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MASTER

METAL-WATER REACTIONS: II

AN EVALUATION OF SEVERE NUCLEAR EXCURSIONS

IN

LIGHT WATER REACTORS

By

J. I. Owens

G E Class I

June 15, 1959

GENERAL ELECTRIC

ATOMIC POWER EQUIPMENT DEPARTMENT

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J. I. Owens

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I. ABSTRACT

An evaluation of the power and temperature versus time histories of three typical water-cooled and moderated reactors during a nuclear excursion has been made by applying a previously reported method of analysis. These reactors are characterized by one of the following types of fuel elements: U-Al alloy, highly enriched, aluminum jacketted, flat plate; U-Zr alloy, highly enriched, zirconium clad, flat plate; and UO_2, slightly enriched, zirconium clad rods. The validity of the method of analysis was evaluated by comparing the results with experimental data derived from the BORAX I and SPERT I tests. The effects of varying some of the input data, such as heat transfer coefficients, and of considering various nuclear shutdown mechanisms were also obtained.

For reactors with flat plate highly enriched fuel, this evaluation has indicated that the chemical energy available from the reaction of all of the material with water, is several times the amount of energy required to melt the fuel. A reactor period has been found, above which no melting of metal will occur in the fuel elements. For incidents more severe than these threshhold cases which barely produce molten metal (i.e. incidents with shorter reactor periods), it has been shown, for this type of fuel, that only a very narrow range of periods is possible between zero and 100 percent melting. For these high enrichment fuels, the principal mechanism for the shutdown of the nuclear transient is the formation of steam at the surface of the fuel elements. This in

effect reduces the hydrogen density of the system so that the reactor loses reactivity through under-moderation. Gas formation due to radiolysis of water, and other effects, contribute only slightly to the transient termination mechanism.

The results of the analysis for the Zr clad, uranium oxide, low enrichment, rod type fuel indicate that during the nuclear incident, the oxide temperature increases rapidly while the clad temperature is slow to respond. The thermal properties of the fuel elements and the nuclear properties of the core are such that neither the oxide nor the cladding attains the melting temperature of zirconium for the accidents considered. For this fuel with a low U^{235} content, the transient shutdown mechanism arises largely from the Doppler broadening and increase of the absorption cross-section of the U^{238} present, as the temperature is raised. As a result of this, the parasitic loss of neutrons to the non-fissionable uranium in the fuel increases to such an extent that ultimately the reactor mishap is quenched because of a deficiency of neutrons to sustain the reaction. The steam void formation, which is the main process by which shutdown occurs for the enriched fuel systems described above, is of relatively minor importance for this case.

The objective of the Metal-Water study is to combine this and other analytical work with chemical kinetics data to determine the extent and rate of metal-water reaction initiated by a nuclear incident, and to apply the results to the evaluation of the safety hazard resulting from such a metal-water reaction.

II. INTRODUCTION

Since nuclear fuels are fabricated from chemically reactive materials such as aluminum, zirconium, and uranium, it is possible to release large amounts of energy in the reaction

$$\text{Metal} + \text{Water} = \text{Metal Oxide} + \text{Hydrogen Gas} + \text{Heat}$$

This chemical reaction is unique in that it appears to be the only primary process, other than the nuclear reaction itself, which can add energy to the reactor system during a severe nuclear excursion.

Extensive laboratory studies at many different sites have indicated that rapid and dangerous metal-water reactions of this type cannot be expected to occur unless the metal is at an elevated temperature and finely dispersed. These conditions appear to require that the metal be molten. There are two basic mechanisms that can conceivably lead to this situation in a water cooled and moderated reactor: a severe nuclear incident and a loss of coolant accident.

The following evaluation of nuclear excursions in light water cooled reactors is part of the work performed to investigate the magnitude and rate of metal-water reactions under severe reactor incident conditions. For the three reactors described in Table 1, this report presents data on the relationship between power, temperature and time as a function of initial reactor period. The reactors are characterized by one of the following types of fuel elements: (1) U-Al alloy, highly enriched, aluminum jacketted, flat plate; (2) U-Zr alloy, highly enriched, zirconium clad, flat plate; and (3) UO_2 slightly enriched zirconium clad rods. Hereafter, reactor cores containing these fuel elements may be referred to as the Al, Zr, or UO_2 reactors. This evaluation was limited to short reactor periods, from low power and temperature conditions, that potentially could result in rapid metal-water reactions.

III. METHOD OF SOLUTION - COMPUTER SETUP

The basic method of solution is outlined in Reference A, in which an analog computer model is developed for an Al alloy fueled reactor. This method was modified (See Appendix) to include different shutdown mechanisms to accommodate the reactor fueled with UO_2 and to allow faster transients to be studied.

The effect of radiolytic gas formation was omitted in these new calculations. In Reference A, it is shown that the importance of water decomposition on the reactor transient parameters is not great; and, in fact, the omission of this process from the calculated model gives significant improvement in the results, as can be determined by comparison with the BORAX I experimental data. As temperature distributions within the fuel are highly desirable, the model in Reference A was expanded to provide this information.

The calculated periods were obtained by introducing a step increase of reactivity. While this is a somewhat over-simplified type of accident, it is believed to be realistic for this study where the excursion is begun at source level and the required period is so short. For this condition, the amount of reactivity that must be compensated is almost exactly that which would be required if the insertion had been linear with time, provided the ramp is terminated a few decades below the point where shutdown mechanisms become important. Therefore, the excursion will be practically identical whether a step in reactivity or a ramp, that terminates before one percent peak power is reached, is used.

IV. RESULTS

Figure 1 indicates a comparison between the predicted behavior and actual Borax transients. It will be noted that the total energy released in the excursion compares favorably with that calculated by the technique developed. The deviation (approximately 10% at 5 ms*) at very short periods could be due to actual shutdown mechanisms not included in the analysis such as mechanical deformation of the core. It might also arise from experimental uncertainty since very few data points are available for periods shorter than 10 ms. In any case, the agreement is considered adequate for the purpose of this evaluation.

Examples of the transient temperature and power data obtained are presented on Figure 2 for the UO_2 fueled reactor and on Figure 4 for the highly enriched plate type fueled reactors. It should be noted that each set of data is for a different initial period.

Figures 3 and 4 illustrate a power excursion for the three reactors studied. The systems have such dissimilar characteristics that direct comparison is difficult. In general, however, the Al and Zr highly enriched, flat plate fuels respond as would be expected qualitatively. The poorer thermal diffusivity of Zr allows less heat to enter the water, and consequently, the Zr reactor experiences larger power excursions due to the greater length of time to shut down. Comparison of the curves obtained by this computational technique with experimental data (that is, the BORAX and SPERT results) shows that while the total energy is given rather accurately by the model, the shape of the power versus time curves is in somewhat poorer agreement. The experimental curves do not show the fairly slow falling off past the power peak illustrated by the Al curve in Figure 3; rather they tend to be nearly symmetrical about the peak, like the curve labelled UO_2 in this figure. Thus, there is a shortcoming in the analysis, which is believed to arise because the model assumes a unique dependence of steam void on fuel temperature. This is reasonably correct for short times - until the power peak, for example; however, for longer times it is incorrect since water inertia is no longer overriding and the steam can continue to expand. Note that this phenomenon is not observed with the low enriched oxide fuel since the principal shutdown mechanism in this case is the Doppler temperature coefficient.

In order to determine the amount of molten metal produced the important characteristic of the system is the integral of the power-time curve, that is, the energy associated with the transient. This, from Figure 1, is given quite accurately by the model used. Figure 5 illustrates the effect of period on (1) the peak fuel temperature normalized to the melting temperature, and (2) on the percent of molten metal produced. The Al and Zr plate fuels behave as expected since the low thermal diffusivity of the Zr system results in higher temperatures for longer period accidents. One result which is somewhat surprising

* The abbreviation "ms" will be used for milliseconds.

is the very narrow range of accidents between initial and complete melting of the core. It is also interesting to note for the UO_2 core that even at the 4.34 ms period excursion the UO_2 and Zr clad temperatures are below the melting temperature of zirconium. To achieve melting of the Zr clad periods of less than about three milliseconds would apparently be required. Whether faster periods than this can ever be achieved for a low enrichment UO_2 reactor is a problem for the system designer and safeguard engineer to evaluate for the specific system under consideration. It seems improbable.

Figure 6 supplements the information given in Figure 5 by showing the energy released in these excursions. The relatively lower amount of energy produced in the Al fuel compared to the Zr plate should be of interest for research and test reactors. The dashed lines show the total energy released assuming all the metal present to react chemically with the water added to the nuclear excursion energy. This assumption is not justified, since the extent of the chemical reaction will be largely determined by the amount of molten metal available. The dotted lines for the two high enrichment fuels show the upper limit of the effect taking this factor into account. Any real energy release due to a metal-water reaction must lie in the shaded region between the solid and the dotted curves.

Figures 7 and 8 give the temperature distribution as a function of period and time for the Zr flat plate element and the UO_2 Zr-clad fuel. These curves have been useful in determining the amount of molten metal available and in performing stress analyses on these fuel elements. The flat plate Zr element will deform, but probably not rupture before melting. In the case of the Al fuel, the high thermal conductivity of the material will result in a nearly uniform temperature distribution across the solid. Stress analysis of the Zr clad, UO_2 filled element (Reference B) indicates that the yield strength of the cladding will be exceeded long before the oxide becomes molten. The exact nature of the fuel distortion or failure will be discussed in detail in a forthcoming report in this series (B). It is expected, however, that the core will be deformed following excursions with initial periods of a few milliseconds or less. Figure 9 indicates the maximum strain in the clad versus the reactor period, due to the differential expansion of the oxide and the clad. These strains are far below the values of 7 to 10% which it is believed irradiated zirconium can stand without rupture (B); but are sufficient to produce significant mechanical deformation.

Figures 10 and 11 provide a study of the importance of various shutdown mechanism for the low enrichment UO_2 rod. In Figure 10 the solid curve indicates the power transient with all important shutdown mechanisms in effect. The lower dashed curve is the same transient assuming the fuel to be perfectly insulated thermally. The apparent inconsistency of the lower peak power with no heat loss is due to the importance of the Doppler temperature coefficient. Insulating the fuel produces somewhat higher temperatures that more than compensate for the smaller amount of steam void that would have been formed due to lower conduction. If the rods were larger than the assumed 0.45" O.D. (See Table 1), there would be less difference between these two cases. The upper dashed curve indicates the effect of neglecting the neutron and gamma heating in the water. It can be seen that this is an important mechanism, reducing the peak power by more than a factor of two. The remaining dashed curve illustrates the very important effect of the Doppler coefficient. This parameter was assumed constant at 1.67×10^{-5} per ^{o}F. It is known that the coefficient will be lower at elevated temperatures; however, for lack of detailed information this decrease in the parameter was ignored. Perhaps future studies should include the variation, although it is not expected to alter the basic conclusions drawn. In Figure 11 the effect of assuming an infinite heat transfer coefficient between meat and clad is shown to be small. This again illustrates strongly the weak influence of the heat being conducted out of the fuel, and the fact that a slight reduction in fuel temperature has a greater influence on the Doppler coefficient, due to the low enrichment, than it has on the steam void coefficient.

V. CONCLUSIONS

The techniques illustrated above are applicable to any reactor, and can be used in safeguards evaluation. Even the somewhat generalized and diffused examples considered show trends which should be of considerable usefulness. One of the most striking examples is illustrated in Figure 5 where it is noted that, for the flat plate fuels, the range from the period which (1) produces liquid metal to that which (2) completely melts the core, is extremely short. For the Al core for example, a period of 4.9 ms does not result in fuel melting*; while at 3.3 ms all the fuel is molten before termination of the reactor excursion. The same trend is seen in the Zr fuel, and the respective periods are 12.2 and 8.6 ms. This deviation in period for melting is to be expected due to the relatively poorer thermal diffusivity (resulting in slower void formation) in spite of the higher melting point of Zr. However, the short range of nuclear transient accidents in which there is only partial core melting suggests that it is not reasonable to assume less than total melting for any system which is considered capable of achieving periods that will produce any melting at all.

The low enrichment UO_2-Zr clad element has a long thermal relaxation time and as a result transfers relatively little heat into the water during a rapid transient. This results in oxide temperatures that are reasonably uniform across the radius, and clad temperatures that are low compared to the melting point (See Figure 8). Stress analysis of the element (Reference B) indicates that the yield strength of the cladding will be exceeded long before it becomes molten. In such a system, a nuclear transient insufficiently rapid to produce molten metal (and consequently a metal-water reaction) may still result in significant core distortion and deformation. It is easy to conclude, therefore, that large metal-water reactions, for this type of accident, will not readily occur for low enrichment high melting point fuel. Other accidents, such as loss of coolant and accidents while the core is at full power, should be examined for this and other fuel types before concluding that the metal-water reaction hazard can be ignored as a safety problem.

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* One of the few cases on record where a metal-water reaction was reported to have taken place following a reactor mishap was the NRX meltdown which occurred at Chalk River on December 12, 1952. In this reactor, the fuel was low enrichment uranium jacketted with aluminum and moderated with heavy water. Thus, both the NRX reactor and incident are quite different than the cases considered in this report. A review of the published data on this mishap indicates that this accident resembles a loss of coolant incident more than the nuclear transient case examined in this report.

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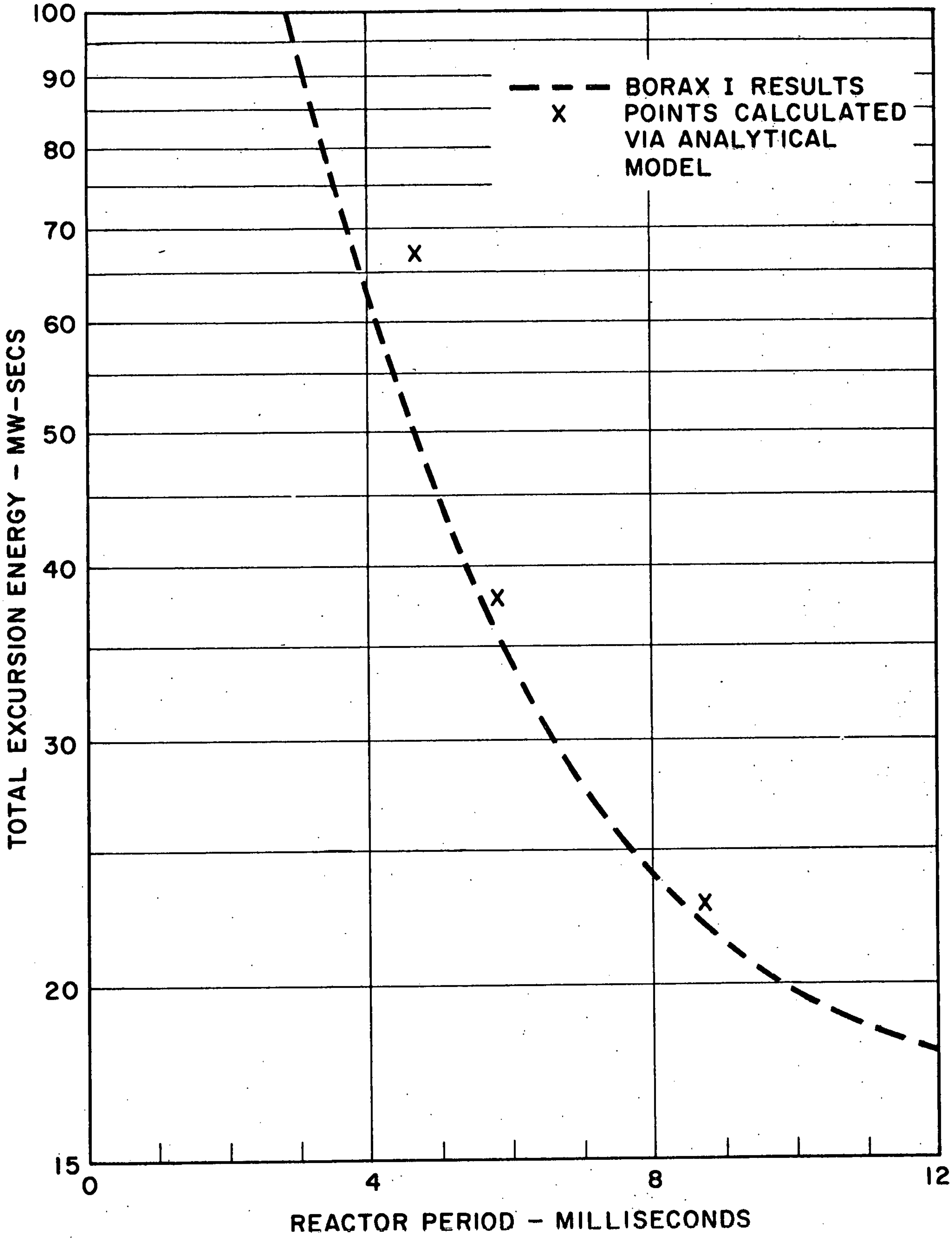
C. Blomberg, P., Hellstrand, E., and Horner, S. (Part 1); also Brimberg, S. and Dahlstrom (Part 2): Measurement and Calculations of the Temperature Coefficient of the Effective Resonance Integral in Uranium Metal and Oxide. Paper No. A/Conf. 15/p/150, 2nd International Conference on Peaceful Uses of Atomic Energy. (Geneva, 1958).

TABLE I

REACTOR CHARACTERISTICS

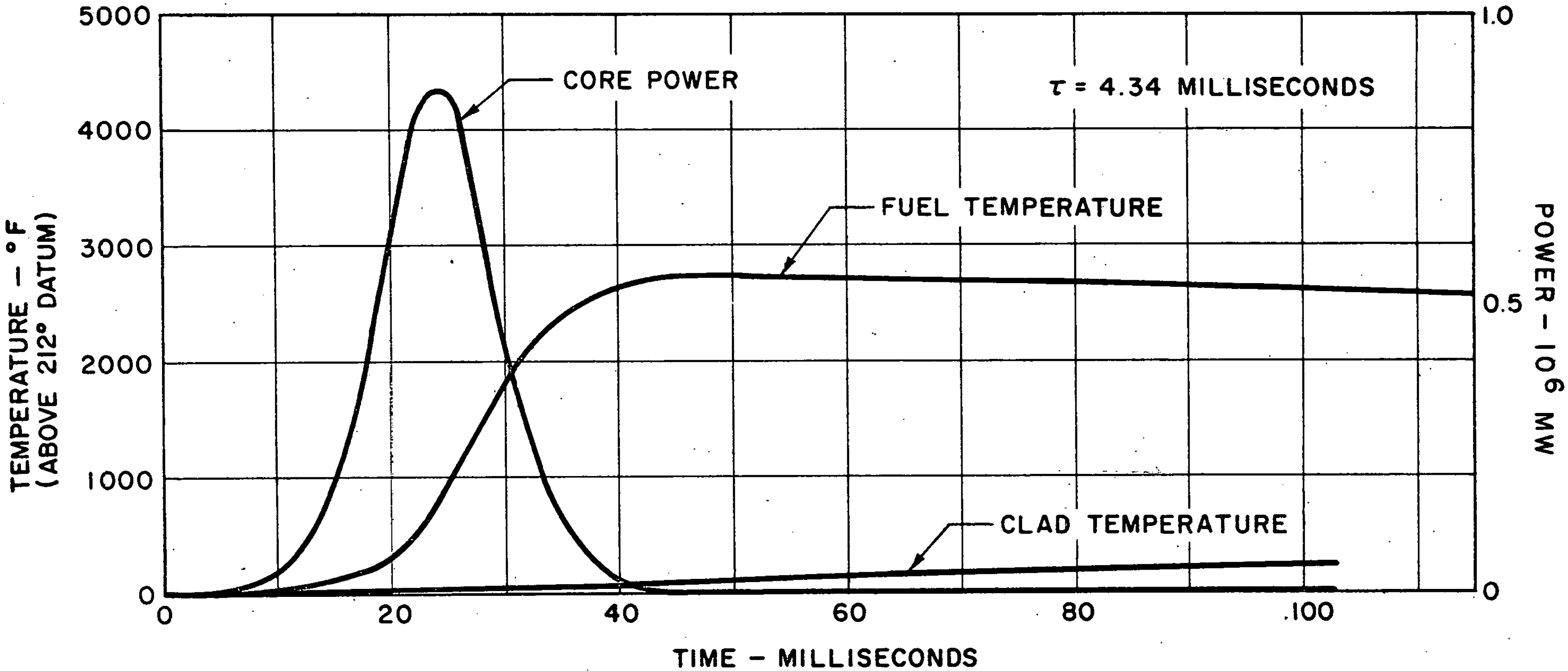
	Case 1	Case 2	Case 3
Fuel	Al-U alloy	Zr-U alloy	UO_2
Clad	Al	Zr	Zr
Enrichment (% U^{235})	93.5	93.5	2.5
Weight fraction of U in fuel - %	17	7	88
Meat Thickness (Inches)	0.020	0.020	---
Clad Thickness (Inches)	0.020	0.020	0.025
Meat Diameter (Inches)	---	---	0.40
Doppler Coefficient $\left[(\Delta K/K)(^{o}F^{-1})\right]$	0	0	-1.67×10^{-5}
Temperature Coefficient $\left[(\Delta K/K)\ (^{o}F^{-1/4})\right]$	-0.00106	-0.00106	-0.00106
n and γ Heating of Water $\left[(\Delta K/K)(^{o}F^{-1/4})\right]$ (corresponding to 3% reactor power)	0	0	-0.00106
Neutron Lifetime (Sec)	6.5×10^{-5}	6.5×10^{-5}	6.5×10^{-5}
Delayed Neutron Fraction	0.0075	0.0075	0.0075
*Initial Pressure (Psia)	14.7	14.7	14.7
*Initial Temperature (^{o}F)	212	212	212
Thermal Conductivity of Fuel (BTU/Hr-Ft-^{o}F)	100	8	1.15
Thermal Conductivity of Clad (BTU/Hr-Ft-^{o}F)	106	10.85	10.85
Heat Capacity of Fuel (BTU/Lb-^{o}F)	0.183	0.08	0.07
Heat Capacity of Clad (BTU/Lb-^{o}F)	0.183	0.08	0.08
Total Metal in Fuel Region (Lbs)	84	216	57,800
Total Metal in Clad Region (Lbs)	168	432	9,900
Heat Transfer Coefficient, Fuel Surface to H_2O, (BTU/Hr-Ft^2-^{o}F)	1000	1000	1000

* These initial conditions were arbitrarily selected; but it is believed that the transient behavior of a nuclear reactor is relatively insensitive to these parameters.



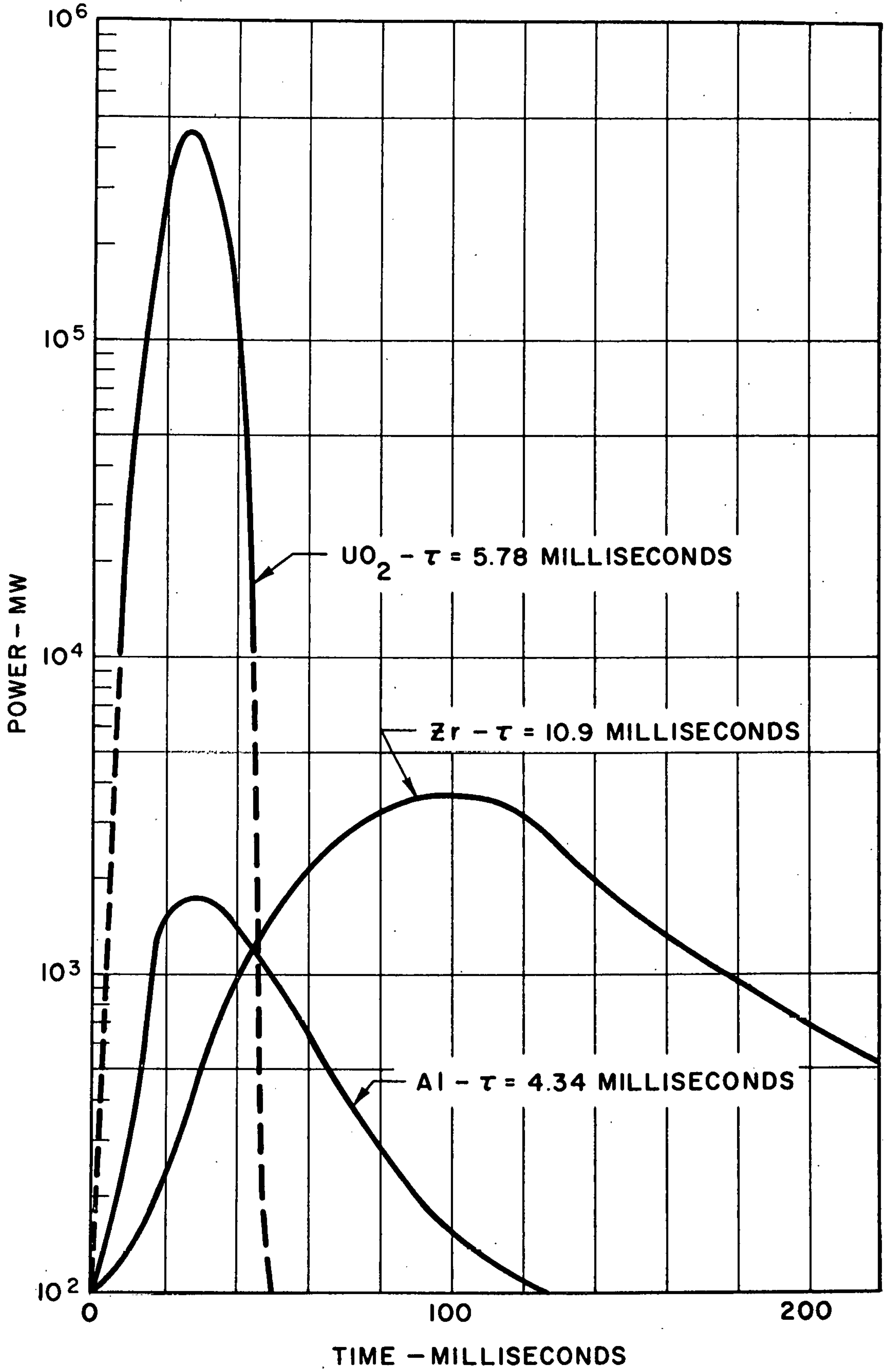
ENERGY PRODUCED IN A BORAX I NUCLEAR EXCURSION

FIGURE 1



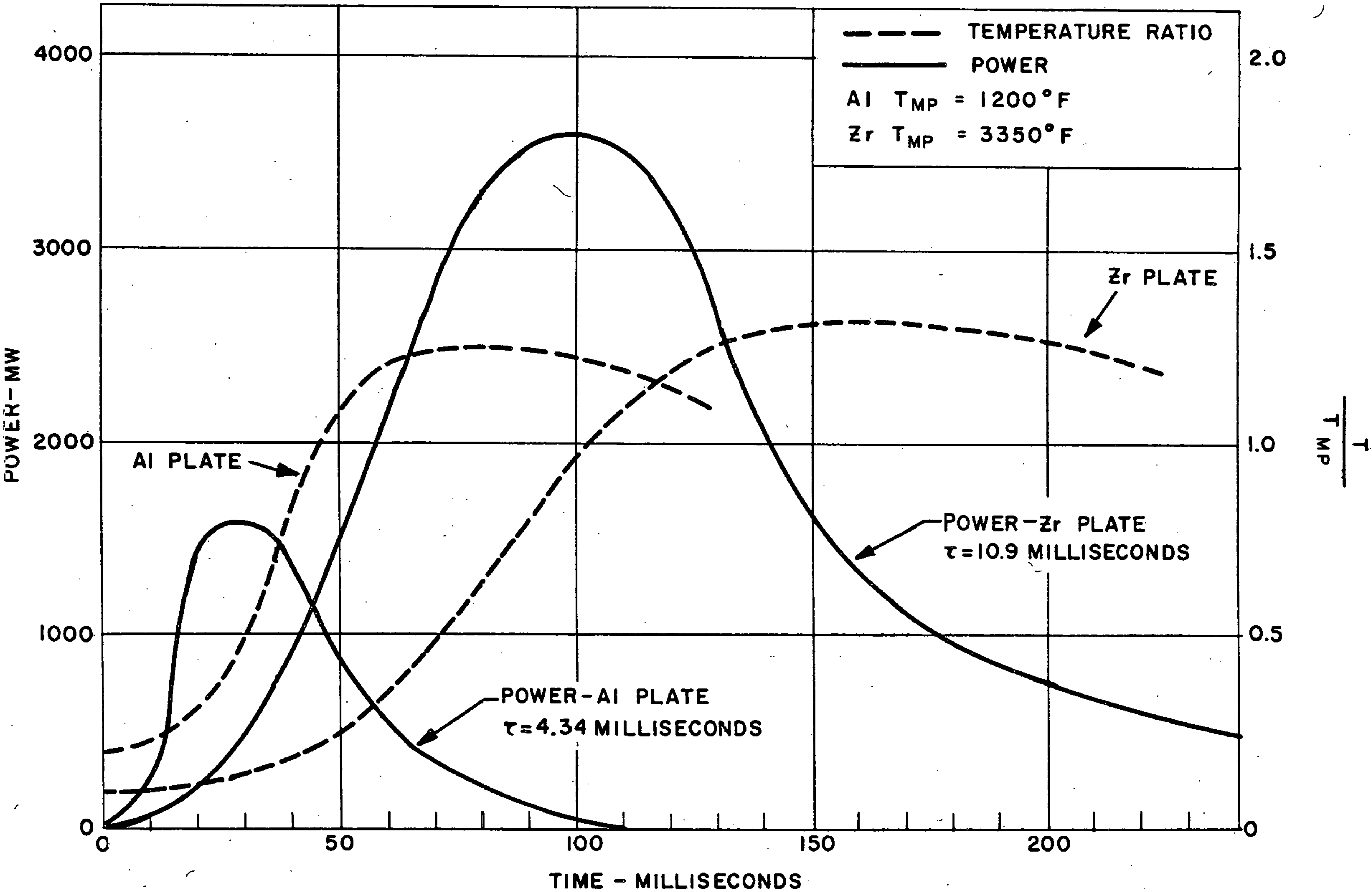
TYPICAL POWER AND TEMPERATURE EXCURSION
FOR A UO_2 Zr CLAD ELEMENT

FIGURE 2



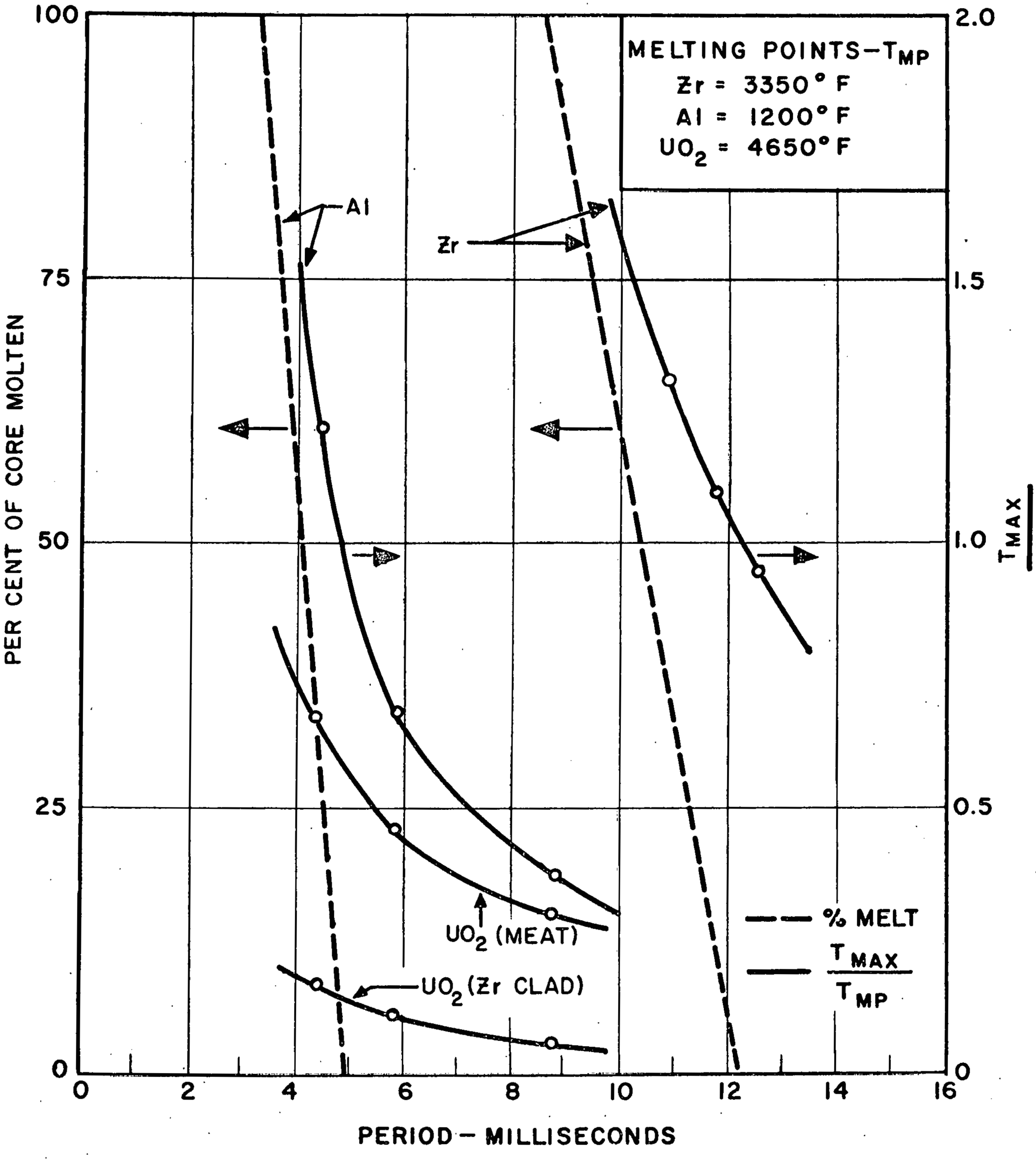
COMPARISON OF POWER BURSTS FOR
THE THREE REACTORS VS TIME
NOTE DIFFERENCE IN INITIAL PERIOD

FIGURE 3



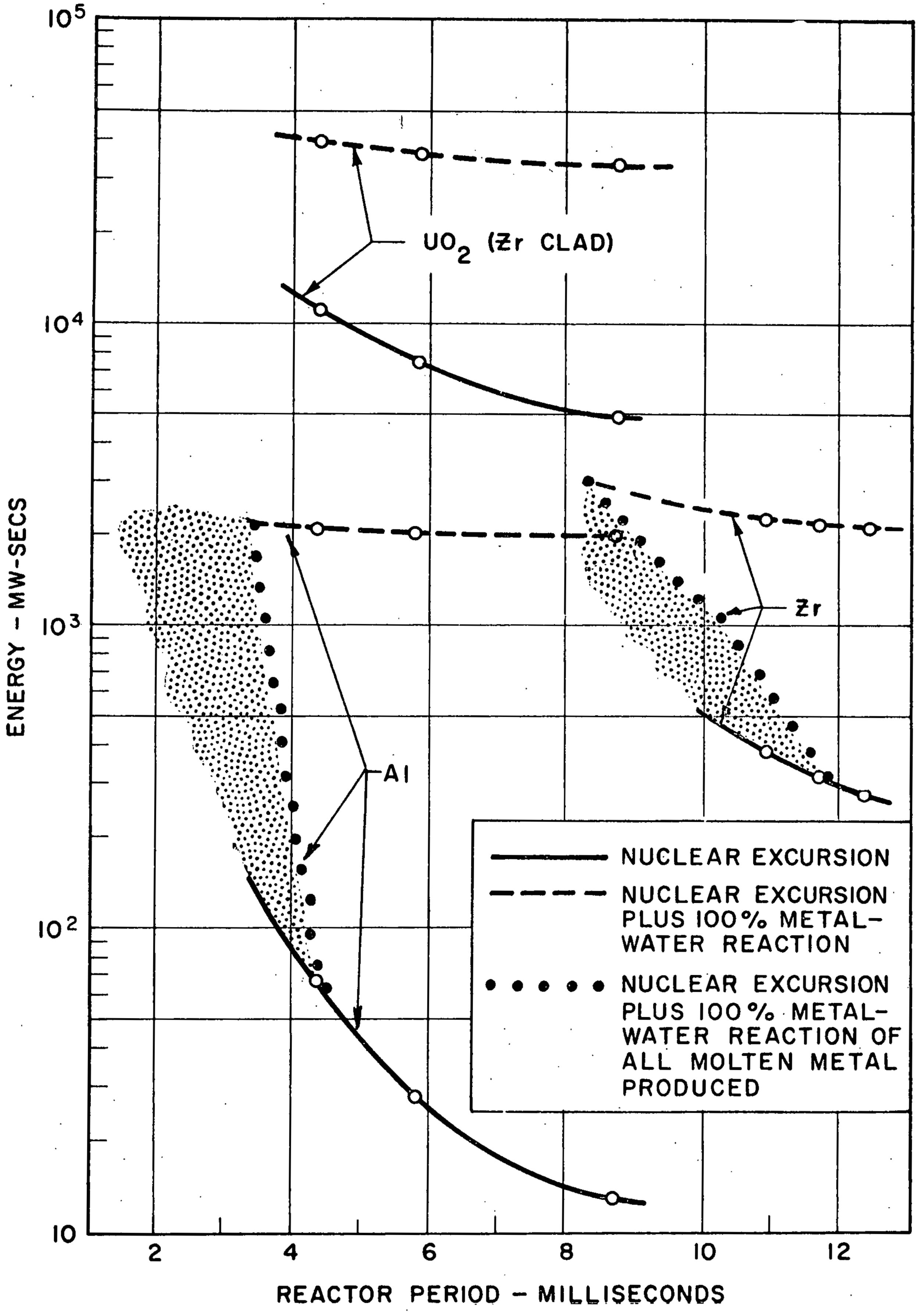
COMPARISON OF POWER AND THE RATIO OF HOT SPOT TO MELTING TEMPERATURE FOR THE TWO FLAT PLATE ELEMENTS
– NOTE DIFFERENCE IN INITIAL PERIOD

FIGURE 4



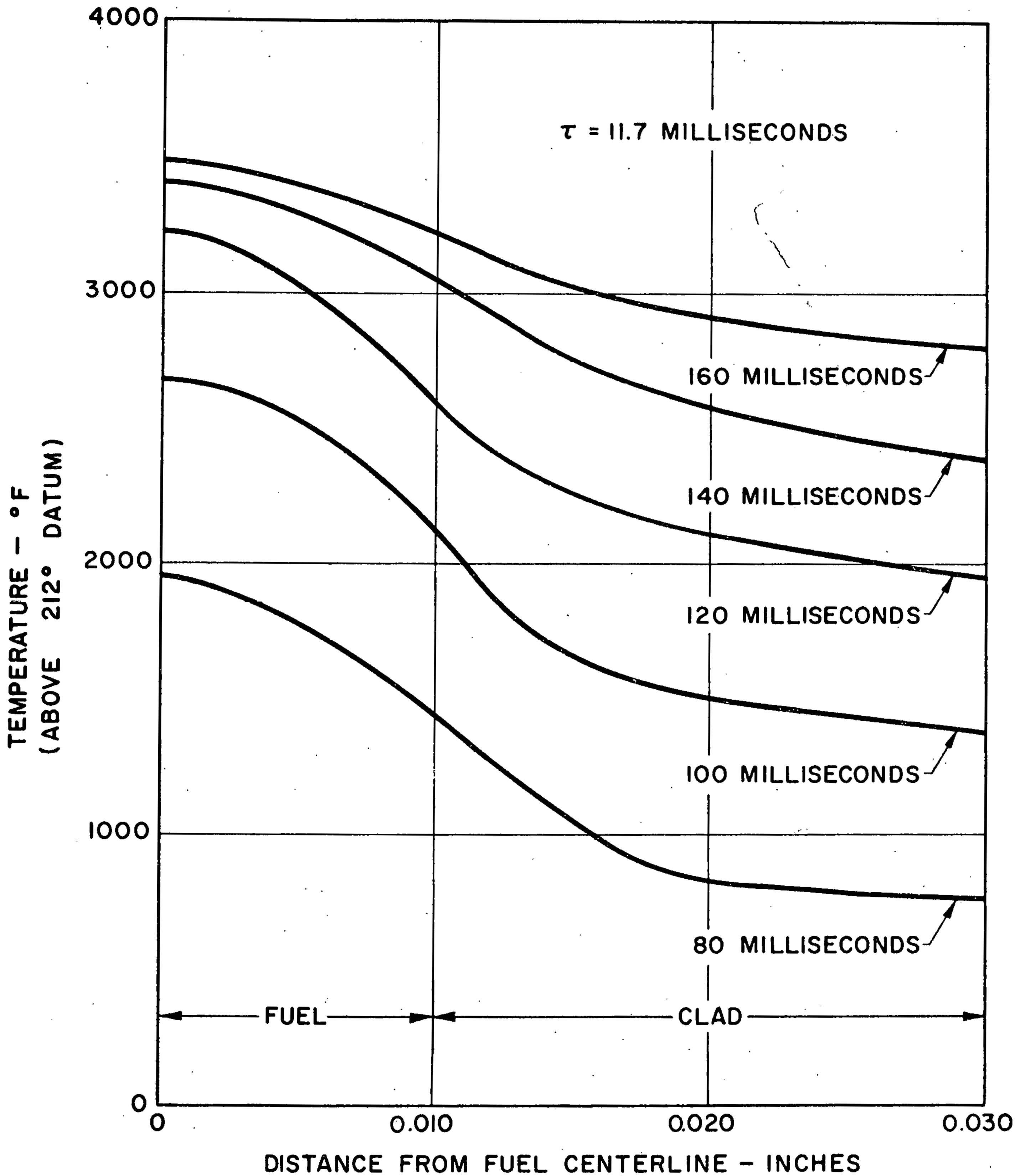
COMPARISON OF HOT SPOT TEMPERATURE AND PER CENT MOLTEN METAL FOR THE THREE REACTORS VS REACTOR PERIOD

FIGURE 5



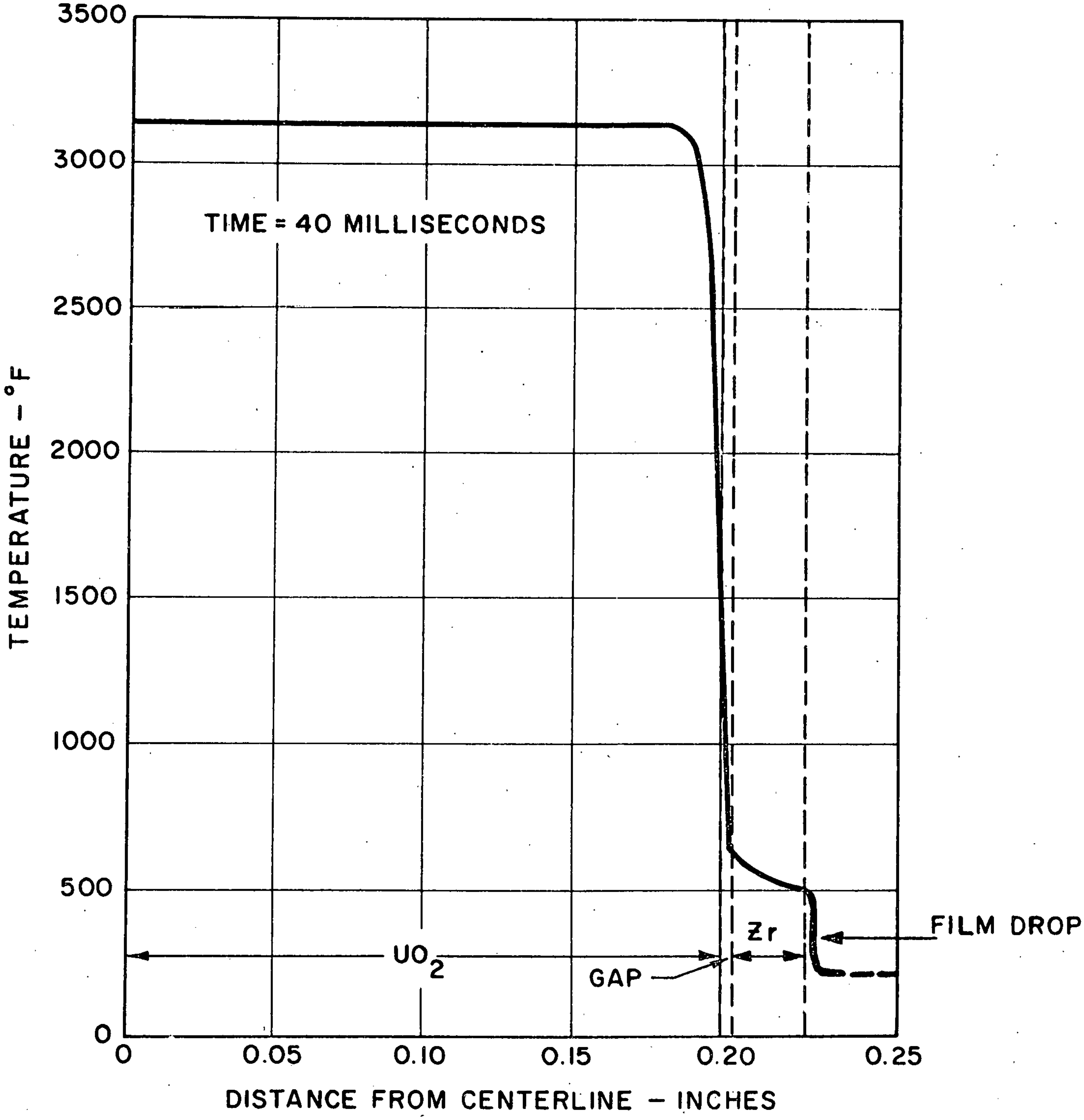
COMPARISON OF ENERGY RELEASED IN AN EXCURSION FOR THE THREE REACTORS VS REACTOR PERIOD

FIGURE 6



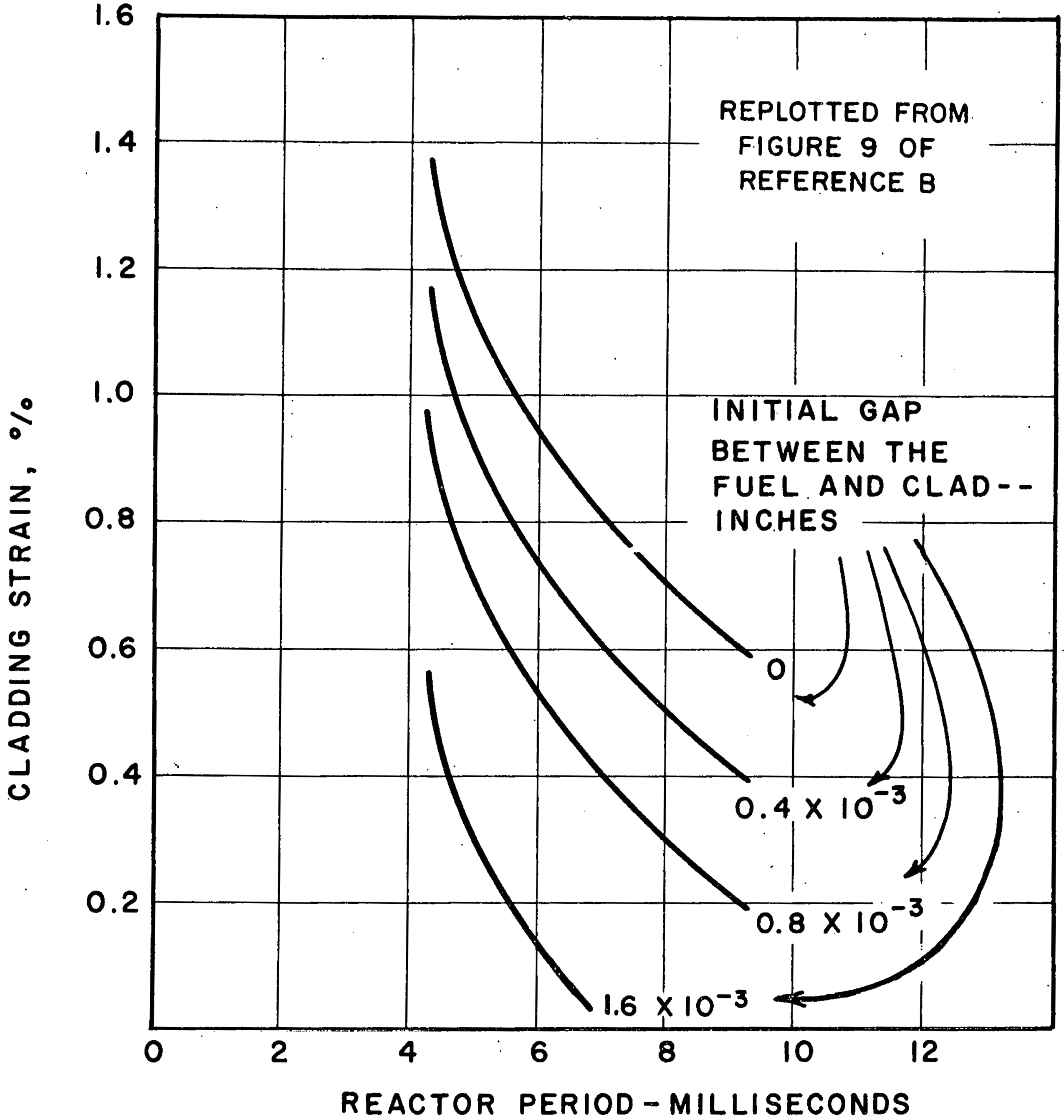
TEMPERATURE VS DISTANCE FROM FUEL ELEMENT CENTER
Zr FLAT PLATE ELEMENT

FIGURE 7



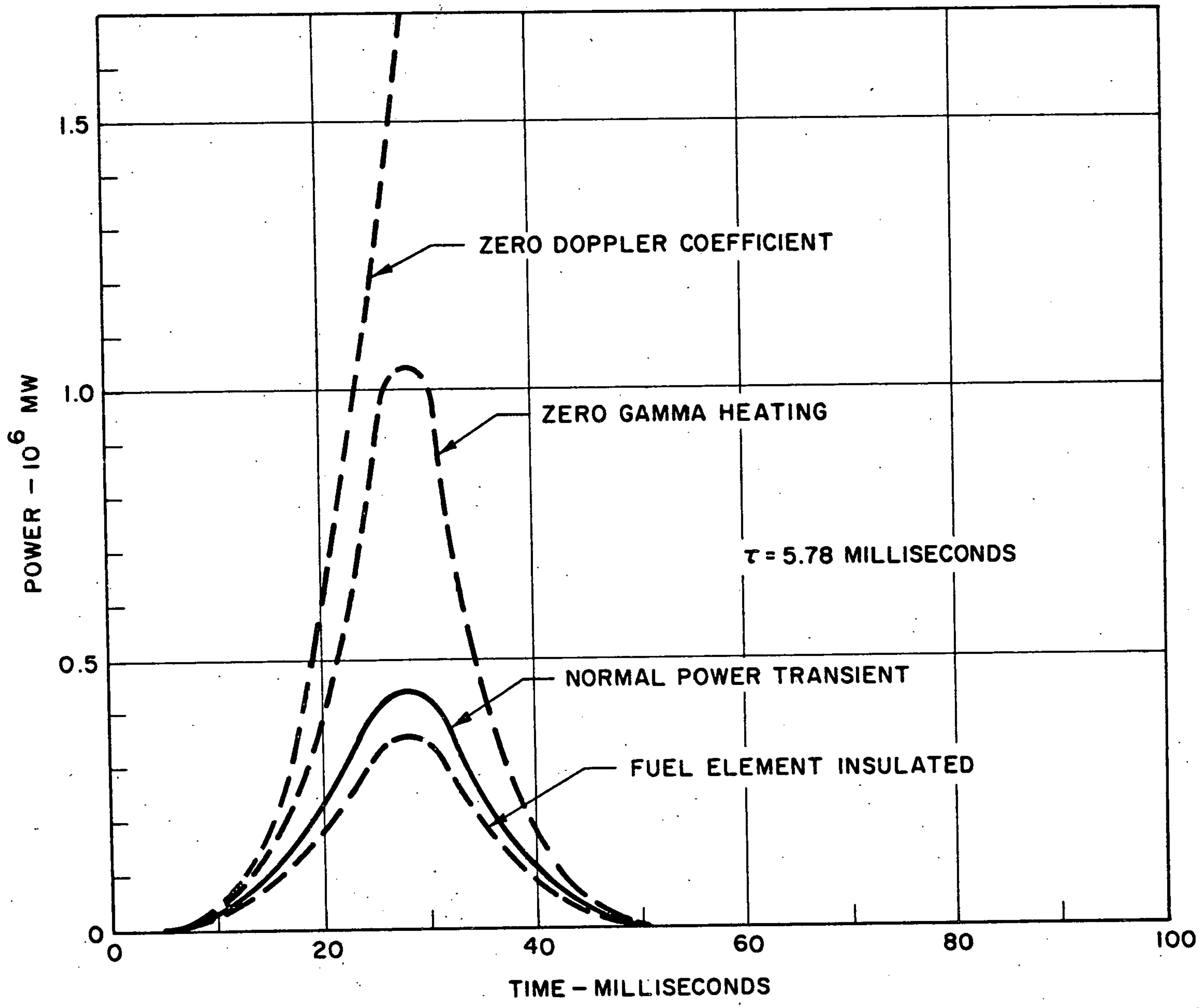
TEMPERATURE VS DISTANCE FROM FUEL ELEMENT CENTER FOR A REPRESENTATIVE TIME AFTER NUCLEAR EXCURSION: $\tau = 4.34$ MS CYLINDRICAL UO_2 Zr CLAD ELEMENT

FIGURE 8



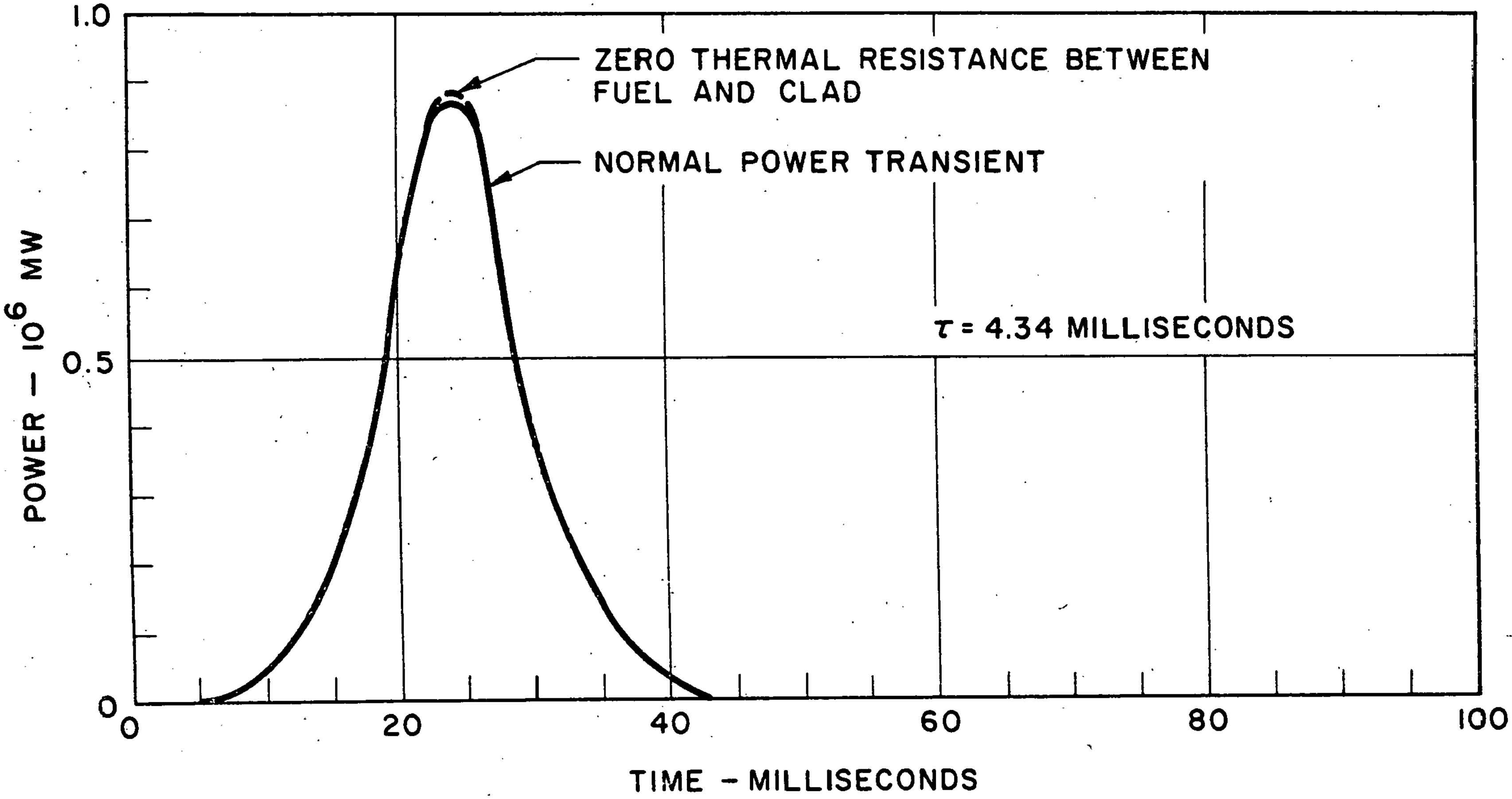
CLADDING STRAIN AS A RESULT OF TEMPERATURE INCREASE IN THE UO_2 Zr CLAD FUEL ELEMENT

FIGURE 9



POWER EXCURSIONS FOR THE UO_2 Zr CLAD FUEL ELEMENT SHOWING RELATIVE IMPORTANCE OF VARIOUS SHUTDOWN MECHANISMS

FIGURE 10



POWER EXCURSIONS FOR THE UO_2 Zr CLAD FUEL ELEMENT
SHOWING THE EFFECT OF FUEL CLAD THERMAL RESISTANCE

FIGURE 11

APPENDIX

NUMERICAL DATA USED IN THE EXCURSION CALCULATIONS

This appendix summarizes the numerical data that were used in performing the calculations that are given in the text of the report. In addition, the fuel models chosen are described. The details of the analysis follow the procedure outlined in Reference A. Basically, the calculational method involves a definition of the shutdown mechanisms and a mathematical description of these conditions for the reactor under consideration. These equations were then solved by means of an analog computer.

The subjects presented are:

1. Nomenclature
2. Shutdown Mechanisms
3. Constants for the Al-U Alloy Fueled Reactor
4. Constants for the Zr-U Alloy Fueled Reactor
5. Constants for the UO_2-Zr Clad Reactor

1. Nomenclature

C_p	-	specific heat	BTU/Lb.-^oF
h_{gap}	-	heat transfer coefficient from fuel meat to clad	BTU/Hr.-Ft^2-^oF
h_w	-	heat transfer coefficient from fuel element to water	BTU/Hr.-Ft.^{2}-^oF
$\Delta K/K$	-	reactivity	Dollars
k	-	thermal conductivity	BTU/Hr.-Ft-^oF
q	-	power	BTU/Hr.
T	-	temperature above the temperature at t = 0	^oF
t	-	time	Seconds
T_s	-	temperature of fuel surface	^oF
ρ	-	density	Lbs./In.^{3}
p	-	resonance escape probability	—
σ	-	U^{238} effective resonance integral	Barns

2. Shutdown Mechanisms

The shutdown mechanisms considered in this study were, for the Al and Zr alloy highly enriched reactors, the formation of steam due to conduction from the fuel. Thermal capacities and conductivities used are summarized in Table 1 in the report. It was assumed that the reactivity associated with void formation followed the relation $\Delta K/K = 0.1415\ (T_s)^{1/4}$ as developed from Appendix B of Reference A. The constant is derived from the geometry, thermodynamic state of the fluid, pressure gradient and surface and saturation conditions. In studying the UO_2 rod type fuel, two additional shutdown mechanisms were significant: heating of the water by absorbed radiation and the Doppler temperature coefficient. From the core composition and nuclear properties, it was calculated that 3% of the power was absorbed directly in the water producing nearly instantaneous heating. Secondly, since this fuel is only slightly enriched, there is a significant amount of U^{238} present. The U^{238} Doppler temperature coefficient of reactivity may be calculated using the relationship

$$(1/K)(dK/dT) = \left[(1-p)/p\right] \quad (1/\sigma)(d\sigma/dT)$$

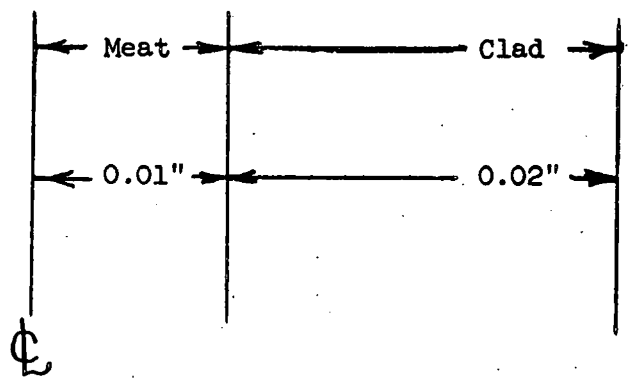
The resonance escape probability p is an exponential function of σ and the effective slowing-down cross section in the resonance region. In turn, σ depends on the fuel surface-to-volume ratio and on the fuel clumping. A value $p = 0.8$ was used for calculating the Doppler temperature coefficient, based on an 0.4 inch diameter UO_2 fuel rod surrounded by cold water with a 1.6 to 1 water-to-fuel volume ratio and no rod interaction.

2. Shutdown Mechanisms (Continued)

The term $(1/\sigma)(d\sigma/dT)$ is the temperature coefficient of the resonance integral. Its value depends upon the fuel surface-to-volume ratio and the temperature, the coefficient increasing with larger surface-to-volume ratio and decreasing at higher temperatures. One of the most recent measurements of the resonance integral of UO_2 as a function of fuel geometry and of temperature over a wide temperature range, 70 to 1830^{o}F, was made by Blomberg, Hellstrand, and Horner and reported in a Geneva II Conference paper (Reference C). For an 0.4 inch diameter UO_2 single rod, this data yields a resonance integral temperature coefficient of $7.5 \times 10^{-5}/^{o}F$ averaged over the 70 to 1830^{o}F fuel temperature range. This represents the low temperature end of a power excursion. An estimate of the temperature coefficient of the resonance integral at the higher fuel temperatures during an excursion was obtained from Part 2 of the Geneva Paper. For example, at 4000^{o}F fuel temperature, the coefficient is $6.7 \times 10^{-5}/^{o}F$. The corresponding Doppler coefficient calculated for $p = 0.8$ is then $1.67 \times 10^{-5}/^{o}F$. This is the figure used in the body of this report and probably represents a minimum reasonable value.

3. Constants Used For The Al-U Alloy Fueled Reactor

The core considered is composed of 36 fuel assemblies with 18 plates per assembly. The plate geometry is shown by the following sketch:



The plates are 24.625 inches long and 2.845 inches wide.

The fuel properties used were:

$$\rho = 0.0975 \text{ Lbs./In.}^3$$

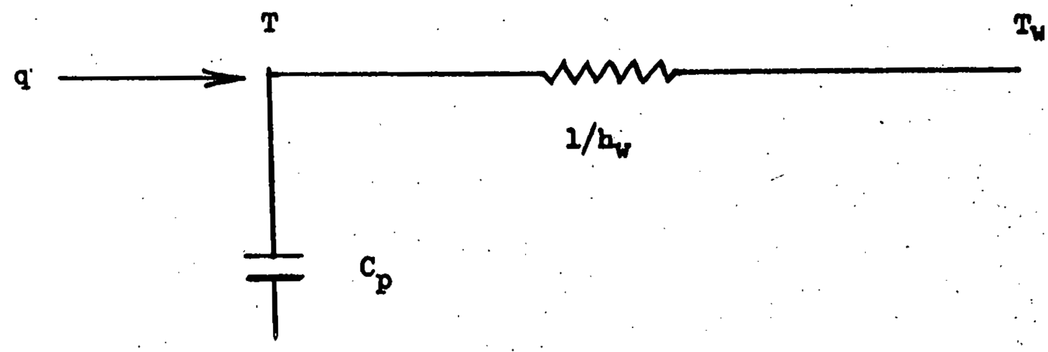
$$C_p = 0.183 \text{ BTU/Lb.-}^\circ\text{F}$$

$$k = 100 \text{ BTU/Hr.-Ft.-}^\circ\text{F}$$

$$h_w = 1000 \text{ BTU/Hr.-Ft.}^2\text{-}^\circ\text{F}$$

3. <u>Constants Used For The Al-U Alloy Fueled Reactor (Continued)</u>

A lumped fuel model was used due to the high conductivity of the material. This was



The equation solved was

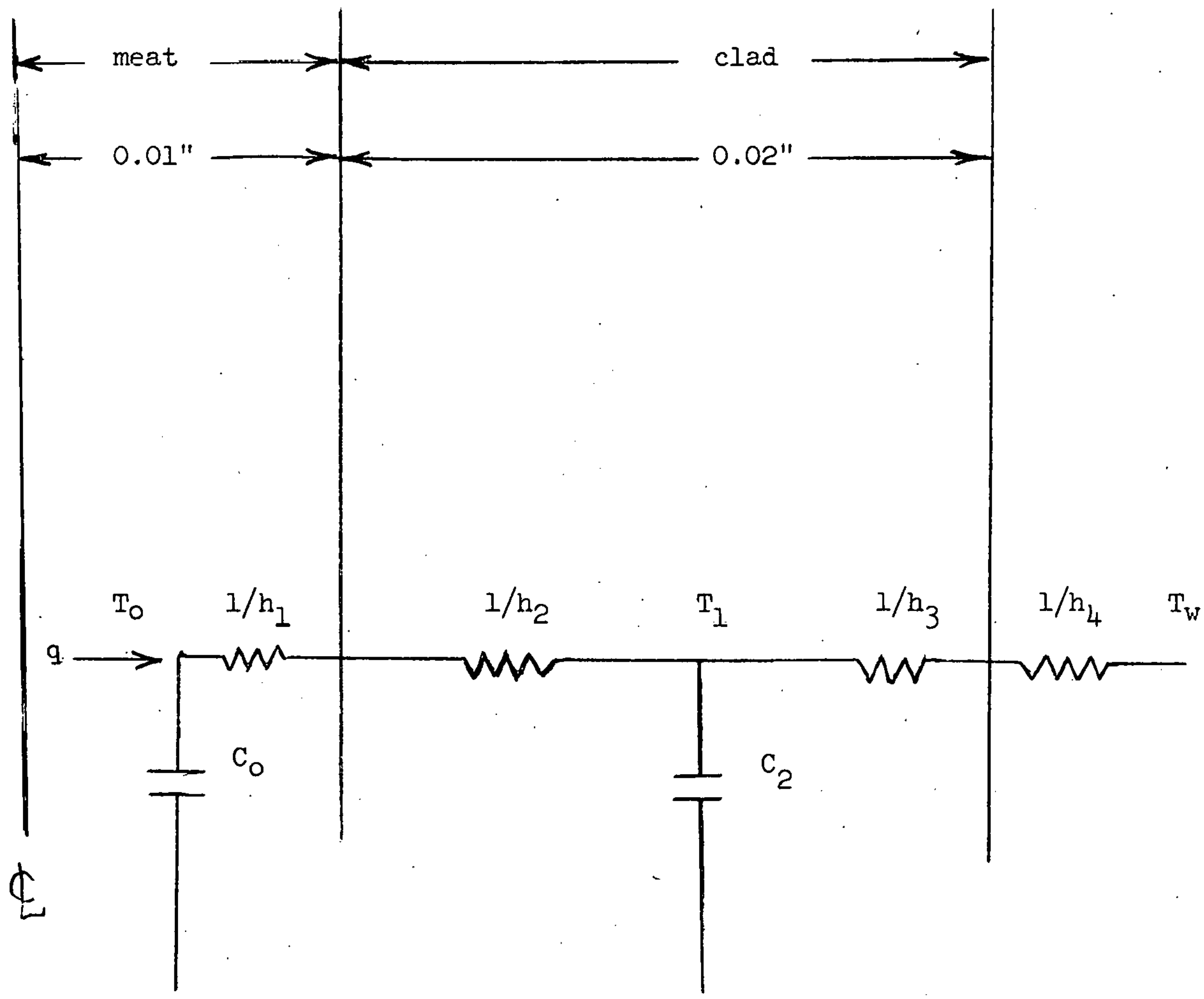
$$T = 1/C_p \int \left[q - (T - T_w)h_w \right] dt$$

4. Constants Used For The Zr-U Alloy Fueled Reactor

The fuel geometry was unchanged from the previous model; however, the physical properties for these materials are quite different. They are:

$$\rho = 0.237 \text{ Lbs./In.}^3$$
$$C_p = 0.08 \text{ BTU/Lb.-}^\circ\text{F}$$
$$k_{meat} = 8 \text{ BTU/Hr.-Ft.-}^\circ\text{F}$$
$$k_{clad} = 10.8 \text{ BTU/Hr.-Ft.-}^\circ\text{F}$$
$$h_w = 1000 \text{ BTU/Hr.-Ft.}^2\text{-}^\circ\text{F}$$

Due to the relatively poor conductivity of this fuel, it was considered advisable to develop a nodal fuel model that would more accurately predict the transient heat transfer and would allow calculation of the temperature distribution within the fuel plate. The model used was;



4. Constants Used For The Zr-U Alloy Fueled Reactor (Continued)

The equations solved were:

$$T_o = (1/C_o) \int \left[q - (T_o - T_1)h_1' \right] dt$$

$$T_1 = (1/C_2) \int \left[(T_o - T_1)h_1' - (T_1 - T_w)h_3' \right] dt$$

where $$\frac{1}{h_1'} = \frac{1}{h_1} + \frac{1}{h_2}$$

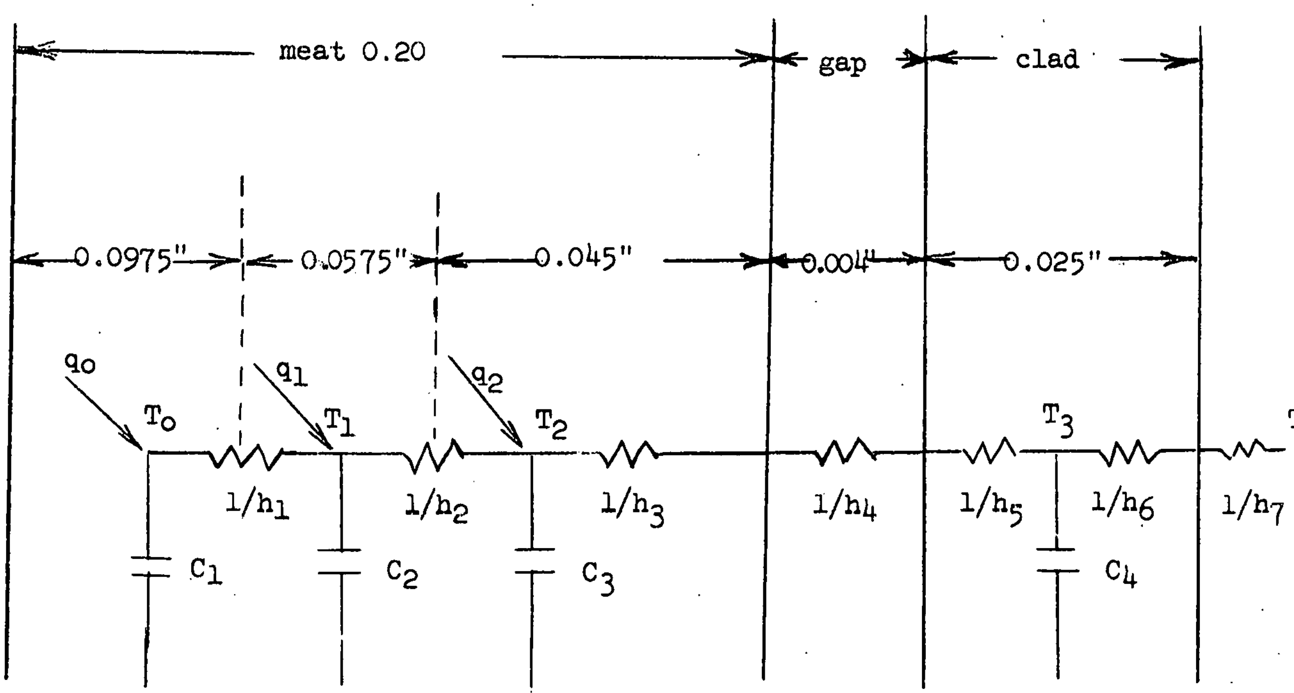
$$\frac{1}{h_3'} = \frac{1}{h_3} + \frac{1}{h_4}$$

5. Constants Used With The UO_2-Zr Clad Reactor

The fuel properties for this reactor are:

ρ_{UO_2}	=	649 Lbs./Ft.^{3}
ρ_{clad}	=	410 Lbs./Ft.^{3}
C_p of UO_2	=	0.07 BTU/Lb.-$^\circ$F
C_p of Zr	=	0.08 BTU/Lb.-$^\circ$F
h_w	=	1000 BTU/Hr.-Ft.^{2}-$^\circ$F
h_{gap}	=	1000 BTU/Hr.-Ft.^{2}-$^\circ$F
k_{UO_2}	=	1.15 BTU/Hr.-Ft.-$^\circ$F
k_{Zr}	=	10.85 BTU/Hr.-Ft.-$^\circ$F

The fuel geometry chosen was a UO_2 rod 0.40 inches in diameter with a 4 mil gap between it and the 25 mil thick clad. The fuel model used was:



5. Constants Used With the UO_2-Zr Clad Reactor (Continued)

The transient equations to be solved are:

$$T_o = (1/C_1) \int \left[(q_o - (T_o - T_1) h_1 \right] dt$$

$$T_1 = (1/C_2) \int \left[(q_1 + (T_o - T_1) h_1 - (T_1 - T_2) h_2 \right] dt$$

$$T_2 = (1/C_3) \int \left[(q_2 + (T_1 - T_2) h_2 - (T_2 - T_3) h'_3 \right] dt$$

$$T_3 = (1/C_4) \int \left[(T_2 - T_3) h'_3 - (T_3 - T_w) h'_6 \right] dt$$

where

$$1/h'_3 = (1/h_3) + (1/h_4) + (1/h_5)$$

$$1/h'_6 = (1/h_6) + (1/h_7)$$