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REACTIVITY EFFECTS DUE TO VARIATIONS IN NUCLEAR PARAMETERS IN A THERMAL POWER REACTOR

by

G. Casini and M. Paillon

1963

Joint Nuclear Research Center
Ispra Establishment - Italy

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Applied Physics and Mathematics

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The most important nuclear cross-section influences have been found to be due to U_{235} and Pu_{239} fuel isotopes. In particular, a variation of 1 % in the fission cross-section of Pu_{239} gives a variation of about 0.6 % on the reactivity for an ORGEL type reactor. An improvement of the up-to-date experimental techniques to measure these cross-sections is therefore desirable.

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As far as the lattice parameters are concerned, the calculations have confirmed the necessity to improve the accuracy of the activation experiments, in particular for resonance escape probability determination, in order to have results comparable with criticality measurements.

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TABLE DES MATIERES

REACTIVITY EFFECTS DUE TO VARIATIONS IN NUCLEAR PARAMETERS IN A THERMAL POWER REACTOR

SUMMARY

The influence on the neutron balance of small variations of some nuclear parameters in a natural uranium power reactor has been investigated.

The most important nuclear cross section influences have been found to be due to U_{235} and Pu_{239} fuel isotopes. In particular, a variation of 1 % in the fission cross section of Pu_{239} gives a variation of about 0.6 % on the reactivity for an ORGEL type reactor. An improvement of the up-to-date experimental techniques to measure these cross sections is therefore desirable.

As far as the lattice parameters are concerned, the calculations have confirmed the necessity to improve the accuracy of the activation experiments, in particular for resonance escape probability determination, in order to have results comparable with criticality measurements.

1. Introduction

The incidence on the reactivity balance due to small variations of some nuclear parameters in a thermal power reactor has been investigated.

The nuclear parameters taken into account have been divided into two categories :

1) Nuclear cross sections of the various materials and fuel isotopes present during the life of the reactor, which can be determined by microscopic measurements;

2) Lattice parameters which can be evaluated by integral measurements (activation, oscillation, criticality, pulsed neutron experiments).

This division can also be related to the calculation method used to evaluate the reactivity.

If we dispose of a big computing machine, numerical calculations can be performed starting with the elementary cross sections of the different materials involved in the project. In such a case, one could imagine that the mathematical model for neutron balance calculation is so accurate that the most important sources of errors in reactivity are due to the uncertainties on the input data (nuclear cross sections).

If we want to evaluate quickly the lattice properties by simplified models, adjustments on integral measurements or correlations between theory and experiments are necessary. In this case, the accuracy of the calculations is practically determined by the precision of the integral experiments to which the neutron cycle model assumed is related.

The calculations have been performed for the case of an ORGEL type reactor (natural uranium fuelled, heavy water moderated, organic cooled reactor); the CAROLINE I (Ref. 1), TERMIDOR (Ref. 2) and

RLT-2 (Ref. 3) codes, established by the Reactor Physics Department of Euratom, Ispra (Italy) have been used.

2. Reactor reference case data

The fuel element geometrical data of the reference case reactor are the following (a cross section is represented in fig. 1) :

- type of fuel :	natural uranium carbide
- number of rods on the cluster :	7
- carbide volume per unit height :	30.1 cm^2
- ratio organic-carbide volumes :	0.465
- total ratio aluminium (canning plus tubes) - carbide volumes	0.624

The assumed ratio heavy water-fuel is 18.

The unirradiated lattice parameters, as calculated by CAROLINE I code, are the following :

$\varepsilon = 1.0280$ $\quad L^2 = 184.4$

$p = 0.8880$ $\quad L_s^2 = 118.4$

$\eta = 1.2824$ $\quad B_m^2 = 3.07$

$f = 0.9130$ $\quad C_o = 0.800$

$k_\infty = 1.0950$

The long term reactivity curve, as calculated by TERMIDOR and RLT-2 codes, is represented in Fig. 2. This curve corresponds to a continuous bidirectional fuel movement as in CANDU reactor. The steady-state conditions are supposed. A value of 5.10^{13} $n/cm^2 sec$ for the mean thermal flux on the reactor is assumed.

3. Nuclear cross sections

The most important nuclear cross sections which enter into the CAROLINE and RLT-2 codes for an ORGEL lattice calculation have been varied; the main results are summarized in tables I and II. In these tables are given, for each nuclear cross section, the reference value,

the variation chosen, the incidence respectively on the reactivity for a fuel burn-up of about 6600 MWD/T and on the burn-up for an initial reactivity reserve of 2 %.

Table I refers to fuel isotope nuclear data; the variations assumed to test the influence of these elements on the neutron balance are of the same order of magnitude as the up-to-date inaccuracies from measurements deduced from Ref. 4.

Table I

Cross section symbol	Reference value	Cross section variation	Reactivity incidence	Burn-up incidence	
				relative value	absolute value
	barn	%	pcm	%	MWD/T
σ_{a25}	683.04	1	- 440	- 7.0	- 460
σ_{f25}	577.01	1	+ 580	+ 9.2	+ 605
σ_{a39}	1029.10	1	- 425	- 6.7	- 445
σ_{f39}	742.15	1	+ 600	+ 9.5	+ 630
σ_{a40}	277.87	10	- 100	- 1.6	- 105
σ_{R40}	1.734×10^5	33	+ 100	+ 1.6	+ 105
σ_{a41}	1426.00	6	- 115	- 1.8	- 120
σ_{f41}	1015.20	2	+ 40	+ 0.6	+ 40
ν_{25}	2.450	1	+ 580	+ 9.2	+ 605
ν_{39}	2.885	1	+ 500	+ 8.2	+ 525
ν_{41}	3.06	2	+ 40	+ 0.6	+ 40

Table II refers to the structural, cooling and moderating materials of the ORGEL type lattice.

For aluminium the variation of 4 % assumed corresponds to the difference between the value of the 2200 m/s absorption cross section given in BNL-325, second edition (Ref. 5) and the value given in the Supplement n° 1 (Ref. 4).

For the organic 2200 absorption cross section, reference to the measurement inaccuracy of hydrogen cross section has been made.

Due to the fact that the scattering cross section of the organic changes with the energy in the thermal region, the reference value is obtained by averaging the cross section curve on the thermal spectrum. Therefore, the variation assumed for this cross section takes into account, first the inaccuracies on the measured values of the total cross section as a function of the energy (3-4 %) and secondly the uncertainty on the spectrum average. The heavy water absorption cross section variation takes into account the difficulties in knowing the exact degree of purity of the heavy water. For the other cross sections, the variations assumed are of the same order of magnitude as the inaccuracies from measurements. The order of the columns is the same as in table I.

Table II

Cross section symbol	Reference value	Cross section variation	Reactivity incidence	Burn-up incidence	
				relative value	absolute value
	cm^{-1}	%	pcm	%	MWD/T
Σ_{ao}	0.00786	2	- 55	- 0.9	- 60
Σ_{so}	1.489	10	- 85	- 1.35	- 90
$(\xi \Sigma_s)_{oep}$	0.685	5	+ 40	+ 0.6	+ 45
Σ_{aAl}	0.0140	4	- 200	- 3.0	- 210
Σ_{am}	$0.66.10^{-4}$	5	- 60	- 0.95	- 65
Σ_{tm}	0.395	5	+ 185	+ 2.9	+ 195

4. Lattice parameters

The results are summarized in table III. Also in this case the variations assumed are of the same order of magnitude as the experimental errors in the measurements.

As far as the thermal utilization factor is concerned, we assumed a variation of 5 % for all the disadvantage factors of the various media (organic, tubes, heavy water) surrounding the fuel.

For the resonance escape probability factor, a variation of 1 % corresponds, for the ORGEL lattice, to a variation of 8.4 % in the effective resonance integral and 2.3 % in the initial conversion factor, two parameters which are measured.

As regards the spectrum effect, we proceeded in the following manner* : we divided the uncertainty sources into three categories, corresponding to the following phenomena (Ref. 2) :

- homogeneous hardening in the lattice : an uncertainty of 10°C in the equivalent neutron temperatures of the various media of the cell was supposed to be representative of the up-to-date situation;
- fuel selective hardening : 20°C variation on the fuel neutron temperature was taken into account;
- rethermalization effect : a change of 100 % of the transfer cross section attributed to the organic in the TERMIDOR program (Ref. 2) was assumed to be representative of the uncertainties on the theoratical model used to calculate this effect. This corresponds to a variation of about 20°C on the equivalent fuel neutron temperature.

By supposing the above-mentioned three errors to be independent from one another, a total variation of 30°C was found to be representative of the total spectrum uncertainties.

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* This suggestion was made by Mr. Rossi, who also gave us a part of the numerical values for the spectrum effect calculations.

Table III

Parameter symbol	Reference value	Variation	Reactivity incidence	Burn-up incidence	
				relative value	absolute value
			pcm	%	MWD/T
f	0.9130	5% in disadvantage factors	- 480	- 7.6	- 500
ε	1.0280	0.3 %	+ 345	+ 5.5	+ 365
p	0.88804	1.0 %	+ 625	+ 9.9	+ 655
k_∞	1.0950	1.0 %	+ 1000	+ 16.1	+ 1060
B_m^2	3.07	0.15 m^{-2}	+ 420	+ 6.7	+ 440
T_{nf}	240°C	30°C	- 15	- 0.2	- 15
r	0.055	30 %	+ 10	+ 0.1	+ 10

5. Conclusions

Some general conclusions for natural uranium thermal power reactors can be drawn from this study.

I. Nuclear cross sections

a) The incidences on the neutron balance of U_{235} and Pu_{239} cross sections are the most important ones. In particular, a variation of 1% in the fission cross section of Pu_{239} gives a variation of about 0.6 % on the reactivity from an ORGEL type reactor. An improvement of the up-to-date experimental techniques to measure these cross sections is therefore desirable. For U_{235}, however, it must be noted (see Fig. 3) that the strongest influence is for unirradiated conditions; if we can adjust the lattice unirradiated balance by an integral experiment, the U_{235} incidence changes signs and is reduced. For instance, the total error due to a variation of 1% in the U_{235} fission cross section is (see table I) 580 pcm and the corresponding pure long term reactivity effect (see Fig. 4) is -420 pcm.

b) A relatively sensible error (10 %) on Pu_{240} absorption cross section does not much alter the neutronic balance. This means, in particular, that the evaluation of the self-shielding effect of Pu_{240} does not need to be very accurate.

c) Finally, the up-to-date uncertainty on the Pu_{241} absorption cross section (about 6 %) does not induce a severe error on the reactivity balance. For the same reason, the lack of information on the curve of the total Pu_{241} cross section as a function of the energy from 0.5 to 4 ev does not seem to be so important.

II. Lattice parameters

a) A precision of about 0.5 % on k_∞ measurements is necessary to be comparable with the buckling experiment accuracy (0.15 m^{-2}). This figure (0.5) can be obtained by a PCTR type measurement but not by activation measurements of the different factors of k_∞ (ε , p, f, η). It seems therefore that for the moment this kind of experiments alone cannot prediot the overall neutron balance with sufficient accuracy.

b) The necessity of increasing the accuracy of the activation measurements is particularly evident in the case of the initial conversion factor where an experimental precision of 2.3 %, which is not easy to obtain, brings an error of 625 pcm on the reactivity balance.

c) The spectrum uncertainties do not seem to introduce a great error on the criticality calculations. This is mainly due to a compensation between opposite effects from U_{235} and Pu_{239} fission and absorption cross sections. But we know that the spectrum influence is important on the temperature coefficient calculations.

III. ORGEL parameters

For the ORGEL type reactor, the following remarks can be made :

a) The organic scattering cross section does not seem to be a critical parameter for the lattice calculations;

b) the absorption cross section of aluminium must be determined carefully because the influence on the reactivity, due to the great amount of such material in the fuel channel, is sensible.

References

1. Caroline I, a calculation method for non-irradiated organic liquid heavy water lattices - G. Casini, W. de Haan, E. Diana, C. Foggi, A. Kind, G. Rossi - EUR 144.e

2. Termidor, an IBM 7090 Fortran code to evaluate the thermal neutron spectrum in an ORGEL type fuel element - G. Rossi - EUR 105.e

3. Variation de la réactivité à long terme pour différents types de circulation du combustible - G. Blässer, G. Casini, J. Pillon - EUR 129.f

4. Neutron cross sections - D.J. Hughes, B.A. Magurno, M.K. Brussel - Supplement n° 1 to BNL 325, Second Edition

5. Neutron cross sections - J. Hughes and R.B. Schwartz - BNL 325, Second Edition.

List of symbols

ε	fast fission factor
p	resonance escape probability factor
η	thermal fission factor
f	thermal utilization factor
k_∞	infinite multiplication factor
L^2	diffusion area
L_s^2	slowing-down area
B_m^2	material buckling
C_o	initial conversion factor

Fuel isotopes :

"a"	total absorption
"f"	fission
"25"	U_{235}
"39"	Pu_{239}
"40"	Pu_{240}
"41"	Pu_{241}
σ	2200 m/s microscopic cross section value, excluding
σ_{R40}	resonance value of total cross section of Pu_{240}
ν	secondary neutrons per fission
Σ_{ao}	2200 m/s macroscopic absorption cross section of organic
Σ_{so}	thermal mean value of the macroscopic scattering cross section of organic
Σ_{aAl}	2200 m/s macroscopic absorption cross section of aluminium
Σ_{am}	2200 m/s macroscopic absorption cross section of heavy water
Σ_{tm}	thermal mean value of the macroscopic transport cross section of heavy water
T_{nf}	equivalent neutron temperature of the fuel (Westcott formalism)
r	epithermal ratio (Westcott formalism)
τ	integrated flux (neutrons/kbarn) (1 n/kb = 4000 MWD/T)

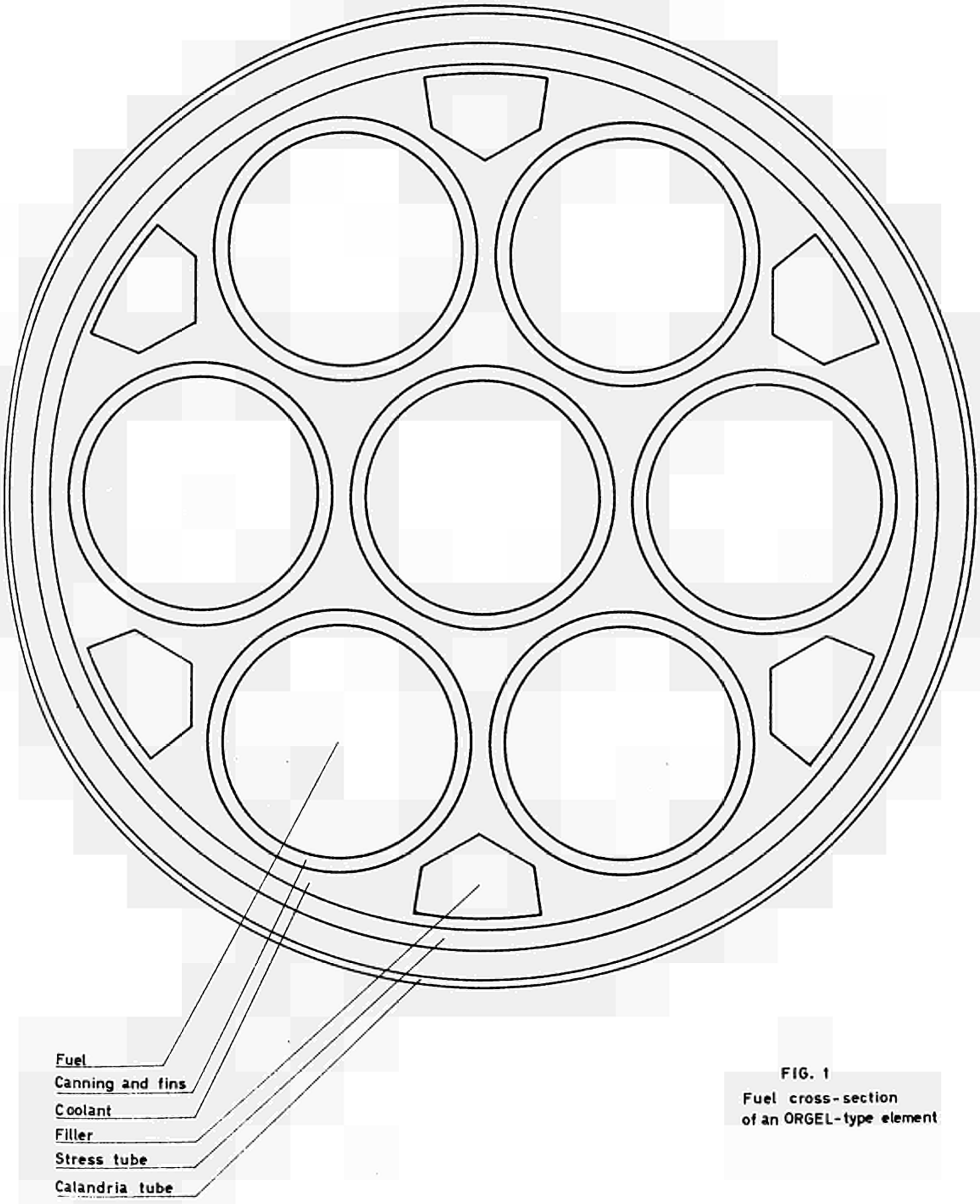

FIG. 1
Fuel cross-section
of an ORGEL-type element

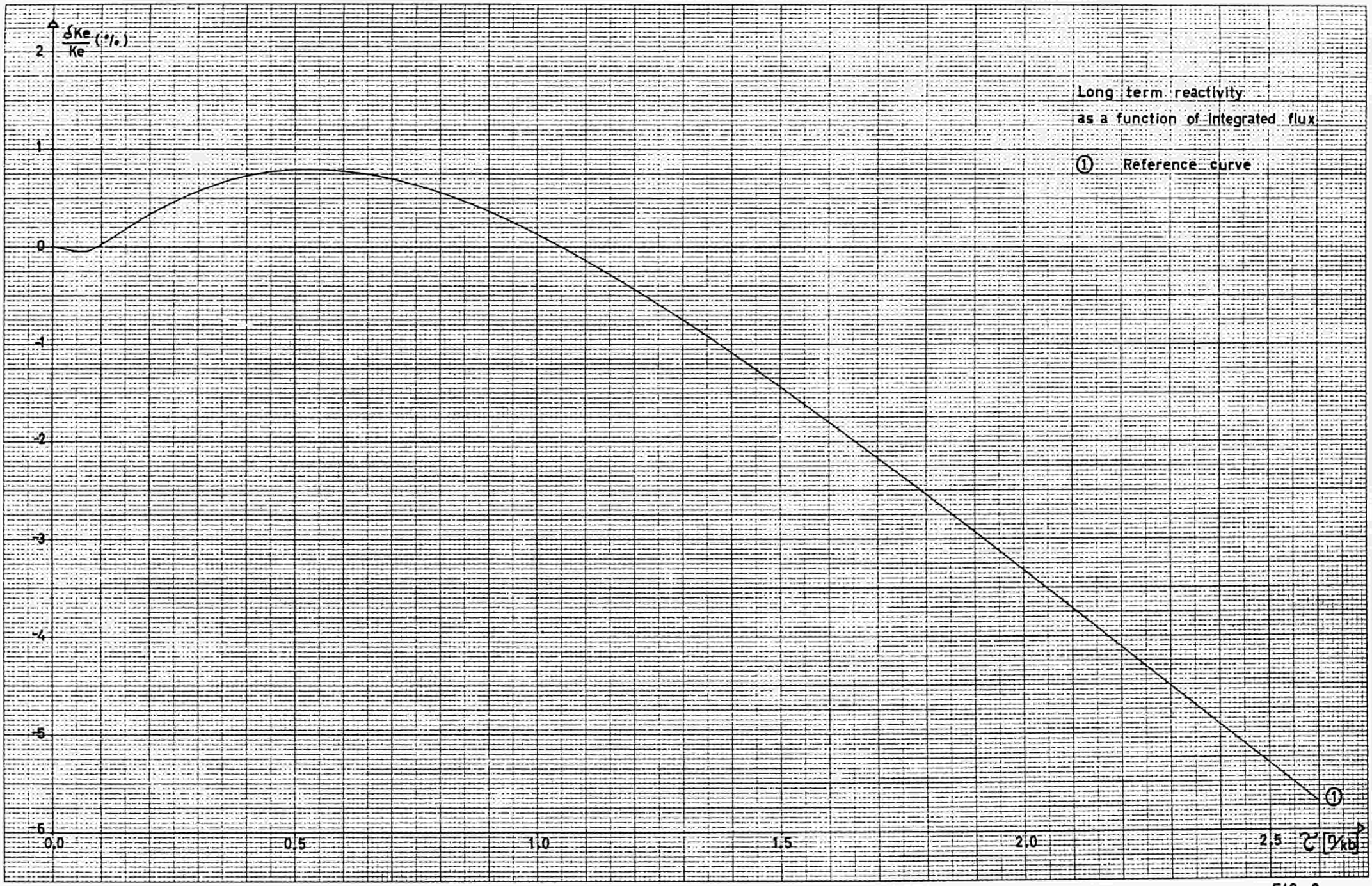
δKe/Ke (%)
2
1
0
-1
-2
-3
-4
-5
-6
0,0
0,5
1,0
1,5
2,0
2,5
τ [%kb]
Long term reactivity
as a function of integrated flux
① Reference curve
①

FIG. 2

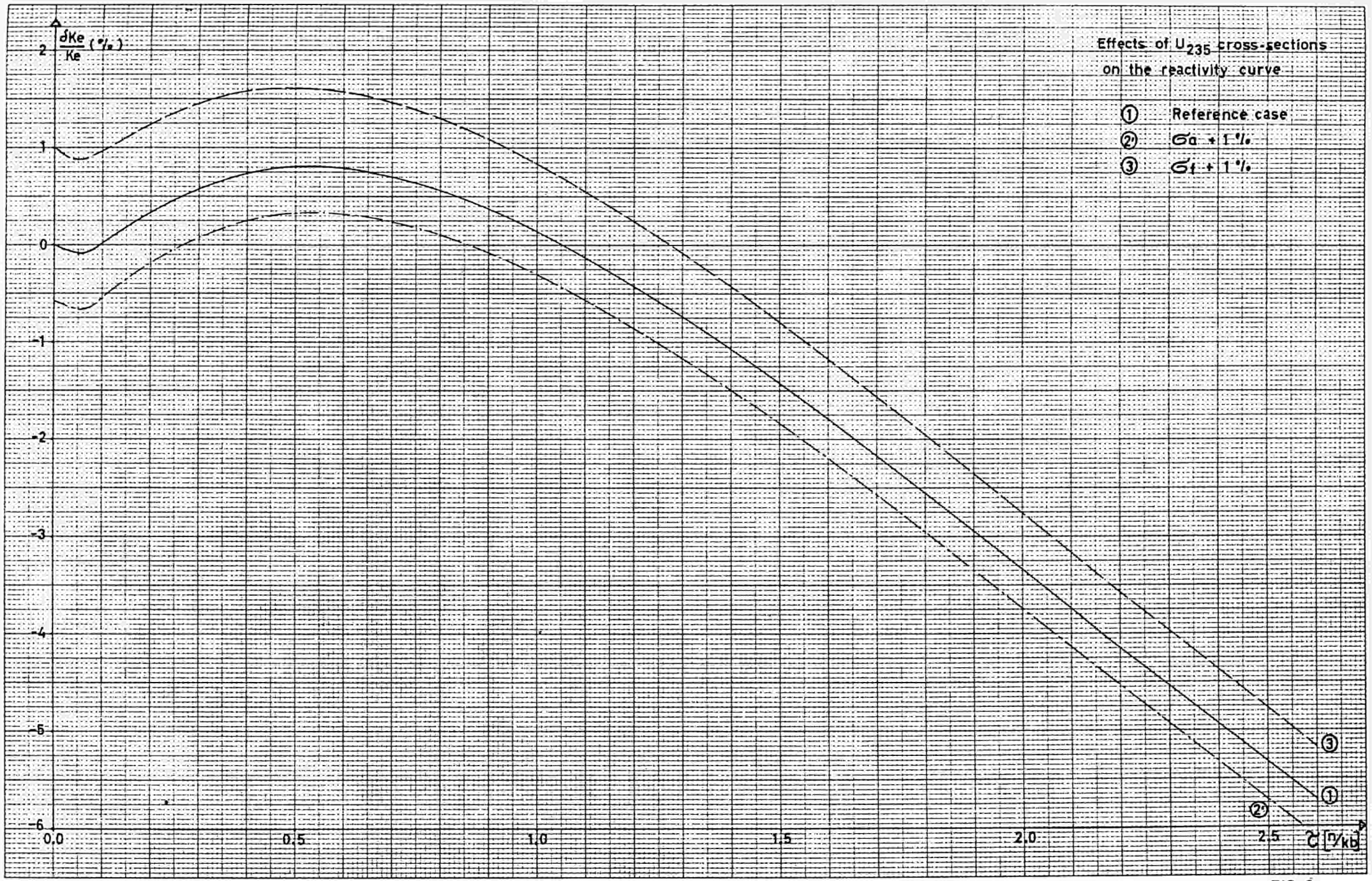
Effects of U_{235} cross-sections on the reactivity curve
① Reference case
② σ_a + 1 %
③ σ_f + 1 %
$\frac{dKe}{Ke}$ (%)
2
1
0
-1
-2
-3
-4
-5
-6
0,0
0,5
1,0
1,5
2,0
2,5
τ [n/kb]

FIG. 3

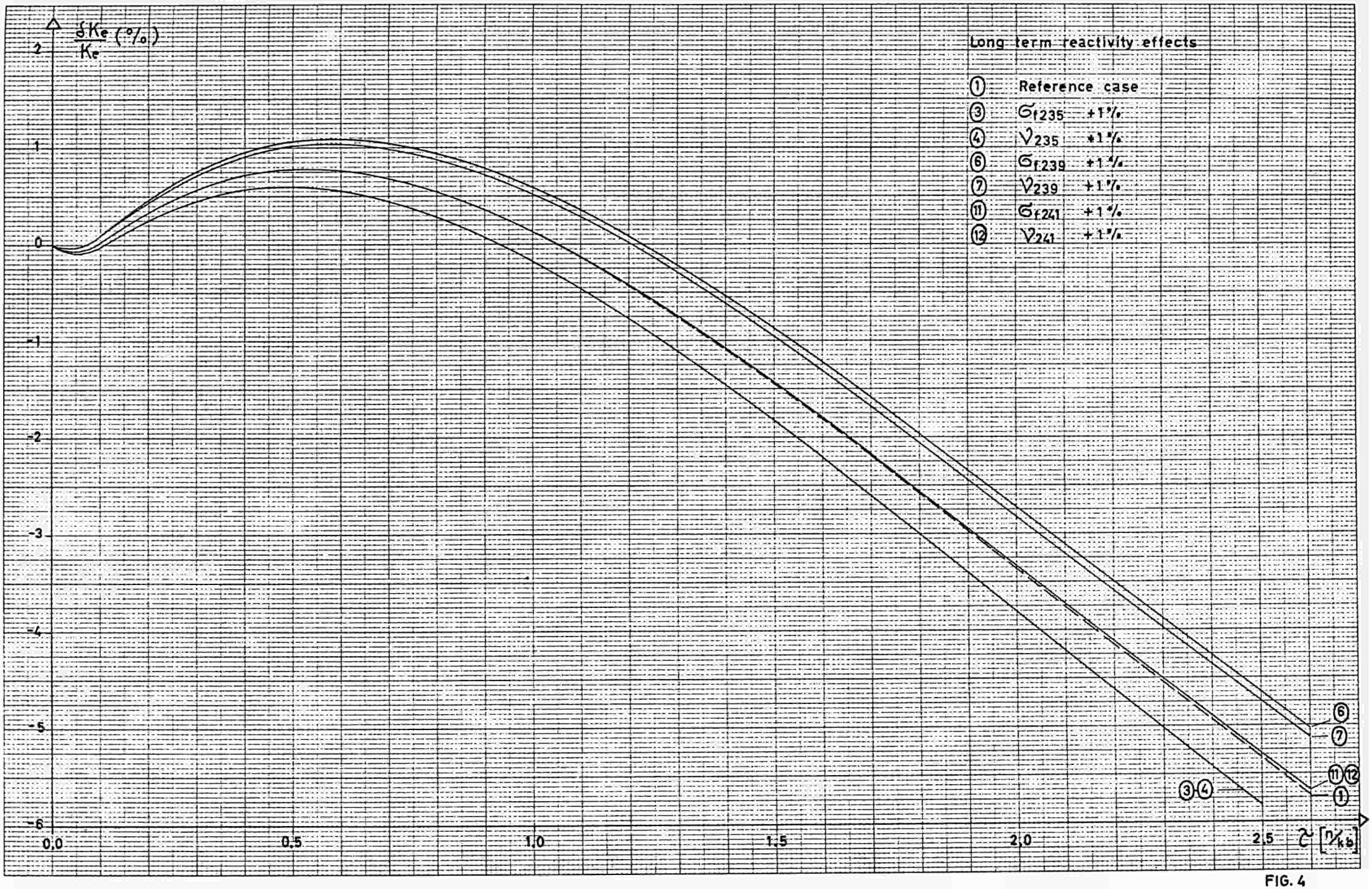
$\frac{\delta K_e}{K_e}$ (%)
2
1
0
-1
-2
-3
-4
-5
-6
0,0
0,5
1,0
1,5
2,0
2,5
τ [n/kb]
Long term reactivity effects
① Reference case
③ σ_{f235} +1 %
④ ν_{235} +1 %
⑥ σ_{f239} +1 %
⑦ ν_{239} +1 %
⑪ σ_{f241} +1 %
⑫ ν_{241} +1 %
⑥
⑦
⑪⑫
①
③④

FIG. 4

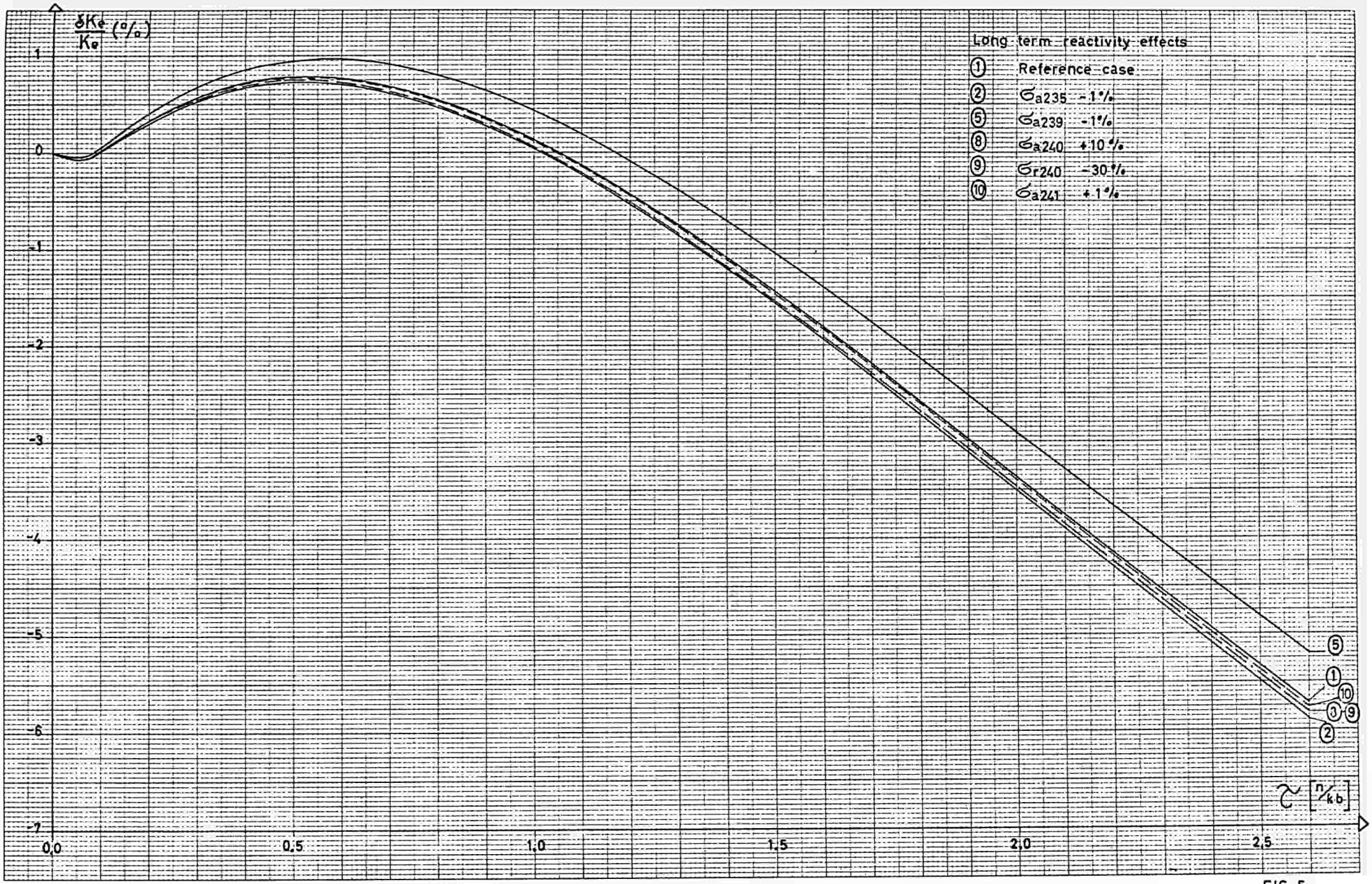
δKe/Ke (%)
1
0
-1
-2
-3
-4
-5
-6
-7
0,0
0,5
1,0
1,5
2,0
2,5
τ [n/kb]
Long term reactivity effects
① Reference case
② σa235 -1%
⑤ σa239 -1%
⑧ σa240 +10 %
⑨ σr240 -30 %
⑩ σa241 +1 %

FIG. 5

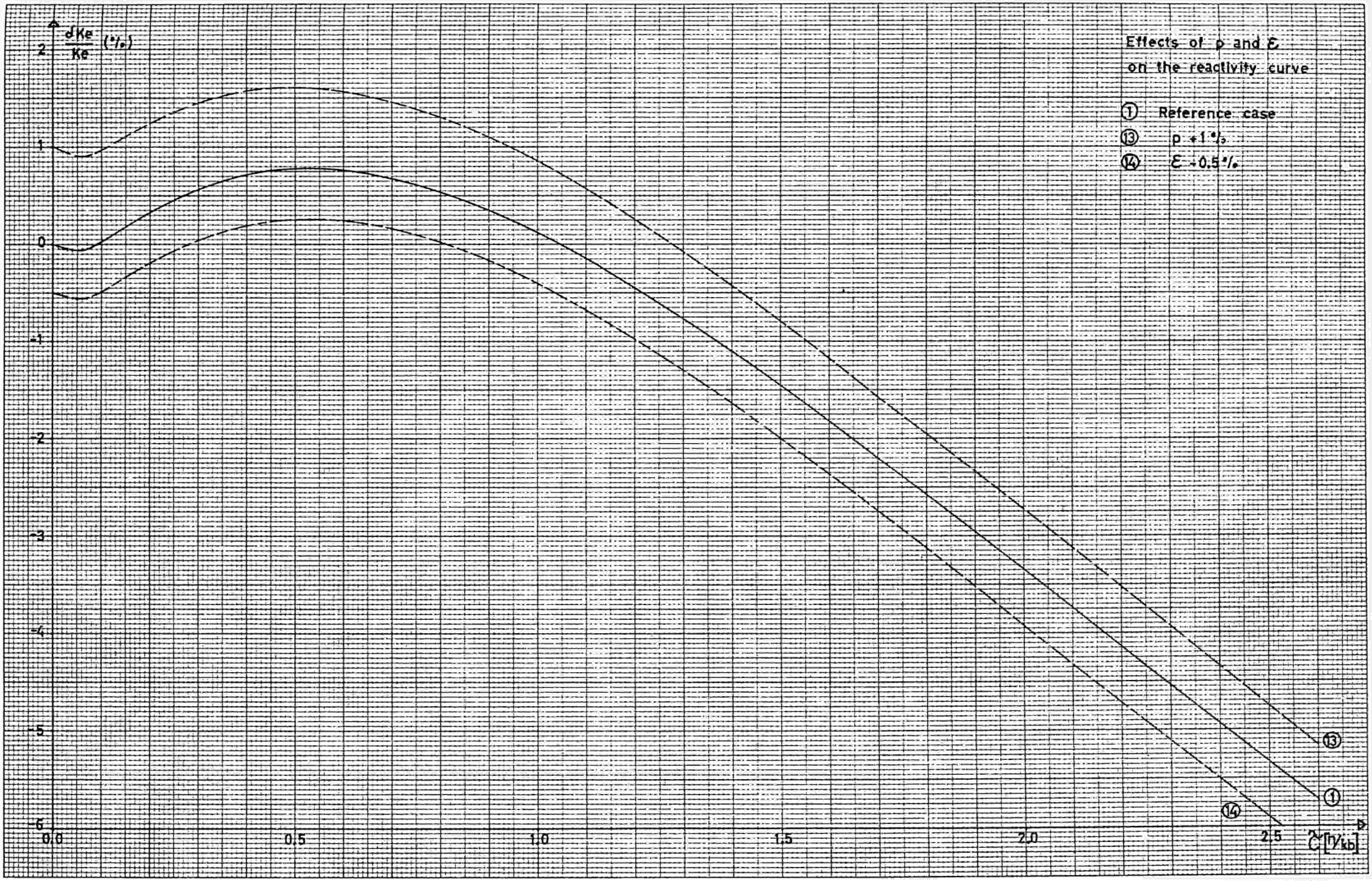
$\frac{dKe}{Ke}$ (%)
2
1
0
-1
-2
-3
-4
-5
-6
0.0
0.5
1.0
1.5
2.0
2.5
C [γ/kb]
Effects of ρ and ε on the reactivity curve
① Reference case
⑬ ρ +1 %
⑭ ε -0.5 %

FIG. 6

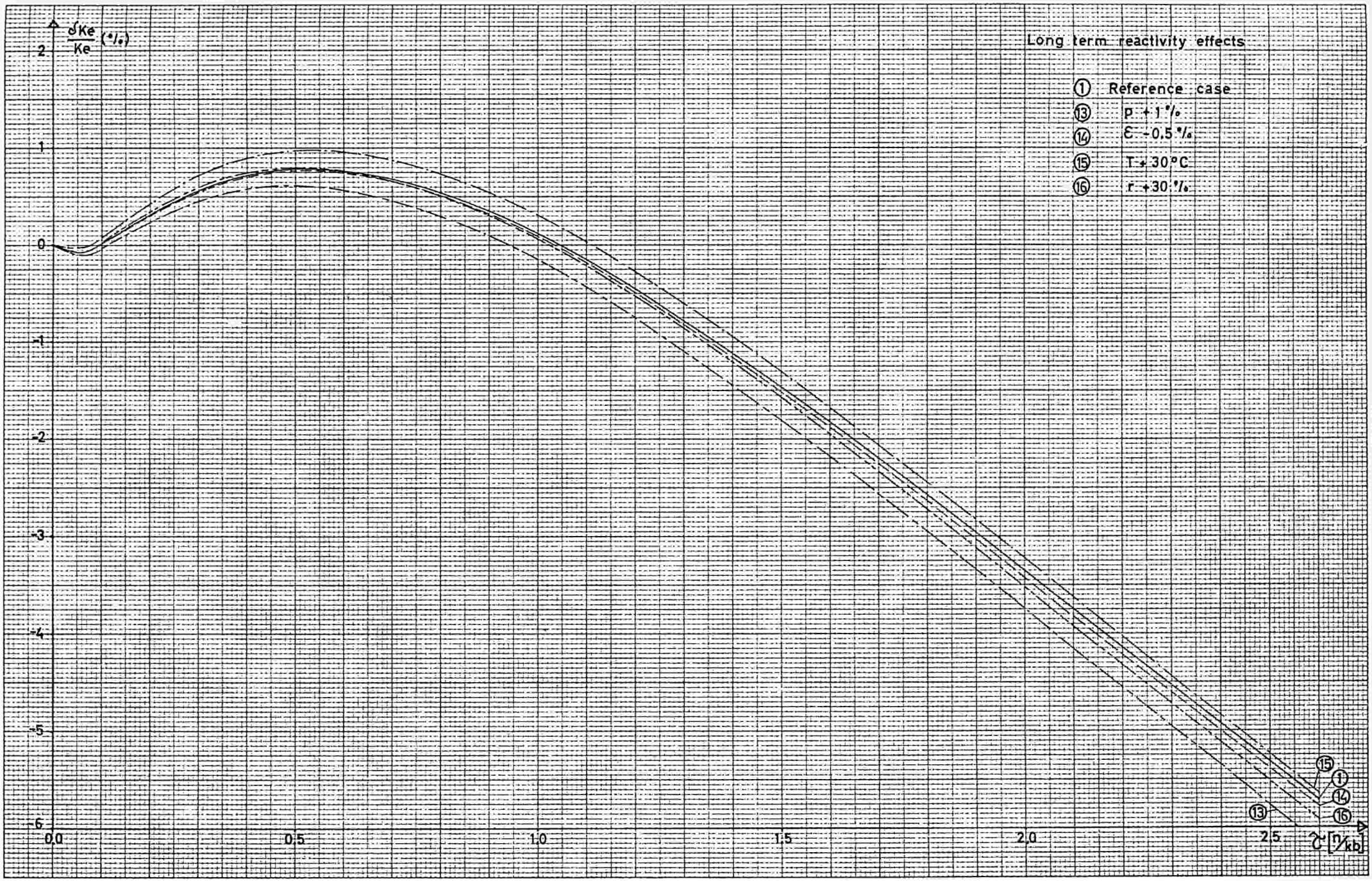
Long term reactivity effects
① Reference case
⑬ p +1 %
⑭ ε -0,5 %
⑮ T +30°C
⑯ r +30 %
δKe/Ke (%)
2
1
0
-1
-2
-3
-4
-5
-6
0,0
0,5
1,0
1,5
2,0
2,5
τ [D/kb]

FIG. 7

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