YAEC - 1002

AN ANALYSIS OF THE ROD EJECTION ACCIDENT IN THE YANKEE REACTOR

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Acknowledgement

The author wishes to acknowledge the work of the Westinghouse Atomic Power Division which forms the basis for this report, and in particular the efforts of J. P. Cunningham who developed the analog model used to simulate the reactor transients.

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Introduction

It is possible to postulate failure of a control rod travel housing such that a control rod could be rapidly ejected from the core. Such a failure is considered to be virtually incredible, for the following reasons:

1. Each housing is proof tested to 5000 psi prior to operation.
2. Stress levels in the housings are not affected by system transients at power or by thermal movement of the coolant loops.
3. The housings are exposed to very little radiation, so embrittlement cannot occur.

Nevertheless, the control rod program in the Yankee reactor has been revised in order to minimize the reactivity that could be inserted by rod ejection. Henceforth, Yankee will be operated virtually as a chemical shim plant. In Core IV, the rods will be used only to compensate for xenon and Doppler and to provide a small control margin (approx. 0.5%). Rod groups 5 and 6 (the high worth rods) will be withdrawn prior to going critical.

Boron will be adjusted to reach criticality in the hot xenon free condition with groups 1234 in the core. Groups 3 and 4 will be withdrawn simultaneously to approximately 33" in going to power (540 MWt). As xenon builds in, 3 and 4 will be fully withdrawn and groups 1 and 2 will be withdrawn approximately half way. Power will not be raised to 600 MWt in Core IV until groups 1 and 2 are controlling. The boron concentration in the main coolant will then be gradually reduced over the life of the core to compensate for burnup.

With this mode of operation the maximum excess reactivity insertion which could occur due to an ejected rod is 0.88%. This corresponds to the worth of a group 3 or 4 rod ejected from the fully-in position at zero power. At 540 MWt a partially withdrawn rod from groups 3 and 4 would be worth 0.46%. The full worth of a rod in groups 1 or 2 is 0.36%. Rapid insertion of this much reactivity should never present a significant hazard. However, the rod ejection accident has been analyzed in detail to show conclusively that the consequences of such an accident would be much less severe than those of the "hypothetical accident", on which the safety analysis of the plant is based.

Method of Analysis

The accident was analyzed by analog simulation of the transient following rod ejection using the conventional, one group, neutron kinetics equations. The effect of the reactivity insertion on reactor power level, fuel temperature, and system pressure was studied. In the analysis, the UO_2 pellets were subdivided into three concentric regions. This subdivision permits an accurate simulation of the non-uniform power density and temperature distribution in the pellets.

The results of the analog study have been compared with a digital analysis based on the WIT-4 code. The basic difference between the digital and analog work is that the pellet is treated as a single region in WIT-4 and power density is assumed to be uniform throughout the UO_2. Actually, the results of the digital and analog studies are in good agreement on the power transient following rod ejection, but the analog model predicts a more rapid rate of heat transfer to the primary coolant. This is as expected with the pellets subdivided into several concentric regions.

One very conservative assumption was made in both studies. The moderator temperature coefficient was assumed to be zero, thereby eliminating any negative reactivity feedback from heating of the primary coolant in the transient. This will not be the case in the Yankee reactor, but the assumption was made to insure that the analysis would be independent of moderator coefficient and, therefore, independent of the boron concentration present in the primary coolant. Actually, the moderator coefficient in Core IV should not be less negative than $-1 \times 10^{-4}/^{o}F$.

The analog work was performed as a parametric study, with the primary purpose to determine the maximum temperature reached in the UO_2 during the transient as a function of initial power level, reactivity insertion, the steady state heat flux hot channel factor, and the heat flux hot channel factor following rod ejection.

The methods, assumptions, and results described herein apply to the analog analysis except as specifically noted. The WIT-4 work is not discussed in detail. It is considered that the analog simulates the behavior of the core in the transient more accurately, although the heat transfer model associated with the analog study, when coupled with a zero moderator coefficient, exaggerates the power and fuel temperature increase following rod ejection.

Basic Assumptions

The important parameters used in the analysis of the rod ejection accident are as follows:

1. Delayed neutron fraction - 0.006
2. Neutron lifetime - 12×10^{-6} sec.
3. Moderator temperature coefficient - zero

4. Thermal conductivity - Reference 1, pa. 102:16

5. Gap coefficients - pellet to clad

   Normal heat flux (600 MWt) - 240 BTU/hr.-ft.^{2}-^oF

   Twice normal heat flux - 420 BTU/hr.-ft.^{2}-^oF

   Pellets in contact with clad - 1000 BTU/hr.-ft.^{2}-^oF

6. Film coefficients

   Subcooled - 6400 BTU/hr.-ft.^{2}-^oF

   Nucleate boiling - Jens-Lottes Correlation

   Film boiling - 200 BTU/hr.-ft.^{2}-^oF

7. Shutdown margin - 0.5% (2 rods out)

8. Rod ejection time - 0.14 sec.

9. Doppler coefficient - $\frac{\partial \rho}{\partial T} = 2.889 \times 10^{-5} - \frac{3.34 \times 10^{-3}}{\sqrt{^oR + 3850}}$

The delayed neutron fraction (β) used in the analysis is based on the results of the Yankee Core Evaluation Program. The value of $\beta = .006$ corresponds to beginning of core life. This is the most appropriate value for use with a low moderator coefficient. While β will drop to approximately 0.005 over the life of the core, the end of life (boron-free) moderator coefficient will be $-2.4 \times 10^{-4}/^oF$. Calculations show that this change in moderator coefficient will outweigh the reduction in β and that the beginning of life case (with a zero moderator coefficient) is most critical from a rod ejection standpoint.

The prompt neutron lifetime is also based on the Yankee Core Evaluation Program and on analytical work done under the Large Reactor Development Program. This quantity is essentially constant throughout core life, because the removal of boron from the main coolant will offset the effect of fuel depletion.

Use of a zero moderator coefficient is, of course, overly conservative. As previously outlined, this value was used to insure that the results would be independent of boron concentration in the primary coolant. In Core IV, the moderator coefficient will not be less negative than $-1.0 \times 10^{-4}/^{o}F$ at any time in core life.

The heat transfer coefficients used for the gap between the pellets and cladding are calculated based on gap thickness. For pellets in contact with the cladding, the usual contact resistance of 1000 BTU/hr.-ft.^{2}-^{o}F is assumed.

The film coefficients are based on standard correlations. The subcooled coefficient is calculated from the Dittus-Boelter equation. A variable coefficient based on the Jens-Lottes correlation is used for nucleate boiling. The value of 200 BTU/hr.-ft.^{2}-^{o}F used for stable film boiling after DNB is calculated from the correlation reported in Reference 2.

The shutdown margin of 0.5% assumed in this study represents the minimum that could exist with any two control rods out of the Yankee Core. The license requires at least 2% shutdown with the highest worth rod stuck. A second rod will not be worth more than 1.5%, regardless of location. The assumption of a stuck rod plus an ejected rod is, of course, extremely conservative, but it is not particularly important for Yankee because adequate shutdown margin will always be available. In fact, about 3% shutdown should be available in Core IV with two rods out.

The rod ejection time of 0.14 seconds was calculated from a simple force balance on the control rod, follower and drive shaft (410 lbs., 2062 psig, 1.5 in.^{2}) assuming 30 inches of rod insertion at the time of the accident and neglecting friction. For the worst case in Core IV (.46% Δ K @ 540 MWt), the rod would be inserted 57" prior to the accident, and the ejection time would be 0.2 seconds. Use of the lower value of 0.14 seconds throughout the analysis exaggerates the consequences of the accident for all cases studied.

The relation used for the variation of Doppler coefficient with fuel temperature is the result of a resonance temperature analysis performed under the Large Reactor Development Program. This analysis is based on measurements of power coefficient at the Yankee Plant and is, therefore, directly applicable to the study reported herein. A statistical weighting factor of 1.68 was applied to the Doppler coefficient to account for the effects of non-uniform power distribution.

Results

The results of the parametric rod ejection study are shown on Figures 1 and 2. Figure 1 is a plot of fuel centerline temperature prior to rod ejection (T_c^o) as a function of the initial hot channel factor (F_q^o) and the steady state power level (q_o). Figure 2 defines the average increase in centerline temperature (ΔT) as a function of reactivity insertion (ΔK), the hot channel factor following rod ejection (F_q^t), and initial power (q_o). Given the hot channel factors, these two graphs can be used to predict the maximum fuel temperature at any point in the core for any reactivity insertion,

$$\text{ie:} \quad Tmax = T_c^o + F_q^t \Delta T.$$

Detailed calculations have been performed on Core IV to define rod worths and hot channel factors before and after rod ejection. The rod worths are given as a function of rod position on Figure 3. Hot channel factors before ejection are shown on Figure 4 and after ejection on Figure 5. As previously explained, the maximum excess reactivity insertion that could take place is 0.88% ΔK (the full worth of a rod from groups 3 or 4), but this can only occur at zero power. As power is increased, groups 3 and 4 will be withdrawn until, at 540 MWt, the maximum individual rod worth is 0.46% ΔK. Note that the steady-state HCF (Figure 4) is very nearly at a maximum at this point and the transient HCF (Figure 5) is also high. This represents the worst case in Core IV from the standpoint of UO_2 temperature. Other positions in the program have been analyzed and all have been found to result in lower fuel temperatures. Data for the three limiting cases is given below:

	q_o (MWt)	ΔK(%)	F_q^o	F_q^t	T max. (°F)
rod groups 1234 @ 0"	0	0.88	2.93	5.12	2124
rod groups 12 @ 0", 34 @ 33"	540	0.46	3.89	4.75	4390
rod groups 12 @ 0", 34 @ 90"	600	0.36	2.70	3.35	3628

It is important to realize that these limiting cases will exist only briefly, if at all, in going to power. Xenon buildup will very rapidly require that the rods be withdrawn past the peak hot channel factors and maximum rod worths to positions very much more favorable from the standpoint of rod ejection.

A plot of maximum fuel temperature v.s. time after the accident for the case of 0.46% ΔK @ 540 MWt is shown on Figure 6. Note that even this "worst case" is not severe enough to result in center melting (5000°F) in the hottest fuel pellet.

Figure 7 illustrates the variation in nuclear power as a function of time following the accident for the case of .46% ΔK at 540 MWt. Figure 8 shows the variation in heat transferred to the primary coolant as a function of time for the same case. Figures 9 and 10 are similiar plots of heat flux as a function of time for cases of .36% ΔK and 1% ΔK at 600 MWt. While this latter case cannot occur in Core IV, it was used as a conservative basis for the analysis of the pressure transient in the main coolant system and vapor container.

The effect of the 1% insertion on main coolant system pressure is shown on Figure 11. Several other conservative assumptions were used in generating this curve:

1. No leakage through the broken rod travel housing

2. No pressurizer spray

3. No relief valve operation

Even with these assumptions, the safety valves are more than sufficient to accommodate the pressure surge. The first valve, which is set at 2,500 psia, would open in 9 seconds and its capacity would be sufficient to accommodate the entire surge. The second safety valve, which is set at 2,575 psia, would not even open.

The energy release to the vapor container as a result of rod ejection has been calculated. In this analysis, the assumption was made that the weld connecting the reactor vessel head and the adapter which supports the rod drive mechanism fails completely, and that the discharge is through a 3-7/8" diameter open break. This represents the most rapid blowdown that could be associated with rod ejection. Again, the 1% reactivity insertion from 600 MWt was studied. The mass and energy release to the container through the break is shown as a function of time following the accident on Figure 12. The pressure transient associated with this release is shown on Figure 13. The peak pressure reached is 17 psig. This compares with 32 psig reached following the 20" pipe break. The design pressure of the container is 34.5 psig. It is clear that the rod ejection accident represents a less severe case from the standpoint of containment pressure than the hypothetical accident.

Fission product release following rod ejection has also been evaluated. The methods used are similar to those described in the Connecticut Yankee PHSR, Amendment No. 6. The release of iodine and other fission products which would take place as a result of rod ejection (0.46% ΔK @ 540 MWt) would be less than 0.1% of that which would occur in the hypothetical accident (per TID-14844). Even with a 1% rod ejection, the release would be less than 7% of the TID-14844 value. It is clear that the radiation hazard associated with rod ejection would not even approach that postulated in the hypothetical accident.

Discussion

Concern has been expressed from time to time over the effect of a rapid reactivity insertion on fuel integrity. It has been theorized that a fast transient could result in bursting of the fuel rods due to melting and partial vaporization of the UO_2. This in turn could cause dispersal of the oxide into the coolant. A rapid transfer of heat from the UO_2 might then create a pressure wave which could, as a limit, rupture the primary system.

Experiments have been run at the TREAT facility which approximate these conditions. Capsules of UO_2 and mixed uranium and plutonium oxide fuel were subjected to neutron bursts of sufficient energy to cause fuel melting and clad damage, (Reference 3). The primary purpose of these experiments was to study metal-water reactions, but data on the effects of the bursts on the oxide was also obtained. These experiments show that there is no problem below the melting point of the fuel. They also show that where the heat added is sufficient to totally melt the fuel and generate a vapor pressure the rods will burst and dispersion of the oxide will occur. Additional experiments are required to determine exactly how much of the fuel must be molten or if, in fact, a vapor pressure must be produced before bursting will occur.

The results of the rod ejection analysis outlined in previous sections show that even the center of the hottest pellet will not reach the melting point for the worst case that could occur in Yankee Core IV, much less generate a vapor pressure. And these results are based on extremely conservative assumptions, such as zero moderator coefficient.

However, even if it were possible to add enough energy to burst the hot rods and rapidly disperse the UO_2 in those rods, this would not result in a pressure wave sufficient to rupture the primary system. A detailed analysis has been performed for rod bundles of various sizes in which the UO_2 in these bundles is instantaneously dispersed into water at 2000 psi. The pressure wave has been calculated as a function of UO_2 particle size and distance from the bundle. A summary of this work is included as Appendix A. For a single rod and 0.01" diameter particles, the maximum pressure surge two feet from the rod is only 30 psi. (The average particle size measured in the TREAT tests was 0.011".) For a 25 rod bundle (entirely dispersed) and the same particle size, the maximum pressure surge at two feet is 700 psi. Even if these values are doubled to account for reflection, the resulting pressure is not significant from the standpoint of primary system rupture. The system will withstand a pressure surge nine times as great as that calculated for the 25 rod bundle without even yielding. A much larger surge would, of course, be required for failure. The calculations for a bundle of 25 rods are appropriate for the rod ejection accident because only a few fuel rods in the vicinity of the ejected control rod will see temperatures anywhere near the maximum values cited herein.

For comparison purposes, the methods developed in the analysis of rod ejection for Yankee have been used to calculate power level and energy production to the transient peak in the recent SPERT tests with UO_2 fuel. The calculated results are in good agreement ($\pm$ 10%) with the tests for reactivity insertions above 1% ΔK. For smaller reactivity insertions, the analytical model predicts power and energy production significantly higher than experienced in SPERT. Use of a zero moderator coefficient in the calculations is probably the reason for this disparity, because the differences are significant only for high-period transients where heat transfer to the moderator would be important.

Conclusions

The following conclusions can be drawn from the foregoing analysis:

1. Ejection of a control rod from Yankee Core IV will not result in fuel melting or dispersion of UO_2 into the coolant.
2. Rod ejection will not result in pressurization of the primary system beyond the design value of 2500 psia.
3. The maximum pressure reached in the vapor container as a result of rod ejection will be less than 17 psig, compared to a design pressure of 34.5 psig.
4. The fission product release associated with rod ejection will be negligible compared with that postulated for the hypothetical accident.
5. Even if dispersal of a substantial amount of UO_2 into the main coolant were to occur due to rod ejection, the primary system would not be damaged by the resulting pressure surge.

References

(1) "Final Hazards Summary Report", Yankee Nuclear Power Station, Volume I

(2) WCAP-2040, "A Review of Heat Transfer and Fluid Flow of Water during Once-through Operation and in the Super Critical Pressure Region"; A. A. Bishop, L. E. Efferdine and L. S. Tang; August 1962

(3) Limatainen, R. C., Ivins, R. O., Deerwester, M. F., and Testa, F. J. "Studies of Metal-Water Reactions at High Temperatures II. TREAT Experiments Status Report on Results with Aluminum, Stainless Steel-304, Uranium, and Zircaloy - 2," Argonne National Laboratory, ANL-6250 January 1962

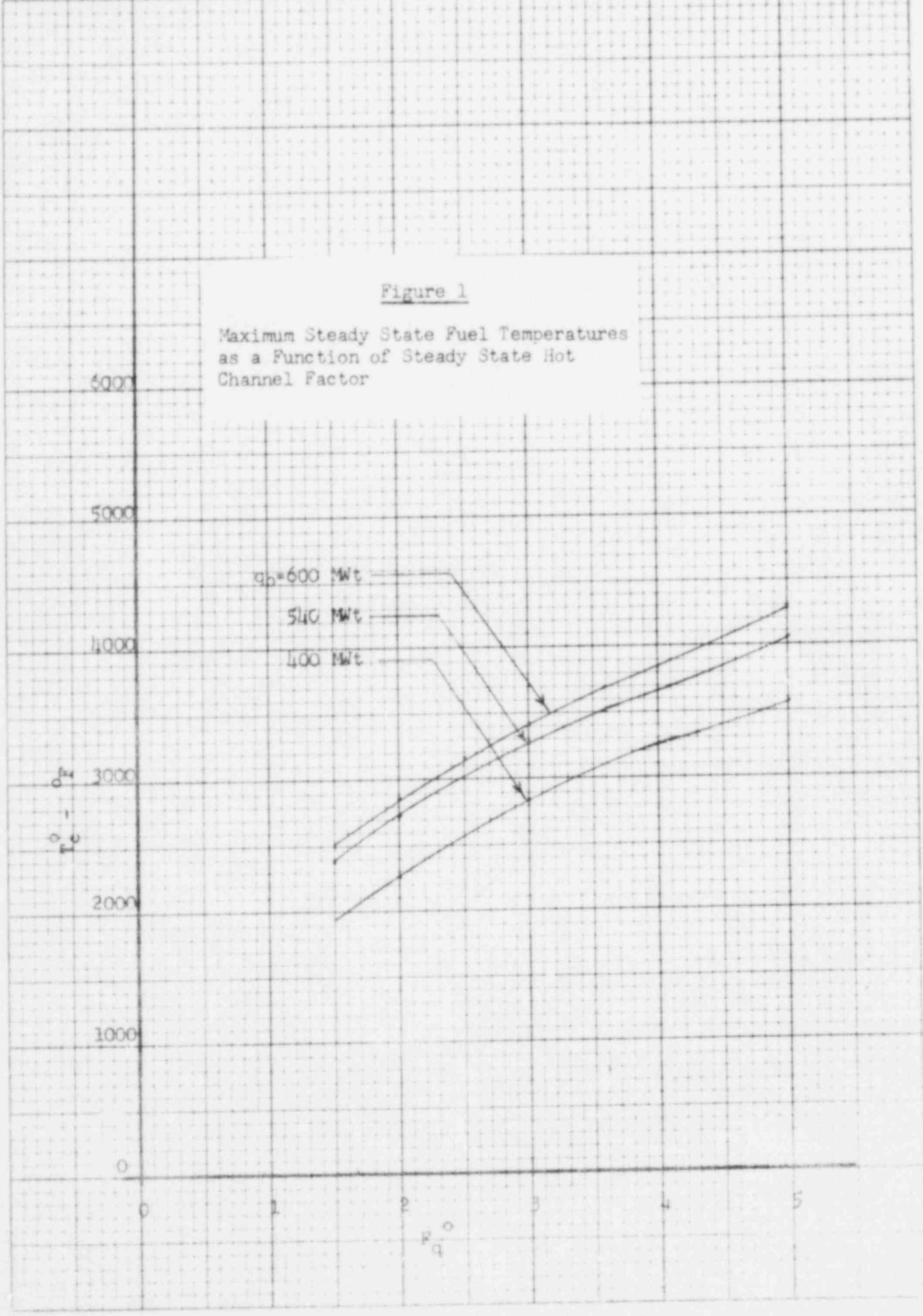


Figure 1

Maximum Steady State Fuel Temperatures as a Function of Steady State Hot Channel Factor

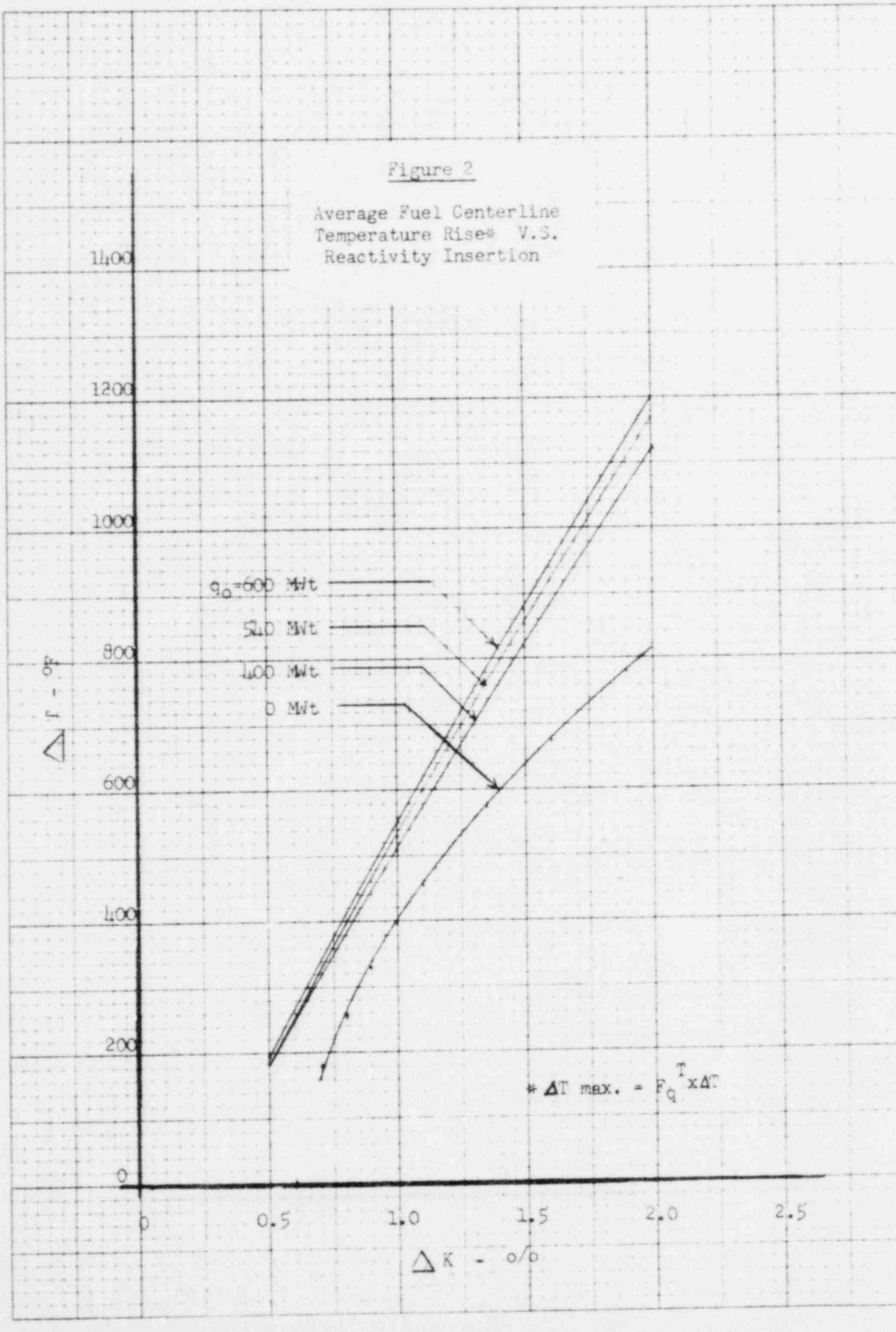
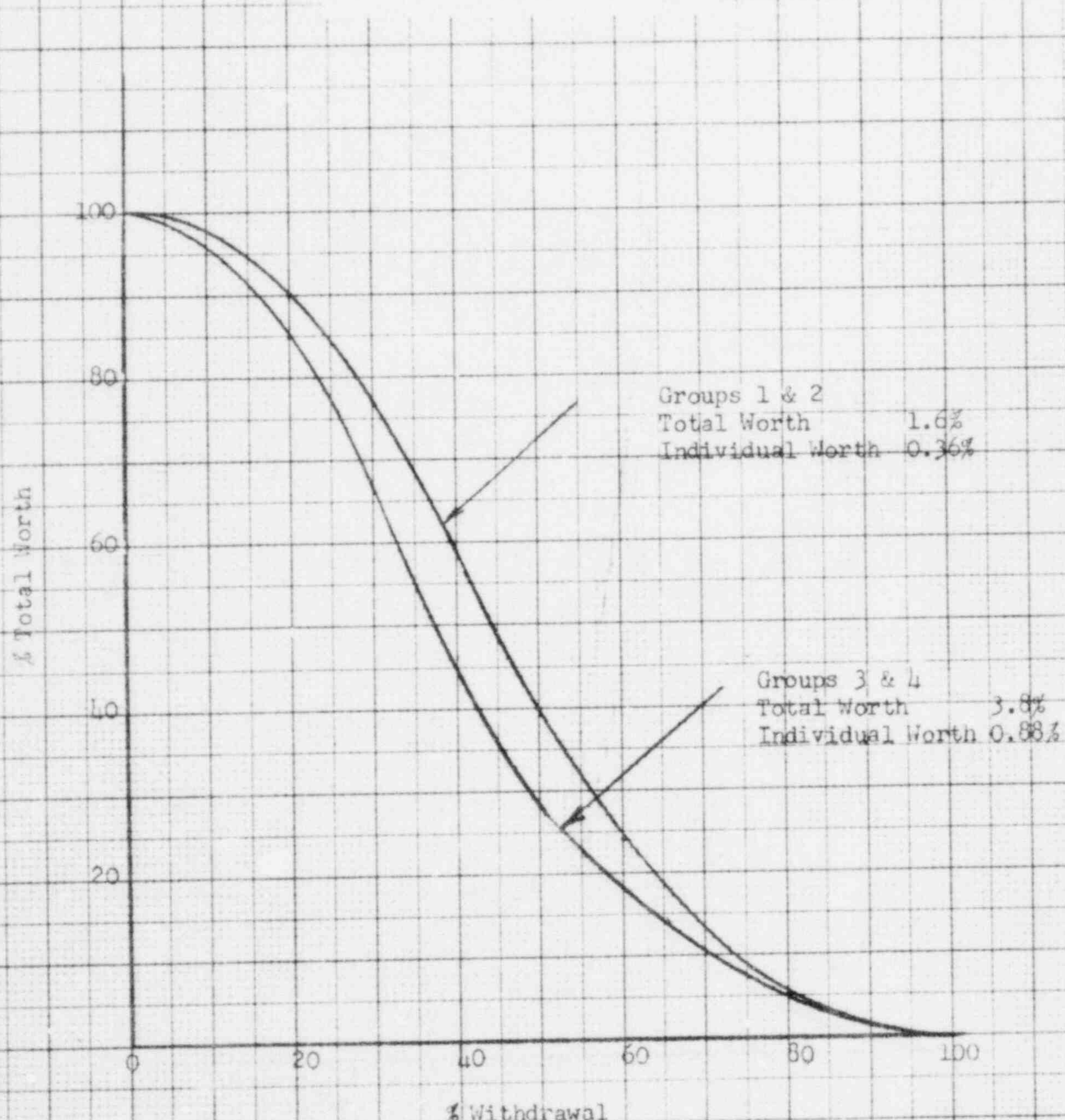


Figure 2

Average Fuel Centerline Temperature Rise* V.S. Reactivity Insertion

Figure 3

Core IV Rod Group Worth
V.S. Rod Position



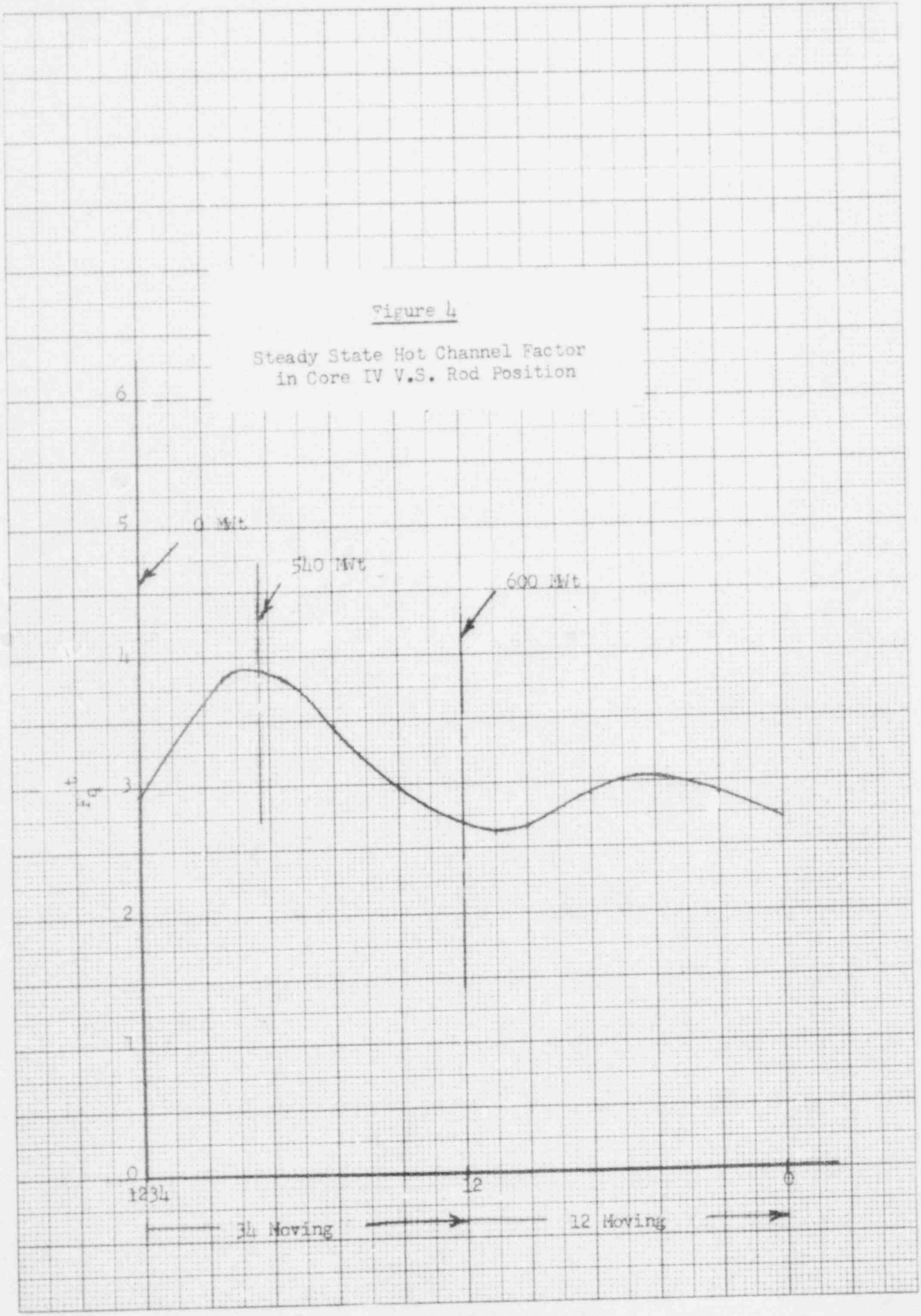


Figure 4
Steady State Hot Channel Factor
in Core IV V.S. Rod Position
6
5
4
3
2
1
0
0 MWt
540 MWt
600 MWt
1234
12
0
34 Moving
12 Moving

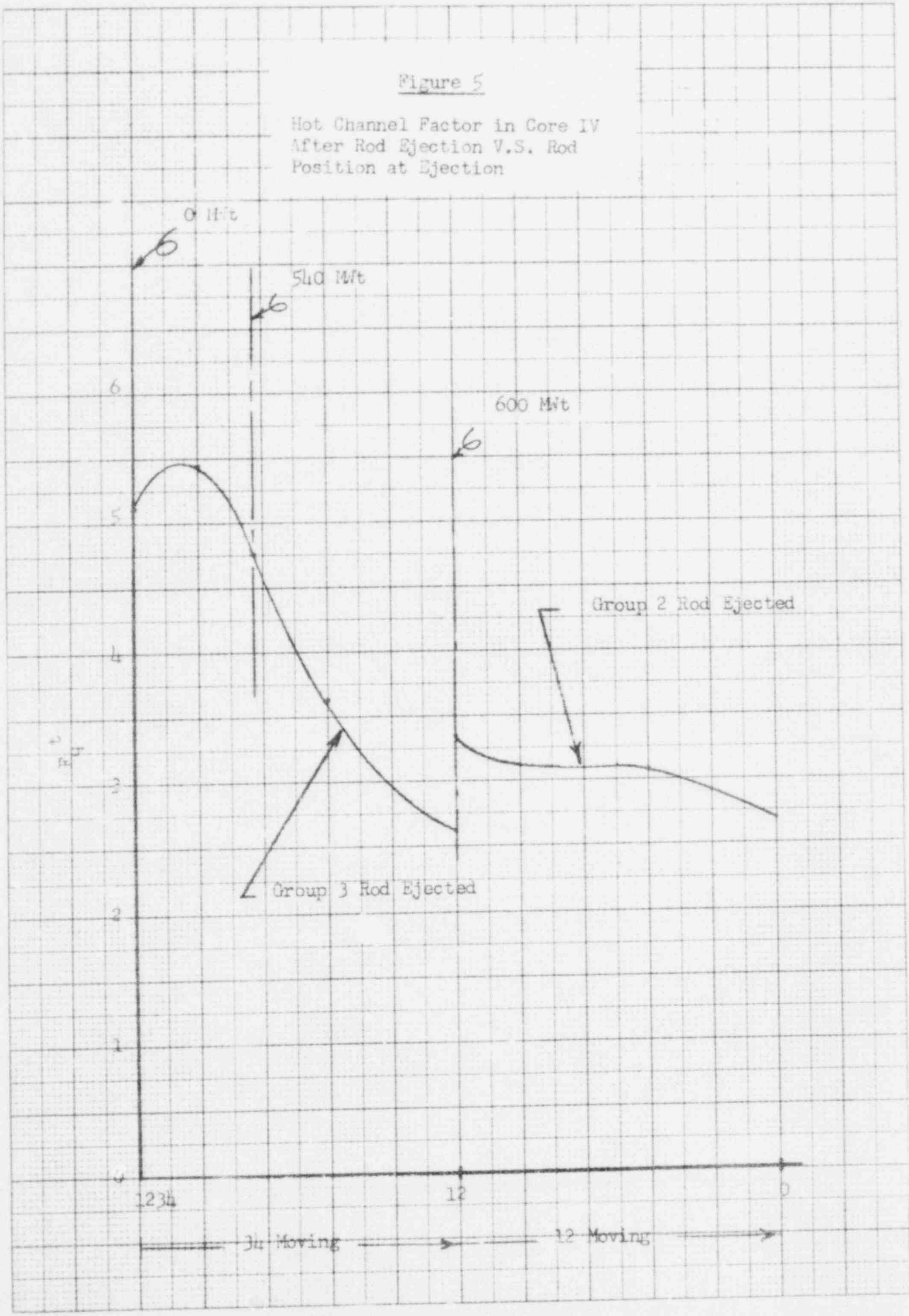
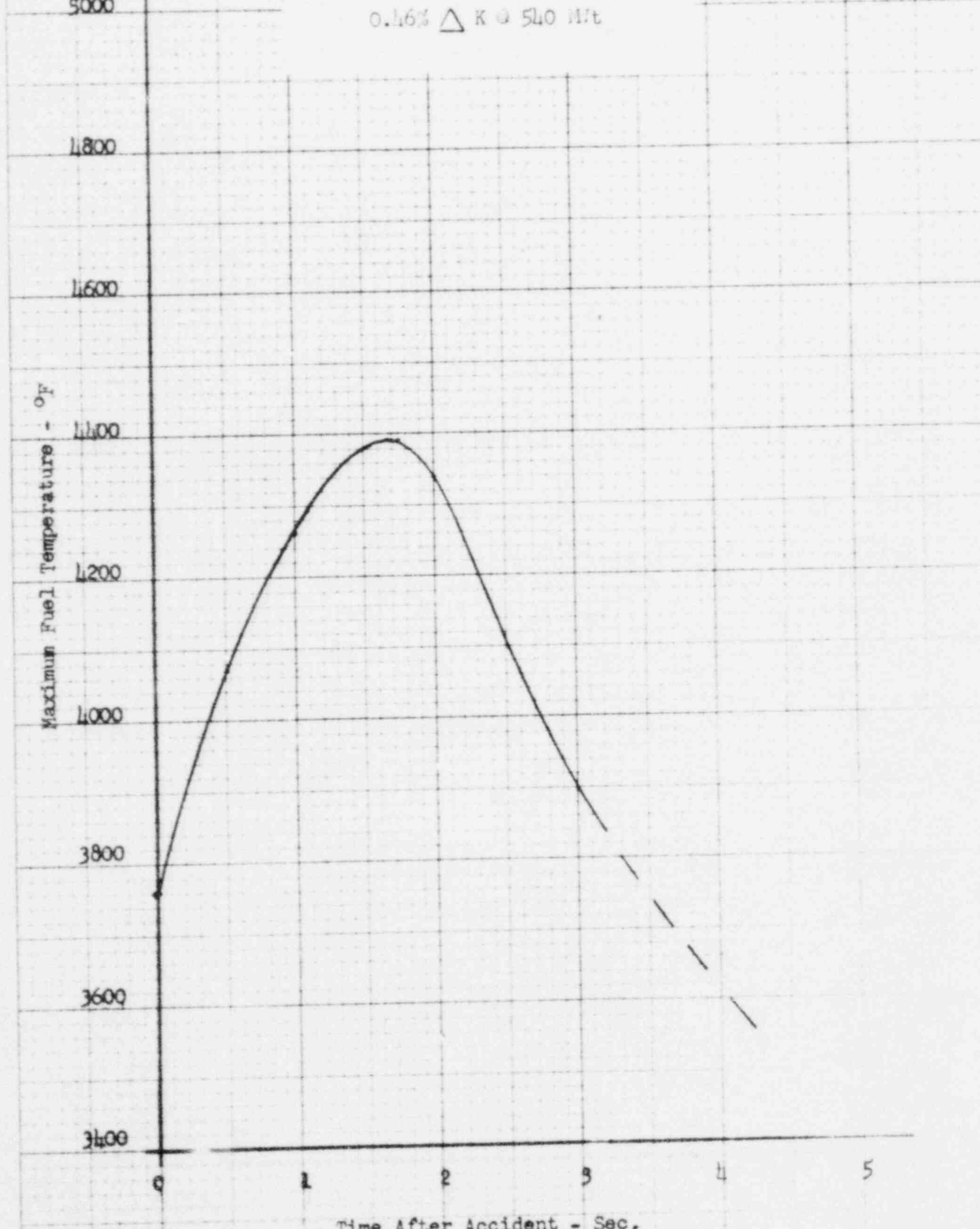


Figure 5
Hot Channel Factor in Core IV
After Rod Ejection V.S. Rod
Position at Ejection
0 MWt
6
540 MWt
6
600 MWt
6
Group 2 Rod Ejected
Group 3 Rod Ejected
6
5
4
3
2
1
0
1234
12
0
34 Moving
12 Moving

Figure 6

Hot Spot Centerline Fuel Temperature (T max) V.S. Time

0.46% Δ K @ 540 MWt

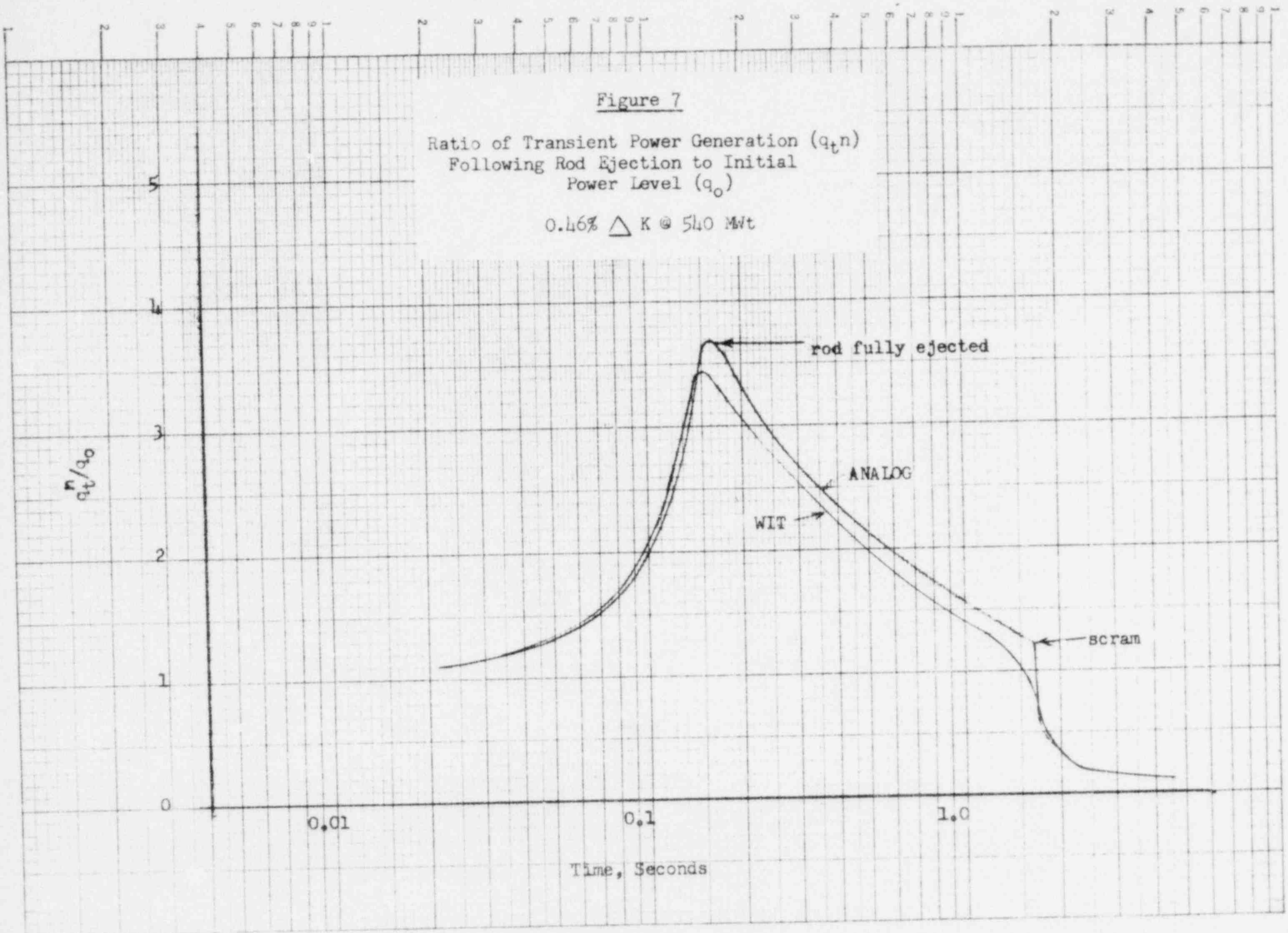


KωE SEMI-LOGARITHMIC 359-81
KEUFFEL & ESSER CO. MADE IN U.S.A.
4 CYCLES X 70 DIVISIONS

Figure 7

Ratio of Transient Power Generation ($q_t n$) Following Rod Ejection to Initial Power Level (q_o)

0.46% Δ K @ 540 MWt



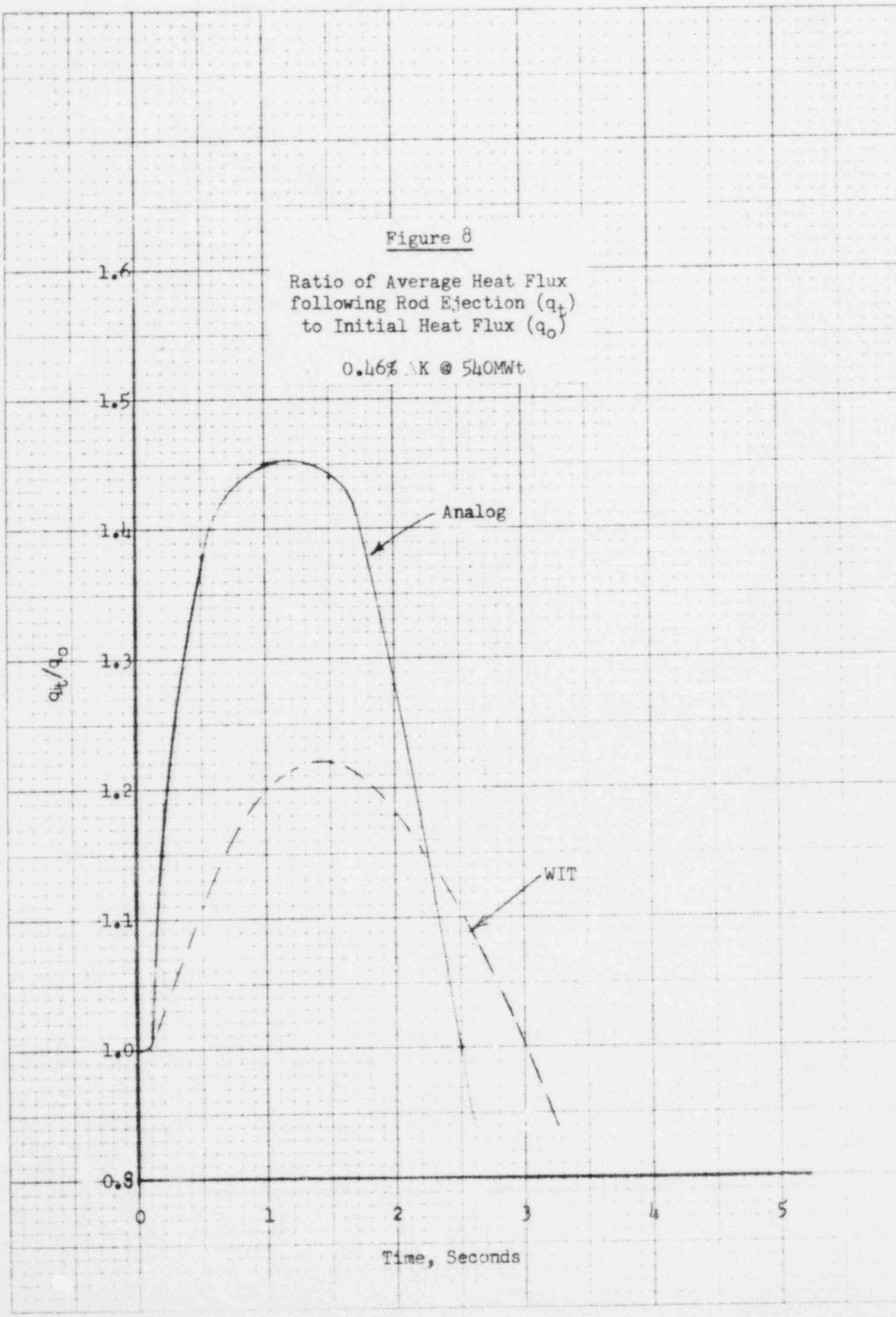


Figure 8
Ratio of Average Heat Flux following Rod Ejection (q_t) to Initial Heat Flux (q_o)
0.46% ΔK @ 540MWt
Analog
WIT
1.6
1.5
1.4
1.3
1.2
1.1
1.0
0.9
q_t/q_o
0
1
2
3
4
5
Time, Seconds

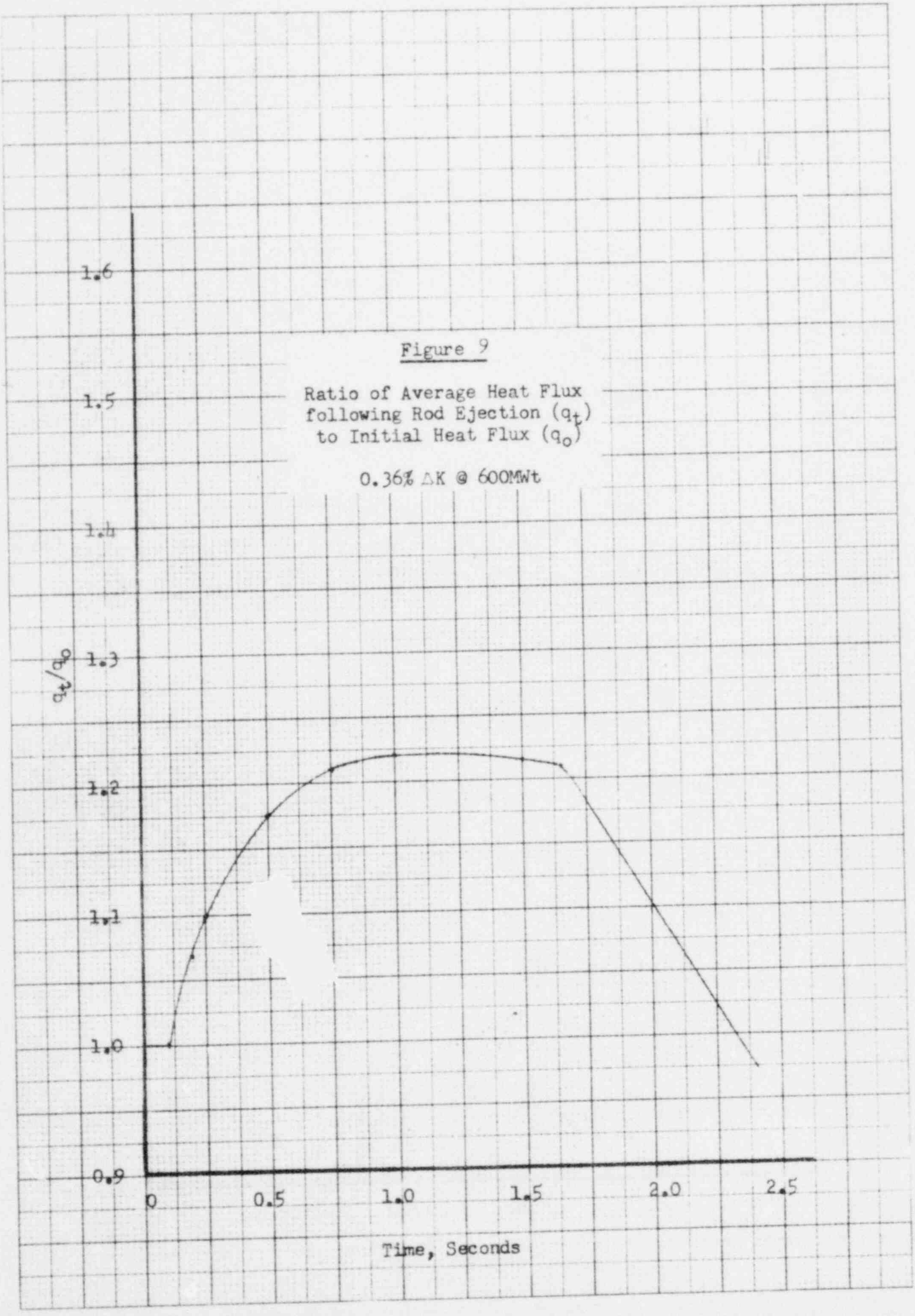


Figure 9
Ratio of Average Heat Flux following Rod Ejection (q_t) to Initial Heat Flux (q_o)
0.36% ΔK @ 600MWt
q_t/q_o
1.6
1.5
1.4
1.3
1.2
1.1
1.0
0.9
0
0.5
1.0
1.5
2.0
2.5
Time, Seconds

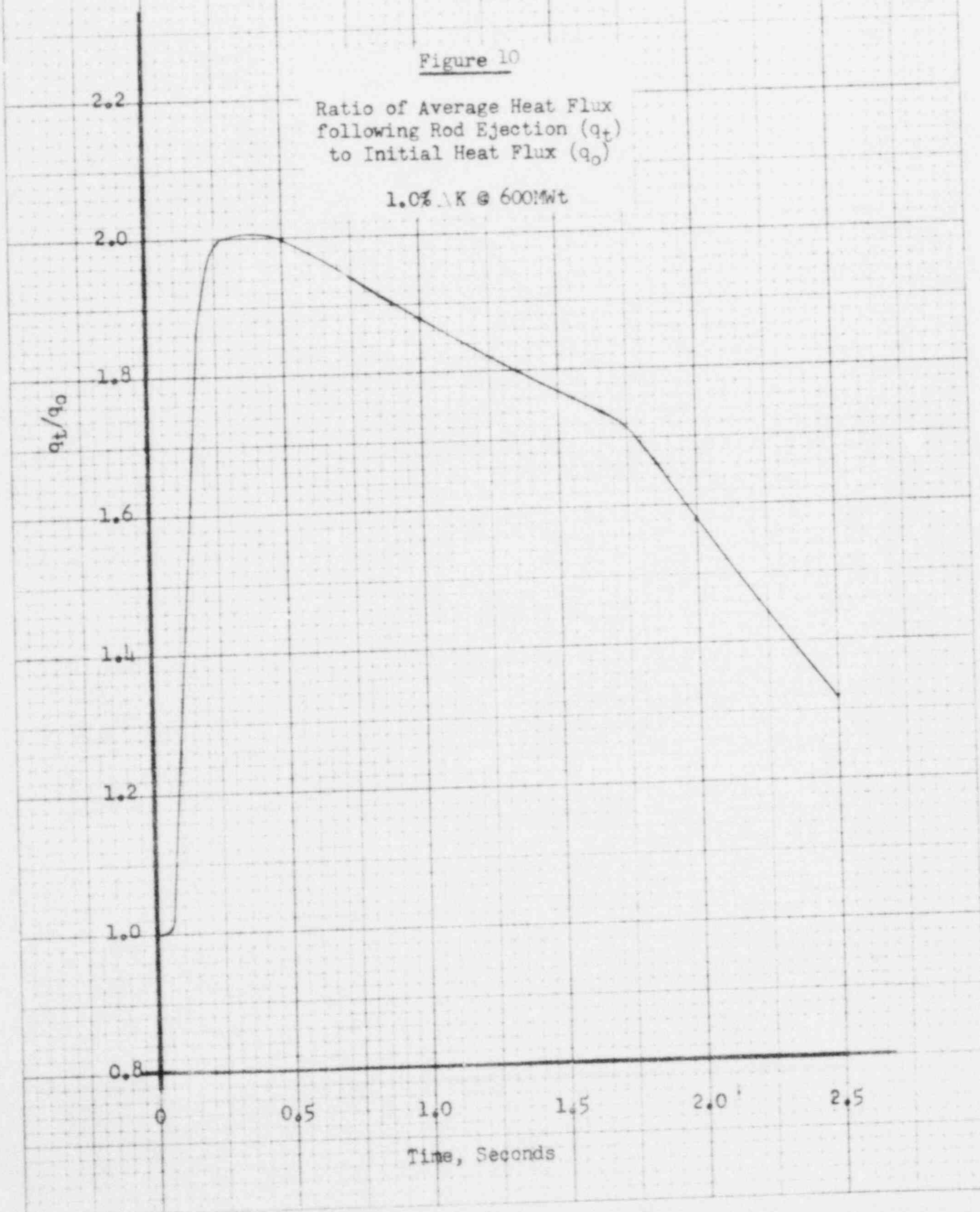


Figure 10

Ratio of Average Heat Flux following Rod Ejection (q_t) to Initial Heat Flux (q_o)

1.0% ΔK @ 600MWt

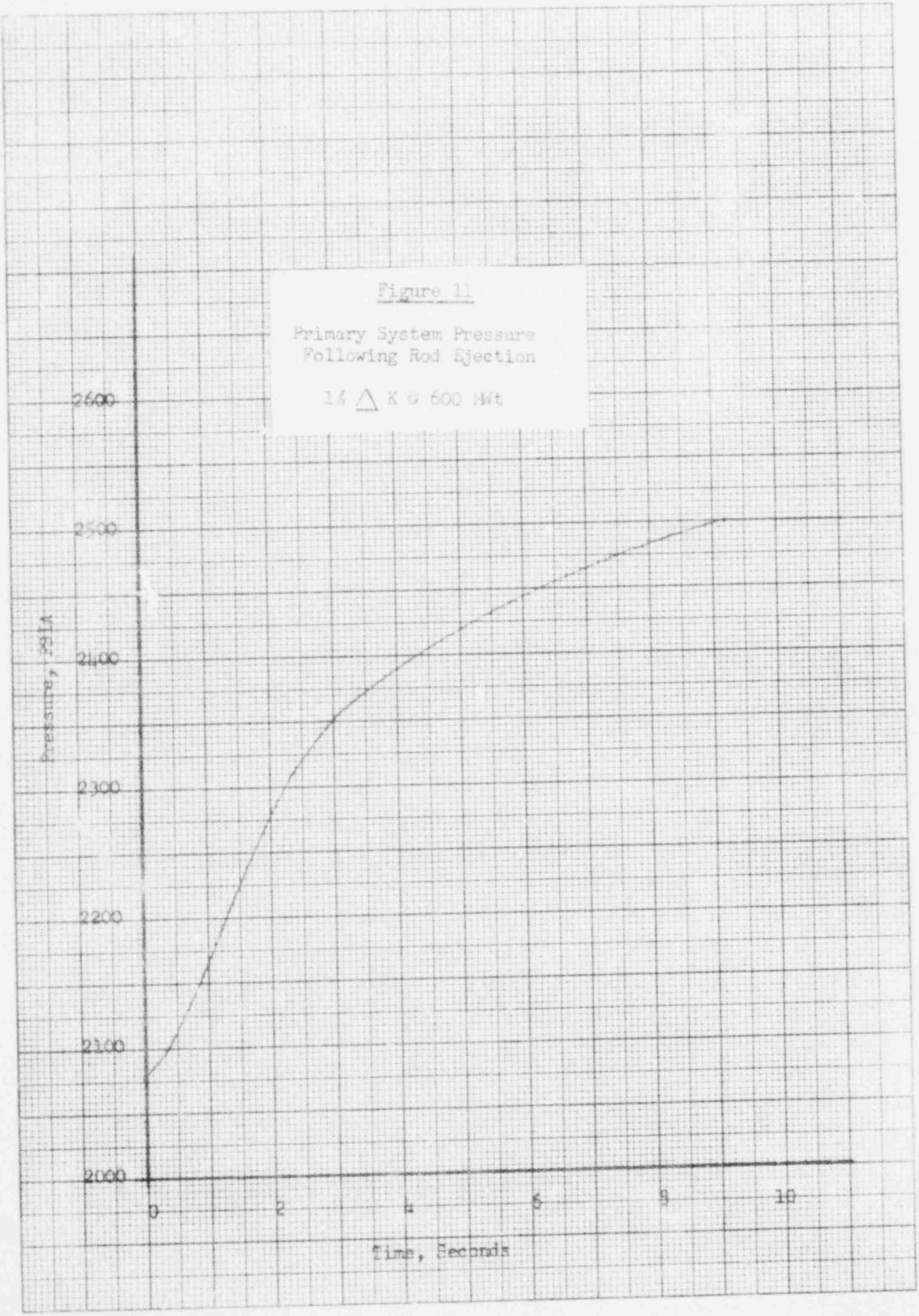


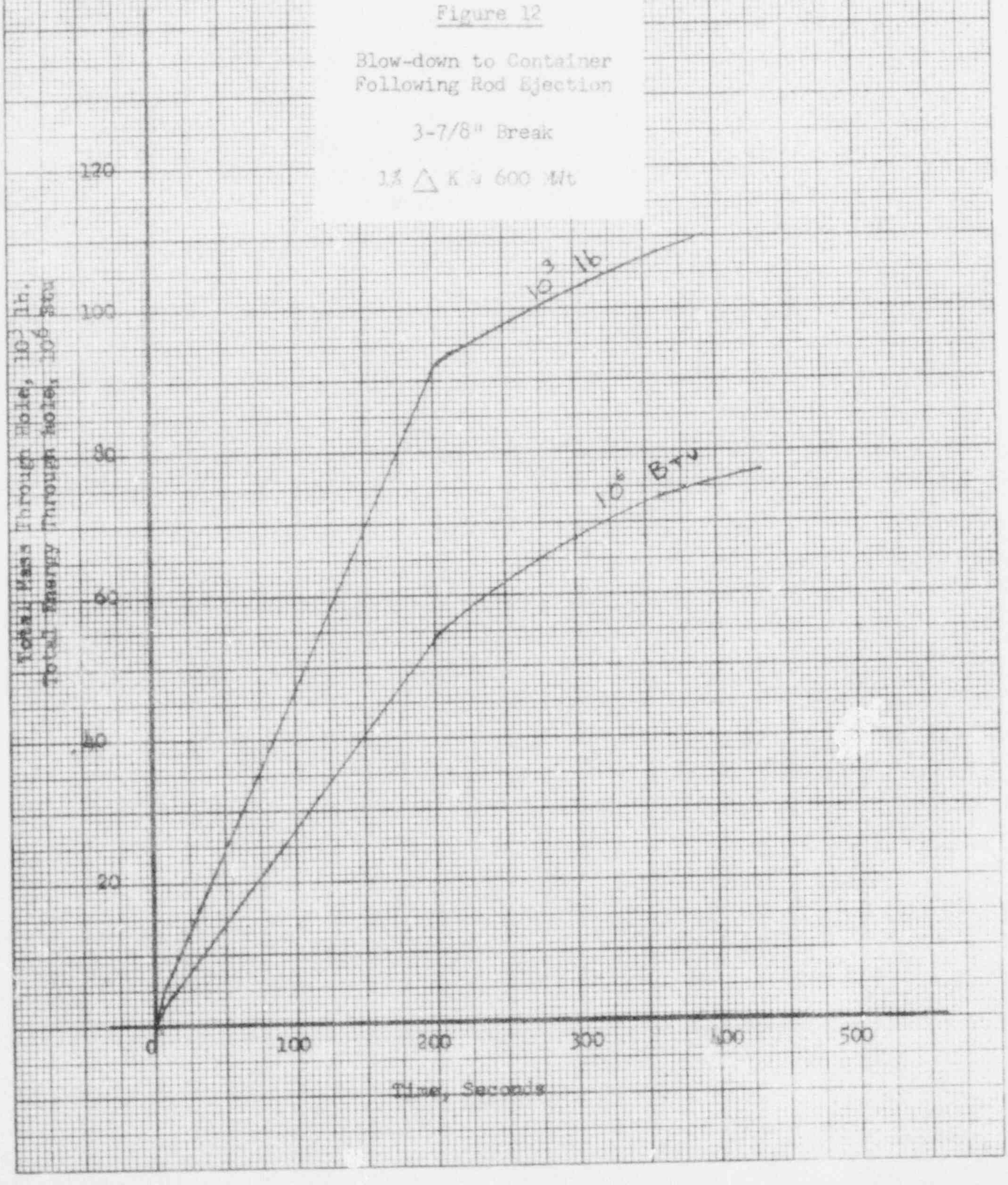
Figure 11
Primary System Pressure
Following Rod Ejection
1% △ K @ 600 MWt
Pressure, PSIA
2600
2500
2400
2300
2200
2100
2000
0
2
4
6
8
10
Time, Seconds

Figure 12

Blow-down to Container
Following Rod Ejection

3-7/8" Break

1% △ K @ 600 MWt



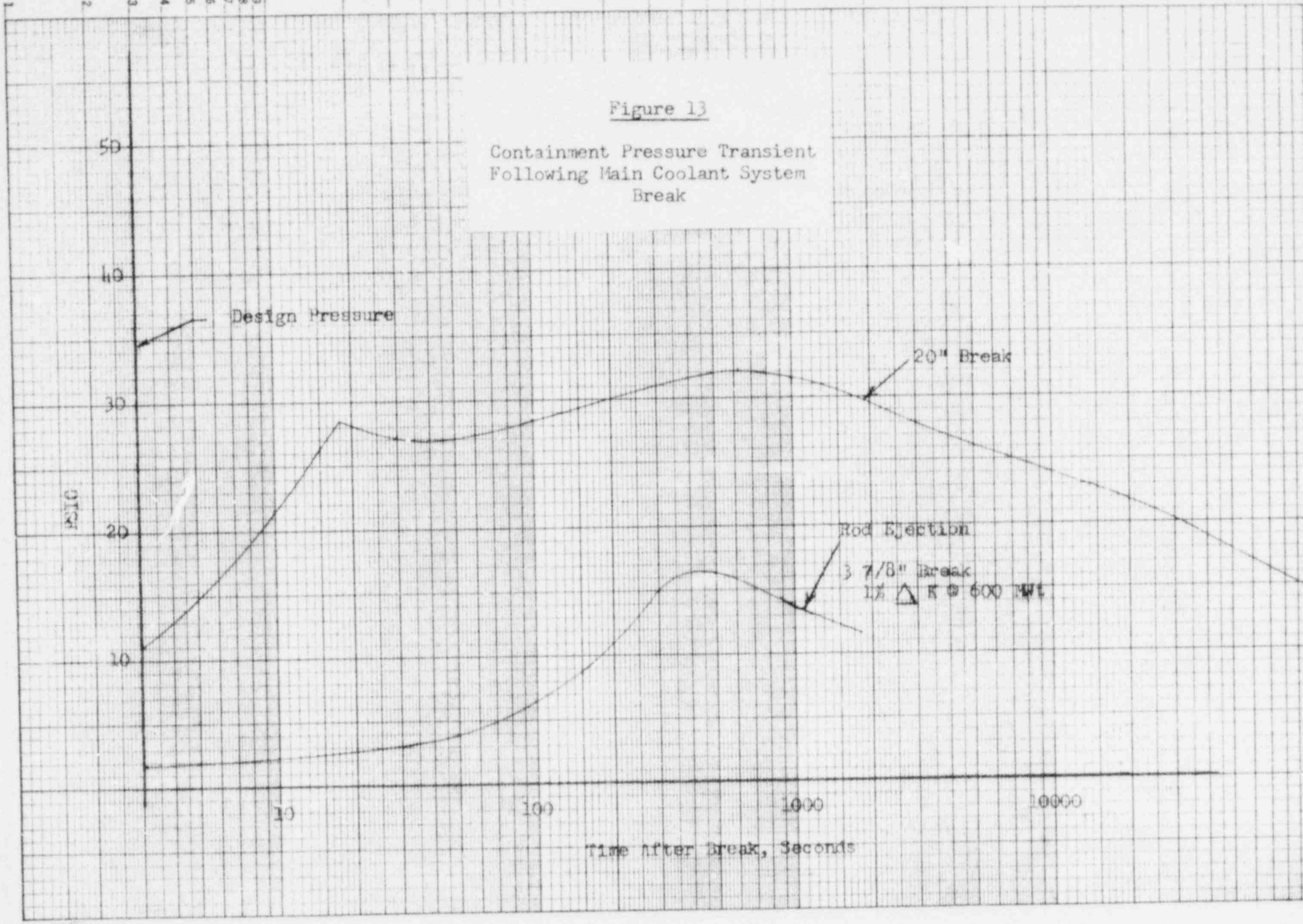


Figure 13

Containment Pressure Transient Following Main Coolant System Break

APPENDIX A

SUMMARY OF STUDY OF PRESSURE WAVES GENERATED DURING REACTOR ACCIDENT

W. A. Stewart
A. T. Pieczynski

1. Introduction

During a period of increased reactivity in a pressurized water reactor, such as following a control rod ejection accident, one or more clad fuel rods may increase in temperature, melt (at $5210^{o}F$), burst the clad and eject molten droplets of UO_2, thus forming an assemblage of molten UO_2 spheres in liquid water. This memorandum summarizes the analysis and results for the pressure waves produced when heat transfers from UO_2 spheres to water and an expanding vapor layer is formed around the spheres. Detailed theory and results will be contained in a report which is being prepared.

2. Analysis

The analysis considers an initial period (lasting 8×10^{-8} seconds for 0.010 inch diameter spheres) during which a single shock front travels in liquid from a sphere boundary into undisturbed saturated water between spheres at 2000 psia. Assuming uniform pressure in the vapor layer and disturbed liquid, the first law of thermodynamics for the vapor layer, as an open system (with the addition of an increment of vapor mass in a unit of time), together with relations for heat transfer processes from UO_2 to vapor layer and from vapor layer to liquid interface, conservation of mass for the vapor-liquid system and for the vapor layer, plus thermodynamic property relations enabled solutions for: mass addition to the vapor layer, average vapor temperature, enthalpy and specific volume and layer thickness for an increment of time. After all of the water in the small cell surrounding a sphere is compressed, in other words pressure waves from neighboring spheres have met, multiple wave addition has the effect of producing pressures like those encountered with a constant volume system of UO_2, vapor layer and surrounding water. Analysis in the second period differs from that during the initial period only in that the compressible water region is not increasing in thickness with the speed of sound in the liquid.

For larger times (typically greater than 5×10^{-5} seconds) water compression waves, which are traveling outward from the assemblage of spheres and water, again permit expansion of the single sphere's water cell region. The analysis utilizes a calculation of vapor layer properties and thickness, as for the second period of time, and a method of characteristic's pressure wave calculation, utilizing as inputs the pressures produced by the second period (constant vapor-liquid volume) calculation and the velocity of the liquid (relative to the assemblage coordinate) from a preceding wave calculation. Since pressure changes by the wave calculation are for an isentropic process in the liquid, work done by the vapor layer system to equalize vapor and liquid pressures must also be calculated. An isentropic process for the vapor layer, to again equalize pressure in a constant vapor-liquid system, predicts the final conditions for pressure, etc. after an increment of time. The three step process can be visualized as a heat transfer and open system process, an isentropic liquid wave process and an isentropic process occurring simultaneously in a small increment of time. The final conditions from one time increment are used as initial conditions for the next time increment at a given spatial increment.

All calculations were made using sufficiently small space and time subdivisions to permit accurate prediction of the continuous phenomena. Wave calculations were made considering the liquid speed of sound constant (2390 ft/sec) since strong shocks (where pressures above 10^5 psia produce speed of sound and compressibility changes) would not be encountered. The methods outlined above are used throughout the assemblage region where there is energy input to the total system and also in the surrounding region up to about two feet from the assemblage. Outside of the assemblage only the wave calculations were used. A Burroughs B-5000 digital computer was utilized for all numerical calculations.

3. Results

Typical results of interest are shown in figures 1 through 5 for a 3 x 3 rod array melting and bursting to form an assemblage that is cylindrical and contains only 0.010 inch diameter UO_2 spheres and water. The pressure wave (figureA1) decreases in maximum amplitude as it travels out from the assemblage and is only 200 psi above the 2000 psia ambient 2 feet from the assemblage. The pressure at the center of the assemblage reached 7400 psia during the short time, high heat flux second period (figureA2), fell sharply following a rarefaction wave reflection at the center and rose again as the ratio of vapor volume to liquid volume increased with time. After one millisecond, all liquid had been removed from the assemblage region. Pressures just inside and outside the assemblage show a similar behavior.

Water velocities (figureA3) are useful in estimating displacement of rods outside the assemblage region. For example, at 1.2 R_o, (just outside the assemblage) integration of the velocity up to 1 millisecond indicates a displacement of 0.097 feet. This conclusion (and the theory) depends on the rods having substantially the same properties as water and not being constrained. Where constraints are applied or the density is much greater than for water, displacements are proportionately less. Average predicted pressures and average liquid velocities remain correct in the absence of significant flow friction. Actual rod displacement may be computed on the basis of pressure impulse in the gradient (figureA1) and frictional flow pressure drop through rod bundles. Actual pressures far outside the assemblage will be somewhat less because of rod bundle drag and viscous losses. The pressures reported are therefore the "worst case" pressures (i.e., for a frictionless fluid).

The average vapor temperature (figureA4) reflects pressure changes in the center of the assemblage. The vapor temperature is above the average of molten UO_2 and saturated water (635°F) temperatures while still transferring less heat to liquid than it receives from UO_2 because of the higher conductivity at the higher temperature near molten UO_2. The maximum total flux of heat to solidify 0.010 inch diameter spheres is greater than 1 Btu/ft^2. As figureA5 shows, part of a UO_2 sphere must still be molten after the principal shock wave is felt and there is no more liquid to vaporize around the partly molten UO_2.

Following complete vaporization of liquid, the vapor will expand ultimately to a lower pressure (2000 psia) and be heated to the UO_2 temperature.

The total work delivered to the surrounding liquid by this "vapor bubble" process has been estimated for the worst case (highest pressure) investigated, a 5 x 5 rod array and 0.010 inch diameter spheres. The work deliverable after 1 millisecond was less than 20% of the mechanical energy already in the water at 1 millisecond.

Comparative results for maximum pressures are shown in Table 1. Pressures near the assemblage are highest for small spheres and large assemblages. Away from the assemblage, the highest pressures result from large assemblages and their large total energy input. Magnitudes of results for other parameters show comparable behavior.

The pressure wave results which are presented here, are without solid boundary interactions. For solid boundary interactions it is to be remembered that the shock or waves are weak enough that superposition applies. One consequence is the doubling of pressure increase for normal incidence of a wave at a reflecting wall, such as the pressure vessel. Thus if the vessel wall is 2 feet from a 3 x 3 (9) rod array, the maximum pressure on the wall would be 2440 psia, instead of 2220 psia where the initial pressure was 2000 psia.

No. of Rods	UO_2 Sphere Size (diameter, in) 0.005	0.010	0.020
1 x 1 (1)		R_o+ 3432 1 ft 2043 2 ft 2030	
3 x 3 (9)	R_o+ 5960 1 ft 2528 2 ft --	R_o+ 4350 1 ft 2310 2 ft 2220	R_o+ 3376 1 ft 2182 2 ft 2130
5 x 5 (25)		R_o+ 4800 1 ft 2950 2 ft 2700	

Table I. Maximum Pressures (psia) at Various Distances from Center of Assemblage (Center of Rod Group). R_o+ is Just Outside (Approx. 1 Rod Away) the Assemblage.

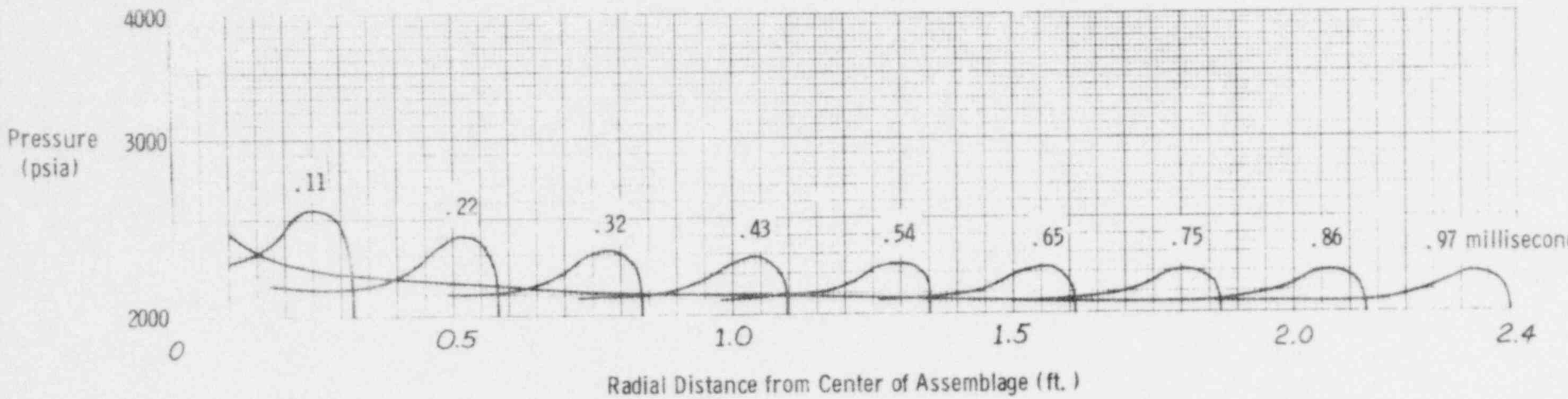


Fig. 1—Pressure wave distribution for 9 fuel rods melting and forming 0.010 inch diameter molten UO_2 spheres at various times in Yankee reactor core.

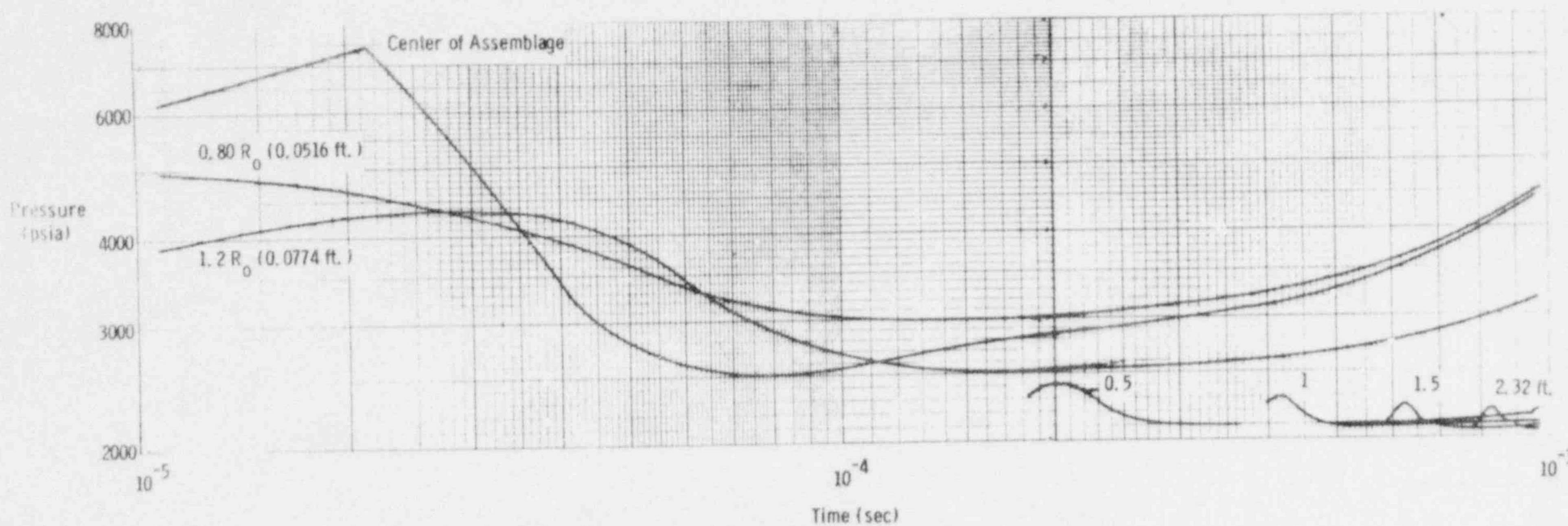


Fig.42—Pressure in water as a function of time for various radial distances from center of assemblage of 0.010 inch diameter molten UO_2 spheres formed from 9 fuel rods in Yankee reactor core.

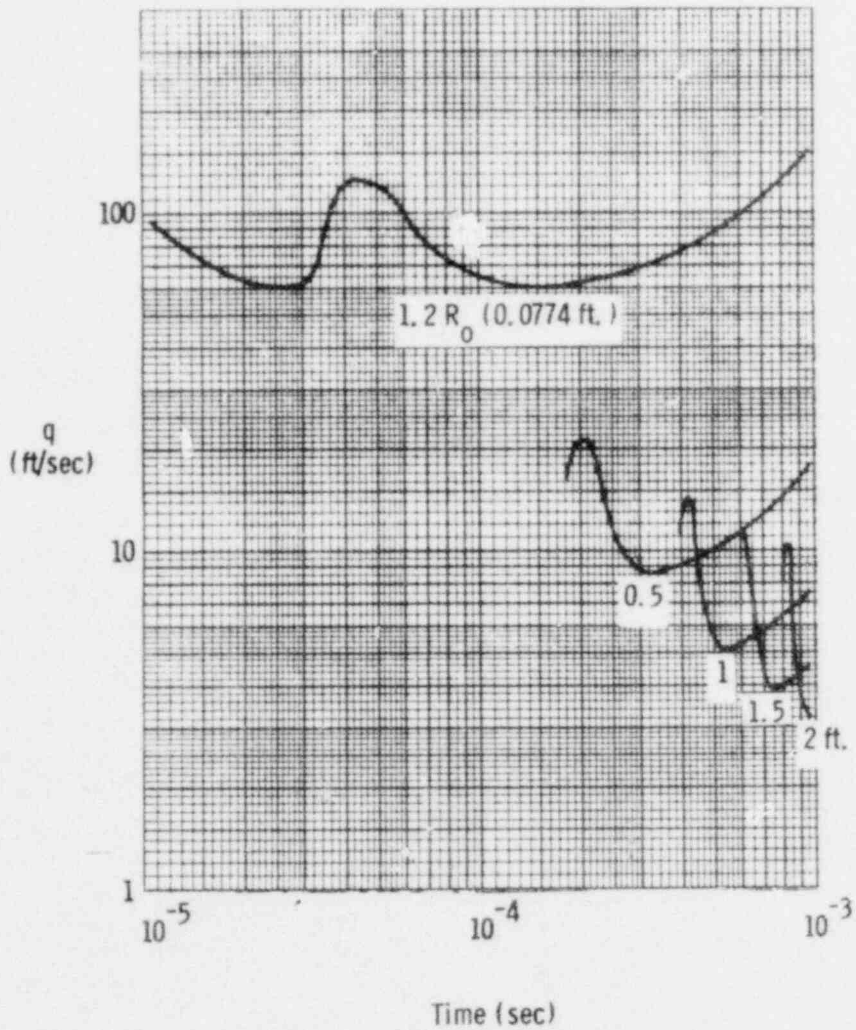


Fig. 3—Velocity in water as a function of time for various radial distances from center of assemblage of 0.010 inch diameter molten UO_2 spheres formed from 9 fuel rods in Yankee reactor core.

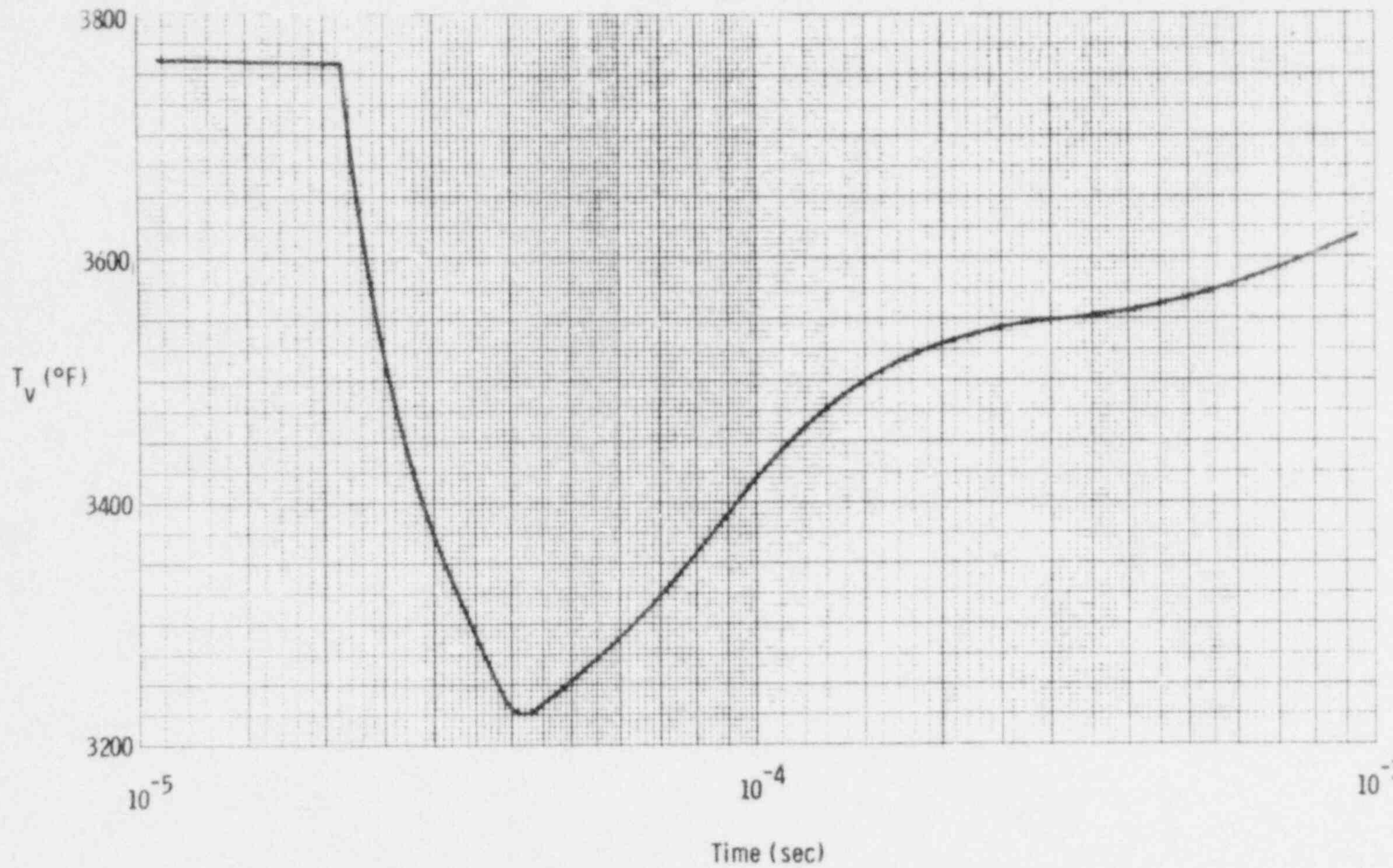


Fig.14—Vapor layer temperature as a function of time for the center of an assemblage of 0.010 inch diameter molten UO_2 spheres formed from 9 fuel rods in Yankee reactor core.

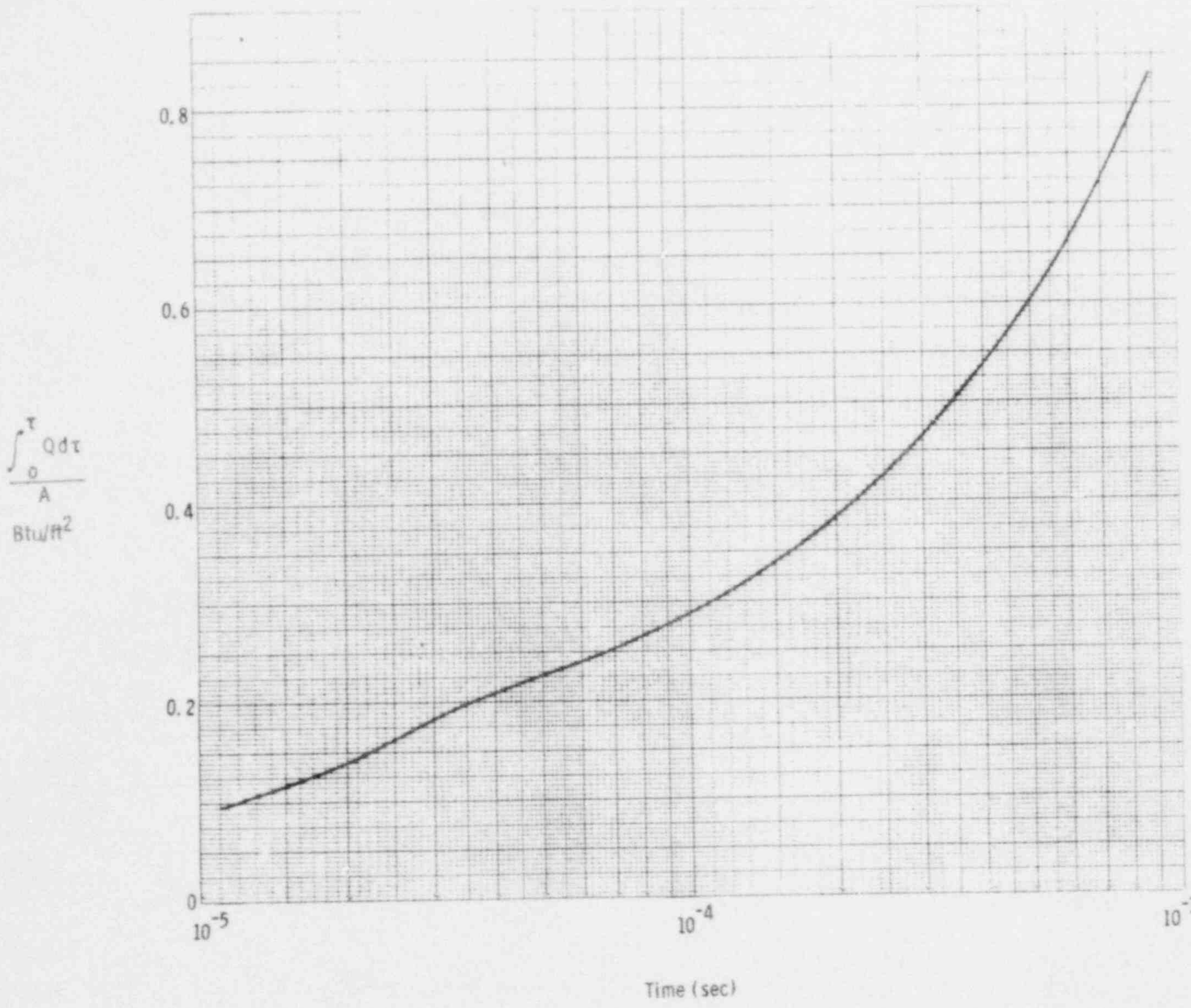


Fig. 5—Total heat flux per unit area of 0.010 inch diameter molten UO_2 spheres in an assemblage formed from 9 fuel rods in Yankee reactor core.