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OPTIONS FOR A 250 MWe ORGEL PROTOTYPE PLANT ORIENTATION STUDY

by

A. DECRESSIN, F. LAFONTAINE, J. NOAILLY and P. TAUCH

1969

ORGEL Program

Joint Nuclear Research Center
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The three reactors have been optimized taking into account a certain number of technological restrictions which are due to the actual state of development of the ORGEL string.

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KEYWORDS

ORGEL REACTOR
POWER PLANTS
NATURAL URANIUM FUEL
URANIUM
ENRICHED MATERIALS
URANIUM CARBIDES
ORGANIC COOLANT
OPTIMIZATION
DESIGN

CONTENTS

PART I

Options for a prototype

I.1. Introduction

Many choices are open when the engineer is designing the core of a 250 MWe ORGEL prototype. Among them, one presents in this report the main features of three reactors as illustration of three different tendencies:

- a natural-uranium reactor (reactor A);
- a high-performance reactor (reactor B);
- a high-performance and intrinsically stable reactor (reactor C).

These three reactors have been optimized taking into account a certain number of restrictions which are due to the actual state of development of the ORGEL string.

- The three reactors are equipped with a G-18 fuel element of the structured type (central rod); it is on that type of fuel element that the development effort is actually focused.

- The coolant channel is of the gas-insulated type with SAP pressure tube. An exchange with zirconium-based pressure tube would not alter the main characteristics of these three reactors.

The design of the cell (fuel element and channel assembly) of these three reactors has been done in such a way as to render the three solutions technologically as similar as possible. Nevertheless, the fuel element performances

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(*) Manuscript received on 14 January 1969.

differ strongly by the couple of values relating to conductivity integral and burn-up, which however are well within the technological limits of the fuel (see Fig. 0, extracted from the prototype tender).

These three reactors have many common features: vertical axis, on-load bidirectional fuel cycle, OM-2 organic cooling, extraction of high boilers by distillation, etc., which are given with their general characteristics in the tables of Chapter 1.5.

1.2. Natural-uranium reactor (reactor A)

The ORGEL reactor is able to burn natural uranium. Earlier studies seem to demonstrate that, in such a case, the ideal fuel is not represented by a G-18 fuel bundle [1]. Nevertheless, for the reason mentioned in I.1., this fuel type has been adopted for the study.

The fuel has been designed in order to minimize the quantity of absorbing material.

A maximum conductivity integral of 40 W/cm coupled with a pin diameter of about 1.5 cm (fuel section between 30 and 35 cm2) consents a maximum specific power of about 21 W/g, maintaining the coolant section/fuel section ratio per channel at a non-too-high figure, neighbouring 0.5.

The fuel section has been optimized on the strength of available experimental reactor physics data. The optimum

---

(1) Although the G-18 fuel bundle does not represent the optimum solution for the natural uranium concept, it represents the best compromise if one intends to develop the natural and enriched ORGEL strings. This is the reason why the development effort is, to the present, focussed on it.

moderation ratio is nearly equal to 18.

The main characteristics of this reactor A are:

- Heavy water investment: 1 T/MWe
- Average burn-up: 7,000 MWd/T
- Net power plant efficiency: 30%
- (Coolant outlet temperature in the neighbourhood of 360°C)

The result is a relatively high cost of electric power (fixed costs of about 0,90 mills/kWh due to direct and indirect investment in heavy water[1] and fuel consumption costs (see Table 1) of about 1.5 mills/kWh) (no fuel reprocessing).

It has to be noted that very similar performances are obtainable in a CANDU reactor or in an ORGEL reactor fed with Zirconium-alloy cladded UO_2 fuel.

Indeed, on one hand neutronic calculations indicate that in an A-type reactor the substitution of uranium carbide by uranium oxide and of SAP by Zr-2,5 Nb determines a 2,000 MWd/T loss in burn-up; on the other hand, WR-I UO_2/Zr-2,5 Nb reference fuel operates with a conductivity integral of 36 W/cm having attained a 10,000 MWd/T burn-up at the end of 1967. Finally, admitting a \$60/kg fuel cost[2], it appears that a burn-up of only about 5,500 MWd/T is sufficient to determine fuel consumption costs in the order of 1.5 mills/kWh.

To improve reactor performance, there is only one possibility: enrich fuel in order to increase burn-up and/or diminish investment.

---

(1) Heavy water: \$20/lb; indirect investment estimated at 35% of direct investment.

(2) See AECL 2534, "Trends in Atomic Power Costs".

Should fuel be enriched with the sole purpose of increasing burn-up, these would be the results:

TABLE 1

Enrichment	wt % - U-235	0,71	0,93	1,14
Burn-up	MWd/T	7,000	14,000	20,500
Fuel cost	$/kg	75	100	120
Fuel consumption	mills/kWh	1.47	0.97	0.80

The reduction in fuel consumption costs is considerable; it is of interest to note that it can also be achieved with UO_2-fuel clad in zirconium alloy.

Fuel enrichment may be envisaged in order to lower substantially investment. Then, it is necessary to increase strongly the heat output per channel and hence the linear heat rating of the fuel rods, if one sticks to the G-18 fuel element concept(1).

To minimize absorbing material in the cell is no longer a goal; on the contrary, it is necessary to fin the fuel cladding and to increase the coolant section of the fuel element in order to extract as much heat as possible.

In this run to high linear heat ratings, one parameter remains free: the diameter of the fuel rods. Two fuel elements, each one designed for a maximum conductivity integral of 90 W/cm and with identical thermomechanical stresses, are illustrating this freedom (Table 2): the first with the same fuel rods as those of the A reactor, the second with thicker fuel rods.

---

(1) Such a thing seems difficult with UO_2 fuel, so one resorts to a SAP-clad UC fuel.

TABLE 2

Rod diameter (cm)	Coolant section to fuel section ratio	Fuel section (cm2)
1.5	1	33
1.8	0.65	47.5

1.3. High-performance reactor (reactor B)

The concept of this reactor is both to minimize heavy water and fuel inventory and to maximize burn-up. This determines a choice of pin-diameter of about 1.5 cm - identical as with station A - coupled with a 90 W/cm conductivity integral (hence an organic/fuel ratio = 1) and a heavy water/fuel ratio in the neighbourhood of 16.

In such conditions, it is possible to reduce heavy water investment to 0.46 T/MWe and fuel investment to 0.17 T/MWe and yet obtain 15,000 MWd/T burn-up; fixed costs pertaining to direct and indirect heavy water investment thus attain 0.4 mills/kWh, and fuel consumption costs 1.1 mills/kWh, a definite improvement over natural uranium.

The counterpart consists in:

1) A close vicinity between the fuel performances and the theoretical technological limits of UC-SAP fuel (Fig. 0 - see also doc. 2 in the prototype tender ledger).

2) A greater amount of energy deposited in the organic coolant: 0.55% instead of 0,37% of total fission energy.

3) A slight fuel enrichment of about 1.14% in U-235.

4) The impossibility of operating this reactor with natural uranium.

1.4. High-performance and intrinsically stable reactor (reactor C)

The concept of this reactor is to obtain inherent stability and yet preserve to the utmost the economic features of a high-performance reactor.

For this purpose, as compared with reactor A, one has increased, not only the linear heat rating of the fuel rods, but also the diameter of these in order to minimize the organic-to-fuel ratio(1) (the de-stabilizing effect is essentially due to the presence of organic), while the moderation ratio has been reduced in a very marked measure (down to value 9) in order to increase the organic's moderating power in regard of that of heavy water, and likewise resonance absorption(2) and thus the Doppler effect. According to the latest buckling measurements, it appears that ensuing reactivity loss is relatively low for such large fuel sections ($\approx$ 50 cm2) /Ref. 1/.

Finally, a preliminary design-study shows that a moderation ratio as low as 9 can be obtained with clusters having a fuel section of about 50 cm2, but that this would not be possible with smaller fuel sections.

A fuel section greater than 50 cm2 was disregarded here:

1) because buckling measurements for such large sections are not yet available;

2) because the diameter of SAP pressure tubes would be too much above that of tubes now in production;

3) because an upper 90 W/cm-limit has been set for the conductivity integral (due to the lack of experiments);

---

(1) In reactor B instead, pin-diameter has been maintained constant, thus allowing an increase in specific fuel power but with an ensuing deterioration in the organic/fuel ratio.

(2) The increase in resonance absorption implies an increase in the initial conversion factor - from 0.6 to 0.7 - and thus explains the economic equivalence in the plant B and C fuel cycles. See further on.

4) because of a limit of 18 (or 19) has been set for the number of rods per fuel cluster.

As opposed to reactor B, this reactor offers the following advantages: a -2.2 pcm/% power negative power coefficient instead of a positive +1 pcm/% power[1], a lower void coefficient at equilibrium, a slight decrease in heavy water investment (0.40 T/MWe instead of 0.46 T/MWe), a slight decrease of power absorbed by the organic coolant (0.47% of total fission power instead of 0.55%) and a slight increase of plutonium concentration in irradiated fuel.

On the other hand, there are the following disadvantages: slight increase in fuel investment (0.24 T/MWe instead of 0.17 T/MWe and a slight decrease in burn-up (13,300 MWd/T instead of 15,000 MWd/T).

These differences offset one another in terms of cost. Economically, these two reactors are nearly equivalent, with a slight advantage for reactor B.(see also footnote on page 32).

One should not however loose sight of the fact that, although reactor C is stable[2], it does not offer a wide safety margin in relation to its stability-limit (zero-power coefficient) and that, furthermore, the organic's temperature coefficient is positive. A detailed analysis is necessary in order to assess how the layout of the control equipment of reactor C can profit from the negative power coefficient.

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(1) Reactor A's power coefficient is also positive, slightly more than 1 pcm/% power.

(2) At least according to the temperature coefficients calculated by PLUTHARCO [Ref. 2].

As related to the moderation ratio (Vm/Vu), the temperature and power coefficients are the following:

TABLE 3

Vm/Vu	-	9	8	7
Organic-temperature coefficient	pcm/°C	+1.8	0	-1
Fuel-temperature coefficient	pcm/°C	-1.1	-1.2	-1.4
Power coefficient	pcm/% power	-2.2	-4	-5.2

This shows that the concept of a stable core has a definite sense only in the case of moderation ratios below 9. Matters being so, and should core stability be considered as an imperative, the choice of a moderation ratio between 8 and 7 becomes a necessity.

Remark

Reactors A, B and C can naturally be fed, either with more enriched fuel (A, B and C) or with less enriched fuel (B and C). The impact of such alterations on energy cost appears in figures 1, 2 and 3. Reactors B and C cannot operate on natural uranium. On the other hand, should plutonium-recycling be taken into consideration, reactors B and C could operate on natural uranium; in this case, fuels would be SAP-clad UC-PuC mixtures, which still require complete technological testing. Average burn-up of irradiated fuels would, in this case, be of about 10,000 MWd/T [Ref. 3].

I.5 PLANT DATA

		A	B	C
1. Power balance				
Power Plant				
Gross electric power	MWe	250	250	250
Net electric power	MWe	234	235	234
Net plant efficiency	-	0,301	0,297	0,311
Reactor				
Fission power Wf	MW	777	790	752
Thermal power losses to the moderator	MW	54	52	45
Thermal power to the coolant	MW	723	738	707
Specific power (referred to reactor vessel volume)	kW/l	3,5	7,5	8,3
Primary circuit				
Pumping power at nominal rating	MWe	8,5	7,7	8,5
Pumping power to the coolant	MW	6,5	5,9	6,5
Thermal losses	MW	3,6	3,7	3,5
Heat exchanger power	MW	726	740	710
Secondary and auxiliary circuits				
Pumping power (steam cycle)	MWe	4	3,7	4,3
Power consumption by all other auxiliary equipments	MWe	3,5	3,5	3,5
2. Reactor block (1)				
Reactor				
Axis		vertical	vertical	vertical
Core height	cm	480	400	400
Core radius	cm	311	215	200
Radius of the flattened zone	cm	125	92	113
Reflector thickness (D_2O) - axial	cm	30	30	30
radial	cm	50	50	50

(1) Given dimensions relate to : room temperature, unirradiated fuel, mean value in the range of tolerances

TABLE I (cont.)

		A	B	C
Number of channels	--	424	220	216
lattice pitch (square)	cm	26,8	25,8	24,2
Core fuel inventory	10^3 kg U	84,6	36	51,6
D_2O inventory	10^3 kg	221	99	84
Fuel element				
Number of elements per channel	--	6	5	5
Number of fuel rods per element	--	18	18	18
Over all length of the element	cm	80	80	80
Length of fuel core	cm	75,5	75,5	75,5
Diameter of the fuel pins	cm	1,524	1,524	1,830
Carbon contents in UC (wt %)	%	4,9	4,9	4,9
Cladding material	--	SAP	SAP	SAP
- Thickness (between fins)	cm	0,0762	0,0762	0,0915
Height of fins	cm	--	0,09	0,075
Finning ratio	--	1	1,9	1,75
Fuel cross section	cm^2	32,83	32,83	47,34
Cladding cross section	cm^2	7,3	14,5	14,1
Coolant cross section	cm^2	17,5	32,0	31,1
Ratio of coolant to fuel cross section	--	0,53	0,97	0,66
Fuel management	--	Bidirectional	Bidirectional	Bidirectional
Enrichment in U-235	wt.-%	0,71	1,14	1,14
Main coolant assemblies (1)				
Coolant		OM2	OM2	OM2
Pressure tube material		SAP	SAP	SAP
- inner diameter	cm	8,7	10,16	11,0
- thickness	cm	0,23	0,23	0,31
Gas for thermal insulation	--	CO_2	CO_2	CO_2
- pressure	kg/cm^2	7	7	7
- gap thickness	cm	0,5	0,5	0,5
calandria tube material	--	Zr-2	Zr-2	Zr-2
- internal diameter	cm	10,16	11,62	12,62
- thickness	cm	0,18	0,20	0,23

(1) The cooling channel is of pressurized gas insulated type.

TABLE I (cont.)

		A	B	C
3. Neutron Physics				
General Characteristics				
Ratio of moderator to fuel volume	-	18	16	9
Thermal neutron flux in core center (nv_o)	$10^{13} ncm^{-2} sec^{-1}$	7,26	10,1	6,1
Axial form factor of flux (1)	-	0,68	0,69	0,69
Radial form factor of flux (1,2)	-	0,73	0,78	0,86
Power form factor	-	0,40	0,43	0,47
Reactivity balance (clean, hot lattice)				
Infinitive multiplication factor (k_{oo})	-	1,099	1,250	1,202
Thermal utilisation (f)	-	0,920	0,911	0,928
Resonance escape probability (p)	-	0,891	0,888	0,836
Fast fission factor (ε)	-	1,038	1,034	1,042
Thermal fission factor (η)	-	1,292	1,494	1,487
Initial conversion ratio (C)	-	0,832	0,598	0,707
Slowing down area (L^2_s)	cm^2	155	110	113
Diffusion area (L^2_{th})	cm^2	209	148	102
Reserved reactivity for the fuel burn-up				
- inner zone	pcm	3800	14300	11000
- outer zone	pcm	2000	11800	7900
Characteristics of the equilibrium core				
Mean conversion ratio	-	0,77	0,66	0,73
Burn-up burnt fuel				
- inner zone	MWd/TU	8400	17400	15900
- outer zone	MWd/TU	6600	14300	12000
core averaged	MWd/TU	6900	14900	13200

(1) without flux disturbances by inserted control rods

(2) obtained by proper adjustment of the zonal burn-up

TABLE I (cont.)

		A	B	C
Characteristics of the burnt fuel (core averaged)				
Concentration of Pu 239	g/kg	2,22	2,22	2,61
Concentration of Pu 240	g/kg	1,19	1,80	1,99
Concentration of Pu 241	g/kg	0,21	0,33	0,41
Concentration of U 235	g/kg	2,21	2,07	2,62
Plutonium production	g/KWh	$0,7 \cdot 10^{-4}$	$0,4 \cdot 10^{-4}$	$0,5 \cdot 10^{-4}$
Uranium 235 consumption	g/KWh	$1,40 \cdot 10^{-4}$	$1,04 \cdot 10^{-4}$	$1,14 \cdot 10^{-4}$
4. Thermohydraulic performances				
Primary coolant system				
Coolant flow direction in the core		single pass - downwards		
Total coolant flow	t/hr	14200	14000	15300
Reactor inlet temperature	°C	283	275	293
Temperature rise (orificing)	°C	77	80	72
Reactor outlet temperature	°C	360	355	365
Coolant pressure				
- core inlet (1)	kg/cm^2	16	15	16
- core outlet	kg/cm^2	10	10	10
Total primary pumping power	MWe	8,5	7,7	8,5
Maximum coolant velocity (1)	m/s	10	10	10
Flow rate	kg/s	15,9	28,3	27,2
Moderator cooling system				
Total power transported	MW	54	52	45
Reactor inlet temperature	°C	50	50	50
Reactor outlet temperature	°C	80	80	80
Fuel performances				
Specific power (2)	W/g	21,2	46,7	28,2
Fuel rating ($\int k \, d\, \Theta$) (3)	W/cm	40	90	80

(1) The values given refer to the most charged channel

(2) Maximum values averaged over the fuel element section

(3) Maximum values averaged over the fuel rod

TABLE I (cont.)

		A	B	C
Heat flux clad-coolant (1)	W/cm^2	95	118	95
Max. clad surface temperature (without hot spots)	°C	425	425	425
Max. central fuel temperature	°C	800	1200	1100
5. Steam cycle with superheat and reheat				
Total power exchanged	MW	726	740	710
Number of loops	-	4	4	4
Pinch point (evaporator inlet)	°C	30	30	30
Pinch points (superheater, reheater, inlet)	°C	10	10	10
Steam pressure	kg/cm^2	48,5	42,5	56,0
Steam temperature	°C	350	345	355
Condenser pressure	kg/cm^2	0,044	0,044	0,044
Net steam cycle efficiency	-	0,338	0,332	0,345
6. Organic coolant				
Purification system	-	-	distillation	-
Fast neutrons energy deposition	% W_f	0,23	0,34	0,31
Gamma ray energy deposition	% W_f	0,14	0,21	0,16
Total gas production	Nm^3/hr	1,62	2,22	2,04
Contents of high boilers	%	5 %	5 %	5 %
Total make-up flow rate	kg/hr	127	186	162
Total purification flow rate	t/hr	2,6	3,7	3,2
7. Reactor stability				
Temperature coefficients				
- fuel	pcm/°C	- 0,4	- 0,3	- 1,1
- coolant	pcm/°C	+ 5,2	+ 5,0	+ 1,8
- moderator	pcm/°C	+ 7,7	+ 4,6	- 0,6
power coefficient	pcm/% power	+ 1,3	+ 1,0	- 2,2
Void coefficient (full drain of coolant)	pcm	2900	2700	2500 (2)
Inherent stability	-	unstable	unstable	stable

(1) Maximum value averaged over the fuel rod

(2) Extrapolated from experiments

TABLE I (cont.)

8, Economy of the plants				
Specific total investment (fuel excluded)	$/kWe net	285	240	238
Fixed costs plant investment	mills/kWh	4,07	3,43	3,40
Fixed costs fuel	mills/kWh	0,22	0,15	0,21
Total fixed costs	mills/kWh	4,29	3,58	3,61
Fuel consumption costs (no Pu-recovery)	mills/kWh	1,47	1,10	1,20
Organic consumption costs (at 0,3 $/kg)	mills/kWh	0,16	0,23	0,20
Heavy water losses	mills/kWh	0,03	0,01	0,01
Total electricity production costs (3)	mills/kWh	5,95	4,92	5,03

(3) Without insurances, operation and maintenance costs

PART II

Remarks with respect to the plant data of chapter I.5.

II.1. Power balance

The net electrical output is slightly different for the various plants and has been obtained by stipulating a fixed gross electric power of 250 MWe for all plants. The net plant efficiency is defined as the ratio of net electrical power to fission power. The power released in the heavy water [1] is not recuperated. The pumping power of the primary circuit is considered as being fully reconverted in thermal energy of the coolant assuming an efficiency of 0,76. The thermal losses of the primary circuits are estimated at 0,5% of the thermal reactor power.

II.2. Reactor block

Reactor dimensioning

The reactor dimensioning has been performed with aid of the Code ORION II [Ref. 4].

The starting point was the calculation of the power output of the central channel. This has been found by the stipulation that the fuel ratings of the maximum charged fuel rods have to reach given values.

The flux depression (disadvantage factor) has been calculated by the code CAPRI [Ref. 5] for the unirradiated elements. The radial neutron flux shapes with given flattening degrees are analytically derived from a one energy group diffusion theory dealing with two zones.

1) As hypothesis has been taken that no neutron and gamma-energy dissipates from the reactor tank.

The stipulated flattened radial flux distribution can only be obtained in reality if the flux is not perturbed by control rods, for instance. In the reactor dimensioning it has been accounted for that the control rods may perturb the flux to such an extent that the power form factor can decrease by 8% without loosing in power output.

The value of 8% has been estimated on the basis of earlier studies in which the control rods have been positionned at the core reflector interface.

Fuel element

In all cases A, B, C an 18 rod bundle has been chosen being technologically viable and adapted to the 3 plants from an economical point of view (see figure 4).
Economic power generation and stability criteria determine the rod diameters for the three cases as well as the ratio of coolant to fuel. Assembly of 6 (case A) and 5 (case B and C) bundles has been foreseen with the help of a hanger bar, taking the place of the 19th rod in the center of the bundles, so to provide a means to guide and align bundles in the channel, to simplify fuel handling and to discharge axial loads from fuel canning.

Even if the computations made - supported by some experiments - show that canning would withstand under axial load in the foreseen conditions, the low safety margin obtained suggests for the prototype an individual support for each bundle, fixed to the hanger bar. The supports possess positioning surfaces to center the assembly in the channel. Lenght of bundles (80 cm) has been fixed as a limit to prevent inacceptable vibrations of single rods, on the light of present out-of-pile experiments.

Canning thickness has been fixed to account for pressure difference between coolant and rod plenum for virgin and irradiated fuel and considering also ovality of canning itself. Proportionality between thickness and diameter has been chosen (5% of diameter).

Finning ratio has been fixed such as to lead to acceptable values of coolant and canning temperature and to prevent considerable temperature differentials in the circumference of the rod which would introduce important thermal bowing or, if prevented, significant thermal stresses. Computations were performed with thermal codes THESEE I and II [Ref. 6 and 7].

Main coolant assemblies (Cooling channel)

The cooling channel contains the fuel elements and the organic coolant and separates them from the moderator. It consists of the process tube, the insulation gas and the calandria tube.

The in-core part of the process tube is of SAP with 10% Al_2O_3. Steel extension tubes are attached to each end of the SAP tube by means of cold-rolled joints. The coolant enters through pigtails at the upper part of the channel and exits at the bottom of the channel. The top of the channel contains a sealing plug, which is removable for the on-power refueling. The lower extension tube is provided with a bellow, allowing for the thermal expension of the process tube.

The insulation gas is slowly circulating CO_2, which is pressurized to about the coolant outlet pressure, in order to reduce the pressure difference acting on the SAP tube. The calandria tube is of Zr-2.

The SAP tube thickness has been calculated with the design stresses as specified in chapter 3 of document 2 in the prototype tender ledgers.

For the plant C, having the biggest channels and the least lattice pitch, the space between adjacent channel extensions is rather small. So it is not possible to dispose feeder tubes of sufficient diameter to contain the entire cooling flow of a channel. Each channel is therefore provided with two inlet feeders and two outlet feeders. For the plants A and B this is not necessary.

II.3. Neutron physics

Introduction

The neutron physics calculations have been evaluated by the codes : PLUTHARCO, CAPRI and RLT-2 [Ref. 8] developed by Euratom and EQUIPOISE-3 [Ref. 9], GAM-I [Ref. 10] and WDSN [Ref. 11] developed by others.

The PLUTHARCO code has been tested against various experiments at Ispra [Ref. 12]. Experiments with hot lattices will be performed in the beginning of 1969.

The RLT-2 code has been tested against Canadian experiments [Ref. 13]. Oscillation measurements with fuel of various isotopic concentrations of uranium and plutonium in order to simulate burn-up states are being executed actually in the ECO critical facility.

All the results reported deal with possible equilibrium cores having two zones of various burn-up in order to flatten the inner core region. The fuel cycle always applied was of bidirectional type.

Reactivity balance

The clean, hot cell parameters of the infinite lattice have been calculated by means of the code "PLUTHARCO".

As input data for the physical temperatures of the cell materials the following values have been adopted for all three reactors; fuel 660°C, coolant 330°C, moderator 65°C. These values were calculated as core averaged values for the reactor A. The thermal flux distribution in the fuel cluster has been obtained by means of the code CAPRI.

The reactivity of the reactor zones is evaluated on the basis of the modified one group theory with aid of the zonal bucklings calculated by the dimensioning procedure. The reactivity which can be reserved for the fuel burn-up has been obtained after having made reactivity reservations for the xenon poisoning effect, for control purposes and for reactivity losses by joints.

Fuel burn-up

The fuel burn-up is calculated by the code RLT-2 which is based on the modified one group model. It calculates the concentrations of the fuel and fission products during fuel burn-up in a one-point model. The reactivity evolution of the finite reactor is evaluated by means of a perturbation method. In its fuel management part the code deals with the continuous bidirectional fuelling.

The reactivity evolutions of the three reactors are shown in fig. 5.

Equilibrium core

The fuel of the equilibrium core has been considered to have the isotopic concentrations according to the mean burn-up of the zones.

In the figures 6 and 7 are given the macroscopic radial flux distributions of the equilibrium cores. They have been calculated with EQUIPOISE-3, a two dimensional two groups diffusion code.

The reactor assembly has been considered in the calculations as consisting of four regions of constant group parameters : the two homogenized core regions of various mean burn-up and the radial and axial reflector regions.

The cell parameters have been provided by PLUTHARCO fed with the equilibrium fuel compositions.

The mean temperature coefficients of the 3 reactor cores have been evaluated as follows : cell parameters for various temperatures of fuel, coolant and moderator are calculated by PLUTHARCO and then inserted into EQUIPOISE-3 for a reactivity search. The difference in reactivity against the reference core defines the temperature dependence of the cores. For evaluating independently the various temperature coefficients the fuel temperatures have been changed by $\pm$ 100°C, the coolant temperature by $\pm$ 50°C and the moderator temperature by $\pm$ 15°C.

The void coefficient calculations have been performed in a similar way : lattice parameters are calculated for a voided cell by PLUTHARCO and are then introduced in the diffusion code. It must be noted that PLUTHARCO is not adapted to treat unisotropic lattices. But as the coolant volume is in general relatively small in comparison with the cell volume the streaming effects will not influence very much the results.

An extrapolation of the reactivity coefficients given here for the equilibrium core to a possible fresh core has shown the following in earlier studies : the fuel coefficients become more negative, the other positive coefficients of coolant and moderator become negative or nearly zero; the void coefficients increase.

Concerning the void coefficient buckling measurements performed with voided natural uranium elements have been compared with PLUTHARCO calculations. The results show that this code underestimates always the voiding effect, especially for lattices of low moderating ratio. The values quoted in the tables of chapter I.5 have been corrected accordingly for this effect.

Energy release in the core by neutron and gamma radiation

The kinetic neutron energy per fission is set equal to 4,8 MeV. After correcting for the neutron leakage the kinetic energy of the neutrons is considered to be released by elastic and inelastic scattering according to the reaction rates in the various core materials. The necessary cross sections have been calculated by means of the GAM-1 code for the homogenized core. The spatial flux distributions (disadvantage factors) have been provided by a WDSN calculation.

The gamma ray spectrum has been calculated for 8 energy groups. After having corrected for the gamma ray leakage the energy has been distributed to the various core materials according to their attenuation coefficients assuming the reactor to be a homogeneous one. The homogeneous absorption yields are then corrected by selfshielding factors. These are obtained assuming the same subdivision in two regions as for the neutron energy absorption and considering the homogeneous channel zone as a uniform source of gamma rays.

II.4. Thermohydraulic performances

Primary coolant system

The organic coolant circulates from top to bottom of the reactor through the main coolant assemblies (coolant channels) in a single pass. The upper and lower extension of each channel is connected by individual pipes to a torical header situated above and below the core, serving at the same time as collector and mixing tank. The manifolds of all channels situated in a 90° section of the core are grouped together.

Four main coolant loops (in carbon steel) connect the headers with the four external steam circuits penetrating the biological shielding of the reactor block. Each loop is fitted with two motor-operated isolating valves in each duct (inside and outside the reactor containment). The steam generator (drum boiler type) and one main coolant pump in the cold leg complete the coolant loop.

The steam circuits are connected to the torical headers such as to assure a uniform cooling of the core also in case of a pump failure in one main coolant circuit. All organic piping is fitted with steam trace heating.

The maximum coolant velocity inside the coolant channels is fixed at 10 m/s; by appropriate orificing an equal heating of the coolant in all channels is assured. The primary coolant system pressure is chosen sufficiently high as to prevent local boiling in the peripheric channels.

Moderator coolant system

The fission energy released in the moderator and thermal losses from the channels are evacuated by the moderator coolant system consisting of three loops, completely in stainless steel, the main pumps are of the canned rotor type. The loops are directly cooled by raw water.

The heating of the heavy water inside the reactor vessel is chosen relatively high (30°) in order to decrease the volume of D_2O in the loops and to reduce the investments. The moderator in the reactor vessel is slightly pressurized (about 1.5 ata) to prevent any local boiling of the moderator. D_2O flow in the calandria vessel is upwards.

The moderator auxiliary systems are composed of a double D_2O purification system by ion-exchange, a catalytic recombination system and a helium purification system. All parts of these systems being in contact with D_2O are made of stainless steel.

Fuel performances

The maximum linear heat rating of the fuel rods has been selected according to the requirements of the envisaged reactor string (see also Part I).

For the natural uranium reactor string (type A) with it's restricted neutron balance it was necessary to find a compromise between an acceptable specific power of the fuel and a small ratio of coolant to fuel section.

The high performance string (type B), taking full advantage of the good thermal conductivity of the uranium carbide fuel, works with a high maximum linear heat rating. Conserving the same fuel rod diameter as for the string A the specific power of the fuel is more than doubled, thus minimizing fuel and heavy water inventory in the core and maximizing the burn-up by a slight enrichment.

The high performance string of intrinsic stability (type C) with it's large fuel section (and greater fuel rod diameter, as the basic option of a G-18 type element is conserved) holds a medium position between string A and B; the specific power of the fuel is about 30% higher than for A and the maximum linear heat rating exactly twice that of A. Consequently, the specific investments in fuel and heavy water are rather close to those of the B type string.

For all three strings the maximum allowable temperature of the SAP cladding of the fuel rods is fixed at 425°C (without accounting for hot spots). Assuming a mean contact resistance of 1°C cm W^{-1} between fuel and cladding the maximum fuel rod central temperatures do not exceed 1.200 °C.

II.5. Steam cycle

The steam cycle adopted is a classical one with superheat and reheat by the primary coolant and feedwater heating by extraction steam. The thermodynamic efficiency of such steam cycles has been evaluated under EURATOM contract, in a range of primary coolant temperatures and steam pressures being typical for an ORGEL power plant.

A four loop design is chosen with superheater and reheater arranged in series. The steam generator is of the drum boiler type.

The heat exchanger surface was calculated by the EURATOM internally developped code HDP - 1 [1]. As this code does not foresee steam reheating the surface was calculated for a steam cycle without reheat, but having the same steam pressure and pinch points at the evaporator inlet and superheater outlet as the reheat cycle. Moreover, this code does not take into account any pressure drop inside the steam generator, so the surface calculation was done with the turbine inlet pressure. The reheater surface was estimated to be equal to that of the superheater.

II.6. Organic coolant and coolant purification system

The organic coolant is a mixture of ortho-, meta-, and paraterphenyls (commercially known as OM-2) containing 5% high boiling residues (HB). The physical properties of this coolant and the reasons for it's choice are described in [Ref. 15].

In order to eliminate the decomposition products produced in the organic coolant under the influence of radiation and temperature in the reactor a purification system is provided based on the distillation process. A certain quantity of coolant is bypassed from the main coolant loops and continuously sent through the purification system, thus keeping the HB-content at the preset level.

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(1) Not available; an improved version HDP-2 is to be published soon [Ref. 14].

The total organic coolant decomposition rate takes into account the combined effects of radiolysis and radiopyrolysis. The first can be split up to into fast neutron degradation and gamma ray degradation. The ratio of $G_{neutron}$ to G_{γ} has been assumed to 4 according to recent results obtained at the CEN Grenoble.

The fast neutrons account for the greatest fraction of total decomposition rate (85%), the gamma rays contribute about 12% and the rest is due to radiopyrolysis. The radiolytic degradation is supposed to follow a second order law, for the radiopyrolysis a first order law is assumed. The radiopyrolysis degradation constant is strongly temperature dependent; so, at the relatively law level of organic coolant outlet temperature (360°C) the radiopyrolysis is of little importance. Only the organic coolant being at reactor outlet temperature is assumed to participate in radiopyrolysis.

II.7. Reactor stability

Method /Ref. 16/

An analytic study has been developped from the set of differential equations describing the kinetics and heat transfer of the reactor. This method, unlike a numerical calculation as the normally used analogue computation, presents the great advantage to give the physical laws of the different parameters for stability which is very useful when the matter is to compare the stability of several reactor types.

The mathematical tool is the servo-theory which assumes that the studied physical systems are described by sets of linear differential equations with constant coefficients : it is not the case for the kinetic equations and the method is valid only for small variations around a specific steady state.

The physical effects of the temperature coefficients have been translated in servo-loop feedback which modifies the elementary reactor transfer functions; account has been taken of the axial (for a cosine flux) and radial statistical weight of the temperatures. The steady state operation program of the reactor is one with a constant coolant flow and a constant coolant inlet temperature.

Results

Stability condition

The stability condition is defined by

$$-\frac{\alpha_c}{\alpha_u} < l_s$$

where : $l_s = 1 + \frac{8}{3}\,\frac{T_u - T_c}{\Delta T}$

α_c = coolant temperature coefficient

α_u = fuel temperature coefficient

T_u = average fuel temperature in the channel of medium power

T_c = average coolant temperature in the channel of medium power

ΔT = coolant temperature span

Reactor string	A	B	C
$I_{w/cm}$	40	90	80
α_u pcm/°C	- 0,4	- 0,3	- 1,1
α_c pcm/°C	+ 5,2	+ 5,0	+ 1,8
l_s	4,3	7,8	7,3
$-\frac{\alpha_c}{\alpha_u}$	13	17	1,6
Stability	Unstable	Unstable	Stable

Power coefficient

It is defined by the following equation

$$\alpha_w = 10^{-2} \frac{\Delta T}{2} (\alpha_c + \alpha_u l_s) \qquad \text{(pcm/\% power)}$$

Reactor String	A	B	C
α_w (pcm/%)	+ 1,3	+ 1,0	- 2,2

As an illustration (Fig. 8 and 9) are reported in the (α_c, α_u) plane the stability regions and the lines corresponding to variable power coefficients for the reactors B and C. These plots give an idea of the gain of stability when the uncertainty on the values of the temperature coefficients is taken into account.

II.8. Economy of the plants

Direct costs of construction

The direct investment for the 3 plants was calculated by adjusting the results of the 250 MWe Reference Design Study Contract carried out between January 1962 and 1963 by the firms Belgo-Nucléaire/Indatom/Siemens [Ref. 17]. The cost estimates established under this contract were based on the state of technological development in 1962 and had the aim of ascertaining the fixed charges for both a 250 MWe gross ORGEL prototype and an already industrially-mature ORGEL power plant of 250 MWe gross (tête de filière). The estimates given hereunder refer to the ORGEL plant "tête de filière".

In order to escalate these 1962 cost figures to 1966 figures, a rate of 2.4 % per year was assumed, totalling at 10% escalation in 4 years.

The direct construction costs include the reactor, it's primary circuits, steam generators, moderator and coolant, but not the first charge of fuel and the fuel reserve. In addition, they include site clearance, civil engineering work, auxiliary work, turbogenerator unit, electrical equipment and main step-up transformers, but not the land. Heavy water is estimated at 20 $/lb.

Indirect costs of construction

The indirect costs include engineering, overheads and administrative costs, interest during construction, possible price increases till putting in operation and contingencies [1].

(1) Taxes on capital and return on capital during construction are not included in view of the fact that their incidence, which depends on the tax system applicable in the country where construction takes place, may vary between 0 and 6% in the Community. No account is taken of customs duties, it being assumed that all equipment is supplied from within the Community.

The percentage of indirect costs (35%) is based on the results of the Symposium on Technical and Economical Problems for Proven-Type Reactors held in Venice in October 1963.

Fixed costs of plant investment

The fixed costs due to plant investment (annual instalment) include interest on money, amortization and taxes on revenues. The total annual instalment varies considerably in the countries of the European Community, between 8.1% (France), 10% (Italy, The Netherlands), and 13% (Belgium, Germany).

Interest rates are in general between 5.5 and 7%. Amortization rates are based on estimated lifetime between 20 and 30 years for proven reactor types; in Germany and The Netherlands plant lifetime is shorter for fiscal reasons. Thus the amortization rates in the community are related to plant lifetimes between 15 and 30 years. Taxes on revenues are also quite different ranging from exemption (France, The Netherlands) to 3% - 4% per year on the revenues (Belgium, Germany).

In this economical evaluation a medium value of 10% is chosen for annual instalment. The annual plant load factor is taken as 0.8 equal to 7000 h of full load operation per year.

Fuel cycle costs

Fuel cost

The cost of the raw materials is taken as 8 \$/lb U_3O_8 for the natural uranium fuel; for the enriched fuel cost is 61 \$/kg U as UF_6 following the USAEC price list from July 1962.(α = 1,14 %)

The costs of converting U_3O_8 or UF_6 into UC are assumed to 38 and 40 \$/kg U respectively for the natural and the enriched fuel.

Cost of SAP sheaths differ somewhat between the natural uranium fuel without fins (3 $/kg U) and the enriched fuel with finned sheaths (5 $/kg U).

The cost of cladding, assembly and inspection is taken at 13 $/kg U (natural uranium fuel) and 14 $/kg U respectively (enriched fuel).

Total costs of fuel elements [1]

enrichment (wt.% U-235)	0,71	0,93	1,14	1,35
costs ($/kg U)	75	100	120	140

Fuel cycle costs (equilibrium core)

In calculating the fixed costs of the equilibrium fuel cycle the half first-charge costs were amortized over the whole plant lifetime. The interest rate on spare fuel (equal to 10% of the fuel charge) is 6%. Annual instalment rate and plant load factor are the same as for the fixed costs of plant investments.

The variable costs of the fuel cycle (fuel consumption) have been calculated firstly without Pu-recovery from spent fuel, secondly with Pu-recovery. In the latter case, only the extracted Pu-isotopes are accounted as a credit (at 8 $/g Pu), the residual U-235 of the enriched fuel is not considered. Reprocessing and transportation costs are taken at 21 $/kg U.

Operation and maintenance costs

The costs include only organic make-up costs and D_2O-losses. All other operation, maintenance and insurance costs are omitted because of their dependance on local conditions and the operation strategy of the plant owner.

---

1) The slight variation of fuel costs with fuel rod diameter has been neglected. Consideration of this effect would give reactor C a small advantage in cost over reactor B.

Organic coolant make-up costs are calculated for an equilibrium content of 5% HB to be **maintained by distillation.** Cost of fresh coolant is 0.30 $/kg.

Comparisons of D_2O losses for operating heavy water moderated and cooled reactors gave figures between 0.2 and 9% per year of total D_2O inventory.

For all these figures it is stressed out that the losses occured mainly during the periods of pressurised primary D_2O circuit.

For an ORGEL type reactor being only moderated by D_2O without high operating pressures one can reasonably assume yearly D_2O losses at about 0,5% of total inventory.

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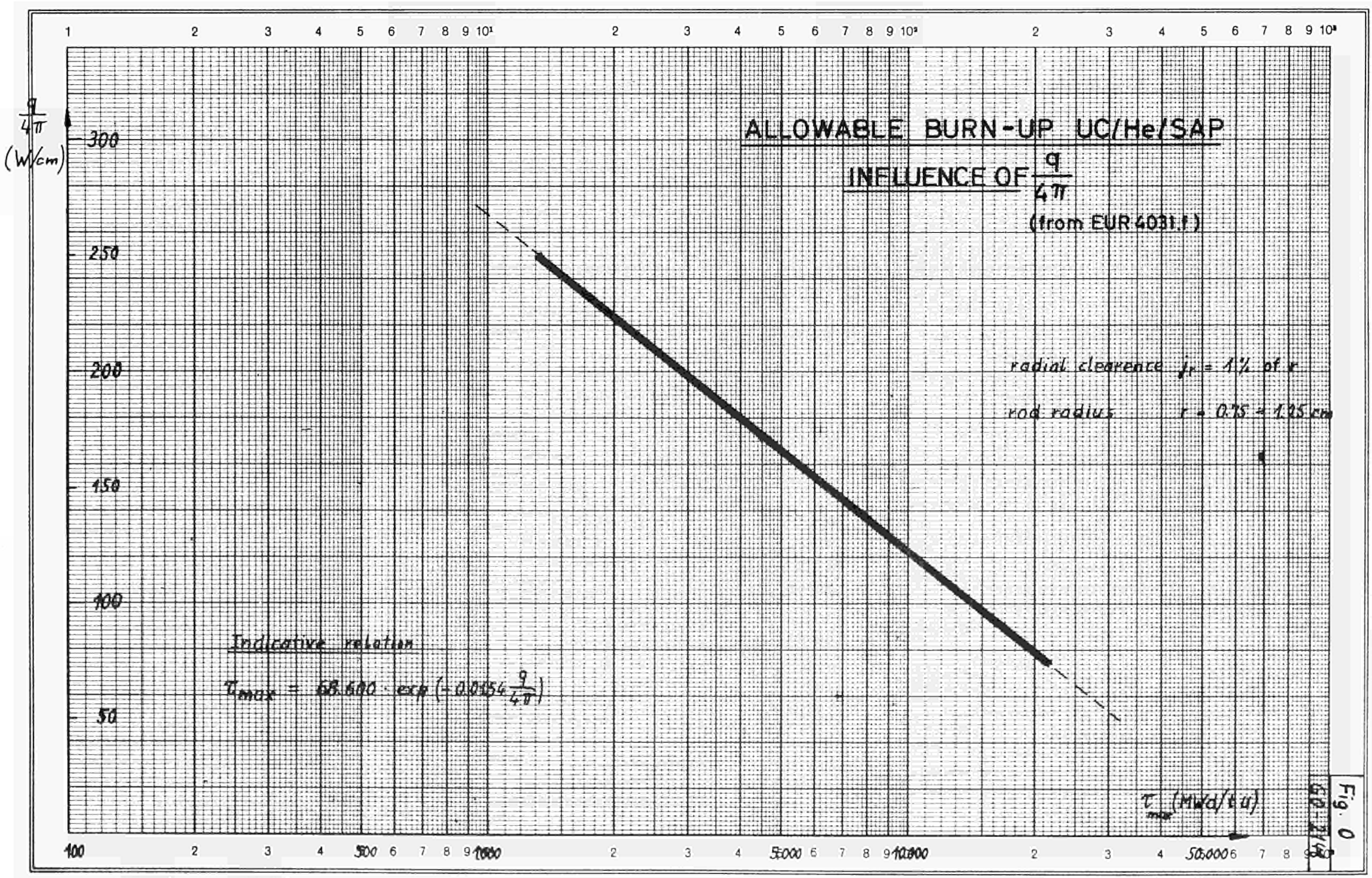

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Fig. 1
GO - 1977

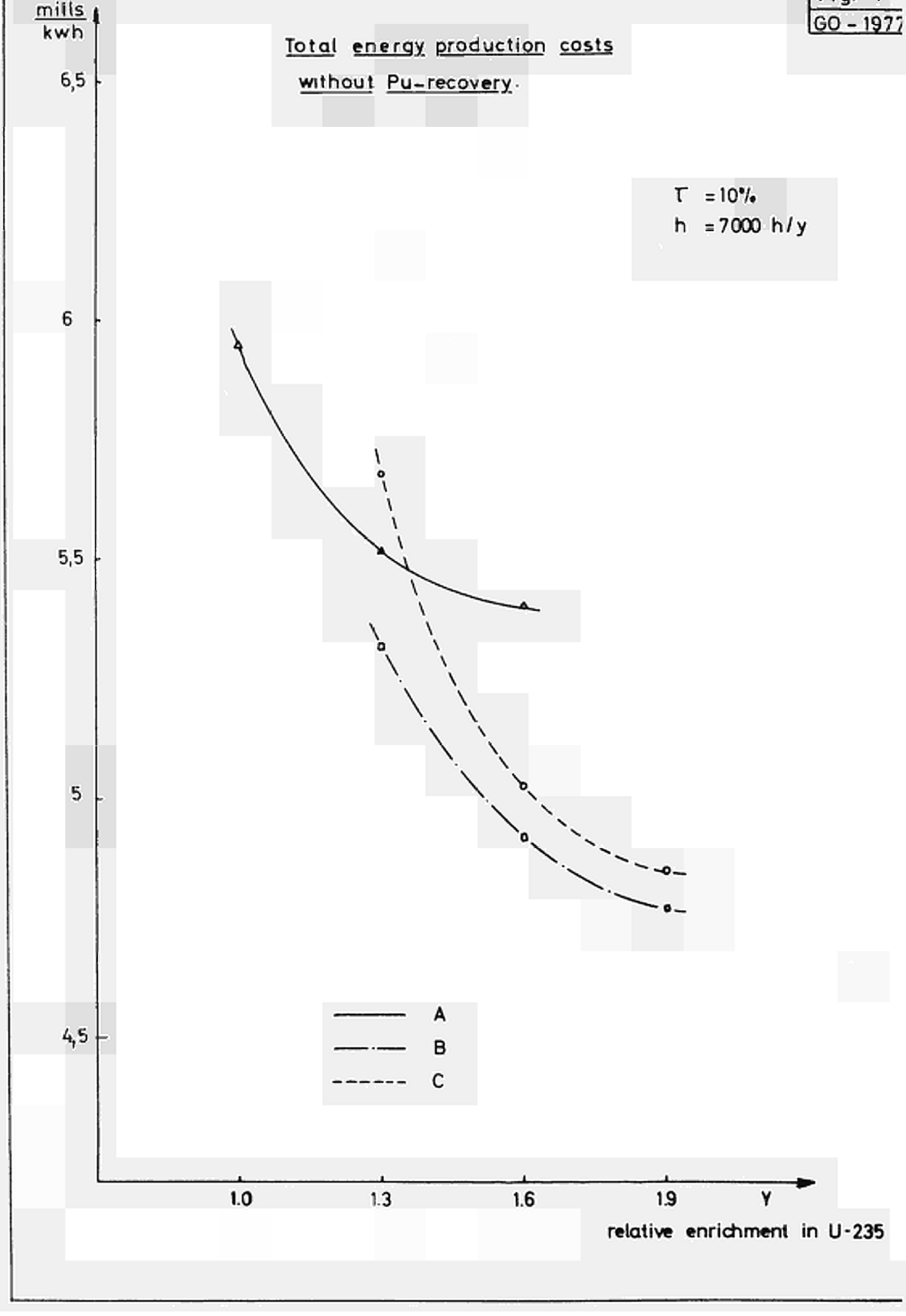

mills
kwh
6,5
6
5,5
5
4,5
Total energy production costs
without Pu-recovery.
T = 10%
h = 7000 h/y
A
B
C
1.0
1.3
1.6
1.9
Y
relative enrichment in U-235

Fig. 2

GO -1978

Total energy production costs with Pu-recovery.

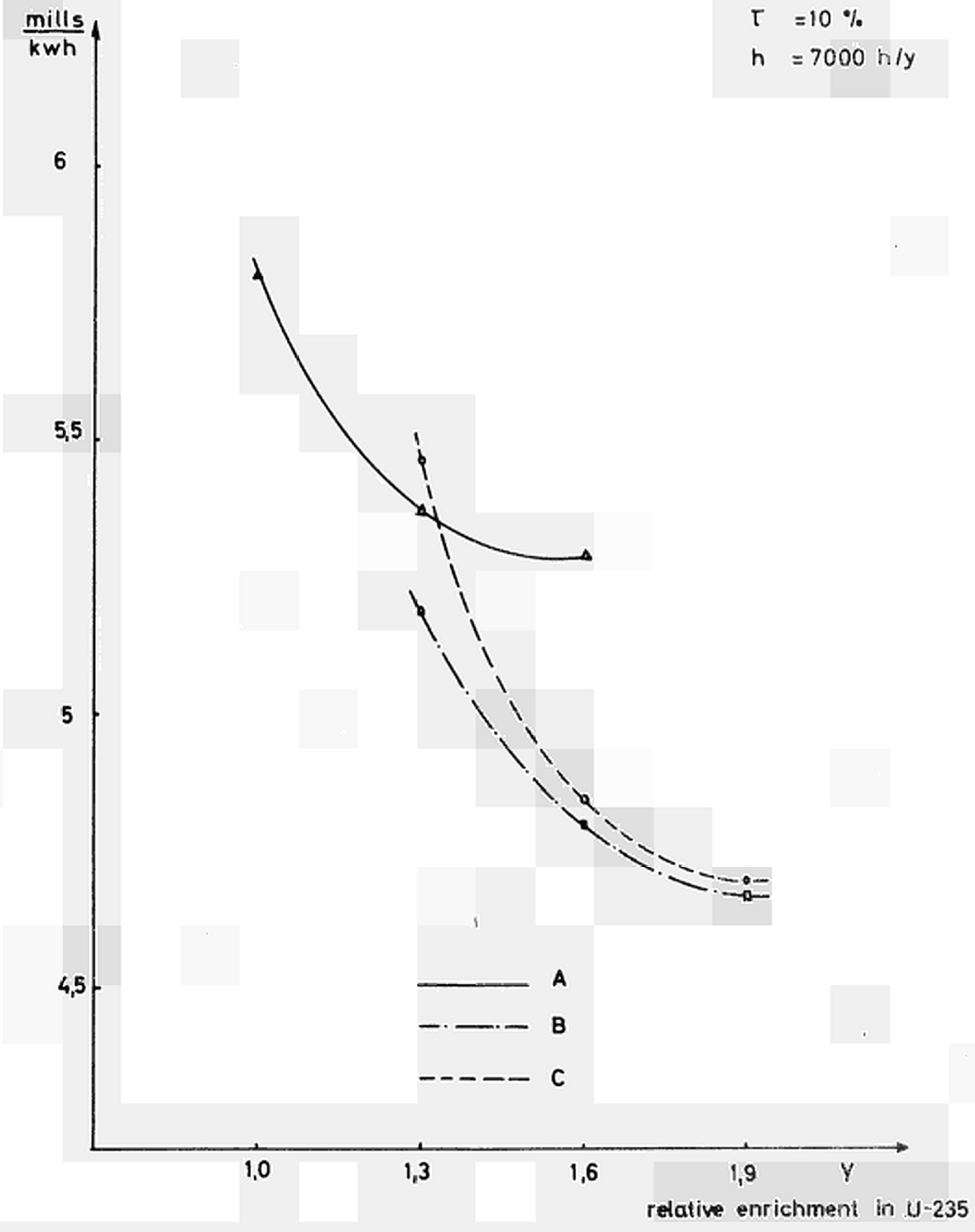


Fig. 3
GO – 1979

Fixed and variable fuelcycle cost

upper curves without Pu credit
lower curves with Pu credit

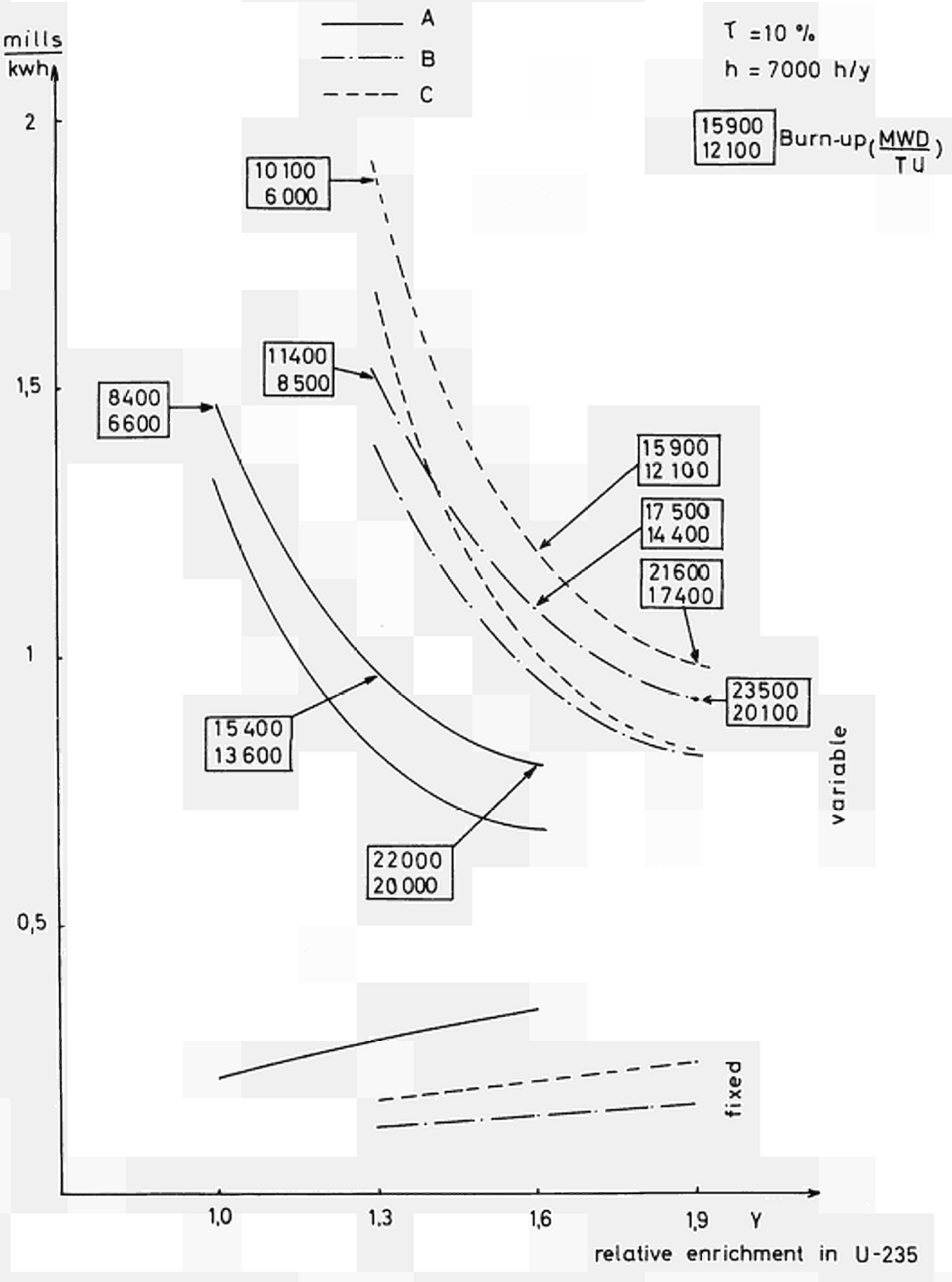
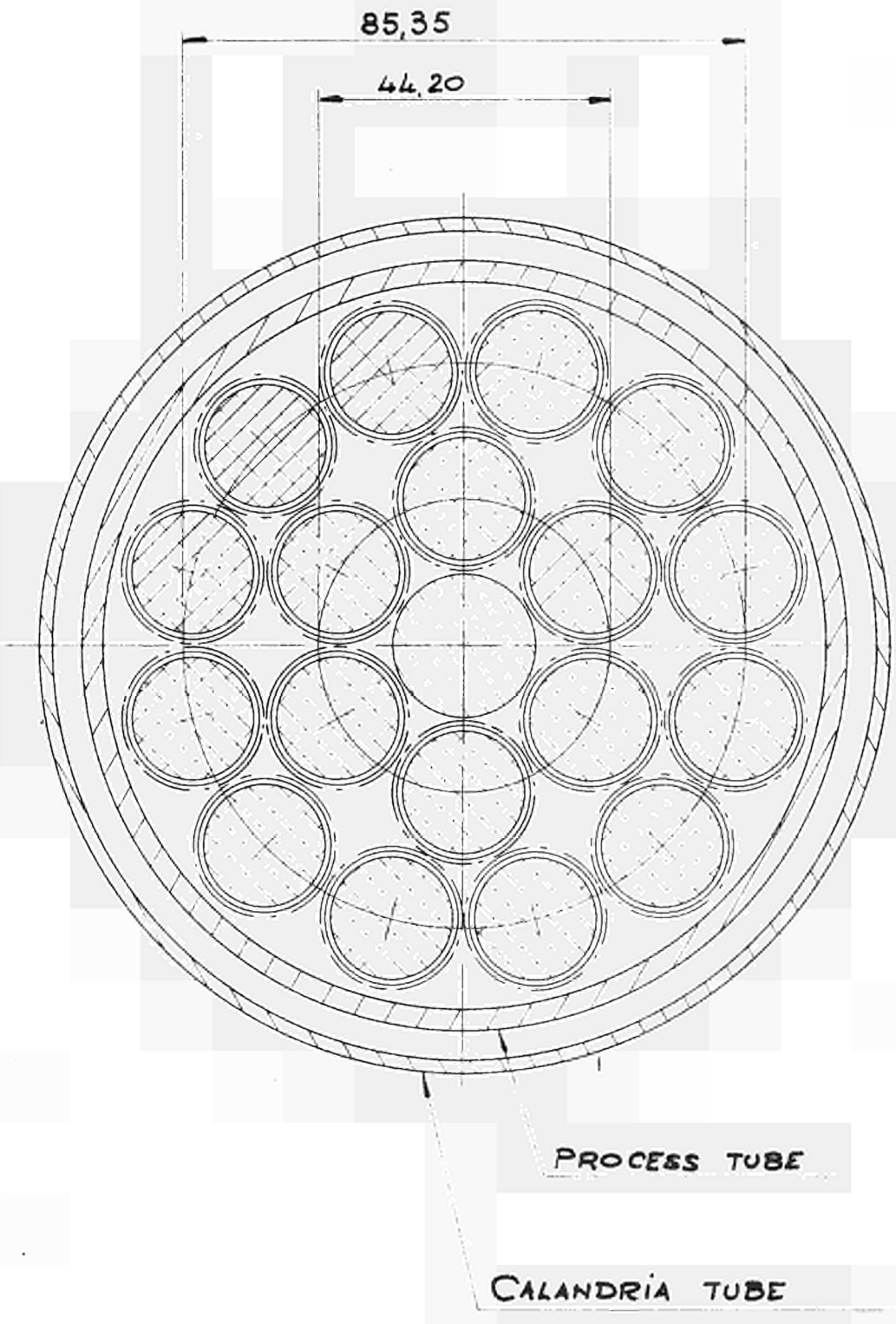


Fig. 4
GO 2255

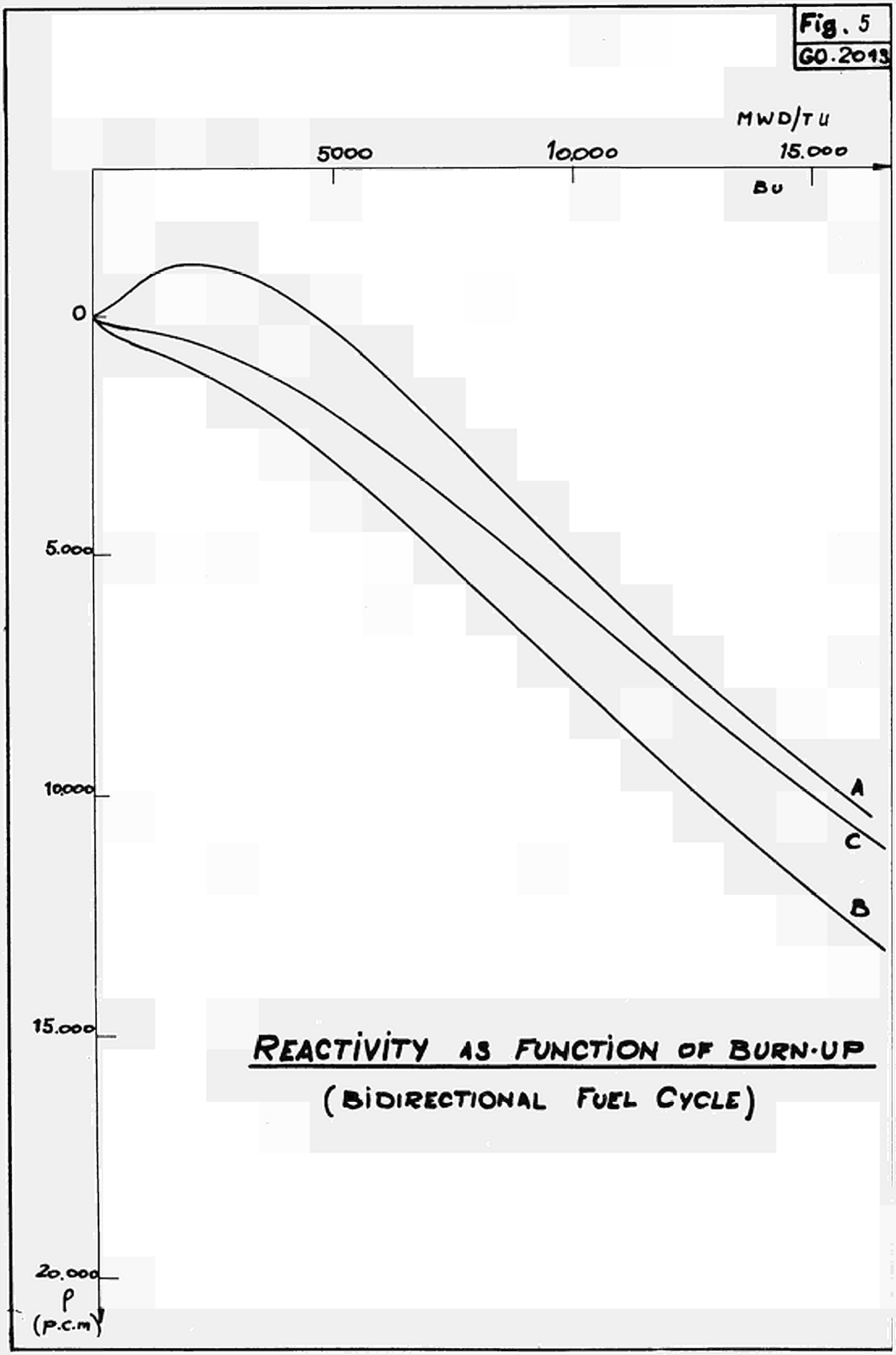
PROTOTYPE ORGEL

CROSS SECTION OF FUEL ELEMENT (REACTOR C)



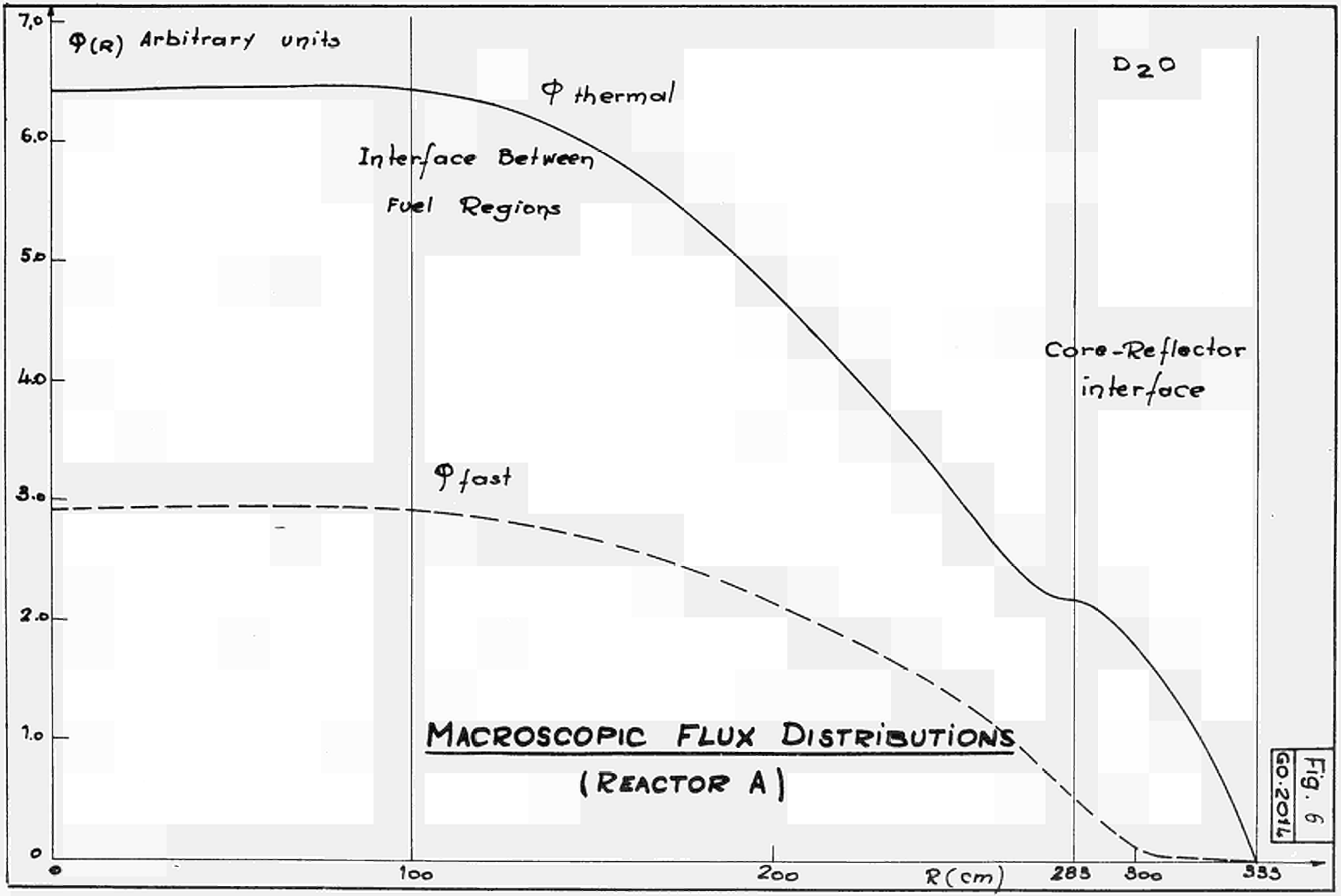
SCALE : 1:1

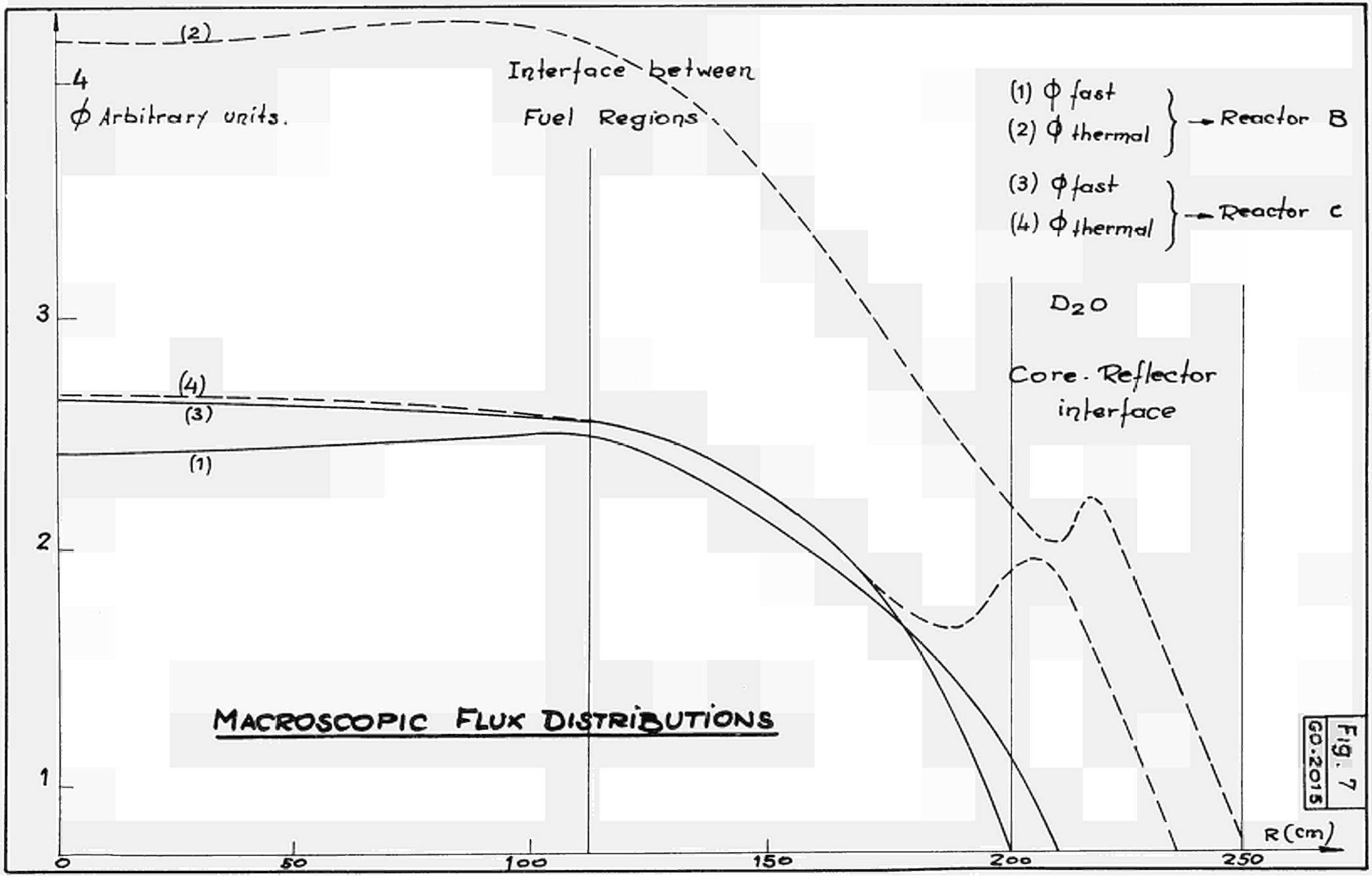
Fig. 5
GO.2013



REACTIVITY AS FUNCTION OF BURN-UP

(BIDIRECTIONAL FUEL CYCLE)


7.0
6.0
5.0
4.0
3.0
2.0
1.0
0
φ(R) Arbitrary units
φ thermal
Interface Between
Fuel Regions
D2O
Core-Reflector
interface
φ fast
0
100
200
R(cm)
285
300
333
MACROSCOPIC FLUX DISTRIBUTIONS
(REACTOR A)
Fig. 6
GO-2014



MACROSCOPIC FLUX DISTRIBUTIONS

Fig. 7
GD-2015

Fig. 8
GO-2262

Diagram of stability reactor B

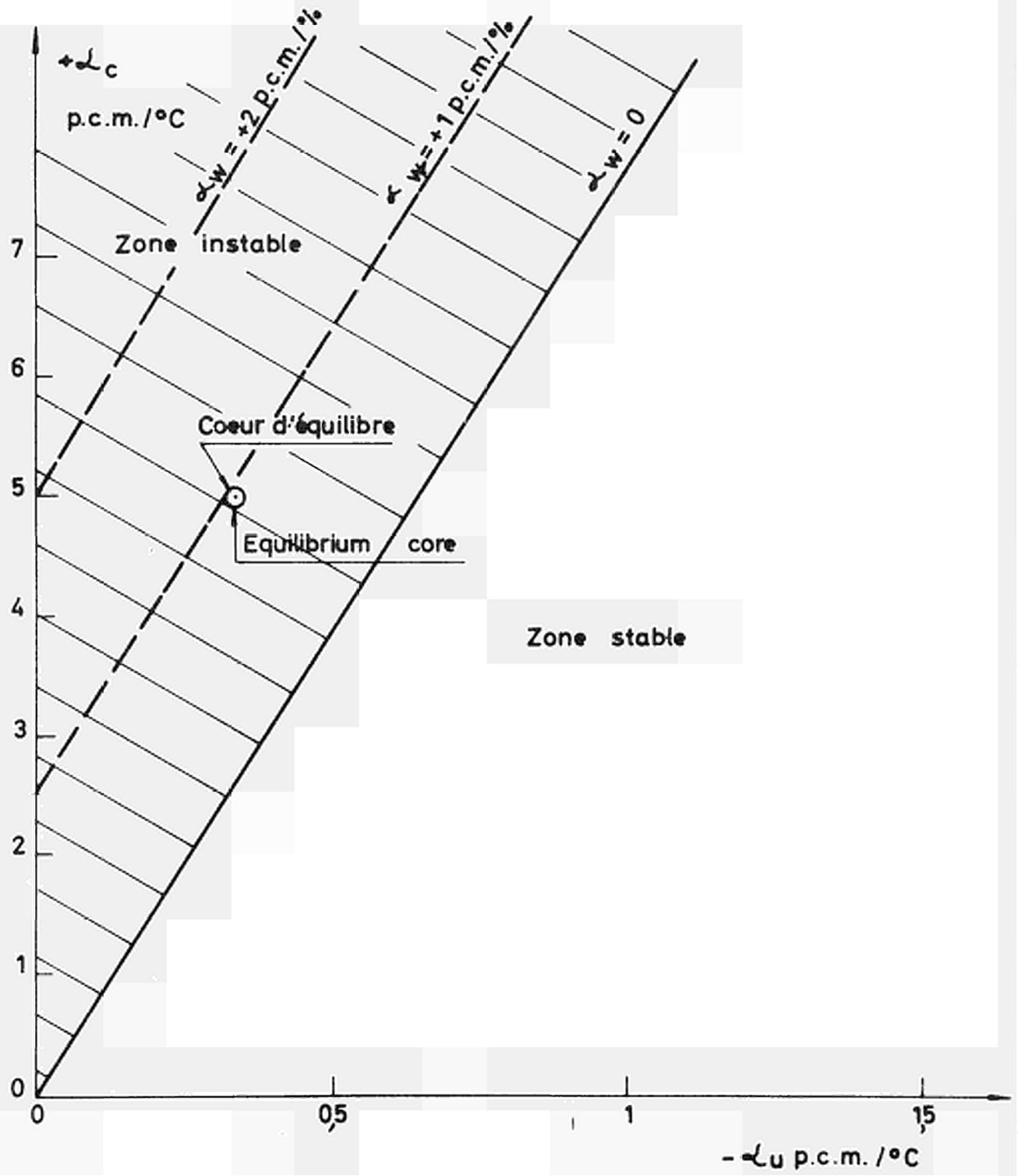


Diagramme de stabilité de la variante B

Fig. 9
GO-2263

Diagram of stability reactor C

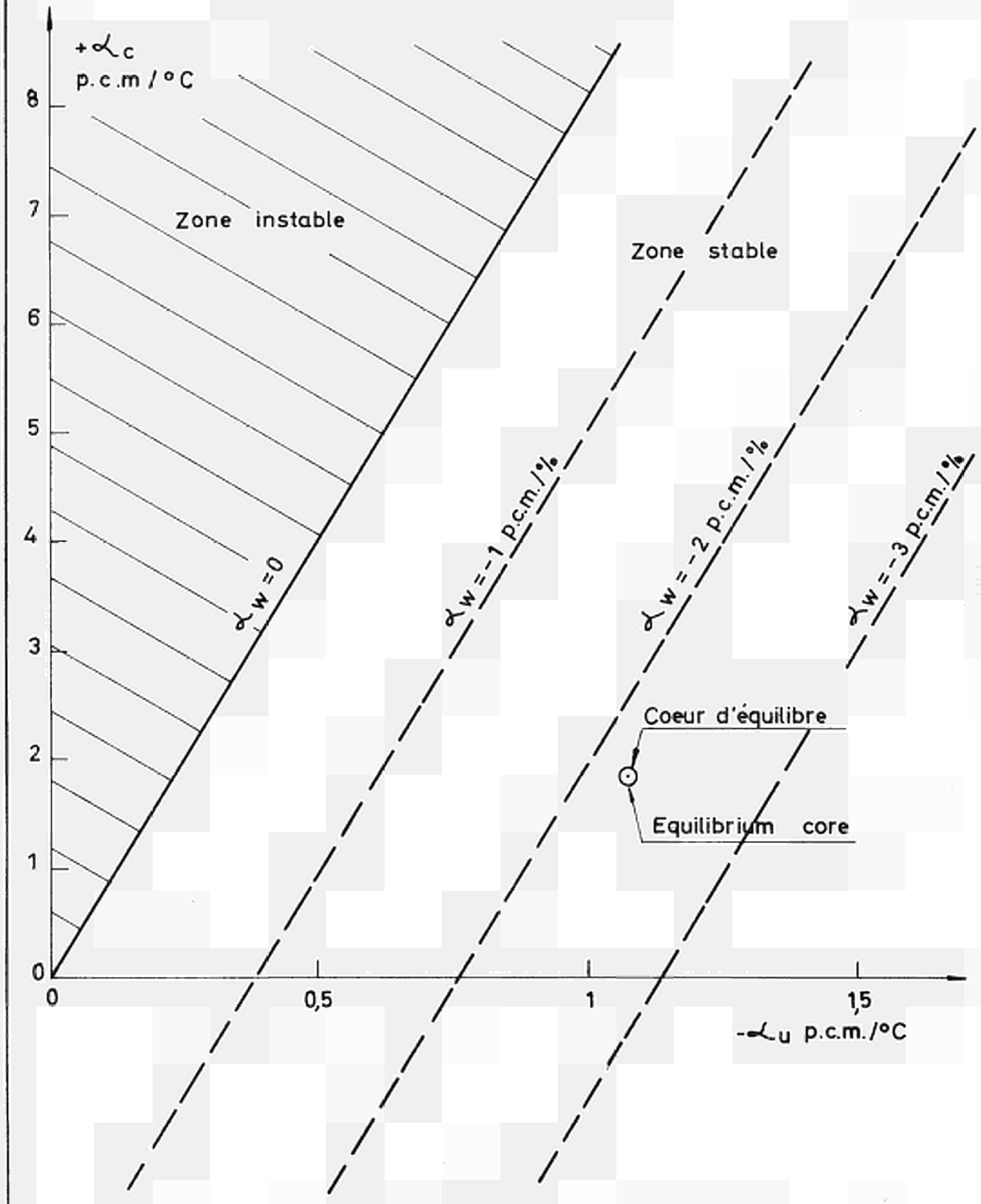


Diagramme de stabilité de la variante C

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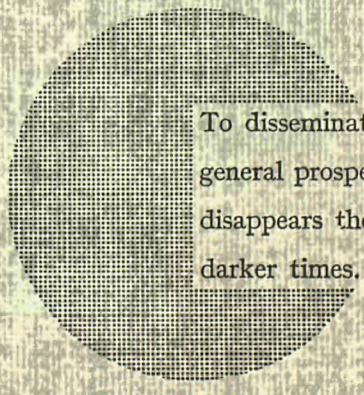

To disseminate knowledge is to disseminate prosperity — I mean general prosperity and not individual riches — and with prosperity disappears the greater part of the evil which is our heritage from darker times.

Alfred Nobel

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