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ORGEL Program

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SUMMARY

A comparison is made in order to assess the economical and technological potential of an ORGEL reactor operated with different fuel managements. All cases are normalized to plants of 250 MWe gross.

Three managements are considered : on-power refuelling with axial shuffling (MA), on-power refuelling with radial shuffling (MR), and off-power refuelling with batchwise shuffling (SR); moreover, the influence of variable fuel enrichment and radial reflector thickness is investigated.

A first comparison in terms of fuel enrichment shows the predominant economic position of the MA management under the hypothesis of fuel burn-up being limited only by the reactivity potential.

A second comparison envisages technological restrictions which limit the maximum local burn-up of the external rods in the fuel bundle. In this context a low burn-up region up to about 12.000 MWd/tU is clearly dominated by the MR management; in a transition zone up to about 20.000 MWd/tU the MR and MA management are economically equivalent; above 20.000 MWd/tU maximum local burn-up the MA management is more attractive.

KEYWORDS

ORGEL REACTOR
ECONOMICS
POWER
FUEL CYCLE
FUELS
ENRICHMENT
BURNUP
REACTIVITY
FUEL CLUSTERS
NUMERICALS

C O N T E N T S

1. INTRODUCTION

Early in 1967, the Commission invited European Community industries to submit offers for a prototype plant with a gross electric power output of 250 MWe, equipped with an ORGEL (organic-cooled-heavy water moderated) reactor.

The possibility was given to the industrials to base their reactor conceptual choices on the results of theoretical works - parametric, dimensioning and optimization studies - carried out by the ORGEL project in the perspective of reactors for a future string.

Being advisable, in elaborating a prototype concept, to call on the main characteristics of the string, it is nonetheless necessary to consider a context peculiar to this prototype :

- The installed power, at least two times lower than that of a string reactor,
- The present status of the technology,
- And, more generally, the wish to build low-cost and reliable equipment.

It is why, as regards the fuel management, we have planned to establish a comparison with three clearly distinct fuel management systems, in order to support the industrials studying the ORGEL prototype in the choice of the best compromise solution. Not only the neutronic aspect of the problem had to be considered, but also the reactor thermal and the plant thermodynamic performances, as well as the cost of investment .

The comparison criterion to be retained finally is the total cost of the energy produced. Particular attention had to be given to fuel element behaviour and, for that matter, the maximum local burn-up rate as well as the in-pile residence time have been considered as important figures.

The fuel management systems which have been confronted in this study are :

- The bidirectional management, which is the reference solution of ORGEL string reactors (principle used in the case of NPD, DOUGLAS POINT, EL-4, KKN, for example). The fuel is submitted to axial shuffling only. In each channel, one element occupies 5 successive positions from one end to another, the shuffling direction being opposed from one channel to the adjacent channels. In this management, fuel handlings take place with the reactor on-power. The sign MA (Management Axial) will be used in the course of this document when referring to this management.

- A fuel management system with fuel handlings at reactor shut-down; it corresponds to the present status of the technology, inasmuch as the heavy water-organic test reactors WR-1 and ESSOR are being loaded in this manner. In the selected management, the fuel element is submitted to radial shuffling only and occupies successively 3 channels in 3 concentric reactor zones. The elements are moved as follows : those of the peripheral zone are transferred batch-wise in the intermediary zone, then, at the next move, batch-wise in the central zone of the core. In the course of this document, this management will be referred to as SR (Shut-down Radial).

- Finally, an intermediary solution between both preceeding managements : fuel handlings are carried out with the reactor on-power; the fuel element is submitted to radial shuffling only. Like in SR management, it occupies successively 3 channels located in 3 concentric zones, but fuel transfers are carried out progressively in time and not by batch. The MR (Management Radial) sign will refer to this management in the course of this document.

NOTE : The SR and MR managements, such as they are being considered here (3 zones), are probably not representative of the optimum attainable by radial shuffling with more complex fuel rearrangements. One merely examined simple schemes, which are sufficient at the stage of an orientation study.

Comparison principles

Besides the nature of the fuel management, two adjustable parameters have been considered :

- The relative fuel enrichment, y, varying between 1,3 and 2,2 times the naturel U^{235} content,
- The thickness of the radial reflector, δR, varying between 25 and 50 cm.

The comparison was carried out in accordance with the following scheme :

By varying the refuelling management, the fuel enrichment and the radial reflector thickness, one has calculated :

- The neutronic and thermal performances of the reactor, the other characteristics of which were maintained unchanged (total fission power included).
- The performances of the associated steam cycles, by maintaining, from one case to another, the pinch points.

Then, for each plant one has determined the costs of energy produced. The comparison could have been carried out from this stage; however, it would have covered plants with powers dispersed between 245 and 270 MWe, thereby attenuating or amplifiying certain differences. Therefore, the costs of energy produced have been further adjusted for plants yielding the same electrical output : 250 MWe gross. Then, one has compared the results thus obtained by representing them, for each reactor family, characterized by the fuel management and the radial reflector thickness, firstly in terms of relative fuel enrichment and secondly in terms of maximum local burn-up of fuel rods. The first comparison assesses the economical position of each management under the hypothesis of a fuel burn-up being limited only by the reactivity potential , whereas the second one envisages technological restrictions which limitate the maximum local burn-up of fuel rods.

Consequently, depending on the grade of confidence the industrials would show to the maximum burn-up, they would choose a project value in regard of which they would find the cost of energy produced by the plant of each reactor family delivering this burn-up. The comparison between the generating costs would then have a homogeneous base. (The other characteristics that can be included in the choice : fuel enrichment and in-pile residence time, are quoted separately).

Bases

The fixed characteristics of the above-mentioned plant (first part) are listed below *; only the main ones are given here :

Reactor	
Number of channels	216
Core height	400 cm
Core radius	200 cm
Axial reflector thickness	30 cm
Thermal power	707 MW
Fission power	752 MW
Control rods	extracted
Thermodynamical cycle	
Pinch point at evaporator inlet	20°C
Pinch point at superheater outlet	10°C

As regards the fuel element, the choice was made on the 18 rod-UC/He/SAP-bundle having a Zr-2 central rod serving as axial structure.

In all cases, a channel is filled with 5 bundles (see Annex 1). The diameter of the UC rods, the coolant flow section and the moderation ratio of the cell (moderator section/fuel section) have been fixed to values which, without setting particular fabrication or behaviour problems, should place the reactor in the field of intrinsic stability.

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* Further details of the plant referred to are described as plant C in the report "Options for a 250 MWe ORGEL Prototype plant; orientation study"
by A. Decressin, J. Noailly, P. Tauch
EUR .... (1968), to be published

(Although this document does not aim at discussing these options, it has appeared useful to briefly report their motivations).

Finally, in calculating the performances, one has limited to 425°C the nominal temperature of the SAP sheathing and to 10 m/s the coolant flow in the channels. In this context, nominal means that the temperature has been calculated with the THESEE 1 code (Ref. 1), excluding the flux peakings at element ends and the effect of various fabrication tolerances.

2. REACTOR PHYSICS

2.1. Basis of the nuclear calculations

In order to assess the advantages and disadvantages of various refuelling schemes against each other, it is necessary that the nuclear calculations determine as well the burn-ups as the power distributions.

The burn-ups of the bidirectional management (MA) have been calculated by means of the depletion code RLT (Ref. 2). Herein, the burn-up equations are solved in function of the flux time. On the basis of an axial cosine flux shape, the local thermal cell parameters are calculated according to the fuel composition evoluting with irradiation. The epithermal cell parameters are assumed to remain constant during irradiation. The effective buckling is found in applying the perturbation method by weighting the local bucklings with their statistical weights.

The assumption of a cosine flux distribution along the channel axis is acceptable for the bidirectional refuelling scheme (MA).

For the radial flux, a flat distribution within an inner zone is needed in order to optimize the bidirectional refuelling scheme. The flattening is obtained by adjusting the burn-ups of the zones according to their different radial bucklings.

The buckling of the inner zone is zero by definition; that of the outer zone has analytically been derived from a one energy group diffusion model dealing with two zones. The reflector zone has been taken into account by its saving.

The managements SR and MR are calculated with the code ERUPT (Ref. 3). This code, which is able to study the burn-up history beginning with the start-up core and ending with the equilibrium cycle, contains a coupling of the first part of RLT and the code EQUIPOISE (Ref. 4). The part of RLT gives the evolution of the fuel composition with the time integrated flux. With EQUIPOISE, the flux distribution is calculated in radial and axial direction at each time step. Nine radial and twenty axial zones of various burn-up states can be distinguished.

The EQUIPOISE code calculated the flux distribution for a homogeneous reactor. The heterogeneity effects of the nuclear cell are taken into account in the calculation of the homogeneous group constants. As the epithermal constants are assumed to remain unaffected during irradiation only the variation of the thermal flux depression with irradiation is considered.

A refuelling is always executed in the moment when the reactivity of the core reaches a prescribed limit. For all managements compared, this limit was set to $\Delta K = 0{,}017$. The reactivity reserved is partly needed to compensate absorptions by structure materials (fuel element end plates) and partly for control purposes.

In the SR management, at each refuelling stage the inner zone is discharged and the fuel of the other zones is loaded in the adjacent zone next to the center. Fresh fuel is always charged to the outer zone.

The MR management can be calculated by the ERUPT code only in a rough approximation. Because of the fact that the code is restricted to problems with azimuthal symmetry, the irradiation history of a single element cannot be pursued, but only that of concentric homogeneous zones of the core. Therefore, at each refuelling step, concentric annuli consisting of several elements are moved.

Calculations performed later for the ORGEL Prototype core as defined by the industrial group with the ERUPT code and the tridimensional REFLOS code on the MR cycle, showed a satisfactory agreement in the results of the two methods.

.2. Discussion of the results

In Fig. 1, a typical evolution of the concentrations of the main fissile and fertile isotopes is given in function of the burn-up for a 1.6 x U_{nat} enriched cell.

The results (average and max. burn-ups, power form factors*, residence times, etc.) of the various fuel managements reported in Table 1 are values found for the equilibrium core. The equilibrium state has been reached when the burn-up of the discharged fuel does not change anymore in successive refuelling steps.

Fig. 2 shows the variation of K_{eff} and mean burn-up with time of a core the fuel of which is managed according to the SR scheme. Initially, it is uniformly charged with fuel of an enrichment $1.3 \times U_{nat}$. The steep slopes in the K_{eff} curve at the beginning of a fresh fuel irradiation are due to Xenon poisoning effects. After about 270 days of irradiation, the core reaches the limit of reactivity set at $\Delta K = 0{,}017$. Then the inner zone (one third of the reactor) is discharged, the other two zones are moved in the zones closer to the center and the outer zone is loaded with fresh fuel (enrichment $1.3 \times U_{nat}$).

The volume averaged burn-up of the first discharged fuel amounts to about 3900 MWD/TU. (In the case that the whole core would have been discharged - batch cycle - a mean burn-up of 3700 MWD/TU would have been attained). The SR management reaches after about two years its equilibrium with a mean burn-up of 4650 MWD/TU. The lowest power form factor between two refuelling stages in the equilibrium state is 0,74, the lowest form factor for the uniformly enriched start-up core being 0.49.

Since it is economic to design the core for the equilibrium state, the difference in form factors between start-up core and equilibrium core means that, in the initial period, the reactor delivers less than nominal power.**

The advantages of the SR refuelling scheme in comparison with the corresponding batch one are evident. The mean burn-up in the cited case is by about 25% and the minimum power form factor by about 50% higher than in the batch cycle.

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* The power form factor (PFF) is the product of the axial form factor, the radial form factor and the bundle form factor.

** Provided the form factor is not raised by differential enrichment of the first core.

For these advantages, a higher shutdown time for refuelling operations has to be paid, thus penalizing slightly the availability of the plant.

Fig. 3 shows the axially integrated flux distributions in the SR management for various fuel enrichments. For an enrichment of 1.3 x U_{nat}, a rather flat flux distribution is obtained. With increasing enrichment, there exists a marked difference in the nuclear properties between the depleted fuel in the center of the core and the fresh fuel charged at core edge. This leads to the decrease of the power form factor with increasing enrichment, as shown in Table 1. The same trend is seen also for the MR management, whereas the power form factors of the MA management remain nearly constant because of the flux flattening in the reactor center.

A smaller reflector thickness improves the form factors as can be seen comparing, in Table 1, the MR management, calculated once with a thickness of 50 cm and once with a thickness of 25 cm. But the gain in form factor has to be paid by a loss in burn-up. The conversion in costs of all the effects which accompany a variation in reflector thickness shows finally whether the change means a penalty or a gain.

3. THERMAL PERFORMANCES OF THE REACTORS

3.1. Method of calculation

At reactor level, the calculation is based on the coolant channel, the thermohydraulic performances of which are determined by the code THESEE (Ref. 1). Once the fuel element being chosen, the fuel bundle is divided in axial zones and the zones again in sub-channels, the latters being confined artificially by the constrictions of fluid streams and connected together by mass transfer (mixing).

The average coolant velocity being given, the local velocity in the sub-channels is determined as to assure a constant total pressure drop over a given height of the zone. Then the thermal balance of the bundle is made-up taking into account the mixing of the different fluid streams. The local cladding temperature of each sub-channel as defined in THESEE 1 is calculated from its average heat transfer coefficient and from its average heat flux.

The step from the channel performances to those of the reactor is done in supposing the maximum nominal cladding temperature being the same for all channels. In order to satisfy this condition, a regulation device for the coolant flow at each channel is necessary.

The important differences among the axial distribution of power generation of the channels for the envisaged fuel managements has brought us to deal with the problem at reactor level in two different ways.

3.2. MA management

This management is characterized by an axial form of power generation being identical for all channels of the reactor. As a consequence, it is sufficient to calculate the performances of the most charged channel as the temperature profiles are practically the same in all channels (see Annex 2).

Thermohydraulic performances of the most charged channel **

- Determination of the heat rating

Suppose :

W_R	=	thermal power of the reactor
N_C	=	number of channels
PFF	=	overall power form factor of the reactor
L	=	effective length of fuel
n	=	number of rods in the bundle
$q/4\pi$	=	linear heat rating

The maximum linear heat rating is defined such that :

$$W_R = L \cdot n \cdot N_C \cdot PFF \cdot (q/4\pi)\ max. \ 4\pi \ (*)$$

- Calculation hypothesis

The average inlet coolant velocity in the most charged channel is fixed at 10 m/s. The maximum cladding temperature (without hot spots) is taken as 420°C.

- Results

Fuel enrichment being without significant influence on the overall power form factor of the reactor, the heat rating remains sensibly constant (Table 2).

For a relative enrichment of y = 1,6, the influence of the direction of fuel shuffling with respect to the direction of coolant flow has been investigated. The maximum cladding temperature is found to be about 5°C higher for opposite circulation than for parallel one. As a consequence, the coolant inlet temperature is determined on the basis of the "opposite circulation" in order to guarantee the maximum cladding temperature tgm = 420°C.

** Cooled by OM2 containing 5% HB (purified by distillation)

* As the reactor thermal power is kept constant for a reactor core of given dimensions, one has :

$$PFF \cdot (q/4\pi)\ max = const$$

3.3. Radial managements (MR and SR)

The reactor core is subdivided into 9 zones having the same volume each; for all zones, the axial form of power generation is calculated at the moment just before fuel extraction (see chapter 2.2.); indeed, it may be supposed that the extraction of fuel from one channel does not sensibly affect the general form of power generation at reactor level. This variation will be accounted for in calculating the hot channel factors.

All channels of one zone are supposed to release the same power. The general hypothesis concerning coolant velocity and cladding temperature are the same as for the bidirectional management.

- Channel performances

The thermal performances of one channel of the most charged zone are calculated in the same way as for the bidirectional management.

The limitations are constituted by cladding temperature and coolant velocity, the principal result of calculation being the coolant inlet temperature. For all other channels, the coolant inlet velocity is calculated starting from the limitation set by cladding temperature and coolant inlet temperature determined for the most charged channel. Then the thermohydraulic performances of these channels may be determined and thus also of the zones.

- Reactor performances

From the calculated performances of each zone of the core, the reactor outlet temperature and the mass flow are established as balanced average values. The results are given in Table 2.

4. POWER OF THE DIFFERENT PLANTS

Basic assumptions of this report are the constant thermal and the constant fission power of the plant to be generated in a core of fixed dimensions. Moreover, the core is calculated as to yield, in all channels, the maximum permissible fuel cladding temperature and, for the most charged channel, the maximum mean coolant velocity at the entrance of it. The maximum linear heat rating and the power form factors being different for each plant, it is obvious that the thermal performances of the core vary (coolant inlet and outlet temperature, coolant heating across the core). Hence, the thermodynamic efficiencies of the associated steam cycles differ from each other resulting in gross electric power output between 246 and 270 MWe.

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.

The pinch point at evaporator inlet of the steam generator was found to yield the optimum cycle efficiency between 20 and 30°C; a value of 20°C was adopted. Superheater and reheater are arranged in parallel, the pinch points at the outlet being fixed at 10°C.

The low-pressure stages of the turbine are equipped with grooved blades for decreasing the exhaust steam wetness at 10%; condenser pressure is taken as 0,044 kg/cm^2.

5. COST EVALUATION

The criterion in comparing the different fuel managements is the total cost of electric energy produced by the plants. It should be remembered that not the absolute level of costs is of importance, but the relative position of the fuel managements among themselves.

With this aim in view, it was possible to refer to a cost evaluation carried out in 1962/1963 in the frame of a 250 MWe Design Study Contract for an ORGEL power plant by the firms BELGONUCLEAIRE-INDATOM-SIEMENS (Ref. 5).

The cost estimates established under this contract were based on the state of technological development in 1962 and had the aim of ascertaining the investment costs 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 made hereunder refer to the ORGEL plant "tête de filière".

5.1. Direct cost of construction

The direct plant investment was calculated by adjusting the results of the above-mentioned Design Study Contract. In order to escalate these 1962 cost figures to 1966 figures, a rate of 2,4% per year was assumed, totalling a 10% escalation in 4 years.

The direct construction costs include the reactor, its primary circuits, fuel handling devices, steam generators, D_2O-moderator and organic coolant, but not the first charge of fuel and fuel reserve. In addition, they include the site clearance and construction work, auxiliary work, turbogenerator unit, electrical equipment, main step-up transformers. The land is not included. Heavy water is estimated at $ 20/lb.

Certain cost components vary from one fuel management to another. Table 1 gives the most significant changes in direct plant investment costs.

TABLE 1

Fuel management		MA	MR (δ_R = 25 cm)	SR
Reactor block	10^6 $	3,68	3,59	3,68
Fuel handling	10^6 $	3,57	3,07	1,59
Heavy water	10^6 $	4,08	3,15	4,08

The cost variations for the reactor block and the heavy water are due to a change in radial reflector thickness from δ_R = 50 cm (for MA and SR) to δ_R = 25 cm (for MR). Fuel handling devices will be less complicated for an off-power refuelling management (SR) than for on-power refuelling (MA and MR).

The power plants will differ one from another in net power output due to slight changes in operating temperatures.

Thus, the cost of the secondary installations which are mainly a function of the installed electric power, vary also (between 13,82 and 13,05 million $).

5.2. Indirect costs of construction

These costs are calculated as percentage of the direct costs of construction. They include engineering (6%), overheads and administrative costs (6%), interest during construction (12%), contingencies and possible price increases till putting in operation (10%), miscellaneous (1%).

The percentages adopted here (35% in total) are based on the results of the Symposium on Technical and Economical Problems for Proven-Type Reactors held in Venice, October 1963.

Taxes on capital and return on capital during construction are not included in view of the fact that their volume, which depends on the tax system applicable in the country where construction takes place, may vary between 0 and 6% in the Community.

Moreover, no account is taken of customs duties, it being assumed that all equipment is supplied from within the Community.

5.3. Fixed costs of plant investment

The fixed costs due to plant investments (annual instalment) include interest on money, amortization and taxes on revenues.

The total annual instalment rates vary considerably in the countries of the European Community, between 8,1% (France), 10% (Italy, the Netherlands), and 13% (Belgium, Germany). In this economic evaluation, a rate of τ = 10% is adopted for annual instalment.

Interest rates are in general between 5,5 and 7%. Amortization rates are based on estimated plant lifetime between 20 and 30 years for proven-type reactors; 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 also differ considerably, ranging from exemption (France, The Netherlands) to 3-4 per year on the revenues (Belgium, Germany).

The annual plant load factor is taken as 0,8 equal to 7000 h of full load operation per year for the plants having fuel managements with on-power refuelling (managements MA and MR). In the case of off-power refuelling (management SR), the load factor will decrease the more the shorter the in-core residence time of the fuel elements, i.e. as a function of the number of refuelling operations per year.

5.4. Fuel cycle costs

Fuel costs

The costs of the enriched UF_6 are taken from the USAEC price list dated July 1962. Conversion costs for converting UF_6 into UC rods ready for cladding amount to 40 \$/kg U, costs of finned SAP sheaths to 5 \$/kg U. The costs of cladding, assembly and inspection are 14 \$/kg U. Then, the total costs of fresh fuel elements will be 100 \$/kg U, 120 \$/kg U, 140 \$/kg U 160 \$/kg U, respectively for relative enrichments of 1.3, 1.6, 1.9 and 2.2.

Fuel cycle costs

The calculations were made for the reactor being at equilibrium. In computing the fixed costs of the fuel cycle, the half first-charge costs are amortized over the whole plant lifetime. The interest rate on spare fuel (equal to 10% of a core) is 6%. Annual instalment rate and plant load factor are the same as described under 5.3.

The variable costs of the fuel cycle (fuel consumption costs) are calculated with and without Pu-recovery. Only the extracted Pu-isotopes are credited at 8 $/g Pu of all isotopic composition, the residual U-235 is not considered. Reprocessing and transportation costs are estimated at 21 $/kg U.

.5. Operation and maintenance costs

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

Organic coolant make-up costs are calculated for an equilibrium content of 5% HB* to be maintained by distillation. Cost of fresh coolant is 0,3 $/kg.

From operating experiences with other D_2O-moderated and-cooled reactors, one may reasonably assure yearly D_2O losses of about 0,5% of total inventory for the D_2O-moderated ORGEL reactor, which is operated at small D_2O pressures only.

* HB stands for high boilers= molecules with higher molecular weight than ordinary organic coolant

6. ELECTRICITY GENERATING COST NORMALIZATION TO PLANTS OF 250 MWe GROSS POWER OUTPUT

In an early study, the power generating costs of ORGEL type plants have been evaluated in the range of 100 to 1000 MWe gross. These plants were quite similar in construction, only the fuel element performances were assumed to be slightly higher for the larger plants ($\geqslant$500 MWe gross). The fuel element was always the same.

Electricity generating cost showed to follow fairly well an exponential law :

$$w = w_o \left(\frac{W_{eb}}{W_{ebo}} \right)^{-0,41} \qquad \text{(mill/kWh)}$$

in the range of Web = 200 to 1000 MWe gross.

These generating costs are computed in the same way as described in chapter 5 and contain the following cost elements :

- fixed costs due to direct and indirect plant investments;
- fixed costs of the fuel cycle
- costs of organic coolant make-up and D_2O losses.

By arranging the cost elements of the plants with different power output in the same way as in the above-mentioned early study, this cost variation law may be applied to normalize all generating costs to plants having 250 MWe gross output. Such a normalization being performed in a small region around the nominal output of 250 MWe will give a fair approximation of production costs.

It is obvious that a normalized plant will no longer have the same postulated fission power of 752 MW. At fixed mean maximum cladding temperature of the fuel element and fixed maximum coolant entrance velocity, the thermal performances of the core will slightly change as to yield the thermal power necessary to produce 250 MWe gross. Generally, the exact design parameters and performances of the normalized plant have to be found by iteration, but, in this study with maximum electric power deviations of 8% the difference in net power plant efficiency, η_{en} is only 0,3% (0,335 for the normalized plant to be compared with 0,334 for the actual plant). So, the fission power W_{fo} of the normalized plant can be determined by the relation W_{fo} = $(Web_o - Wa)/\eta_{en}$, the total auxiliary power Wa being sensibly constant.

7. GLOBAL RESULTS

The first part of the comparisons is done under the hypothesis that the fuel burn-up is limited only by the reactivity potential. As a consequence, figures 4 and 5 show quantitatively the fact that, under certain circumstances, energy generating costs can be cut down substantially by a higher enrichment of fuel which increases the mean burn-up.

In the range of relative fuel enrichment investigated, the differences in power generating costs of the normalized plants are quite large (0,5 to 0,8 mills/kWh), the MA fuel management yielding the lowest and the SR fuel management the highest costs (Fig. 4). The same result shows up also in the case where reprocessing of the spent fuel is envisaged (Fig. 5); the absolute level of power generating costs being about 0,15 mills/kWh lower than without fuel reprocessing.

The surprisingly large gap between the MA and the MR fuel management can be explained, at least partially, by the fact that the MA management has been studied thoroughly in the last years and refers to well optimized power plants, whereas much less effort is spent for the MR and also for the SR management. Indeed, this orientation study was conducted in choosing a well defined core (optimized for an MA management) and to see which power can be extracted from that core in utilizing or an MR or an SR fuel management without reoptimizing the core dimensions according to the adopted fuel management. More detailed studies would probably reduce the differences between the managements.

The mean burn-up attainable - being economically of great importance for the fuel cycle cost - is shown in Table 1 for each management in function of the relative enrichment. Thus, for a given enrichment, the mean burn-up of the investigated MR management is in general about 25% lower than for an MA management (at equal radial reflector thickness of 50 cm). Upon reducing the reflector thickness of the MR management in order to raise the global power form factor and to shorten the fuel in-core residence time the mean burn-up attainable will be about 30% lower as compared to that of the MA management.

The second part of the comparisons investigates the economical position of the different fuel managements, taking into account the maximum local burn-up of the most charged rod in the fuel bundle.*

Such a comparison gives a better view on the potential of each management in the case that a limitation is set by the maximum local burn-up.

Plotting the power generating costs over the maximum local burn-up (figures 6 and 7), it is obvious that the SR management is ruled out at once against the MR one for its poor burn-up performances and for the lower load factor due to off-power refuelling.

On the other hand, the MR management now competes excellently with the MA scheme and yields even appreciably lower energy generating costs at least in the region below about 12.000 MWd/TU.

In a transition zone situated between 12.000 to 20.000 MWd/TU, the MR and MA management show about the same economical performances with slight advantages for MA at higher burn-up.

Finally, a third zone with maximum local burn-up in excess of 20.000 MWd/TU is clearly dominated by the MA management.

Another feature of some interest, the in-core time of the fuel elements, is also indicated in figures 6 and 7. In general, fuel managed according to a MA scheme has the shortest residence time in the core and the MR management yields longest one at any maximum local burn-up considered here.

On the basis of this comparison and looking after figures 6 and 7, it seems that the MR management is adapted for a 250 MWe prototype, burning enriched Uranium, the aimed burn-up of which being relatively low.

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* The maximum local burn-up of a rod bu^*_{max} (MWd/tU) is found upon dividing the maximum local burn-up of the bundle in Table 1 by the bundle form factor (taken at 0.92).

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EUR .... (to be published) (1968)

(4) EQUIPOISE-3 : A two-dimensional two-groups neutron diffusion code for the IBM 7090 computer
I.B. Fowler, M.L. Tobias - ORNL-3199

(5) ORGEL Reference Design (Contract 042-62-1 ORG C)
EUR-1616 (1967)

ANNEX 1

GEOMETRICAL CHARACTERISTICS OF THE REACTOR UNIT CELL

Unit cell

Lattice pitch (square)	24,2 cm
Ratio of moderator to fuel volume	9 -
Coolant channel inner diameter	11,0 cm

Fuel element

Number of elements per channel	5 -
Number of rods per element	18 -
Overall length of the element	80 cm
Length of fuel core	75,5 cm
Diameter of the UC fuel pins	1,830 cm
Carbon content in UC (wt.%)	4,9 %
Cladding material	SAP
Cladding material thickness (between fins)	0,0915 cm
Height of fins	0,075 cm
Finning ratio	1,75 -
Fuel cross section	47,34 cm^2
Cladding cross section	14,1 cm^2
Coolant cross section	31,1 cm^2
Ratio of coolant to fuel cross section	0,66 -

ANNEX 2

CLADDING TEMPERATURES FOR MA MANAGEMENT IN ANY TWO CHANNELS

Suppose W_1 and W_2 the power generated in any two channels of the reactor; under the hypothesis of equal heating of the coolant in all channels, the mean coolant velocity V is directly proportional to the power generated :

$$\frac{W_1}{W_2} = \frac{V_1}{V_2} \qquad (1)$$

In this case, the cladding temperature at a given point of the channel is equal to :

$$tg_1 = t_o + A\,\frac{W_1}{V_1} + B\,\frac{W_1}{V_1{}^{0,9}}$$

$$tg_2 = t_o + A\,\frac{W_2}{V_2} + B\,\frac{W_2}{V_2{}^{0,9}} \qquad (2)$$

for the channels 1 and 2 respectively, where :

t_o = coolant inlet temperature
A,B= constants

The last term represents the temperature difference between wall and fluid which is inversely proportional to the heat transfer coefficient, then to $V^{0,9}$.

In combining equations (1) and (2), one gets :

$$tg_2 = tg_1 + B\,\frac{W_2}{V_2{}^{0,9}}\left[\,1 - \left(\frac{W_1}{W_2}\right)^{0,1}\,\right] \qquad (3)$$

Numerical example

Taking an extreme case of the power of the least charged channel being only 65% of the most charged one, and considering a maximum temperature difference $B \frac{W_2}{V_2 0,9} = 80^{o}C$, one gets :

$$tgm_2 = tgm_1 - 3,5 \quad (^{o}C)$$

Thus, in first approximation, the cladding temperature profiles may be estimated identical for all channels.

TABLE 1 - PHYSICAL PERFORMANCES

Management	Relative enrichment	$\overline{bu}$ [1] MWd/tU	bu max [1] MWd/tU	P F F	rt d	U-235 g/kg U	Pu g/kg U	A kgU/year
MA δ_R = 50 cm	1.3	7.020	10.060	0.56	450	3.65	4.02	29.750
	1.6	13.230	15.930	0.56	810	2.55	5.05	15.850
	1.9	18.730	21.600	0.55	1240	2.20	5.37	11.250
MR δ_R = 50 cm	1.3	5.360	7.280	0.68	370	4.50	3.44	40.650
	1.6	10.130	13.130	0.62	720	3.55	4.59	21.450
	1.9	13.750	17.850	0.59	970	3.55	4.86	15.780
MR δ_R = 25 cm	1.3	4.980	6.040	0.84	350	4.70	3.29	40.880
	1.6	9.400	11.000	0.76	670	3.85	4.43	21.780
	1.9	12.800	14.800	0.72	900	3.90	4.73	16.100
	2.2	16.000	18.400	0.69	1150	4.10	4.95	13.300
SR δ_R = 50 cm	1.3	4.650	6.180	0.74	330	4.95	3.09	40.980
	1.6	8.780	11.200	0.57	630	4.15	4.28	23.700
	1.9	11.850	15.150	0.48	850	4.25	4.60	18.670

$\overline{bu}$ (MWd/tU) = average burn-up at extraction of fuel

bu max (MWd/tU) = maximum local burn-up of the bundle

PFF = global power form factor at equilibrium

U-235 (g/kg U) = final concentration of U-235 at fuel extraction

Pu (g/kg U) = total Pu concentration at fuel extraction (all isotopes)

A (kgU/year) = average Uranium throughput/year

---

[1] calculated at 200 MeV/fission

TABLE 2 - THERMAL PERFORMANCES

Management	Relative enrichment	$(q/4\pi)$ max W/cm	t_1 ^oC	t_o ^oC	Δt ^oC
MA δ_R = 50 cm	1.3	68.2	365	305	60
	1.6	68.4	363	303	60
	1.9	69.6	361	301	60
MR δ_R = 50 cm	1.3	56.3	354	283	71
	1.6	62	358	282	76
	1.9	64.9	359	280	79
MR δ_R = 25 cm	1.3	45.7	372.5	310	62.5
	1.6	50.4	370.5	308.5	62
	1.9	53.2	369	302.5	66.5
	2.2	55.5	364	291	73
SR δ_R = 50 cm	1.3	51.8	377	320.5	56.5
	1.6	67.2	363	291	72
	1.9	79.8	356	258	88

$(q/4\pi)$ max (W/cm) = max. local heat rating of the most charged peripheral rod

t_1 (^oC) = average coolant outlet temperature

t_o (^oC) = coolant inlet temperature

Δt (^oC) = average heating of the coolant across the reactor core

TABLE 3 - PLANT PERFORMANCES

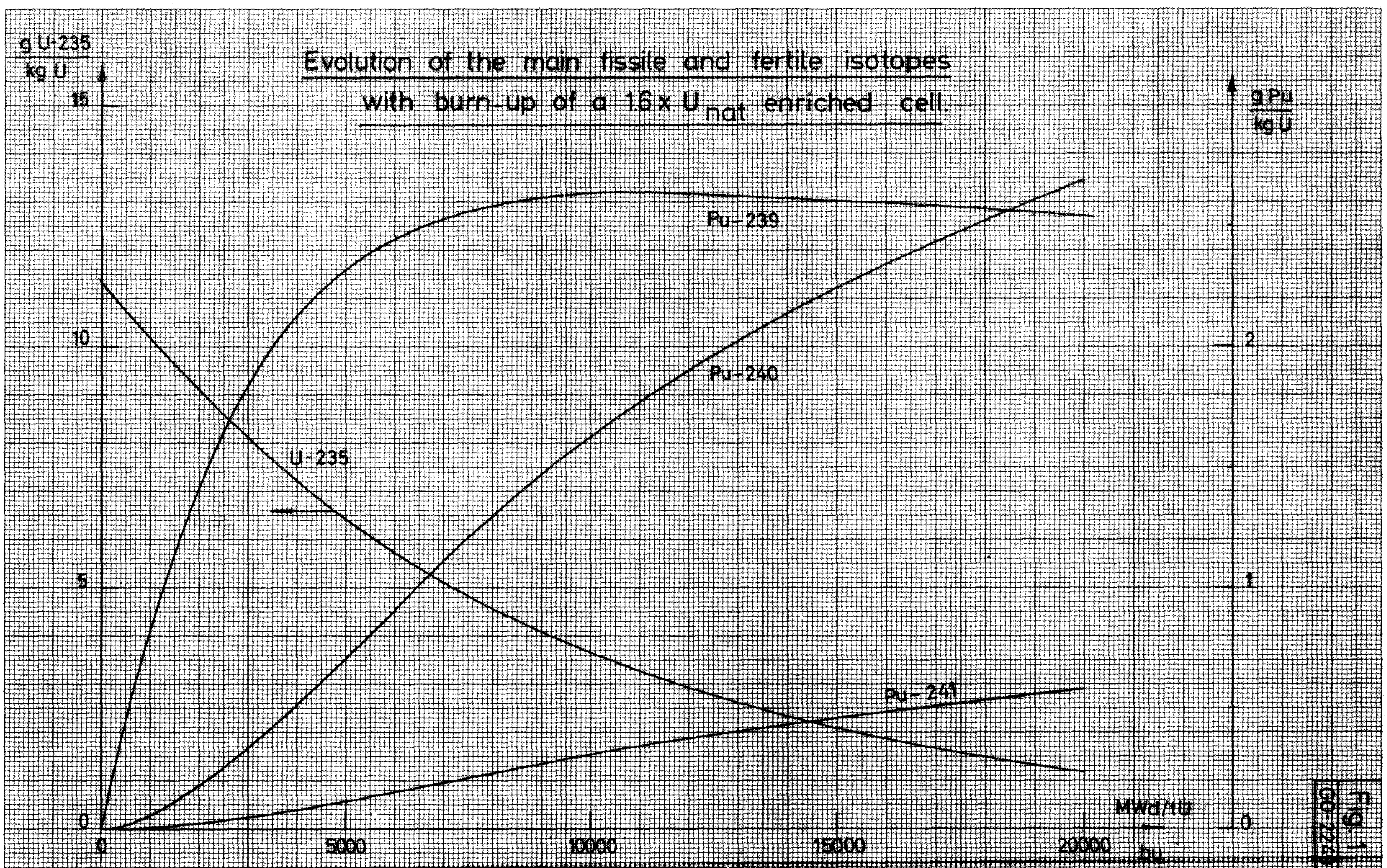
Management	Relative enrichment	η_{eb}	Web MW	η_{en}	Wen MW
MA δ_R = 50 cm	1.3	0.369	261.7	0.325	244.0
	1.6	0.367	260.8	0.323	243.1
	1.9	0.365	259.2	0.322	241.5
MR δ_R = 50 cm	1.3	0.354	251.7	0.311	233.9
	1.6	0.355	252.3	0.315	236.9
	1.9	0.356	252.4	0.315	238.2
MR δ_R = 25 cm	1.3	0.374	265.8	0.330	248.5
	1.6	0.372	264.3	0.322	246.9
	1.9	0.370	262.7	0.325	246.0
	2.2	0.363	257.5	0.321	241.7
SR δ_R = 50 cm	1.3	0.380	269.6	0.334	251.4
	1.6	0.364	258.2	0.322	242.2
	1.9	0.347	246.4	0.308	231.9

η_{eb} (-) = Gross electric efficiency of the steam cycle

η_{en} (-) = Net electric plant efficiency, related to the reactor fission power

Web (MWe) = Gross electric power output

Wen (MWe) = Net electric power output of the plant



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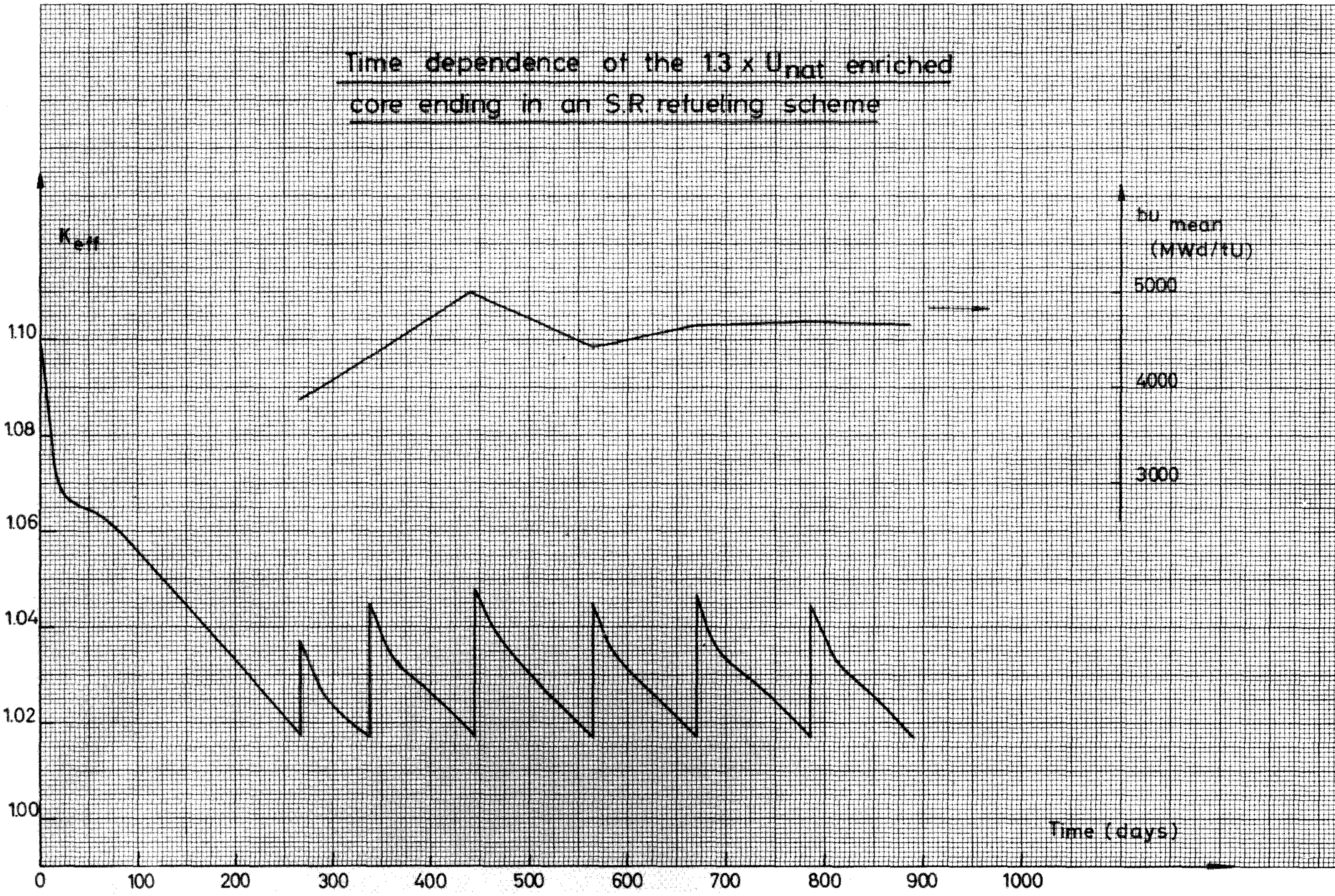
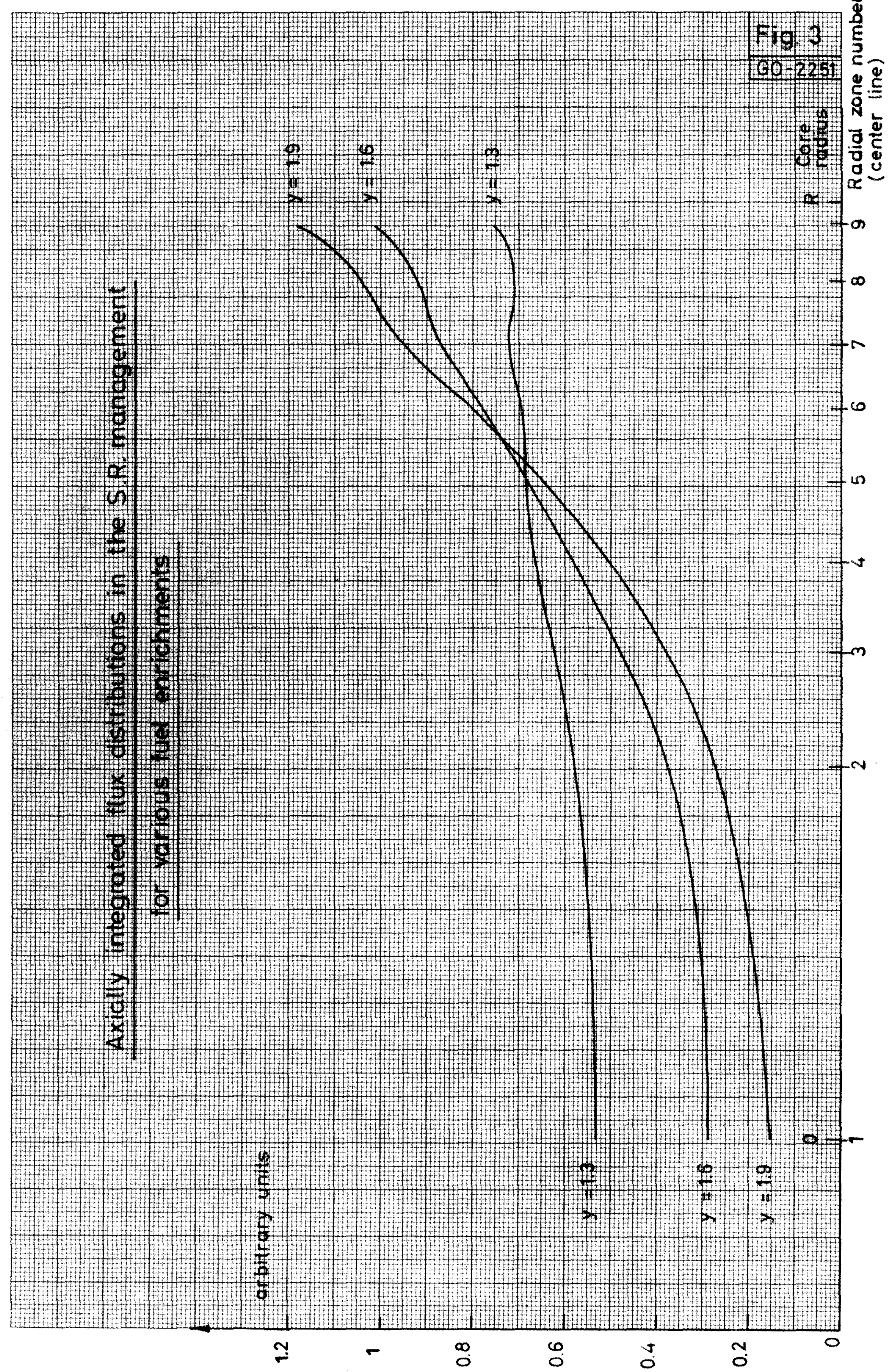

Time dependence of the 1.3 x U nat enriched
core ending in an S.R. refueling scheme
K eff
bu mean
(MWd/tU)
5000
4000
3000
1.10
1.08
1.06
1.04
1.02
1.00
0
100
200
300
400
500
600
700
800
900
1000
Time (days)

Fig. 2

GD-2250



Axially integrated flux distributions in the S.R. management
for various fuel enrichments
arbitrary units
y = 1.3
y = 1.6
y = 1.9
Radial zone number (center line)
R Core radius
0
1
2
3
4
5
6
7
8
9
0.2
0.4
0.6
0.8
1
1.2
Fig. 3
GO-2251

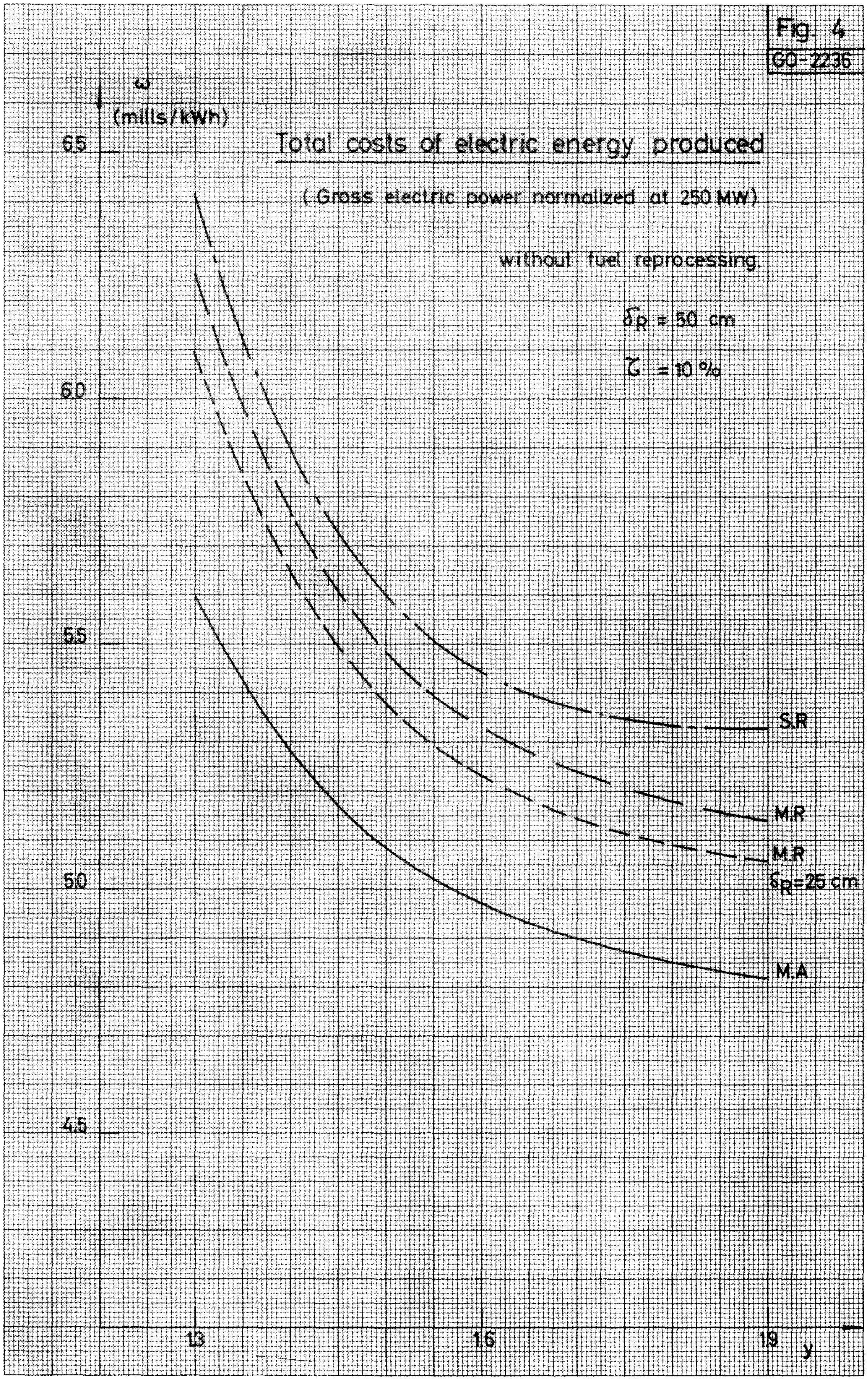


Fig. 4
GO-2236
ω
(mills/kWh)
Total costs of electric energy produced
(Gross electric power normalized at 250 MW)
without fuel reprocessing
δR = 50 cm
ζ = 10 %
6.5
6.0
5.5
5.0
4.5
S.R
M.R
M.R
δR=25 cm
M.A
13
16
19
y

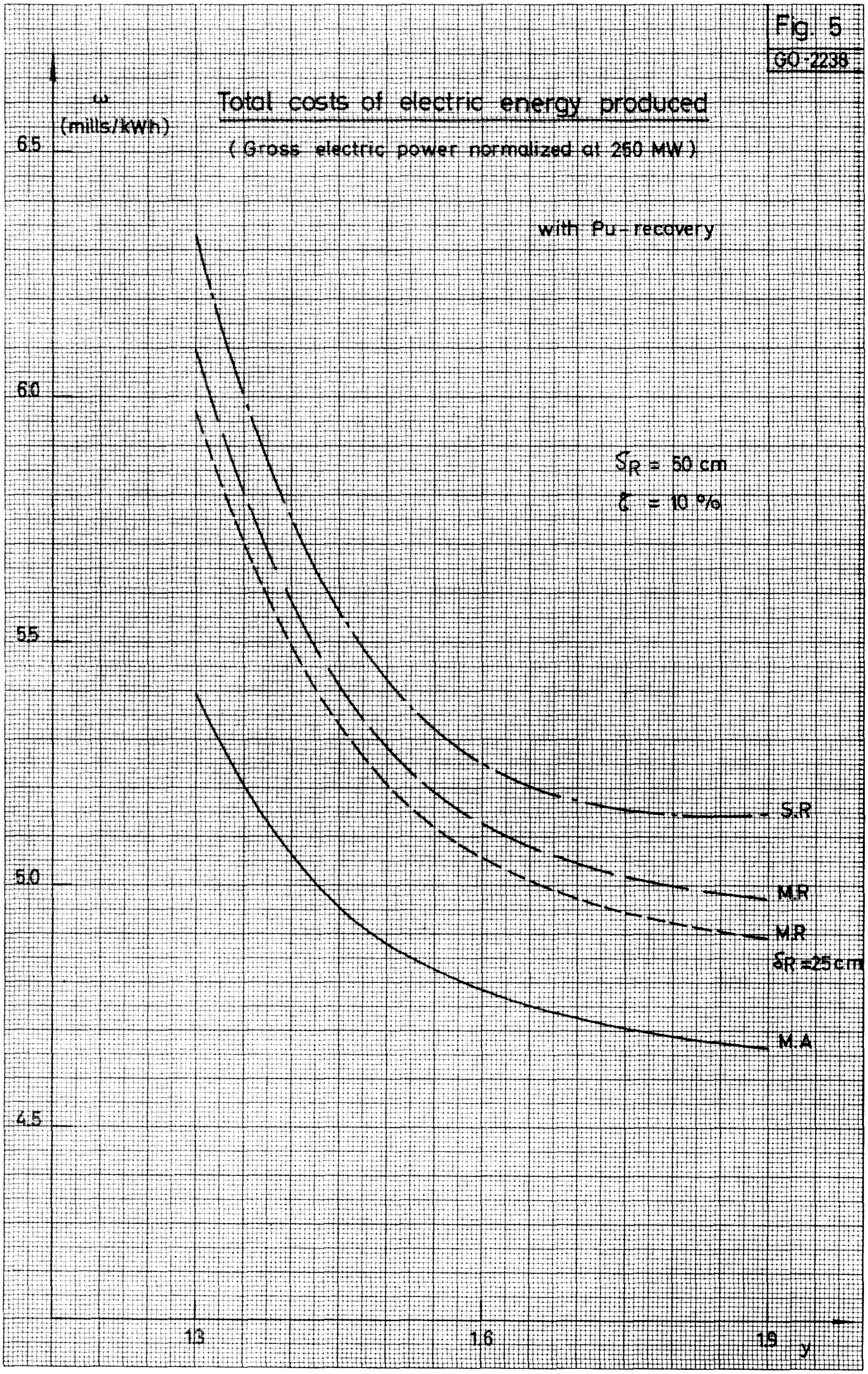


Fig. 5
GO-2238
Total costs of electric energy produced
(Gross electric power normalized at 250 MW)
with Pu-recovery
ω (mills/kWh)
6.5
6.0
5.5
5.0
4.5
δ_R = 50 cm
ζ = 10 %
S.R
M.R
M.R
δ_R = 25 cm
M.A
1.3
1.6
1.9
y

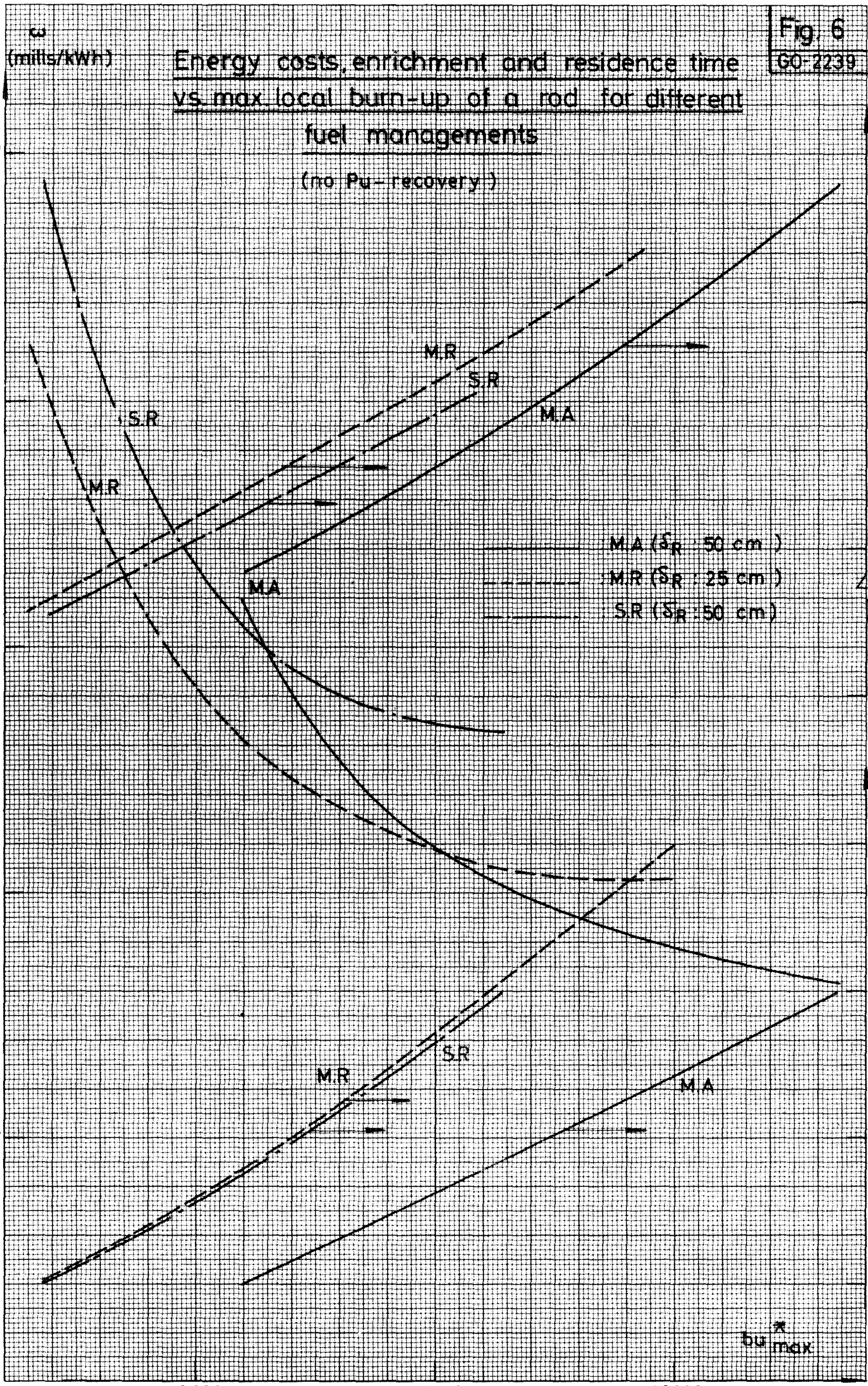


Fig. 6
GO-2239
Energy costs, enrichment and residence time vs. max. local burn-up of a rod for different fuel managements
(no Pu-recovery)
ω (mills/kWh)
6.5
6.0
5.5
5.0
4.5
rt (d)
1000
500
0
y
2.2
1.9
1.6
1.3
M.R
S.R
M.A
M.A (δ_R : 50 cm)
M.R (δ_R : 25 cm)
S.R (δ_R : 50 cm)
bu^{*}_{max}
10000
15000
20000
(MWd/tU)

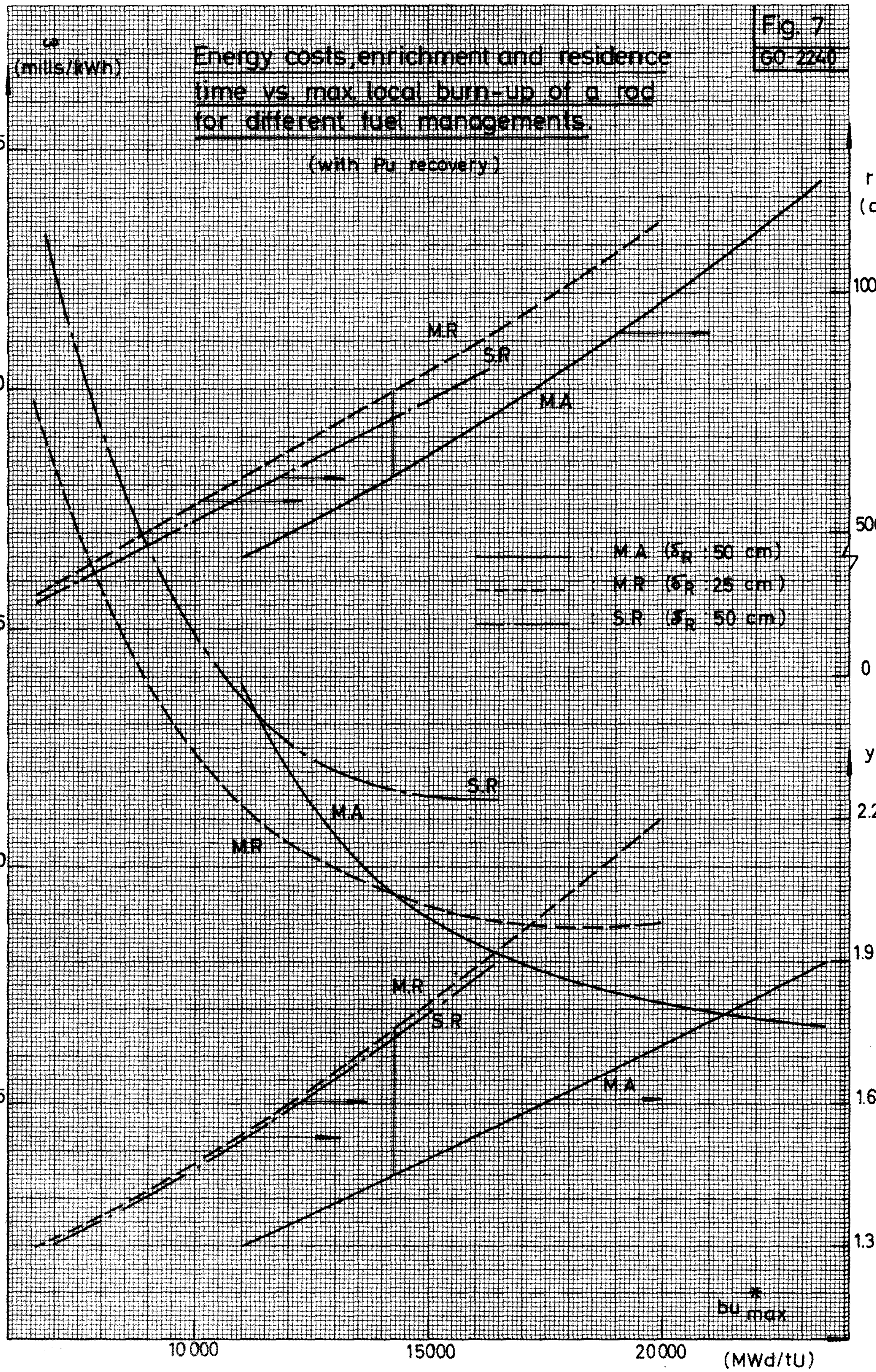


Fig. 7
GO 2240
Energy costs, enrichment and residence time vs. max. local burn-up of a rod for different fuel managements.
(with Pu recovery)
ω (mills/kWh)
6.5
6.0
5.5
5.0
4.5
rt (d)
1000
500
0
y
2.2
1.9
1.6
1.3
M.R
S.R
M.A
M.A (δ_R : 50 cm)
M.R (δ_R : 25 cm)
S.R (δ_R : 50 cm)
S.R
M.A
M.R
M.R
S.R
M.A
10 000
15000
20000
(MWd/tU)
bu^*_{max}

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Alfred Nobel

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