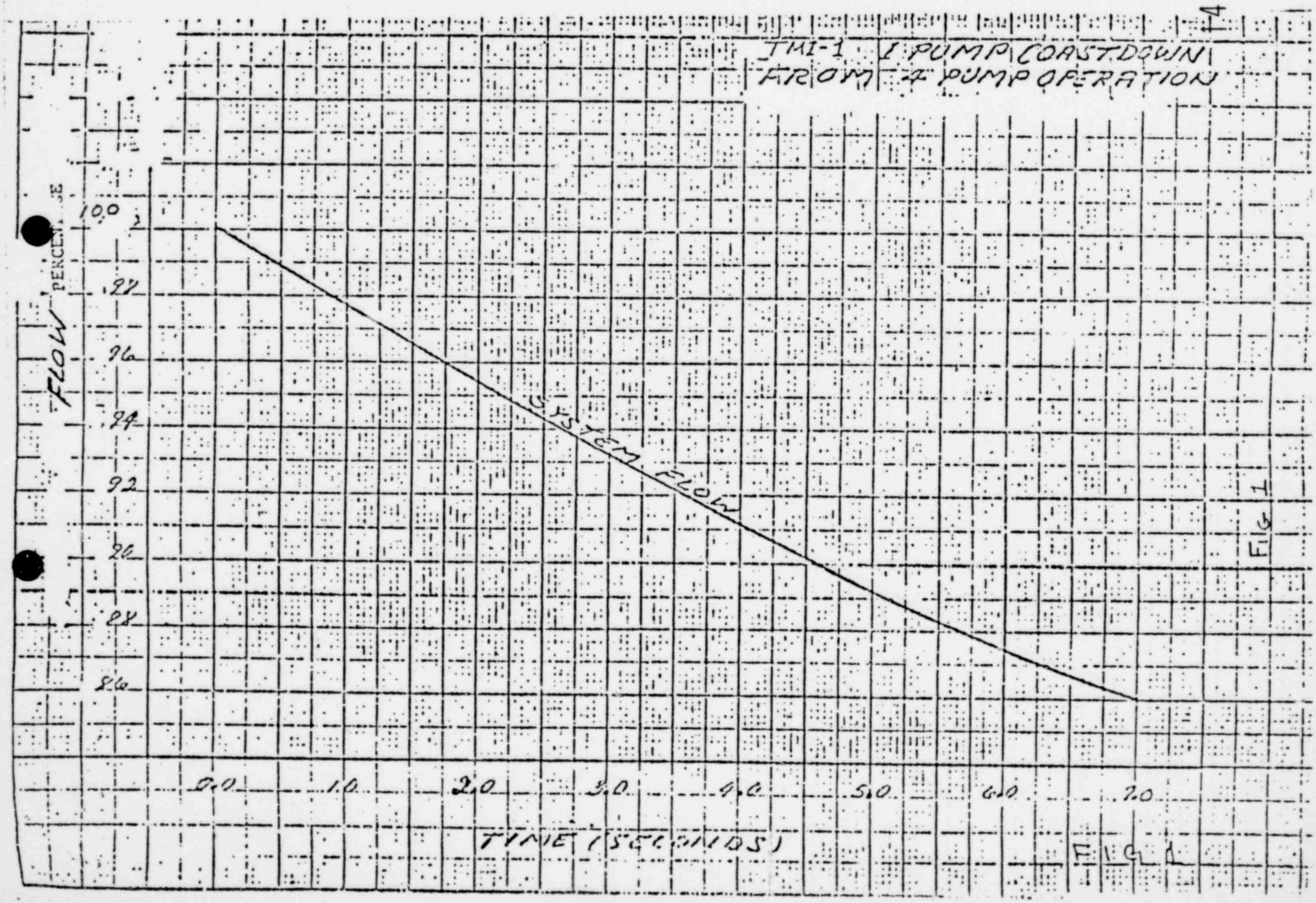

Fig 1

FIG 1