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TO: Distribution

FROM: T. E. Cole

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Shut-down Cooling of ORR

The purpose of this memo is to present information which may be used in estimating the magnitude of the shut-down cooling problem in the ORR at various power levels. Using previously estimated data for the MTR (IDO-16443), a rough estimate of the space distribution of heat release after shut-down is presented.

As a basis on which to start this investigation, it is assumed that for the ORR the total energy release is 197.5 Mev per fission (neglecting neutrinos). This value is taken from IDO-16443 and is based on an MTR type reactor - i.e., aluminum fuel plate, Be reflected, etc. - and is based on a 3 x 9 core operating in the steady state. Of the 197.5 Mev, 190 Mev comes from the fission process, 7.5 Mev from neutron capture reaction in U-235, Al, H_2O, Xe, Sm and Be and from the Be(n-2n) reaction. The figure of 197.5 Mev/fission is equivalent to 3.16×10^{10} fiss/watt-sec.

A recent study of the decay of fission products has been made by the Internuclear Company for the ANP Department of the General Electric Company and is reported in APEX-448. This study was made using recent data from both experimental and analytical sources to arrive at beta and gamma fission product activity with much greater accuracy than previously possible for time ranges of one second to a few hours after a given history of fission in a nuclear reactor.

Using the data from APEX-448 and the card decks supplied by the Internuclear Company for IBM machine calculation, H. C. Claiborne and T. B. Fowler of the REE Division ran machine calculations on the decay of fission products following various reactor operating times. Their calculations were based on 1 Mw power level assuming $3.38 \cdot 10^{10}$ fiss/watt-sec. The data are presented for various decay times covering the range of 1 to 10^6 seconds. The gamma ray data are presented for 12 energy groups; the beta data are presented as a single group. In order to convert the data to the case of an MTR type reactor, it is necessary to multiply by 3.16/3.38 to take into account the difference in total energy per fission.

Fig. 1 is a curve of the total gamma energy release following shut-down of a reactor. The data are presented in terms of fraction of full power and the several curves are labeled according to length of operating time at constant power. Fig. 2 is a similar curve for beta energy release.

Fig. 3 is a curve based on data taken by J. C. Gundlach, March 4, 1959, on shut-down of the ORR from 20 Mw at the end of a normal run. These data were obtained using a compensated ionization chamber, a Log N amplifier and a Brown recorder having a one second full scale drive system. According to Gundlach, later data indicate that this curve overestimates the neutron flux in the core following shut-down and thus gives an overestimate of the power generated; however, this is the best datum presently available.

Fig. 4 is a curve of the total heat release following shut-down. The beta, gamma and neutron curves are added together to obtain this curve. Data for beta and gamma heat releases are based on 800 hr. operation.

Table I presents the data for Figs. 1, 2, 3 and 4 in tabular form.

At this point it is interesting to note that if there were no heat transfer requirements and if the problem were simply one of removing heat which was released to water, the required flow rate for a given ΔT would decay following the curve of Fig. 4. For a 30°F ΔT the flow rate required, in gpm, could be obtained by multiplying the fraction of full power as shown in Fig. 4 by 228 x P where P is the normal full power in megawatts. Another interesting observation may be deduced from the heat capacity of the core. For a full ORR core assumed to be made up of 28 fuel elements and 35 Be reflector pieces, each 24 inches long, the heat capacity is approximately: Be, 210 BTU/°F; Al, 53 BTU/°F and H_2O, 220 BTU/°F. If this assembly is considered to heat up uniformly and be insulated by the core box boundaries, the approximate rate of temperature rise in °F per second, at any instant, can be obtained by multiplying the fraction of full power, as given in Fig. 4, by 1.68 P where P is expressed in megawatts normal full power.

Table I

Heat Release Following Shut-down
(Fraction of Full Power)

Time after Shut-down Seconds	γ 1 hr. Opr.	γ 10 hr.	γ 100 hr.	γ 800 hr.	β 1 hr.	β 10 hr.	β 100 hr.	β 800 hr.	Neutron	Total @ 800 hr.Opr.
1	.026	.0297	.031	.0323	.032	.035	.0368	.0375	.021	.0908
2	.0249	.0282	.0299	.0308	.029	.0323	.0337	.0344	.0172	.0824
4	.023	.0264	.028	.0289	.0252	.0284	.0298	.0305	.014	.0734
10	.0199	.0233	.025	.0258	.02	.0233	.0247	.0253	.0086	.0597
20	.0173	.0207	.022	.0232	.016	.0197	.0211	.0218	.0055	.0505
50	.0137	.017	.0186	.0195	.0123	.0156	.017	.0176	.00239	.0395
100	.011	.0143	.016	.0169	.0096	.0128	.0143	.0149	.00082	.0326
200	.0085	.0117	.0133	.0143	.0072	.0104	.0118	.0124	.00029	.027
500	.00549	.0086	.0102	.0111	.0047	.0077	.0091	.00976	-	.02
700	.00453	.0076	.00912	.0101	.0039	.0069	.0083	.0089	-	.018
1,000	-	.0065	.00807	.00904		.006	.00737	.0081	-	.0162
2,000	.0014	.0047	.0063	.00722	.0019	.0045	.0058	.00647	-	.0128
5,000	.000918	.0029	.0043	.00527	.00089	.0028	.0041	.0047	-	.009
10,000	.000436	.0026	.00317	.00413	.00044	.0026	.00297	.0036	-	.0068

The next step is to look at the distribution of heat release in the various core materials following shut-down. An estimate of the distribution of heat release in the MTR is given in IDO-16443 and the following estimate is based primarily on the results reported therein.

It is assumed that the heat release due to neutron decay following shut-down, as shown in Fig. 3, is distributed as follows: fuel plates, $\frac{177.7}{197.5}$, moderator $\frac{9.25}{197.5}$, and reflector $\frac{9.7}{197.5}$ with the remainder escaping from the reflector. This is the distribution given for the MTR and should be adequate for our present purposes in view of the fact that at one second following shut-down the neutron decay contributes only about 23% of the total heat and at 10 seconds after shut-down only about 14% of the total heat.

The release of energy from the beta decay of the fission products will be confined to the fuel and moderator region due to the short range of the beta particles in this region. IDO-16443 does not state their basis for the estimate of 4 Mev/fiss in the fuel plate and approximately 1 Mev/fiss in the moderator; however, this number appears reasonable if one considers that the effective range of a 1 Mev beta particle in aluminum is only about 0.15 cm, the half-thickness of the fuel plate is 0.064 cm. and the fact that most beta particles must travel through more than the half thickness of the fuel plate. For our case we will consider that 80% of the beta energy is released in the fuel plate and 20% in the moderator.

The remaining heat source comes from the fission product gamma rays. Table III of IDO-16443 gives the distribution of heat as follows for primary gamma energy: fuel element, 5.7 Mev/fiss; moderator, 5 Mev/fiss; reflector, 4.9 Mev/fiss; and escape to graphite, 0.2 Mev/fiss. The total is 15.8 Mev/fiss and is made up of the following: fission products, 7.5 Mev/fiss; direct fiss. gammas, 5.5 Mev/fiss; neutron capture gammas, 2.8 Mev/fiss. For present purposes we can neglect all but the fission product gammas and we will treat them as distributed fractionally in the same location as given above - i.e., $\frac{5.7}{15.8}$ in the fuel plate, $\frac{5}{15.8}$ in the moderator and $\frac{4.9}{15.8}$ in the reflector; we will neglect the $\frac{0.2}{15.8}$ which escapes the reflector.

Using the information set forth above, we can now calculate the average heat release in the fuel plates, moderator and reflector. Fig. 5 shows the results obtained and Table II presents the data in tabular form. The results are given in terms of fraction of full power.

Table II

Heat Release Distribution Following Shut-down

Time after shut-down, seconds. Fraction of Full Power

	Fuel Plates				Moderator			Reflector
t	β	γ	fiss.	Total	β	γ	Total	γ
1	.03	.0117	.021	.0627	.0075	.0102	.0177	.01
2	.0275	.0111	.0172	.0558	.0069	.0097	.0166	.0096
4	.0244	.0104	.014	.0488	.0061	.0091	.0152	.009
10	.0202	.0093	.0086	.0381	.0051	.0082	.0133	.008
20	.0174	.0084	.0055	.0313	.0044	.0073	.0117	.0072
50	.0141	.007	.0024	.0235	.0035	.0062	.0097	.006
100	.0119	.0061	.00082	.0188	.003	.0053	.0083	.0052
200	.0099	.00516	.00029	.0154	.0025	.0045	.0070	.0044
500	.00781	.004	-	.0118	.002	.0035	.0055	.0034
700	.00712	.0036	-	.0107	.0018	.0032	.005	.0031
1,000	.0065	.0033	-	.0098	.0016	.0029	.0045	.0028
2,000	.00518	.0026	-	.0078	.0013	.0023	.0036	.0022
5,000	.00376	.0019	-	.0057	.00094	.0017	.00264	.0016
10,000	.0029	.0015	-	.0044	.00072	.0013	.002	.0013

For the present purpose it is desired to know the maximum rate of heat absorption in the reflector. The following equation taken from IDO-16443 was used to compute the source strength at the center of the face of the core for each of the energy groups:

$$I = \frac{S_v}{2}\left[\frac{1}{\mu} - \frac{e^{-x\mu}}{x\mu}\right]$$

where, I is the radiation intensity on the center of the face of the reactor core

S_v is the source strength per unit volume

μ is the total attenuation coefficient, cm^{-1}, for the core

x is the core thickness.

The aluminum-to-water ratio for the ORR is typically 0.65 and from this we obtain the average density of water and aluminum in the ORR core.

Avg. density of H_2O in ORR core = 0.597

Avg. density of Al in ORR core = 1.086

Using this information and data from the Nuclear Engineering Handbook by Etherington, Table III gives the value for μ in the core region for the energy groups of interest and gives the values obtained for I using these numbers for μ and taking x to be 31.2 (the values are quite insensitive to x).

Table III

Gamma Attenuation Coefficients for ORR Core Region

Mev	$(\mu/\rho)_{H_2O}$	$(\mu/\rho)_{Al}$	$0.597(\mu/\rho)_{H_2O}$ + $1.086(\mu/\rho)_{Al}$	I
0.3	0.118	0.103	0.1823	2.74 S_v
0.63	0.0878	0.07581	0.1347	3.7 S_v
1.1	0.0676	0.05876	0.1042	4.78 S_v
1.55	0.0567	0.0493	0.0873	5.69 S_v
1.99	0.0494	0.0433	0.0765	6.5 S_v
2.38	0.0456	0.0402	0.0709	7.0 S_v
3.3	0.0379	0.0340	0.0595	8.31 S_v

Table IV presents the Gamma Heat Release by energy groups following shut-down. These data are taken from calculations based on APEX-448 as described earlier in this memo.

Table IV

Gamma Heat Release by Energy Groups
Following Shut-down after 800 hr. Operation

Fraction of Normal Full Power

Time after shut-down, seconds	Group 1 0.3 Mev	Group 2 0.63 Mev	Group 3 1.1 Mev	Group 4 1.55 Mev	Group 5 1.99 Mev	Group 6 2.38 Mev	Groups 7-12 3.1–3.5 Mev Avg. = 3.3
1	.00182	.0078	.00567	.00503	.00339	.0027	.005887
2	.00175	.0075	.0054	.00487	.00319	.00262	.00552
4	.00166	.0071	.00503	.00469	.00298	.00246	.00502
10	.00151	.0065	.00456	.00426	.00265	.0022	.00416
20	.00138	.0059	.00418	.0039	.0024	.002	.00341
50	.00123	.00525	.00363	.00337	.002	.00165	.00243
100	.00111	.00474	.00318	.00296	.00172	.00139	.00176
200	.00099	.00424	.0027	.0026	.00146	.00115	.00119
500	.00084	.00362	.002	.0021	.00114	.00085	.000626
700	.00079	.0034	.00176	.00188	.00104	.00075	.00048
1,000	.000736	.00317	.00151	.0017	.000932	.00065	.00035
2,000	.000635	.00274	.00107	.00138	.00075	.000468	.000174
5,000	.00051	.00221	.00067	.00102	.000545	.000268	.0000536
10,000	.00043	.00183	.00049	.000779	.000414	.000161	.0000198

Table V gives the maximum gamma radiation intensity due to fission products at one second after shut-down at the center of the core face, using the information contained in Tables III and IV and assuming uniform distribution of fission products in the core.

Table V

Radiation intensity at Center of ORR
Core Face at 1 second after Shut-down

Mev	Fraction	I	I x Fraction
0.3	0.00182	2.74 S_v	4.99×10^{-3} S_v
0.63	0.0078	3.7 S_v	2.89×10^{-2} S_v
1.1	0.00567	4.78 S_v	2.71×10^{-2} S_v
1.55	0.00503	5.69 S_v	2.86×10^{-2} S_v
1.99	0.00339	6.5 S_v	2.2×10^{-2} S_v
2.38	0.0027	7.0 S_v	1.89×10^{-2} S_v
3.3	0.00589	8.31 S_v	4.89×10^{-2} S_v
			Total = 0.179 S_v

Let us now look at the heat release in the first row of Be (7.5 cm).

If the energy absorption in a unit volume is given by:

$$E_a = E_o \mu_a (1 + k\mu t) e^{-\mu t}$$

where E_a = energy absorption

E_o = initial energy

μ = total attenuation coefficient, cm^{-1}

μ_a = energy absorption coefficient, cm^{-1}

$k = \dfrac{\mu - \mu_a}{\mu_a}$

t = thickness of material preceding the volume element of interest

The amount of energy absorbed in the first row of Be per unit area is given by

$$\int_o^t E_o \mu_a \left[1 + \left(\frac{\mu - \mu_a}{\mu_a} \right) \mu t \right] e^{-\mu t} dt$$

upon performing the integration and evaluating for t from 0 to 6.5 cm, we obtain:

$$E_T = E_0 \frac{\mu a}{\mu} \left[1 - e^{-6.5\mu} + \frac{\mu - \mu a}{\mu a} \left(1 - e^{-6.5\mu}(1 + 6.5\mu) \right) \right]$$

Table VI gives values of μ and μ_a for Be and E_T for 7.5 cm of Be as a function of gamma energy. The last column is the product of E_T, I, fractional energy release, and 1/core volume for 1 second after shut-down; this number when multiplied by the normal full power yields the heat release in 7.5 cc of Be at the core center line in the row adjacent to the lattice. The units for power will be the same as those in which the power is expressed.

Table VI

Gamma Attenuation and Heat Release in Beryllium

Mev	$\mu(cm^{-1})$	$\mu_a(cm^{-1})$	E_T(7.5cm)	E_T(at one second) x I x Fraction x 10^{-5}
0.3	0.1701	0.0461	0.402 E_0	2.01×10^{-8}
0.63	0.126	0.0472	0.335 E_0	9.68×10^{-8}
1.1	0.0979	0.0437	0.284 E_0	7.7×10^{-8}
1.55	0.0814	0.0401	0.235 E_0	6.7×10^{-8}
1.99	0.0711	0.0378	0.233 E_0	5.13×10^{-8}
2.38	0.0653	0.036	0.22 E_0	4.16×10^{-8}
3.3	0.0538	0.0319	0.196 E_0	9.58×10^{-8}
				Total = 4.5×10^{-7}

In view of the fact that the fission products are distributed in the same manner as the power distribution in the operating reactor we will assume that an additional factor of 2 should be applied to account for this distribution. For the decay it will be assumed that the heat release decays following the same curves as given in Fig. 5. Fig. 6 gives the heat transfer requirements for the fuel plates and the reflector elements following shut-down. The ordinate must be multiplied by the normal full power level in Mw to obtain the heat transfer rate required. The heat transfer rate in the fuel plates is based on a maximum of 600,000 BTU/ft^2-hr at a power level of 30 Mw.

Let us look at the operation of the BSF reactor in order to obtain experimental information on heat transfer under conditions of low flow rates. ORNL-2081, Section I-4, is used as a reference. The BSF reactor described has 24 full elements, 3 control rod elements and 1 partial element. The reactor operates at power levels up to 1 Mw using free convection to remove the heat from the fuel plates.

Ht. transfer area = 384 ft^2
Max/Avg in core = 1.8
Avg. Ht. transfer rate in core = 8,898 BTU/ft^2-hr
Max. Ht. transfer rate = 16,016 BTU/ft^2-hr
Max/Avg in element 25 = 1.44
Avg. Ht. transfer in element 25 = 11,122 BTU/ft^2-hr
Ht. removed from element 25 = 48 kw
Inlet H_2O Temp = 95°F
Assumed ΔT = 63°F
Max. Plate Temp = 185°F
Flow area per element = 5.53 in^2
Flow rate per element = 43.35 #/min
Calc. coolant velocity = 0.3 ft/sec

The above information shows that low velocities are sufficient to remove heat from the fuel elements and reflector pieces and that values for h of about 400 are attained in the BSF at flow rates of about 0.3 ft per second.

Based on the above information it now appears reasonable for power levels at least as high as 30 Mw to provide emergency coolant flow for the ORR sufficient to meet the requirements of maintaining a reasonable ΔT in the core region, 30°F, as set forth on page 3, i.e., a flow equal to 228 P times the fraction of full power given by Fig. 4. Table VII gives flow rates based on a 30°F ΔT for several times of interest. It is assumed that adequate flow is provided by the main coolant coast-down for the first 10-15 seconds.

Table VII

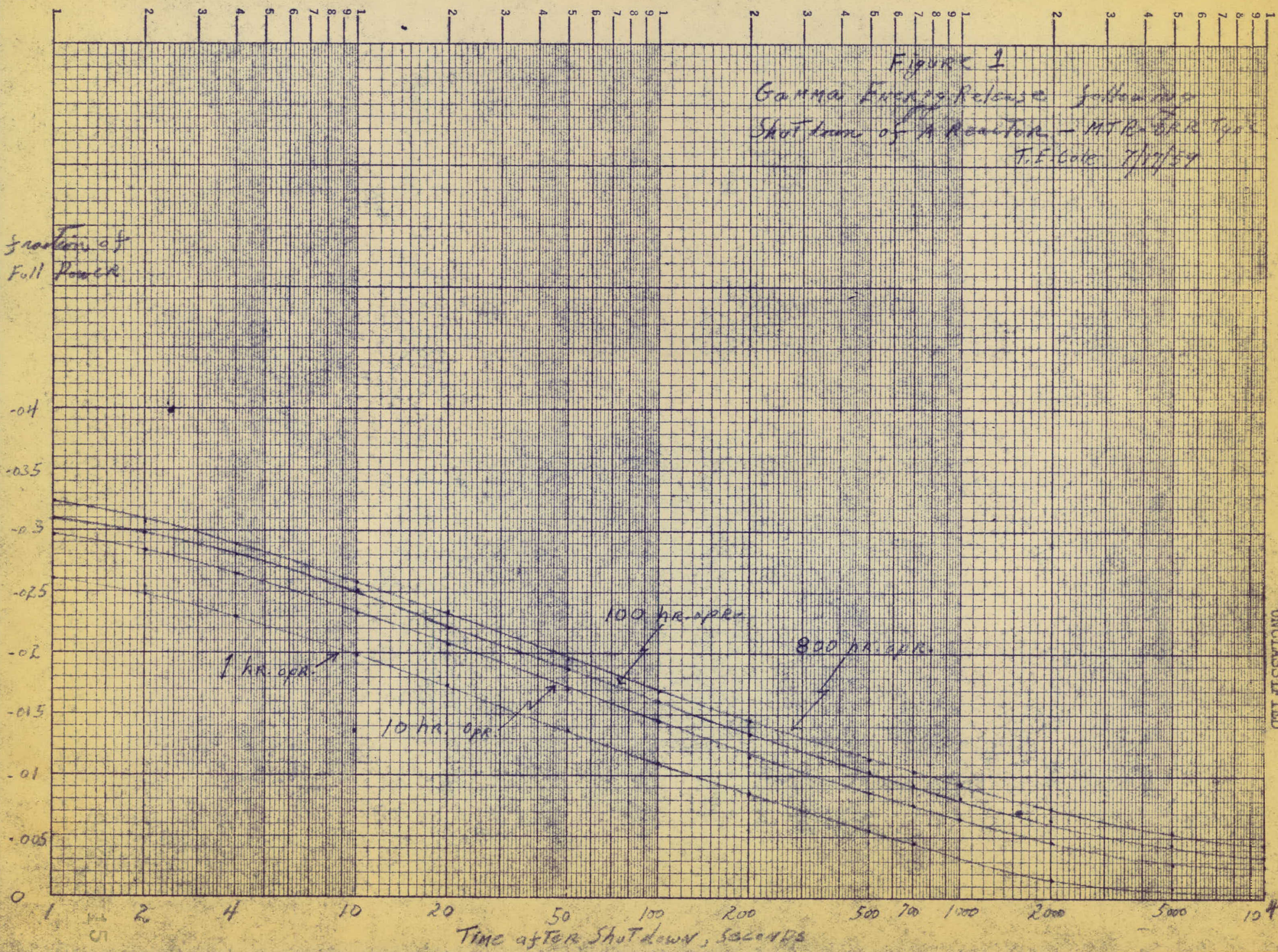
Flow Requirements Following Reactor Shut-down to hold a 30°F Temperature Rise

Time after Shut-down Seconds	Flow gpm
10	13.7 P
20	11.6 P
50	9.1 P
100	7.4 P
500	4.55 P
1,000	3.65 P
5,000	2.05 P

Two additional points should be noted about the shut-down cooling of the ORR. First is the fact that the heat transfer information from the BSF is based on upward coolant flow through the fuel elements. In planning for emergency shut-down coolant flow for the ORR this fact must be considered because the present system for ORR is based on downward flow and as the flow rate shown in Table VII approaches that which would be attained under natural convection it is clear that natural convection will compete to some degree with the forced downward flow. This is not judged to be an important point due to the fact that the flow set forth in Table VII is such as to produce only about a 30°F ΔT and therefore the bouyant force will be kept very small. The statements about emergency shut-down cooling contained in my letter to J. A. Cox dated June 8, 1959, are based on the upward flow which would result from the type of cooling system suggested. The second point has to do with the Facility Cooling Pumps in the ORR. These pumps take suction from the reactor inlet line and the flow, after passing through the ion exchangers, large facilities and other connected devices (experiment cooling?), returns to the reactor exit line. If it is assumed that all the pumps in the pump house fail to operate, it is believed that the Facility Coolant Pumps will circulate water through the reactor tank in a direction opposite to normal flow - i.e., from bottom to top. Dependent on the power level prior to shut-down and the flow through the Facility Coolant Pumps, this may or may not be adequate to cool the reactor. In the event that the shut-down pump is running, the flow through the Facility Coolant System bypasses the reactor and must be subtracted from the flow provided by the shut-down pump in order to get the actual flow through the reactor.

In my opinion the primary need for adequate shut-down coolant flow is to limit the temperature to which the reactor tank itself is subjected. Surface boiling on the fuel plates or beryllium is not considered to be harmful in itself. It should be pointed out that considerable attention was given to this matter of boiling in the MTR design and you are referred to ORNL CF-49-12-84, "MTR Shut-down Cooling Requirements" by J. A. Lane. On page 4 of this memo the following statement is made in regard to the beryllium reflector: "The actual flow requirement must be such that the cooling water in the beryllium does not reach the boiling point. It is assumed that local or 'film' boiling will not be objectionable.". The beryllium cooling requirements for the MTR are based on a different geometry of the beryllium pieces - i.e., the pieces which make up the "permanent" reflector and therefore the conclusions reached regarding cooling are somewhat different.

As stated at the beginning of this memo, the methods used to derive the results contained herein are rather crude. In view of the continued and widespread interest in the shut-down cooling problem of reactors of this type I would like to recommend that steps be taken to perform more careful and complete calculations using available information and techniques and, if necessary, experiments to confirm the calculations in order to have available good information on this subject for use by interested parties.



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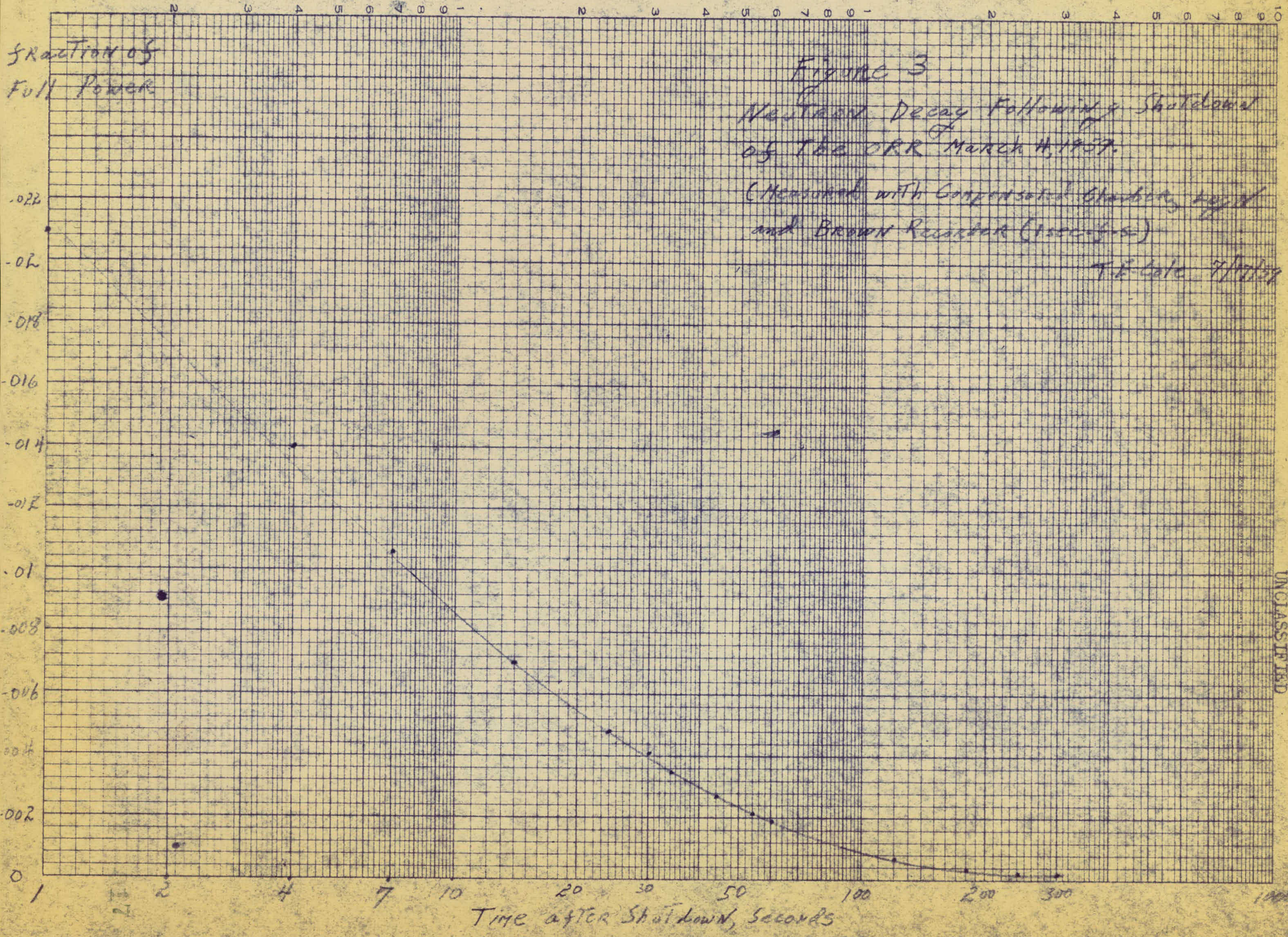

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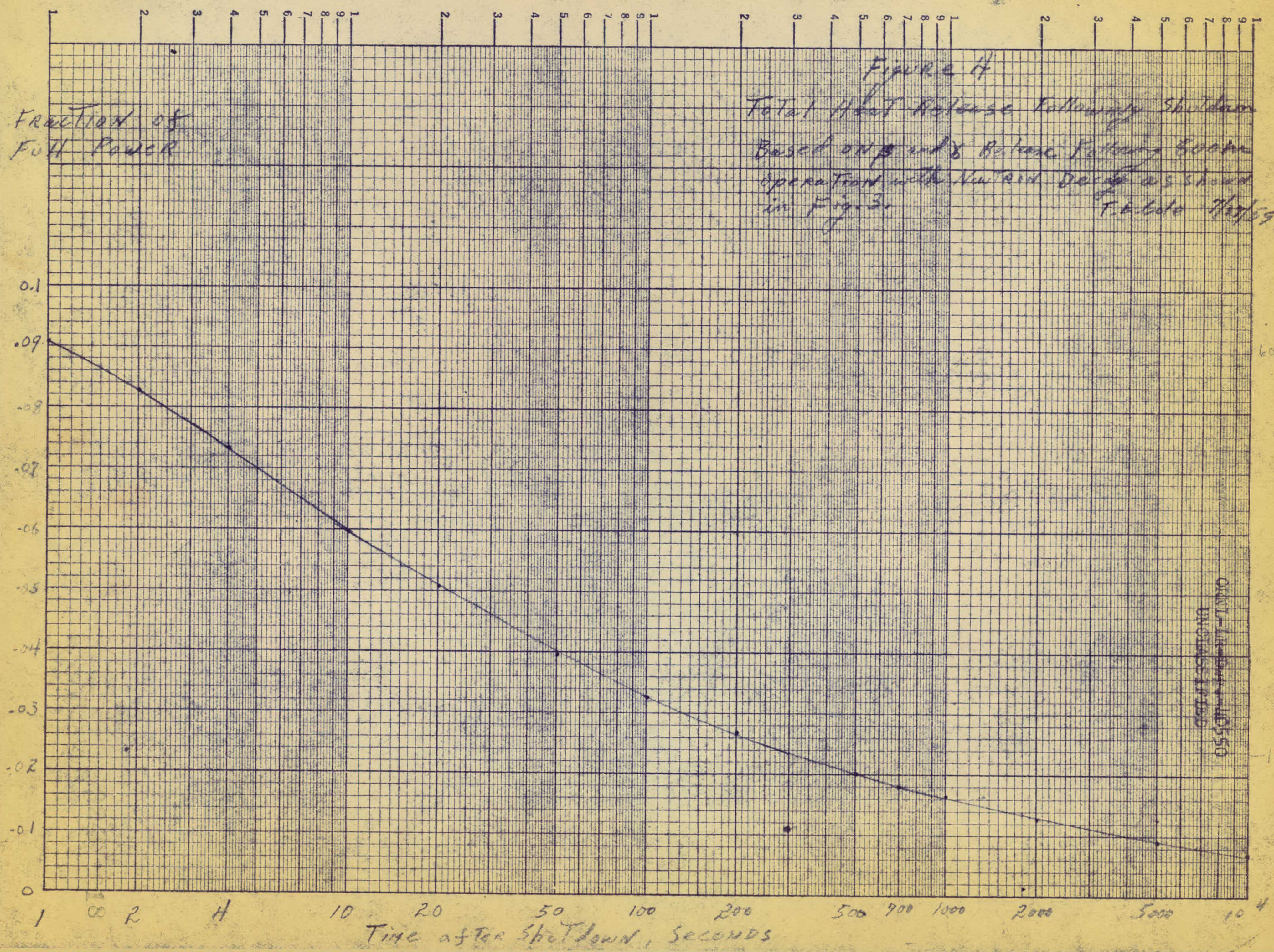
Figure 4
Total Heat Release Following Shutdown
Based on β and γ Release Following 800 hr operation with New Rod Decay as shown in Fig. 3.
T. E. Cole
Fraction of Full Power
0.1
.09
.08
.07
.06
.05
.04
.03
.02
.01
0
1
2
4
10
20
50
100
200
500
700
1000
2000
5000
10^4
Time after Shutdown, Seconds
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Semi-Logarithmic, 4 Cycles X 10 to the inch, 5th lines accented.
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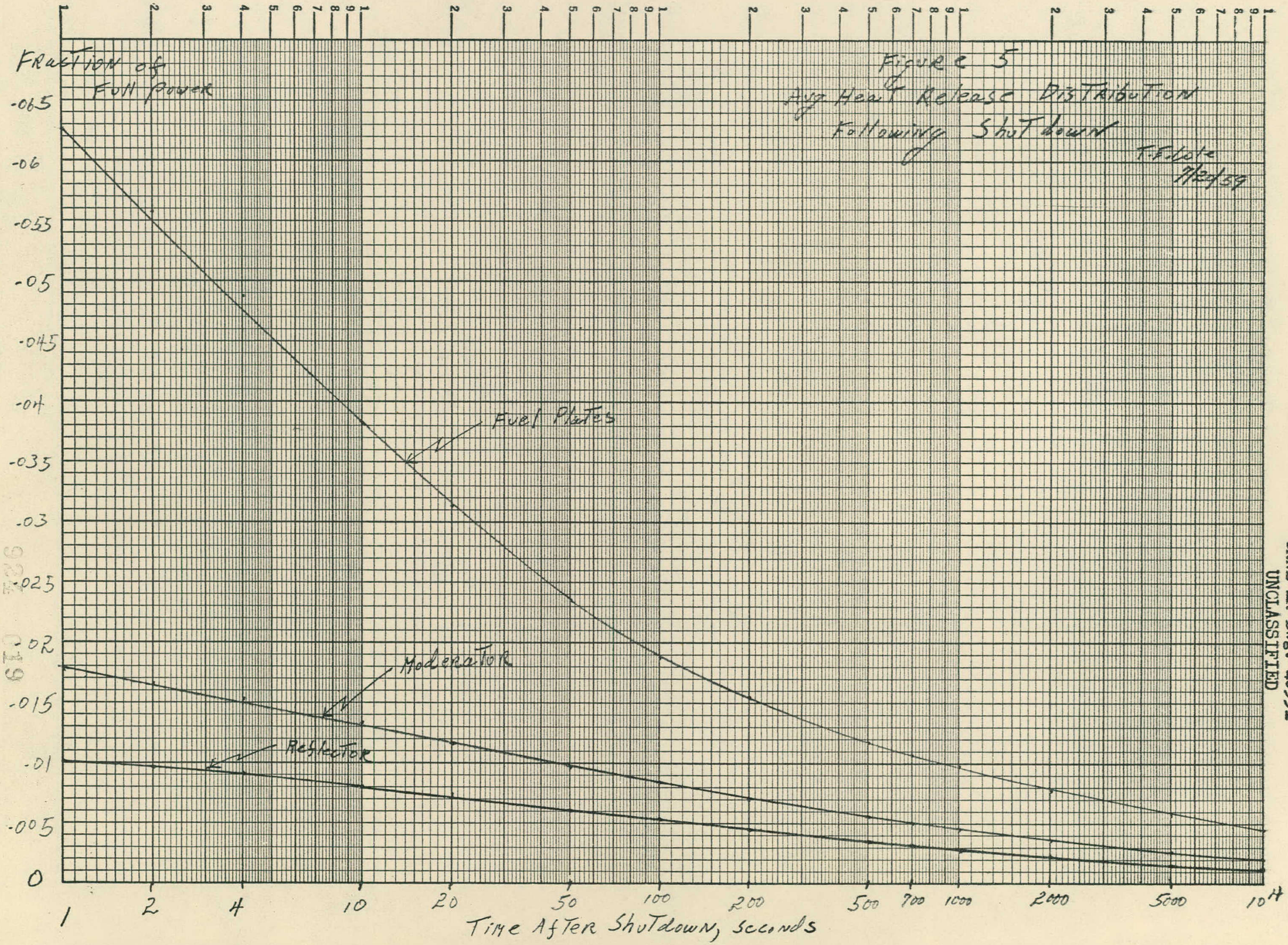

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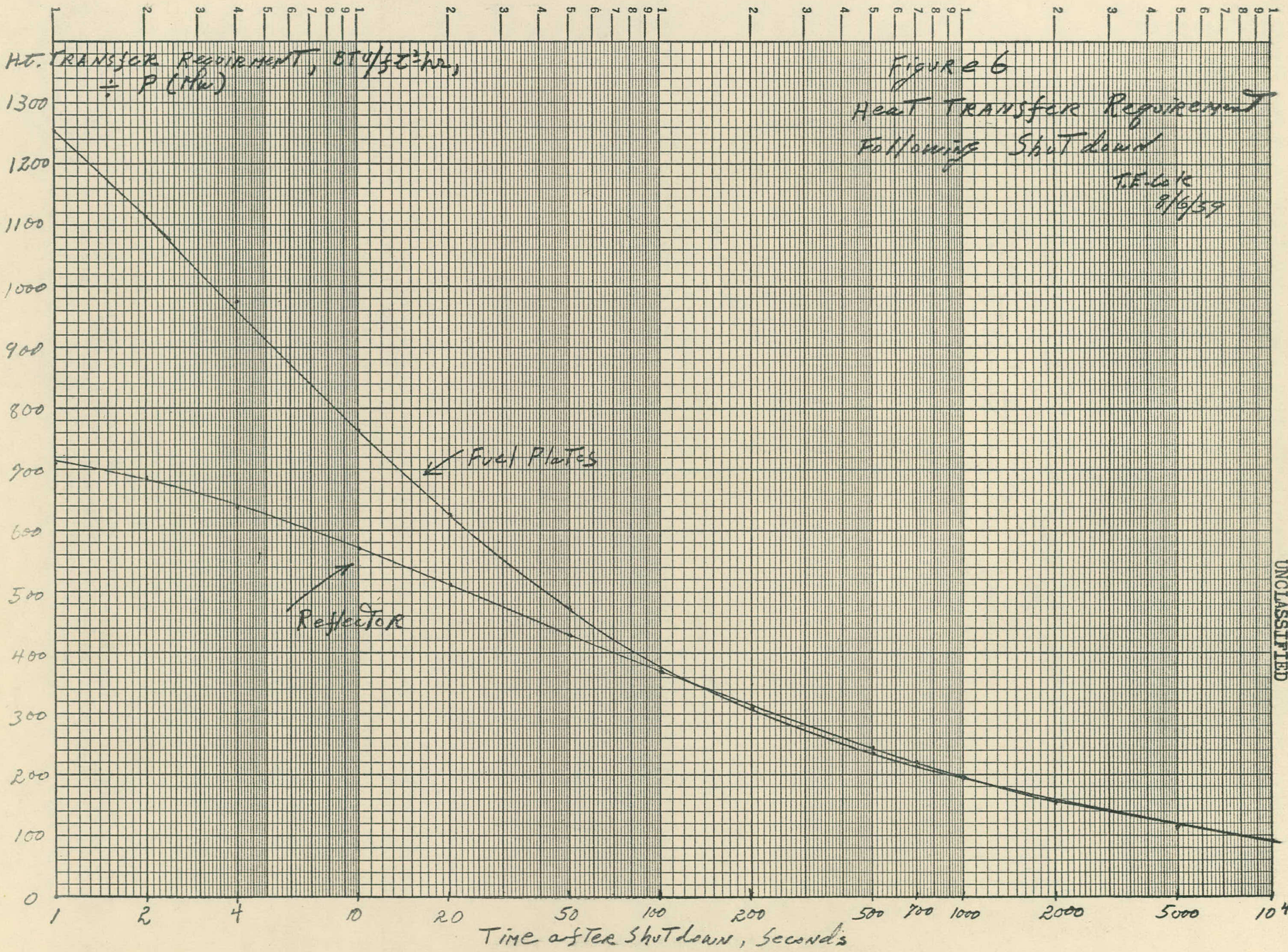

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