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Institut für Angewandte Systemanalyse

Argentine – German Co-operation on Fuel Cycle Optimization
For the Nuclear Power Plant Atucha
Progress Report 1974

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(Editor)

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Z u s a m m e n f a s s u n g

Dieser Bericht enthält Arbeiten zum Brennstoffkreislauf des Kernkraftwerks Atucha, die im Jahr 1974 unter dem Deutsch-Argentinischen Zusammenarbeitsabkommen durchgeführt wurden.

A b s t r a c t

This paper reports on work related to the fuel cycle of the nuclear power plant at Atucha. The main object is to indicate work performed and progress achieved in 1974 in this field under the Argentine-German co-operation agreement.

Table of Contents

I n t r o d u c t i o n

by

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A basic agreement on co-operation in scientific research and technological development was signed between the governments of the Federal Republic of Germany and of the Republic of Argentina on March 31 st, 1969.

Within the frame work of this basic agreement a special co-operation agreement for the peaceful uses of nuclear energy between the Comision Nacional de Energia Atomica (CNEA) and the Gesellschaft für Kernforschung (GfK) came into effect on July 29 th, 1971.

Under these co-operation agreements it was decided to study in 1974 - among other intems - the possibilities to increase the discharge burn-up values of the fuel elements of the nuclear power plant Atucha(CNA).

Nuclear power reactors using natural uranium as fuel are restricted in their reactivity due to the low concentration of the fissile isotope U-235 in the reactor core. The small reactivity reserve of the reactor demands continuous refuelling and causes the low discharge burn-up values of around 7 MWd/kg typical for the Atucha fuel elements. According to their design these fuel elements could easily endure higher burn-up values, this would mean a reduction of the annual demand for fuel elements, i.e. it would bring about savings in the fuel cycle costs.

To reach higher burn-up values a substancial improvement in the reactivity balance is necessary. There are two practical possibilities to achieve this aim:

The use of enriched fuel throughout the whole reactor core.

The use of special boosters
or spiked fuel elements.

These possibilities and their cost reducing consequences are under investigation in Germany already since the construction of the natural uranium fuelled MZFR-power-plant at Karlsruhe, Germany. A recent survey of these activities and the conclusions reached was given at the Reaktortagung in 1973, held in Karlsruhe (1, 2).

Earlier work related to the possibility of improving the burn-up of the fuel elements for the MZFR-reactor led to a change from natural uranium to 0,85 w/o enriched uranium as a fuel for this reactor. This change came into effect in the mid 1974 after extensive testing of fuel pins and fuel elements containing this enriched uranium fuel.

The experiences at the MZFR and the results of the investigations quoted above give strong evidence that substancial improvements in burn-up values may be achieved also for the Atucha power plant resulting in sizable cost reductions. These facts were the incentives to start studying in more detail the possibilities and consequences of the use of enriched uranium as a fuel for the reactor of the Atucha power plant or alternatively the use of special driver or booster elements.

Consequently in 1974 various activities were set up.

1. Results of MZFR-burn-up experiments were compiled (3) to serve as a first guide.
2. A delegate from CNEA compiled and checked computer programs in Karlsruhe for the ultimate use to perform design calculations in Argentina. As a check the burn-up of natural uranium in the Atucha reactor was recalculated and checked against results of earlier calculations performed at KWU. Good agreement was obtained. Also the burn-up of homogeneously enriched uranium was studied. This work will be reported on separately.
3. Calculations performed at KWU in regard to spiking - either by enriched uranium or by plutonium-bearing fuel - served to clarify technical and economical aspects of these modes of fuelling.
4. The economical aspects of the use of a homogeneously enriched core were treated in a parametric way, which allows to easily read off results for various combinations of burn-up and enrichment values within a meaningful region.

These investigations should be a firm base, frow where further investigations may be derived to study in more detail the technical and economic consequences of the various possibilities of fuelling the Atucha nuclear power plant.

(1) H. Raum:
Technische Eignung und wirtschaftliches Potential von P_2O-Reaktoren in bezug auf Pu-Rückführung.
Reaktortagung Karlsruhe, 1973, S. 733

(2) H. Moldaschl, W. Rupp:
Kontinuierlicher Übergang von Natururan zu leicht angereichertem Brennstoff in Schwerwasserreaktoren.
Reaktortagung Karlsruhe, 1973, S. 737

(3) H. Bogensberger, H.-J. Zech
Burn-up parameters of the MZFR
unpublished memo, Feb. 1974

Fuelling and Spiking of CNA

by

H. Moldaschl

January 1975

Kraftwerk Union AG, Erlangen
Federal Republic of Germany

I. Introduction

It is well known that a high discharge burnup is an essential contribution to increase the economy of heavy water reactors. Therefore it is necessary to optimize the fuelling scheme of the natural uranium core. A further step can be either the homogeneous enrichment of the fuel /¯1_7 or spiking. In the following chapters an empirical optimization process concerning the natural uranium core and the spiking of the core by 1.8 w/o U-235 elements is demonstrated.

II. Natural uranium core, unspiked

The fuelling scheme is mainly influenced by the maximal power peaking factor. This is in contrary to a very steep power density (or neutron flux) distribution in order to decrease the (radial) leakage effects. Therefore the goal of the optimization procedure is to reach the steepest power density distribution which is allowed. To reduce the calculation costs the optimization process is performed in a few stages.

1. Cell calculation

By burnup calculations with the cell code CIRCE /¯2_7, /¯3_7 the reactivity behaviour vs burnup is gained and from that a guess of the mean equilibrium discharge burnup if one assumes all elements to be weighted equally (fig. 1). Then positive and negative reactivity contributions of all fuel elements must be equal, that means as criticality condition the areas A and B to be of equal size. All reactivity losses (control rods etc.) are considered in $\Delta\varrho_L$. A lower leakage effect means a higher excess reactivity and therefore a higher burnup as can be gathered from fig. 1. Naturally the influence of any macroscopic spatial effects cannot be treated by a cell code.

2. Twodimensional calculation

The core is subdivided into 4 exchangeable zones with equal volume and the burnup calculation performed in r-z-geometry. What is the shuffling scheme which maximizes the discharge burnup without hurting the power density limitations ? For the natural uranium core of CNA a two-way scheme
$S_1 = (\xrightarrow{in} \text{zone } 3 \rightarrow \text{zone } 1 \xrightarrow{out}, \xrightarrow{in} \text{zone } 2 \rightarrow \text{zone } 4 \xrightarrow{out})$
should be the optimal one. Fig. 2 shows the subdivision of the core due to the twodimensional burnup calculation (zone 1 = innermost zone, zone 4 = outermost zone).

3. Threedimensional heterogeneous calculation

The calculations have been performed with the threedimensional heterogeneous diffusion burnup code TRISIC [4], [5]. Since single fuel elements or groups of elements can be treated by that code, a subdivision of the core into certain regions is not vital. Using the experiences from the homogeneous calculations, first of all one verifies the scheme S_1. After a certain full power time S_1 yields a neutron flux distribution too flat, which results in a higher leakage and a loss of burnup. A steeper flux distribution can be gained by changing the boundaries of the fictive zones such, that fuel elements with a lower burnup than before are introduced into the two central zones increasing the flux in the center accordingly. This strategy is justified also by the equation

$$\bar{\phi}_i \bar{t}_i = \bar{\phi}_j \bar{t}_j$$

($\overline{\emptyset}_i$ denotes the mean thermal flux in the zone i, $\overline{t}_i$ the mean residence time of the fuel elements in the zone i)
and the fuel mass flow equation

$$\dot{\overline{m}}_i = \dot{\overline{m}}_j$$

which holds for the pairs (i = 3, j = 1) and (i = 2, j = 4).

As the control rods introduce a distorsion of the power distribution the zone boundaries are also distorted. This effect will be one theme of a paper to be published.

Results

Fig. 3 shows the normalized burnup $BU/\overline{BU}$, normalized channel power P/P and normalized power density distribution $Pd/\overline{Pd}$ along a radial trajectory y = 0 (the power density values are those from the core midplane). The normalization has been performed due to the core mean values.

General shuffling directions

a) Due to a two way shuffling scheme each element changes its position two times; only those fuel elements with the highest burnup are moved.

b) The zone boundaries are to be chosen from the beginning of the shuffling period as shown in fig. 2 (scheme S_2). The normalized TRISIC equilibrium burnup distribution $BU/\overline{BU}$ should be approximated as well as possible without hurting the power density limitations. This can be attained also by exchanging certain fuel elements if necessary without the insertion of fresh fuel.

c) Adapting the boundaries corresponding to the flux distortions which are introduced by control rods the optimal core subdivision can be found successively.

d) The mean burnup of those elements which are inserted into the 31 central coolant channel positions should be greater than 3500 MWd/t to avoid unallowed high power densities.

III. Natural uranium core, spiked

An essential increase of the discharge burnup of the natural uranium fuel elements is attainable by a relatively small number of 1.8 $^{w}/o$ U-235 spikes. The influence of the spikes on the burnup and the power distribution has been investigated by TRISIC. The method and the obtained results are summarized in the following chapters.

1. Method

The spikes are assumed to reach two different discharging burnups: BU_{end} = 15000 or BU_{end} = 20000 MWd/t. In both cases the discharged spike has a greater k_{∞} and therefore a better reactivity contribution than the oldest natural fuel element.
The number of spikes in the core has been chosen to be 18, 24 or 30. One criterion of their effectiveness was to place them as uniformly and centrally as possible. To avoid very expensive burnup calculations the start-up period of the core to the spiked eqilibrium has not been calculated. Since BU/fpd (fpd = full power days) at a certain position is proportional to the channel power, an estimate of the spike residence time at each position can be calculated for different loading schemes using the channel powers from the natural uranium core:

(examples of loading schemes: →0→0→0→, →0→0→0→0→,...)

If one also gets an estimate of the axial burnup distribution (e.g. from the natural uranium calculations), the threedimensional calculation of the spike effects can be carried out. Because the reactivity maximum appears immediately after the spike shuffling operation, the burnup calculations of the spiked reactor are started from that point, the spikes having the mean burnup shown in the table. An estimate of the burnup increase now can be obtained by a depletion calculation down to the original reactivity value (= reactivity of the unspiked reactor). Since in CNA the mean core burnup $\overline{BU}_{core}$ and the mean discharge burnup $\overline{BU}_{out}$ obey the equation

$$\overline{BU}_{out} \cong 1.8\ \overline{BU}_{core}$$

$\overline{BU}_{out}$ can be determined from $\overline{BU}_{core}$. The flattening of the flux distribution is equivalent to latent reactivity reserve which can be estimated by shuffling the natural fuel elements in such a way, that a flux distribution similar to that of the unspiked core is obtained.

2. Results

Fig. 4 shows a sextant of the CNA core, the denumeration of the coolant channels is equivalent to that in the table. The scheme $\begin{matrix} i \\ j \\ k \\ \cdot \\ \cdot \\ \cdot \end{matrix}$ designates the way of the spikes (i = insertion position of the fresh spikes). $\overline{BU}_i$ is the mean burnup of the spike, at the moment it is inserted into the position i, BU_{end} denotes the discharge burnup of the spike, BU_s/BU_u the increase of the natural uranium fuel burnup by spiking the core.

Table: CNA spiking, 1.8 $^{w}/o$ U-235

Case number	Number of spikes	Shuffling scheme	$\overline{BU}_i$	BU_{end}	BU_s/BU_u
1	18	31 26 20	0 3850 8950	15000	1.39
2	18	31 26 20	0 5130 11940	20000	1.35
3	24	33 28 24 18	0 3490 8100 13520	20000	1.39
4	24	29 28 27 17	0 4040 8680 13290	20000	1.37
5	24	29 28 25 15	0 3830 8230 13100	20000	1.39
6	30	29 28 27 19 18	0 3060 6580 10080 15050	20000	1.43

In fig. 5 the normalized burnup, the power and power density distribution of the natural uranium fuel elements are drawn along the radial trajectory y = 0 for different core states:

a) immediately after spike shuffling (reactivity maximum);

b) after burning down until the initial reactivity (= reactivity of the unspiked core) is reached; flat flux distribution;

c) the same state as in b), but after shuffling the natural uranium fuel elements such, that the initial flux distribution (of the unspiked core) is approximated; notice the lower burnup in the core center.

3. General remarks on the natural uranium fuel element shuffling

The changes in the shuffling scheme of the natural uranium fuel elements should be small due to the similar flux distribution. The mean burnup in the central zone may be somewhat smaller in comparison to the unspiked core and the burnup in the neighbourhood of the spikes higher to reduce the power density in the spikes.

Literature

[1] H. Moldaschl, W. Rupp: Kontinuierlicher Übergang von Natururan zu leicht angereichertem Brennstoff in Schwerwasserreaktoren.
Reaktortagung DAtF 1973, Karlsruhe, p. 737

[2] H. Raum: CIRCE - ein neues Zellrechnungsprogramm
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[3] C. Grant, H. Moldaschl, R. Solanilla: Application of a Heterogeneous Method in the Simulation of Heavy Water Reactor Operation.
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[4] H. Moldaschl: Entwicklung eines heterogenen 3-dimensionalen Abbrandprogramms für D_2O-Druckwasserreaktoren.
Abschlußbericht zum Vorhaben 1.T.4/5 im Rahmen "Weiterentwicklung der Technologie von Schwerwasserreaktoren"
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[5] C. Grant, H. Moldaschl, R. Solanilla: Erweiterung und Erfahrungen mit dem 3-dimensionalen Abbrandprogramm REFLOS-S für D_2O-Druckwasserreaktoren.
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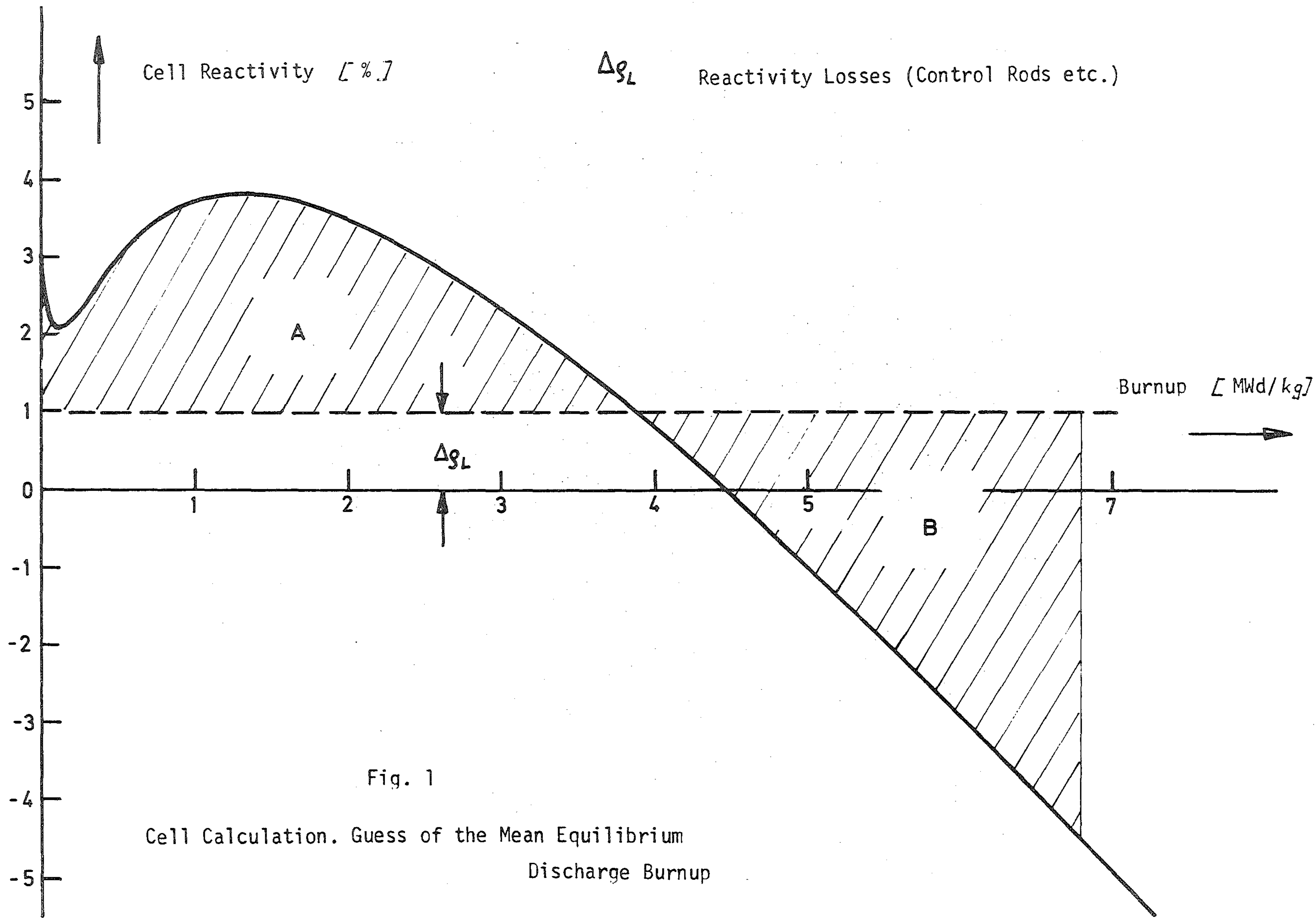


Fig. 1

Cell Calculation. Guess of the Mean Equilibrium Discharge Burnup

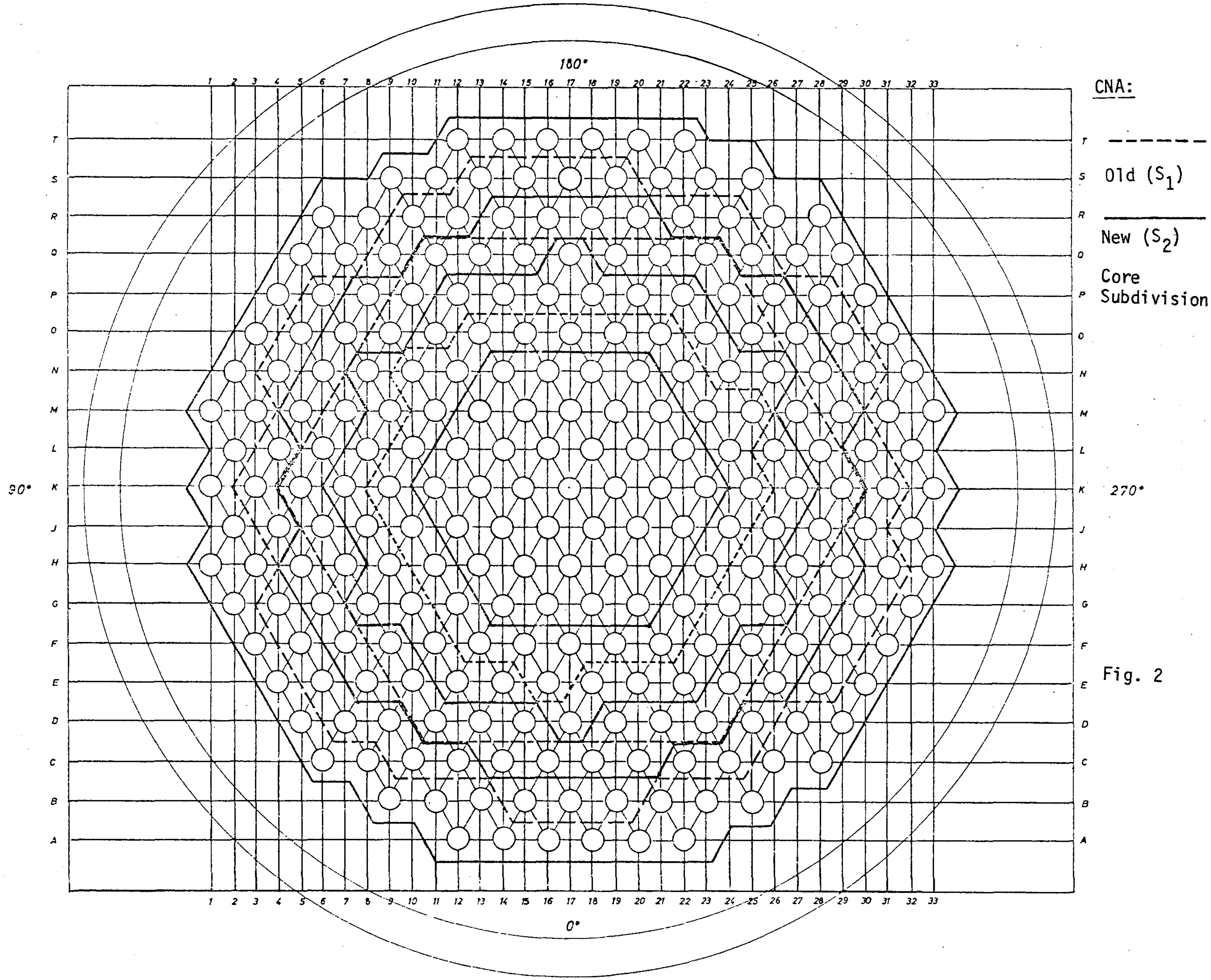
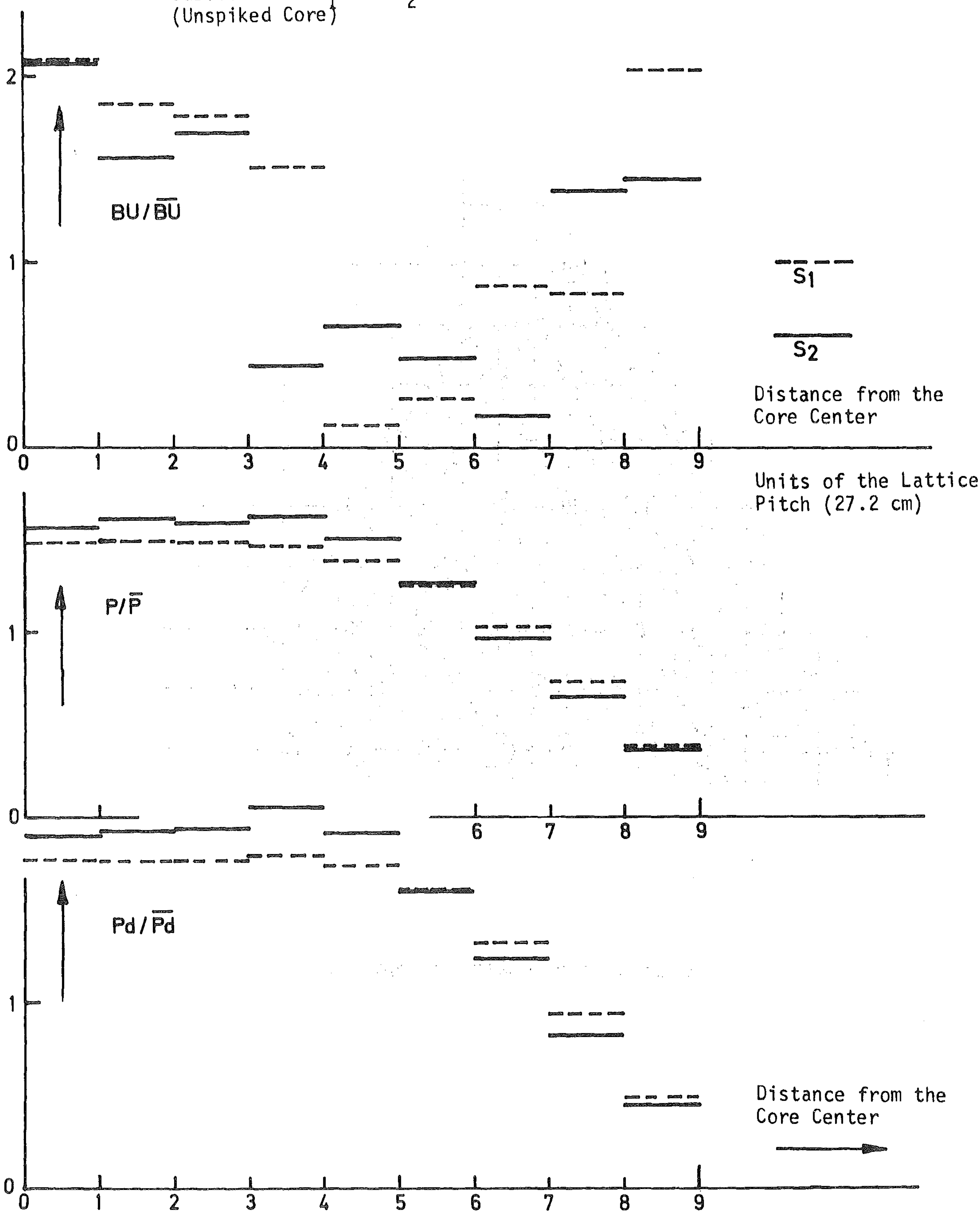


Fig. 2

Fig. 3: Normalized Burnup, Channel Power and Power Density Distribution along the Radial Trajectory y=0 for the Core Subdivisions S_1 and S_2 (Unspiked Core)



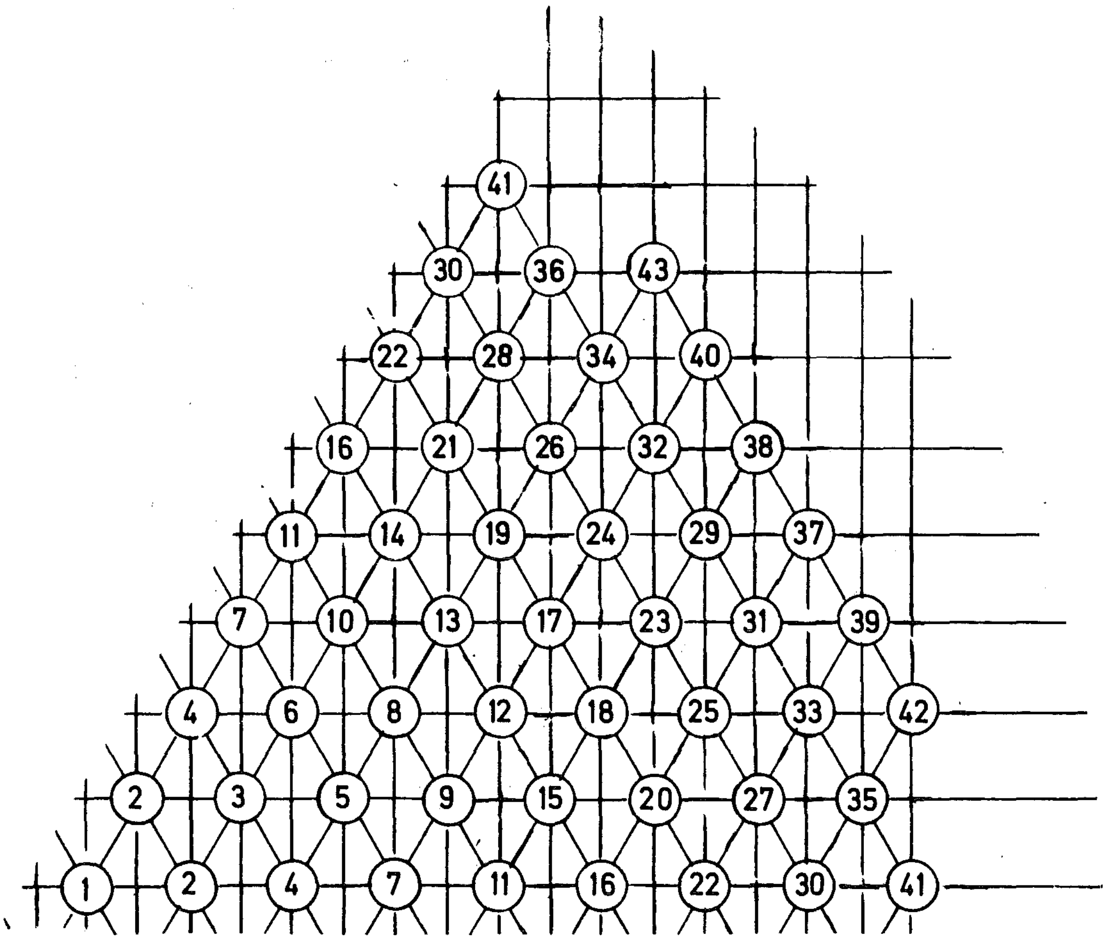
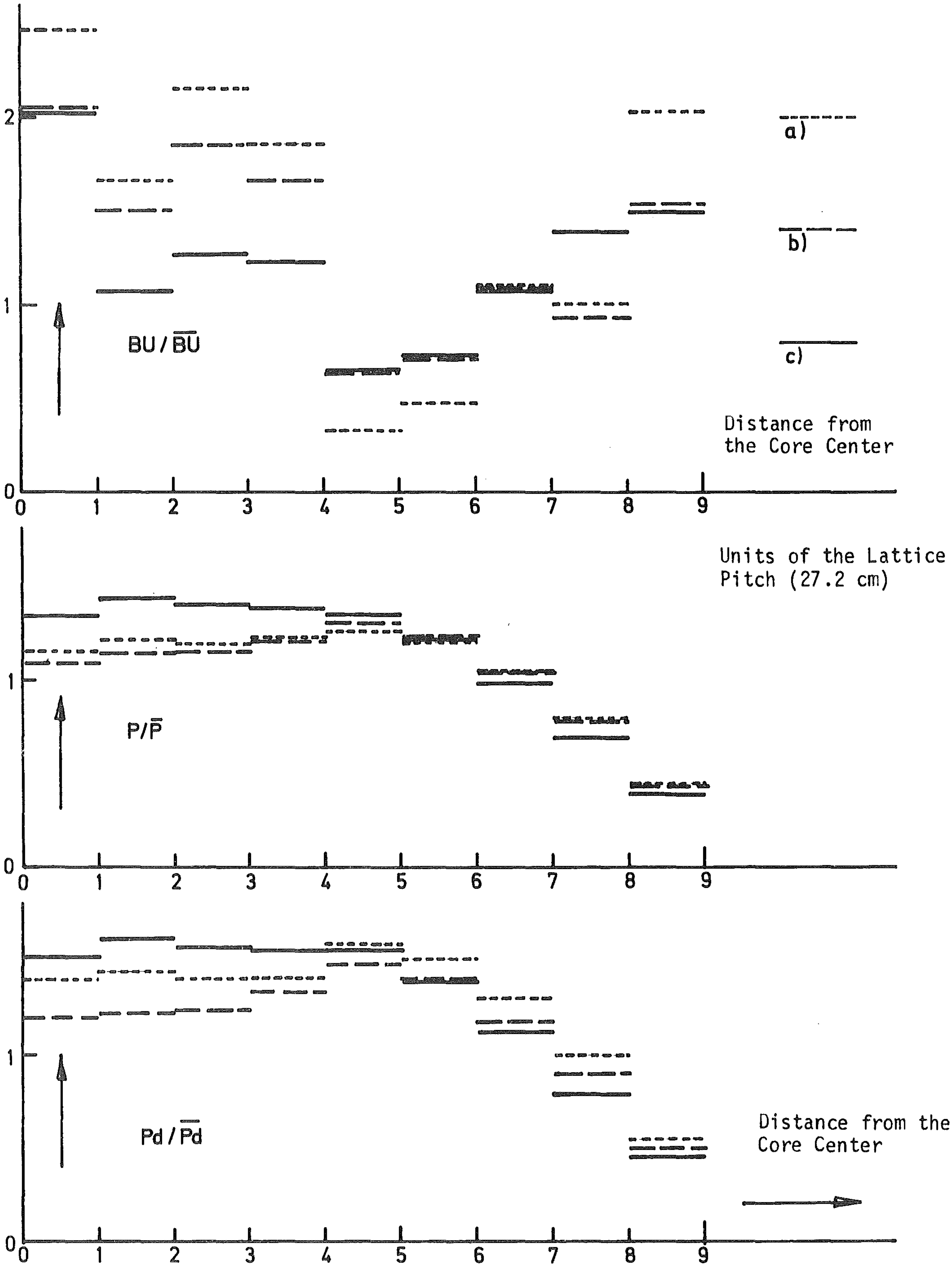


Fig. 4: Channel Group Numbers for Spiking

Fig. 5: Normalized Burnup, Channel Power and Density Distribution along the Radial Trajectory y=0 for the Different States a, b, and c Described in the Paper



Pu-Spikes in CNA

by

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January 1975

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1. Introduction

A reduction of the fuel cycle costs of a D_2O-reactor appears to be feasible by slightly enriching the fuel /1, 2/. Though this fact has been well known for many years it is scarcely used until now in practice. The presumable reason is primarily, that one of the most important features of a D_2O-reactor is lost by enriching the fuel, i.e., the independence of foreign enrichment services.

There is however a way to achieve lower fuel cycle costs without losing economical independence, namely the use of domestic natural uranium together with plutonium which will soon be available in many countries and presumably for a rather favourable price. There are two ways to increase burnup by Pu-enrichment:

1. a homogeneous distribution of Pu over all elements of the core,
2. the concentration of Pu in only a certain percentage of all fuel elements, in socalled "spike elements"

The first possibility yields indeed the highest achievable burnup, but it is ruled out by economical reasons since the radiotoxicity of Pu demands incisive precautions leading to a considerable increase of fabrication costs of Pu-containing fuel elements which - together with the increased fuel costs due to the use of Pu - at least compensates the gain by increasing the burnup. For this reason the second possibility appears to be preferable.

/1/ Raum, H.: Technische Eignung und wirtschaftliches Potential von D_2O-Reaktoren in bezug auf Pu-Rückführung, Reaktortag 1973, Karlsruhe p. 733

/2/ Moldaschl, H., Rupp, W.: Kontinuierlicher Übergang von Natururan zu leicht angereichertem Brennstoff in Schwerwasserreaktoren. Reaktortag 1973, Karlsruhe, p. 737

2. General Features of Spiking

In the case of spiking a reactor with Pu-elements, only these spike elements involve increased fabrication costs. Their number should therefore not exceed a reasonable limit. On the other hand, it is just the Pu of these spike elements which causes the increase of burnup not only of these elements but also of the natural uranium elements. That means, the amount of Pu inserted in the spikes should be as large as possible.

These two demands - not too many spike elements because of the fabrication costs, but as much Pu as possible because of the burnup - together with the desire of generating as much power as possible - within the thermodynamic and technologic limits of course - lead to the main features for the design of a spiked reactor.

1. The Pu-content of the spike elements should be chosen so that the power density in a just inserted fresh Pu-element does not exceed the thermodynamic limit, that means, that the total power shape factor of the core achieves the highest permissible value.

   Due to the rather high Pu-content the thermal flux depression within the bundle is rather high. In order to generate as much power as possible the power density distribution should be, however, similarly flat as in a natural uranium bundle throughout the residence time of the spikes. This demand is fulfilled by enriching the pins of different rings of the bundle to different Pu-contents in order to compensate the lower flux in the bundle centre by a higher fission cross section.

2. The positions of spike elements and the refuelling scheme should be chosen in such a way that the reactivity and power contribution of these elements during all their life is as high as permissible. This demand can be realized by inserting the fresh Pu-elements at the core periphery and

by shuffling them closer to the centre as soon as their power has decreased enough. The fresh natural uranium elements on the other hand are inserted near the core centre and than shuffled to the periphery. This scheme guarantees the maximum burnup for the uranium elements. Since the purpose of Pu-spikes is mainly to increase the burnup of uranium elements they are removed from the core when their reactivity contribution has decreased to zero.

A further demand to the positions of spike elements is that their power density must not be increased by more than 20 % by radial shuffling in order to avoid too much swelling and fission gas release in the fuel.

3. The number of Pu-spikes can be fixed by the demand that the central natural uranium elements should produce the maximum permissible power immediately before inserting fresh Pu-elements. Immediately afterwards this maximal power is generated by the just inserted and shuffled Pu-elements. In this way the time averaged power distribution becomes very flat and the produced total power achieves a maximal value. If the number of Pu-elements is too large or too small the power of the central channels is too low or too high and the power distribution is not optimal.

3. Nuclear Design of Pu-Spikes for CNA

Starting from these basic features we designed Pu-spike elements for the nuclear power plant Atucha (CNA). All considerations refer to the burnup equilibrium state of the spiked core; the transition from the natural uranium core to the spiked core will be mentioned only briefly later on.

The Pu is assumed to be taken from discharged elements of H_2O-reactors, that means the percentage of fissile nuclids is about 70 %.

3.1. Optimization of Pu-Distribution within the Bundle

As mentioned above the power distribution within a Pu-element should be similarly flat as in a corresponding natural uranium element. That means for the case of CNA, the maximum power density in the bundle may not exceed the mean value at more than about 10 %. In contrasc to U_{nat}-elements, the power densities in the different pin rings of an element, containing more than about 1.5 % Pu, diverge considerably with increasing burnup. Therefore the optimization of the Pu-distribution must take into account the total incore time. It is not sufficient to consider only the fresh fuel as it would be for very low enrichments.

The calculations of burnup-dependent power density distributions within the bundles for various Pu distributions were performed with the cell calculation code CIRTHE, a combination of CIRCE /3/ and the thermalization code ESE /4/, an improved version of THERMOPYL /5/. The bundle was subdivided into three zones containing the outermost pin ring, the middle pin ring and the innermost pin ring together with the central pin. The following values are the result of this calculation; they give the relative optimum Pu-distribution for mean Pu-enrichments $\bar{\gamma}_{Pu}$ from 2.5 to 3.2 %:

$$\frac{\gamma_{Pu}^{\text{zone 1}}}{\bar{\gamma}_{Pu}} = 1.37 \qquad \frac{\gamma_{Pu}^{\text{zone 2}}}{\bar{\gamma}_{Pu}} = 1.24 \qquad \frac{\gamma_{Pu}^{\text{zone 3}}}{\bar{\gamma}_{Pu}} = 0.68$$

/3/ Raum, H.: CIRCE, ein neues Zellrechnungsprogramm. Reaktortagung des deutschen Atomforums, Mainz, 1967

/4/ Solanilla, R.: Use of Secondary Model for Calculating the Thermal Neutron Spectrum in Multiannular Cylindrical Geometry and Applications. Unpublished Siemens Report, June 1970

/5/ Märkl, H., Raum, H.: THERMOPYL, a Program for Calculating the Thermal Neutron Energy Spectrum in Heterogeneous Lattice Cells. Nukleonik, Band 7/5 1965, p. 247

The ratio of maximum to average power density in the bundle during the total incore time ranges from 1.09 to 1.12 for $\bar{\gamma}_{Pu}$ = 2.5 and 3.2 respectively. That means that the upper limit for the average Pu-enrichment of CNA-elements is about 3 %. Higher values would result in inadmissibly high power densities at some burnup.

Fig. 1 shows the change of the power density distribution within the bundle with increasing mean burnup for an average Pu-content of 3.2 %. The figure points out clearly as the "hot spot" is passing over from the outermost to the innermost pin ring in course of incore time. At each time, however, the ratio of maximum to average power density remains below 1.12. Each other Pu-distribution would yield a value of more than 1.12 at some time intervall. Hence, the Pu-distribution given above is an optimum with regard to the time dependent power density distribution in the bundle.

3.2. Optimization of Spike Distribution within the Core

The selection of suitable positions for the Pu-elements within the reactor core is unseparably connected with the selection of a refuelling scheme for both Pu- and U_{nat}-elements that fulfills all demands fixed in chapter 2. Therefore this task can only be solved by performing extensive time dependent calculations. We used for this purpose the three dimensional heterogeneous code TRISIC /6/; the lattice constants wanted by this program were calculated by CIRTHE.

As a result we found that 48 Pu-elements with an average Pu-content of 3 % and a Pu-distribution given above can be inserted into the CNA-core. The refuelling scheme that guarantees an economically optimal utilization of the fuel

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/6/ Grant, C., Solanilla, R., Moldaschl, H.: Application of a Heterogeneous Method in the Simulation of Heavy-Water-Reactor Operation.
IAEA Panel on "Reactor Burn-up Physics", Wien 1971

within the permissible thermodynamic limits is represented in figure 2, which shows a sextant of the CNA-core together with the position of the spikes. The two kinds of fuel elements, natural uranium and Pu-enriched elements, are shuffled in two opposite ways: the spikes from the periphery to channels nearer to the centre, and natural uranium elements from central positions to the edge.

Fresh Pu-elements are charged into channels at the edge of the reactor core. After having reached a burnup increment of about 7000 MWd/tU, they are shuffled to a neighbouring position closer to the core centre. In this way the Pu-elements are shifted altogether three times; then, having reached a burnup of about 28000 MWd/tU they are discharged. The mean power produced by a Pu-element during its life is about 6.5 MW.

Between charging and discharging a channel, the power can change from 7.5 to 5.9 MW. The in-core time amounts to 690 full power days and the percentage of discharged Pu-elements relative to all discharged elements is about 11 %. This means on the other hand, that 11 % of all elements to be fabricated must contain Pu. This number is markedly less than the percentage of core positions occupied by spikes which is about 19 %. The Pu-demand amounts to about 80 kg fissile Pu per full power year.

Every 14 full power days the Pu-element with the highest burnup is discharged, that one with the next lower burnup in the neighbouring position nearer to the edge is shuffled into this now empty channel and so on, until the outermost Pu-element is shifted. Then, this channel at the edge is refuelled by a fresh Pu-element.

The channels containing natural uranium elements are subdivided into three zones with 69 elements each, apart from the central channel, as shown in figure 2. After having discharged the element with the highest burnup out of the outermost zone, one element of the middle zone is shuffled into this now empty position and the element with the highest burnup of the inner zone is shifted to the middle zone. Then, a fresh element is charged into the empty channel of the central zone.

In this way an average discharge burnup of the natural uranium elements of about 9000 MWd/tU is achieved.

The in-core residence time of the natural uranium elements is about 400 full power days and about every second day a fresh element has to be charged into the core in order to keep the reactor critical.

The reactivity changes in the equilibrium state due to refuelling of coolant channels do not exceed 0.2 %, this value referring to the event of exchanging a Pu-element.

In the same way an arrangement with 66 spikes containing 2.6 % Pu on an average was investigated. The distribution of the Pu-elements within the CNA-core and the refuelling scheme are illustrated in figure 3 which again shows a sextant of the core. Starting from three outermost positions the Pu-elements are shuffled along three branches nearer to the centre of the core. There are only two innermost positions for Pu-elements which are fuelled alternately from the branch to which they belong and from that on the symmetry axis. Every ten full power days a Pu-element has to be inserted. The U_{nat}-elements are shuffled in a similar way as in the case discussed above.

The percentage of Pu containing elements is about 17 % of all elements that are fabricated, the Pu-demand per full power year is about 100 kg Pu_{fiss}. The discharge burnup of Pu- and U_{nat}-elements and the time they are staying within the core are very similar to the values given above.

4. Transition to the Equilibrium State

All considerations and values given up to here are referring to the burnup equilibrium state with Pu-elements. The transition from the core containing only U_{nat}-elements to the state with Pu-elements described above, cannot be treated as generalized as the equilibrium state itself. It sensitively depends on the actual state of the U_{nat}-core into which the first Pu-elements has to be inserted or - other spoken - on the actual moment of the first insertion of a Pu-element. Hence, only some rough considerations can be carried out here.
The first Pu-elements have to be inserted of course into the outer region of the core. Because of the high power production of these elements, the throttles of coolant channels which are selected for Pu-elements have to be removed. Since this is a rather hard work, it appears to be expedient to use from the very beginning of Pu-recycling throughout all the transition to equilibrium state just those positions for insertion of Pu-elements that are already provided for this object for the equilibrium state.

As soon as the power density in a Pu-element has decreased enough, it is shifted nearer to the centre of the core and a fresh Pu-element is charged into a channel at the edge. The high power of the Pu-elements at the edge of the core permits or even demands to insert fresh U_{nat}-elements or shuffle almost fresh U_{nat}-elements into the central region. When a fresh element is inserted into the core the element with the highest burnup apart from Pu-elements is removed and if necessary, further elements are shifted in an appropriate manner. Pu-elements are not discharged before they reached a burnup of about 28000 MWd/tU.

Possibly a detailed calculation of the transition to the equilibrium state for the actual case will give rise to difficulties because of too high power densities in some

Pu-elements that cannot be overcome by appropriately inserting or shuffling U_{nat}-elements. A recycling of Pu into the CNA-reactor would not be prevented by that but only complicated. During the transition to the equilibrium state then Pu-elements with lower enrichment than in equilibrium had to be fabricated and inserted, at least for some times.

5. Conclusions

Thus it appears technically feasible at any rate to spike the CNA-core with Pu-elements. Their percentage relative to all inserted elements will range about 20 to 25 %, their enrichment about 2.5 to 3.0 % Pu with 70 % fissile nuclids. The Pu-content within an element must be graduated in an appropriate manner. The construction of Pu-elements will be very similar to that of U_{nat}-elements, only minor changes - such as a thicker canning and a larger fission gas plenum - will be inevitable because of the markedly higher burn-up of these elements.

The fuel throughput of the spiked CNA-reactor, expressed in fuel elements or fuel metal mass per full power year, is estimated to be reduced by up to 40 % compared with the U_{nat}-core. In return, the spiked reactor will consume up to 100 kg Pu_{fis} per full power year. About 30 to 40 % of this Pu-demand may be covered by reprocessing the discharged spike elements only. If all discharged elements are reprocessed, a nearly self-generating Pu-recycling could be sustained.

Up to now detailed economic calculations have not been performed. However, the saving of fuel cycle costs by spiking the CNA-core can be estimated to amount up to 25 % depending on the assumptions made concerning the Pu-price and the fabrication costs for Pu-elements.

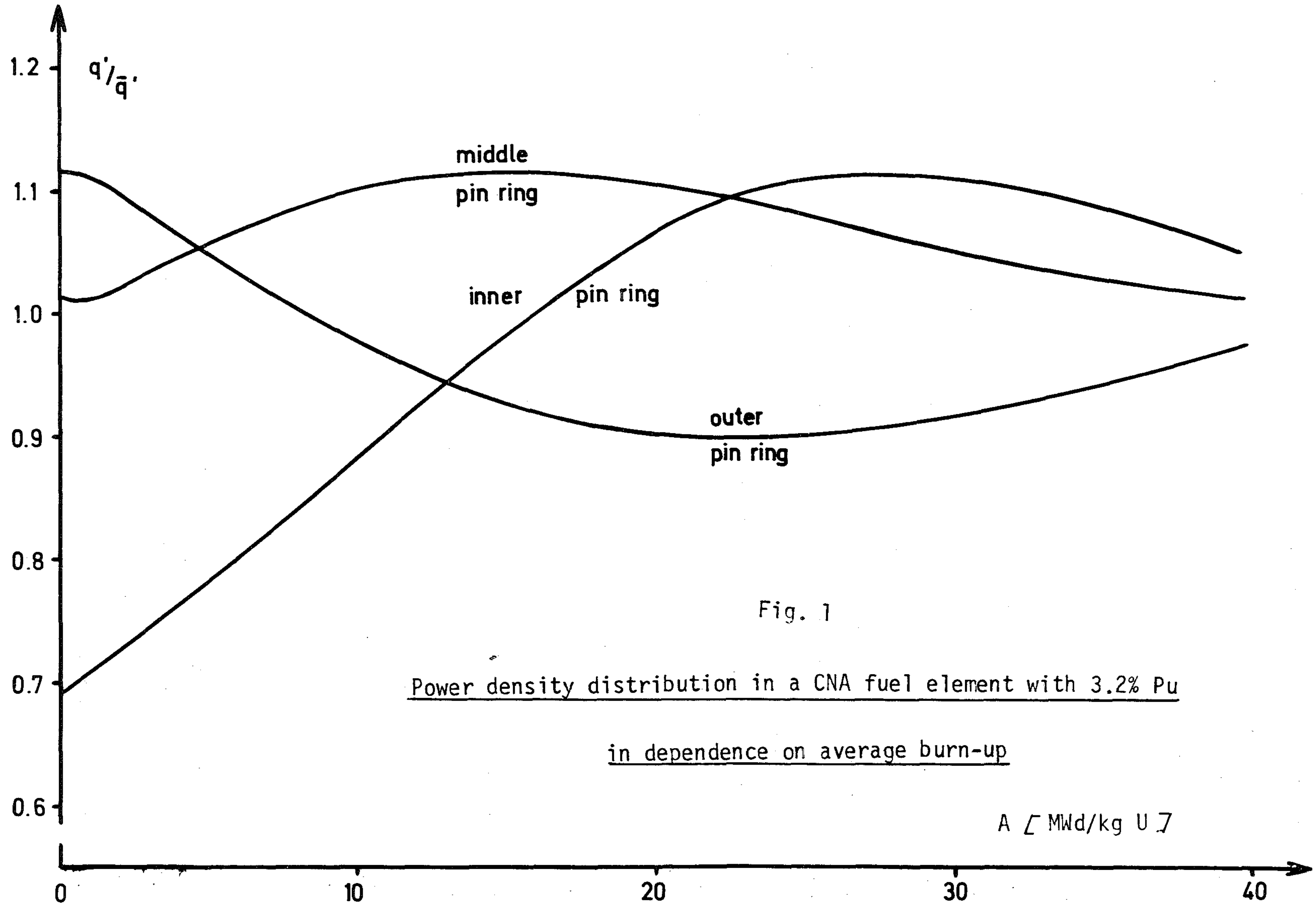

Fig. 1

Power density distribution in a CNA fuel element with 3.2% Pu

in dependence on average burn-up

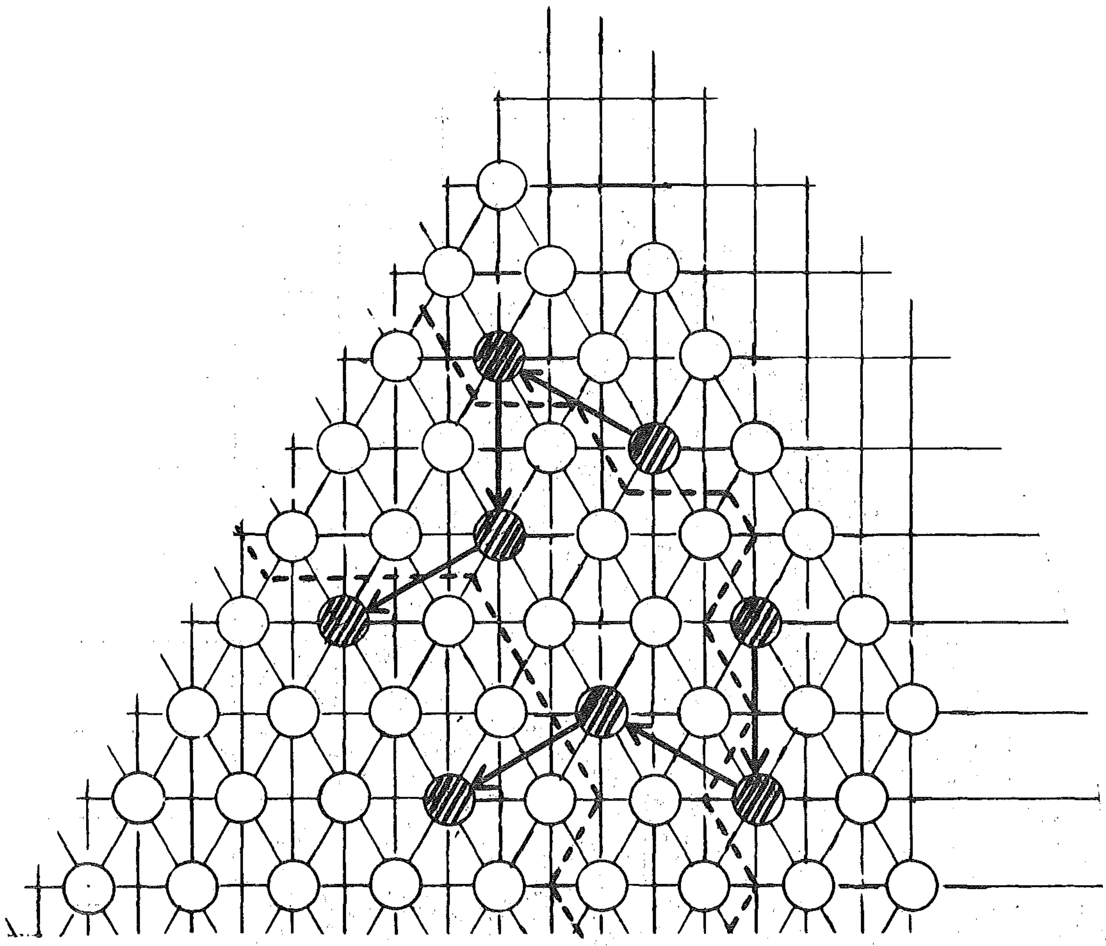

Fig. 2

Arrangement and shuffling scheme of 48 Pu-elements

in one sextant of the CNA-core

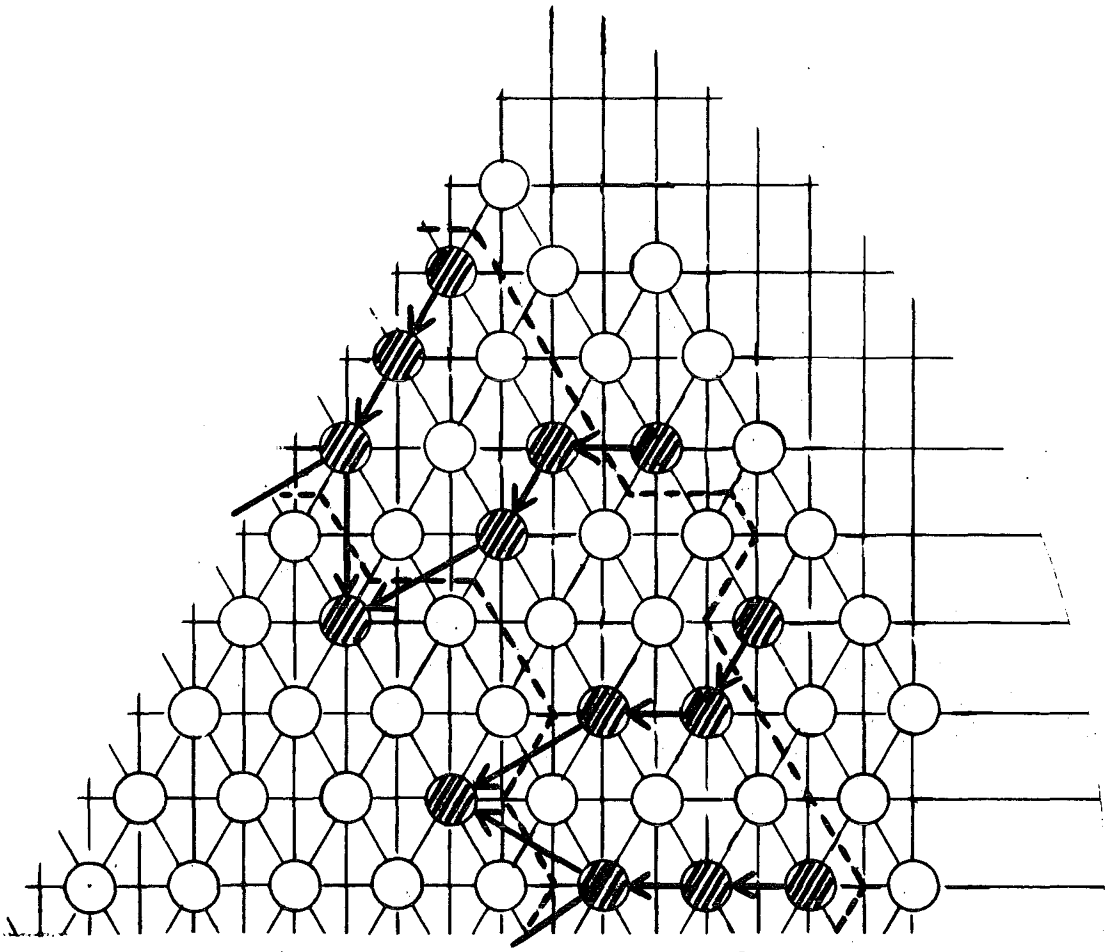

Fig. 3

Arrangement and shuffling scheme of 66 Pu-elements in one sextant of the CNA-core

The Nuclear Power Plant Atucha with Homogeneously Enriched Fuel

General Aspects of Fuelling

by

F.K. Pickert
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September 1974

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1. Introduction and Task

Calculations performed by KWU indicate, that an improvement in fuel cycle performance of MZFR-type reactors may be achieved by using enriched uranium fuel or by spiking the U_{nat}-fuelled core with special Pu-bearing or highly enriched uranium fuel elements.

Some results of these calculations were published at the Reaktortag 1973 /1,2/. For homogenous enrichment of the core the results given by Moldaschl and Rupp /2/ show a possible increase of the discharge burn-up in the MZFR from around 6000 MWd/t for the Unat-fuel to around 10 000 MWd/t for fuel with 0.85 w/o U-235. The average core burn-up is nearly doubled, from below 3000 MWd/t it can be increased to 6100 MWd/t.

Similar calculations for CNA indicate the possibility to reach a discharge burn-up of 15 000 - 20 000 MWd/t with enrichment of 0.9-1.0 $^{w}/o$ U-235.

Experiences with a small number of test fuel elements in the MZFR with an enrichment of 1.1 $^{w}/o$ U-235 yielding a discharge burn-up of about 12 500 MWd/t are in agreement with the results of the calculations referred to above: Correcting for an enrichment of 0.85 $^{w}/o$ U-235 one obtains burn-up values of about 9 000 MWd/t

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/1/ Raum, H: Technische Eignung und wirtschaftliches Potential von D_2O- Reaktoren in bezug auf Pu-Rückführung, Reaktortag 1973, Karlsruhe S. 733

/2/ Moldaschl, H., Rupp, W.: Kontiunierlicher Übergang von Natururan zu leicht angereichertem Brennstoff in Schwerwasserreaktoren. Reaktortag 1973, Karlsruhe, S. 737

/3/, this value is to be compared with the figure of 10 000 MWd/t quoted in /2/.

Results of computations investigating the possibilities of Plutonium-bearing spike elements yielded burn-up values of around 10 000 MWd/t for the fuel elements containing natural uranium /1/.

The possibilities to increase the discharge burn-up value of fuel elements containing natural uranium have direct consequences on the economy of the plant operation.
The purpose of this paper is to demonstrate the important consequences of increased burn-up on the fuel cycle. To accomplish this various relevant quantities and cost figures are plotted vs a plane, characterized by enrichment and burn-up values. The quantities investigated are primarily

- the annual demand for fuel elements
- the annual demand for natural uranium
- the annual demand for separative work
- the effective use of the natural uranium
- the annual expenditures for U_3O_8
- the annual expenditures for conversion and enrichment
- the annual expenditures for fuel element fabrication
- the total annual expenditures for fuel.

2. Investigations performed and Results.

2.1. Input data

The calculations are based on the following input data

/3/ Bogensberger, H., Zech, H.-J.: Burn-up parameters of the MZFR (Feb. 1974), Unpublished memo.

Plant:	Thermal power	1.100 MW_{th}
	Net electric output	319 MW_e
	Load Factor	7.000 h/a
Fuel:	Uranium content per fuel element (net uranium demand)	152.4 kg
	supplement for compensation of fabrication losses	3% of the net demand
	supplement for compensation of conversion losses	6% of the feed demand
	tails assay	0.2 wt.% U-235
Costs:	Natural uranium	8 \$/lb U_3O_8
	Conversion	2,70 \$/kg U in UF_6
	Separative work	38 \$/kg SWU
	F.E. fabrication	121 \$/kg U in F.E.

The costs assumed are those of the first half of 1974 /4/. Price increases as for natural uranium and for separative work that came into effect in the mean time are not yet taken in account. As the results of this paper are mainly intended for a first orientation, we believe that a recalculation with up-dated cost values is not necessary at this stage.

2.2. Results in General

Summarizing and with a view to the individual results to be described in more detail later in this paper it can be said that the amounts and costs of fuel supply are susceptible to respond in an extremely positive manner to the improvement of burn-up. In principle, this is a confirmation of what had been expected on account of the high fabrication costs and the relatively low reactivity burn-up attainable of the natural uranium elements.

/4/ Price for fuel element fabrication is taken from an unpublished memo by DiPrimio and Zech, "Fuel Cycle Cost of the Nuclear Power Plant at Atucha", Karlsruhe, May 1974

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/4/ Price for fuel element fabrication is taken from an unpublished memo by DiPrimio and Zech, "Fuel Cycle Cost of the Nuclear Power Plant at Atucha", Karlsruhe, May 1974

If in the following sections figures are given on the behavior of the flows of amounts and costs indicated above, it should be stressed here that these figures will describe only the favorable trend initiated by a slightly homogeneous enrichment. In our opinion these figures are not a substitute for detailed economic studies, which will be reasonable only when (1) more accurate physical computations will have yielded the possible enrichment of the fuel and the burn-up achievable with it and (2) the necessary investigations will have been made on the core spiked with plutonium and/or more highly enriched uranium.

2.3. Annual Demand for Fuel Elements

A burn-up of 7 MWd/kg U attainable under equilibrium conditions in natural uranium operation effects an annual demand for about 300 fuel elements under the conditions mentioned in the introduction. If, however, a burn-up of 12 MWd/kg U is reached, the annual demand for fuel elements is reduced to 175 fuel elements. Fig. 1 shows very clearly the behavior of the demand for fuel elements as a function of burn-up. (cf. p. 4a)

The strong reduction of the fuel element demand entails savings in amounts and costs, which compensate over a broad range opposing developments originating in the fuel enrichment.

While the annual demand for fuel elements depends only on the burn-up attainable, the values of the flows of amounts and costs to be studied later in this paper are determined both by the burn-up and the degree of enrichment, with the degree of enrichment having an impact on the increase in amounts and costs, as already mentioned, while the increase in burn-up acts in the opposite direction.

2.4. Annual Demand for Natural Uranium

With the natural uranium version and the burn-up of 7 MWd/kg U associated to it roughly 123.000 lbs U_3O_8 are required annually to

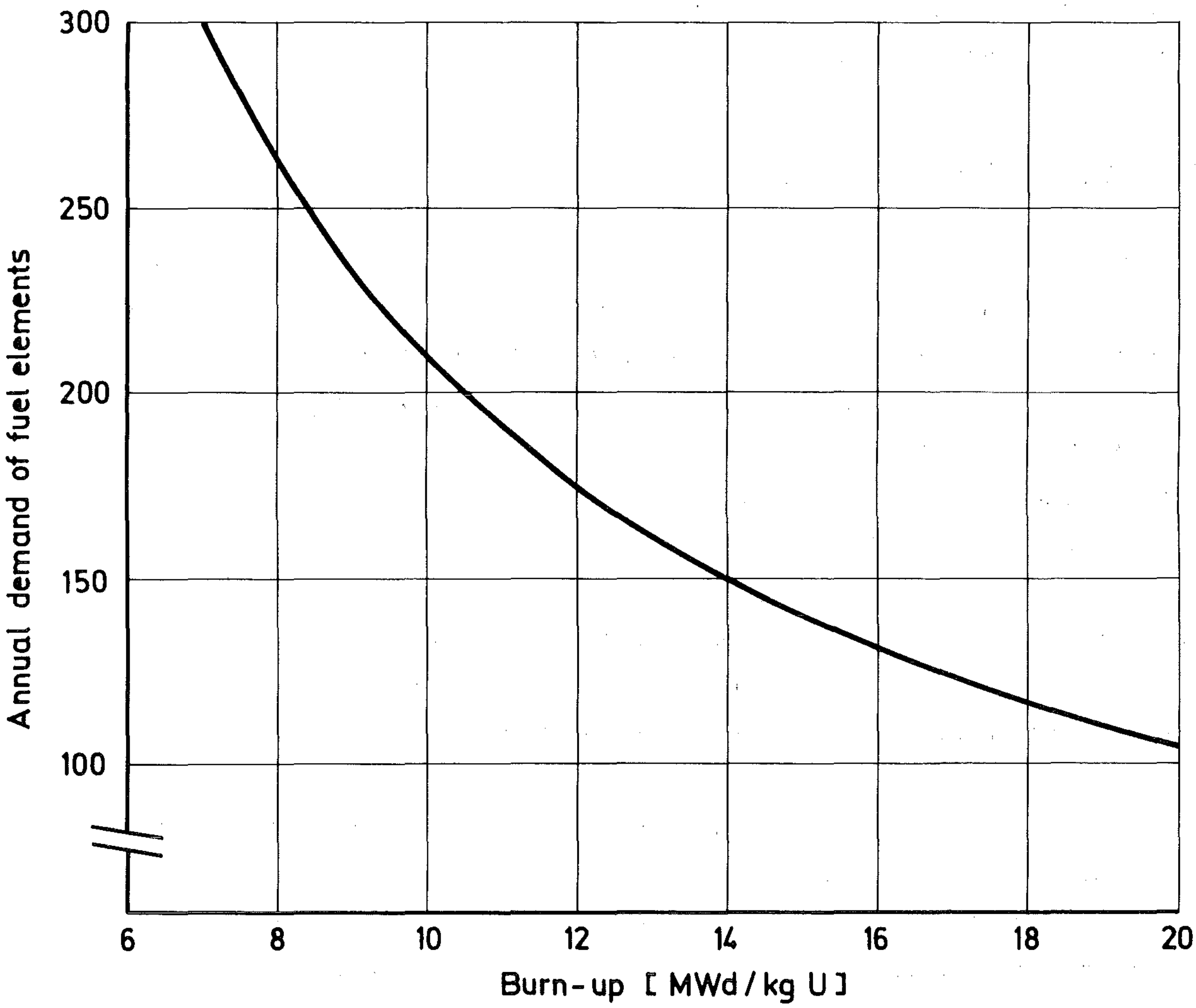


Fig.1 Annual Demand of Fuel Elements

Table 1

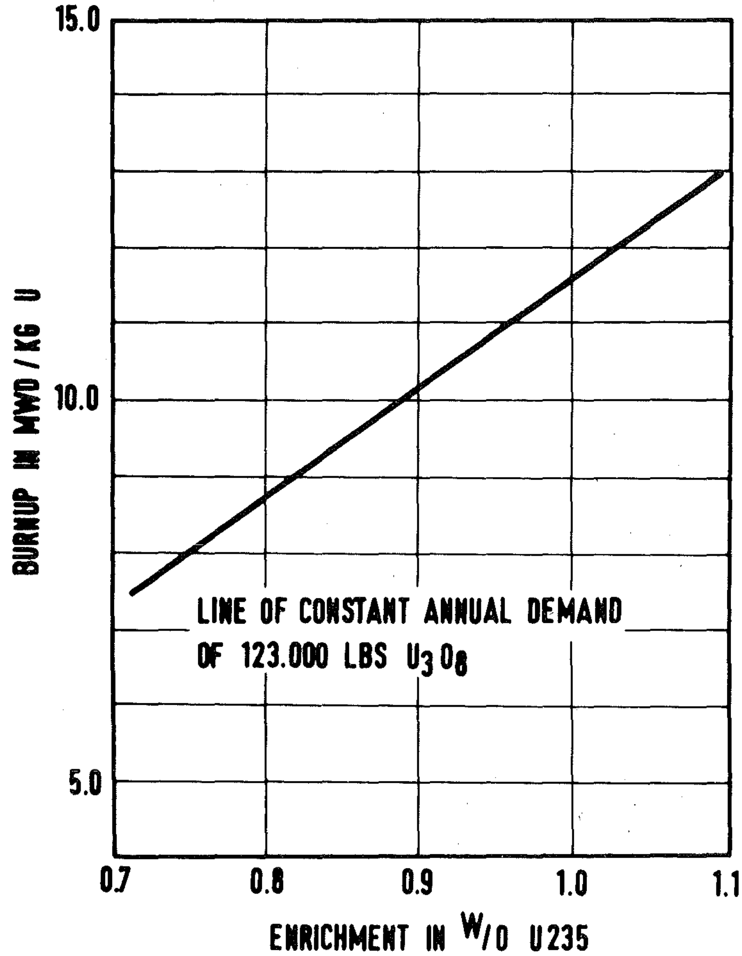
Annual Demand of U_3O_8 in 1 000 lbs as
function of enrichment (ENR) in w/o U-235 and
burn-up in MWd/kg U

ENR w/o U-235	Burn-up /MWd/kg U/													
	7	8	9	10	11	12	13	14	15	16	17	18	19	20
0,711	123													
0,750	140	123	109	98	89	82	75	70	65	61	58	54	52	49
0,775	146	128	114	102	93	85	79	73	68	64	60	57	54	51
0,800	153	134	119	107	97	89	82	76	71	67	63	59	56	53
0,825	159	139	124	111	101	93	86	80	74	70	66	62	59	56
0,850	165	145	129	116	105	97	89	83	77	72	68	64	61	58
0,875	172	150	134	120	109	100	93	86	80	75	71	67	63	60
0,900	178	156	139	125	113	104	96	89	83	78	73	69	66	62
0,925	185	161	144	129	117	108	99	92	86	81	76	72	68	65
0,950	191	167	149	134	122	111	103	95	89	84	79	74	70	67
0,975	197	173	153	138	126	115	106	99	92	86	81	77	73	69
1,000	204	178	158	143	130	119	110	103	95	89	84	79	75	71
1,025	210	184	163	147	134	123	113	105	98	92	86	82	78	74

The line of constant annual demand of 123,000 lbs U_3O_8 (Fig. 2) shows that an improvement of burnup by more than 1.4 MWd/kg U per 0.1 wt.% of increase in U-235 enrichment leads to savings in natural uranium consumption.

operate the power station. Table 1 shows that the additional demand for uranium as a result of enrichment can be compensated already by slight increases in burn-up.

Fig.2.



The line of constant annual demand of 123,ooo lbs U_3O_8 as given in Fig.2 shows that an improvement of burn-up by more than 1.4 MWd/kg U per o.1 wt.% of increase in U 235 enrichment leads to savings in natural uranium consumption.

In Fig. A-1 of the appendix the full set of lines of annual U_3O_8-demand are exhibited for the range of 50 000 to 170 000 lbs/year.

2.5. Annual Demand for Separative Work

Both separative work and conversion are of interest as new cost involving sources; both services will have to be imported in the long run and hence they will require foreign exchange.

Since the demand for separative work and conversion must automatically develop in parallel, we plotted only the annual demand for separative work as a function of the degree of enrichment and burn-up. In the cost calculations however, (cf. 2.9.) both developments will be considered jointly.

The spacing and slope of the lines representing the annual demand for separative work (cf. Fig. A-2) indicate that in the range between 12 and 20 MWd/kg U the annual additional requirement of SWU caused by a higher degree of enrichment required will not react very sensitively provided that the degree of enrichment of 1 wt.% U-235 will not have to be exceeded.

This leads to the assumption already now that the additional expenditure required to improve the economy of the CNA fuel cycle will be kept within a reasonable limit.

2.6. Utilization of Natural Uranium to be Provided Annually

The efficiency of measures to optimize the fuel supply can be judged excellently in this case from the utilization of the quantity of natural uranium to be provided annually. The utilization is expressed in MWd/kg **U** in U_3O_2; in this quotient the increased burn-up but also the increased uranium demand, which results from enrichment and conversion, are taken into consideration.

The reference value for our consideration was taken to be the uranium utilization in the natural uranium core, which was found to be 6.8 MWd/kg U.

The table listing the values calculated for uranium utilization of all burn-up/enrichment combinations (cf. Table 2) shows that e.g. with an enrichment of 0.9 wt.% an increase by only 4 MWd/kg U in burn-up can already lead to a significant improvement of uranium utilization.

Table 2

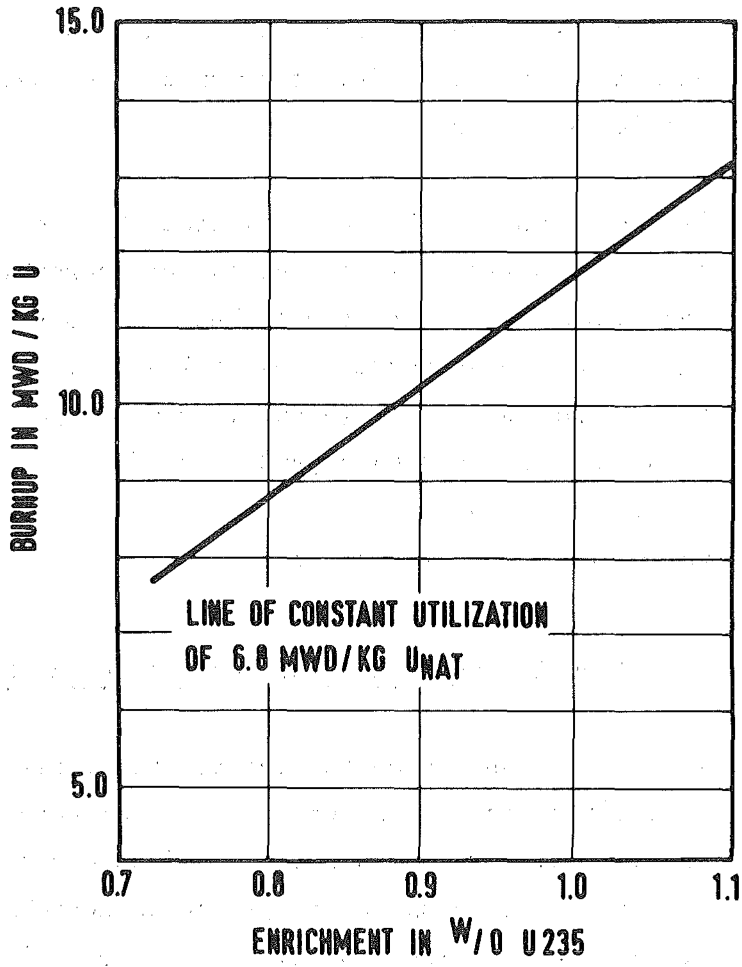
Uranium Utilization

$MWd_{th}/kg\ U_{nat}$ in concentrate

Enrichment w/o U-235	Burn-up [MWd/kg U]													
	7	8	9	10	11	12	13	14	15	16	17	18	19	20
0,711	6,8													
0,750	6,0	6,8	7,7	8,5	9,4	10,2	11,1	11,9	12,8	13,6	14,4	15,3	16,2	17,0
0,775	5,7	6,5	7,3	8,1	9,0	9,8	10,6	11,4	12,2	13,0	13,8	14,7	15,5	16,3
0,800	5,5	6,2	7,0	7,8	8,6	9,4	10,1	10,9	11,7	12,5	13,3	14,0	14,8	15,6
0,825	5,2	6,0	6,7	7,5	8,2	9,0	9,7	10,5	11,2	12,0	12,7	13,5	14,2	15,0
0,850	5,0	5,8	6,5	7,2	7,9	8,6	9,4	10,0	10,8	11,5	12,2	13,0	13,7	14,4
0,875	4,9	5,5	6,2	6,9	7,6	8,3	9,0	9,7	10,4	11,1	11,8	12,5	13,2	13,9
0,900	4,7	5,3	6,0	6,7	7,4	8,0	8,7	9,4	10,0	10,7	11,4	12,0	12,7	13,4
0,925	4,5	5,2	5,8	6,5	7,1	7,7	8,4	9,0	9,7	10,3	11,0	11,6	12,3	12,9
0,950	4,4	5,0	5,6	6,2	6,9	7,5	8,1	8,7	9,4	10,0	10,6	11,2	11,9	12,5
0,975	4,2	4,8	5,4	6,0	6,6	7,2	7,9	8,5	9,1	9,7	10,3	10,9	11,5	12,1
1,000	4,1	4,7	5,3	5,9	6,4	7,0	7,6	8,2	8,8	9,4	9,9	10,5	11,1	11,7
1,025	3,9	4,5	5,1	5,7	6,2	6,8	7,4	7,9	8,5	9,1	9,6	10,2	10,8	11,4

The line of constant uranium utilization for the value 6.8 MWd/kg of natural uranium (Fig. 3) clearly shows that practically each combination within the probable range (0.85 - 1.0 enrichment; discharge burn-up 15-20 MWd/kg U) should result in noticeable savings of the uranium concentrate.

Fig.3.



2.7. <u>Annual Expenditure for Natural Uranium</u>

It was shown in Sections 2.4. and 2.6. that the specific additional demand for uranium concentrate caused by enrichment and conversion cannot only be compensated by improvements in the burn-up appearing realistic according to the present knowledge, but that even actual savings can be made.

In operation with natural uranium an annual total expenditure of roughly $ 982,000. must be anticipated at an assumed price of $ 8/lb U_3O_8. For a supposed enrichment of 1.0 wt.% U-235 and a burn-up of 12 MWd/kg U savings of the order of some $ 32.000 per year would already be made as compared to operation with natural uranium.

2.8. Annual Expenditure for Fuel Element Fabrication

Fuel element fabrication cost includes all expenses incurred in fuel element fabrication from the moment the fuel reaches the fuel element fabrication plant.

It was shown in 2.3. (Fig. 1) that the annual demand for fuel elements reacts sensitively as a function of increasing burn-up. The annual fabrication expenditure must behave in a similar way. Considering again our two standard cases of 7 and 12 MWd/kg U, respectively, we obtain for case no. 1 (natural uranium operation)

an annual fabrication expenditure of	$ 5,550,000.0
and for case no. 2	$ 3,240,000.0
which means a saving of about	$ 2,310,000.0

2.9. Annual Expenditure for Conversion and Enrichment

The conversion and enrichment are the decisive factors increasing expenditure in the present case of homogeneous enrichment. (In an investigation required later on of both possible cases of inhomogeneous U-235 and/or plutonium enrichment a higher fuel element fabrication expenditure will certainly have to be anticipated.)

To be reasonable economically, the still justifyable additional expenditure due to enrichment and conversion must lie below the annual savings made in the provision of uranium (2.7.) and in fuel element fabrication (2.8.). In our standard case (1 wt.% U-235; 12 MWd/kg U) savings were

in uranium provision	$	32,000
in fuel element fabrication	$	2,310,000
in total	$	2,342,000.

The conversion and enrichment expenditure calculated for the same case was

for conversion	$ 116,000
for enrichment	$ 398,000
in total	$ 514,000

Consequently, the calculated annual reduction in expenditure for the fuel element inventory of the plant amounts to some $ 1,828,000. referred to our standard cases.

2.10. Annual Total Expenditure for Reactor Fuel

If the modes of behavior of different types of expenditure discussed above are balanced against each other as done in the preceding section with an example, a very clear idea is obtained of the development of annual total expenditure for the reactor fuel required in each case.

Table 3 shows that cost savings can already be anticipated for very low improvements of burn-up. In the range which according to present knowledge can be considered realistic, a positive result of the use of enriched fuel can be expected also in case of an unfavorable development of present enrichment and conversion costs (cf. Fig. A-6).

3. Conclusions

It was the aim of this investigation to establish an overview of the consequences of using enriched fuel.
To evaluate the variation in flow of material and costs before detailed results of core physics calculation are available, the relevant quantities are plotted as function of enrichment and burn-up as two independent variables. This material is given in the appendix.

These results are useful to demonstrate tendencies and potentialities that go with using enriched fuel. They may serve as a starting point for more detailed investigations which are justified when the field of burn-up values attainable and feasable enrichment values is reduced by the results of core physics calculations.

Table 3

Annual Total Expenditures for Reactor Fuel in 10^6 US-$/a

Enrichment w/o U-235	Burn-up [MWd/kg U]													
	7	8	9	10	11	12	13	14	15	16	17	18	19	20
0,711	6,53													
0,725	6,77	5,93	5,27	4,74	4,31	3,95	3,65	3,39	3,16	2,96	2,79	2,63	2,50	2,37
0,750	6,88	6,02	5,35	4,82	4,38	4,01	3,71	3,44	3,21	3,01	2,83	2,68	2,54	2,41
0,775	6,99	6,12	5,44	4,89	4,45	4,08	3,76	3,50	3,26	3,06	2,88	2,72	2,58	2,45
0,800	7,10	6,22	5,53	4,97	4,52	4,14	3,83	3,55	3,32	3,11	2,93	2,76	2,62	2,49
0,825	7,22	6,32	5,61	5,05	4,59	4,21	3,89	3,61	3,37	3,16	2,97	2,81	2,66	2,53
0,850	7,33	6,42	5,70	5,13	4,67	4,28	3,95	3,67	3,42	3,21	3,02	2,85	2,70	2,57
0,875	7,45	6,52	5,79	5,22	4,74	4,35	4,01	3,73	3,48	3,26	3,07	2,90	2,74	2,61
0,900	7,57	6,62	5,89	5,30	4,82	4,42	4,08	3,78	3,53	3,31	3,12	2,94	2,79	2,65
0,925	7,69	6,73	5,98	5,38	4,89	4,49	4,14	3,84	3,59	3,36	3,17	2,99	2,83	2,69
0,950	7,81	6,83	6,07	5,47	4,97	4,56	4,21	3,91	3,64	3,42	3,22	3,04	2,88	2,73
0,975	7,93	6,94	6,17	5,55	5,05	4,63	4,27	3,97	3,70	3,47	3,27	3,09	2,92	2,78
1,000	8,06	7,05	6,27	5,64	5,13	4,70	4,34	4,03	3,76	3,52	3,32	3,13	2,97	2,82
1,025	8,18	7,16	6,36	5,73	5,21	4,77	4,41	4,09	3,82	3,58	3,37	3,18	3,01	2,86

According to the present state of knowledge the consequence of an increase in burn-up achieved by slight homogeneous enrichment of the core can be described as follows:

1. Argentine uranium reserves can be spared by better specific utilization of the fuel.

2. The specific fuel costs decrease and hence the electricity generation costs can be reduced (cf. Table 4).

3. As already shown in 2.9. the additional expenditures in foreign currencies due to conversion and enrichment services is more than compensated by the savings due to the reduced demand for fabrication of fuel elements.

4. The overall capacity of the storage facility of CNA for spent fuel elements (two pools) amounts to 200 te uranium. Of this capacity 38,6 te are to be reserved for the emergency discharge of the core. The remaining capacity of 160 te will be used up in about 3 1/2 years of normal plant operation using natural uranium fuel. Increasing the burn-up to 12 MWd/t by using enriched fuel would nearly double this time period to six years.

Table 4

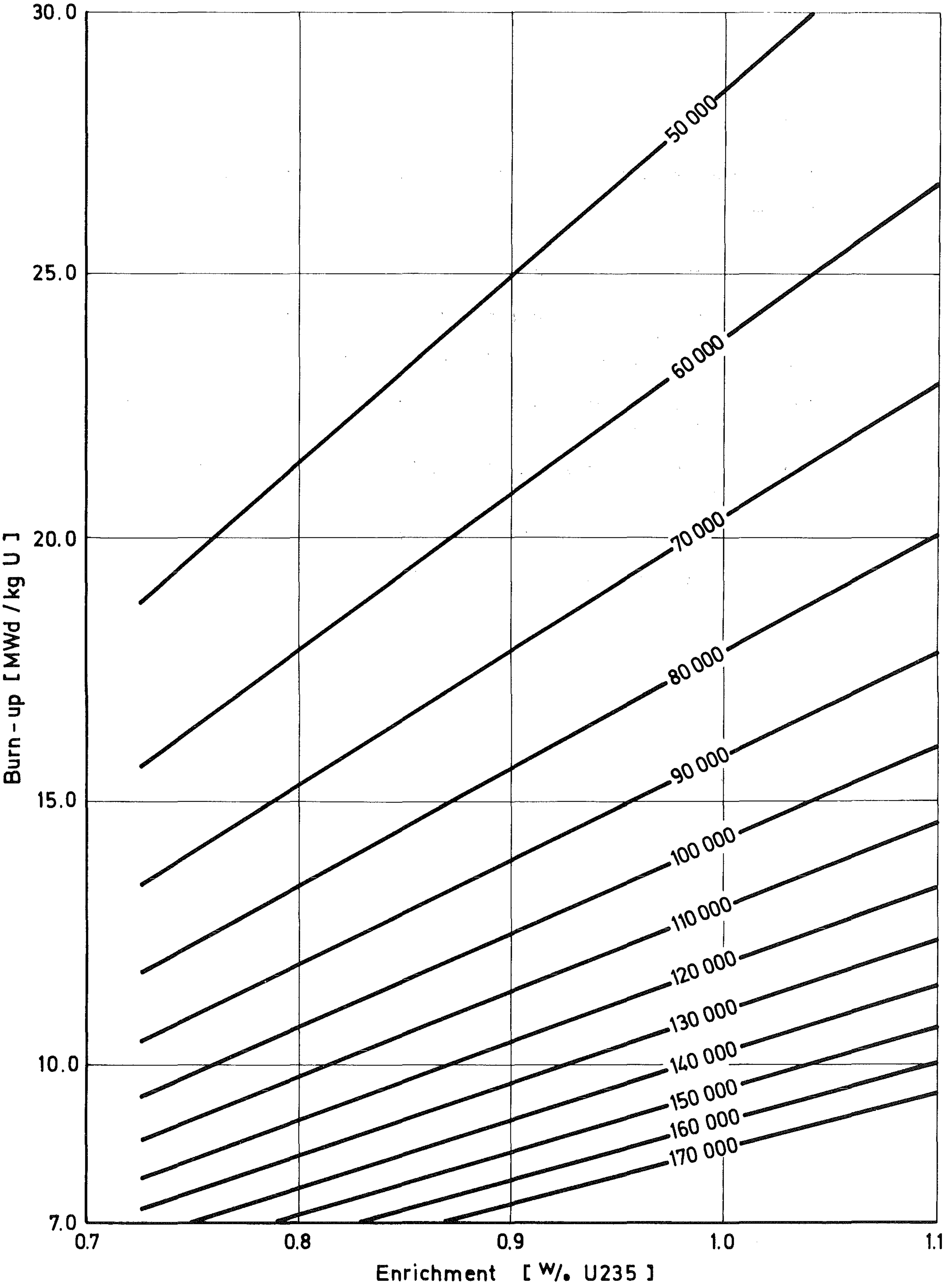
Specific fuel costs [mills/kWh$_e$]

Enrichment w/o U-235	Burn-up [MWd/kg U]													
	7	8	9	10	11	12	13	14	15	16	17	18	19	20
0,710	2,9													
0,750	3,1	2,7	2,4	2,2	2,0	1,8	1,7	1,5	1,4	1,3	1,3	1,2	1,1	1,1
0,775	3,1	2,7	2,4	2,2	2,0	1,8	1,7	1,6	1,5	1,4	1,3	1,2	1,2	1,1
0,800	3,2	2,8	2,5	2,2	2,0	1,86	1,7	1,6	1,5	1,4	1,3	1,2	1,17	1,1
0,825	3,2	2,8	2,51	2,26	2,06	1,89	1,74	1,62	1,51	1,41	1,33	1,26	1,19	1.13
0,850	3,28	2,87	2,55	2,30	2,09	1,92	1,77	1,64	1,53	1,44	1,35	1,28	1,21	1,15
0,875	3,34	2,92	2,60	2,33	2,12	1,95	1,80	1,67	1,56	1,46	1,37	1,30	1,23	1,17
0,900	3,39	2,97	2,64	2,37	2,16	1,98	1,83	1,69	1,58	1,48	1,40	1,32	1,25	1,19
0,925	3,44	3,01	2,68	2,41	2,19	2,01	1,85	1,72	1,61	1,51	1,42	1,34	1,27	1,21
0,950	3,50	3,06	2,72	2,45	2,23	2,04	1,88	1,75	1,63	1,53	1,44	1,36	1,29	1,22
0,975	3,55	3,11	2,76	2,49	2,26	2,07	1,91	1,78	1,66	1,55	1,46	1,38	1,31	1,24
1,000	3,61	3,16	2,81	2,53	2,30	2,10	1,94	1,80	1,68	1,58	1,49	1,40	1,33	1,26
1,025	3,66	3,21	2,85	2,56	2,33	2,14	1,97	1,83	1,71	1,60	1,51	1,42	1,35	1,28

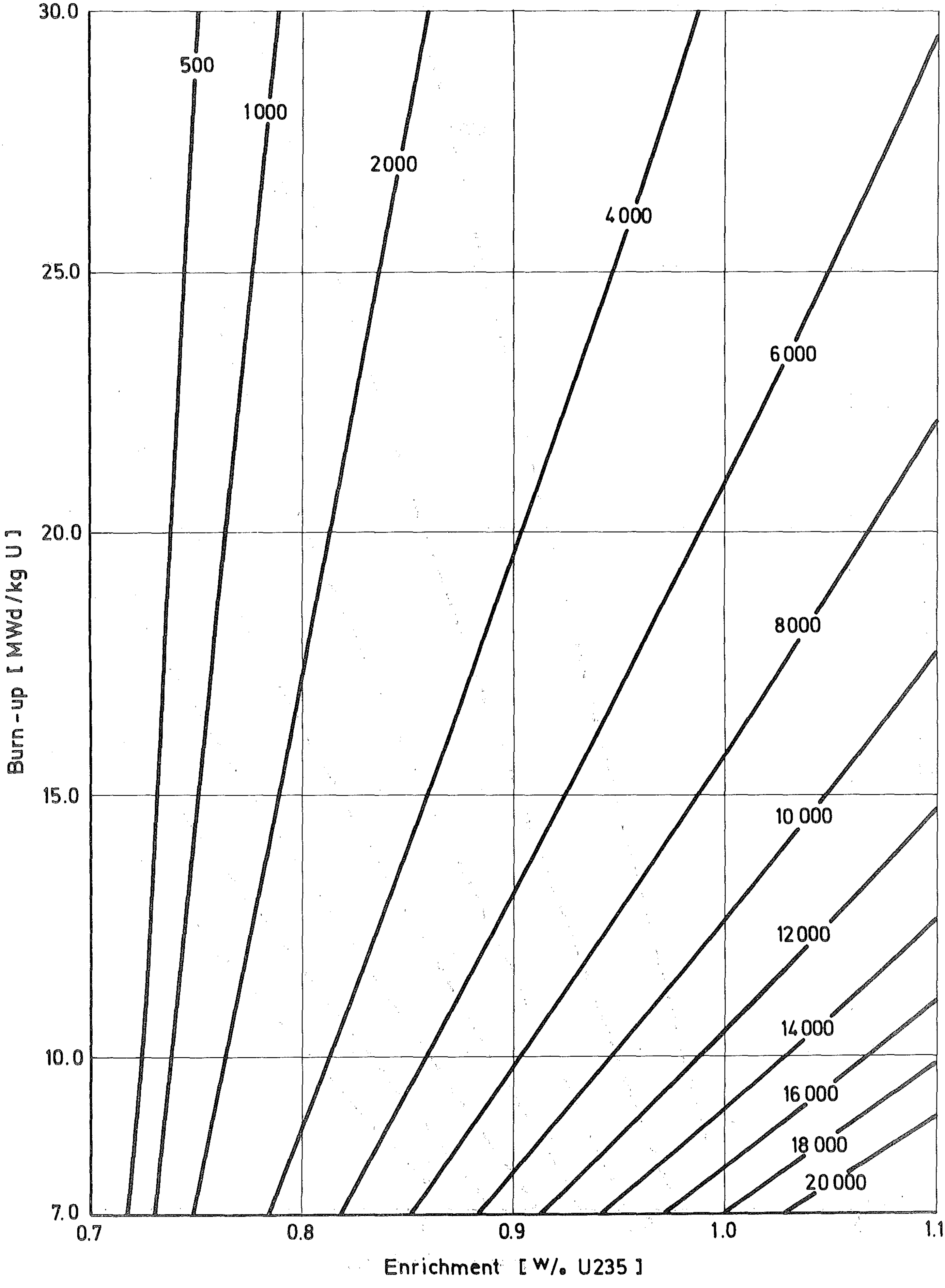
Appendix

Figures showing relevant quantities and costs as function of enrichment and burn-up.

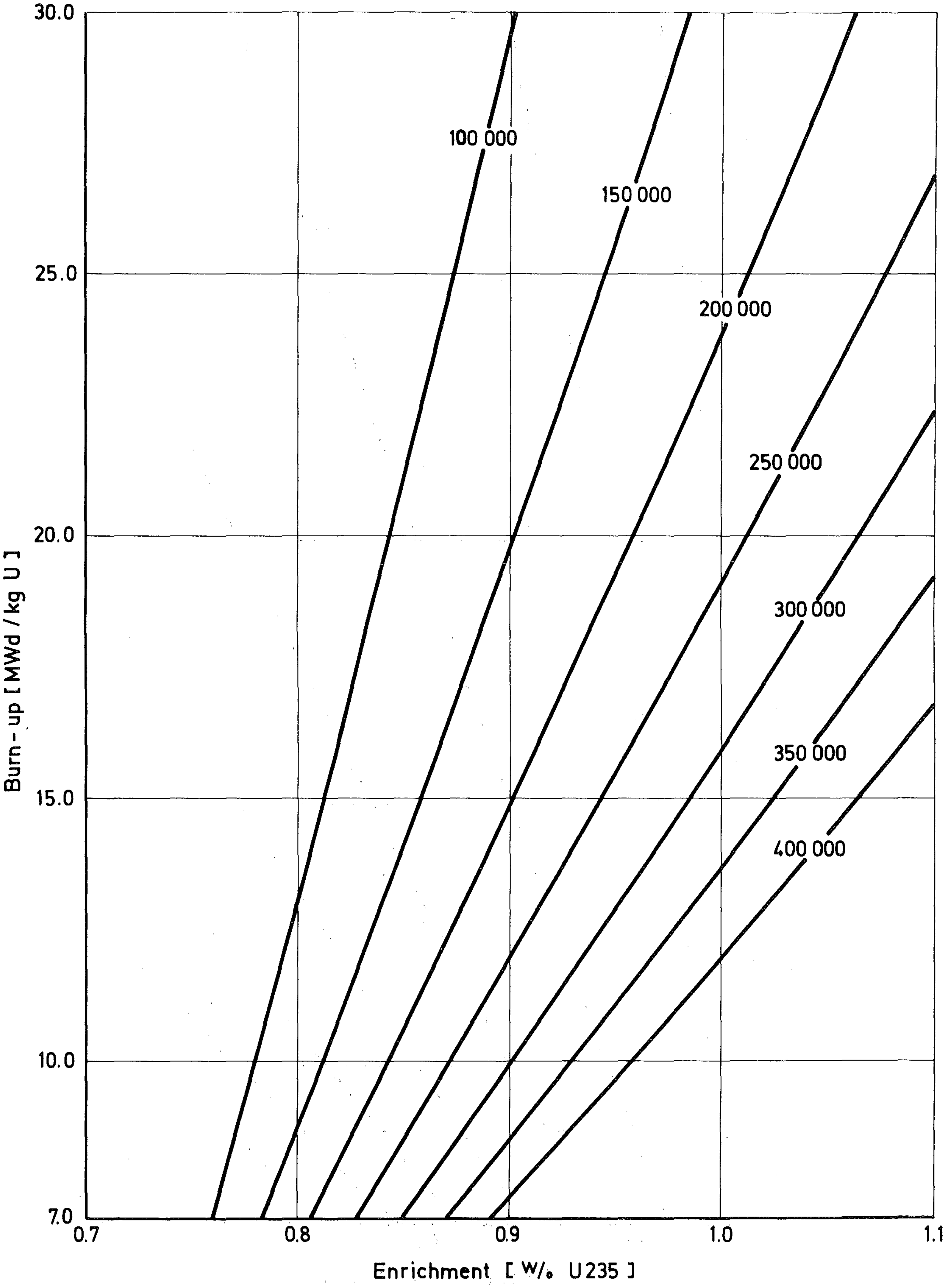
Fig. A-1	Annual demand for uranium in lbs U_3O_8
Fig. A-2	Annual demand for separative work in kg SWU
Fig. A-3	Annual expenditures for separative work in US-$
Fig. A-4	Annual expenditures for separative work and conversion in US-$
Fig. A-5	Annual expenditures for fuel element fabrication in US-$
Fig. A-6	Annual total expenditures for fuel
Fig. A-7	Uranium utilization in MWd/kgU in U_3O_8



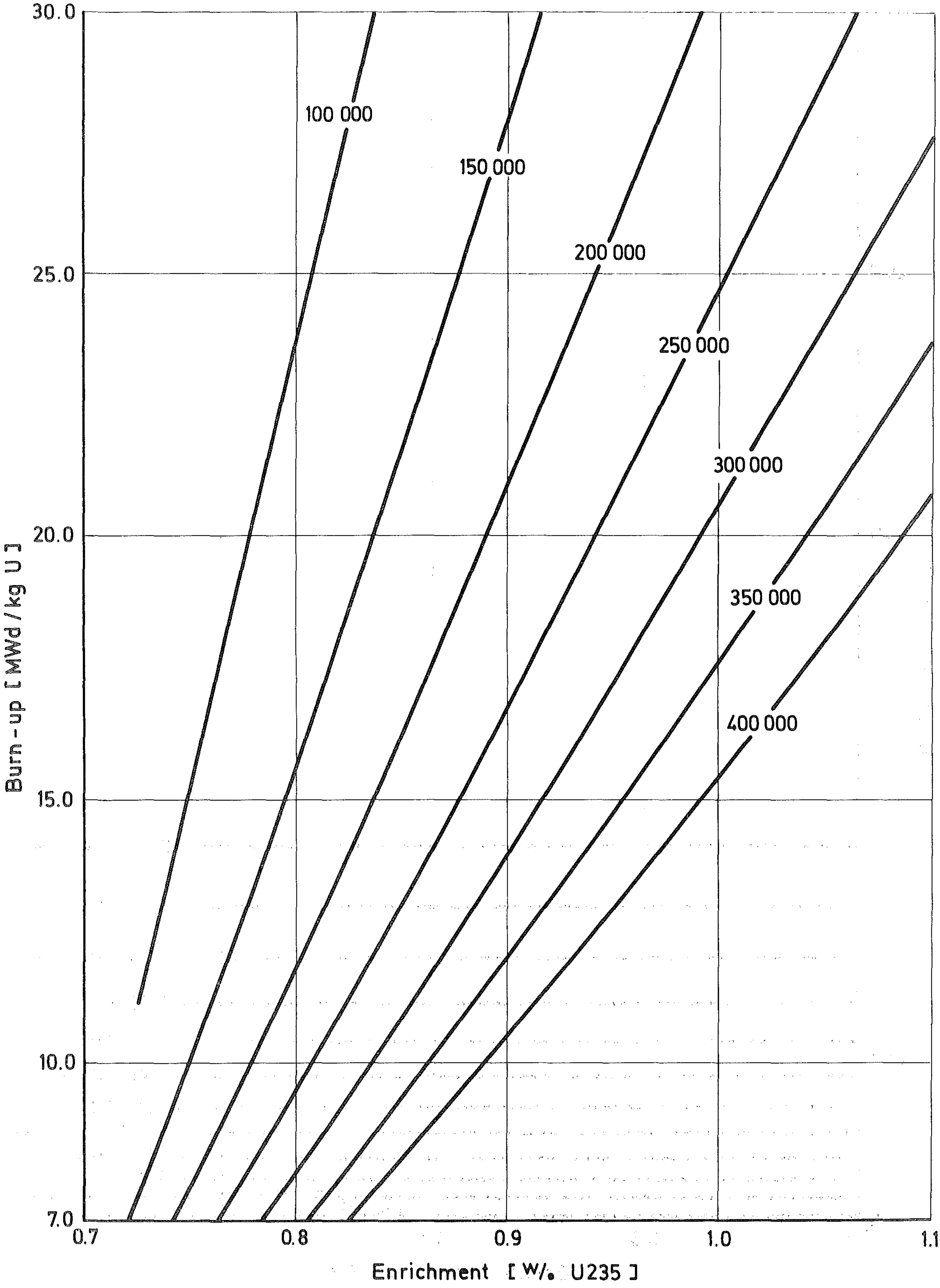
A-1: Annual demand for uranium in lbs U_3O_8



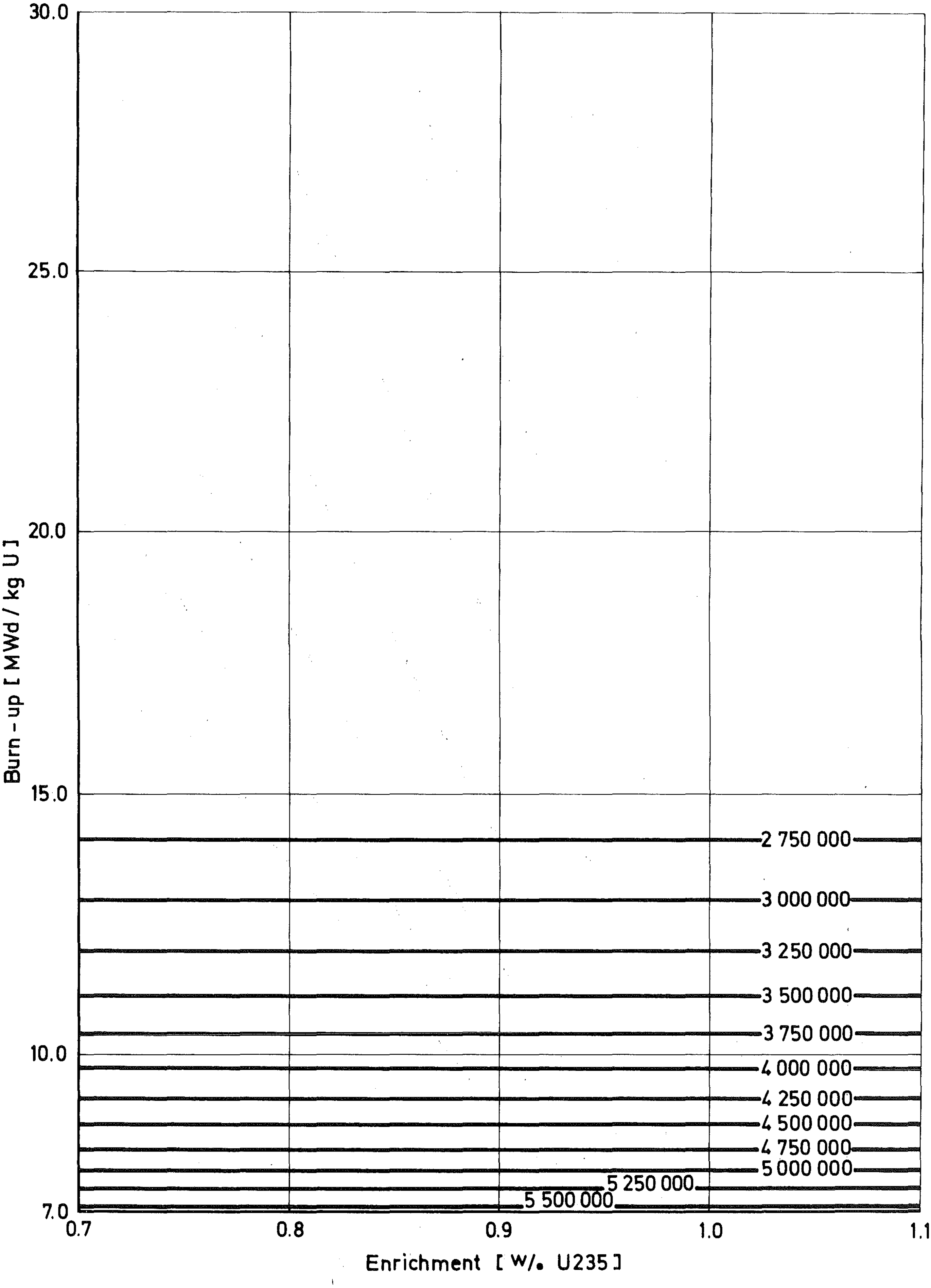
A-2: Annual demand for separative work in kg SWU



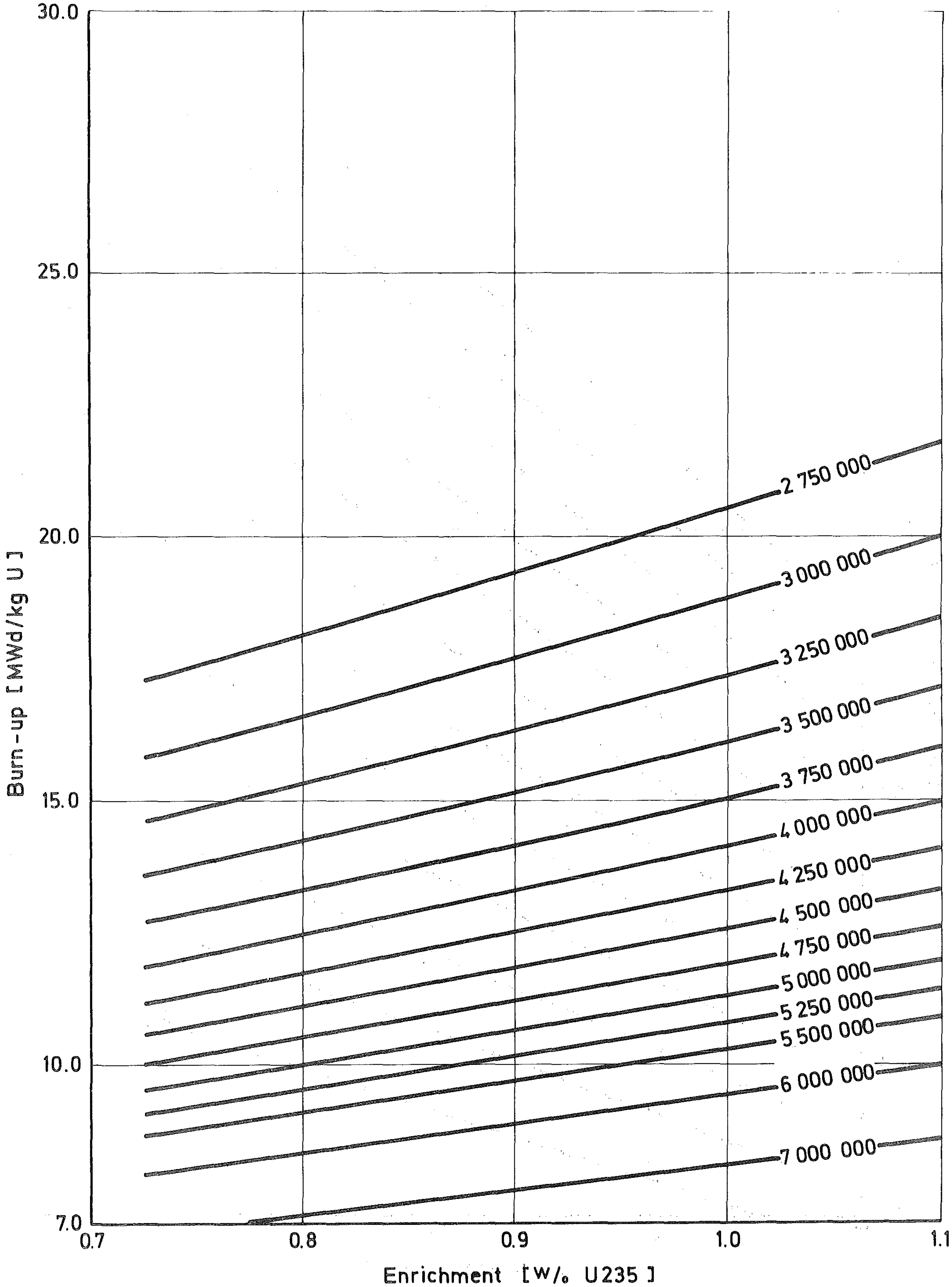
A-3: Annual expenditures for separative work in US-$



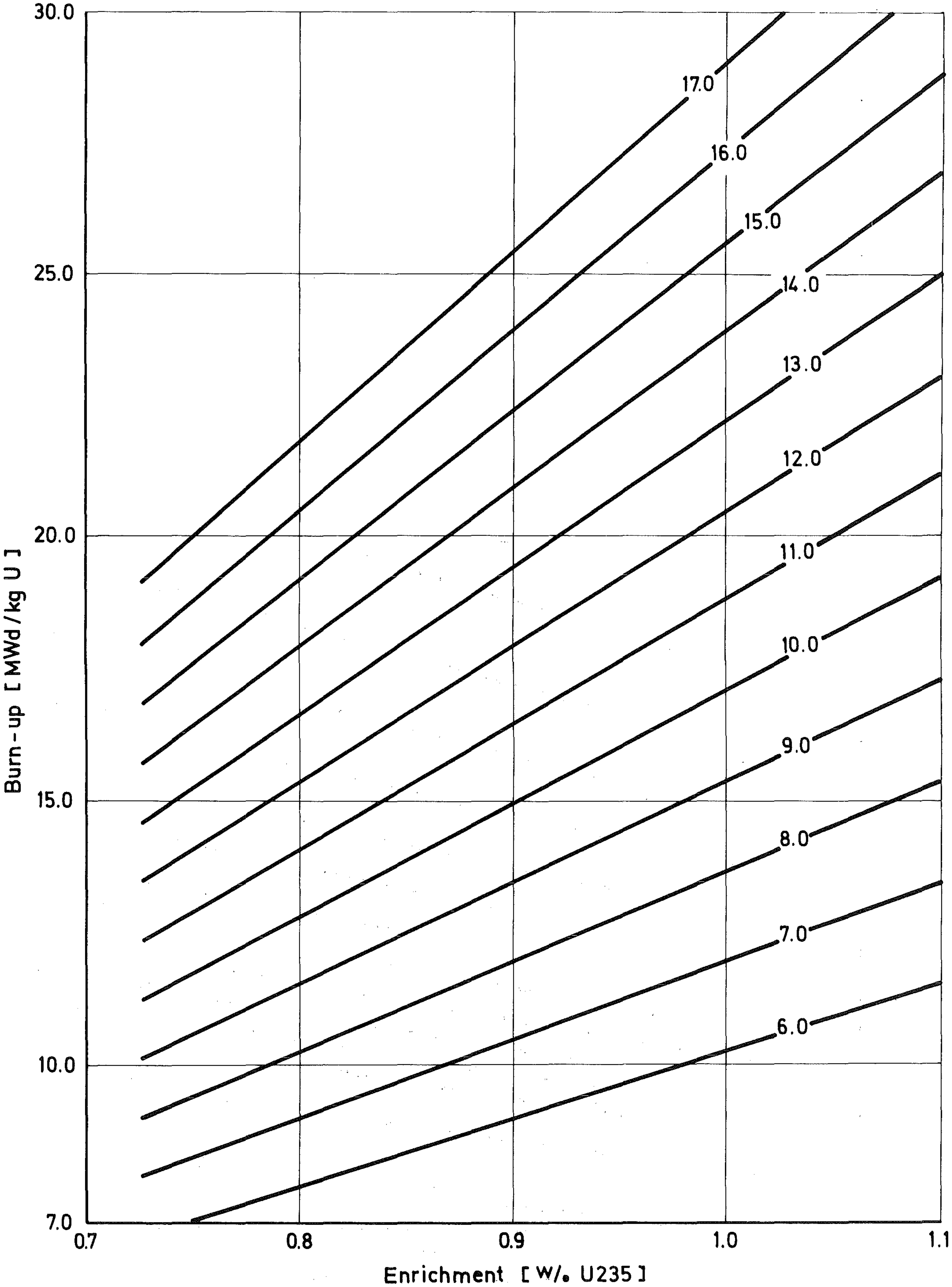
A-4: Annual expenditures for separative work an conversion in US-$



A-5: Annual expenditures for fuel element fabrication in US-$



A-6: Annual total expenditures for fuel in US-$



A-7: Uranium utilization in MWd/kgU in U_3O_8

Conclusions and Recommendations

by

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The results of the preliminary assessment of the use of enriched fuel or of spikes in the Atucha Nuclear Power Plant show very promising aspects. With only a few changes in the fuel cycle substantial annual savings could be realized, and the energy content of the uranium could be brought to more effective use.

These preliminary results should be encouraging enough to stimulate further more detailed calculations. They are to be based on the final results of reactor physics computations yielding information of the burn-up performance of the Atucha reactor including the margin of error to be taken into account. This more detailed treatment should serve to analyze the costs under various assumptions and should separately show expenses in domestic and foreign currencies.

Up to now more technical questions of the fuel cycle domain were excluded from all considerations. For the more detailed studies this will no longer be feasible.
The fresh fuel part of the fuel cycle is largely determined by Argentine decisions on erecting a fuel-element factory and on developing domestic uranium mines. As these decisions are mostly settled, the evaluation of this part of the fuel cycle should bring no problems.

The study should also look explicitely into the handling of spent fuel. By proper analysis and exact planning certain cost savings may be accomplished also in this field. At the same time it will be necessary to preserve the possibility of deviating from the fuel handling strategy as presently planned. Future conditions may favor different prodedures and accordingly enough flexibility should be incorporated.

In any case sufficient transport capacity must be supplied in due time, and enough storage capacity has to be made available. Various alternatives should be analyzed from a technical and economic point of view.