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ATOMICS INTERNATIONAL A Division of North American Aviation, Inc.		NAA-SR-Memo TDR NO. 9514		APPROVALS	
TECHNICAL DATA RECORD		PAGE 1 OF 20			
AUTHOR W. R. Lahs <i>RL</i>		DEPT. & GROUP NO. 726-60		DATE 2-7-64	
		GO NO. 7561		<i>RS Hart</i>	
TITLE Long Term SNAP 10A Reactor Operation Analysis		S/A NO. 1100		TWR	
		SECURITY CLASSIFICATION			
PROGRAM SNAP 10A		(CHECK ONE BOX ONLY)		(CHECK ONE BOX ONLY)	
		UNCL. <input type="checkbox"/> AEC <input type="checkbox"/> DOD <input type="checkbox"/>		RESTRICTED DATA <input checked="" type="checkbox"/>	
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		AUTHORIZED CLASSIFIER SIGNATURE <i>RS Hart</i>		DATE 2-24-64	
		STATEMENT OF PROBLEM			
		Investigation of the long term SNAP 10A reactor operating history, considering the long term reactivity effects and the feedback to reactor power caused by the heat transfer characteristics of the system.			
		ABSTRACT:			
		This report presents the description and results of the long term reactor operation code used to solve the above problem. The code represents the heat transfer and fluid flow in a five node reactor representation coupled with a radiator heat transfer equation and a reactivity relationship. The model is solved at discrete points in time based on the assumption that for the operating histories considered, time dependent terms involving changes in heat capacities can be neglected without significant error. The code does <u>not</u> solve for any power or temperature transients but instead calculates reactor temperatures and powers under conditions of relatively slow coolant flow and/or radiator emissivity coating degradations. A 0.1 year time increment is generally used between calculation points; however, a 0.01 year increment is sometimes used to eliminate convergence problems. Assuming the SNAP 10A system with time dependent NaK flow and emissivity coating degradation as input data, the resulting reactor inlet and outlet coolant temperatures and the reactor power are presented as a function of time.			

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NO. NAA-SR-Memo-9514DATE February 7, 1964PAGE 2 OF 20PURPOSE

The study was initiated to determine the long term SNAP 10A reactor power operation and associated reactor coolant outlet temperature primarily under conditions of degrading NaK flow. The power operation could subsequently be translated into the time dependent fission product inventory used as a starting point for hazard analysis. Coolant outlet temperature would be of value in determining the effectiveness of the temperature actuated band release device as a reflector ejection initiating mechanism or, for that matter, any other safety systems which depend on the magnitude or variation in outlet temperature.

PROCEDURE AND ASSUMPTIONS

The reactor model incorporated in the code is a simplified version of that presented in Reference 1 with the exception that terms involving time derivatives of temperature were neglected. One equation representing heat balance through the radiator served as a relation between reactor outlet and inlet temperatures. Transport time delays, heat capacity changes, etc., were not necessary for the cases studied and, in fact, contribute insignificantly to the resulting steady state values.

The reactivity equation used is essentially a reactivity balance of all the incorporated separate contributions:

- a) grid plate temperature change
- b) average fuel temperature change
- c) losses from fission product production and uranium burnup
- d) equilibrium xenon changes
- e) hydrogen leakage
- f) hydrogen redistribution
- g) samarium burnout

Relationships for (f) and (g) were derived from data obtained from Reference 2. Existing relationships were used for the other reactivity contributions. These were also generally supplied by Reference 2.

Coupling the relationships above at a particular time, the procedure of solution is one of minimized brute force. An educated guess is made for the reactor coolant inlet temperature. Based on the NaK coolant flow at the time under study, the heat balance equation through the radiator yields a coolant outlet temperature. Assuming all the reactor power is transferred through the coolant (not a bad

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assumption for the cases under study, but one which is being eliminated by intended code modifications), reactor power and all the nodal reactor fuel and coolant temperatures are computed. A reactivity balance is then attempted based on these results. Unless the balance falls between set tolerances (usually $-0.1\% \leq R \leq 0.1\%$), the coolant inlet temperature estimate is revised accordingly. Rapid convergence has resulted (less than 5 guesses) for all the cases studied thusfar. The code then advances to the next time point repeating the same procedure after accumulating irreversible reactivity changes-hydrogen leakage, fission product buildup, uranium burnup, and samarium burnout.

DESCRIPTION OF MODEL

Figure 1 shows a schematic of the reactor model. The power transferred to the coolant from the fuel is represented by the following equations (See table of nomenclature):

$$P_i \frac{n(t)}{no} = \frac{U_{fA_f}}{5} (T_{fi} - \bar{T}_i) \quad i = 1, 2, \dots, 5$$

where i refers to a particular node.

Heat transferred out of the node by the coolant is represented by

$$\frac{U_{fA_f}}{5} (T_{fi} - \bar{T}_i) = W(t)W_o C_c (T_{i+1} - T_i) \quad i = 1, 2, \dots, 5$$

The heat balance across the radiator is determined as follows:

Assume a small length of radiator dl . Heat lost by the coolant in passing through dl is radiated to space. No conduction or change in heat capacities is assumed significant. Then:

$$-W(t)W_o C_c dT = K(l) \epsilon(t) (T^4 - T_s^4) dl$$

or since $T \gg T_s$

$$-W(t)W_o C_c dT \cong K(l) \epsilon(t) (T^4) dl$$

where:

T = radiator temperature

T_s = space temperature

$\epsilon(t)$ = radiator emissivity at time (t)/
emissivity at $t = 0$

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TABLE OF NOMENCLATURE

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P_i	fraction of full power produced in node i
$\frac{n(t)}{n_o}$	total power at time t/total power at time t = 0
$U_f A_f$	total heat transfer coefficient
T_{fi}	fuel temperature of node i
\bar{T}_i	average coolant temperature of node i
$W(t)$	fraction of full flow at time t
W_o	coolant flow #/sec
C_c	specific heat $\frac{Kw-sec}{\# \text{ } ^\circ F}$
T_i	internode temperatures (see Figure 1)
$K(l)$	an unknown function of radiator length
$\epsilon(t)$	emissivity at time (t)/emissivity at t = 0

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$T_{1 \text{ int}}$	initial reactor coolant inlet temperature
$T_{6 \text{ int}}$	initial reactor coolant outlet temperature

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\bar{T}_f	average of the five node fuel temperatures at time t
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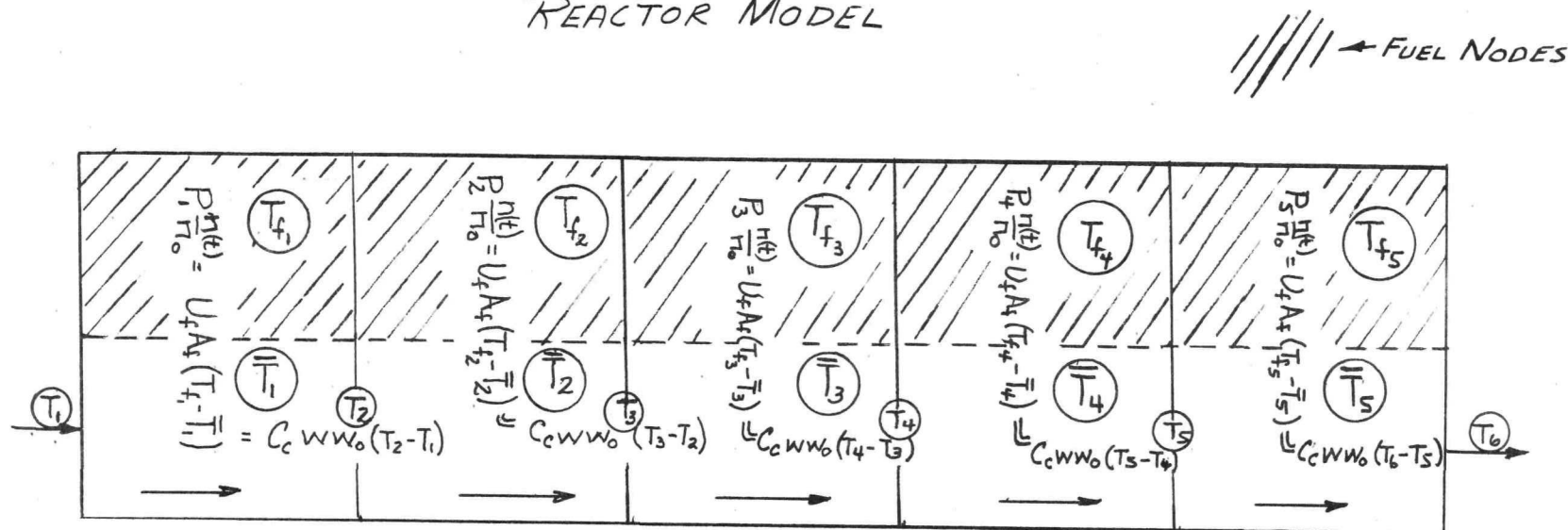
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$\bar{T}_{f \text{ int}}$	average of the five node fuel temperatures at time $t = 0$
α_f	fuel temperature coefficient
α_{gu}	upper grid plate coefficient
α_{gl}	lower grid plate coefficient
$\frac{n_i}{n_o}$	power during time increment Δt_j /power at time $t = 0$
Δt_j	time increment
K_2, K_3, K_4, K_5, K_6	constants
$T_{f_{ij}}$	fuel temperature of node i during time increment Δt_j
ΔT_c	coolant temperature difference across core during time increment Δt_j
$\Delta T_{c \text{ int}}$	coolant temperature difference across core at time $t = 0$ (111°F)

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FIG 1
REACTOR MODEL



Using the axial power distribution equation of reference 1 $\frac{P}{P_{max}} = \cos \frac{\pi z}{35.5}$ z = distance from center.

$$P_1 = 0.11 P_T$$

$$P_2 = 0.24 P_T$$

$$P_3 = 0.30 P_T$$

$$P_4 = 0.24 P_T$$

$$P_5 = 0.11 P_T$$

$$P_T = 33.5 \text{ kw}$$

$\frac{n(t)}{n_0}$ = power fraction at time t

$U_f A_f$ = total heat transfer coefficient (ret. 3) kw/of

C_c = specific heat $\frac{\text{kw-sec}}{\text{# of}}$

W = fraction of full flow at time t

W_0 = flow $\text{#}/\text{sec}$

- T_{fi} Nodal fuel temperature
- \bar{T}_i Average nodal coolant temperature
- T_i Boundary nodal coolant temperature

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Separation of variables yields

$$\frac{-dT}{T^4} = K(l) \frac{\epsilon(t)}{W(t)} \frac{dl}{WoCc}$$

Integrating between a reactor outlet temperature T_6 and inlet temperature T_1

$$\int_{T_1}^{T_6} \frac{dT}{T^4} = \frac{\epsilon(t)}{W(t)Cc} \int_0^1 \frac{K(l)}{Wo} dl$$

where the minus sign has been eliminated by changing the limits of integration.

Therefore,

$$\frac{-1}{3 T^3} \Big|_{T_1}^{T_6} = \frac{\epsilon(t)}{W(t)Cc} \int_0^1 \frac{K(l)}{Wo} dl$$

or

$$\frac{1}{T_1^3} - \frac{1}{T_6^3} = \frac{3\epsilon(t)}{W(t)Cc} \int_0^1 \frac{K(l)}{Wo} dl$$

But values for T_1 and T_6 are known when initial equilibrium is reached ($\epsilon(t) = 1$; $W(t) = 1$) subsequent to reactor startup. Therefore, the right side of the equation for all inlet and outlet conditions is:

$$\frac{1}{T_1^3} - \frac{1}{T_6^3} = \frac{\epsilon(t)}{W(t)} \left(\frac{1}{T_{1int}^3} - \frac{1}{T_{6int}^3} \right)$$

The limitations of the model are fairly obvious:

- (1) Heat transferred into a radiator node is by fluid flow only. Therefore, as other means of heat transfer (conduction) become of comparable magnitude, the model breaks down.
- (2) The radiator node temperature is assumed much greater than space temperature.

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The final relationship in the long term reactor operation model sets equal to zero the sum of all changes in reactivity from initial conditions at full power subsequent to startup. These changes include the following contributions and their source.

$$(a) \int_{T_{fint}}^{\bar{T}_f} \alpha_f dT \quad \text{average fuel temperature change}$$

$$(b) \alpha_{gu} (T_6 - T_{6int}) \quad \text{upper grid plate temperature change}$$

$$(c) \alpha_{gl} (T_1 - T_{1int}) \quad \text{lower grid plate temperature change}$$

$$(d) K_2 \sum_{j=1}^m \frac{n_j}{n_o} \Delta t_j \quad \text{cumulative effect of fission product poisoning and uranium burnup}$$

$$(e) K_3 \left(1 - \frac{n_i}{n_o} \right) \quad \text{change in equilibrium xenon}$$

$$(f) K_4 \sum_{i=1}^5 \sum_{j=1}^m \frac{T_{fij} - 950}{50} \Delta t_j \quad \text{cumulative effect of hydrogen leakage}$$

$$(g) K_5 \left(\frac{\Delta T_c - \Delta T_{cint}}{100} \right)^{1.54} \quad \text{hydrogen redistribution}$$

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$$(h) \sum_{j=1}^m (K_6 - AI_j) (1 - e^{-0.348 \frac{n_j}{n_0} \Delta t_j})$$

$$\text{where } AI_j = \sum_{j=1}^{m-1} (K_6 - AI_{j-1}) (1 - e^{-0.348 \frac{n_j}{n_0} \Delta t_j})$$

and $AI_1 = 0$

samarium burnout

Term (a) is the reactivity input resulting from changes in average fuel temperature. T_{fint} for the SNAP 10A system is 982.7°F. The fuel temperature coefficient, α_f , was determined by the following equation:

$$\alpha_f = - \left[0.074 + \frac{0.066 T}{1000} \right] \text{ } \phi/^{\circ}\text{F}$$

Term (b) used values of -0.06 $\phi/^{\circ}\text{F}$ and 1010°F for α_{gu} and T_{6int} respectively. Likewise term (c) used a value of -0.04 $\phi/^{\circ}\text{F}$ for α_{gl} and 899°F for T_{lint} . The constant, K_2 , in term (d) was based on an estimated loss of 6 ϕ /yr under conditions of constant reactor power (33.5 Kwt). The

$$\sum_{j=1}^m \frac{n_j}{n_0} \Delta t_j$$

(where m is such that $\sum_{j=1}^m \Delta t_j$ equals the time point of interest)

represents the sum of the products of the normalized power fraction multiplied by the time increment Δt_j .

The xenon equilibrium change from cold critical to full power (33.5 Kwt) contributes a 14 ϕ loss in reactivity. Therefore, reactivity changes from equilibrium at full power can be represented by term (e), setting K_3 equal to +14 ϕ .

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The rate of hydrogen leakage as a function of fuel temperature was approximated from the function presented in Figure 2 (Reference 3). This function was normalized to yield a leakage rate of 3 ϵ /yr if the initial fuel temperature profile was maintained for one year. With this criteria K_4 was determined as 1.62 ϵ /yr.

Term (g) was originally a very crude approximation for reactivity loss due to hydrogen redistribution. The -14.86 ϵ portion of this term was deduced from redistribution reactivity losses during startup; namely, from cold critical to equilibrium conditions at full power (coolant ΔT across core $\sim 100^\circ\text{F}$ - 950°F average coolant temperature). The only other data on hydrogen redistribution which existed at this time was for other ΔT s (200° , 300° , and 400°) for the same average coolant temperature and power. The first portion of term (g) is essentially a function derived from these four data points ($K_5 = 4.18\epsilon$). This function was believed at first to be an underestimate of the negative reactivity contribution from hydrogen redistribution as the average coolant temperature declined from 950°F . However, the more recent data of Table 1 indicated that the underestimate resulting from lower average coolant temperature was counterbalanced. This counterbalance was due to the fact that the data points used were for constant power of 33.5 Kw. In reality a reduction of power would cause an overestimate of the negative reactivity contribution of hydrogen redistribution. Thus, a happy ending. One final point about this term should be mentioned. As average fuel temperature decreases, approaching 700 - 800°F , the time for the hydrogen redistribution to take place becomes equal in magnitude to the time increment (0.1 yr). Thus, below 800°F the negative hydrogen redistribution effect is, indeed, an overestimate.

Table 2

Negative Hydrogen Redistribution Reactivities (ϵ)

Coolant Inlet ($^\circ\text{F}$) Power (Kwt)		<u>800</u>	<u>900</u>	<u>1000</u>	<u>1100</u>	<u>1200</u>	<u>1300</u>
$\Delta T = 100^\circ\text{F}$	32.5	17.34	14.86	12.88	11.27	9.94	8.84
	50.0	25.22	21.71	18.88	16.57	14.66	13.07
	100.0	46.33	40.12	35.08	30.94	27.49	24.59
$\Delta T = 200^\circ\text{F}$	32.5	21.12	17.87	15.31	13.26	11.60	10.23
	50.0	28.39	24.22	20.91	18.23	16.04	14.22
	100.0	47.88	41.33	36.05	31.72	28.13	25.12
$\Delta T = 300^\circ\text{F}$	32.5	27.04	22.64	19.22	16.50	14.31	12.53
	50.0	33.72	28.52	24.43	21.15	18.49	16.29
	100.0	51.68	44.39	38.54	33.78	29.15	26.57

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FIGURE 2

HYDROGEN LEAKAGE
REACTIVITY LOSS RATE
VS. TEMPERATURE

(NORMALIZED TO 0.8^{cc}/HR/ELLEMENT AT 1200°F)

HYDROGEN LEAKAGE LOSS RATE (1/HR)

10⁻⁴
10⁻³
10⁻²
10⁻¹
10⁰
10¹
10²
10³
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10¹⁰
10¹¹
10¹²
10¹³
10¹⁴
10¹⁵
10¹⁶
10¹⁷
10¹⁸
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Term (h) represents the positive reactivity resulting from samarium burnout. The initial excess samarium poisoning over equilibrium samarium was evaluated as 44.5¢. K_6 was set equal to 44.5¢ and term (h) then represents the cumulative reactivity input as a function of time and power.

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RESULTS

Examination of Reference 4 revealed a minimum flow degradation to date of 0.3 gpm/1000 hours. Reference 5 recommended a 10% emissivity coating degradation over a 5 year period. Using these conditions as input data, Figure 3 (solid lines) shows the coolant inlet and outlet temperature, average fuel temperature, and the reactor power fraction as a function of time. The early drop in temperature and power results from initial hydrogen redistribution. The following gentle increase is caused by samarium burnout. Figure 4 shows the reactivity contributions from hydrogen leakage, hydrogen redistribution, and samarium burnout as a function of time.

Since average fuel element temperature drops below 800°F at 4.53 years, the reactivity effect from hydrogen redistribution after this time is probably overestimated and, thus, the coolant temperatures are too low. In order to obtain an upper limit for coolant outlet temperature (from a lower limit on hydrogen redistribution negative reactivity loss), a second case was programmed which considered no reactivity loss from hydrogen redistribution following the initial 14.86¢. The results are also shown on Figures 3 and 4 as dotted lines. An increasing difference in outlet coolant temperature with time should be noted. This fact is readily reconciled when one notes the rapid change in negative reactivity input from hydrogen redistribution between the two cases. The resulting higher average fuel temperatures increases hydrogen leakage while samarium burnout is almost unchanged.

A compilation of the code for the first case discussed is included as Appendix A.

CONCLUSIONS

The conclusion reached from this preliminary study is illustrated by Figure 3; namely, that slow coolant flow degradation does not seem to result in the high coolant outlet temperatures previously expected to cause expansion compensator failure or fuel element rupture. Although a complete transition to a NaK stagnation mode (no flow) was not possible with this simplified model, the important effect of hydrogen redistribution is indicated. Further, the heat transfer effects neglected (i.e., conduction) will, at low flow, generally further inhibit rapid increases in outlet temperature. It should be noted that work in progress may demonstrate the transition of interest.

Finally, as estimates of pump failure mode become more sophisticated, the most probable fission product inventory will be exactly determined.

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Obviously, the inventory from full power operation for 10 years expediently used in the Final Safeguards Report - SNAP 10A Flight Tests (NAA-SR-774) seems very conservative from a safety viewpoint.

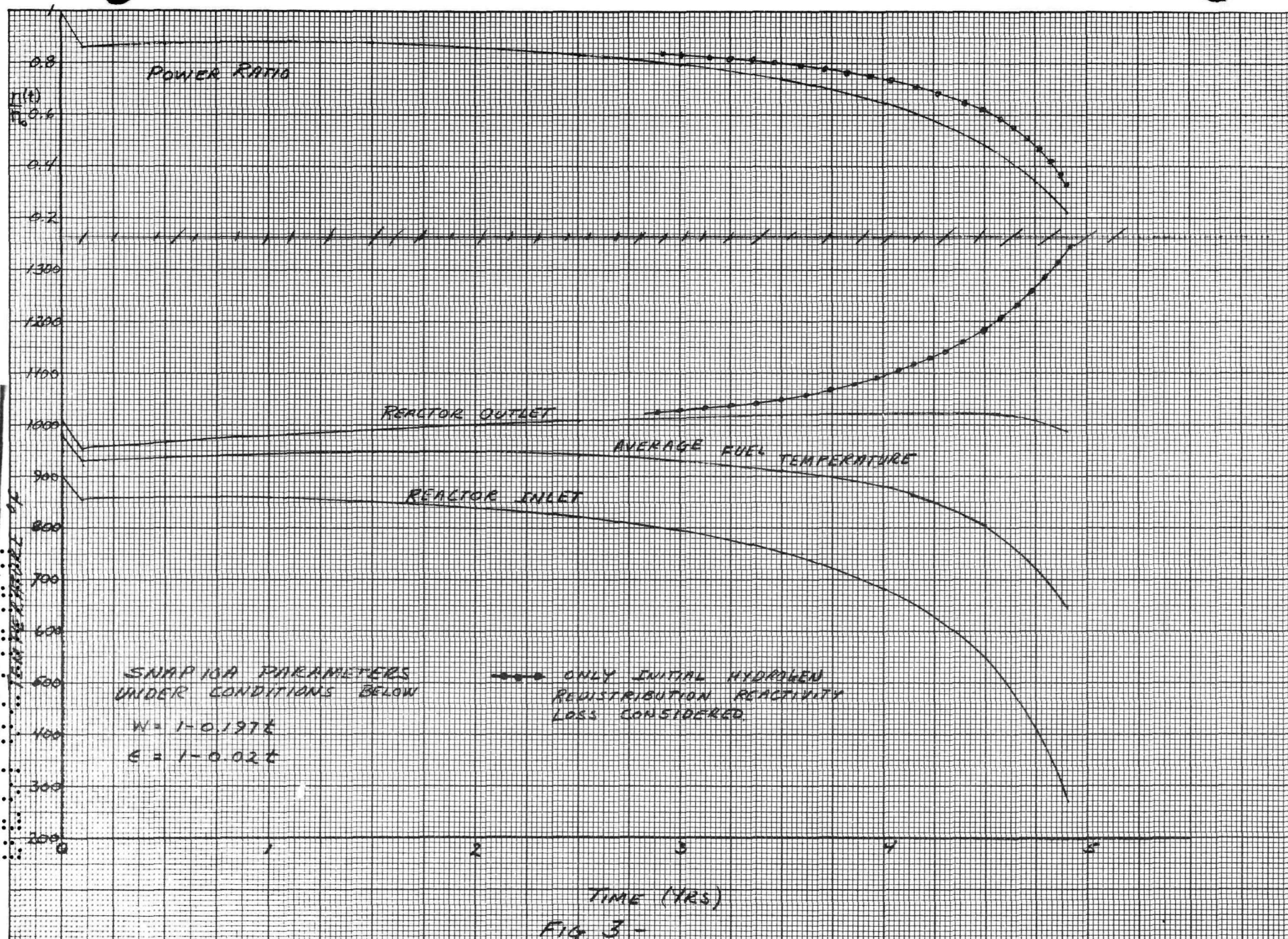
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2. K. Birney Personal Communication with Author
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L. Lepisto Analysis for Pumps Dev. #8, FD-1,
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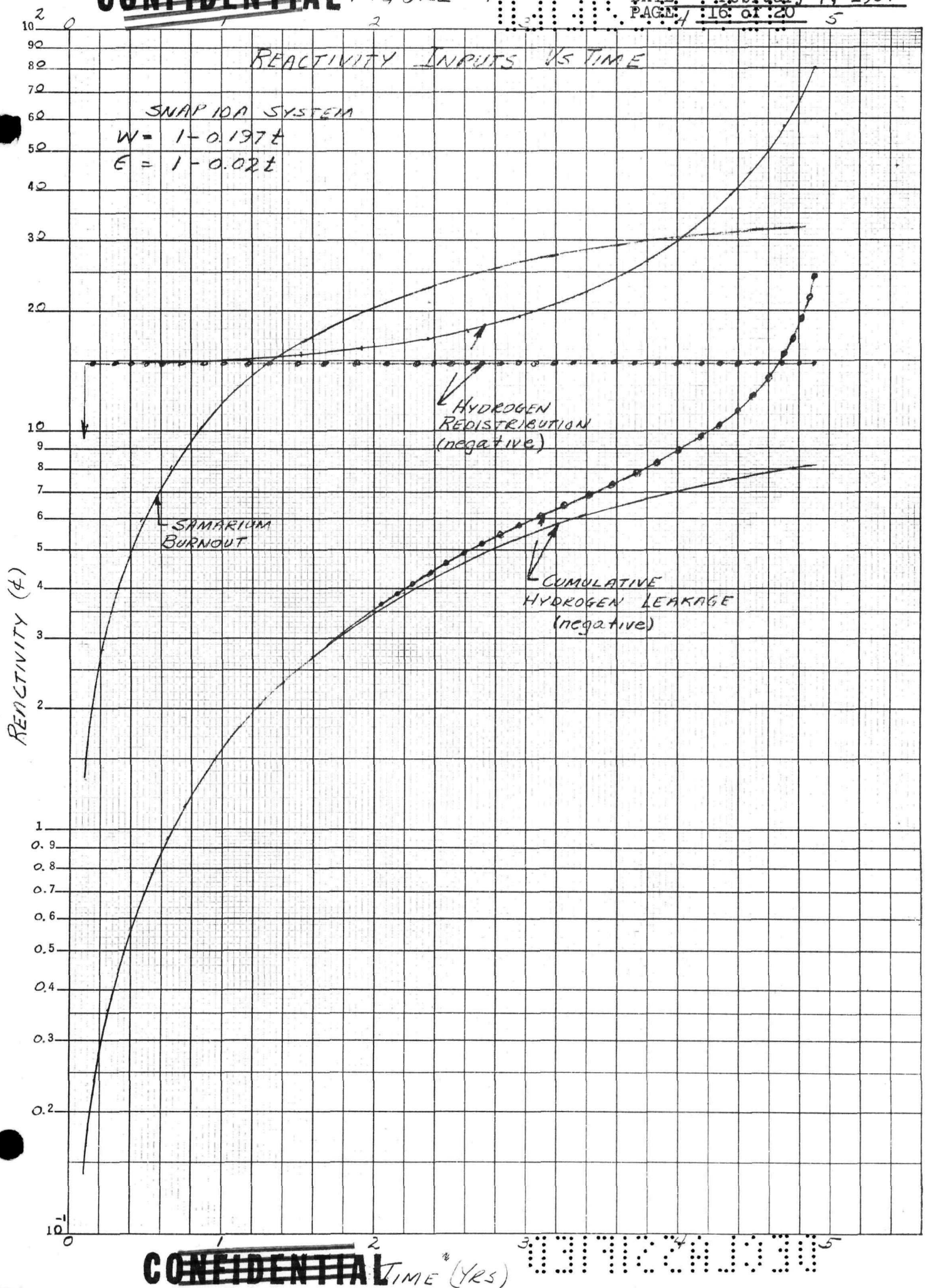
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C	LONG TERM REACTOR OPERATION	00000002
	DIMENSION T(6),TBAR(5),P(5),A(5),TF(5),C0E6(6),TITLE(12)	00000003
	COMMON PT,TIN,TOUT,W0,DELTIM,TMAX,A,C0E6	00000004
	CALL READ	00000008
500	READ INPUT TAPE 5,1,TITLE	00000009
1	FORMAT(12A6)	00000010
	CALL READ	00000011
	SUMENI=DELTIM	00000012
	CAY4=.324*DELTIM	00000013
	GUESS=50.	00000015
	TIME=DELTIM	00000016
	DEL=0.	00000017
	WRITE OUTPUT TAPE 6,2,TITLE	00000018
2	FORMAT(1H1 12A6)	00000019
	TOL=.1	00000021
	TMAX=TMAX+.01	00000022
	TIP=TIN	00000023
	C2=(TIN+460.)*(-3)-(TOUT+460.)*(-3)	00000040
	DELTIC=TOUT-TIN	00000050
	DO 420 I=1,5	00000052
420	P(I)=A(I)*PT	00000054
	TFBAR I=982.7	00000990
	T01=TOUT	00000991
	T11=TIN	00000992
	N4=1	00000995
	LL=0	00000996
	T(1)=TIN	00000997
	SUMAI=0.	00000998
	DELT I=.348*DELTIM	00000999
	SUM I=0.	00010100
104	T(1)=T(1)-GUESS	00001000
	W=1.-.197*TIME	00001001
	E=1.-.02*TIME	00001002
	CI=44.5-SUMAI	00001003
	L=1	00001005
	DO 105 KKK=1,30	00001005
79	TRANK=T(1)+460.	00001006
	RAD=1.-C2*E/W*TRANK**3	00001007
	IF (RAD) 80,80,84	00001008

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80	T(1)=(.992*W/(C2*E))**.3333-460.	00001009
	IF (ITRY) 81,81,82	00001010
81	ITRY=1	00001011
	GO TO 79	00001012
82	WRITE OUTPUT TAPE 6,83,KKK,T(1),TP,DELP,TM,DELM	00001013
83	FORMAT(41H-CONVERGENCE OF T(1) WAS NOT ACCOMPLISHED/14,1P5E15.5)	00001014
	CALL DUMP	00001016
	GO TO 500	00001015
84	T(6)=TRANK/RAD**.3333-460.	00001016
	TGAP=(T(1)+T(6))/2.+26.	00001017
	DEL I=DEL	00001018
	DELTC=T(6)-T(1)	00001020
	EN=W*DELTC/DELTCI	00001030
	CON=P*EN/2.	00010404
	C1=EN/(.211*W*W0)	00010105
	DO 100 I=1,4	00010110
	T(I+1)=P(I)*C1+T(I)	00010130
100	TBAR(I)=(T(I)+T(I+1))/2.	00010140
	TBAR(5)=(T(5)+T(6))/2.	00010150
	TCBA =0.	00010160
	DO 110 I=1,5	00010180
110	TCBA =TCBA +TBAR(I)	00010190
	TCBAR=TCBA /5.	00010210
	TFBAR=(T(I)+T(6))/2.+26.	00010400
	TGAP=(TFBAR+TCBAR)/2.	00010402
	DO 130 N=1,20	00010406
	UFAF= (((C0E6(6)*TGAP+C0E6(5))*TGAP+C0E6(4))*TGAP+C0E6(3))*TGAP	00010410
	1+C0E6(2))*TGAP+C0E6(1)	00010420
	TGAP1=CON/UFAF+TCBAR	00010422
	IF (ABS(TGAP1-TGAP)-1.) 131,131,130	00010426
130	TGAP=(TGAP+TGAP1)/2.	00010430
131	TFBA=0	00010432
	CONST=5.*EN/UFAF	00010440
	SUM=0	00010442
	DO 120 I=1,5	00010450
	TF(I)= P(I)*CONST+TBAR(I)	00010460
	SUM=SUM+2.**((TF(I)-950.)/50.)	00010462
120	TFBA =TFBA +TF(I)	00010470
	PAR=(DELTC-DELTCI)/100.	00010472

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Q=ABSF(PAR)/PAR                                00010473
134 SUMJ=SUM                                     00010478
    TFBAR=TFBA /5.                               00010480
    DBL SUM=SUMI+SUMJ                             00010500
    ALPHA6=-.074-6.6E-5*(TFBAR+TFBARI)/2.        00010510
    CAY5=CAY4*DBLSUM                             00010515
    AI=CI*(1.-EXP(-EN*DELT))                     00010516
    SUMMAI=SUMAI+AI                              00010517
    DEL=ALPHA6*(TFBAR-TFBARI)-.06*(T(6)-T0I)-.04*(T(1)-T1I)+14.*(1.-EN 00010520
1)-6.*SUMENI-CAY5-4.18*Q*ABSF(PAR)**1.54+SUMMAI-14.86
    LIM=KKK                                       00010540
401 WRITE OUTPUT TAPE 6,325,TIME,T(1),DEL        00010543
400 IF (ABSF(DEL)-T0L) 160,160,320               00010547
320 GO TO (151,170),L                           00010550
325 FORMAT (OPF15.1,1P2E15.5)                   00010552
151 L=2                                           00010570
    T1=T(1)                                       00010575
    IF (DEL) 175,160,180                         00010580
175 T(1)=.99*T(1)                                00010590
    GO TO 105                                     00010600
180 T(1)=1.01*T(1)                               00010610
    GO TO 105                                     00010620
170 T2=T(1)-DEL*(T(1)-T1)/(DEL-DELI)            00010630
    T1=T(1)                                       00010632
    T(1)=T2                                       00010634
105 CONTINUE                                     00010640
160 SUMI=SUMI+SUMJ                               00010650
    SUMAI=SUMAI+AI                              00012029
    WRITE OUTPUT TAPE 6,600,TIME,EN,DEL,T(1),T(6),TFBAR,CAY5,TBAR,T,TF 00012000
1,SUMAI,LIM                                     00012001
600 FORMAT(54H- TIME N/NO DEL T(1) T(6) TFBAR HYDROGEN /00012005
16H (YR)40X7HLEAKAGE /OPF5.1,OPF7.3,OPF8.3,OPF7.0,OP2F8.0,1PE11.2/00012010
242X7HTBAR(I)/OP5F15.2/43X4HT(I)/OP6F15.2/43X5HTF(I)/OP6F15.2 00012015
3/1H01PE14.5,14)                               00012016
    TIME=TIME+DELTIM                             00012030
332 SUMENI=SUMENI+EN*DELTIM                     00012031
    IF (LL) 210,210,215                         00012032
210 LL=1                                          00012034
    GUESS=2                                       00012036

```

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1619

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GO TO 104
215 IF (TIME-TMAX) 202,202,500
202 GUESS=TIP-T(1)+2.
TIP=T(1)
ITRY=0
GO TO 104
END(1,0,0,0,0,0,1,0,0,1,0,0,0,0,0)

00012038
00012040
00012044
00012045
00012046
00012050

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