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Safety and Cost Analysis of a 1000 MWe Sodium Cooled Fast Power Reactor

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GESELLSCHAFT FUR KERNFORSCHUNG M. B. H.
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This paper has been prepared within the framework of the association EURATOM - Gesellschaft für Kernforschung mbh. in the field of fast breeder development.

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Gesellschaft für Kernforschung mbH, Karlsruhe

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At the end of 1964 the Karlsruhe group has completed a design study of a 1000 MWe sodium-cooled fast breeder reactor /¯1_7. Now on this a detailed safety and cost analysis has been made. This allows for a good judgement of the safety of large fast reactors in general and for defining the criteria of a second design of a large plant and of the smaller prototype reactor as well. A number of interesting conclusions can be drawn.

1. Main Features of the Present 1000 MWe - Design

Table 1 shows some of the important design parameters of the core. The main features are plain cylindrical shape, oxide fuel of 87 $^{o}/o$ theoretical density, no moderator like BeO and a moderate flattening with H/D = 1/3. This results in a relatively large negative Doppler-coefficient and an internal breeding ratio close to one.
The void coefficient then will be somewhat larger than in several other designs. Its effect on overall safety will be one of the topics of this paper.

The coolant volume fraction is with 50 $^{o}/o$ relatively large, this results in a small pressure drop and pump size.Special spacers with a small amount of structural materials can be used.
The decision not to use BeO or similar moderators, is purely economical. The loss in total breeding gain is evident, whereas the small internal breeding ratio contributes some additional operational difficulties.

The core has 2 zones of equal volume and different enrichment. The 229 subassemblies are of hexagonal shape, the inner radial blanket is of oxide, the outer one of metal. The axial blanket is 40 cm on either side. A fission gas plenum of 80 cm is below the core. In another paper during this conference /¯5_7 we shall give an evaluation of the vented fuel concept.

The characteristics of the plant design will be discussed in detail in our panel paper. So we only mention some important items:

The primary circuit is of the loop type with some features of the pool type (mainly rigid connection ducts between reactor vessel and heat exchangers, pumps and heat exchangers are sliding).

There are two intermediate heat exchangers and four pumps, and four secondary loops. A new type of refueling system has been proposed. The capital costs, based on studies from industry, have been calculated to be 115 $/kW.

2. Influence of Group Constants

The effect of the used cross section set on the calculated breeding ratio, Doppler-effect and void-coefficient defines the reliability of any safety and cost analysis. The results of a world-wide comparison and their interpretation have been presented by Dr.Okrent this morning. We also have calculated our specific reactor with three sets of cross sections:

a) The russian 26-group set ABN

b) A 60-group set

c) The new Karlsruhe 26-group set (of J.J.Schmidt and coworkers)

Table 2 shows the results.

3. Dynamics and Safety of the Reference Reactor

3.1 Dynamic Programs

The dynamic behaviour of our proposed reactor has been studied with the use of 3 programs. These are:

a) An analog program of the core.
It included for the simulation:

1) Space independent neutron kinetics

2) Fuel heat generation and transfer through fuel and can and heat removal by the coolant

3) Reactivity feedback caused by Doppler-effect, structural material and coolant.

The fuel element was divided in 30 segments. For each segment the heat balance equation was set up, density, specific heat and thermal conductivity was considered to be constant.

Fuel melting and the temperature dependence of Doppler-coefficient and a temperature dependent heat transfer coefficient between fuel and can was included.

b) An analog program of the total primary circuit. It included:

1) Similar to the first program, neutron kinetics,heat generation, transfer and removal and the reactivity feedback (15 segments division of the fuel element)

2) The heat exchange in the intermediate-heat-exchanger (16 segments division)

3) The flow coast-down in the main coolant pumps

4) The time delay of the pipes between reactor and heat exchanger

5) The mixing process in the reactor inlet and outlet plenum.

c) A revised version of the digital code FORE, developed by GE. The Karlsruhe version included in addition the temperature dependence of the heat transfer coefficient between fuel and can.

3.2 General Safety Criteria

It is impossible to define one general safety criterium for a fast reactor. At a former occasion [2] we have already pointed out, that one has to study each possible actual accident or incident more or less separately and evaluate the consequences. Since there is always some uncertainty whether one knows each possible actual cause of an accident it is nevertheless helpful to study the general behaviour of a system under enforced conditions, such as fast reactivity ramps of infinite height, even if the results are not directly applicable to a real case.

So the first general safety criterium is the time t_m between the beginning of an infinite reactivity ramp input and the starting of melting of the hottest fuel. It gives a scale for the possibility of any counteractions by the scram system.

Since any reactivity input above 0.25 $ will lead to fuel melting and under certain conditions after sufficient time to destruction of the reactor, the reaction of the safety system should be included into the analysis. So our second safety criterium gives the maximum rate of the reactivity ramp, which can be counteracted by the safety system without starting fuel-melting. We choose a conventional, spring-driven system with 10 safety rods.

Fig. 1 shows the space-time relationship including the specific effects of inertia and friction. Naturally this criterium is less general, since it depends on a specific safety system, but it is also more practical.

Also the delay time τ between the onset of the excursion and the beginning of rod movement has to be taken into account.

A third criterium should be the maximum hypothetical accident (MHA) and its consequences. But this does not depend on the core design only but also on the containment and shielding properties and is even less general. So the MHA will not be discussed in this chapter. Some remarks will be made in chapter 6.

3.3 Results with Safety Criterium 1, Time t_m to Reach Start of Fuel Melting

In table 3 the input data of the dynamic calculation of the reference reactor are given. In fig. 2 t_m is plotted as a function of the

Doppler coefficient and the structural expansion coefficient. The influence of the Doppler effect is distinct, but not too important. The structural coefficient acts only for slow ramp rates.

In this connection it is interesting to compare t_m with t_{bc}, the time to reach coolant boiling (at channel exit) and t_{bf}, the time to reach fuel boiling. t_{bf} may be an equivalent for the core decomposition and termination of the excursion.

In fig. 3 the three time intervals are plotted as a function of the ramp rate. For large ramp rates always $t_{bc} > t_{bf} > t_m$. This is equivalent to the fact that it is impossible in a highly rated oxide core to get coolant boiling before reactor decomposition. The results of Okrent, Cohen and Loewenstein indicate the same [3]. Only for small r_s is $t_{bc} < t_{bf}$. Then coolant boiling may happen. In connection with safety criterium 2 we shall prove that these slow ramp disturbances can easily be governed by the safety system.

Fig. 4 shows the effect of finite ramps. t_m is plotted as a function of the ramp height for different ramp rates. For $t_m \longrightarrow \infty$ all curves converge to the same value of r_h = 0.25 \$, which is defined by static conditions. For this reactivity input the reactor does not need any safety system.

3.4 Safety Criterium 2, Maximum Ramp Rate

Fig. 6 shows the maximum ramp rate, which can be counteracted without fuel melting by the safety system as a function of the delay time τ before beginning of rod movement for different Doppler coefficients. The safety system consists of 10 scram rods, each with a 250 $\frac{kp}{cm}$ spring and a weight of 100 kg. The total reactivity value of the rods is 15 \$.

The space-time dependence of this system has already been given in fig. 1.

In fig. 7 is plotted the maximum ramp rate which can be controlled by the above safety system without fuel melting for a delay time of 30 msec as a function of the Doppler coefficient. If we also transfer some of the information of fig. 3 to here, we get the dotted line. It resembles the ramp rate, for which $t_{bc} = t_{bf}$. For ramp rates above this dotted curve the reactor disassembles before the coolant can boil. The ramp rates for coolant boiling are below the ramp rates, which can be governed by the safety system.

With respect to coolant boiling and void effect, therefore, it is an important fact, that it is impossible to generate sodium boiling by any ramp reactivity input as long, as the normal, conventional safety system is working. It is much easier to destroy the reactor by fuel boiling.

Moreover, in principle the safety action can be accelerated by an additional electromagnetic force according to a proposal of Dosch [4].

Necessarily the values of fuel conductivity, melting temperature, and gap conductivity after some irradiation are not too well known. To resemble the influence of these uncertainties we have calculated the allowable ramp rate as a function of τ for different values of the assured fuel melting temperature. This is shown in fig. 5.

3.5 Actual Accidents

So far we have studied the effect of hypothetical ramp excursions. The actual ramps, i.e. control rod runaways, will definitely be kept in the order of 0.01 \$/sec and below.

Another actual possibility is the loading accident. The central fuel element with 0.7 \$ is dropped into the cold critical reactor in 500 msec. The effect of the resulting perturbation is shown in fig. 8. All temperatures can be kept below dangerous values.

In fig. 9 can be seen the result of a simultaneous failure of all 4 primary pumps. The canning temperature will reach the boiling point of sodium of about 960°C after 90 sec.

As is shown in fig. 10, the structural expansion coefficient will reduce the reactor power to about 75 °/o during this time. The effect of the uncertainty in this coefficient can also be seen from this figure.

3.6 Stability

It can be expected that a reactor system with a fast negative and a slow positive coefficient will be stable in most cases. This has been proved by a detailed analysis. According to the Nyquist-diagram instability may occur if the Doppler coefficient is smaller than 1/50 of its actual value or if the coolant coefficient is 50 times its actual value.

It has been shown that instability may arise also from higher order delay terms. But there is no reason to assume those.

4. Parametric Studies, Influence on Cost and Safety

4.1 Survey of Variations

It has been the central point of our analysis to define the influence of the important design parameters on cost and safety and by this to learn, where our conception has to be changed. So we made systematically small displacements on many values and calculated the effect on safety and costs. Table 4 gives a review of the parametric variations and the expected advantages and disadvantages.

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The difference to a number of other parametric surveys is that all displacements are executed on a real system. Therefore many detailed effects are included which are nevertheless of an extraordinary importance. Example: If the coolant fraction α is lowered (case 1), the pressure drop rises and, therefore, more structural material is needed. Moreover, other types of spacers may be needed for geometrical reasons.

Also, for a real system with given heat exchangers, temperatures, pressures etc. the effect of parametric variations on fuel and capital costs will be more realistic.

For the investigations the following programs were used:

1) One-dimensional multigroup-diffusion program "MGP" [12]

2) Two-dimensional diffusion program "Twenty Grand" [13]

3) Nuclear program system Karlsruhe "NUSYS" [not published]

4) Two-dimensional perturbation Code "2 D-Pert" [14]

The enrichment of the first and second core zone was determinated by one-dimensional diffusions-calculations ("MGP") with the 26 KFK-group set [15] . From these results were generated the macroscopic 6 group-cross sections by "NUSYS". Then the breeding ratios, critical masses, reactivity-coefficients and power distributions were calculated by the "Twenty Grand" and the two-dimensional perturbation code "2 D-Pert". For the calculation of the Doppler-coefficient the one-dimensional perturbation code contained in "NUSYS" was used.

The determination of the core region in which removal of the sodium caused the maximum $\Delta K_{sod\text{-}loss}$-effect, was performed by two - dimensional perturbation calculations. The $\Delta K_{sod\text{-}loss}$-effect itself was determinated by one and two-dimensional diffusion calculations.

4.2 Discussion of Nuclear Results

In table 5 the main results are given.
In fig. 11a the most important results of the nuclear calculation are also represented graphically.

Case 1:

The reduction of the coolant fraction α from 50 to 40 $^{o}/o$ should result in a larger internal breeding ratio. However, the coolant pressure drop is raised to 13.7 atm and the larger requirement in structural material cancels the gain in the internal and total breeding ratios.

Case 1 a:

If $\Delta\vartheta$ is raised to 200^{o}C and α changed in such a way, that the pressure drop is the same as for the reference reactor, the internal and total BR are somewhat higher. The Doppler coefficient stays practically unchanged during these variations, whereas the void effect is somewhat reduced for the smaller values of α.

The larger $\Delta\vartheta$ also allows for a smaller effective temperature difference between the primary and secondary sodium circuit and reduces the danger of thermal shocks in the case of a break in the intermediate heat exchanger (IHE).
On the other side the IHE must be larger or the thermal plant efficiency will be reduced. This will be discussed in our cost considerations.

Case 2:

Smaller H/D must be paid for in the internal breeding ratio and Doppler coefficient. The reduction of the void effect is favorable compared with the reference reactor, but does not pay in comparison with the versions 1 and 1 a.

Case 3:

By addition of 5 °/o BeO the Doppler is nearly doubled, whereas the void effect is reduced by about 1 $. The penalty is purely economical. The advantages are not too important for the operational stability and safety, but for the consequences of the maximum hypothetical accident.

Case 4:

The addition of molybdenum for a better fuel conductivity is plainly disadvantageous. The BR is low, the Doppler is low and the void effect is high.

Case 5 a:

The first version of the carbide core has the very large internal and total breeding ratios, a Doppler coefficient nearly as large as the oxide cores, but a tremendous void effect. The advantage of the low fuel temperature will be discussed in connection with the dynamic behaviour.

Case 5 b:

Here the breeding is strongly reduced, especially internally. The Doppler effect is smaller, but this is true also for the void effect.

Case 6:

The main advantage of the vented fuel concept is the low amount of canning material with the subsequent gain in breeding. It might still be possible that strong gas pressures develop during certain transients and that the canning has to withstand them. This might change the evaluation of the vented fuel.

4.3 Discussion of Economical Results

In table 6 and fig. 11 b the results of the comparative cost calculation are given. All calculations are based on the same assumptions as for the reference reactor [1] .

The main changes of capital costs result from the calculated pressure drop and the corresponding pump size. A relatively moderate power law with an exponent 0.6 has been assumed for the cost dependence on pumping power. Other main cost variations come from the fuel and the breeding ratio. These values are capitalized over 15 years for a load factor of 0.8.

Case 1:

Additional costs mainly from pumping power.

Case 1 a:

Cost savings mainly because of better BR.

Case 2:

Savings in pump size are more than compensated by fuel costs (large number of fuel rods).

Case 3:

Somewhat larger capital costs (pressure drop) and much larger fuel costs (breeding). Additional costs of nearly 5 °/o of the total plant costs.

Case 4:

Additional costs mainly by bad breeding.

Case 5 a:

Considerable cost savings by favourable breeding ratio.

Case 5 b:

Even with some additional capital costs for pumping power and less BR than 5a the total net savings are largest (5 °/o of total plant). This results mainly from lower fuel costs because much less fuel is needed.

Case 6:

Savings in pumping power and BR are partially compensated by gas purification plant (see /¯5_7). Savings in the order of 2 °/o of total costs.

4.4 Discussion of Dynamical Results

Considerable differences in the dynamic behaviour compared to the reference reactor are found only, if other fuels are used. As can be expected moderate changes in coolant temperature and geometry are of minor importance.

In fig. 12 t_m is shown for the cases 0, 4, 5a, 5b as a function of ramp rate. The two low temperature cores with molybdenum and carbide are somewhat above the values for the reference core and for the highly rated carbide.

Fig. 13 shows the effect of ramp height. The low temperature cores naturally can withstand considerable reactivity inputs without any action of the safety system.

In fig. 14 is plotted the maximum allowable ramp rate as a function of the delay time of the safety system. The low temperature cores allow for about 1.5 times faster ramps.

5. The Importance of Coolant Boiling

Partial or complete voiding of the total core may have two causes:

a) Loss of coolant flow
b) A slow excursion.

Both occurences are improbable to a very high degree.
As has been shown, a) requires the simultaneous failure of all primary pumps, of all emergency drives despite of a time reserve of about 90 sec, and of the safety system, (a break of the double wall primary circuit is excluded).
b) requires the simultaneous runaway of several control rods and the failure of the safety system or the runaway of a control rod and the simultaneous failure of all independent safety rods.

It should really be discussed whether a maximum accident beyond this is credible.

But if we assume credibility, then one of these accidents would certainly destroy the core. Ramps of rates up to 100 ∅/sec may be generated by coolant evaporation depending on the initial conditions. The total energy of the excursion is strongly dependent on the Doppler coefficient.

It is only for this type of hypothetical accident, that the void coefficient is of real importance.

The dynamic studies have shown, that under other conditions boiling in larger parts of the core ($\Delta k > 1$ ∅) cannot occur. (Boiling in a single subassembly will be discussed later.)

However, in the light of these considerations two remarks have to be made:

First Remark

The boiling starts at the upper end of the core channels. In this region the void coefficient is negative. Therefore it has been said that an overall positive void coefficient might not be too dangerous under these conditions. This opinion is too optimistic. Because of the large specific volume of sodium vapor a boiling channel will be blocked very easily by the large pressure drop of the two-phase mixture. For our reactor boiling begins at 28 °/o of normal coolant flow, while the channel is blocked already at 22 °/o.

If boiling starts at all, chances are very large that a complete flow blockage with subsequent complete voiding occurs.

Second Remark

Some designs aim at a $\Delta k \leq 0$ for total voiding, but a positive maximum Δk. If voiding occurs according to the above mentioned mechanism of channel blockage it will gradually spread from the core center and at some time the maximum Δk will be effective. Therefore, $\Delta k \leq 0$ for total voiding should not be mistaken for a criterium of a qualitative nature. It might be helpful in a quantitative way only in so far, as the maximum Δk will also be smaller compared to the case, where Δk for total voiding is positive.

While we consider these types of boiling events to be very improbable, we must look at another one as to be much more probable. This is the blockage of a single fuel subassembly for example by something in the sodium flow. The reactivity effect of this can be governed easily by the temperature coefficient (for the central subassembly it amounts to + 0.08 \$). But it might be disastrous if the sodium superheats to a larger degree.

Experimental results on sodium superheat have been reported by several authors [7, 8, 9]. The bulk superheat certainly depends on the heat flux and on the surface conditions. Superheating in the order of 200°C and even more might be possible. Taking as an example a maximum superheat of 200°C and taking into account the temperature distribution in the coolant under reduced flow conditions (21 °/o of normal flow) a superheating energy of about 70 kcal will be stored in the upper quarter of the subassembly.

If finally the superheated liquid is flashed, everything depends on the mechanism how the energy is released in space and time. We have calculated the effect of the worst case, i.e. an instantaneous release of the whole energy and its impact on the subassembly wall by using a one-dimensional model of Symonds and Mentel [10]. Then apparently the subassembly box will be badly deformed and the neighbouring subassemblies will be affected. If the combination of flow reduction and superheat now occurs in the neighbour assemblies, the damage may spread over the core like a chain-reaction.

So we concentrate an important part of our effort on three subjects:

a) Theoretical and experimental studies of the dynamics of two phase flow and of the release of superheat in space and time. A first step in this direction will be reported by Fischer and Häfele [11] during this conference.

b) There have to be developed methods to keep small the superheat in a fast power-reactor.

c) The subassembly boxes must be designed in such a way that the superheating energy can be dispersed at a maximum rate.

6. Some Remarks on the Maximum Hypothetical Accident

The MHA is the fast excursion with the effect of the positive void coefficient.
There are two possibilities for the MHA:

a) The core is disassembled by the fastest possible reactivity input rate (first excursion).

b) After the first excursion the molten core material gathers somewhere in the reactor containment (second excursion).

The maximum reactivity input rate for the first excursion is caused by voiding of the core center. This again can be caused by a slow primary excursion or by a loss of coolant flow.

The other possibility to push a fast reactivity increase into the reactor is the core meltdown or at least a certain stage of fuel slumping.

The consequences of the second excursion may be influenced by the shape of the containment.

Both the first and second excursions are strongly influenced by the Doppler coefficient. In our panel paper of this conference we shall discuss this point in more detail. The released energy has to be contained in the concrete structure surrounding the reactor vessel.

7. Conclusions

7.1 According to our criteria 1 and 2 our proposed 1000 MWe reactor is very safe.

7.2 Infinite ramps of the order of 10 \$/sec can be counteracted by a spring-driven safety system, much more than will actually occur.

7.3 Coolant boiling in larger regions of the core cannot occur as long as the safety system is operating.

7.4 The positive void effect is of importance for the maximum hypothetical accident only. It occurs only, if several independent improbable conditions are fulfilled simultaneously.

7.5 Coolant boiling in a single subassembly may be dangerous in combination with sodium superheat.

7.6 Parametric studies show advantages in cost and safety in the direction of lower coolant fractions, larger $\Delta \vartheta$, and a not too compact carbide core. The addition of Mo to the fuel oxide proves to be very unfavourable.

7.7 Further studies on the MHA and on sodium superheat and sodium two phase flow are required.

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Table 1

Some Design Parameters of a 1000 MWe Sodium Cooled Fast Reactor

Reference Reactor

Core height	H : 95.5 cm	Internal breeding ratio	BR_{in}	: 0.89
Core diameter	D : 286 cm	Total breeding ratio	BR_{tot}	: 1.385
	H/D : 1/3	Mean specific power	q_{spec}	: 1.2 $\frac{MW}{kg}$
		Mean power density	q_m	: 0.599 $\frac{MW}{e}$
Number of core zones	: 2	ΔK_{Void} maximum		: 3.07 \$
Max. rod power	χ : 560 $\frac{W}{cm}$	ΔK_{Void} $total_{Core}$		: 1.04 \$
Rod diameter	d : 6.7 mm			
Coolant fraction	α : 50 %	Coolant inlet temperat.	ϑ_1	: 430°C
Structural fraction	β : 17.3 %	Coolant exit temperature	ϑ_2	: 580°C
Pressure drop (core and axial blanket)	Δp : 3.2 $\frac{kg}{cm^2}$	Maximum fuel center temp.	ϑ_{max}	: 2412°C

$$\frac{\text{Mean power}}{\text{max. power}} = \varphi = \varphi_{rad} \cdot \varphi_{ax} : 0.66$$

80 cm fission gas plenum at the bottom,

229 hexagonal subassemblies

45 cm radial blanket

40 cm axial blanket

Table 2

Calculation of a 1000 MWe-Breeder with 3 Cross-section Sets

	26 groups KFK	26 groups ABN *	60 groups
Critical Mass [kg]	2168	2048	2010
Internal Breeding Ratio	0.95	0.94	0.91
Doppler Coefficient	$-5.97 \cdot 10^{-6}$	$-8.32 \cdot 10^{-6}$	$-6.58 \cdot 10^{-6}$
Δk_{max} Sodium Loss	0.024	0.013	0.008

* Revised version of the reference reactor which has been recalculated for system analysis

Table 3

Basic Constants for Dynamic Calculations of Reference Reactor

Basic Constants	Reference Reactor
Thermal Conductivity: $\left[\frac{W}{cm\,^\circ C}\right]$	
Fuel (1450°C)	0.03
Can (600°C)	0.21
Specific Heat: $\left[\frac{Wsec}{g\,^\circ C}\right]$	
Fuel (1450°C)	0.343
Can (600°C)	0.503
Coolant (500°C)	1.264
Density: $\left[\frac{g}{cm^3}\right]$	
Fuel (1450°C)	9.9
Can (600°C)	8.0
Melting Temperature: $\left[^\circ C\right]$	
Fuel	2800
Heat Transfer Coefficient: $\left[\frac{W}{cm^2\,^\circ C}\right]$	
Fuel $\longrightarrow$ Can (1450°C)	0.75
Can $\longrightarrow$ Coolant (500°C)	14.5

Table 4

List of Parametric Variations

0) Original Reactor	Expected Advantages	Expected Disadvantages
1) Smaller coolant fraction α	Internal BR total BR	more Δ p more structural material
1a) Smaller α and larger $\Delta \vartheta$	Less structural material than 1a)	Larger heat exchange
2) Smaller H/D	Smaller Void effect	Smaller Doppler smaller int. BR capital costs?
3) 5 °/o BeO added	Larger Doppler	Smaller BR
4) 10 °/o Mo in fuel	Low fuel temperat.	Smaller Doppler smaller BR
5a) UC, same geometry as oxide	Low fuel temperat. large internal and total BR	Larger Void effect
5b) UC compact core	Compared to 5a): capital costs?	Smaller internal BR larger Δ p
6) Vented fuel	Larger BR less pumping power	Gas purification plant

Table 5

Characteristic Data of Different Reactors

Reactor Number	0	1	1a	2	3	4	5a	5b	6
Characteristic Data	Reference Reactor	Smaller Coolant Fraction α	Smaller Coolant Fraction α	Smaller H/D	5 °/o BeO added	10 °/o Mo in Fuel	UC-Fuel Same Geometry as Oxide	UC-Fuel Compact Core	Vented Fuel
Core Volume [m^3]	5.9	5.3	5.2	6.0	6.6	5.9	5.9	3.2	5.9
Core Height [m]	94	91	90.5	78	98	94	94	77	94
Height/Diameter: H/D	1/3	1/3	1/3	1/4	1/3	1/3	1/3	1/3	1/3
Pressure Drop [at]	4.6	13.7	4.6	2.8	5.1	4.6	4.6	12.3	3.3
Sodium v/o	50	40	44	50	50	50	50	50	50
Fuel v/o	32.2	34.4	35.0	33.0	27.2	29.3	32.2	28.8	36.8
Steel v/o	17.8	25.6	21.0	17.0	17.8	17.8	17.8	21.2	13.2
Coolant Temperat.Rise $\Delta\vartheta$ [°C]	150	150	200	150	150	150	150	150	150
Volume Ratio of UO_2/PuO_2 (UC/PuC) :									
1 Core Zone	8.2	8.1	8.4	7.9	7.4	7.3	9.7	8.2	9.3
2 Core Zone	6.0	6.0	6.1	5.8	5.3	5.2	7.3	5.6	6.7
Critical Mass [kg]	2168	2093	2046	2320	2295	2220	2370	1416	2245
Breeding Ratios:									
Core (int.)	0.95	0.95	0.97	0.91	0.89	0.84	1.12	0.91	1.04
Blanket	0.47	0.44	0.48	0.52	0.39	0.47	0.50	0.62	0.55
Total	1.42	1.39	1.45	1.43	1.28	1.31	1.62	1.53	1.59
Doppler Coefficient	$-5.97 \cdot 10^{-6}$	$-5.99 \cdot 10^{-6}$	$-5.99 \cdot 10^{-6}$	$-5.30 \cdot 10^{-6}$	$-1.05 \cdot 10^{-5}$	$-4.61 \cdot 10^{-6}$	$-5.76 \cdot 10^{-6}$	$-4.72 \cdot 10^{-6}$	$-6.08 \cdot 10^{-6}$
Max.Sodium-Loss Effect $\Delta K_{Sod-loss}$:	0.0244	0.0224	0.0221	0.0223	0.0201	0.0271	0.0291	0.0201	0.0263
Total Additional Cost [\$]$\cdot 10^6$	0	+ 3.00	- 2.11	+ 1.39	+ 5.36	+ 3.98	- 4.25	- 6.05	- 2.98

Table 6

Capital and Fuel Costs

	Reactor Number	0	1	1a	2	3	4	5a	5b	6
	Additional Costs [$]·10[4]	Reference Reactor	Smaller Coolant Fraction α	Smaller Coolant Fraction α and larger Δϑ	Smaller H/D	5 °/o BeO added	10 °/o Mo in Fuel	UC-Fuel Same Geometry as Oxide	UC-Fuel Compact Core	Vented Fuel
a) Capital Costs	Core Shroud	0	- 0.18	- 0.20	+ 0.23	+ 0.23	0	0	- 0.95	- 1.00
	Thermal Barrier	0	- 0.33	- 0.38	+ 0.35	+ 0.28	0	0	- 1.80	- 2.25
	Inner Vessel + Shielding	0	- 0.98	- 1.13	+ 1.25	+ 1.30	0	0	- 5.55	- 6.00
	Reactor Vessel	0	- 0.43	- 0.58	+ 0.63	+ 0.65	0	0	- 2.55	- 2.75
	Shielding Assemblies	0	- 0.70	- 0.75	+ 0.20	+ 0.90	0	0	- 3.68	- 6.00
	Refueling Equipment	0	- 0.13	- 0.13	- 0.60	+ 0.15	0	0	- 0.65	- 3.00
	Reactor Vessel + Internal Inventory	0	- 2.75	- 3.17	+ 2.06	+ 3.51	0	0	- 15.18	- 21.00
	Intermediate Heat Exchanger + Containment	0	0	+ 84.50	0	0	0	0	0	0
	Pumps	0	+ 450.00	- 110.00	- 120.00	+ 30.00	0	0	+ 380.00	- 50.00
	Gas Purification System	0	0	0	0	0	0	0	0	+ 230.00
	Capital Costs [$]·10[6]	0	+ 4.47	- 0.29	- 1.18	+ 0.34	0	0	+ 3.65	+ 1.84
b) Fuel Costs	Gross Fuel Costs [mills/kWhe]	+ 0.89	+ 0.88	+ 0.88	+ 0.94	+ 0.90	+ 0.89	+ 0.95	+ 0.77	+ 0.85
	Plutonium Gain	- 0.13	- 0.11	- 0.15	- 0.13	- 0.04	- 0.05	- 0.26	- 0.19	- 0.17
	Fuel Costs [mills/kWhe]	+ 0.76	+ 0.77	+ 0.73	+ 0.81	+ 0.86	+ 0.84	+ 0.69	+ 0.58	+ 0.68
	Fuel Cost Difference [mills/kWhe]	0	+ 0.01	- 0.03	+ 0.05	+ 0.10	+ 0.08	- 0.07	- 0.18	- 0.08
	Annual Addit.Cost [$]·10[6]	0	+ 0.03	- 0.23	+ 0.33	+ 0.65	+ 0.51	- 0.55	- 1.25	- 0.62
	Equivalent Capital Costs (present value) [$]·10[6]	0	+ 0.26	- 1.82	+ 2.57	+ 5.02	+ 3.98	- 4.25	- 9.70	- 4.82
	Capital Costs [$]·10[6]	0	+ 4.42	- 0.29	- 1.18	+ 0.34	0	0	+ 3.65	+ 1.84
	Fuel Costs	0	+ 0.26	- 1.82	+ 2.57	+ 5.02	+ 3.98	- 4.25	- 9.70	- 4.82
	Total Additional Costs [$]·10[6]	0	+ 4.73	- 2.11	+ 1.39	+ 5.36	+ 3.98	- 4.25	- 6.05	- 2.98

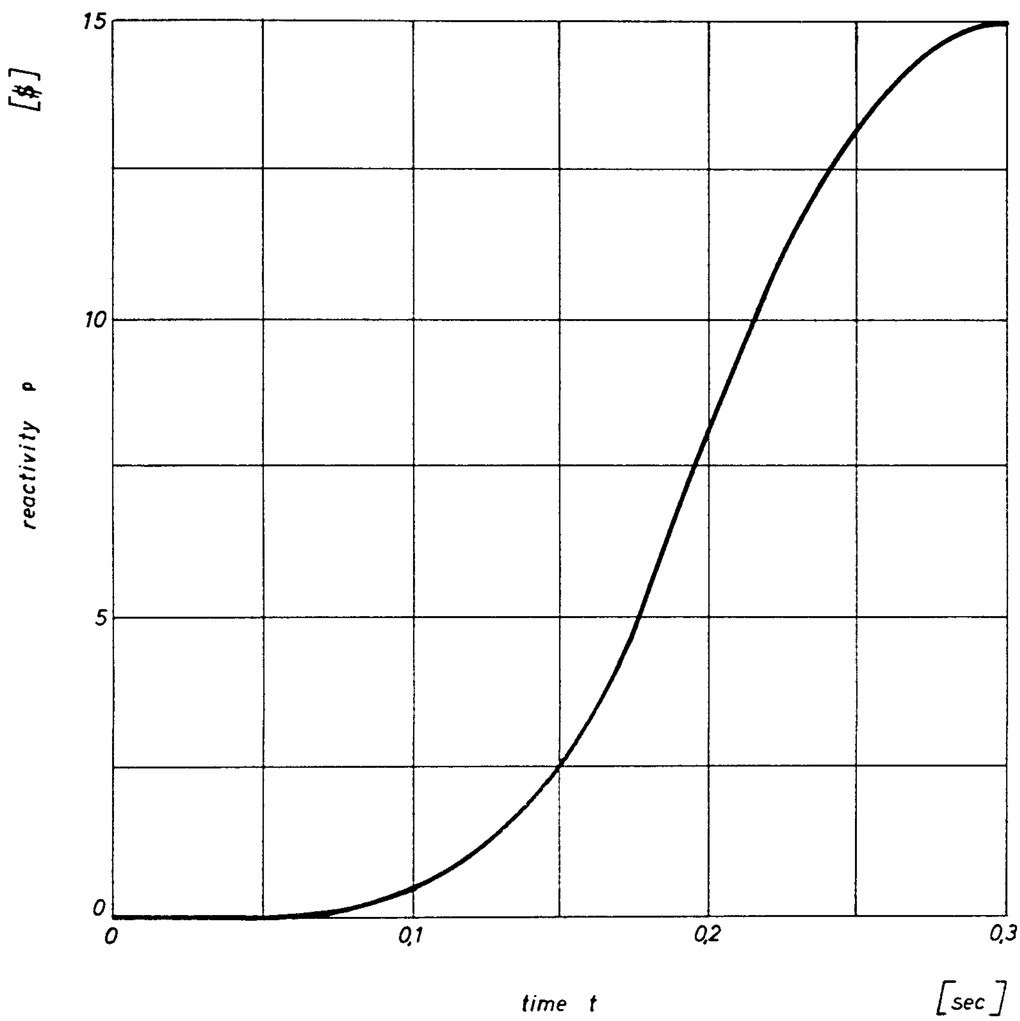

Fig. 1

Reactivity versus time of safety system

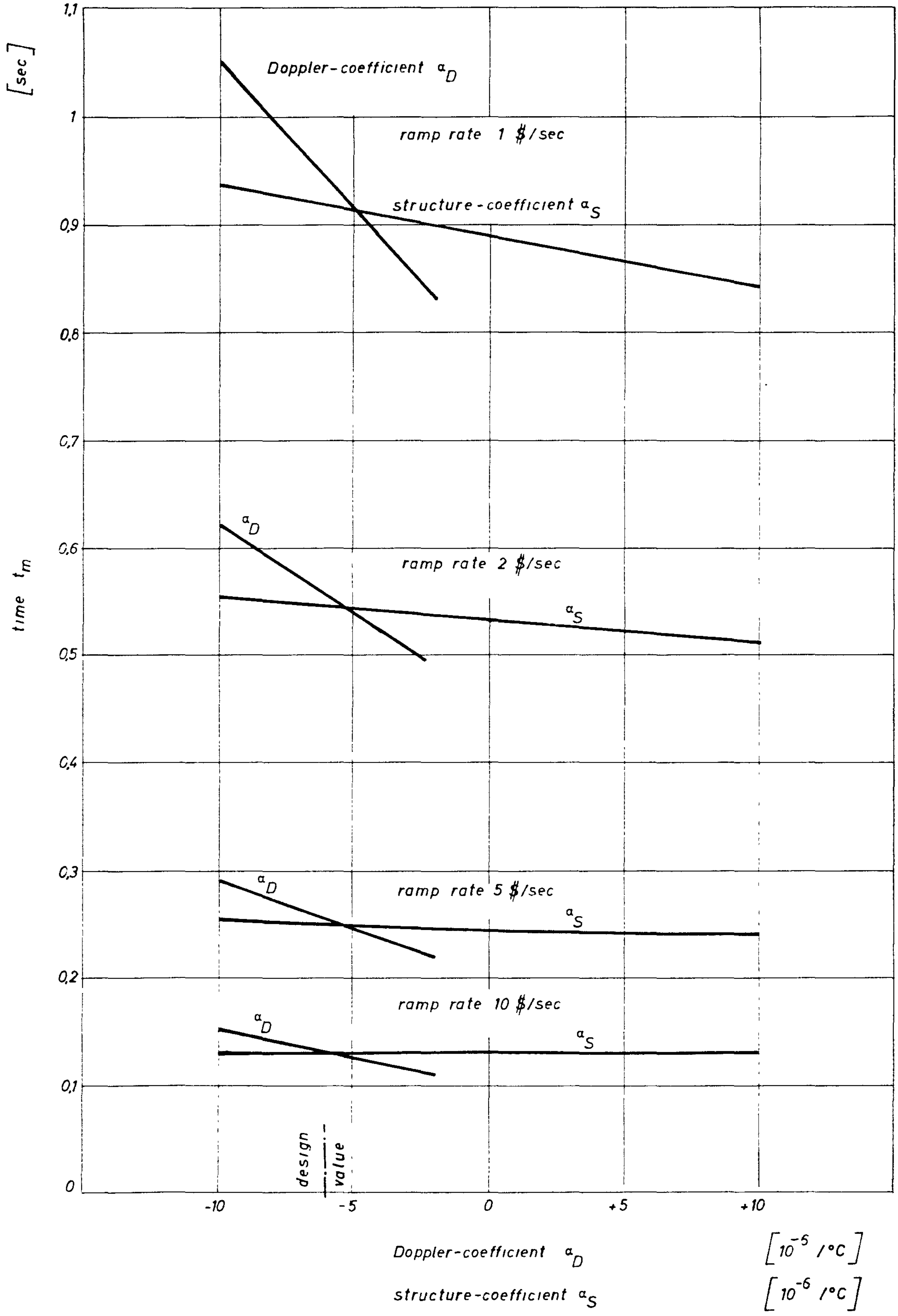

Fig. 2

Time t_m to reach fuel melting versus Doppler-coefficient α_D and structure-coefficient α_S (Reference Reactor)

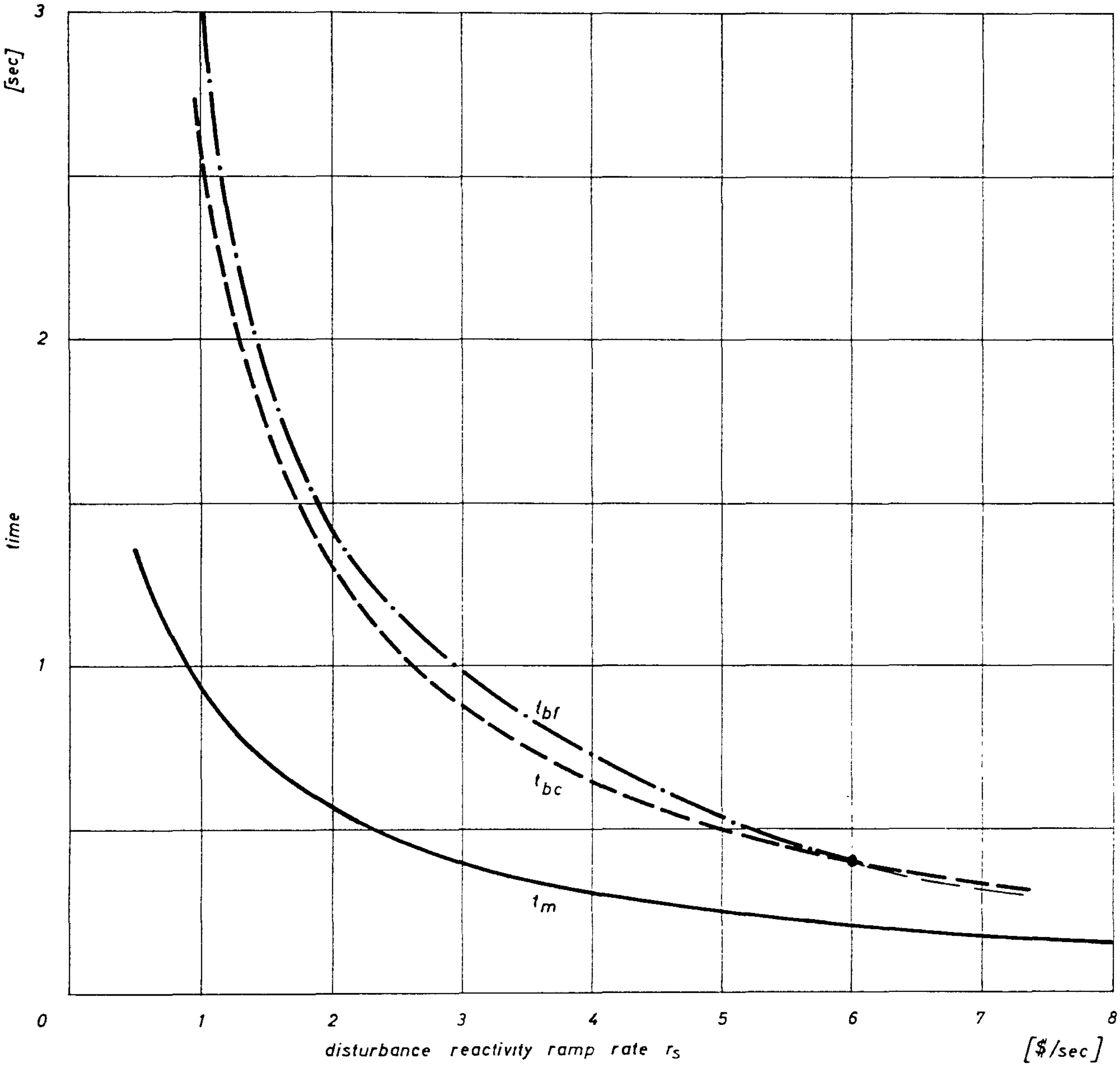

Fig. 3

Time t_m to reach fuel melting, time t_{bc} to reach coolant boiling (core exit) and time t_{bf} to reach fuel boiling versus disturbance reactivity ramp rate.

(Reference Reactor)

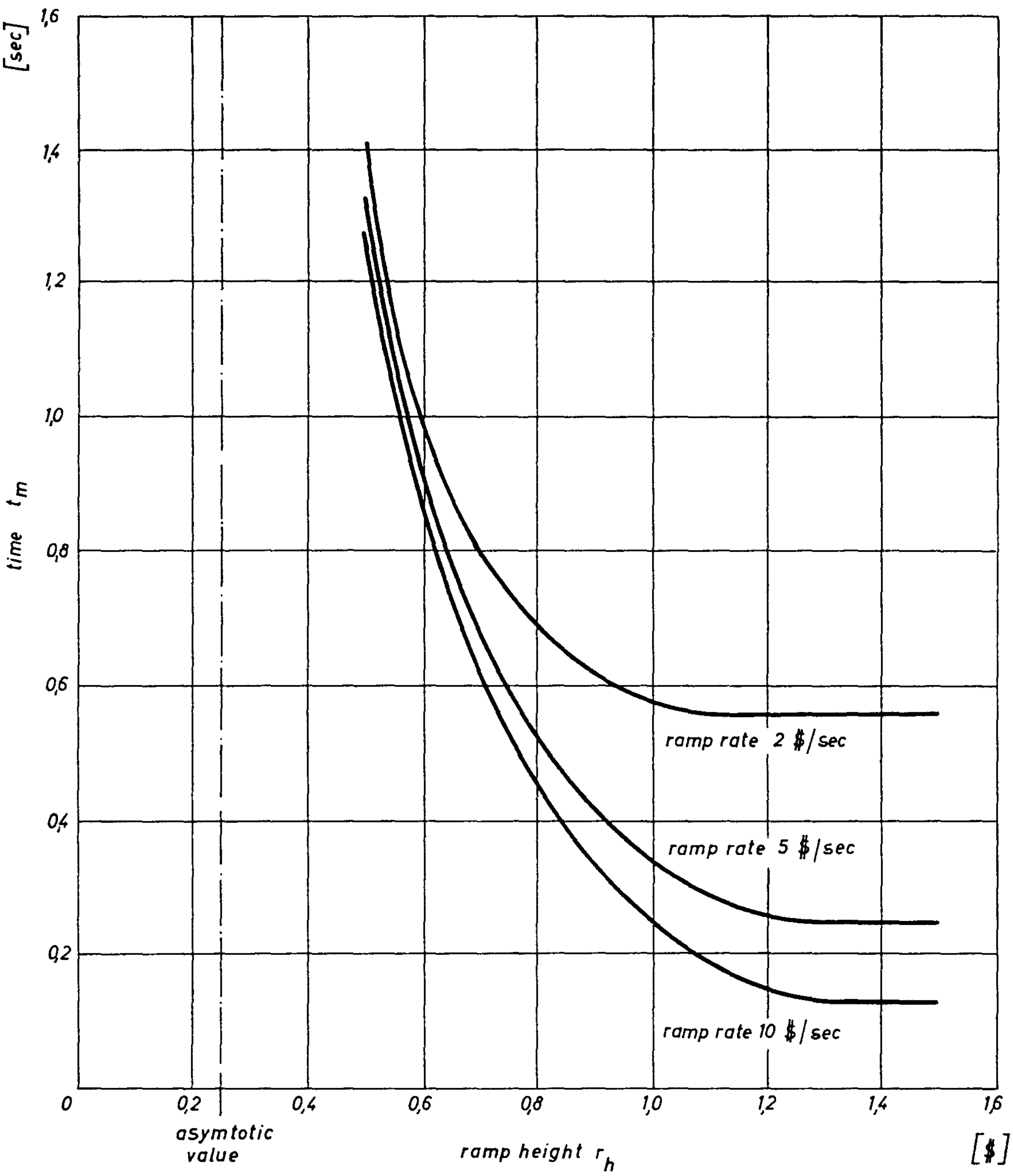

Fig. 4

Time t_m to reach fuel melting versus ramp height for different disturbance reactivity ramp rates (Reference Reactor)

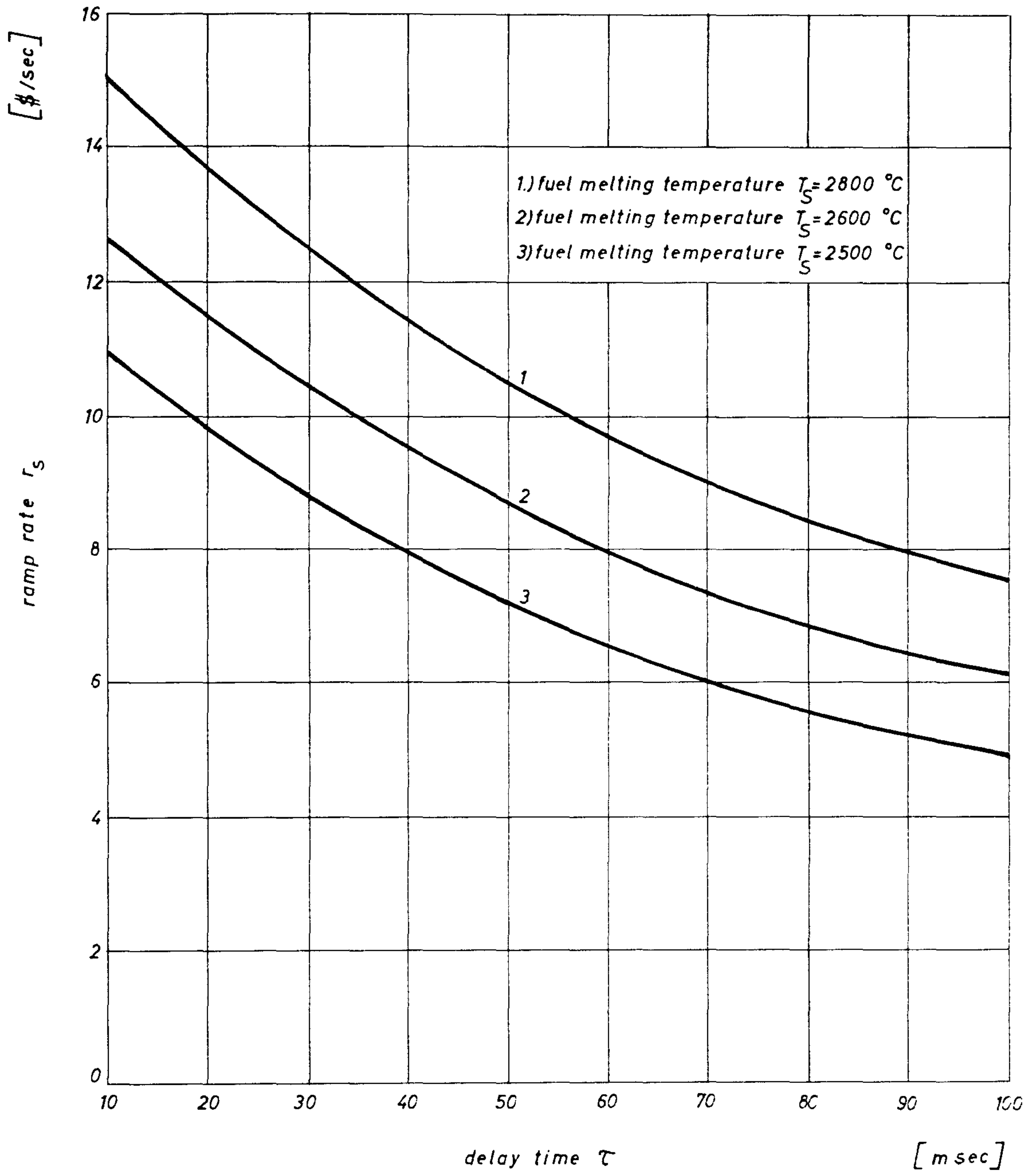

Fig. 5

Maximum allowable disturbance ramp rate r_s for different fuel melting temperature versus delay time

(Reference Reactor)

(Doppler-coefficient = $6 \cdot 10^{-6}/{}^{o}C$)

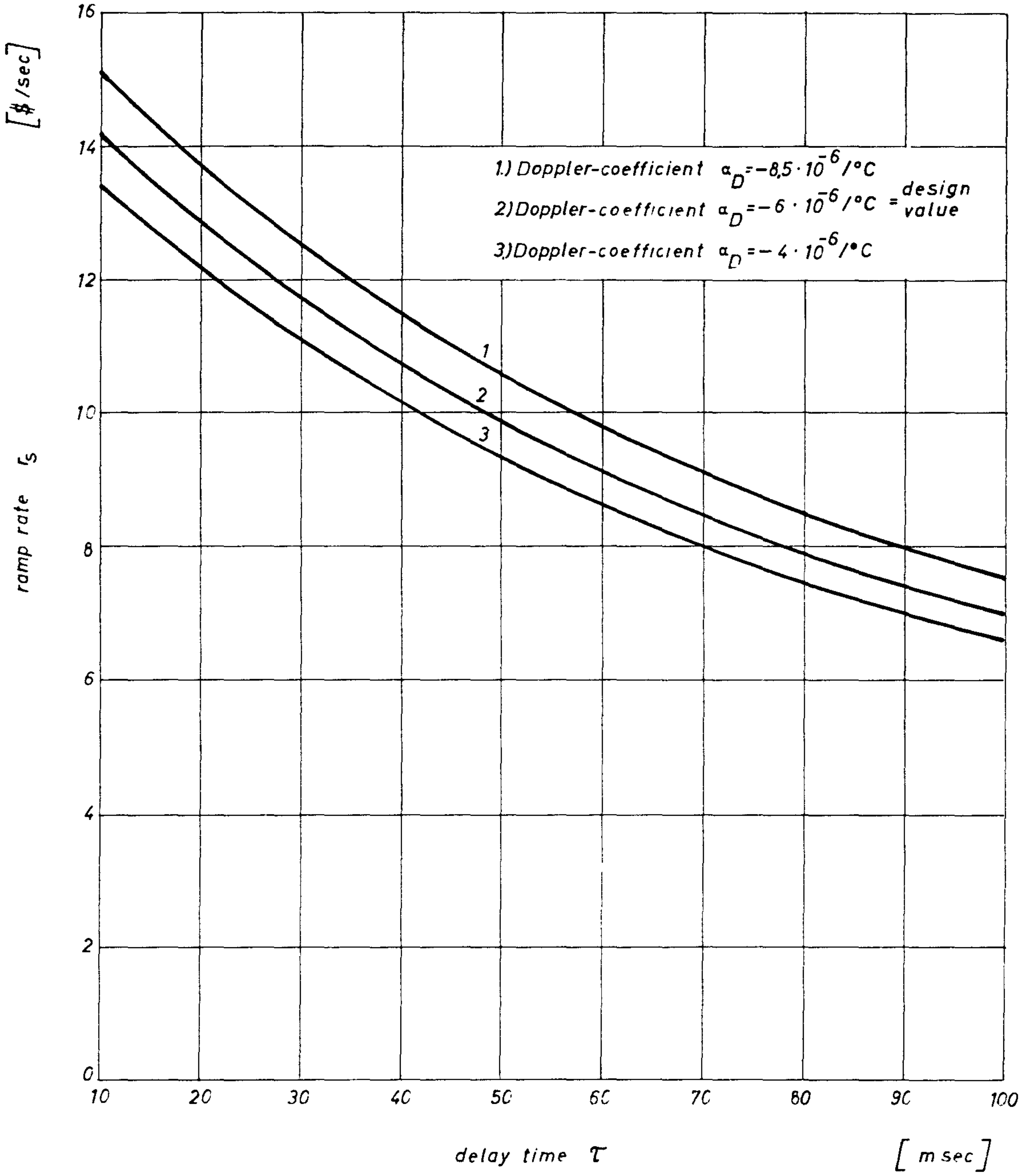

Fig. 6

Maximum allowable disturbance ramp rate r_s for different Doppler-coefficients versus delay time (Reference Reactor)

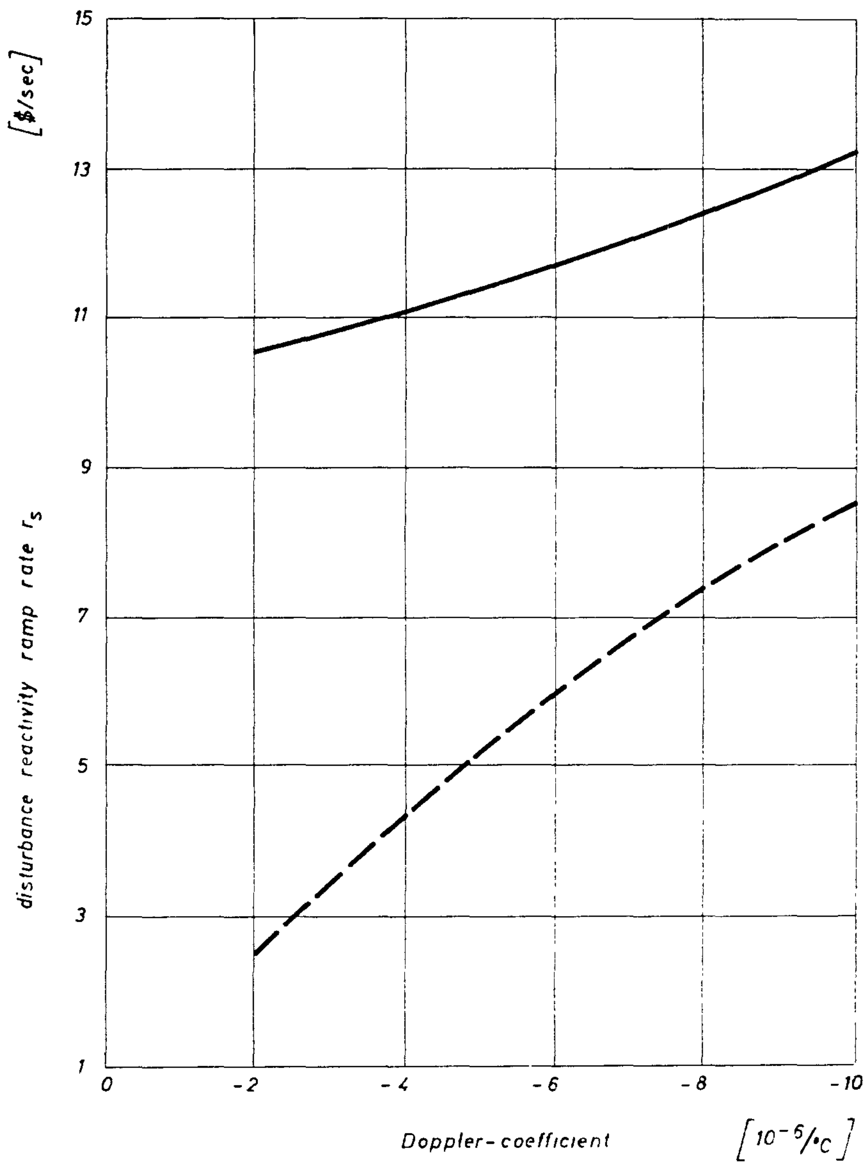

Fig. 7

Maximum disturbance reactivity ramp rate r_s for $\tau = 30$ msec versus Doppler-coefficient. (Dotted line: r_s for $t_{bc} = t_{bf}$)

(Reference Reactor)

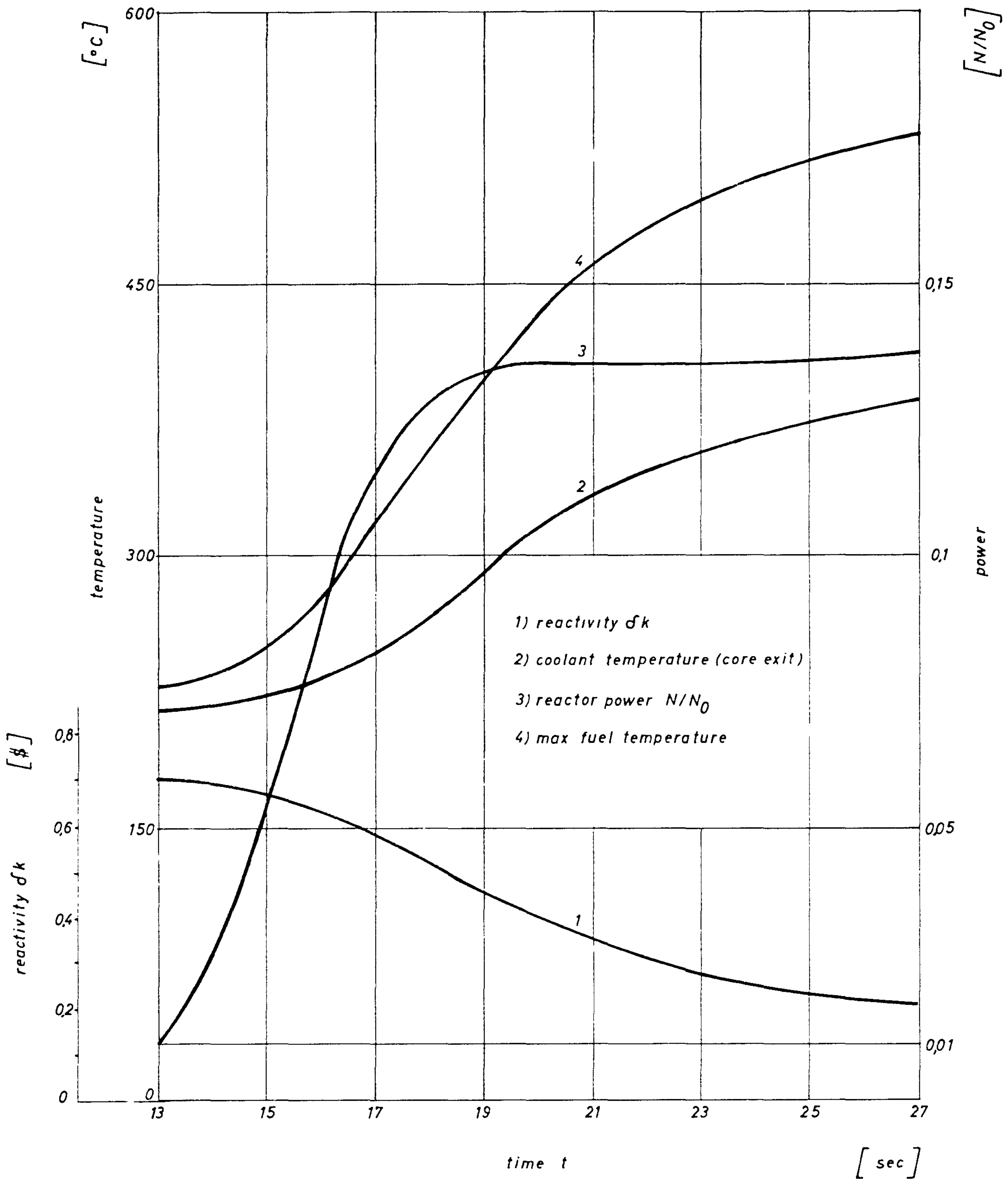

Fig. 8

Loading Accident

Reactivity, power, max. fuel center temperature and coolant temperature (core exit) versus time.

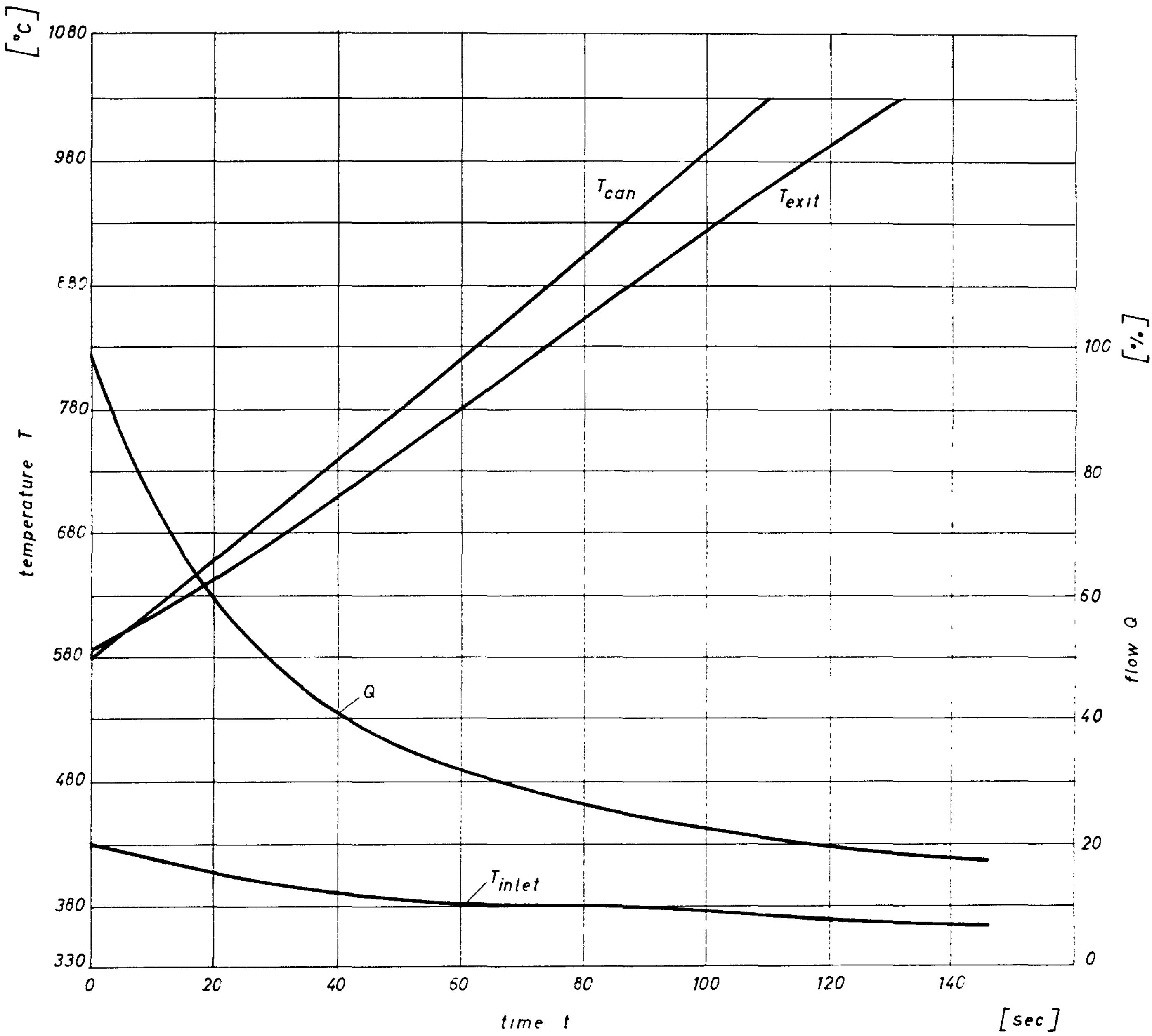

Fig. 9

Loss of pumping power
Coolant flow Q, core inlet temperature T_{inlet},
core exit temperature T_{exit}, can temperature T_{can}
versus time t (Reference Reactor)

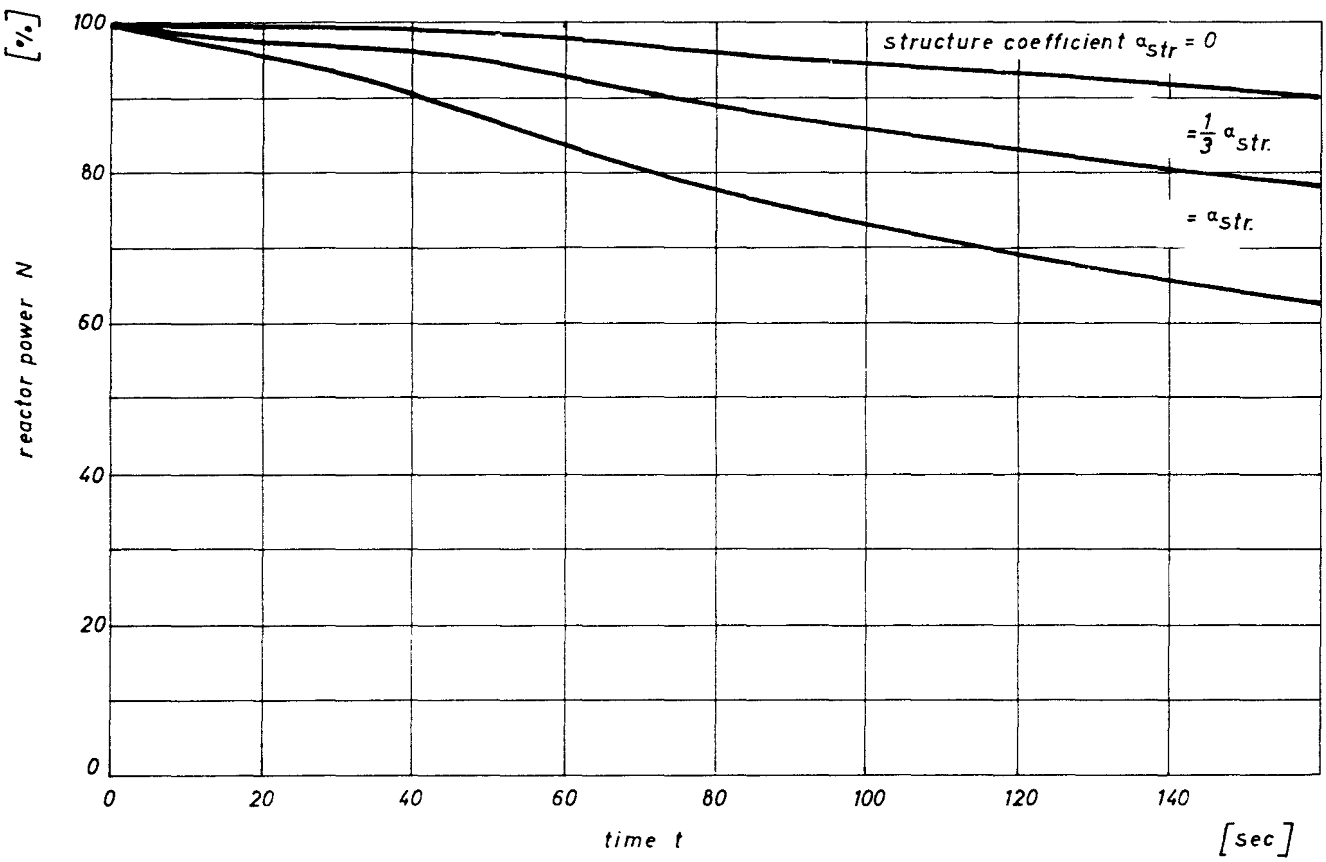

Fig. 10

Loss of pumping power
Reactor power N versus time for different structure coefficients α_{str} (Reference Reactor)

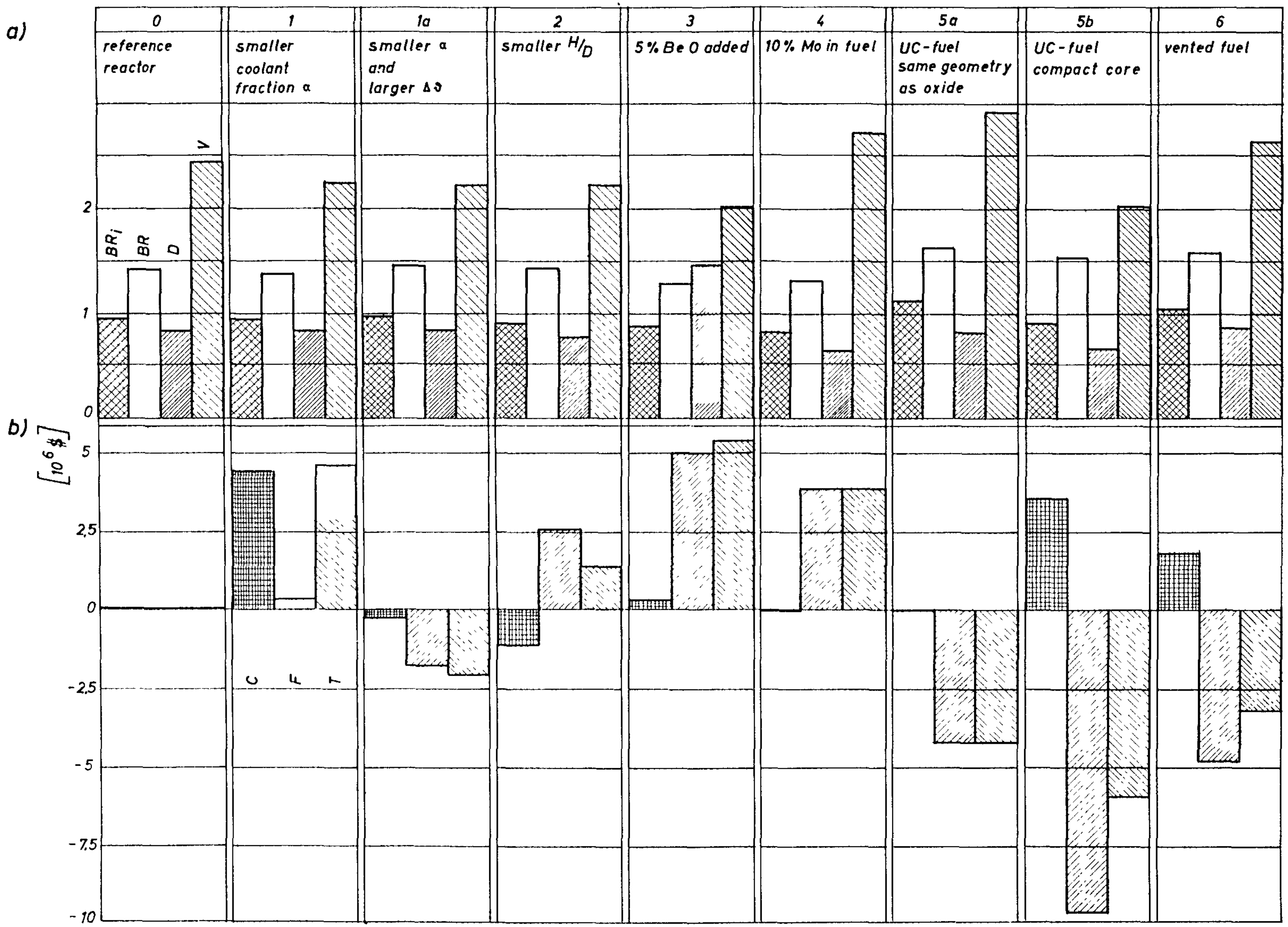

Fig. 11

a) Internal breeding ratio BR_1 , total breeding ratio BR, Doppler-constant D and reactivity for complete core sodium loss V, for different reactors

b) Total additional costs T and differences in capital costs C and fuel costs F

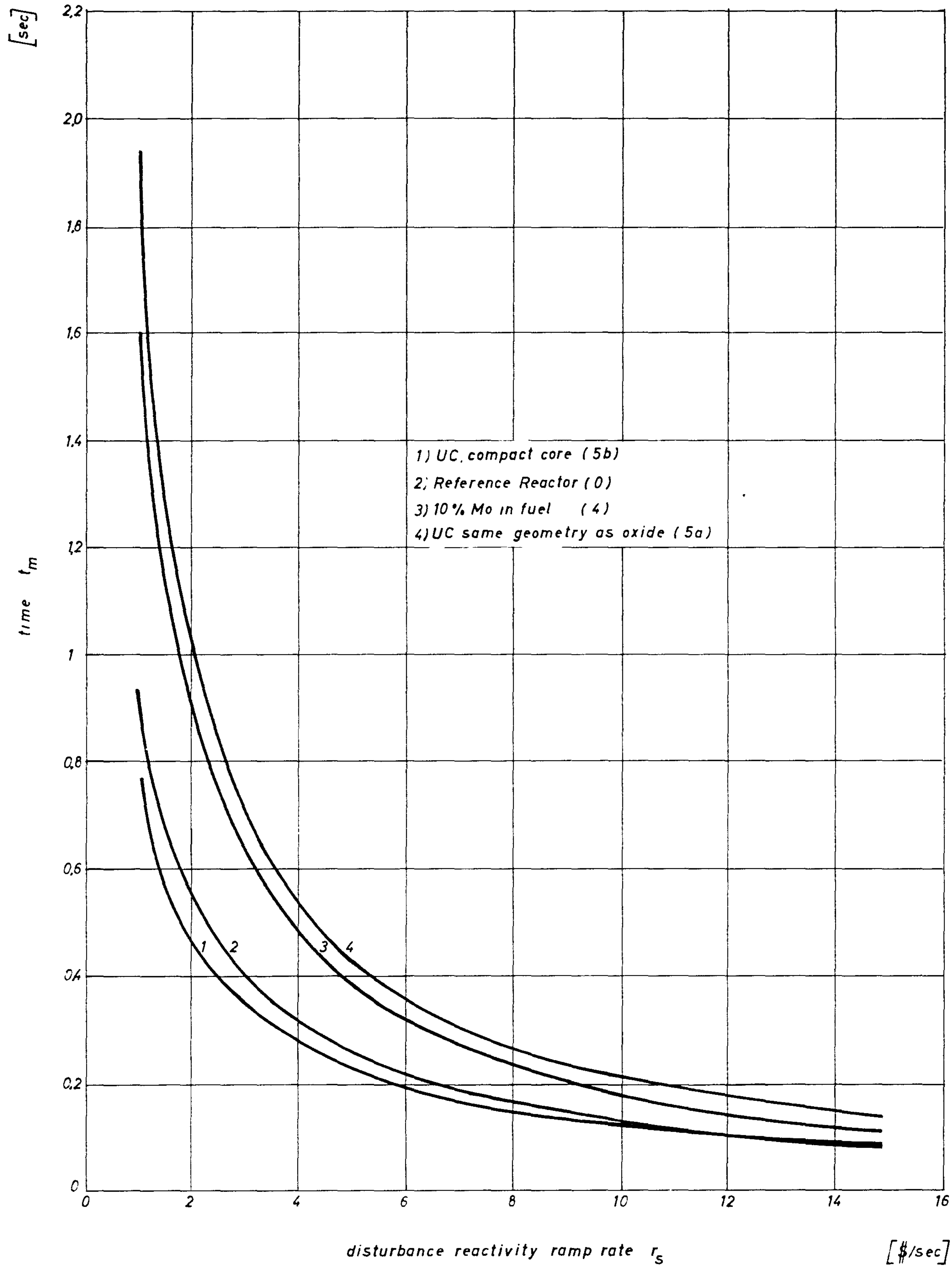

Fig. 12

Time t_m to reach fuel melting versus disturbance reactivity ramp rate for different fuel materials

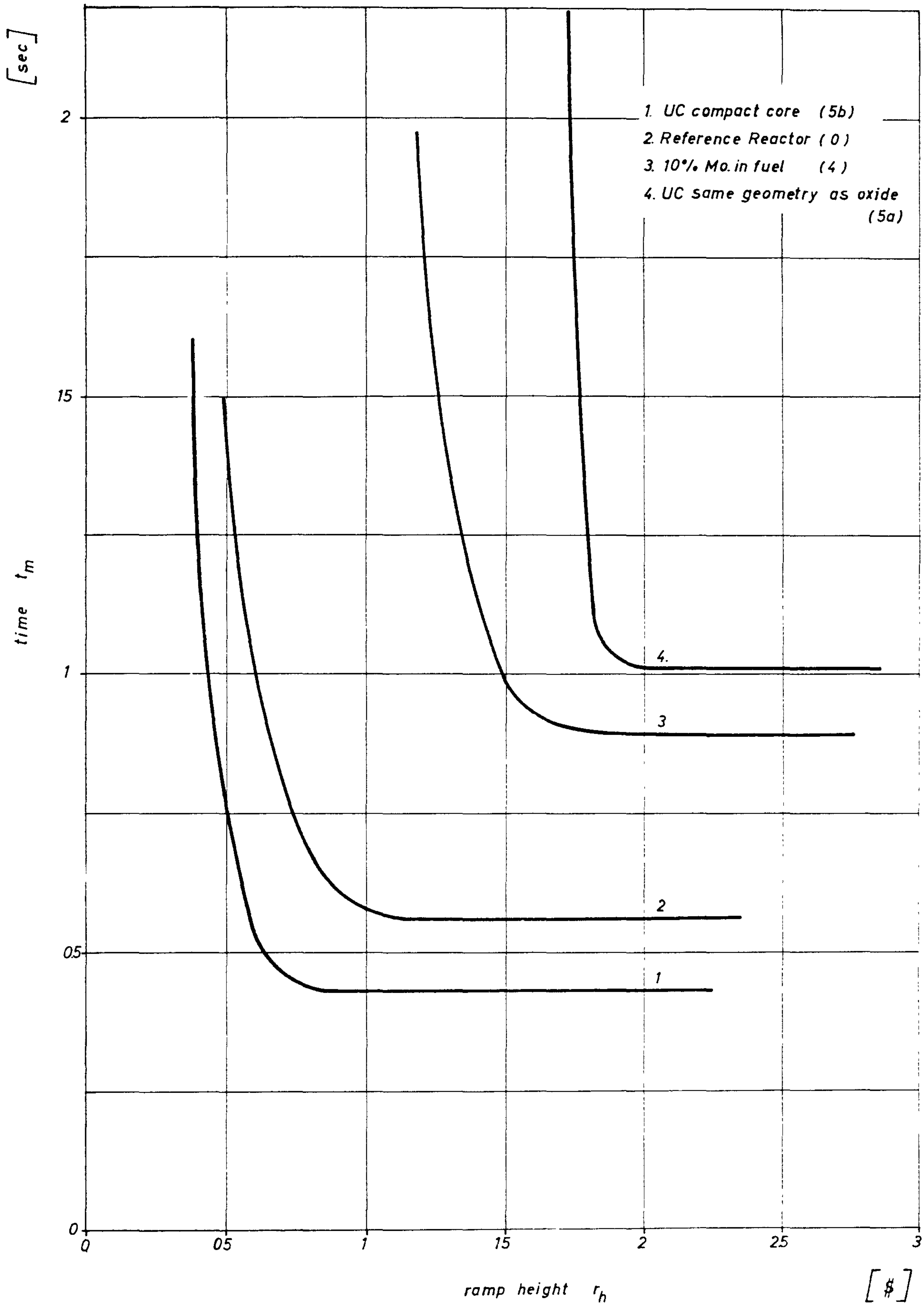

Fig. 13

Time t_m to reach fuel melting versus ramp height for a ramp rate of 2 $/sec for different fuels.

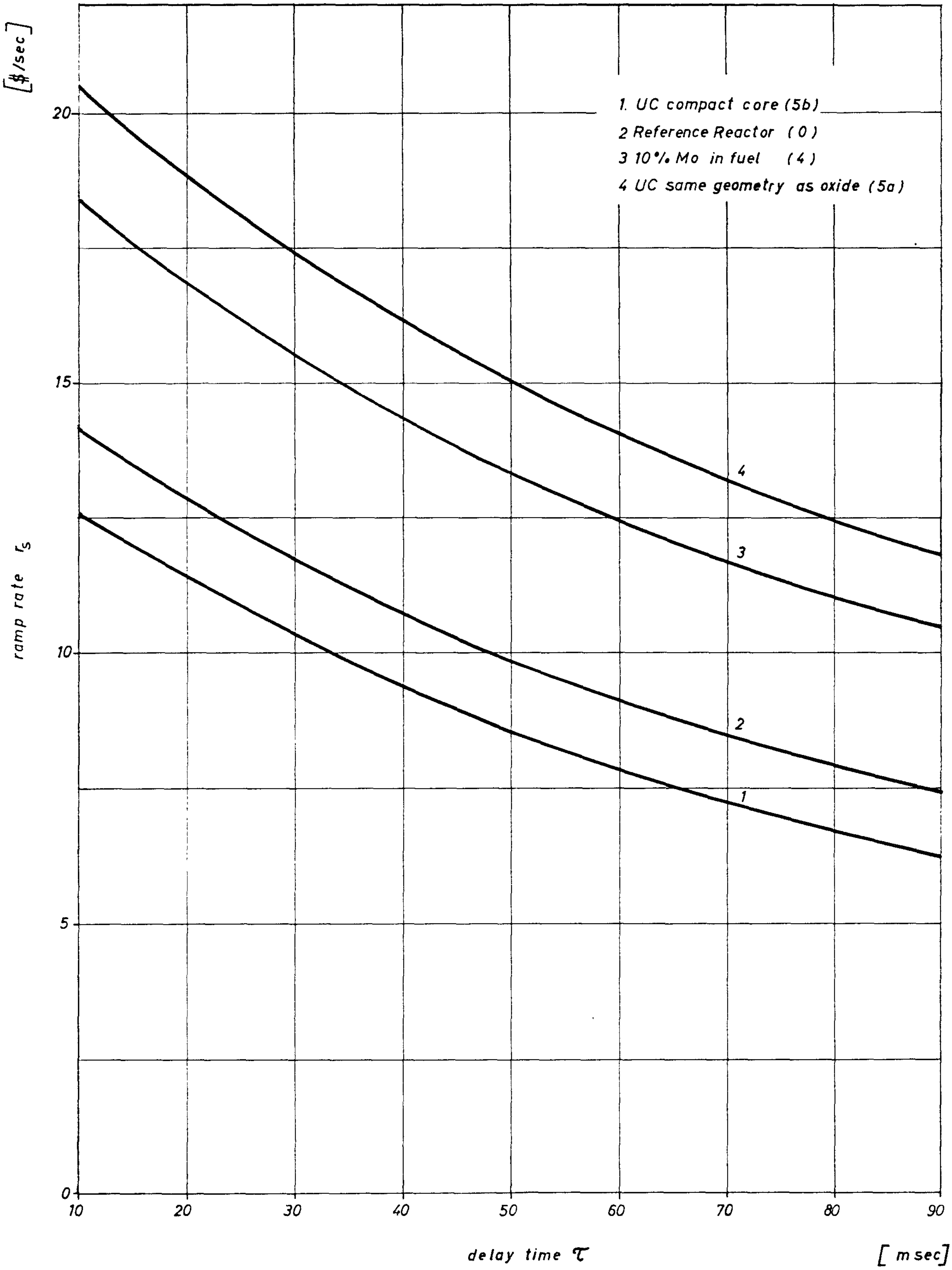

Fig. 14

Maximum allowable disturbance ramp rate r_s versus delay time for different fuels.