KfK 3801
August 1984

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NEUTRONIC STUDIES OF FISSILE AND FUSILE BREEDING BLANKETS

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ISSN 0303-4003

Abstract

In light of the need of convincing motivation substantiating expensive and inherently applied research (nuclear energy), first a simple comparative study of fissile breeding economics of fusion fission hybrids, spallators and also fast breeder reactors has been carried out. As a result, the necessity of maximization of fissile production (in the first two ones, in fast breeders rather the reprocessing costs should be reduced) has been shown, thus indicating the design strategy (high support ratio) for these systems. In spite of the uncertainty of present projections onto further future and discrepancies in available data even quite conservative assumptions indicate that hybrids and perhaps even earlier - spallators can become economic at realistic uranium price increase and successfully compete against fast breeders.

Then on the basis of the concept of the neutron flux shaping aimed at the correlation of the selected cross-sections with the neutron flux, the indications for the maximization of respective reaction rates has been formulated. In turn, these considerations serve as the starting point for the guidelines of breeding blanket nuclear design, which are as follows:

1) The source neutrons must face the multiplying layer (of proper thickness) of possibly low concentration of nuclides attenuating the neutron multiplication (i.e. structure materials, non-gaseous coolants).

2) For the most effective trapping of neutrons within the breeding zone (leakage and void streaming reduction) it must contain an efficient moderator(not valid for fissile breeding blankets).

3) All regions of significant slow flux should contain ^{6}Li in order to reduce parasite neutron captures in there.

In the field of fissile materials production a measure of fissile breeding efficiency (fissile mass/energy released) is proposed as a function of the system conversion ratio and of the non-fissile (e.g. fusion neutrons, fast fissions) energy release in the system. Also a net effective fissile breeding cross-section is defined and its dependence and the one of the breeding efficiency on the resonance self-shielding (RSS) effects is demonstrated. It is shown in numerical calculations that the neglect of RSS of fertile materials in fissile breeding systems causes inadmissible overestimation of fissile breeding and underestimating of the energy production in spallators and fission-fusion hybrids. Consequently, their support ratio is significantly reduced and the danger of supercriticality appears in water cooled spallators. Finally, the necessity of consideration of the resonance self-shielding effects and the resignation of moderators in fissile breeding systems has been postulated.

Neutronenphysikalische Studien zu Tritium und Spaltmaterial erbrütenden Blankets

Zusammenfassung

Zunächst wird ein einfacher Vergleich der Wirtschaftlichkeit von Fusions-Spaltungs-Hybridsystemen, Spallations-Brüter und Schnellen Brutreaktoren durchgeführt. Es zeigt sich, daß bei den ersten beiden eine Maximierung der Spaltmaterial-Erzeugung d.h. ein hohes "support ratio" wichtig ist.

Beim Schnellen Brüter müßten die Aufarbeitungskosten reduziert werden.

Auch bei ungünstigen Annahmen für die Systeme können Fusions-Hybrid-Reaktor und Spallationsquelle bei realistischen Annahmen zum Anstieg des Uranpreises wirtschaftlich werden und mit dem Schnellen Brüter konkurrieren.

Dann werden Kriterien für die neutronenphysikalische Blanketoptimierung entwickelt. Leitender Gesichtspunkt ist hier, die Neutronenflußverteilung so zu formen, daß großen Querschnitten der gewünschten Reaktion auch hohe Neutronenflußwerte entsprechen und das Umgekehrte für Konkurrenzreaktionen gilt. Es ergaben sich folgende Richtlinien:

1. Die Quellneutronen sollen auf eine multiplizierende Schicht auftreffen, diese soll möglichst wenig die Multiplikation schwächende Materialien enthalten.
2. Die effektivste Art Neutronen in der Brutzone einzufangen besteht in der Verwendung eines starken Moderators.
3. Alle Bereiche mit nennenswertem niederenergetischem Neutronenfluß sollen ^6Li enthalten um die parasitäre Absorption zu vermindern.

Zur Charakterisierung des Erbrütens von Spaltmaterial werden die "Brut-Effektvität" (erzeugte Menge an Spaltmaterial/freigesetzte Energie) und der "effektive netto Brutquerschnitt" eingeführt. Die Bedeutung der Resonanzselbstabschirmung wird damit deutlich gemacht. Numerische Rechnungen zeigen, daß eine Vernachlässigung der Resonanz-Selbstabschirmung zu einer erheblichen Überschätzung der Spaltmaterialproduktion und Unterschätzung der freigesetzten Energie führen. Dies gilt besonders bei starker Moderation. Es wird vorgeschlagen, unmoderierte Brutsysteme zu verwenden.

Contents

1. Introduction

1.1 Foreword

The increasing awareness of limited world energy resources gives rise to a need for seeking and development of new energy sources. This need is enhanced by more and more critical environmental problems and the threat of scarcity of many raw materials - the questions that seem to be solvable solely at the cost of additional energy consumption. While facing these additional energy needs, one joins unbroken hopes to the Advanced Nuclear Energy Systems (ANESs): fusion reactors, fusion-fission hybrids and spallation breeders.

The pure fusion systems which are based upon the (d,t) reaction should be rather considered as a preliminary step towards distant prospective of the inexaustible in the millennial scale (d,d) fusion or towards the "clean" fusion based upon the $^{7}Li(p,2\alpha)$ or $^{11}B(p,3\alpha)$ reactions. Instead, the two remaining ANES' concepts can answer already the next generation energy demands.

At present no way of breeding of fissile materials can compete with their recovery from natural resources. Nevertheless, an unavoidable increase in uranium price, resulting from the exaustion of richest deposits, to a level making the fissile breeding worth reconsideration remains within the foreseable future. But the choice of the right energy strategy and proper decisions must be and will be undertaken much earlier. The last is indispensable in spite of the natural uncertainty of world energy consumption forecasts and of some discrepancies in available economical data concerning ANESs.

But not only the future economic benefits plead for their development. Like all peaceful applications of nuclear energy, what is, unfortunately, usually misunderstood by the public, the ANESs are environmentally benign, producing even less radioactive waste than light water reactors (LWRs). When not requiring fissile fuel supply and thus being uranium embargo resistant they can assure the energy independence of national economy.

On the other hand they create a difficult problem to scientists and engineers, with many questions not having been solved yet and still requiring much effort from the scientific community.

As concerns fusion and hybrid reactors, which are based upon the (d,t) reaction, the fusile fuel (tritium) production conditions the operation of the reactor. The fissile breeding, in turn, is just the principal purpose of hybrid and spallator operation. The presence of high energy (14 MeV or more) neutrons in an ANES is the source of many difficulties (e.g. radiation damage, gas production, induced radioactivity etc.), the overcoming of all of these also directly conditions the reliable functioning of the whole system. Nevertheless, according to the hierarchy of objectives, the engineering problems must be imposed by (realistic of course) physical feasibility requirements and not the other way round. Therefore, it has been decided here to concentrate on the basic physical question of the optimizing of fusile and fissile nuclear fuel production.

The neutron processes being investigated in the present study take place in the medium surrounding the neutron source, called blanket, which performs two fundamental functions:

1) assurance of the necessary fusile and/or fissile breeding,
2) conversion of the neutron energy into heat.

In this light, the main objective is to optimize generally the neutron utilization for the nuclear fuel production. This implies the enhancement of neutron multiplication (without excessive energy production) with the simultaneous minimization of neutron losses. This aim can be achieved through the proper flux correlation with the effective macroscopic cross-section for given reaction (multiplication, breeding) in the phase space.

As concerns hybrid and spallator blankets, the difficulties result from the justified need of maintaining the "richness" of the system in neutrons. In other words, the neutron price should be possibly low i.e. their number should be highest at the given power (it is tacitly assumed here that the total cost of the system is basically determined by its size i.e. its power which should be optimum).

This condition imposes the necessity of fission suppression in the system, since the highly exoergic fission reactions drastically decrease the neutron-to-energy ratio of the system. To suppress the fissions proves particularly uneasy in the presence of moderator and higher concentrations of the fissile materials just having been produced. Such situation takes place e.g. when one multiplies neutrons with beryllium in the hybrid fissile breeding zone or uses water as coolant in spallators. All the above presents a complexed neutronic problem even at modern level of neutron transport numerics and computing potential. The common transport codes still assume certain approximations or simplifications of various significance and the nuclear data still are far from being perfect.

To contribute to the solution of these problems is the purpose of the present study.

1.2 Social and economical aspects of advanced nuclear energy

Even when dealing principally with neutronic problems it is reasonable to mention also other vital view-points of the subject, substantiating the scientific activity in this field, since:

1) inherently applied and expensive research must have well convincing social motivation
2) economical needs and requirements deeply affect the direction of research and the technological solutions (e.g. the fission suppression concept, see 4.1.2)

Here, we confine only to certain remarks that may enlighten some overlooked aspects of nuclear energy. While discussing the economical problems merely the general relationships between selected economical and physical parameters of the system will be shown /1/, whereas we do not intend to present e.g. a detailed cost analysis or optimization.

Such simplified methodology which is based on the relative costs behaviour only, is quite different from the one of the studies carried up to now /2 - 15/, in which generally the absolute costs are estimated and their determinants are discussed. It is, however, sufficient for the limited purpose of comparative evaluation of the economic prospectives of fissile breeding alternatives and the indication of general directions for systems design.

1.2.1 Social questions

About nuclear energy there have arisen enormous misunderstandings. In addition to this, the economical analyses of energy problems rarely attempt to consider the entirety of social costs. Usually the studies are confined to the expenses immediately coupled with the energy device and thus being afforded by the institutions directly involved with, instead of considering the costs paid by the society as a whole. No doubts some social costs can only hardly or not at all be expressed in numbers. To those we can qualify, for instance, many environmental problems like e.g. landscape destruction or others like increased mortality and long term (delayed) health effects, for instance, among the coal miners. Nevertheless, all these aspects have to be taken into consideration while evaluating thoroughly an energy system. And in this view, the ANESs seem much more attractive than e.g. fossile energy that still contributes to the world energy consumption in 93.6 % (in 1980).

The ANESs enable us to avoid not only the "acid rain", the mining and transport accidents associated inseparably with the fossile energy, but also the environmental damage resulting from e.g. the excavation mining (brown coal), the pipe line constructions (oil and gas), the oil tanker accidents or the covering enormous surfaces with concrete (solar energy). In addition to the above advantages, characterizing anyway the nuclear energy in general, the ANESs can also protect the national economy against possible embargo of uranium cartel. It must be admitted here, that this safety can be earliest assured by fast breeders, of technology having been well mastered though less encouraging from the purely economical point of view (see 1.2.2).

1.2.2 Economical aspects

The optimum size (or power) of any nuclear energy unit is a result of competing factors. The costs of energy device increase less than linearly with its power ($\sim$ 2/3 exponent) but simultaneously there is a power limit determined by the need of energy distribution among usually spread out consumers, by the admissible net charge variations due to the device failure, by security reasons etc. In consequence, the overconcentration of energy production is undesirable and ca. 1 GWe is usually assumed as the maximum power of a single device. This limitation is then valid for fissile fuel breeding oriented systems like hybrids and spallators.

In contrast with fission based "classic" reactors, the ANESs exploiting neutron "rich" - energy "poor" processes should be recognized rather as the most powerful sources of neutrons to be used for fissile fuel production out of fertile media. This fuel can be next used for the energy production in specialized systems (e.g. LWRs) more economically than it ever can be in a device charged simultaneously with the difficult job of fissile breeding. Such task sharing improves then the performance of the energy systems as a whole. It should be noticed that the opposite suggestion i.e. the parallel stressing on the energy production (through fissions) in an ANES leads to the fast breeder concept. Such conclusion results from the reasoning that the decreasing contribution of the non-fission component in the neutron and energy production implies finally its total elimination, seeing the radical simplification of the system. In other words, no externally driven (fusion, spallation) subcritical assembly with so complex and expensive control unit (tokamak, linac) can compete with exactly critical system of technical possibility proven already several decades ago and also well developed technology. Therefore, the advanced fissile breeding systems should enhance their neutron abundance, that is equivalent to suppressing the energy production. On the other hand, a more reliable economical analysis of advanced nuclear energy systems is at present very difficult because of both objective and rather subjective reasons. The main objective difficulty lies in the enormous variations in uranium price and world economic growth rate (the last one making impossible any energy demand forecast to be certain) during the last decades, depriving from the very beginning any study of its unquestionable grounds. The other difficulty are the discrepancies in present economic data concerning the existing and future nuclear energy systems. They can be however, explained in part by different assumptions and calculation methods.

Such situation justifies simplified analysis of all these questions, since a profound one may prove to be equally inaccurate. As a result we confine ourselves to the consideration of the selected most important elements and some approximative assumptions.

1.2.2.1 Theoretical Premises

The present discussion refers to the following circumstances:

1) The LWRs can be supplied from natural resources based uranium fuel or from then existing ANESs. (No special constructing of LWRs for ANESs is foreseen).
2) The uranium price increase is expected but except of this the inflation will not change the proportions between the particular cost components. (This is conservative, since the consequent energy price increase may draw some increase in the enrichment costs.)
3) The costs of fertile materials (thorium or depleted uranium) are negligible.

4) All costs are related to devices of equal optimum size (power).
5) The electricity market price is determined by the total LWR costs (with Pu recycling).
6) Conversion ratio of supplied reactors does not influence the breeder income from the fuel sale (thus for simplicity c_r = 0.67 will be assumed).
7) The possibility of spent fuel rejuvenation is not excluded i.e. it is not the cladding resistance that limits the admissible burn up.

In a simplest way, the condition of the fissile breeder economy can be formulated as follows:

$$C - \alpha C_L = F \cdot S \qquad (1.1)$$

where C - total annual levelized breeder cost
C_L - annual revenue (or cost) from the electricity sale (or purchase) expressed in total annual LWR cost C_L
α - coefficient equal to 1 for hybrids, 0 or negative for spallators when the electricity must be bought.
F - annual income from fuels sale to one supported LWR (when all units are operated by "the same owner" the "sale" signifies calculatory transfer, in order to evaluate economics of different options)
S - number of supported LWRs

In the equation (1.1) one can distinguish the following components of total costs:

$$C = C' + (F_r + F_o)\ S \qquad (1.2)$$

where C' - total non-fissile fuel cycle costs
F_r - reprocessing costs of bred fuel for one supported LWR
F_o - other fuel costs (fabrication, transportation, etc.)

While in the fuel sale income F, instead of reprocessing costs one can separate the uranium F_u and the enrichment F_e costs:

$$F = F_u + F_e + F_o \tag{1.3}$$

This division of fuel costs is the simplest one sufficient for the present analysis. Simultaneously, the equality of other fuel costs F_o in cases of the use of the bred and of the natural fissile materials was assumed (that is well true at least in the case of ^{232}Th - ^{233}U cycle).

The substituting (1.2) and (1.3) into (1.1), dividing by C_L and transforming leads to

$$\frac{C'}{C_L} = \frac{F_u + F_e - F_r}{C_L} S + \alpha \tag{1.4}$$

or

$$\frac{C'}{C_L} = \frac{F'}{C_L} \cdot S + \alpha \tag{1.5}$$

where

$$F' = F_u + F_e - F_r \tag{1.6}$$

The equations (1.4) and (1.5) represent the maximum breeder (non fuel) cost that can be compensated by the net income from the fuel and electricity sale.

It may be of interest also to consider the variant of fuel rejuvenation without reprocessing /16/. In this case the net income from the fuel sale F' is to be expressed differently. Simplifying, one can assume it to be equal to the costs of fuel production from the spent fuel in the "classic" way, since this cost can be saved by the rejuvenation process. The approximation lies in the assumption of equal quality of the rejuvenated fuel elements and the ones produced from reprocessed fuel. Therefore, F' in the case of fuel rejuvenation is

$$F' = F_u + F_e + F_{rL} \qquad (1.7)$$

where

F_{rL} - cost of the LWR fuel reprocessing.

For fission (fast) breeders that in contrast to hybrids and spallators are assumed to supply no fissile material to external clients (S = O) the equation (1.1) takes the simple form:

$$C = C_L \qquad (1.8)$$

where one can distinguish

$$C' + F_B = {}^oC_L + \Delta F_u \qquad (1.9)$$

where in turn

${}^oC_L$ - inflation corrected present LWR cost

F_B - fission (fast) breeder fuel cycle cost

ΔF_u - supposed maximum price share increase (over present level)

then transforming (1.9) one obtains

$$\frac{C'}{{}^oC_L} = 1 + \frac{\Delta F_u}{{}^oC_L} - \frac{F_B}{{}^oC_L} \qquad (1.10)$$

In the formulas (1.1, 1.2, 1.4, 1.5) there is one important parameter, support ratio, strictly determined by the physics of nuclear fission. The dependence of the support ratio and of the net fissile breeding efficiency on the conversion ratio is presented in /17/. Hybrids and spallators that need not self-sustaining chain reaction are characterized by high conversion and thus high support ratios in contrast to fast breeders. In consequence, for hybrids and spallators the fissile fuel is the main product (or even the only one of spallators), while the energy remains the main product of fast breeders.

1.2.2.2 Calculations and Results

The necessary data indicating approximately the expected values of the parameters in the equations (1.4), (1.7) and (1.9) were elaborated on the basis of recent studies /2 - 15/ pertinent to the economical questions of nuclear energy and are collected in the Table 1.1. For clarity, the costs of selected factors are presented in the form of respective contributions into total annual cost and normalised to LWR costs. This information plus the foreseen support ratios is sufficient for present analysis.

Except of showing certain dispersion of the data, the Table 1.1 indicates generally much lower reprocessing costs of the fissile fuel (.6 - 4 %, the lower value refers to the molten salt concept) bred in hybrids or spallators than the (enrichment + uranium) costs (12 - 23 %). This is very fortunate, otherwise there would not be any chances to produce economically fissile materials. It can be also noticed that the low reprocessing costs of bred fuel rather deprive the direct enrichment of economical justification. One should then remember that the fissile breeding is less efficient at higher enrichments (self-destruction). Instead, the reprocessing of spent fuel is more expensive (3 - 12 %), thus rather encouraging for the fuel rejuvenation /16/. In addition to this, since the fuel cycle contribution to the total costs of advanced breeders (except of the fast breeder) is rather small (1 - 5 %) the approximation that C' represents roughly the total breeder costs may be also accepted. In any case such simplification is conservative and only may increase the certainty of the conclusions. Finally, it should be mentioned that though the values of support ratios given in the Table 1.1 seem overoptimistic /16/, this does not affect the reprocessing cost estimates, since these are relative ones (expressed in LWR costs).

On the basis of the data inserted in the Table 1.1 the condition of advanced breeder economy, expressed in formulas (1.5) and (1.10) has been presented in the form of diagrams (figs. 1.1, 1.2). In the first two pictures, the straight lines corresponding to various contributions of the fuel cost into the LWR costs determine the maximum admissible breeder non-fuel costs for given support ratio that can be compensated by the net income from the electricity and fuel sale. Or, the other way round, it may be understood also as the minimum support ratio required for given non-fuel cost of the breeder, if it has to be economic. The dotted lines concern the present circumstances and forecasts while having assumed the average present cost estimates F'/C_L = 15 % with the uranium price share F_u = 7.5 %.

In connection with the spallator economy (fig. 1.1a) it should be noticed that the assumed energy self-sustainment requires significant energy production in the blanket. Thus, the more strict fission suppression (e.g. fast fission) may be undesirable, since the electricity purchase for the linac supply becomes then necessary. The decision, whether to produce more energy or to have

Table 1.1

Selected Data for Fissile Breeding Economics

Source	LWR				Fusion-Fission Hybrid			Spallator		Fission (Fast) Breeder		
	total fuel costs $\frac{F}{C_L}$ %	uranium costs $\frac{F_u}{C_L}$ %	enrichment costs $\frac{F_e}{C_L}$ %	reprocessing costs $\frac{F_r}{C_L}$ %	non-fuel costs $\frac{C'}{C_L}$	support ratio S	[1]reprocessing costs $\frac{F_r}{C_L}$ %	non-fuel costs $\frac{C'}{C_L}$	support ratio S	non-fuel costs $\frac{C'}{C_L}$	support ratio S	reprocessing costs $\frac{F_r}{C_L}$ %
/2/	28.0	8.3 16.2	7.9	8.								
/3/	24.0	9. 22.6	13.6	5.								
/4/				3.	4.2	16	3.1					
/5/				5.	3.5	16	.5					
/6/	30.6	9. 15.6	6.6						10			
/7/	28.1	5÷7 18.8		6.2						1.5 .9	[2].15 [2].5	24 9
/8/	26.9	7.3 11.6	4.3	[3]11.6								
/9/		8÷15						[3]1.5	5			
/10/				4.0	2.7 3.1	10 13	[3].6 [3]4.0					
/11/					3.0	16	[4]1.0					
/12/											.3	4.
/13/								[3]1.8	9			
/14/	32.6	6.2 17.4	11.2	12.4								
/15/											.75	

[1] may be valid for spallators

[2] assumed

[3] capital

[4] total fuel cycle

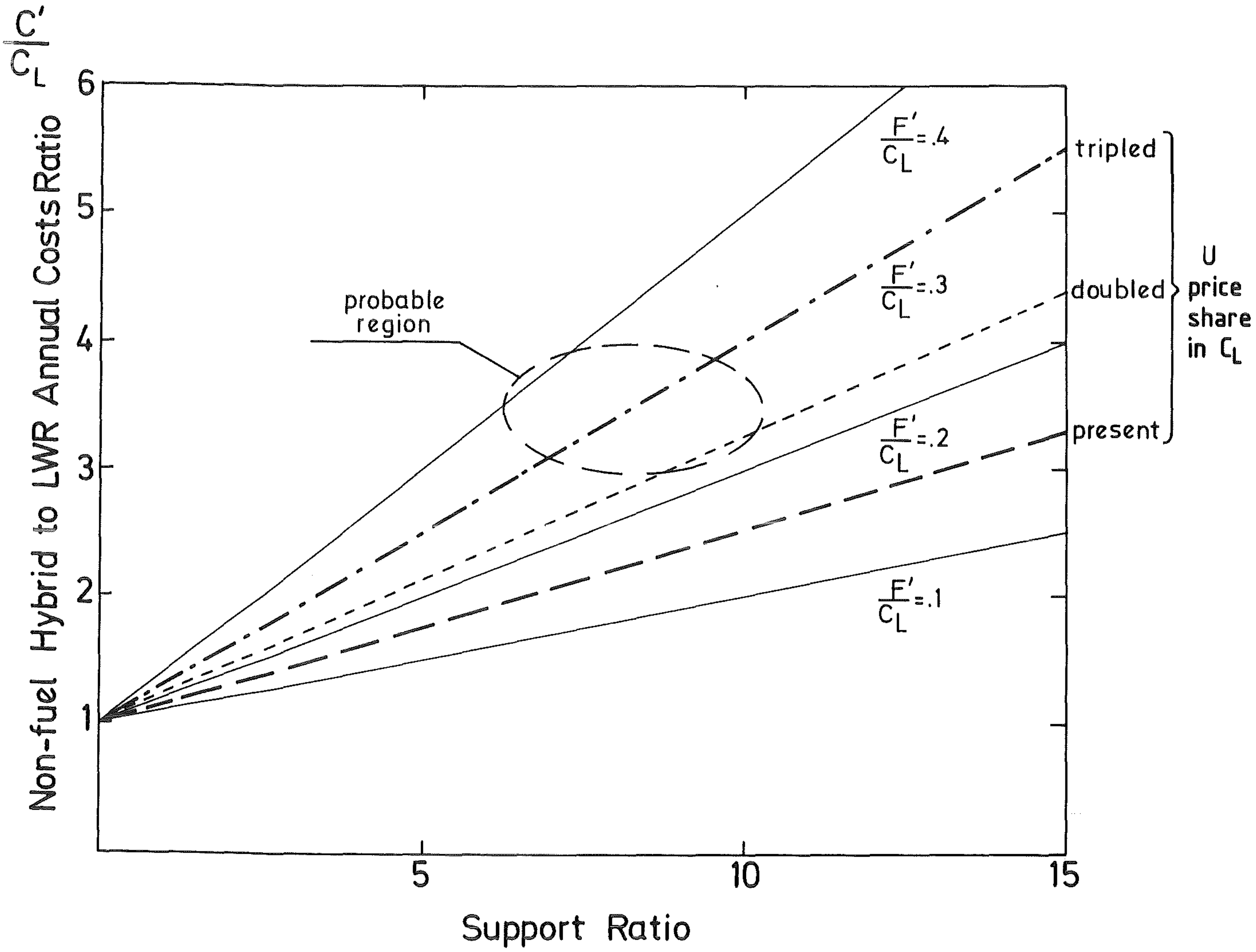


Fig. 1.1a Maximum admissible fissile breeder non-fuel costs as a function of its support ratio for various uranium prices
a) hybrid case

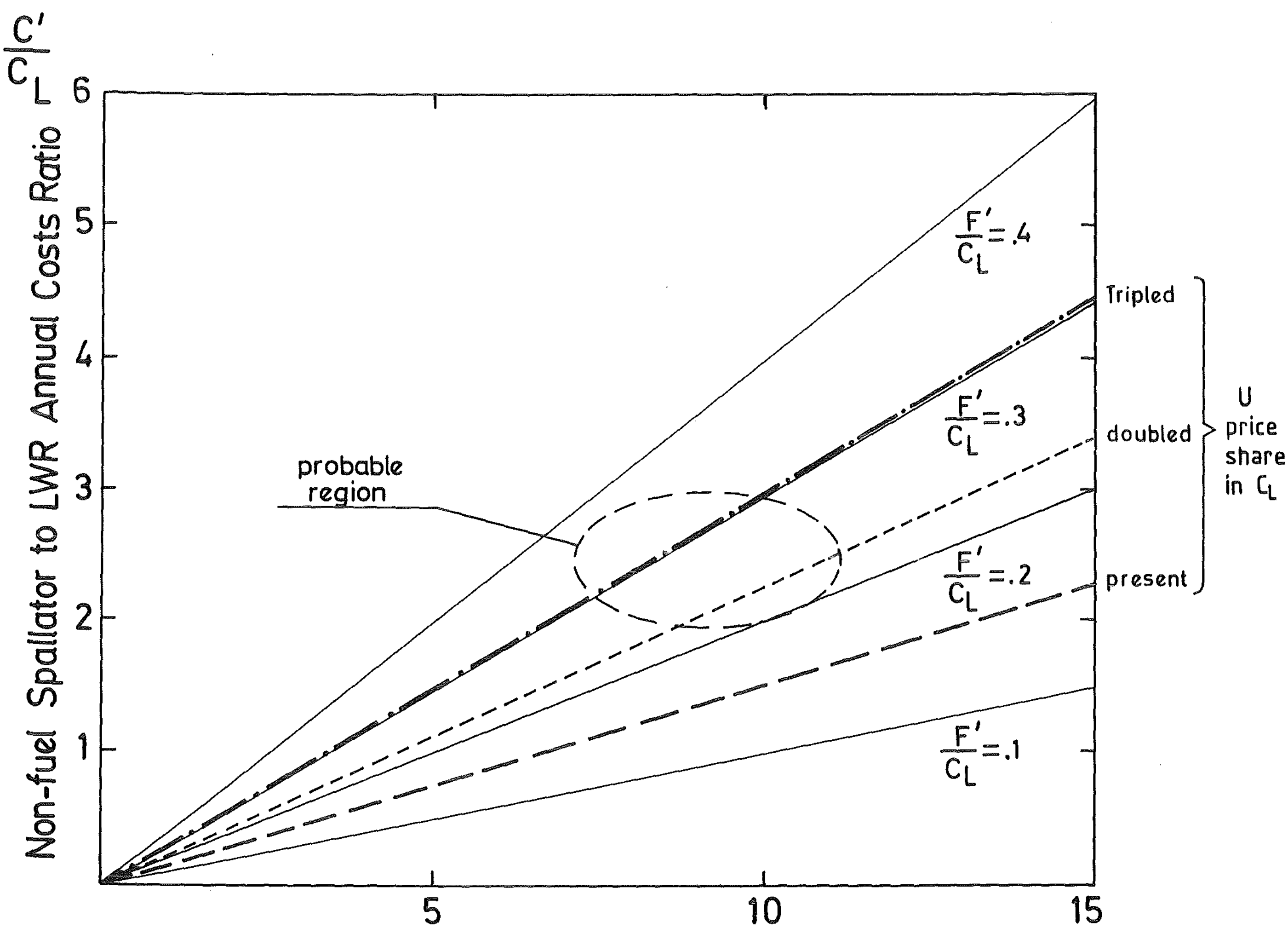


Fig. 1.1b Maximum admissible fissile breeder non-fuel costs as a function of its support ratio for various uranium prices
b) spallator case (energy self-sufficiency assumption)

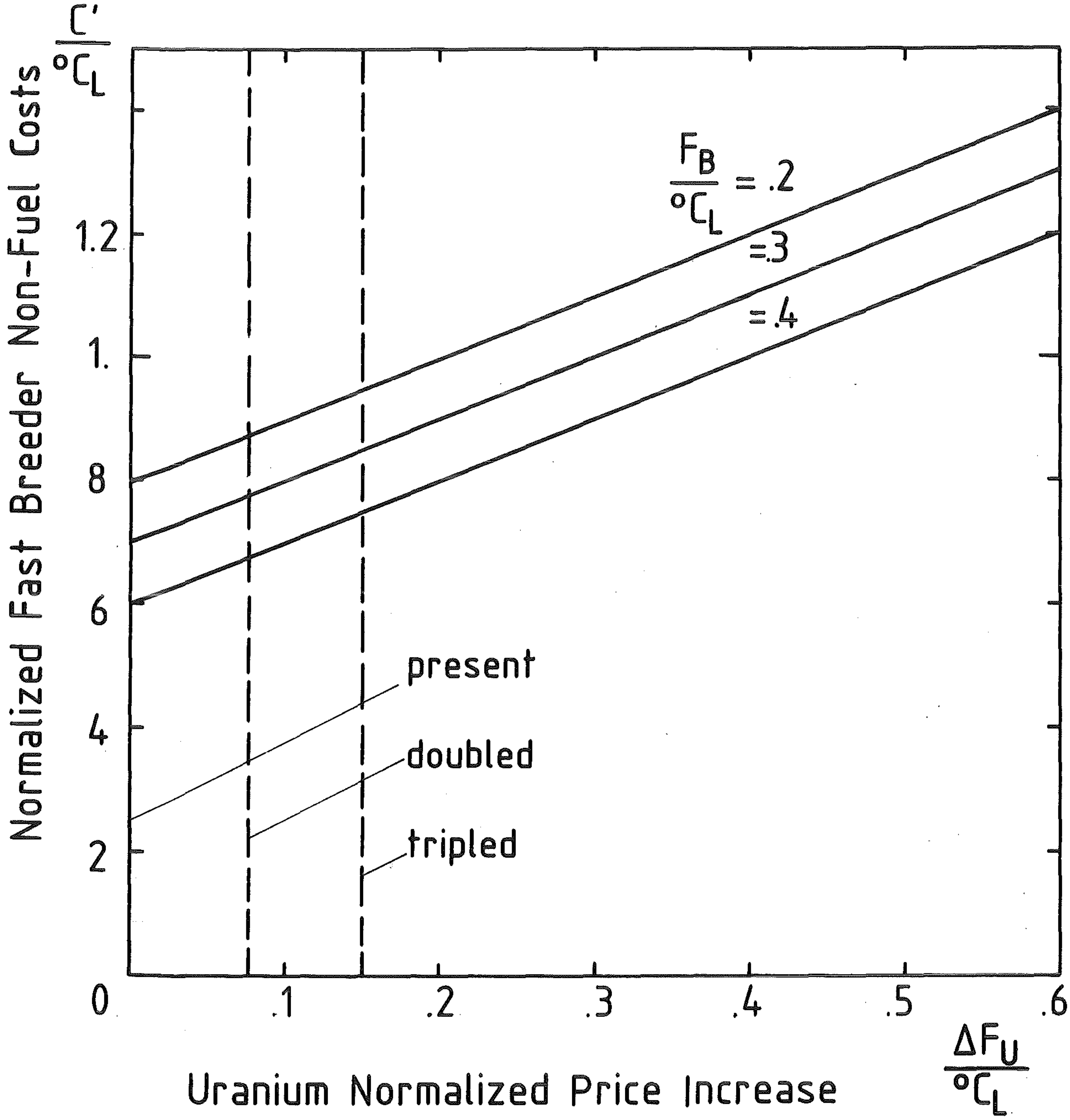


Fig. 1.2 Maximum admissible fast breeder non-fuel costs as a function of uranium price for various fuel reprocessing costs

higher support ratio at higher breeding cost or, in other words, to breed more but to buy electricity at given power or the opposite - requires (now unavailable) reliable data concerning the neutron production efficiency (per energy released) in the spallation process and the accelerator efficiency. The analysis of this question has been recognized to surpass the scope of the present study and the common assumption of the spallator energy self-sustainment (thus rather favourizing the energy production, seeing the relatively low fuel prices) has been made.

In case of fast breeders reduced to fuel self-supply the diagram is different than in figs. 1.1a and 1.1b and presents the maximum admissible non-fuel breeder costs as a function of uranium price share increase (in LWR costs) for several breeder fuel cycle cost values (fig. 1.2).

The diagrams presented in figs. 1.1a and 1.1b give rise to certain optimism, since even conservative estimations of the system parameters remain within the area of the system economy. In efforts to be realistic the most uncertain values of the parameters estimated on the basis of the Table 1.1 were corrected for the conservative indication of "probable regions" (in the authors opinion). The cost of spallator was assumed 30 - 50 % higher than the one given in the Table 1.1 and the support ratio of a hybrid was halved on the average. And in such circumstances roughly the tripling of the uranium prices clearly makes these systems economic.

In spite of recent much less alarming prognostics of the world economic growth rate and thus the energy consumption, in view of the inevitability of the long term development in the global scale, the energy scarcity is still only a question of time. And being conscious of the lengthy path to commercialization of new technologies (proof of technical feasibility, demo-plant, prolonged construction cycle and finally the market penetration process of great inertia), one should not at all recognize the present scientific effort in this field as premature. It may be reminded here, that now, over 40 years after the first physical realization of controlled fission chain reaction and in spite of the well proved economy of nuclear energy, it contributes into the total electricity production even in the most of high industrialized countries only in ca. 10 - 30 % /18/!

It seems natural that also the economical status of the pure fusion should be commented here. Though non-economical aspects give pure fusion certain advantages over fission based nuclear energy, it must be clearly stated here that it has no chances to be economically competitive against other forms of nuclear energy. Estimating the total costs of fusion reactor ca. 3 times greater than the ones of LWR only an increase in uranium price by a factor of several tens times might compensate the costs of thermonuclear energy. And such price increase is impossible even within very distant future. First, already lesser ore price augmentation makes many low grade uranium deposits economic, thus damping its further price increase, second, the fissile breeding becomes profitable at still less expensive uranium ore. It obviously does not mean that the fusion research should be abandoned; to the contrary, it is shown here that the hybrid version of fusion reactor can be economic within the foreseable future at realistic uranium price increase.

Considering the problem of the fast breeder economy as deserving separate studies and treating it here rather marginally, however, one can notice that only the more optimistic values of system parameters can assure the economy of fast breeders. The main cause of these difficulties are high costs of the fast breeder fuel reprocessing, that at present exceed the uranium ore + enrichment prices, thus excluding the fast breeder economy from the very beginning. The success of the fast breeder is then principally conditioned by lowering its fuel cycle cost expressed in LWR units, that may be achieved in part by the direct fuel cycle costs reduction and significant increase in the LWR costs in result of the increase in the uranium price.

The performed comparison of fissile breeding concepts justifies an optimism at least with respect to spallators and fusion-fission hybrids.

An increase in the uranium ore and enrichment contribution to 30 %, of LWR costs, corresponding to ca. triple present uranium price, that should not be recognized as unrealistic, can assure the hybrid and seemingly more easily - the spallator economy. The less encouraging perspectives of fast breeders and the preclusively high costs of pure fusion energy suggest to concentrate more means and efforts rather in the field of hybrid and accelerator breeding. In view of the above and of the decades long time that must pass before any advanced technology can significantly participate in the energy production at the national level, the present scientific activity in this field is fully substantiated. Only in this way the proper energy policy and making right decisions in the right time may be assured.

2. Computational questions of blanket neutronics

As has been already mentioned, in the entirety of physical and technological problems of ANES' blankets the technical solutions should subject to the physical indications and requirements. Therefore, though not at all neglecting the engineering difficulties one should start from the physical aspects of the main neutronic problem i.e. the fuel breeding. The objective is to formulate on the basis of physical premisses the indications that could serve as reliable guidelines for the blanket design. The question of fuel breeding is to be discussed as the problem of maximization of the selected reaction rates in the source driven systems.

2.1 Reaction rate maximization in source driven systems

The neutron balance in these systems can be expressed by the equation

$$S + M = R_u + A + L \tag{2.1}$$

where

- S - source rate
- M - multiplication contribution
- R_u - selected reaction
- A - parasitic absorption
- L - leakage

The terms M, R_u, A and L are mutually bound by means of the neutron flux, thus R_u depends on the remaining terms M, A and L and their significance may be extremely different in various systems.

The objective is the maximization of the reaction rate R which in general is described by the expression:

$$R = \int_{V} \int_{\Delta E} \phi(E,r)\ \Sigma_R(E,\bar{r})\ dE\ dV \qquad (2.2)$$

where

$\bar{r}$ - position vector
E - neutron energy
ΔE - non-zero flux energy interval
V - system volume
ϕ - neutron flux
Σ_R - macroscopic cross-section for selected reaction

As it can be seen from the above formula the value of R depends on one hand on the space-energy correlation of the neutron flux and the given cross-section and on the other hand on the total number of neutrons in the system. The first factor decreases the probability of neutron losses whereas the second one is determined by the multiplication processes.

The most complexed case when all the components influencing the selected reaction rate are of comparable significance is discussed below. The other cases are:

1) Dominant leakage	small system volume
2) Dominant parasitic absorptions	big system volume
3) Leakage and parasitic absorptions negligible	

Ad 1) In this case the reaction rate is basically determined by the correlation of neutron flux and selected cross-section in the phase space /19, 20, 21/. If the given reaction is of the 1/v type the strongest possible neutron slowing-down is desirable, which simultaneously reduces the neutron mean free path and thus hindering their escape increases the probability of neutron interactions.

Ad 2) The flux shaping in these circumstances lies upon the concentration of neutrons in the space-energy region where the ratio of the cross-section for given reaction to the parasitic absorptions is most advantageous.

Ad 3) The maximum R is obtained simply for maximum neutron multiplication in the system.

In the case when no single component of neutron balance clearly dominates in its influence on the rate of given reaction no simple indications of choice between partially contradictory requirements can be formulated.

The neutron multiplication in non-fissile media is always a threshold process, thus requiring fast neutrons. From this point of view, therefore, any slowing-down interactions are profoundly undesirable, meanwhile the leakage reduction needs slowing-down.

Also the desired reaction utilizing neutrons and thus preventing from leakage losses simultaneously is a process competing with the neutron multiplication. Also most frequently the cross-section of selected reaction culminates in the resonance region or is 1/v type, what signifies that just slow neutron flux is well correlated with such cross-section. Then the intuition suggests the spatial separation of both (fast and slow) neutron flux maxima correlated with the respective cross-section ones. The maximum of high energy flux responsible for the neutron multiplication should coincide in space with the respective multiplying (e.g. (n,2n) cross-section), whereas the lower energy flux maximum should take place where the macroscopic cross-section for given reaction is maximum.

Additional difficulties appear if simultaneously some nuclear reaction should be maintained at desired level and other should be strictly avoided - as for instance the production of fissile and fusile nuclei associated with the highly exoergic reactions (fissions) whereas the energy production in the system should be mimimized. Then the objective function is not a maximum reaction rate but the maximum reaction rates ratio obtained while satisfying certain additional conditions. In this case the optimization is the search for an extremum with restrictions which is a particularly delicate problem requiring investigations beyond the scope of the present study.

2.2 Numerics reliability

The present study covers no experimental research, it is then important to estimate properly the reliability of performed numerical calculations. Its evaluation can be done through the discussion and careful selection of admissible simplifications and approximations. Such decisions are not always simple, since it is sometimes difficult to foresee which factors do not affect the results of calculations and which ones are essential. And the neutron transport codes and data do not reflect exactly the physical reality.

Below we try to discuss the significance of the most important effects which are not strictly treated in numerical calculations.

2.2.1 Void streaming

This effect results from this kind of blankets heterogeneity (non full coverage) which causes the direct losses of source neutrons and the scattered flux leakage from the breeding zone through various cavities since the full coverage of the neutron source by the blanket is not possible. Penetrations are needed for beam injections (neutral particles - magnetic confinement fusion, laser or ions - inertial confinement fusion, protons - spallators), vacuum pumps, etc. The one dimensional (frequently also the two dimensional ones) codes are not able to reflect this three-dimensional effect. The objective is to diminish (and estimate) the neutron losses which may significantly exceed the solid angle represented by the voids in the blanket /22/. With the use of one-dimensional transport code however the semi-quantitative investigation of these effects can be carried out.

The idea lies in the representation of leakage losses by adequate left boundary conditions. These are:
a) albedo (plane geometry)
b) vacuum (cylindrical and spherical geometries).

In this way the real voids in the system are represented on one side in the plane geometry and "mixed" and neutron losses are simulated by non-ideal reflection (fig. 2.1a). The albedo should correspond to the voids solid angle as seen from the point of neutron scatter, that is however difficult to evaluate since the voids are not completely "black" (neutrons may be scattered from the openings walls back into the breeding zone). In the source group(s) this estimation may be easier. E.g. in inertial confinement devices voids face directly the source i.e. the neutron leakage exactly corresponds to the voids aperture solid angle as seen from the plasma. Thus, the unity minus albedo coefficient should be equal to the respective solid angle normalized to 4 π. In the other groups, however, neutrons

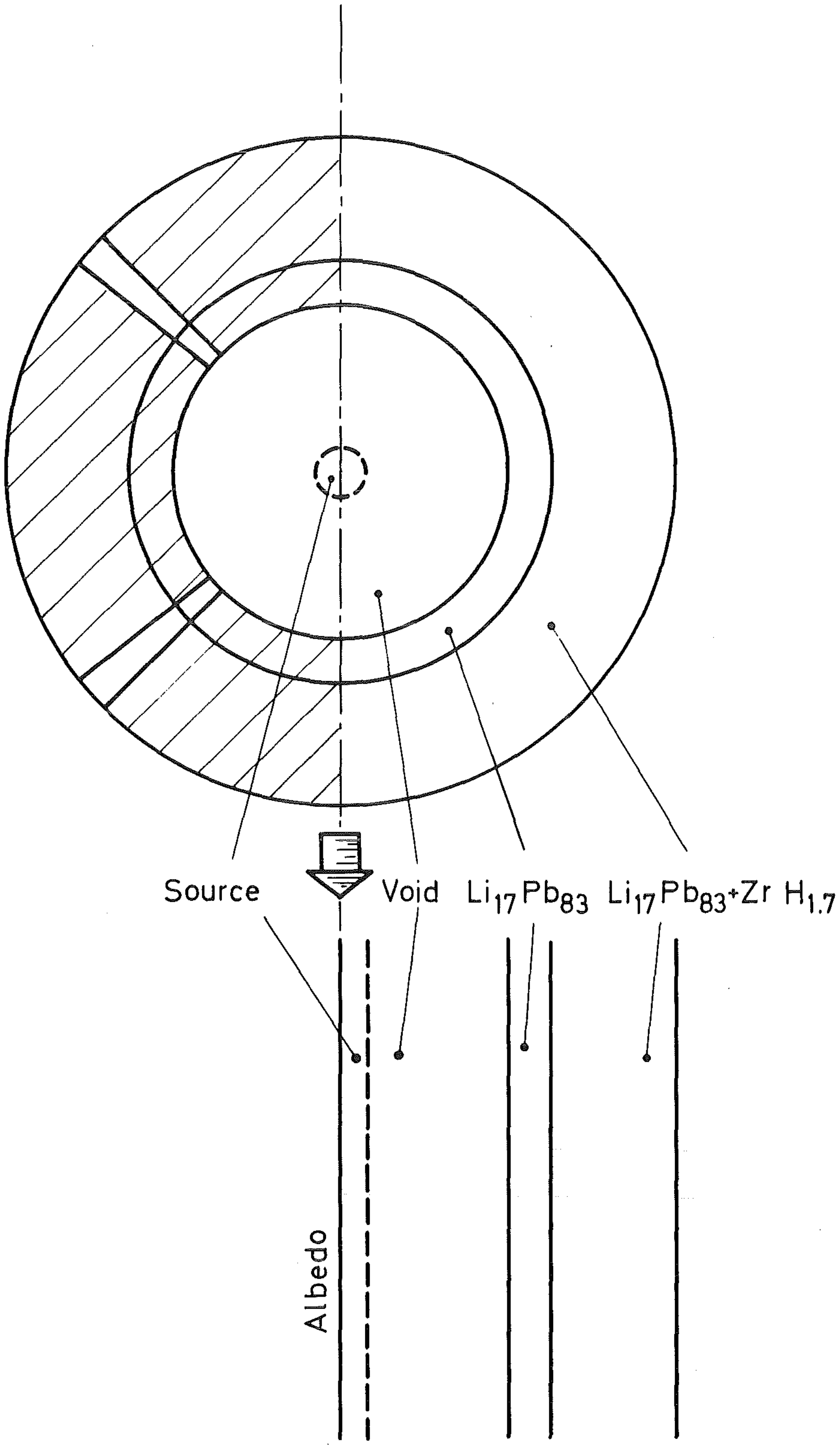


Fig. 2.1a Void streaming estimation with the one-dimensional geometry
Albedo model (plane geometry)

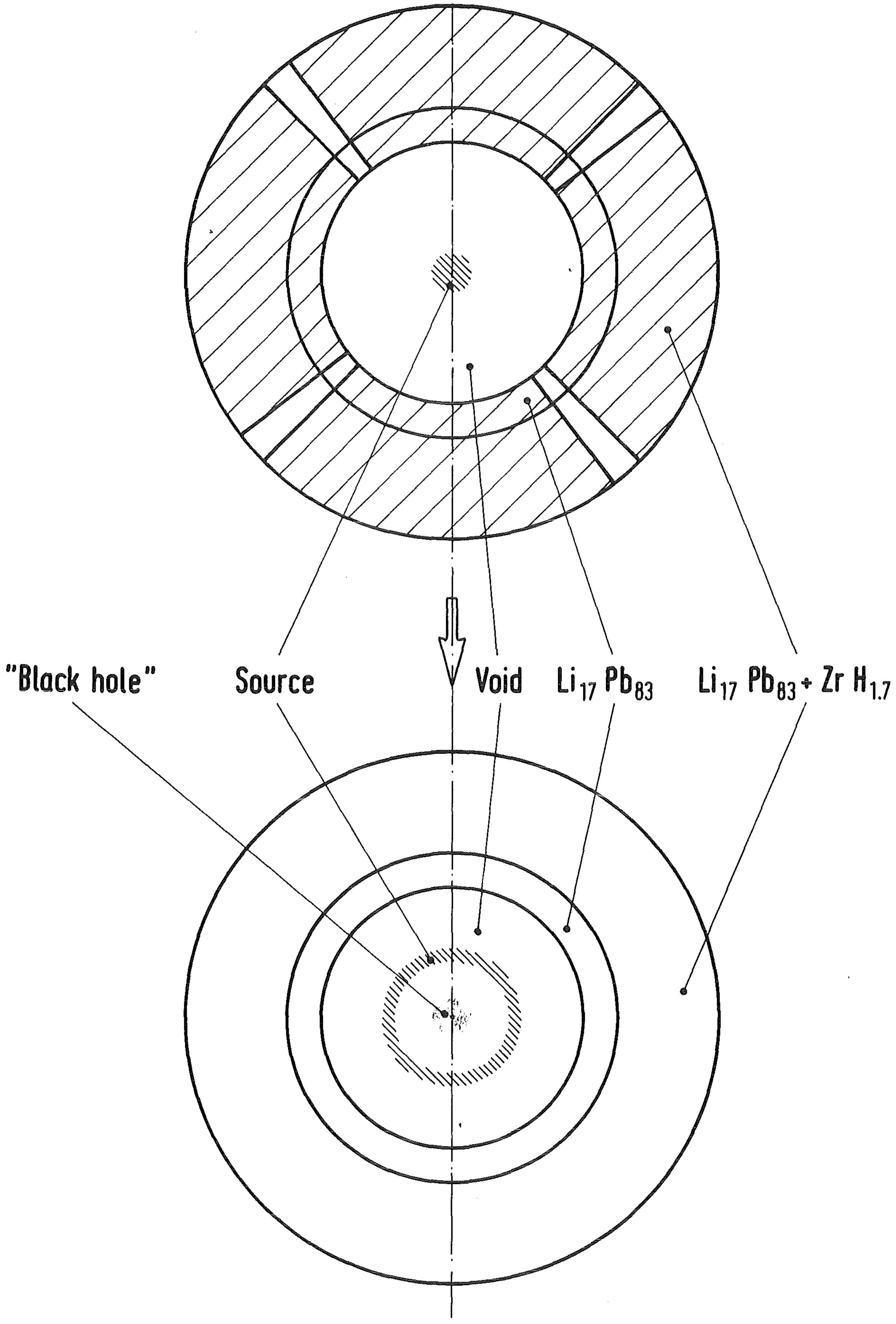


Fig. 2.1b Void streaming estimation with the one-dimensional geometry
Void left boundary condition (spherical geometry)

are scattered in various directions and only small fraction scattered in the volume opposite to an opening "sees" it at well defined solid angle. The rest faces first of all the opening walls. Seeing that, the losses ten times lower than the ones in the source group were assumed arbitrarily in the remaining groups.

Similarly in the case of vacuum boundary condition (albedo equal to 0 in all groups) the size of the centrally situated "black hole" and of the neutron source should correspond to the effective openings solid angle as seen by the source and scattered neutrons. It implies the well defined proportions between the radii of the left boundary, of the surrounding neutron source and of the first wall. The "black hole" radius is determined by the losses of scattered neutrons and then for this value, the neutron source (plasma) radius is determined by the source neutron streaming (fig. 2.1b).

The objective of the calculations based upon these models was rather to indicate the possiblities of reduction of neutron losses and not their absolute evaluation. Even though having no possibilities to reduce the source neutron losses one still can effectively suppress the leakage at all the other energies by reducing the number of neutron returns into void chamber, where from they leak out through the voids. This can be realized by the intense slowing-down that shortens the neutron life in the system. Remembering that one must not reduce the total number of neutrons by disturbing the multiplication process, the slowing-down zone was preceded by the multiplying one. The dependence of the leakage and of the tritium breeding on the thickness of the layer preceding the slowing-down zone are presented in figs. 2.2 and 2.3.

An advantageous effect that can be noticed i.e. the effective suppression of the neutron leakage also by hydrogeneous layers not facing directly the source neutrons is explainable. The hydrogen slowing-down power of 14 MeV neutrons is low whereas the one of heavy metals (due to inelastic processes) decreases first below ca.

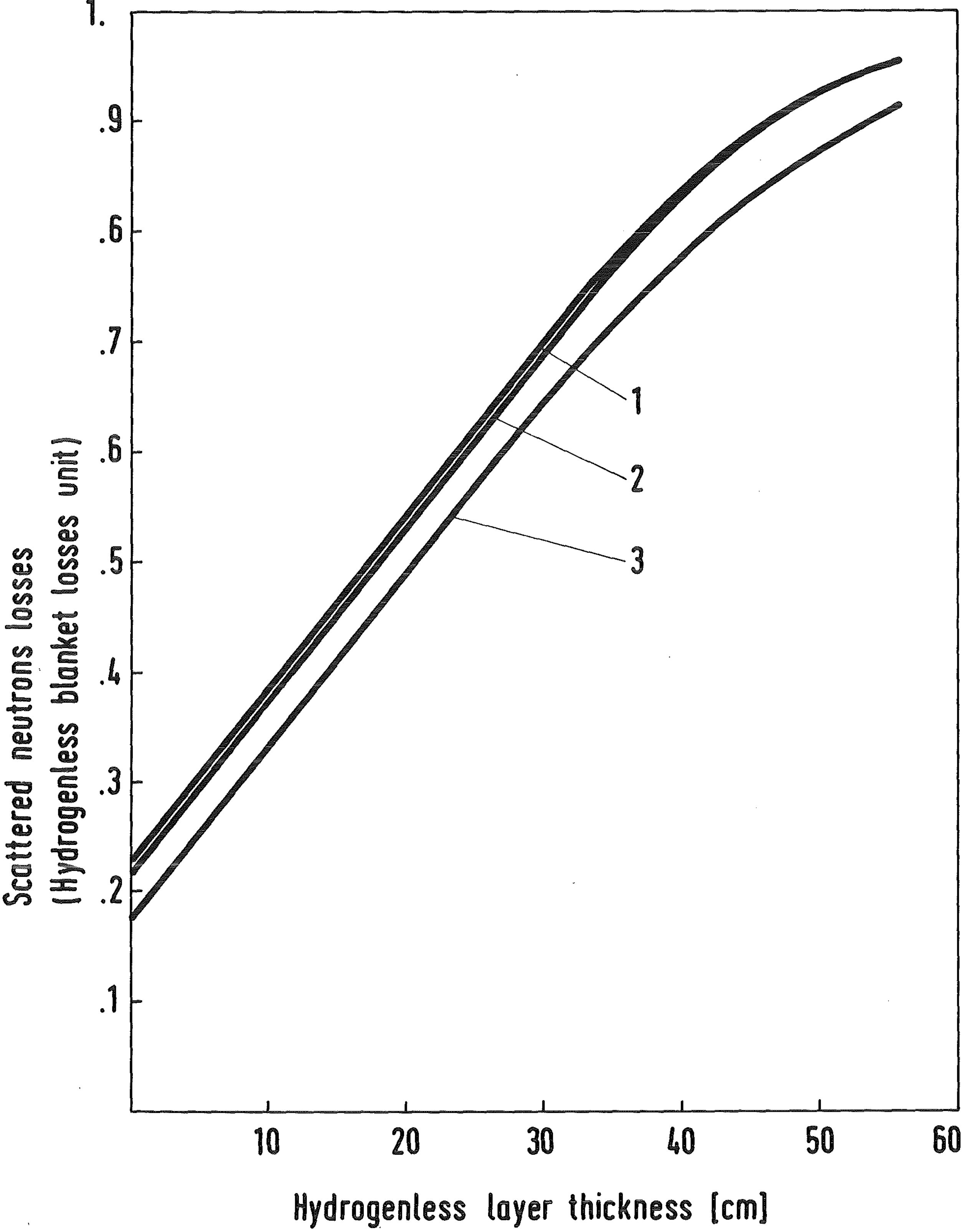


Fig. 2.2 Tritium breeding rate vs. $Li_{17}Pb_{83}$ layer thickness (Hydrogeneous layer composition 58 % $Li_{17}Pb_{83}$ + 30 % $ZrH_{1.7}$)

1) } albedo/plane geometry } 10 % void aperture solid angle
2) }

3) Void left boundary condition } 1 % void aperture solid angle

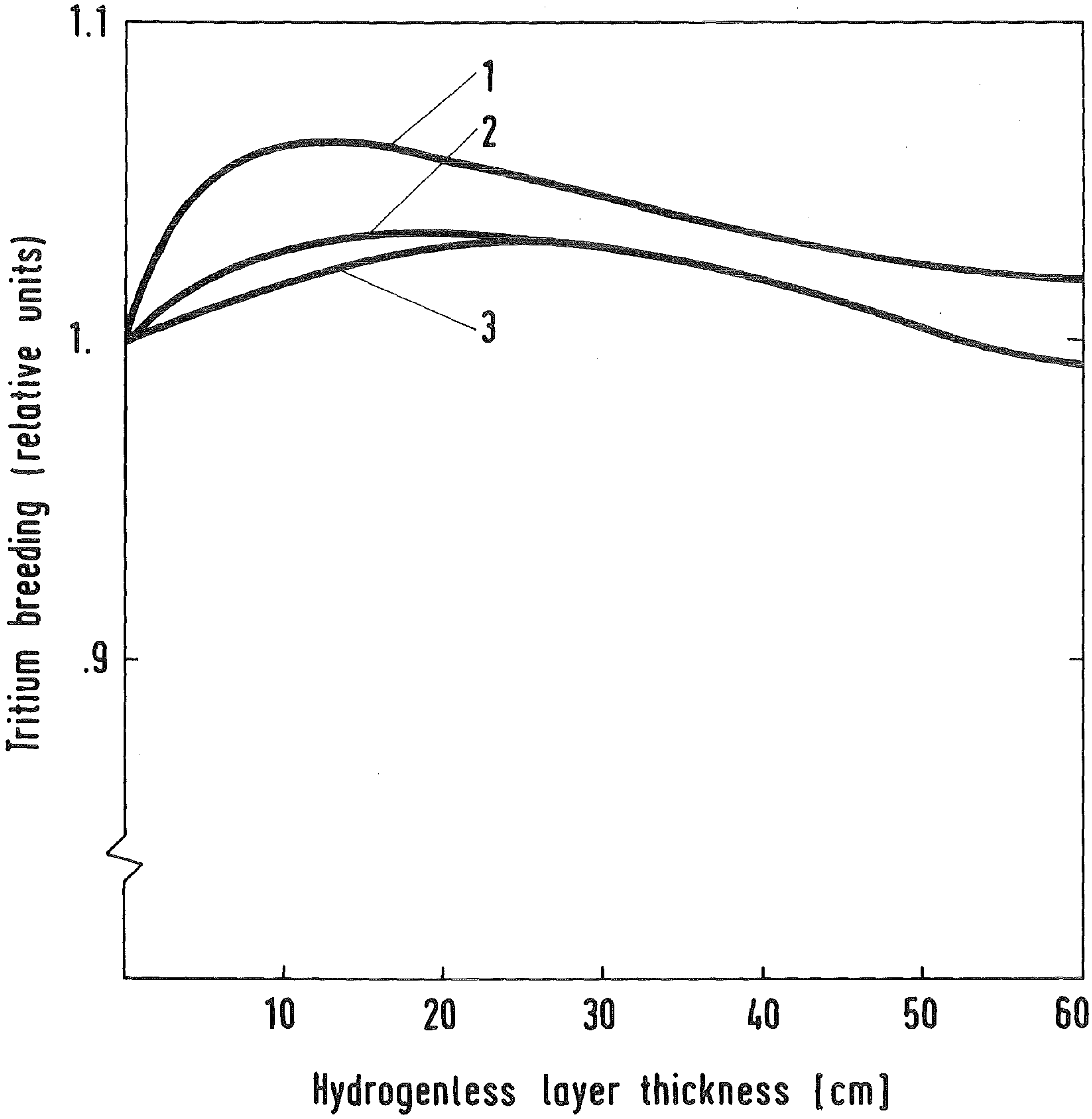


Fig. 2.3 Void streaming vs. $Li_{17}Pb_{83}$ layer thickness
(circumstances as for Fig. 2.2)

1 MeV. It means that hydrogen slows-down neutrons more efficiently only after one inelastic scattering (then being of energy 1 - 2 MeV). Therefore certain (not too thick) heavy metal layer preceding the hydrogeneous one and advantageous for the neutron multiplication proves not harmful from the point of view of leakage reduction.

Comparing the achieved attenuation of streaming with the results presented in /22/ one may state in conclusion, that in a well moderated system the void streaming should not exceed much more than 50 % the mean solid angle subtended to the voids as seen from the neutron source.

2.2.2 Self-shielding effects

One can distinguish two kinds of self-shielding (S-S) effects:
- the ones caused by the sharp cross-sections maxima (strong deviations from the straight line, as a function of lethargy) called resonance self-shielding and the ones practically resulting solely from the system heterogeneity - the non-resonance self-shielding.

The neglect of the resonance self-shielding which is admissible in pure fusion neutronics leads to grave consequences (e.g. to unexpected criticality) in fissile breeding systems, therefore will be analysed in detail in connection with hybrid and spallation breeding (see 4.2).

Thus, we confine ourselves here to discuss only the non-resonance self-shielding.

In addition to the large scale heterogeneities (not 100 % coverage) the breeding blanket is heterogeneous also in the scale of typical neutron mean free path. These heterogeneities are e.g.

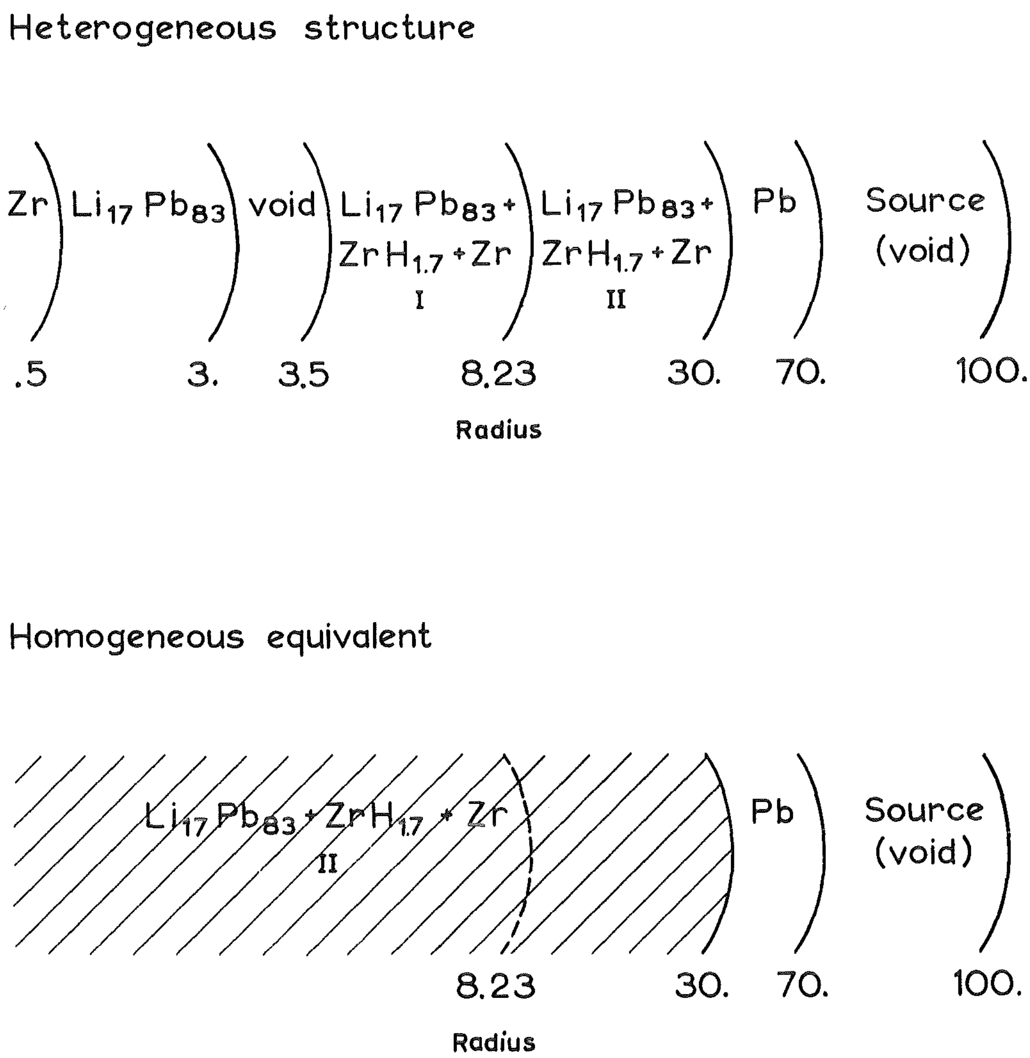


Fig. 2.4 Example of a calculational model for the estimation of non-resonance self-shielding effects in one-dimensional geometry

cooling tubes, fuel rods or multiplying/moderating/breeding balls in the case of pebble-bed blanket concept. The S-S effects resulting from this can be explained in the classic way i.e. by the flux depression in the inner part of high absorbing medium (these nuclides are hardly or not at all seen by neutrons and thus should not be taken into consideration in the evaluation of the homogenized medium macroscopic cross-section. The effect may change the distribution of neutron captures between the strongly and weakly absorbing media and is significant for low energies (thermal and epithermal ones).

Such heterogeneity cannot be represented even in three-dimensional calculations because of its too fine structure, but may be estimated approximately also with the use of one-dimensional code. The idea lies in the comparison of the breeding rates of homogeneous structure with the partially heterogeneous one (fig. 2.4). The number of nuclides in the heterogeneous region and in the same volume of the homogeneous one are equal and correspond to the average density of homogeneous medium. The difference in given reaction rate (e.g. fusile breeding) is caused then only by the different spatial distribution of all the materials.

The heterogeneous structure results not only in the flux depression in the inner part of absorbing medium but also in the simultaneous flux enhancement in the surrounding volume. This increased flux area is not limited to weakly absorbing medium (moderator) but covers all the immediately neighbouring zones, the strongly absorbing one including. Thus, the increased reaction rate in the vicinity of lumped absorber compensates, to a degree, its decrease within the lump itself. Quantitatively this effect expresses the formula for the heterogeneity correction factor:

$$f_n = \frac{R_{het} + \Delta R}{R_{hom}} \qquad (2.3)$$

where

R_{het} - selected reaction rate in lump volume with the consideration of heterogeneity

R_{hom} - selected reaction rate in lulmp volume in homogeneous system

ΔR - increase in reaction rate in the lump vicinity in the heterogeneous case

and the results of transport calculations are presented in the Table 2.1.

Table 2.1 Non-resonance self-shielding correction factors f_n for tritium production

Li lump radius 1.5 cm	90 % ^{6}Li 13 % H_2O	Li_{nat} 13 % H_2O	Li_{nat} 26 % H_2O
in lump only (ΔR = O)	.889	.853	.811
the breeding in the neighbouring zone including	.998	.985	.955

As it can be seen in the Table 2.1 the S-S effects are much less than it might be expected on the basis of neutron attenuation within the breeding medium itself (The lump dimensions exceed several mean absorption free paths). It is also worth to notice that apparently against intuition the higher 1/v type absorber concentrations result in weaker S-S effects. It can be however, explained by the spectrum hardening associated with higher ^{6}Li concentrations. In this case, much more neutrons are absorbed already at higher energies where the heterogeneities (differences between media cross-sections) are much less pronounced.

2.2.3 Other reliability questions

Contrarily to the previously discussed simplifications and approximations the remaining ones are of secondary significance from the point of view of the present study needs. Therefore, we confine ourselves to several remarks.

In general, the possible remaining causes of errors lie in neutron data imperfections. One may list here the uncertainties in neutron multiplication estimation (Be or $^{7}Li(n,n')$ equivalent reaction), the lacking secondary neutron distributions and/or other data (e.g. secondary gammas, kerma factors or DPA etc. Having been concentrated upon the breeding problems, the above shortcomings may be recognized principally as less singificant. The more important lack of reliability of Be neutron multiplication estimation /23, 24, 25/ has been avoided by applying lead based neutron multipliers, also in view of certain techno-economical drawbacks of Be (as high price, swelling and toxicity). Thus, the question of the choice of beryllium as the neutron multiplier in fusion devices that requires reliable data, remained beyond the scope of this study.

3. Neutronics of fusile breeding

The variety of present fusion reactor blanket concepts requires clear indications for designing an optimum blanket structure on the basis of physical premisses though simultaneously not neglecting the engineering ones. Within the severe constraints imposed by technological possibilities, the optimum nuclear design should be identified first, determining, in turn, the objective for technological solutions. In this hierarchy of aims the fusile breeding seem to overshadow other questions and impose the decisive requirements upon the blanket design. According to the views expressed earlier in this study a thorough analysis of physical process occurring in the blanket and the idea of proper neutron flux shaping in the phase space /19, 20, 21/, create a reliable basis for defining the guidelines of blanket designing.

3.1 General considerations

In view of the above remarks the maximization of the tritium breeding becomes the main premiss for the blanket design and as it was mentioned in the chapter 2, the space-energy correlation of the neutron flux with the adequate cross-sections can assure the achievement of this goal. These cross-sections contain several components: the one responsible for the total number of neutrons in the system (principally determined by the neutron multiplication) and the second one of main tritium breeding reaction - $^{6}Li(n,\alpha)T$, seen as the process competing with the leakage and parasitic absorptions. Instead, the idea of adequate flux shaping can be reduced to the statement that in limited volumes the maximum rate for given reaction can be obtained when having the neutron flux well peaked at the energy of the maximum of the respective cross-sections (neutron multiplication, tritium production) and situated in the region of maximum concentration of the respective nuclides. In other words, the concentration of neutron flux in this volume with its simultaneous minimization outside this area can assure the best neutron utilization in such circumstances.

In case of maximizing the rate of the 1/v type reaction, e.g. the main tritium producing reaction $^{6}Li(n,\alpha)T$, the objective is clear, one should slow-down neutrons as intensely as possible. According to authors earlier suggestions /25, 26, 27/ for the 14 MeV neutrons the two following physical processes are to be used in order to achieve the above aim most effectively:

1) neutron inelastic multiplication processes - (n,xn) - for the energies above 1 MeV
2) proton elastic scattering - below this energy.

These quite general suggestions of the proper cross-sections-flux correlations usually signify, in practice, a difficult compromise between several competing processes contributing to the maximization of the desired reactions; neutron multiplication/inelastic slowing-down - parasitic absorptions, elastic scattering/moderation - leakage. The significance of each process must be evaluated in view of the neutron balance: production - losses (leakage, absorptions).

The rather high thresholds of neutron multiplication reactions (n,xn) make the multiplication rate to be sensitive to the neutron spectrum within the fast region. It signifies that not only the direct absorption processes - (n,p), (n,α) etc. decrease the neutron multiplication but also practically all inelastic scatterings driving neutrons down to energies below those thresholds. Since all the above processes usually dominate at 14 MeV, from the point of view of the neutron multiplication the presence of any other nuclides i.e. the non-gaseous coolant and/or structure materials in the multiplying medium is highly undesirable /28/.

The need and the advantages of the neutron multiplication are obvious, but they can be cancelled by insufficient leakage suppression when proper moderation is lacking, so this question is to be discussed below.

The leakage suppression is important for two reasons. First, for moderate breeder thicknesses - i.e. with the mean chord length equal to only several mean free paths of 14 MeV neutrons, the reduction of leakage by slowing-down can be even more important than the neutron multiplication. Second, the neutron losses due to less than 100 % breeding blanket coverage can be diminished by possibly early (i.e. after not many scatterings) neutron capture in the breeding medium, that in turn, is to be achieved also by intense moderation (see Chap. 2.2.1). A hydrodynamic model of neutron transport can be helpful in explaining these effects. The action of hydrogen by the slowing-down process and by the following neutron captures reminds the suction of a pump placed in this area. As a result, one can control the neutron spatial distribution and balance by means of the neutron moderation process.

One should also notice that inelastic processes slow-down 14 MeV neutrons most efficiently i.e. even better than the proton scattering, at this energy characterized by a relatively low cross-section . Thus, the choice of the neutron multiplication, that always is an inelastic process, as the most probable interaction for source neutrons is not in contradiction with the requirement of intense moderation. Therefore, the need for undisturbed neutron multiplication (also as the desired slowing-down process in the higher energy region) and further neutron moderation through elastic scatterings is justified. As a result, their spatial separation seems to be the best solution. The source neutrons should face first the multiplying layer, if possible, free of all other nuclides (except of a small amount of ^{6}Li in order to suppress parasitic losses, see below). Then, the breeding/moderating region should follow the multiplying one.

The optimum thickness of the last one is a function of competing factors. It should be thick enough to utilize most of the neutron multiplication/inelastic slowing-down processes but simultaneously thin enough in order not to hinder further desired neutron moderation in a hydrogeneous medium. Or in other words, it should not unnecessarily prolong the neutron life in the system, that must result in increased void streaming. The determination of an optimum needs the exact evaluation of the (n,2n) reaction spatial distribution what would require the knowledge of the double-differential cross-sections for the reaction in question, that is not available up to now. However, there are no grounds to expect that the optimum multiplier thickness is peaked. To the contrary, a flat maximum of the tritium breeding is to be expected, thus leading to the conclusion that the non-optimum thickness would not bring significant worsening of the breeding ratio.

Finally, one should not forget the neutron parasitic absorptions in coolants and structural materials, the presence of which sets the lower limit of ^{6}Li concentration at the level where parasitic captures start to compete significantly with tritium breeding in ^{6}Li. This effect becomes more important as compared with the leakage losses with increasing volume of the breeding zone and thus usually softer spectrum reducing the neutron escape. As a result the lithium enrichment (in ^{6}Li) may then prove indispensable.

3.2 Calculations and Results

According to the above indications a broad numerical study aimed at the confirmation of these has been carried out.

In all the calculations, the one dimensional finite element neutron and gamma transport code ONETRA /29/, in cylindrical geometry was applied together with the University of Wisconsin 25-neutron, 21-gamma group cross-section set /30/ condensed from the ENDF/B-IV based, Vitamin C library in P3, S8 approximations.

The following blanket structure was considered: 1 cm-ferritic steel (FS) first wall preceding the breeding region of variable thickness, composed of breeding + moderating media - 88 %, FS structure material - 4 % (all vol. percent). As breeders the eutectic $Li_{17}Pb_{83}$ (also 90 % ^{6}Li enriched) and the metallic natural lithium were selected /31, 32/. The low volumes of coolant and structure material /33/ are admissible due to the liquid form of breeders thus assuring good cooling conditions. The shield of 15 cm FS and then 25 cm steel with borated water followed the breeding zone in all cases. Only the more outer shielding zones, also of steel with borated water, having been of negligible influence upon the breeding zone processes were not always of the same thickness.

It should be also noticed that the fusion reactor blanket optimization is a multiparameter problem, therefore any one- or two-parameter analysis cannot fully reflect its real complexity. Nevertheless, general tendencies can be determined in this way.

The thesis to be demonstrated is the possibility of significant decrease in the tritium breeding volume due to the introduction of hydrogeneous moderator into the breeding zone. This effect can be seen in fig. 3.1 and 3.2 and explicitly in the Table 3.1.

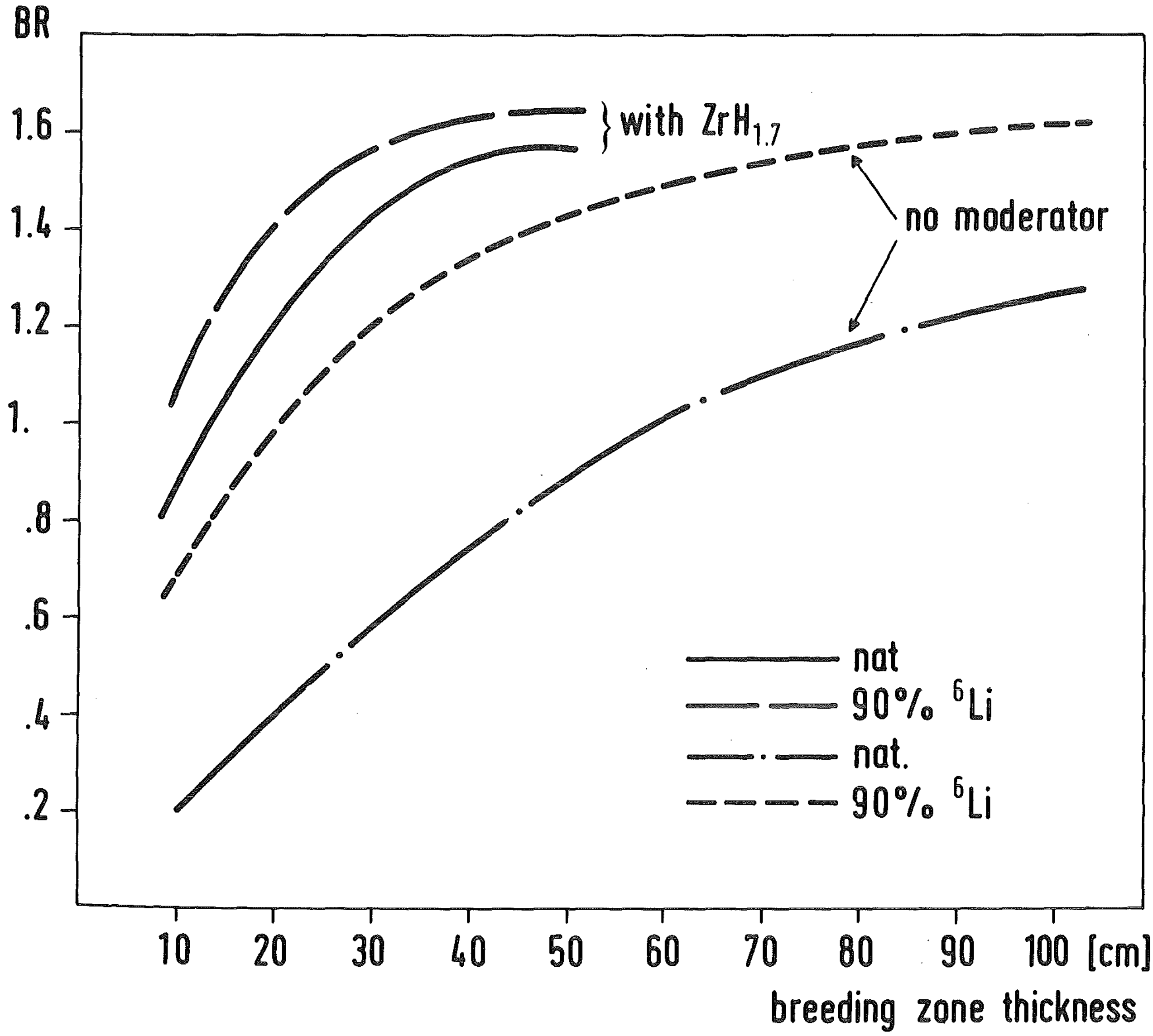


Fig. 3.1 Tritium breeding rate vs. breeding zone thickness (100 % coverage) for $Li_{17}Pb_{83}$

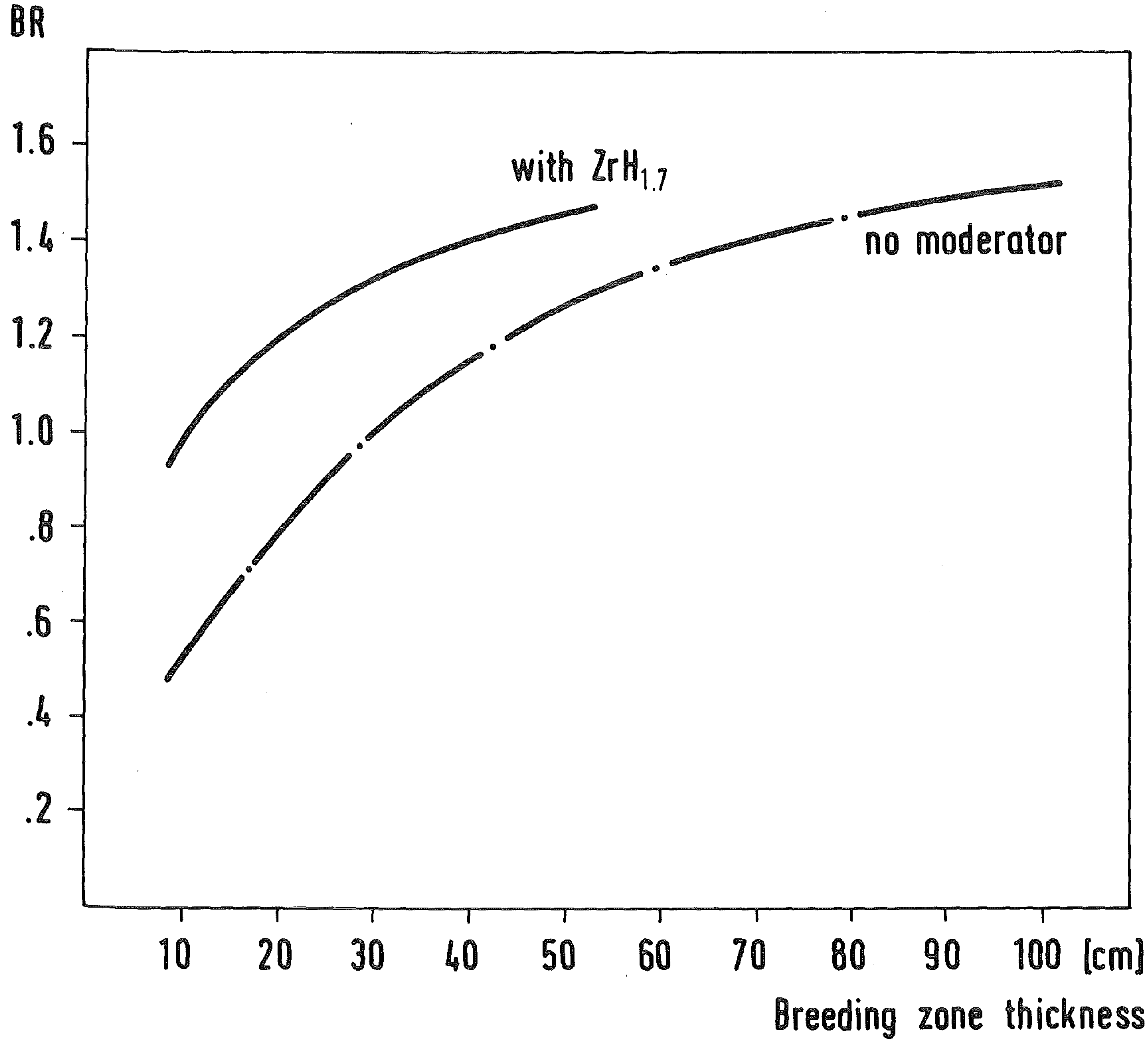


Fig. 3.2 Tritium breeding rate vs. breeding zone thickness (100 % coverage) for Li_{met}(nat.)

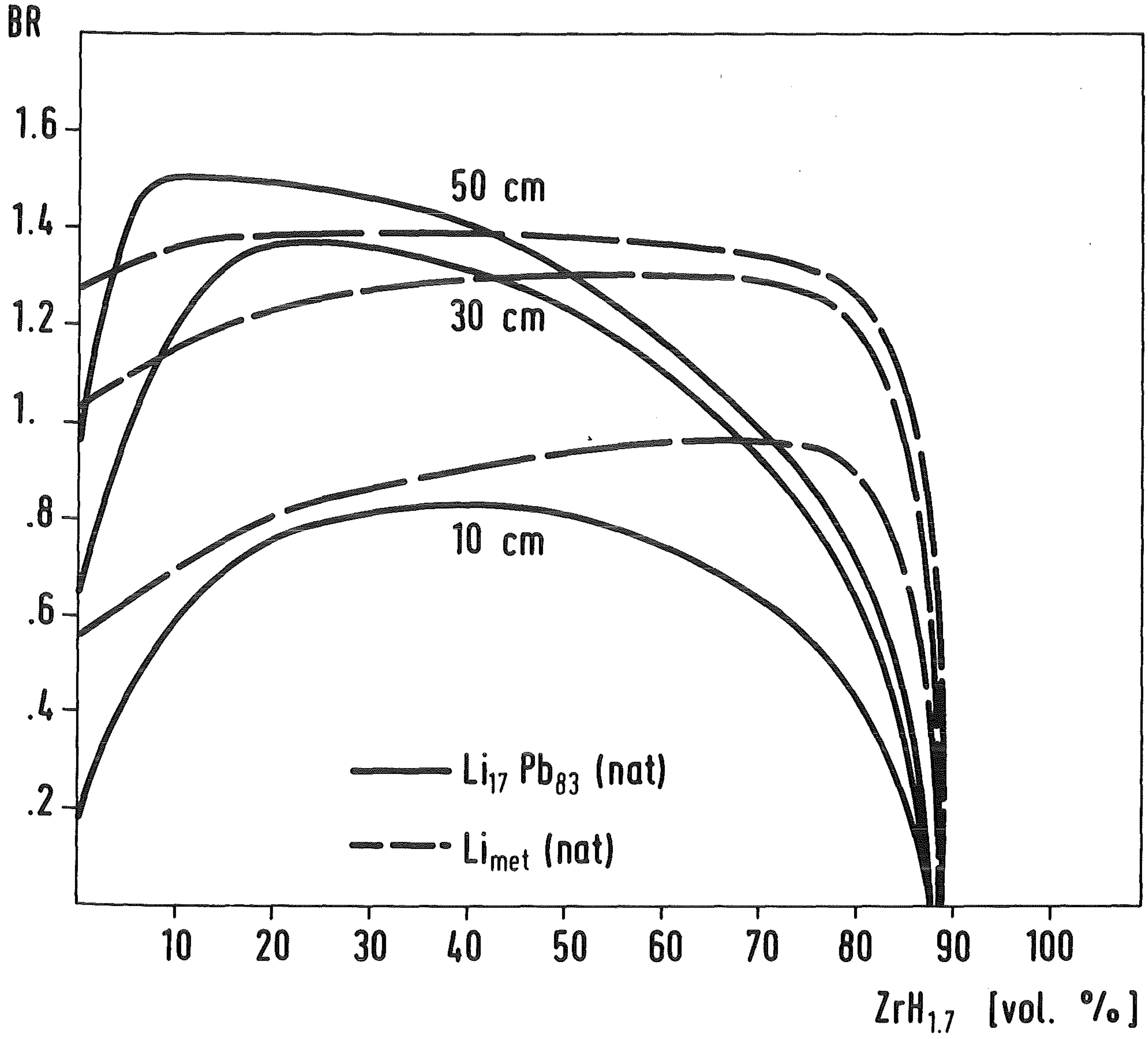


Fig. 3.3 Tritium breeding rate for various breeding zone thicknesses and materials vs. $ZrH_{1.7}$ volume fraction

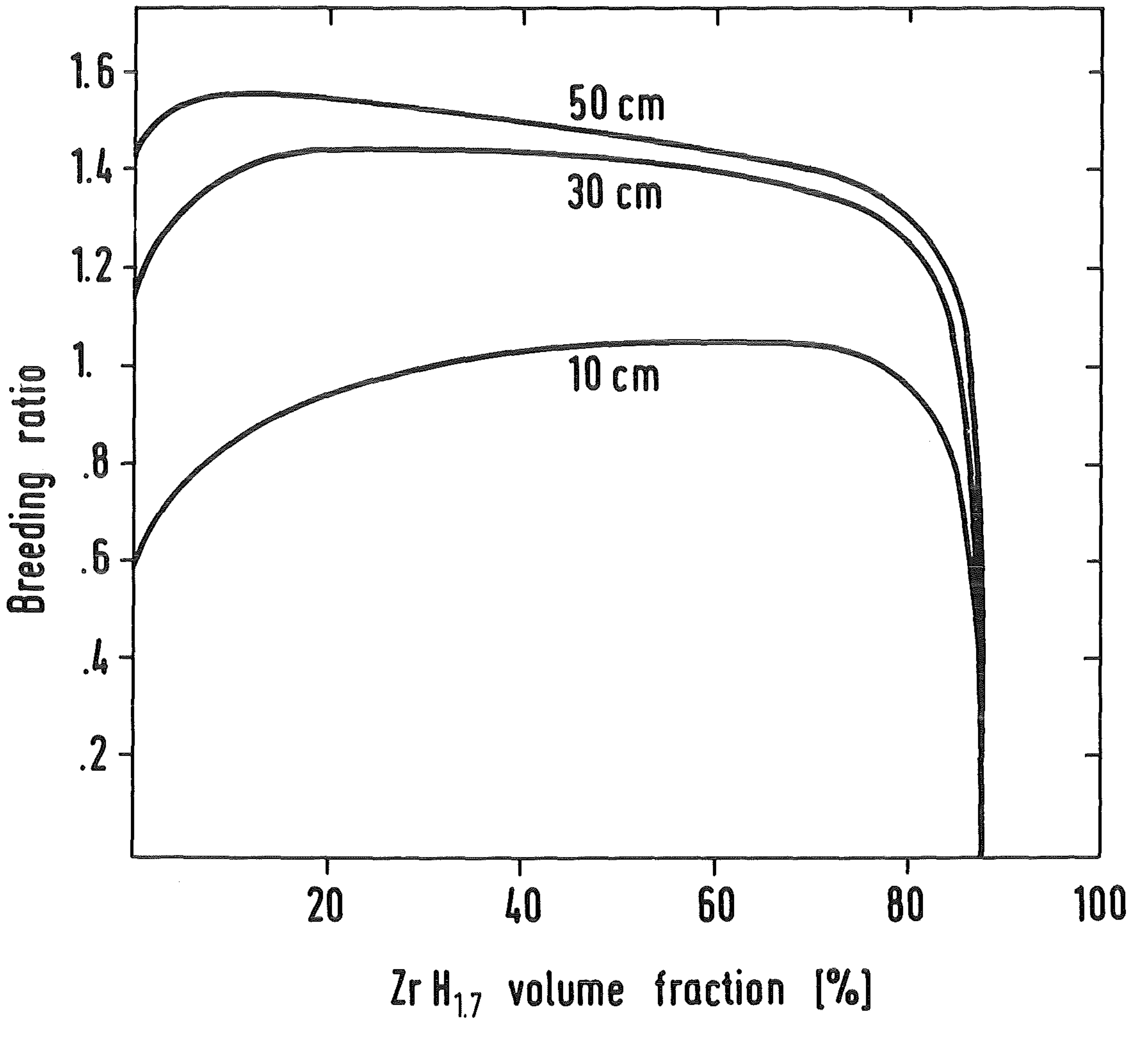


Fig. 3.4 Tritium breeding rate for various thicknesses of $Li_{17}Pb_{83}$(90 % ^6Li) breeding zone vs. $ZrH_{1.7}$ volume fraction

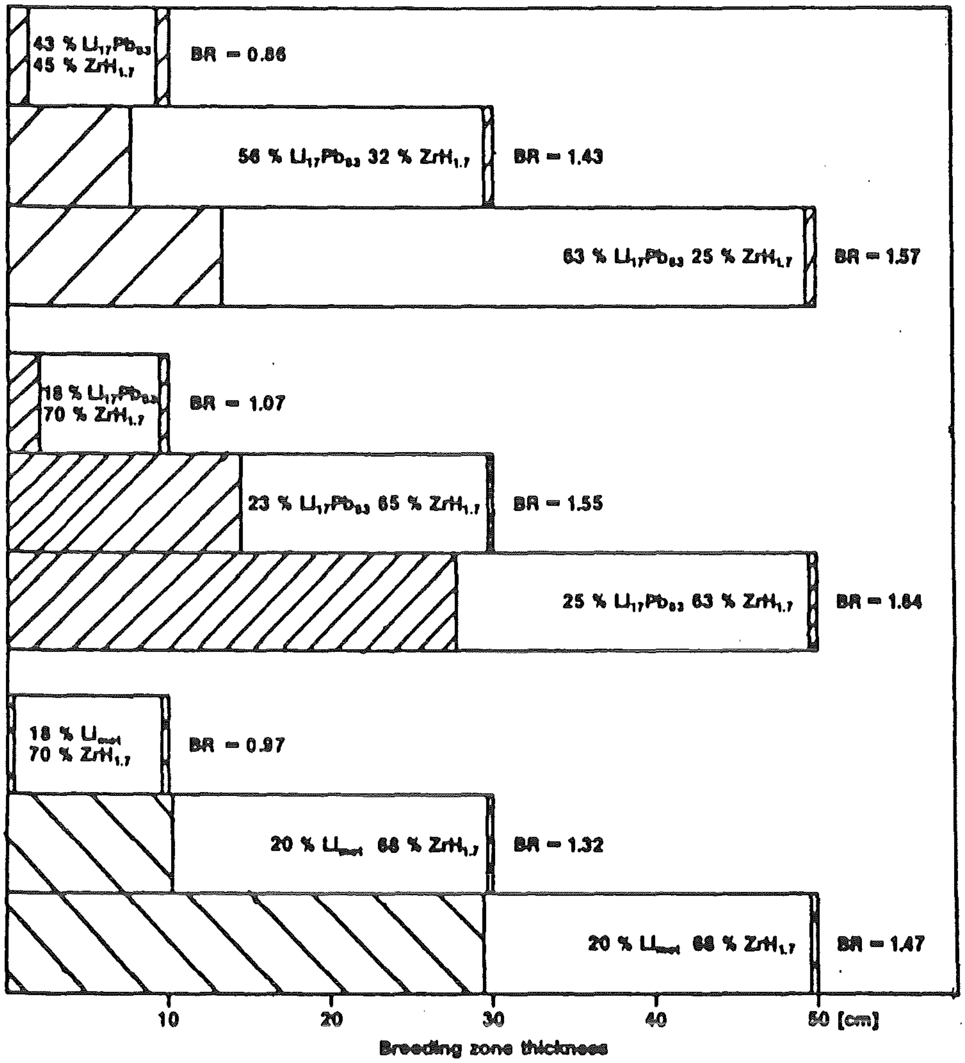


Fig. 3.5, Optimum sandwich structures of the breeding zone for 10 cm, 30 cm and 50 cm blanket thicknesses (in all layers 8 % void, 4 % SS)

88 % $Li_{17}Pb_{83}$(nat)

88 % $Li_{17}Pb_{83}$(90 % ^{6}Li)

88 % Li_{met}(nat)

Table 3.1 Ratio of breeding thicknesses $\frac{d_B}{d_{BH}}$ and breeder volumes $\frac{V_B}{V_{BH}}$ of unmoderated to hydrogen moderated blankets for various materials and breeding rates

Medium \ Br		1.		1.3		1.4	
		d_B/d_{BH}	V_B/V_{BH}	d_B/d_{BH}	V_B/V_{BH}	d_B/d_{BH}	V_B/V_{BH}
$Li_{17}Pb_{83}$	nat	4.7	8.5	5.5	10.	5.7	13.
	90 % ^6Li	2.4	6.0	2.4	5.7	2.4	5.5
Li_{met}		2.8	11.	2.0	4.6	1.8	3.3

It is to be noticed there, that an increase in the breeding zone thickness above 30 cm in the case of $ZrH_{1.7}$ moderated $Li_{17}Pb_{83}$ and above 40 - 45 cm for metallic lithium brings only minor increase in the tritium breeding. The difference in observed "saturation" thicknesses can be explained by higher "transparence" of metallic lithium than the one of lithium lead alloy.

As one may expect, the advantage of having a hydrogeneous moderator in the blanket (table 3.1) is generally greater for lower ^6Li atomic densities (natural $Li_{17}Pb_{83}$) and in the cases of higher concentration (Li_{met}) - for lower breeding rates (or thinner blankets).

The direct influence of the presence of hydrogen containing medium in the blanket on the breeding rate is illustrated in figs. 3.3 and 3.4.

The distinct plateau's (Li_{met}) or broad maxima (natural $Li_{17}Pb_{83}$) of the breeding rate dependence on the hydrogeneous moderator volume fraction indicate useful flexibility in design of the breeding zone. Such shape of these curves signifies that already at low $ZrH_{1.7}$ concentrations most of neutrons are trapped within the breeding zone and the number of absorptions in ^6Li changes weakly until ^6Li density so decreases (together with certain decrease in the neutron multiplication) that even well thermalized neutrons escape from this area and/or are more often captured in the structure materials. It is not surprising that higher ^6Li atomic densities allow for greater moderator volume fraction, unacceptable otherwise, when the absorbing power of the breeding medium is too small. The lower ^6Li concentration in natural $Li_{17}Pb_{83}$ turns the plateau's into broad maxima (fig. 3.3).

One can, however, have slightly higher tritium production than the one shown in fig. 3.3 and 3.4, for given breeding zone thickness, due to some rearrangement of the breeding (or multiplying) and moderating medium. The optimum "sandwich" structures of breeding region (resulted from the removal of the moderator from both inner and outer layer of the breeding zone), obtained with the modified simplex method /34/ are sketched in the fig. 3.5 and the corresponding increase in the tritium breeding is shown in the Table 3.2.

Table 3.2 Gain in the tritium breeding due to the "sandwich" structure of breeding zone (as compared with the moderated homogeneous structure)

Thickness / Medium		10 cm		30 cm		50 cm	
		hom	sand.	hom.	sand.	hom.	sand.
$Li_{17}Pb_{83}$	nat	.84	.86	1.37	1.43	1.49	1.57
	90 % ^6Li	1.04	1.07	1.44	1.55	1.56	1.64
Li_{met}		.95	.97	1.30	1.32	1.38	1.47

A shift of the slowing-down zone backward from the first wall enables higher neutron multiplication in the layer preceding the moderating zone, while the increased moderator density assure sufficient softening the spectrum and trapping the neutrons in the breeding zone. The hydrogeneous moderator situated more backward acts also as a efficient reflector for neutrons of energies < 1 MeV. On the other hand it should be admitted that the consideration of the void streaming effects may recommend slightly thinner multiplying zone and more hydrogen in the slowing-down one in order to reduce this component of neutron losses. Nevertheless, as it can be seen in figs. 3.3 and 3.4, fortunately, the breeding is not very sensitive to the blanket composition, thus always certain useful degree of freedom is left for the designer. Also the outer breeding layer of the "sandwich" as being very thin (1 cm or less) has rather symbolic meaning and may be easily forgotten e.g. for technical reasons, without practical losses in the breeding rate.

In order to have some idea about the validity of the performed calculations in other circumstances certain sensitivity evaluations have been carried out. For this purpose, the influence of the structural materials and the void volume fractions on the tritium breeding for hydrogen containing and non-containing blankets have been checked. In this way, one can roughly estimate the breeding ratios corresponding to other blanket compositions, as well as learn if the presence of hydrogen is advantageous also in this case. The results of the respective calculations are enclosed in the Table 3.3. The derivatives inserted there designate the relative changes in the tritium breeding ratio per one per cent (absolute) change in the void or structure material volume fractions.

Table 3.3 Tritium breeding rate sensitivity to void and structure volume fractions

Breeding thickness / Medium		10 cm No H	10 cm $ZrH_{1.7}$	30 cm No H	30 cm $ZrH_{1.7}$	50 cm No H	50 cm $ZrH_{1.7}$
$Li_{17}Pb^{83}$	$\frac{\Delta BR}{\Delta V}$	.010	.007	.009	.004	.007	.002
	$\frac{\Delta BR}{\Delta S}$	.006	.015	.006	.015	.009	.015
Li_{met}	$\frac{\Delta BR}{\Delta V}$	.007	.004	.006	.003	.005	.002
	$\frac{\Delta BR}{\Delta S}$	.015	.006	.010	.007	.011	.010

As it can be seen in the Table 3.3, according to intuition, the tritium breeding rate is less sensitive for hydrogeneous blankets to the void volume fraction and to the steel volume fraction for higher ^6Li concentrations (Li_{met}) in the blanket.

The decreased sensitivity to the void fraction is due to the reduced neutron leakage from the breeding zone in the presence of hydrogen. Instead, for low ^6Li densities the neutron capture in structural materials becomes more important when in hydrogen containing blankets ^6Li density is still lower.

Also in view of the lack of space and of the resulting from this difficulties in shielding the inner part of tokamak devices, it is interesting to know what additional steel shield thickness is necessary in order to balance the replacement of the 10 cm steel shield layer by the mixture of 35 % $Li_{17}Pb_{83}$ + 53 % $ZrH_{1.7}$ + 4 % SS + 8 % void. In other words, the question is what additional

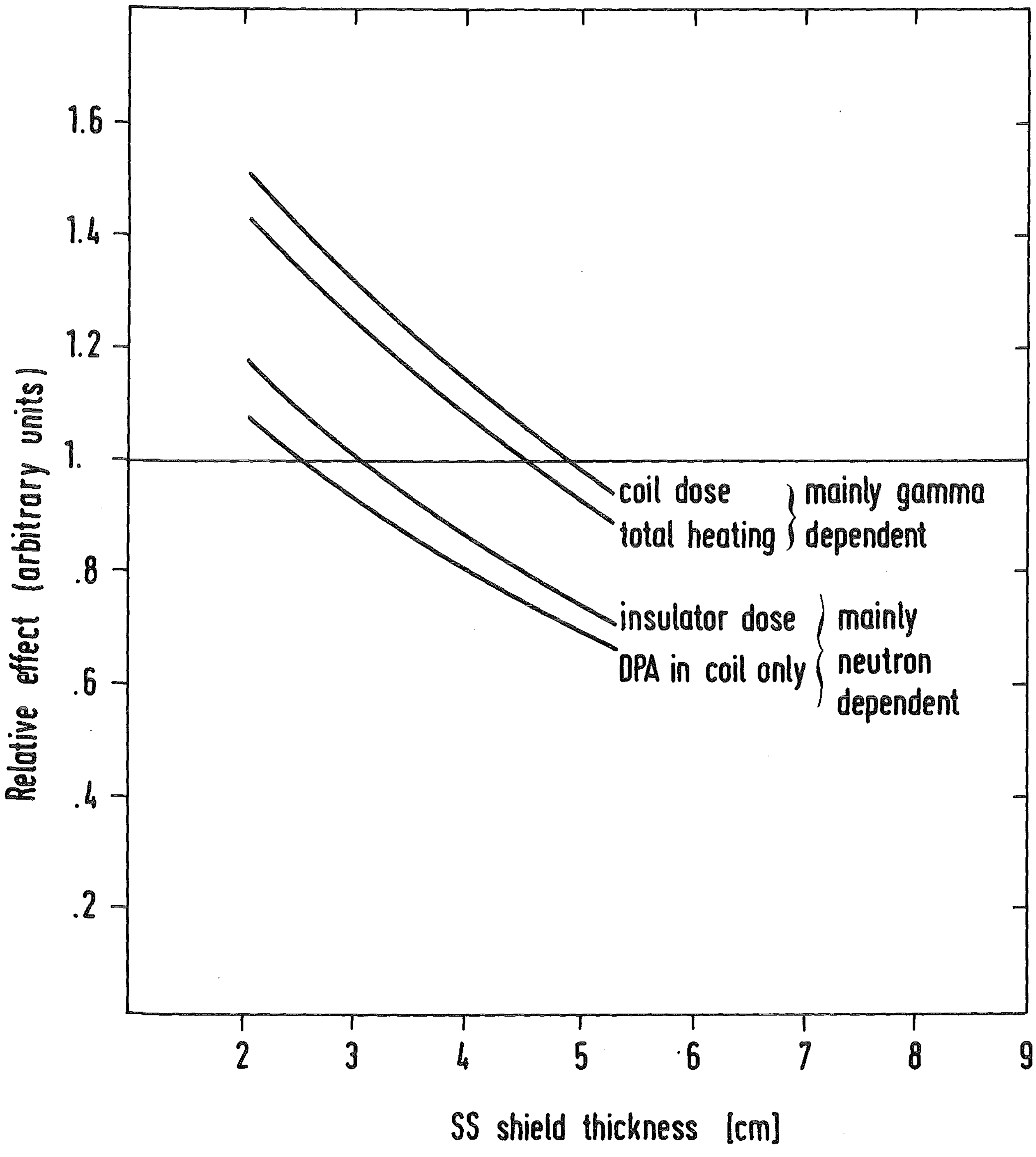


Fig. 3.6 Compensation of weaker radiation attenuation in the 10 cm thick breeding zone by additional steel shield (Breeding medium: $Li_{17}Pb_{83}$ + $ZrH_{1.7}$)

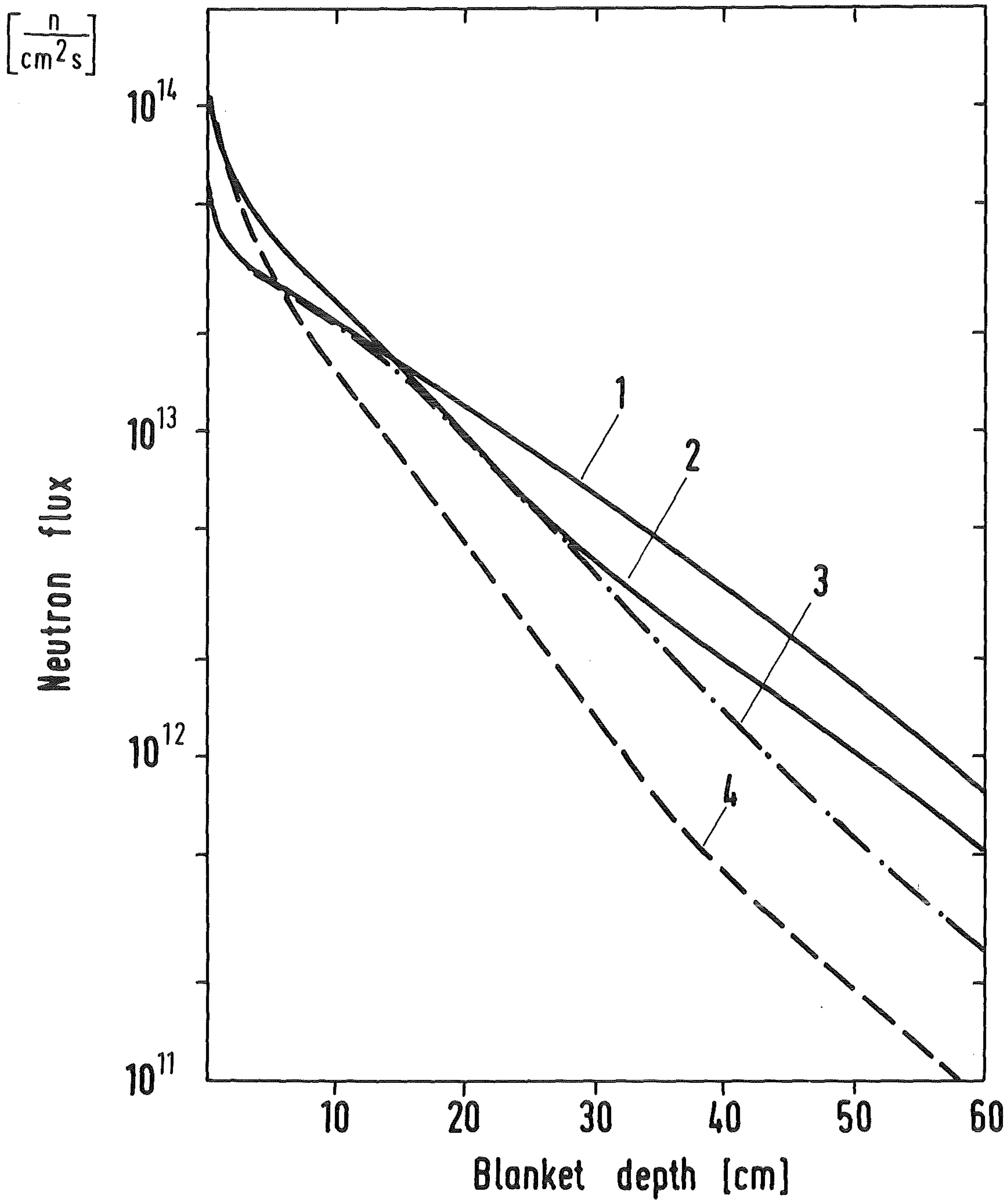


Fig. 3.7 Examples of 14 MeV source neutron flux distributions for various materials and geometries, normalized to 1 MW/m^2 of 14 MeV neutrons

1. 60 % Li_2O sph. geom. point source
2. 60 % Li_2O plane geom. and source
3. 88 % $Li_{17}Pb_{83}$ sph. geom. point source
4. 58 % $Li_{17}Pb_{83}$ 30 % ZrH_{17} plane geom. and source

In cases 1 + 2 SS = 10 vol. % void = 30 vol. %

3 + 4 SS = 4 vol. % void = 8 vol. %

space (filled with steel) is needed for to have the same radiation attenuation for the breeding blanket + thinner shielding as for steel shielding alone. The results of calculations are illustrated in fig. 3.6.

The picture seen in the fig. 3.6 indicate relatively thin additional shield needed for the compensation of less effective radiation attenuation in the breeding zone than in the steel shield. However, what thickness must be designed for additional shielding in a particular case can be decided only having known which effect determines the needed shield thickness. For instance, having coil insulator less sensitive (ceramic) to the radiation damage, one should rather expect the coil dose or nuclear heating (gammas) as shielding criterion and thus require thicker additional shielding. Instead, in the opposite case, for more delicate coil insulation, much thinner additional shield may prove sufficient.

Simultaneously, it must be emphasized, however, that the validity of the above considerations is limited to these parts of the blanket where the streaming effects can be neglected. In most parts (e.g. in outer blanket zone) the radiation streaming through voids (beam ducts, divertors, etc.) determine the radiation damage and nuclear heating in the magnet system, what is not possible to evaluate with the available 1-dimensional neutron transport code.

3.3 Selected spatial distributions and spectra

The multiplicity of significant distribution and spectra for the breeding blanket nuclear design make a proper selection of them in light of the limited scope of this study to be indispensable.

Therefore, the following spatial distributions of greater practical importance for the breeding blanket nuclear design have been selected+:

1) Neutron flux distribution
2) Tritium breeding distribution
3) Power distribution

Ad 1) The significance of neutron flux distributions is obvious. On this basis all the other distributions (reaction rates, energy release etc.) can be determined. On the other hand, since these processes are energy dependent and there is no sense in presenting the neutron flux distributions in all the groups (and the total flux also is not very meaningful) only the first (source) group has been chosen to be presented. This flux is important because it practically predetermines the distribution of the threshold processes like significant for radiation damage gas production reactions (n,α) and (n,p) and neutron multiplication $(n,2n)$ (except of beryllium case) (fig. 3.7).

Independently, of the above opinion of predominant role of the source neutrons it is interesting to know the neutron energy distributions, first of all in the most sensitive place of the blanket-in the first wall (fig. 3.8).

+The DPA distribution is not listed here as not only from the neutron direct interactions dependent (neutron transport code can give only its very rough estimation).

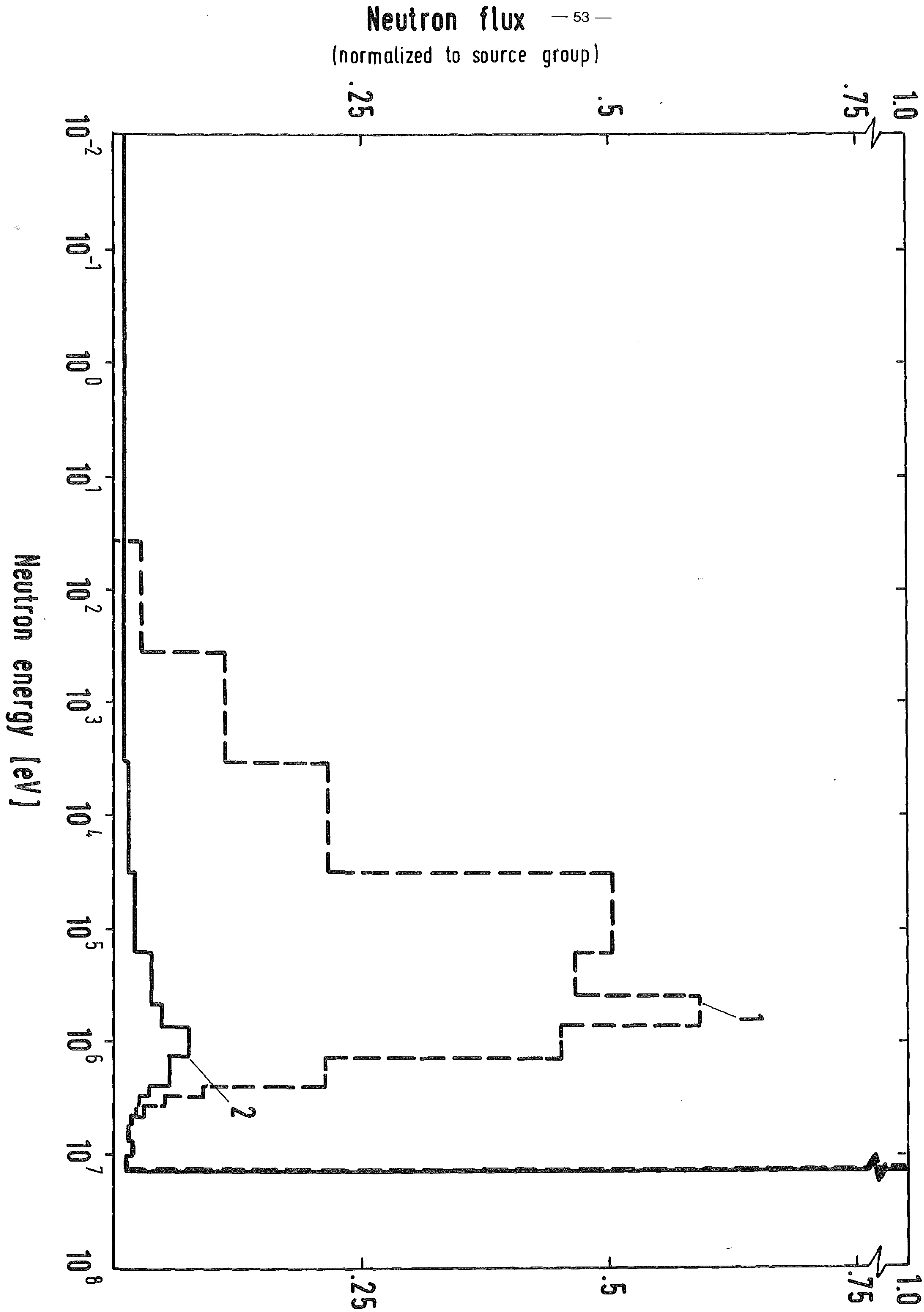


Fig. 3.8 D-T Neutron spectra in the first wall (1 cm thick) (breeding zone thickness .8 m)

1) 88 % $Li_{17}Pb_{83}$, 4 % SS, 8 % void, sph. geom. point source
2) FW followed by 1.5 cm H_2O layer, cyl. geom. line source the rest as above

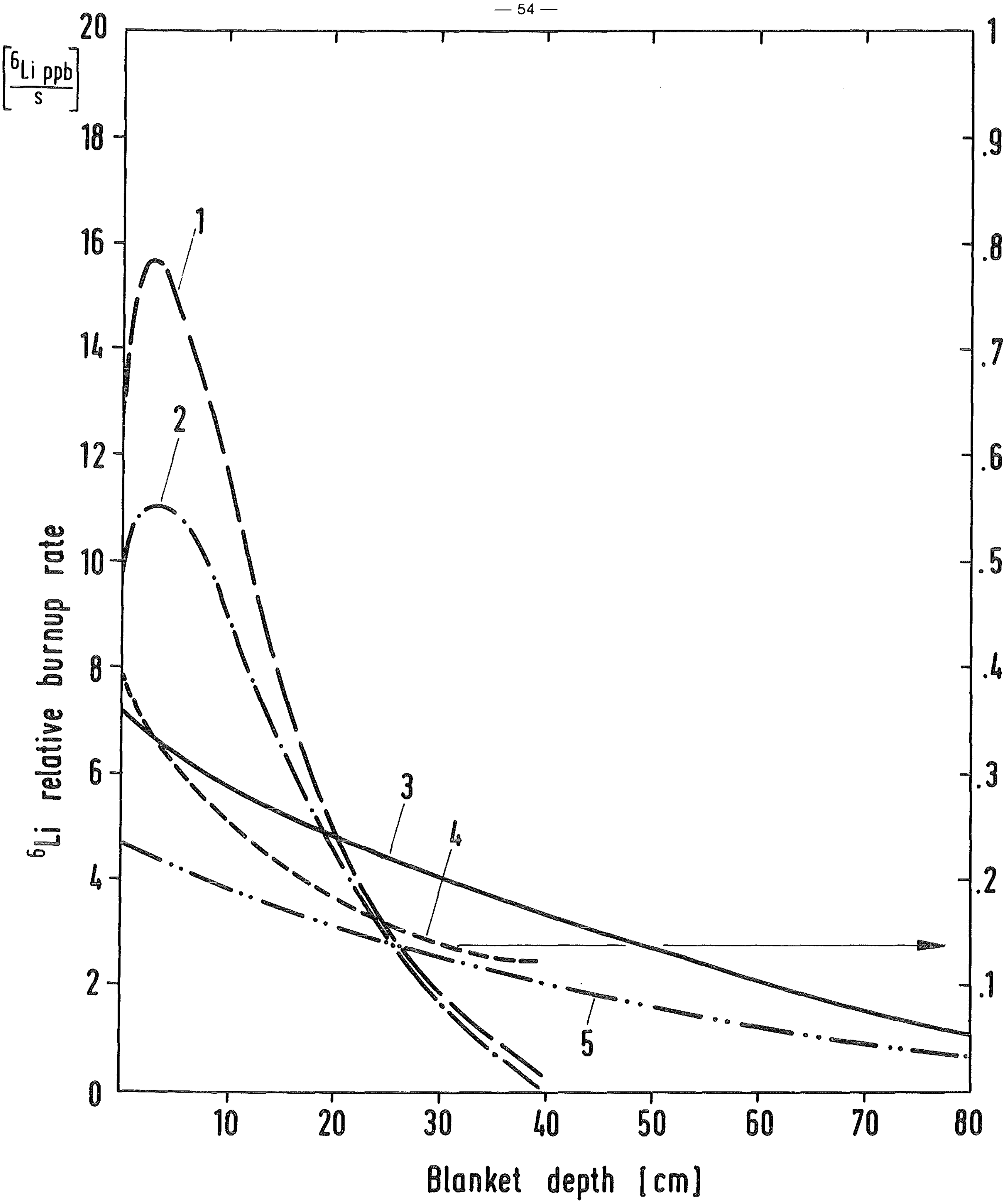


Fig. 3.9 Tritium breeding distributions (expressed as the relative burnup rate)

1) 58 % LiPb, 30 % $ZrH_{1.7}$ plane geom. and source

2) 58 % LiPb, 30 % $ZrH_{1.7}$ sph. geom., point source

3) 88 % $Li_{17}Pb_{83}$ plane geom. and source

4) 60 % Li_2O plane geom. and source

5) 88 % $Li_{17}Pb_{83}$ sph. geom. point source

In all cases Li_{nat} SS reflector, cases 1), 2) and 4) breeding zone thickness 0.4 m, 3) and 5) - .8 m

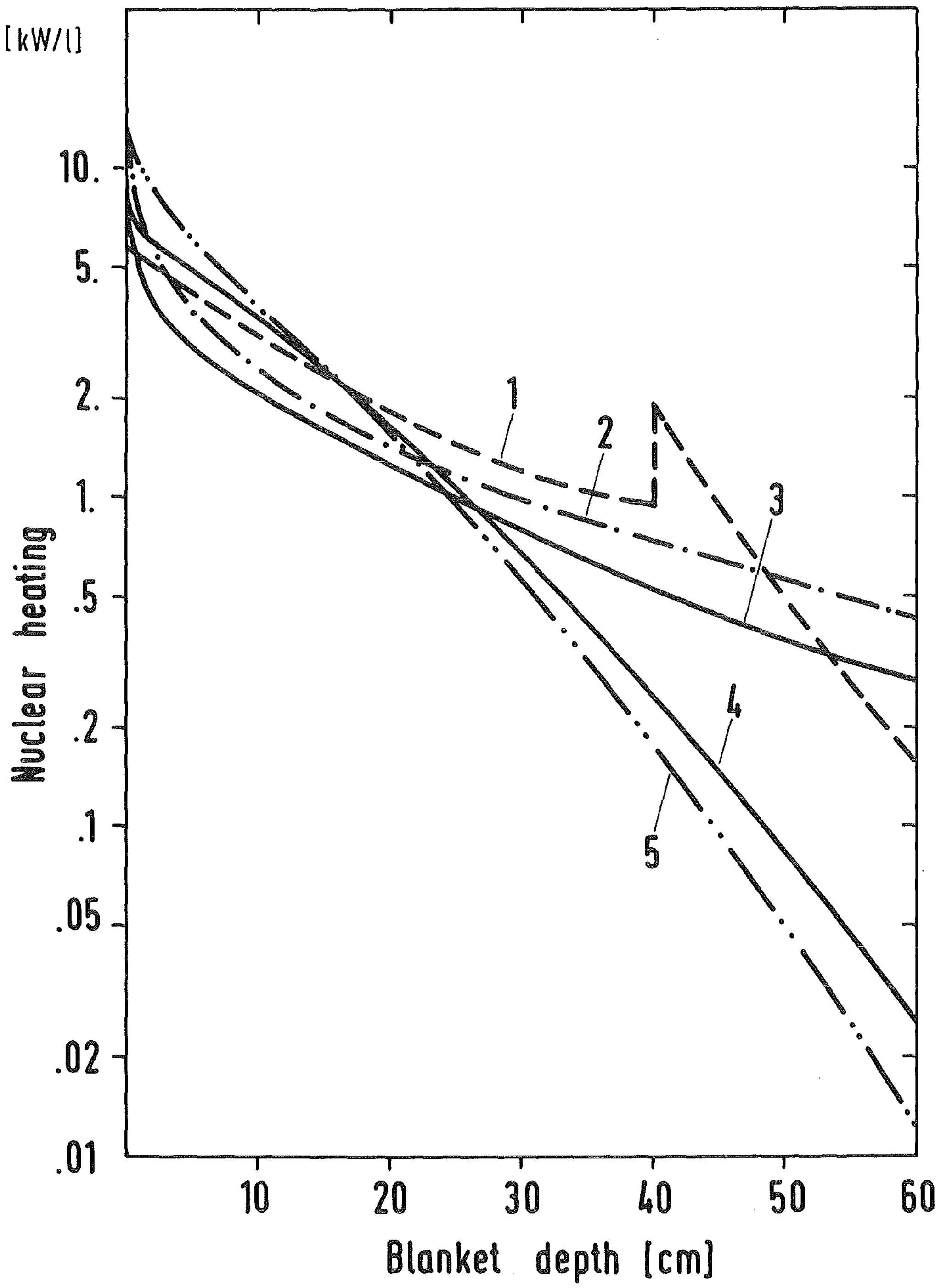


Fig. 3.10a Power density distributions for 1 MW/m^2 of FW of 14 MeV neutrons (direct FW heating from plasma not considered)

a) one breeding zone

1. 70 % Li_2O sph. geom. point source, 10 % SS, 30 % void
2. 88 % $Li_{17}Pb_{83}$ plane geom. and source
3. 88 % $Li_{17}Pb_{83}$ sph. geom. point source
4. 58 % $Li_{17}Pb_{83}$ 30 % $ZrH_{1.7}$ sph. geom. point source
5. 58 % $Li_{17}Pb_{83}$ plane geom. and source

In cases 2 - 5, 4 % SS, 8 % void

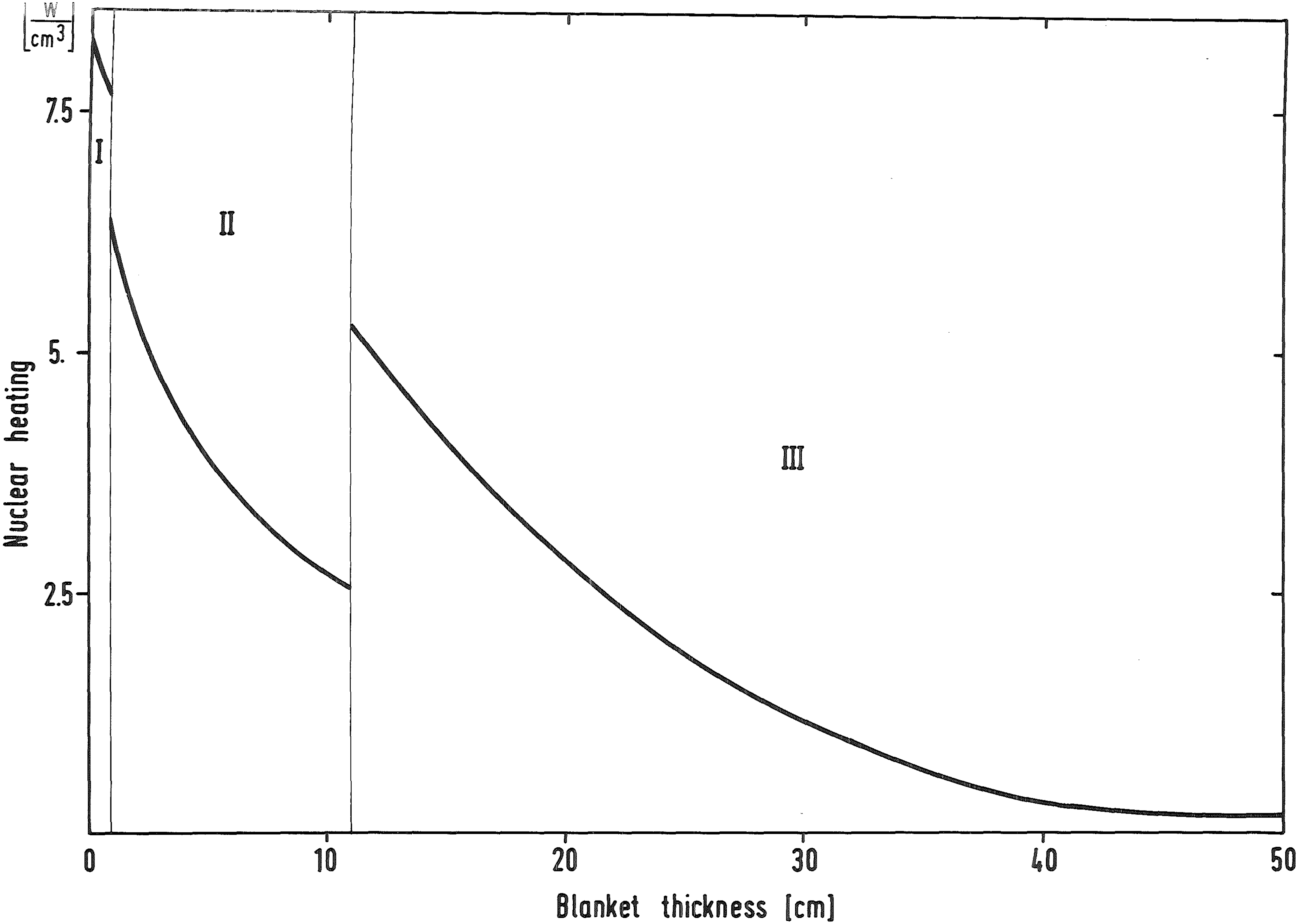


Fig. 3.10b two breeding zones, 4 % SS vol., 8 % void vol. sph. geom. point source
I-FW, II - 88 % $Li_{17}Pb_{83}$, III 58 % $Li_{17}Pb_{83}$, 30 % $ZrH_{1.7}$

Ad 2) The tritium breeding distribution is significant for its recovery and for the burnup of ^{6}Li, that can determine the necessary cycling of tritium breeding medium. The last effect is important in the case of higher breeding densities and simultaneous low ^{6}Li densities e.g. low volume fractions of Li_{nat} based breeding media. Some examples of the tritium breeding distribution is shown in fig. 3.9.

Ad 3) The knowledge of the nuclear heating distribution is indispensable for the design of the cooling system and for the proper tritium recovery. If the first is obvious, the second can be explained by great sensitivity of tritium effective release and diffusion through porous ceramic materials, for the temperature. This effect results from the fact that at (too) low temperatures the release and duffusion of gas is simply (too) slow while (too) high temperature causes sintering - equivalent to closing the medium pores and thus blocking the tritium diffusion. In consequence, ceramic substances (Li_2O, Li_2CO_3 etc.) have only "windows" of admissible temperatures - sometimes very narrow ones. This, in turn, implies particular requirements with respect to the cooling system, even if the fusion systems are characterized by relatively low power densities. The results of calculations are presented in fig. 3.10.

The observation of the figs. 3.7-3.10 leads to the conclusion that both the source geometry and the breeding zone composition strongly influence the spatial distribution of neutron induced phenomena in the blanket, first of all in the vicinity of the first wall (which is the most sensitive point of the whole blanket). At the same energy flux through the first wall unit area (from the plasma) one obtains twice as much power density in the first wall for plane geometry (distributed source) as in the case of an point source (e.g. spherical geometry). The last case and also light blankets permit us to achieve less peaked power distribution, that is more advantageous from the point of view of cooling and tritium recovery. On the other hand, however, the lighter breeding blankets as being much more transparent to neutrons, result in much higher neutron leakage into reflector and magnet system thus being distinguished by higher neutron losses and requiring additional shielding.

3.4 Guidelines of fusion reactor blanket nuclear design

In view of all the above considerations the guidelines for the breeding blanket design can be summarized as follows:

- The source neutrons must face the multiplying layer (of proper thickness) of possibly low concentration of nuclides attenuating the neutron multiplication (i.e. structure materials, non-gaseous coolants).

- For the most effective trapping of neutrons within the breeding zone (leakage and void streaming reduction) it must contain an efficient moderator.

- All regions of significant slow flux should contain ^{6}Li in order to reduce parasite neutron captures in there.

Such moderated and "sandwiched" tritium breeding blanket is of the following advantages:

- The tritium inventory in the blanket breeding zone can be reduced even by one order of magnitude.

- The necessity of lithium enrichment can be avoided.
- The utilization of the inner blanket of toroidal devices for the tritium breeding becomes worthwhile.
- The overall blanket dimensions can be reduced.

4. Neutronics of fissile breeding

Specific problems of the externally driven fissile breeding blankets deserve a separate consideration.

4.1 Fissile breeding efficiency

The fissile production as compared to sole fusile breeding is a more complexed problem. The most important differences result from the resonance self-shielding effects and the strongly exoergic reactions in the fertile and fissile media. As it was indicated in the Chapter 1 the economy of ANES requires that the fissile breeding devices be characterized by possibly high support ratio i.e. high fissile breeding rate per power unit.

The efficiency of a fissile breeding assembly expressed as the ratio B of bred fissile nuclei mass-to-energy released in the system /kg/GW_{th} yr/ can be presented as a function of well known reactor parameter - conversion ratio c_r of the system according to the expression:

$$B = \frac{c_r m_b - m_d}{Q_{f_{fi}} \frac{\langle\sigma_f\rangle}{\langle\sigma_f+\sigma_c\rangle} + Q_{c_{fi}} \frac{\langle\sigma_c\rangle}{\langle\sigma_f+\sigma_c\rangle} + c_r (Q_{c_{fe}} + Q_{nf})} \qquad (4.1)$$

where m_b - mass of bred fissile nuclide

m_d - mass of destroyed fissile nuclide

Q_f - neutron binding energy

Q_{nf} - remaining non-fissile origin energy (e.g. fusion, fast fission or proton beam energies) released in the system per one bred fissile nucleus

$<\sigma_f>$ - fissile material spectrum averaged fission cross-section

$<\sigma_c>$ - fissile material spectrum averaged neutron capture cross-section

fi - fissile medium index

fe - fertile medium index

When substituting the quantity

$$Q_{fi} = Q_{f_{fi}} \frac{<\sigma_f>}{<\sigma_f+\sigma_c>} + Q_{c_{fi}} \frac{<\sigma_c>}{<\sigma_f+\sigma_c>} \tag{4.2}$$

into the denominator of the formula (4.1) one obtains

$$B = \frac{c_r m_b - m_d}{Q_{fi} + c_r (Q_{c_{fe}} + Q_{nf})} \tag{4.3}$$

It is remarkable that with an analog formula one can express the LWR net burning efficiency B_L (as the negative breeding):

$$B_L = \frac{m_d - c_{r_L} m_b}{Q_{fi} + c_{r_L} (Q_{c_{fe}} + Q_{nf_L})} \tag{4.4}$$

where index L designates the quantities regarding LWR.

Since the non-fissile energy release in LWRs is negligible, one can simplify the formula (4.4) to the form:

$$B_L = \frac{m_d - c_{r_L} m_b}{Q_{fi}} \tag{4.5}$$

One of the objectives of the present considerations is to find an expression describing the number of unit LWRs of given burning efficiency, that can be supported by the fissile breeding system of the same unit power. The support ratio S so defined can be expressed as follows:

$$S = \frac{B}{B_L} \tag{4.6}$$

or having assumed the equal mass of bred and destroyed nuclei

$$S = \frac{c_r - 1}{(1-c_{r_L})\,(1+c_r\,\frac{Q_{C_{fe}}+Q_{nf}}{Q_{fi}})} \tag{4.7}$$

The above expression is illustrated in fig. 4.1a and for breeders of dominant fissile origin energy production (e.g. fast breeders, fission enhanced blankets) it can be further simplified (fig. 4.12b):

$$S = \frac{c_r - 1}{1 - c_{r_L}} \tag{4.8}$$

Thus, based upon the total energy associated with the breeding of fissile nuclei one can determine the minimum conversion ratio necessary for attaining the given fissile production per system power · time. Or, the other way round, one obtains in this way the breeding upper limit of the system or its maximum achievable support ratio for given conversion ratio.

The support ratio must be sufficiently high if the fissile breeder has ever to be economic /1/. On the other hand the realistic values of non-fissile origin energy-to-fission energy ratio Q_{nf}/Q_f can hardly exceed 0.15 and the conversion ratios higher than 15 - 20 seem also hardly achievable because of difficulties in still better fission suppression. Rough estimations and available data /35 - 47/ indicate that in both hybrid and spallation systems the Q_{nf} lesser than a 30 MeV looks inattainable, what corresponds to Q_{nf}/Q_f ~.15).

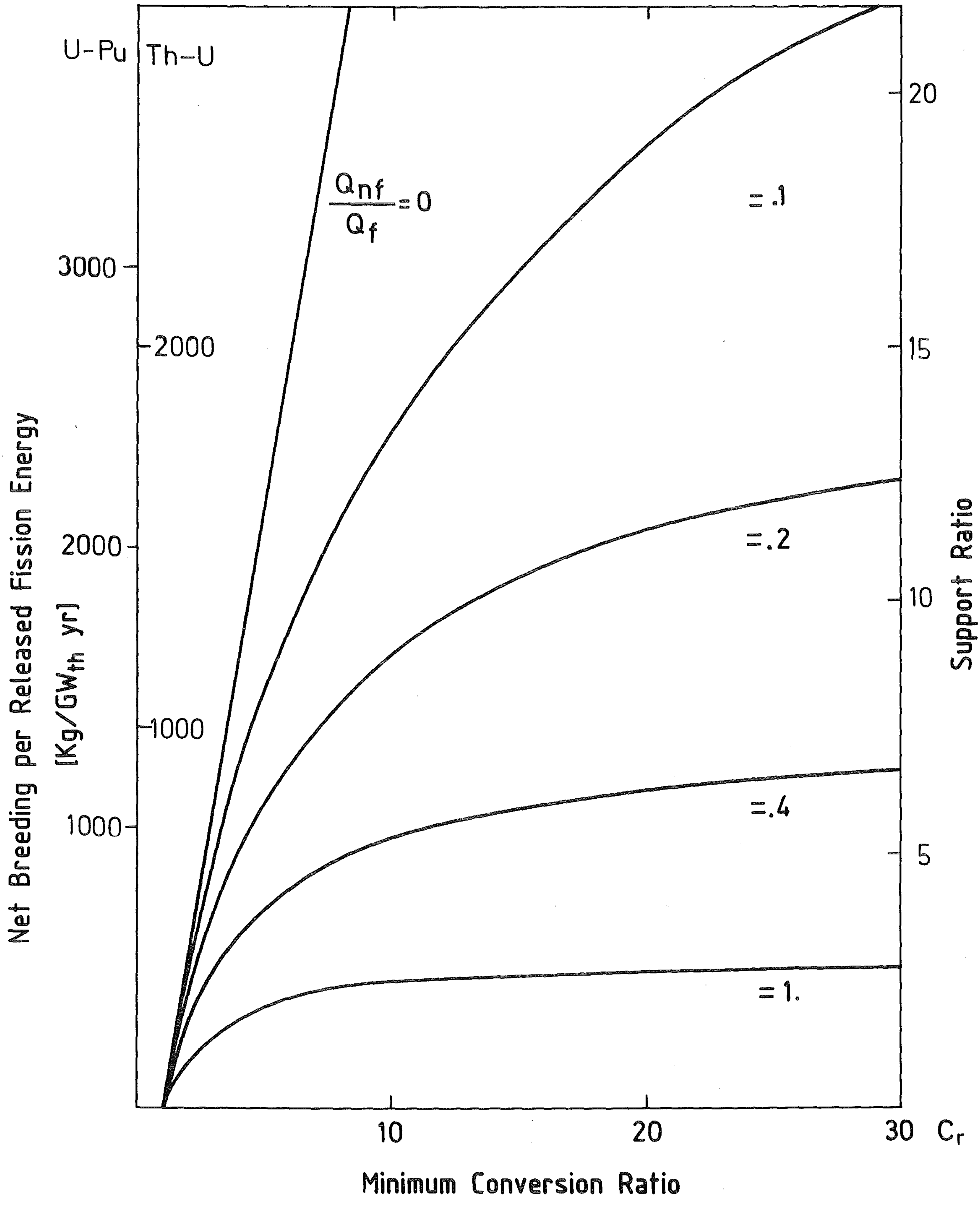


Fig. 4.1a Breeding efficiency limitations
Fission suppressed systems

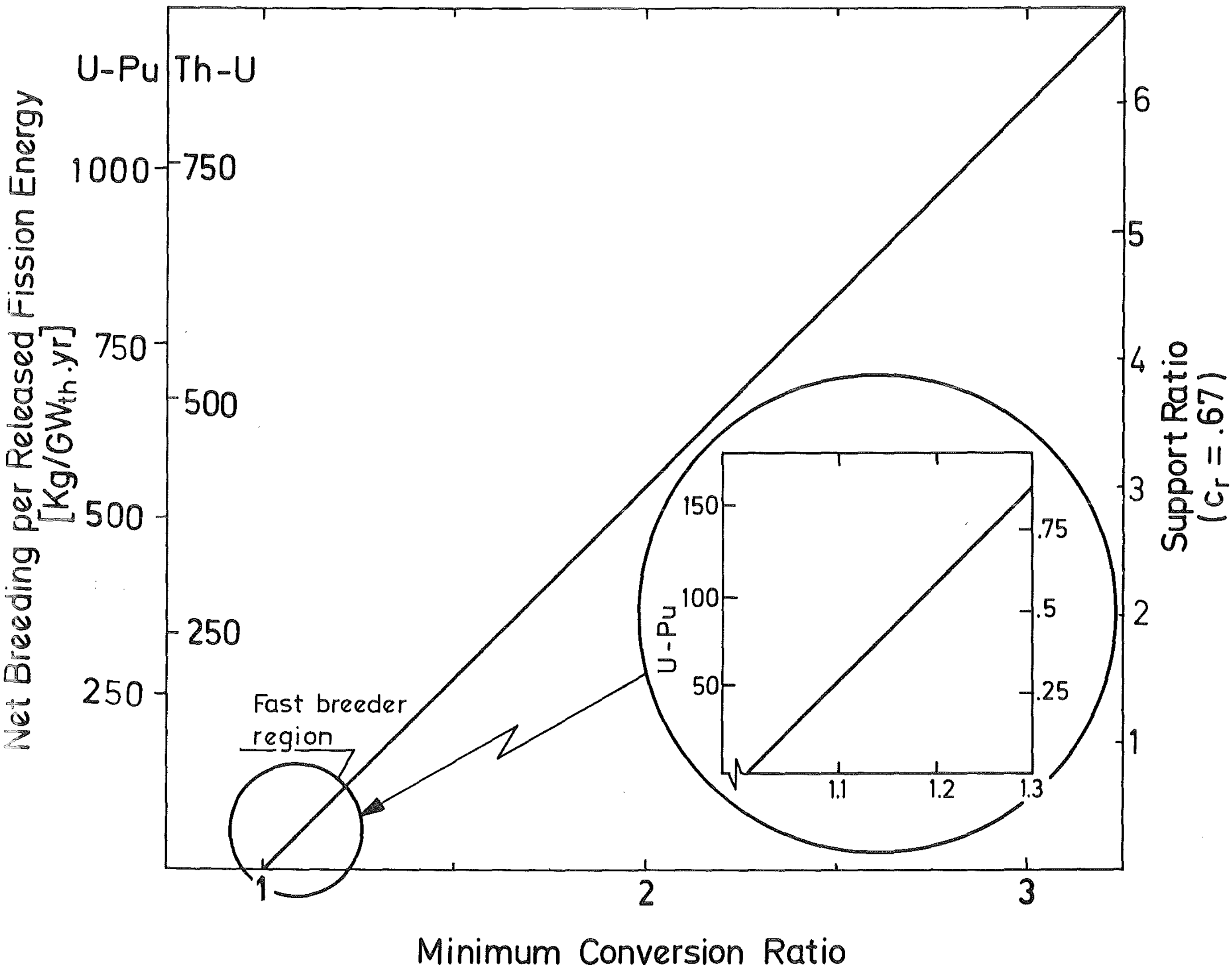


Fig. 4.1b Breeding efficiency limitations
Fission enhanced systems

The other limitation results from the necessity of maintaining certain minimum enrichment, (usually ca. 1 %) and is caused by the requirements of fissile material recovery at reasonable costs. The fissile destruction rate at this enrichment level that cannot be reduced below several per cent corresponds to the above mentioned c_r values.

All the above is strictly limiting the maximum support ratio (of LWRs with c_{r_L} = 0.67) to the values slightly exceeding 10 only in the most favourable conditions of very high conversion ratios above 10, with simultaneous lowest possible non-fissile origin energy share (fig. 4.1a). In the case of fission enhanced systems the breeding efficiency is obviously still much lower (fig. 4.1b).

4.2 Fission suppression

In view of the techno-economical indications the fission suppression proves one of the factors conditioning the effective fissile breeding. Since the fission suppression is just the opposite to the objectives of fission reactor design, the means to be undertaken should be also opposite. It signifies thus, among others, that instead of the increase in enrichment and neutron slowing-down, possibly lowest enrichment and no moderator are advisable. The problem becomes particularly uneasy when one is conscious that the apparently most efficient resonance absorptions in fertile materials cannot be fully utilized due to unavoidable self-shielding effects. In consequence, the slow fission suppression proves more complicated task. The realization of this aim can be achieved through /35/:

1) adequate flux shaping (i.e. first of all slow flux suppression)
2) lowering of the fissile (and when possible also of the fertile) concentration.

As concerns the first item the important phenomena occurring in the resonance region in fertile materials, which is designed to be a neutron trap preventing neutrons from reaching the low epithermal and thermal energies, deserve a detailed discussion.

In turn, the reduced fissile concentration can be realized by:
a) direct lowering of heavy metal concentration
b) unload of the product at low enrichment
c) fuel shuffling that the average enrichment in the reactor is the mean of the load and the unload enrichments /16/
d) rapid fuel cycling (the product exists in the blanket partly in form of non-fissile intermediate nuclide in the fuel cycle, see 4.2.1)

Ad c) The need of the fuel shuffling is due to the flux gradient across the breeding zone making the enrichment to increase faster at the inner blanket side than at the outer one. In case of quasi continuous refuelling the spatially averaged enrichment is the average between the reload and unload enrichment levels. As being constant in time, it assures simultaneously the total power constancy for a constant fusion yield. Thus, e.g. for the final enrichment level of 3 %, the mean enrichment would be 1.5 % and 2.5 % for the fresh fuel loading (initial enrichment of 0 %) and for the fuel rejuvenation (initial enrichment of 2 %) respectively.

4.2.1 Irradiation rate significance

The other way to fission suppression lies in the opportunity offered by the non-fissile intermediate nuclides in the fissile production cycle i.e. first of all ^{233}Pu, $T_{1/2}$ = 27.4 d and perhaps even ^{239}Np, $T_{1/2}$ = 2.37 d /16/. It is interesting to analyze the possible gains resulting from the delayed build up of the fissile component.

The question is what irradiation conditions must be assured (and if they are realistic) in order to have sensible profits due to the above effects. In this purpose one should express the actual-to-final enrichment increase ratio $\frac{\Delta}{\Delta_\infty}$ as a function of system neutron yield S per number of fertile nuclei N_f in the system for given breeding rate b (fissile nuclei/system neutron):

$$\frac{\Delta}{\Delta_\infty} = 1 - \frac{1}{\lambda T}\left[1 - \exp(-\lambda T)\right] \qquad (4.9)$$

where

$$T = \frac{\Delta_\infty N_f}{Sb} \qquad (4.10)$$

In order to facilitate the reference of the abscissa (in eq. 4.9) to more practical quantities in the case of hybrid reactor, it was also presented as source neutron flux per unit area of the first wall (b is then related to the source neutrons) with the fertile mass per the same unit area as a parameter. Quantitatively, the real advantages are determined also by the other dependence i.e. by the breeding efficiency (fissile nuclei/total energy released) as a function of the actual enrichment. If one were able to breed the fissile at constant rate per power unit independently of the enrichment (ideal slow fissions suppression), its delayed build up could not bring then any gain. And, in contrast, the higher is the energy release associated with the increase in enrichment, the greater improvements can be expected from the enrichment reduction.

Some representative examples of these relationships are given in figs. 4.2a and b. On the basis of these diagrams one can conclude that considerable gains can be obtained in the case of thorium cycle when enriching from zero level since then a significant increase in the energy production should be expected. The other cases and esp. the uranium cycle ones are less encouraging. All this agrees with the intuition, seeing that the fuel to be regenerated already contains quite large amounts of fissile material, so the existing unfavourable conditions can neither be worsened much by the increase in enrichment nor much improved by the time effects in question. As concerns the uranium cycle, the decay time of ^{239}Np is about one order of magnitude shorter than the one of ^{233}Pa that results in proportionally more severe requirements regarding the enrichment rate (and thus the neutron flux), that seems hardly achievable. It should be also noticed that the more fertile material there is in the blanket, the higher source neutron fluxes are required for given enrichment rate. This gives additional argument for the limitation of fertile inventory in ANESs.

4.2.2 Resonance self-shielding effects

The attenuation of the neutron absorbing power of a medium having sharp cross-section maxima which is caused by their self-shielding, is a well-known phenomenon since the birth of reactor physics in the early forties /48/. Whereas this effect is negligible in pure fusion neutronics practically dealing with materials without significant resonances, it proves be of particular importance in heavy metals. Therefore, in any fissile breeding system, i.e. not only in e.g. fast breeders but also in hybrids and spallators, the resonance self-shielding (RSS) must be taken into consideration independently of the system structure. It signifies, among others, that the system need not be heterogeneous that the RSS be significant. Its consideration is necessary, first of all since the resonance captures considerably contribute to the fissile breeding and fission suppression. It is so since the resonance region

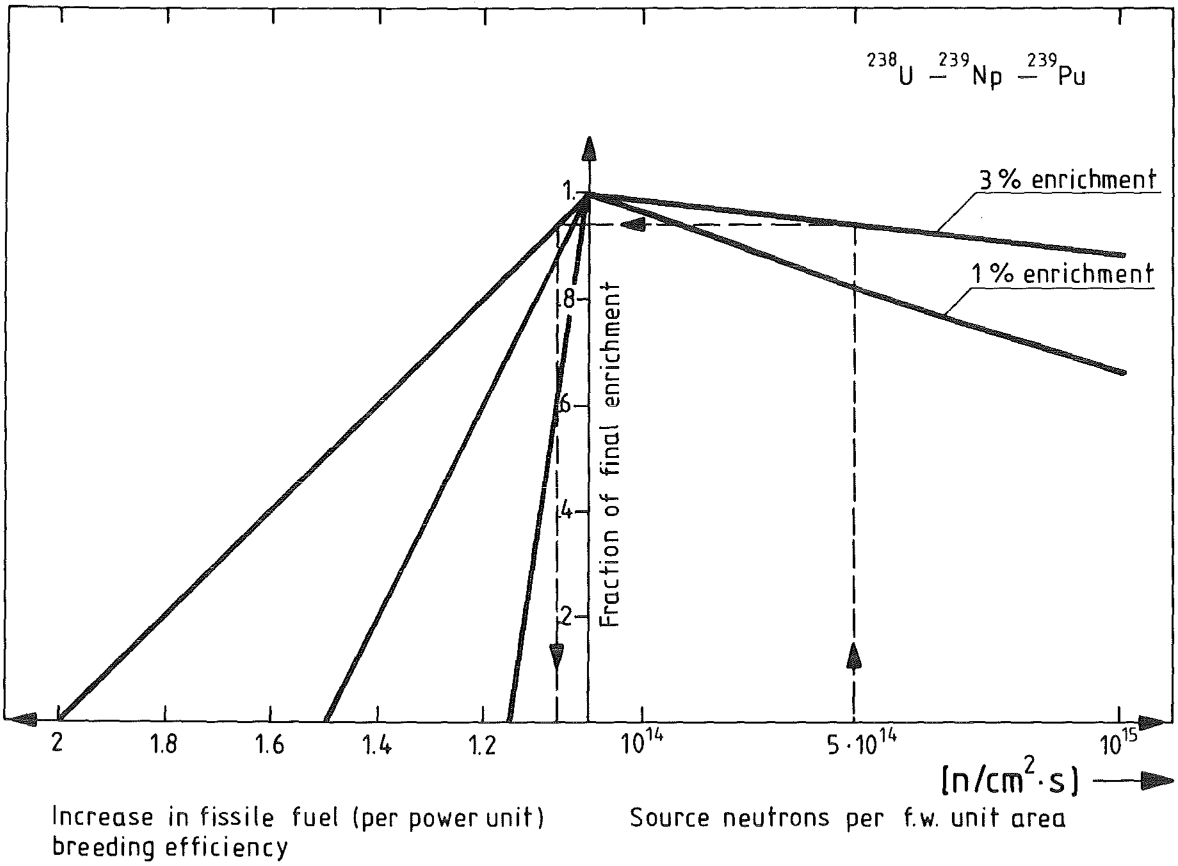


Fig. 4.2b Advantages of rapid enrichment due to delayed fissile build up (200 kg U 238 per m^2 of the first wall, .6 fissile nuclei/fusion neutron)

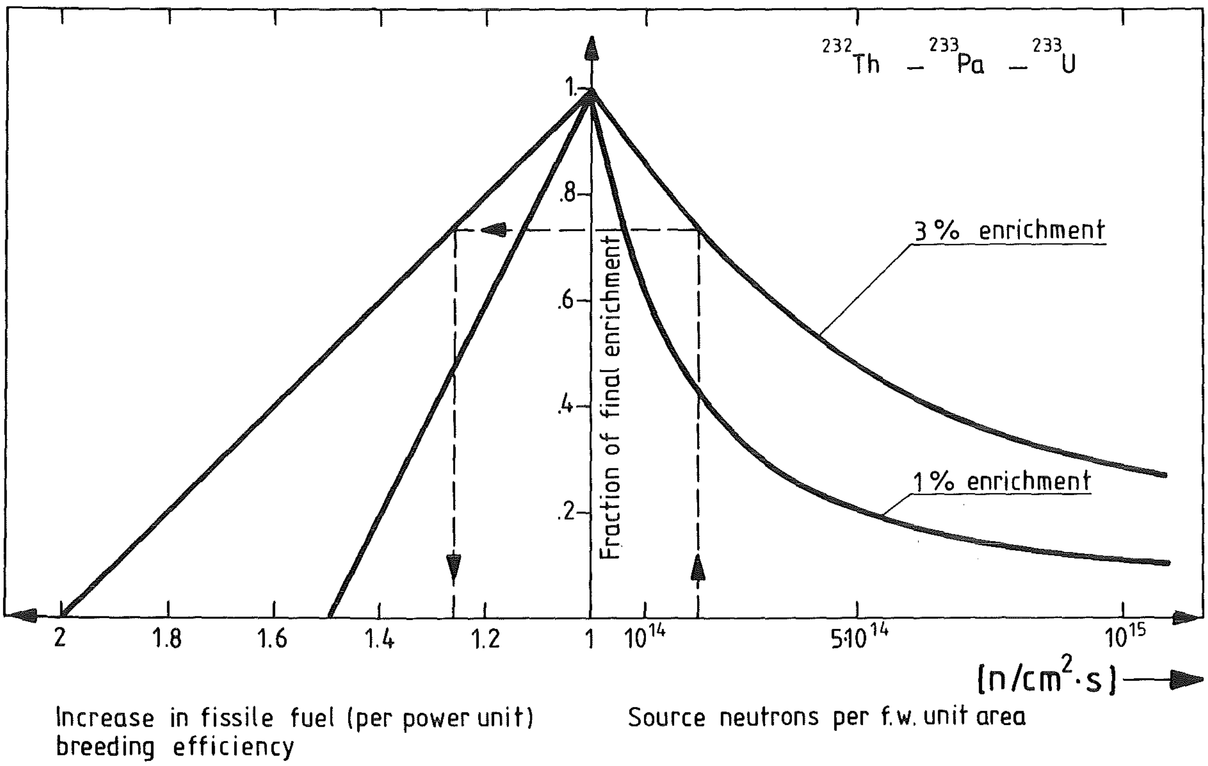


Fig. 4.2a Advantages of rapid enrichment due to delayed fissile build up (200 kg Th 232 per m^2 of the first wall, .6 fissile nuclei/fusion neutron)

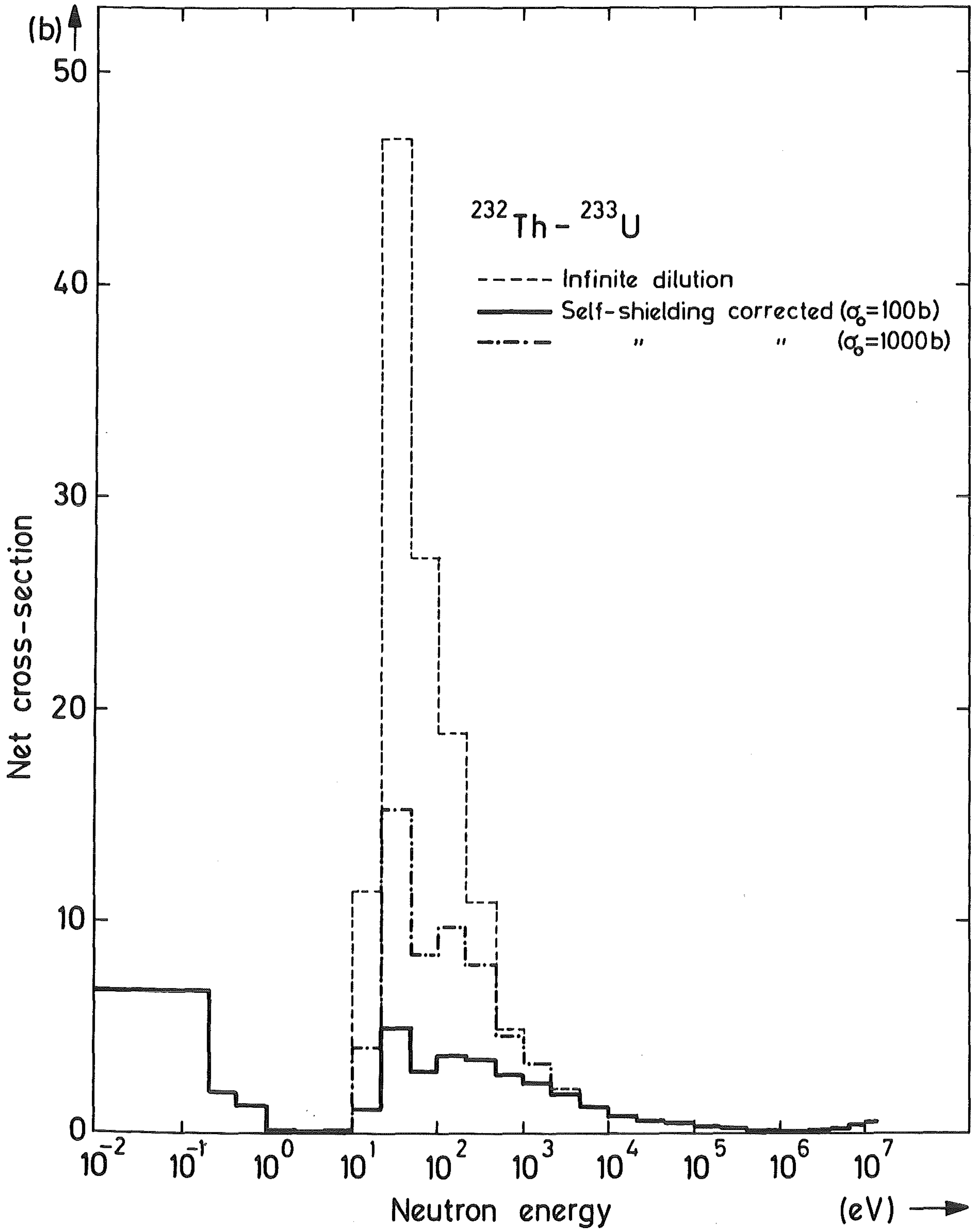


Fig. 4.3a Resonance self-shielding influence on the net fissile breeding cross-section. RSS corrections correspond e.g. to homogeneous mixtures with fertile-to-Be atomic ratios 1/15 and 1.150 respectively (data based on ref. /6,7,8/. σ_o signifies the total cross-section of the medium per one fertile nucleus of resonance cross-section /8/).
a) Th cycle

of high capture cross-sections of fertile media should be the final trap preventing neutrons from reaching the low epithermal and thermal regions in the slowing-down process. Neutrons of these energies are highly undesirable since, due to cross-sections relations, the fissile material destruction associated with the intensive energy release (fissions!) dominates over the fissile breeding already at relatively low enrichment /16/. This effect, contrary to the fundamental objectives and requirements of effective fissile breeding, threatens to transform a spallation breeder in a supercritical burner and to make impossible the effective fission suppression in case of fusion-fission hybrids.

As it results from the considerations in chapters 1 and 4.1 the advanced fissile breeding systems like hybrids and spallators should be optimized in view of the enhancement of their fissile production /1/. The neutron generation per released energy should be maximized. This aim is mainly conditioned by the minimization of slow fissions that not only deteriorate the neutron-to-energy ratio of the system but also destroy the fuel just having been bred. In general, the slow fissions can be effectively avoided only in the absence of slow flux since the low fissile concentration /35/ not always can be realized (e.g. in spallator targets, where high heavy metal concentrations are necessary). The slow neutron flux suppressing by means of 1/v type ^{6}Li(n,α)T reaction /35/ (applicable only in hybrids or at most in spallation - fusion symbionts where tritium would be partly produced in spallator) also proves not always efficient. Namely - in moderated systems, where neutrons are rapidly slowed in relatively small number of collisions down to the energy region below 7 - 20 eV of fertile media resonances. The probability of neutron capture in these materials in then significantly reduced, but the problem remains unnoticed unless the resonance captures are not able to prevent some neutrons from being transferred down to low energies. Apparently, the fertile material capture cross-sections in the resonance region suffice for neutrons to be captured before being slowed-down also in the presence of moderators (e.g. H_2O, D_2O, Be, C). Unfortunately, the properly evaluated effective capture cross-sections, corresponding to real concentrations of fertile nuclei does not justify such optimism. The standard neutron data related to infinite dilution of nuclides in question, approximate adequately the reality only when the weighting neutron spectrum used for multigroup constants calculation does not differ significantly from the real spectrum in the system. As a rule, except of thermal and high energy regions, the constant in lethargy spectrum is assumed while generating neutron group data libraries, that results in considerable overestimation of neutron captures in cases of deep flux depressions at resonance energies. And this takes place already for low fertile concentration because of extremely peaked capture cross-section of these nuclides, exceeding even 20000 b. As a result the capture power of fertile media proves incomparably lower than it might be expected.

In order to illustrate the scale of the RSS influence on the cross-section a net effective fissile breeding cross-section σ_{nb} of a fertile-fissile mixture should be defined first. This task requires slightly more attention since the co-existing processes of fissile production, destruction, of neutron multiplication and losses make the measure of net fissile breeding to be influenced by a number of factors. The point is how to consider properly the multiplication processes (i.e. first of all the fission) which on one hand, when occurring in fissile media results in its destruction but on the other hand supplies additional neutrons that can be captured in the fertile nuclei thus reproducing the material just being fissioned. It signifies, obviously, that the neutron balance in the system (i.e. the neutron deaths distribution - fissile breeding, destruction and all the remaining losses) directly influences the effective multiplication. Naturally, when occuring in fissile media, they contribute also indirectly to the fissile breeding, while producing neutrons captures next in part by the fertile material. Therefore, the net breeding cross-section normalized to one fertile nucleus of a medium containing one fertile and one fissile material in stationary state (in general the quantities in 4.11 are functions of time, what however is unimportant from the point of view of present discussion) is expressed as follows:

$$\sigma_{nb} = \sigma_{cfe} + (\sigma_{ffe}\,\nu_{fe} + 2\sigma_{n,2nfe})\,\frac{P-D}{P+D+L} - \qquad (4.11)$$

$$-\frac{e}{1-\ell}\left[\sigma_{cfe} + \sigma_{ffi}\,(1 - \nu_{fi}\,\frac{P-D}{P+D+L}) + \sigma_{n,2nfi}\,(1-2\,\frac{P-D}{P+D+L})\right]$$

where:

P - fissile production rate
D - fissile destruction rate
L - remaining neutron losses
e - enrichment
fe - fertile medium index
fi - fissile medium index

As it results from the performed calculations /47/, the share of net fissile breeding $\frac{P-D}{P+D+L}$ in the neutron balance varies from ~ .3 for hybrids up to ~ .7 for hard spectrum spallators of low enrichment. Here it was assumed equal to .4 that for $\nu_{fi} = \nu_{fe} = 2.5$ and $e \ll 1$ when neglecting $\sigma_{n,2nfi}$ permits us to simplify the expression (4.11) to the form:

$$\sigma_{nb} = (\sigma_c + \sigma_f + .8\ \sigma_{n,2n})^{fe} - e\ \sigma_{cfi} \qquad (4.12)$$

presented in figs. 4.3a and b.

It should be noticed that the net breeding cross-sections corresponding to other fissile breeding shares in the neutron balance will not have the shape much different from the ones in figs. 4.3a and b, since generally the dominant σ_{cfe} determines the effective breeding cross-section.

As it can be seen in figs. 4.3a and b the group capture cross-sections are drastically reduced by RSS effects in the region of well peaked resonances also for relatively diluted fertile nuclides. Whether it significantly affects the reaction rates i.e. first of all the net breeding and the energy release in the system, it depends on the number of neutrons that reach these epithermal energies, what is determined by the slowing down properties of the system.

The question arises, how to reduce the above losses of neutron capture power. In a relatively simple way, certain broadening of the energy interval suitable for fissile breeding might be achieved by admixing of UO_2 with thoria, thanks to the weakened self-shielding of mutually diluted ^{232}Th and ^{238}U. The effective neutron capture cross-sections for a mixture of .8 ThO_2 + .2 UO_2 (depleted) and for pure ThO_2 are shown in fig. 4.4.

While aiming at the fissile breeding at minimum power, it may be of interest to see the net breeding cross-section normalized to the fuel energy production cross-section weighted with the released energy: $Q_f \cdot \sigma_f(fe,fi) + Q_c \cdot \sigma_c(fe,fi)$ where Q_c and Q_f signify the energy release in the capture and fission processes respectively. It should be noticed here that the above cross-section does not represent the total neutron energy release in the system, but its component released in the heavy metals. Unfortunately, the general consideration of the energy fraction set free in the rest of the blanket is difficult, since it differs significantly for various systems. This fraction may be recognized negligible for fission enhanced systems (first of all for fast breeders, obviously) where the wast majority of neutrons should interact solely with heavy metals, while in fission suppressed blankets of different degree of slow flux suppression and of the respective energy fraction may vary quite significantly. Generally, one should expect additional neutron energy transfer in

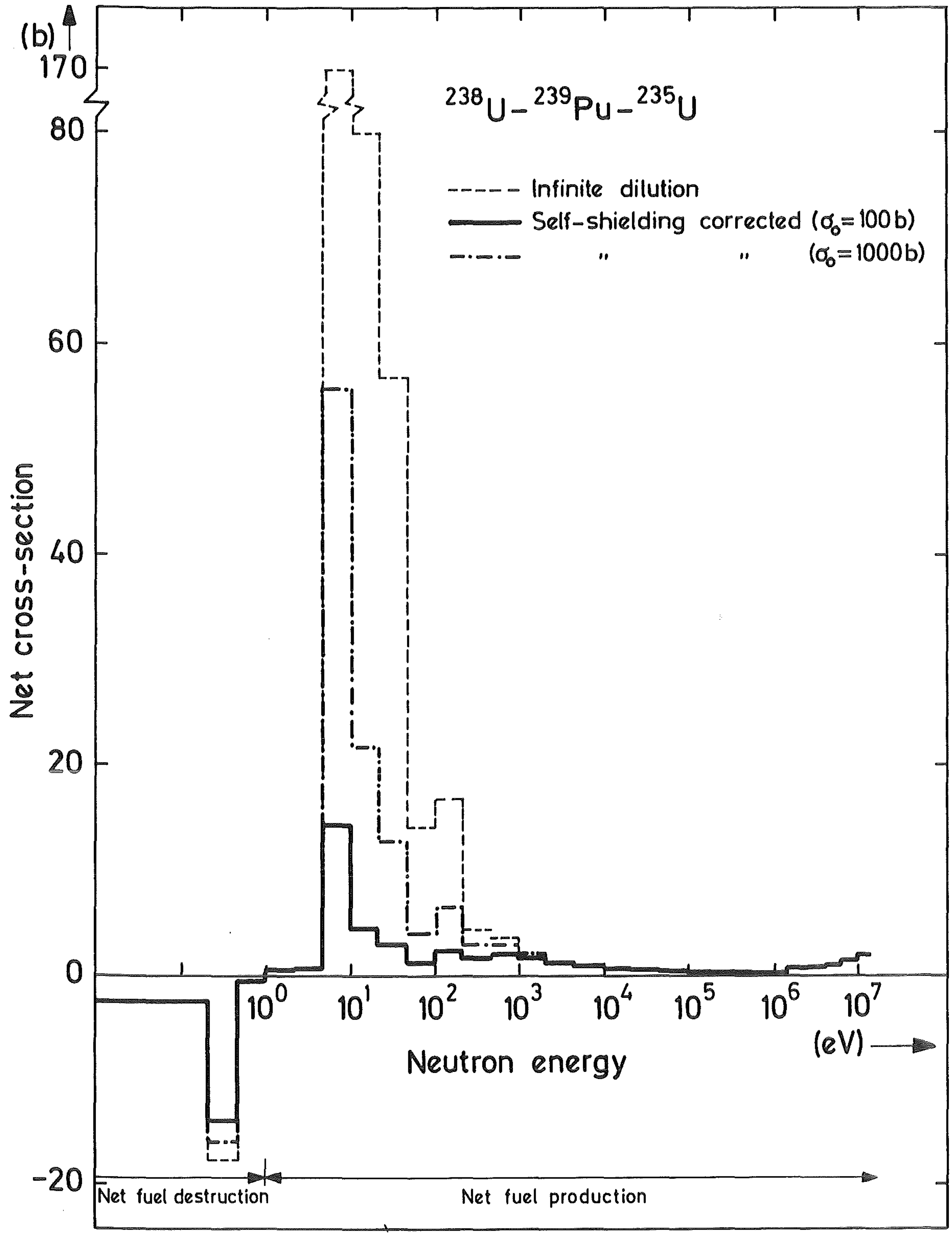


Fig. 4.3b as Fig. 4.3a but U-cycle

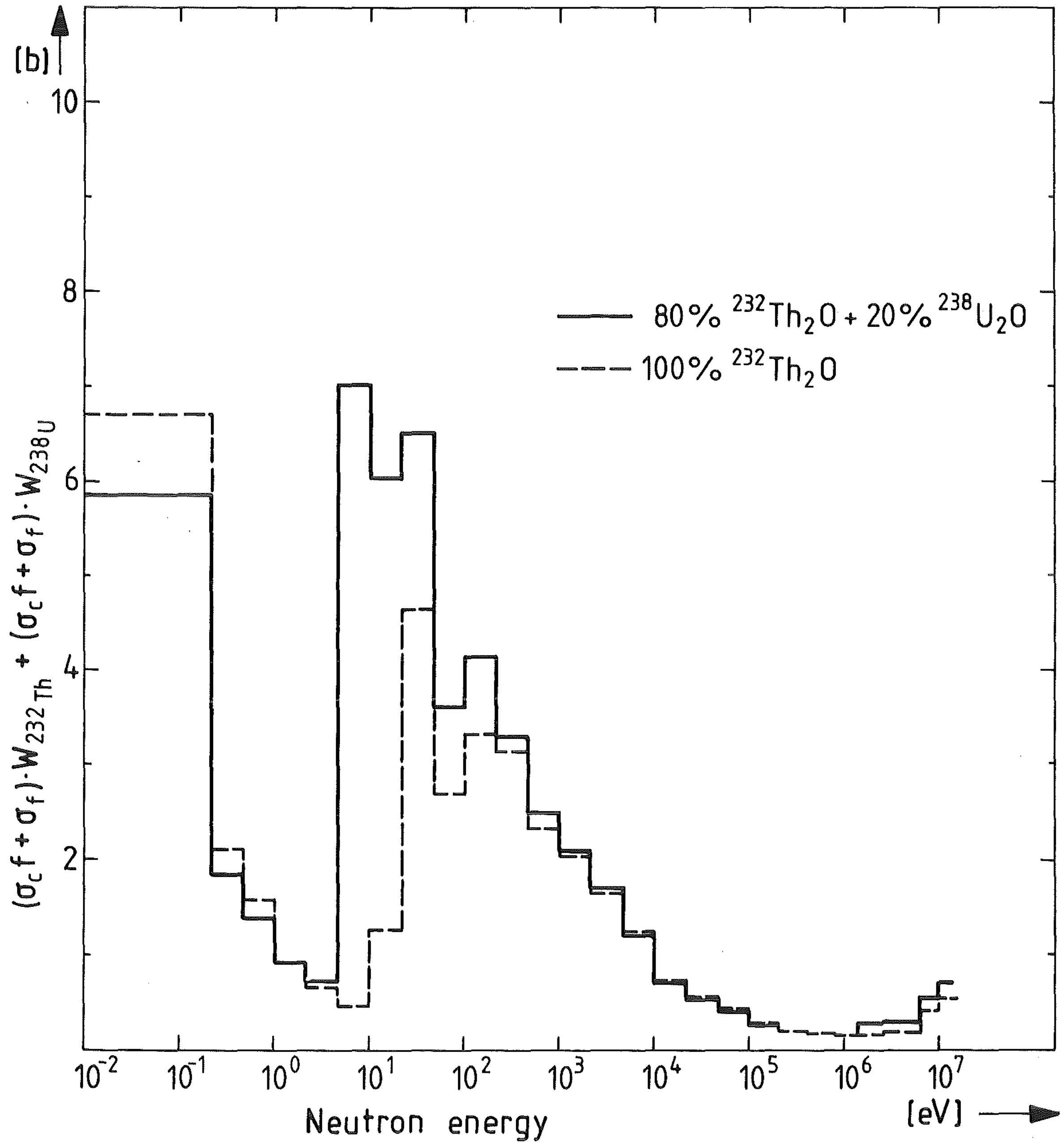


Fig. 4.4 Increase in the effective fissile breeding cross-section for a mixture of fertile materials (self-shielding correction factor f as for fig. 4.3)

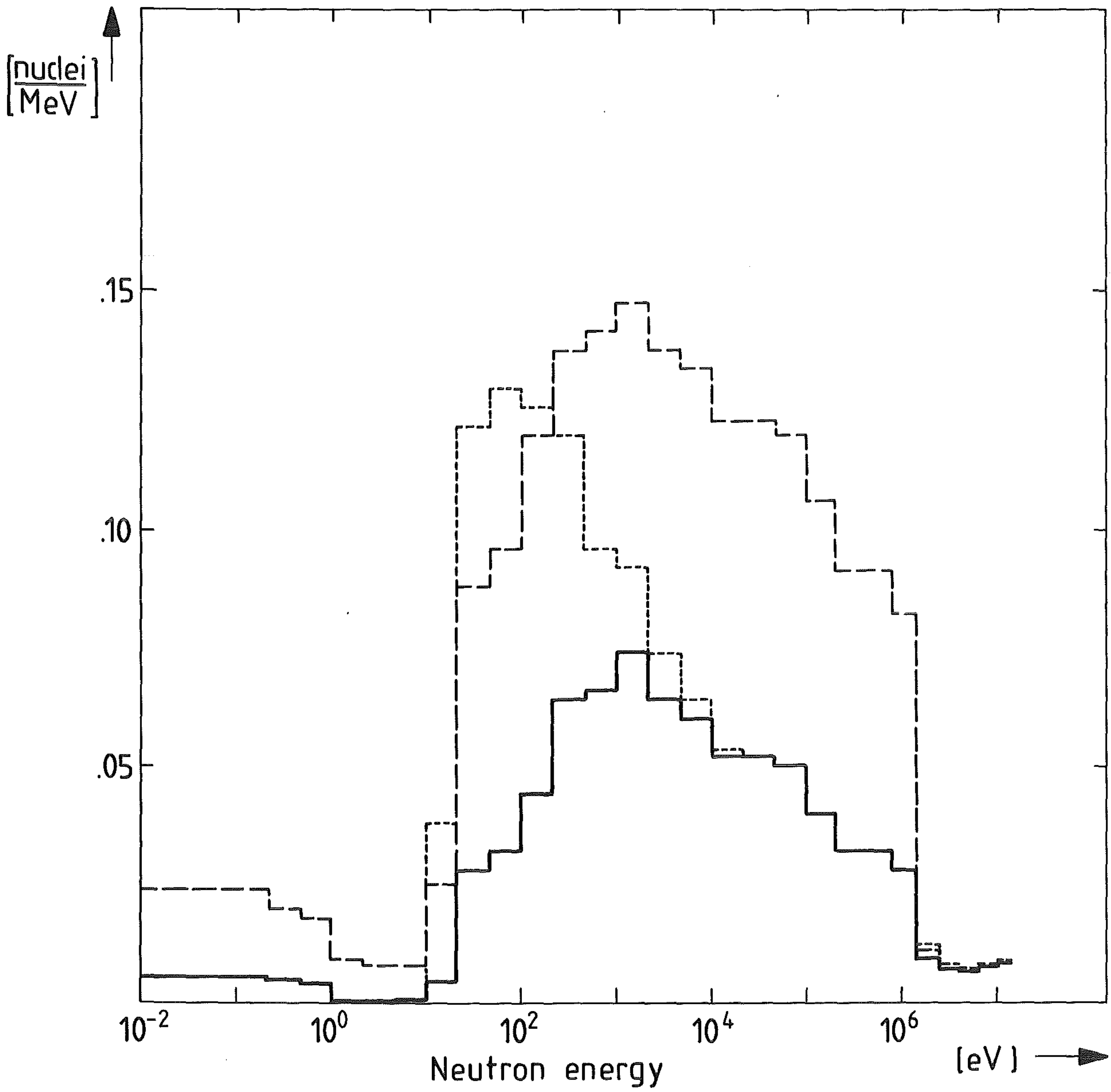


Fig. 4.5a Net fissile breeding per energy released in the fuel (self-shielding correction as for fig. 4.3)

^{232}Th - ^{233}U

– – – – – self shielding corrected 0.3 % enrichment

------------ self shielding neglected 1.5 % enrichment

__________ self shielding corrected 1.5 % enrichment

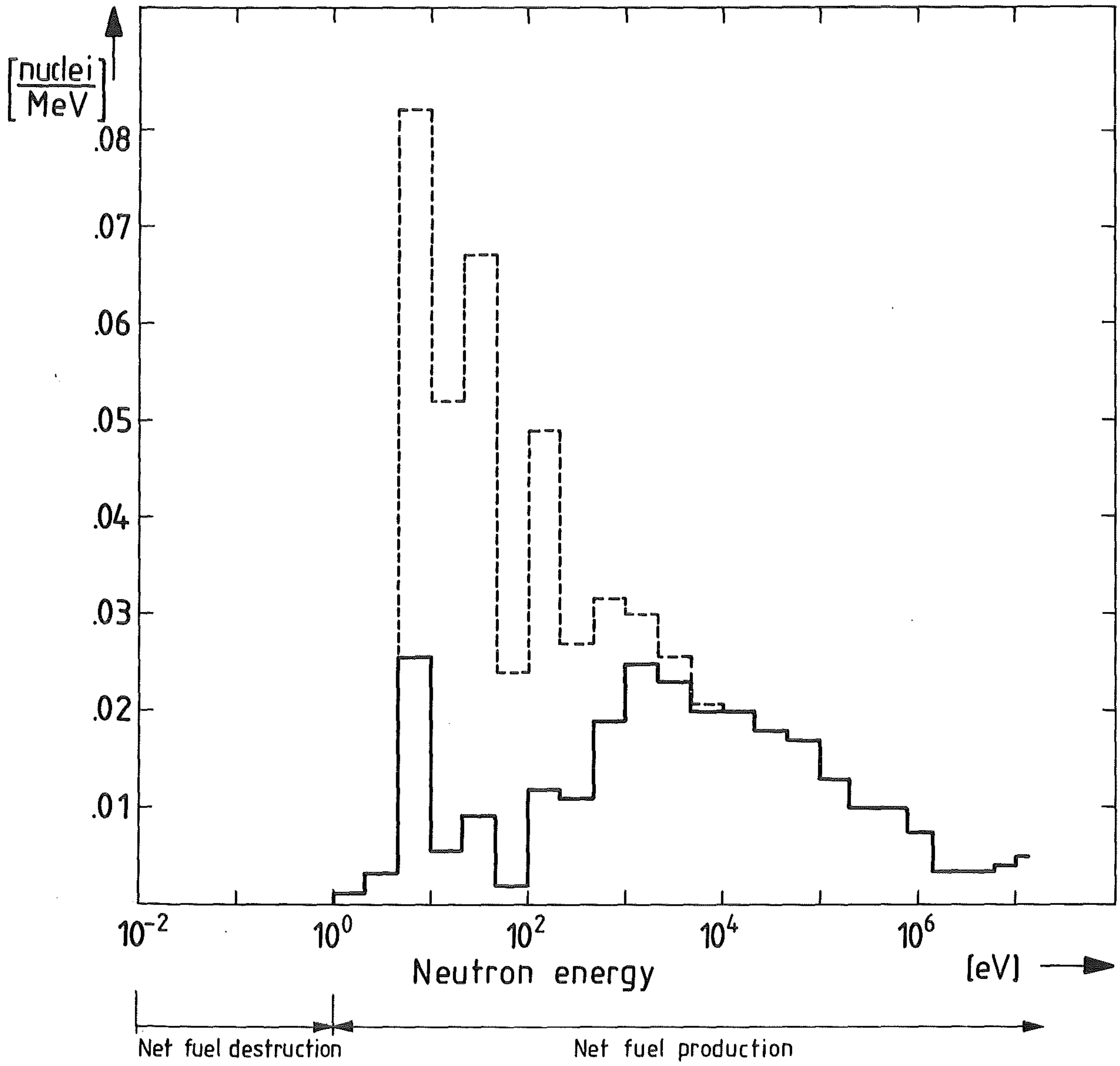


Fig. 4.5b Net fissile breeding per energy released in the fuel (self-shielding correction as for fig. 4.3)

^{238}U - 1,5 % ^{239}Pu - 1 % ^{235}U

-------- self shielding negligible

———— self shielding corrected (σ_e = 100 b) both high (inelastic interactions) and low (neutron capture in ^{6}Li) energy intervals, with the last effect slightly shifting the optimum towards higher energies. Nevertheless, it will not change the essence of the image in figs. 4.5a and b.

It can be seen in there that also in this case the picture obtained with the corrected self-shielding significantly differs from the one corresponding to the negligible self-shielding effects.

In addition to this, the observation of the diagrams permits us to conclude that the most disadvantageous energy interval is the lower epithermal region, then the thermal and the fast ones. It signifies that the removal of neutrons from high energies must be followed by possible moderate slowing-down, in order to avoid their premature transfer into the area of too low energies where the probability of neutron capture in the fertile material rapidly decreases. Since, obviously, the energy dispersion in the course of moderation can never be avoided, rather a harder spectrum than a too soft one is advisable, in order to prevent neutrons from slowing-down to the energies where even the net fuel destruction may occur. Quantitatively the consequences of all these effects will be illustrated with the results of neutron transport calculations in connection with selected hybrid and spallator breeding concepts (see 4.3 and 4.4).

In order to illustrate the all above discussion, a series of neutron transport calculations for hybrid and spallation breeders was carried out with the use of ONETRAN code /29/ together with the GRUCAL module /49/ evaluating the self-shielded nuclear data /50/ of homogeneous media for given temperature. The influence of heterogeneities was taken into consideration with the use of GRUCAH module /51/ providing the input data for the GRUCAL.

4.3 Fusion-Fission Hybrid

The most of neutronic problems of hybrid reactors are identical with the ones of fusion reactor and therefore are not to be discussed here once again. The need of the best neutron utilization the radiation damage and all the questions joined with the tritium breeding are common for both these types of ANES.

Thus, we confine ourselves here to deal with the particularities resulting from the presence of fertile and fissile media in the hybrid reactor blanket and from the one of its main objectives - the fissile breeding.

4.3.1 Calculations and results

In recent hybrid designs /35, 40 - 42/ large quantities of beryllium occupying 50 - 70 % vol. of the breeding zone are applied because of its excellent neutron multiplying properties. Unfortunately, in such media intensely slowed-down neutrons "jump" over self-shielded resonances and are finally absorbed in ^{6}Li and also in fissile and structure materials. A series of performed calculations concerning various blanket concepts confirm this opinion.

As it can be seen in Table 4.1, the RSS is the source of deeply misleading overestimation of fissile breeding in moderated systems, whereas the hard spectrum blankets are characterized by much lesser errors resulting from the neglect of RSS. Though such blankets require larger volume than the well moderated ones for the reduction of leakage losses only there the fission suppression is effectively assured.

Table 4.1

Influence of the resonance self-shielding on the fissile breeding

Breeding zone composition		66 % Be 17 % $Li_{17}Pb_{83}$ (15 % Li 6) 3 % ThO_2 (.2 % ^{233}U) 4 % SS	45 % C 42.5 % Li_7Pb_2 (1 % Li 6) 2.5 % ThO_2 (2 % ^{233}U) 5 % SS	80 % $Li_{17}Pb_{83}$ nat 10 % ThO_2 (1.5 % ^{233}U) 10 % SS	(0.5 % ^{233}U)	80 % $Li_{17}Pb_{83}$ nat 10 % UO_2 (1.5 % ^{239}Pu) 1 % ^{233}U) 10 % SS	55 % Be 40 % Li 3 % Th (1 % ^{233}U) 2 % SS	
self-shielding corrected-to-self-shielding neglected-ratio	net fissile breeding	.61	.60	.94	.95	.90	.53	.39*
	slow fissions	1.35	1.46	1.07	1.06	1.14	1.62	1.85*

*heterogeneities corrected

$$\frac{4V}{S} = .075 \text{ cm}$$

To the contrary, in the slowing-down systems, it is demonstrated (figs. 4.6a and b) that for moderator high volume fractions, this disadvantageous neutron captures distribution (too few in fertile media) can be improved only at the cost of simultaneous additional energy production. In the respective calculations the question was at what increase in the fertile volume fraction and thus in the fission rate and energy release, one can obtain the same breeding rate as the one evaluated while neglecting the RSS. For this purpose the following was assumed:

1) The net fissile breeding and fission rates evaluated without considering RSS effects were taken as units.
2) The increase in fertile concentration (of constant enrichment) was associated with the equal decrease in lithium concentration, their sum volume fraction and thus the one of the moderator remaining constant.

As one can see in figs. 4.6a and b neither an increase in the fertile nor a decrease in the ^{6}Li concentration seem to be very helpful. The fertile capturing power is simply too low in the conditions of rapid transfer of neutrons down to the energies below resonances. It is so, since the RSS makes the addition of fertile media rather ineffective (only the captures at higher energies can be intensified) but simultaneously the fission suppression is worsened, as due to the heavy metal concentration increase, the slow and fast fission rates augment proportionally, too.

The simultaneous decrease in ^{6}Li concentration beneficially enabling more neutrons to be captured in fertile media at high energies, deteriorates obviously the neutron balance at low energies, again favourizing first of all the neutron absorption in fissile media and also parasitic losses in structure materials.

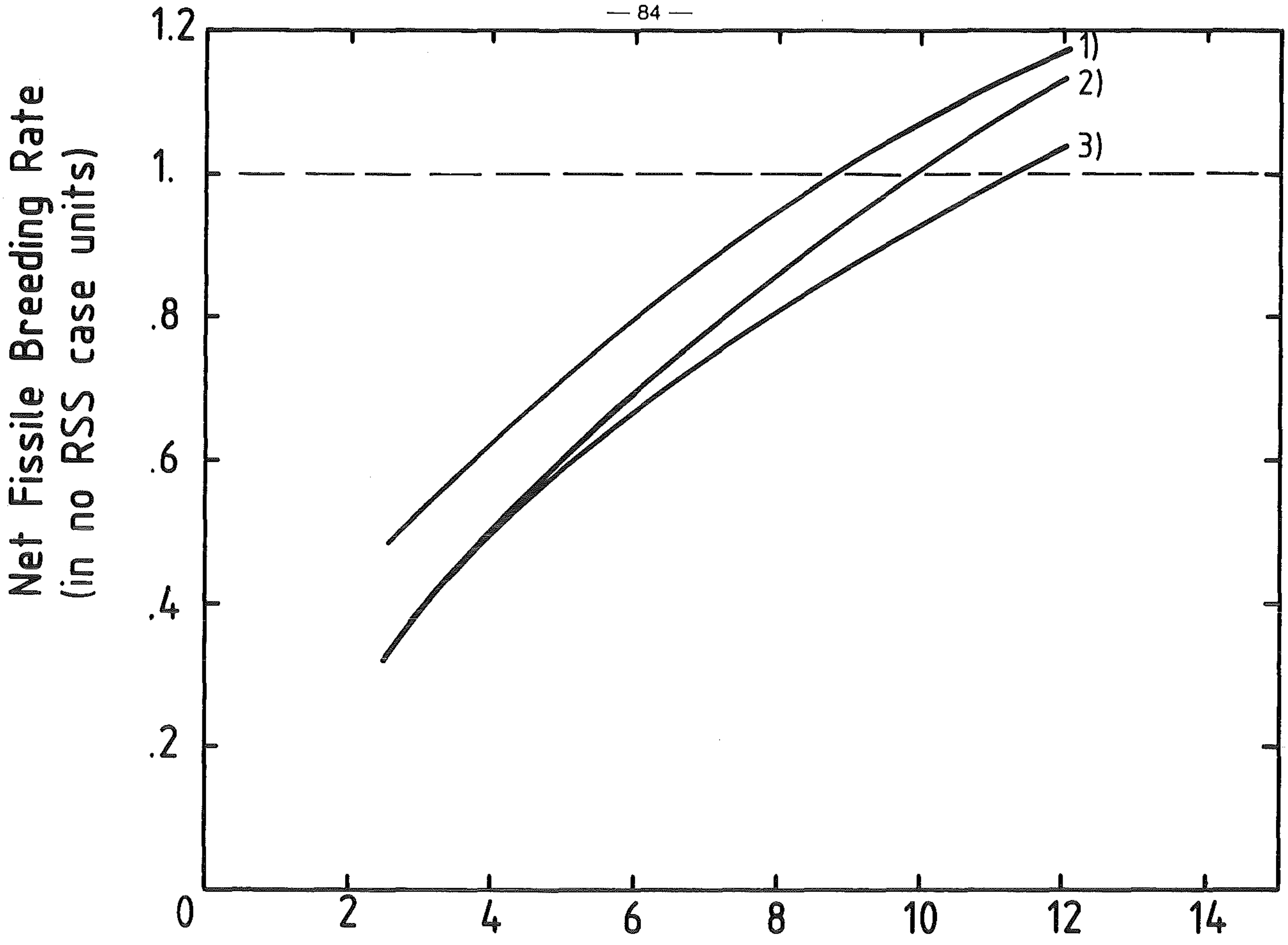
The increase in thorium concentration (3 - 4 times), necessary for maintaining the same fissile breading as the one estimated without consideration of the RSS effects results in about 4 - 5 times increase in fission rate in the blanket. The power production associated with it inadmissibly decreases the breeding efficiency of the system especially when the heterogeneities are also taken into consideration.

In view of the above a decrease in Be concentration looks unavoidable. However, a considerable reduction of beryllium content, sufficient for significant slowing-down attenuation that would enable neutrons to be captured in fertile media reduces the neutron multiplication in Be, thus cancelling the reason for its usage.

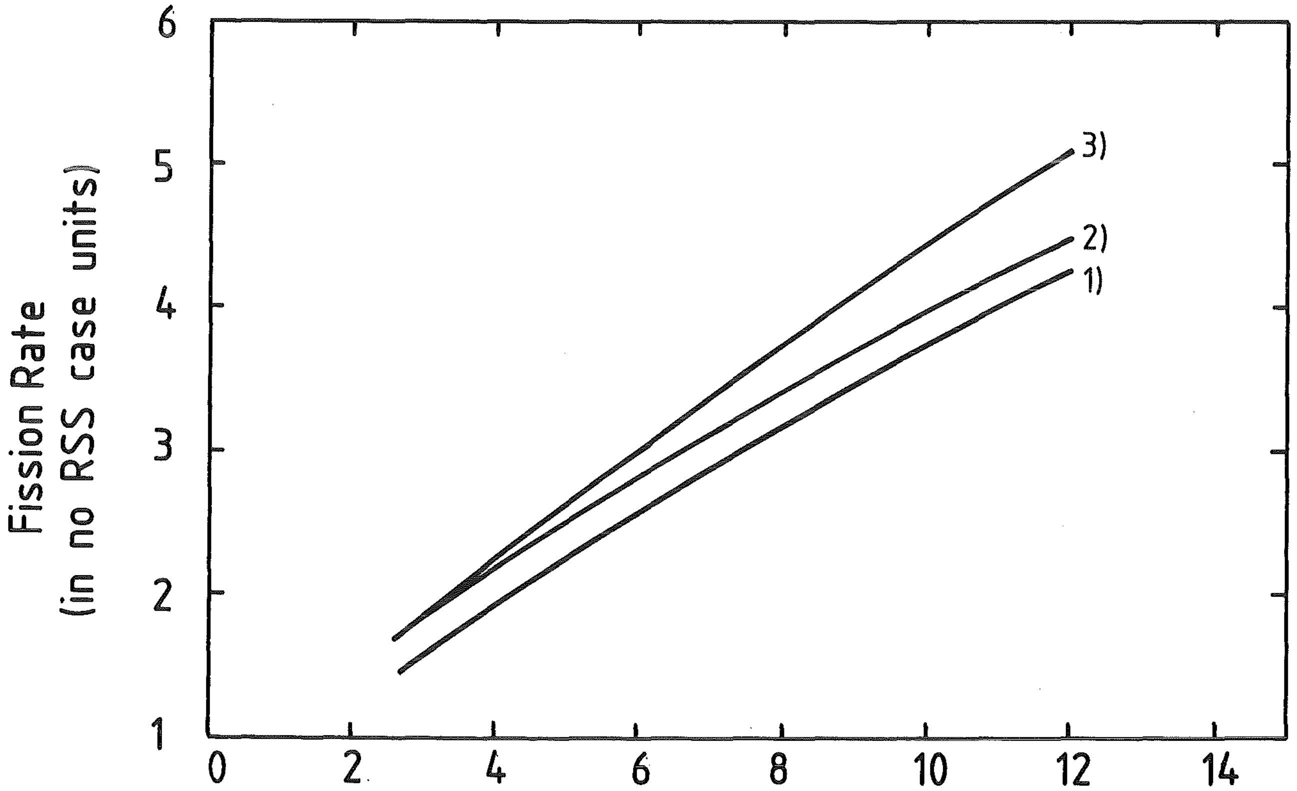
In light of the above remarks the removal of any moderator from the fissile breeding zone and the use of weakly slowing-down neutron multiplier (Pb) is proposed.

On the other hand it does not signify that Be has to be totally forgotten while developing hybrid blanket concepts. Having in mind that in view of 1/v type of the main tritium producing reaction a well moderated system is desirable /52, 26, 27/ the fusile breeding region (only!) with Be multiplier spatially separated from the fissile breeding zone with least moderating multiplier (e.g. Pb) might be advantageous.

Such separation of fusile breeding region of moderating properties seems realistic in case of mirror systems due to their length rendering the solid angle of its any part to be small, as seen from the rest of the blanket, what conditions the effective separation of the regions in question. In this way a higher fusile and fissile breeding may be attained thanks to supposed superior (to Pb) neutron multiplication properties of beryllium at some savings in the volume of the breeding zone (as compared with Pb multiplier). Simultaneously a significant reduction of the Be inventory by a factor of 3 - 4, as compared with a complete slowing-down system can be obtained.



b) Net fissile breeding rate



a) Fission rate increase

Fertile Material Concentration [% Th]

Fig. 4.6 Reaction rates dependence on the Th concentration in Be moderated hybrid blanket with consideration of RSS effects (volume fractions: 55 % Be, 2 % SS, 43 % (^{232}Th (1 % ^{233}U) + Li (1 % ^{6}Li)), cycl. geom. 1.5 m FW radius, 1 cm FW thickness, 100 cm breeding zone thickness, 4 V/S = .075 cm for heterogeneities influence evaluation /10/

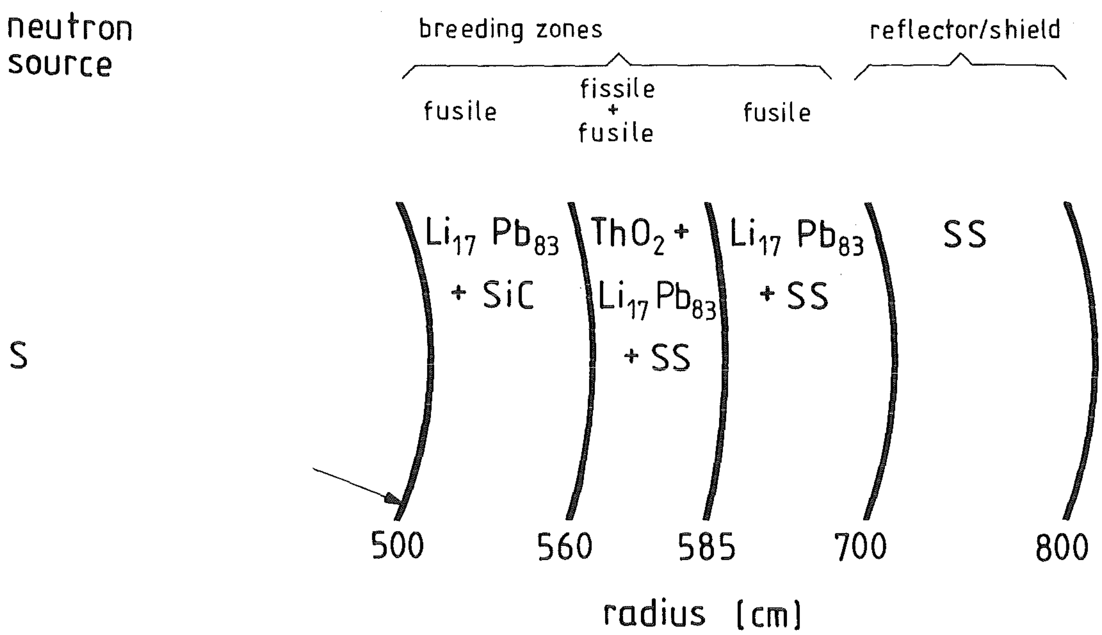


Fig. 4.7 Example of the hard spectrum breeding blanket

To evaluate these effects quantitatively remains in the plans of the author.

In view of all the above, the hard spectrum HIBALL /53/ type blanket assuring high endoergic neutron multiplication was taken as the basis for the hybrid breeding zone design (fig. 4.7). It was assumed that fuel elements can be placed in deeper layers of the $Li_{17}Pb_{83}$ "water fall". The multiplying zone of 33 % $Li_{17}Pb_{83}$, 2 % SiC and 65 % void /51/ preceded the fissile breeding zone of various thicknesses and position, composed of 90 % $Li_{17}Pb_{83}$ + fissile fuel and 10 % SS. Both, the thorium and the uranium cycles were considered. According to the earlier suggestion (shuffling), the 1.5 % U 233 mean enrichment level was chosen for the fresh enrichment cycle and the 1.5 ^{239}Pu + 1 % ^{235}U for the spent fuel enrichment.

The results of calculations for spherical geometry, corresponding to very coarsely localized optimum are enclosed in the Table 4.2 and the breeding spectrum is presented in fig. 4.8 (where for comparison also the one for Be multiplier case is given).

The obtained results require certain comment.

Due to some approximations in the calculation model (isotropic inelastic neutron emission, 100 % coverage, spherical geometry, homogeneous case self-shielding correction) and to cross-section uncertainties, the present results may require recalculations. On the other hand, some of the above effects can cancel each other as, for instance, the neutron losses resulting from the not 100 % coverage could be compensated by the lower enrichment than the assumed one thus improving the breeding efficiency (per energy released). Also a source of non-utilized reserves lies in the rapidity of enriching, enabling still better fission suppression because of the delayed build up of the fissile material.

Table 4.2

Performance of the hard spectrum hybrid blanket

cycle-average enrichment	net fissile breeding	tritium breeding	fissile absorptions	total energy (MeV)	fissile production $kg/GW_{th} \cdot y_r$
	per source neutron				
Th 232 - U 233 (1.5 % U 233)	.62	1.01	.07	35	1350
U 238 - Pu 239 (1.5 % Pu 239 1 % U 235	.69	1.01	.12	43	1250

In turn, the comparison of the breeding spectrum in Pb multiplier systems (fig. 4.8) with the breeding efficiency spectrum (fig. 4.5) indicates that the better flux shaping is hardly possible. In contrast to Be based blankets both these distributions are perfectly correlated in hard spectrum systems and only an increase in the number of neutrons but not any changes in their spectrum seem to be much significant.

4.3.2 Hybrid blanket nuclear design

The guidelines of hybrid blanket nuclear design are in part similar to those of fusion reactor. The fundamental difference is the necessity of fission suppression that dissuades the (recommended in pure fusion case) use of moderators. Therefore:

1. The possibly high neutron multiplication should result from endoergic processes which (inelastic neutron emission) driving neutrons below the fast fission threshold assure the fast fission suppression.

2. The probability of processes competing to multiplication should be possibly low i.e. the relative concentration of structure materials and coolants must be as low as possible.

3. The effective neutron capture in fertile materials admits no moderator in the fissile breeding zone (the same is valid also for a reflector one).

4. In all places of non-negligible flux (except of fertile medium, obviously), the neutron absorptions should occur in ^{6}Li. Thus the tritium breeding substance can prevent the parasitic absorptions wherever the reaction rates are significant.

4.4 Spallation breeding

The performed studies of spallation breeders are of preliminary character and have as a task only to show generally the negative consequences of neutron slowing-down in breeding systems mainly because of RSS effects in fertile media. The main factor limiting this research was the lack of high energy transport code and data thus confining the carried out calculations to the simple reactor ones.

Therefore, any future particular design, esp. based upon the indications issuing from this study would require more detailed and comprehensive calculations. However, even the scope of carried out calculations seems to be sufficient for the limited purpose of substantiating the hard spectrum spallator target/blanket concept proposed below.

4.4.1 Resonance self-shielding effects

Not lessening the significance of RSS in hybrids one can safely state that these effects become still more important in spallator target/blanket assemblies. It is due to the fact, that here in contrast to fission suppressed hybrid reactor blankets, the fertile and thus fissile materials concentrations should be much higher and are expected to approximate the values typical for fission reactors. So high heavy metal concentration in the spallator target are necessary in order to obtain the desired neutron production which is most abundant, in fast fissionable heavy metals due to fast fissions associated to spallation processes. Thus, the concentration of these should prevail over all the other nuclides in the target. In the blanket, it must be also sufficient to prevent parasitic captures in structure materials (whereas in hybrids or in case of spallation-fusion symbiont only, ^{6}Li can sucessfully fill this task). In light of all the above, it is to foresee that such systems, when moderated by light or heavy water coolant /13, 43 - 46/, must be close to criticality or even supercritical and breed very poorly or even net burn also at quite low enrichment levels.

The above statements and effects have been confirmed in performed calculations for two fuel cycles ^{238}U - ^{239}Pu and ^{232}Th - ^{233}U in light water cooled, grafite and lead reflected spherical system with the 2 m radius corresponding to the target volume of ~ 33 m^3, and 2 x .3 m reflector thicknesses according to the latest design /13, 45, 46/. The fuel of 25 % and the SS of the 5 % volume fractions have been assumed, the rest being filled with water coolant and void. The choice of those particular system parameters does not deprive present calculations of their general character. Another geometry, heavy water coolant, the presence of fission products or of further Pu isotopes will not change the essence of presented results. In these calculations the influence of heterogeneities was not considered, but since they can only deepen the RSS, their neglect may be recognized as reasonably conservative while estimating the errors resulting from the total forgetting the RSS (figs. 4.9, 4.10 and 4.11).

First of all the supercriticality of the spallator target/blanket assembly obtained at quite low enrichment levels (figs. 4.9a, b) shows the scale and consequences of the neglect of RSS. Then, in addition to safety problems hidden by the neglect of RSS, the spallators objective - the fissile breeding is also lost. The weakly perturbed by self-shielded resonances transfer of neutrons into the energy region where the fuel destruction may dominate over the fuel breeding, deteriorates the relation of these two processes. Its quantitative measure, the conversion ratio is

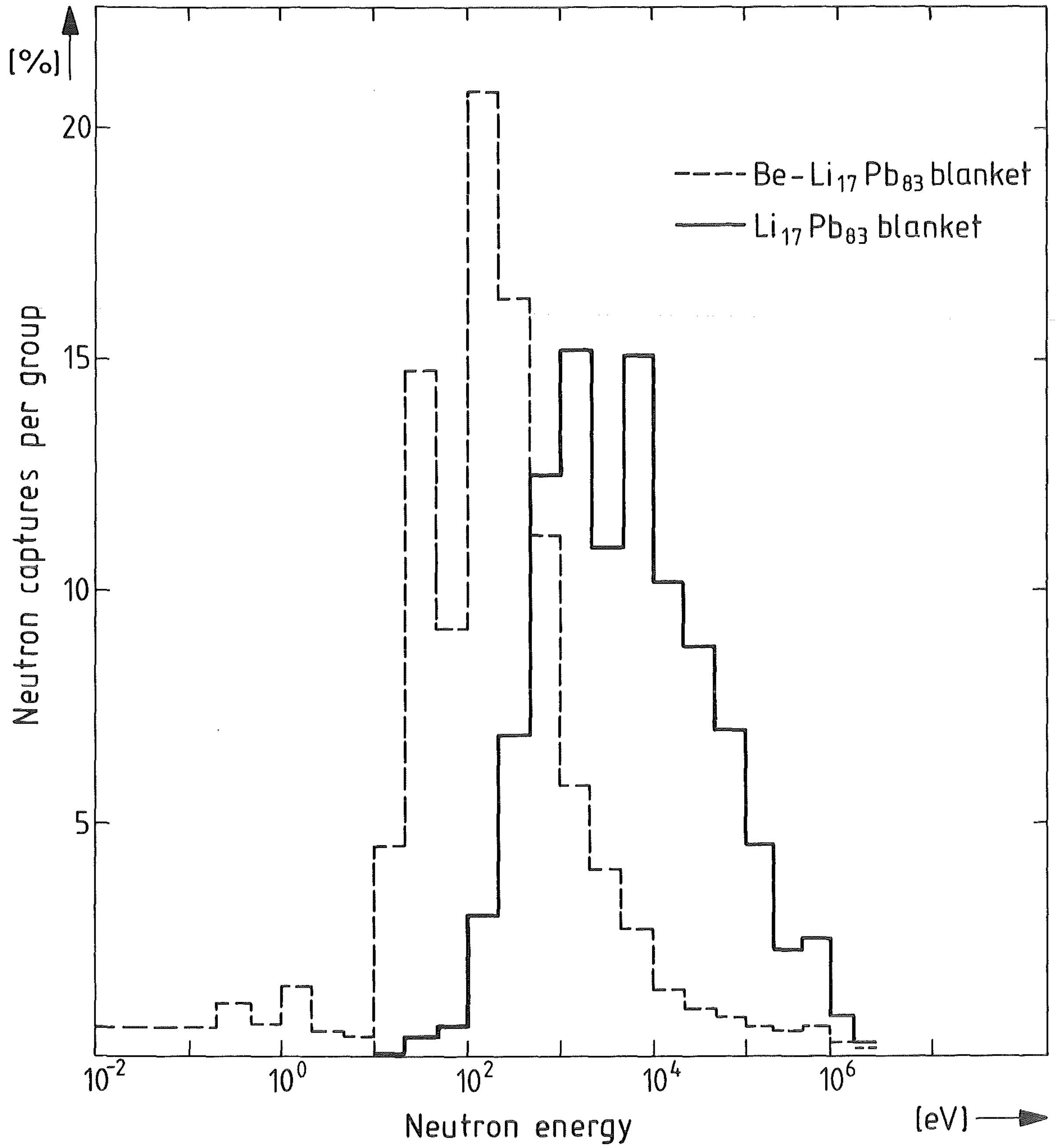


Fig. 4.8 Fissile breeding spectrum for moderated an unmoderated blankets (volume fractions: 66 % Be, 17 % $Li_{17}Pb_{83}$ (15 % Li 6), 3 % ThO_2, 4 % SS and 80 % $Li_{17}Pb_{83}$, 10 % ThO_2, 10 % SS respectively)

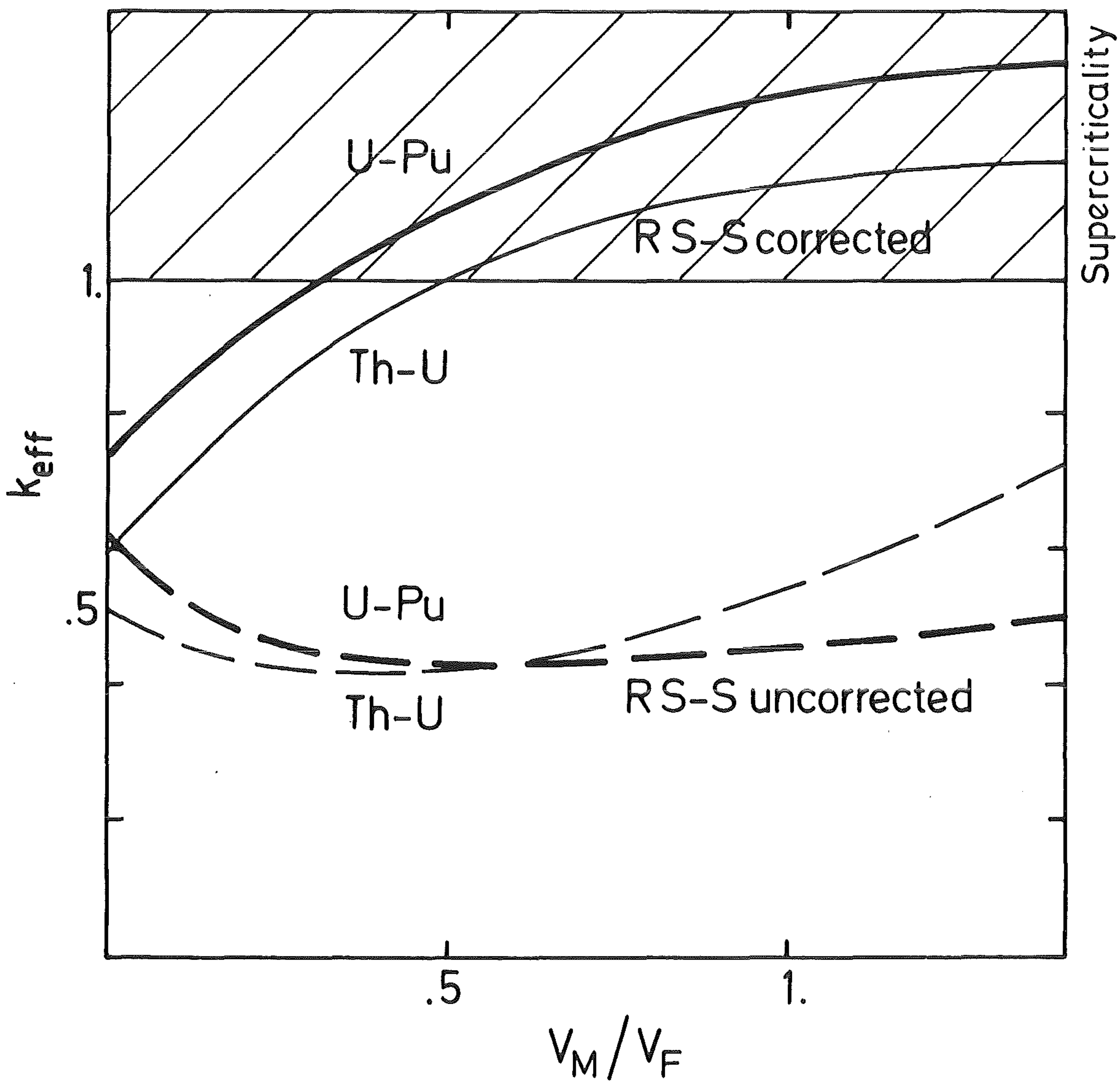


Fig. 4.9a Resonance self-shielding influence on k_{eff}
light water moderator
3 % enrichment

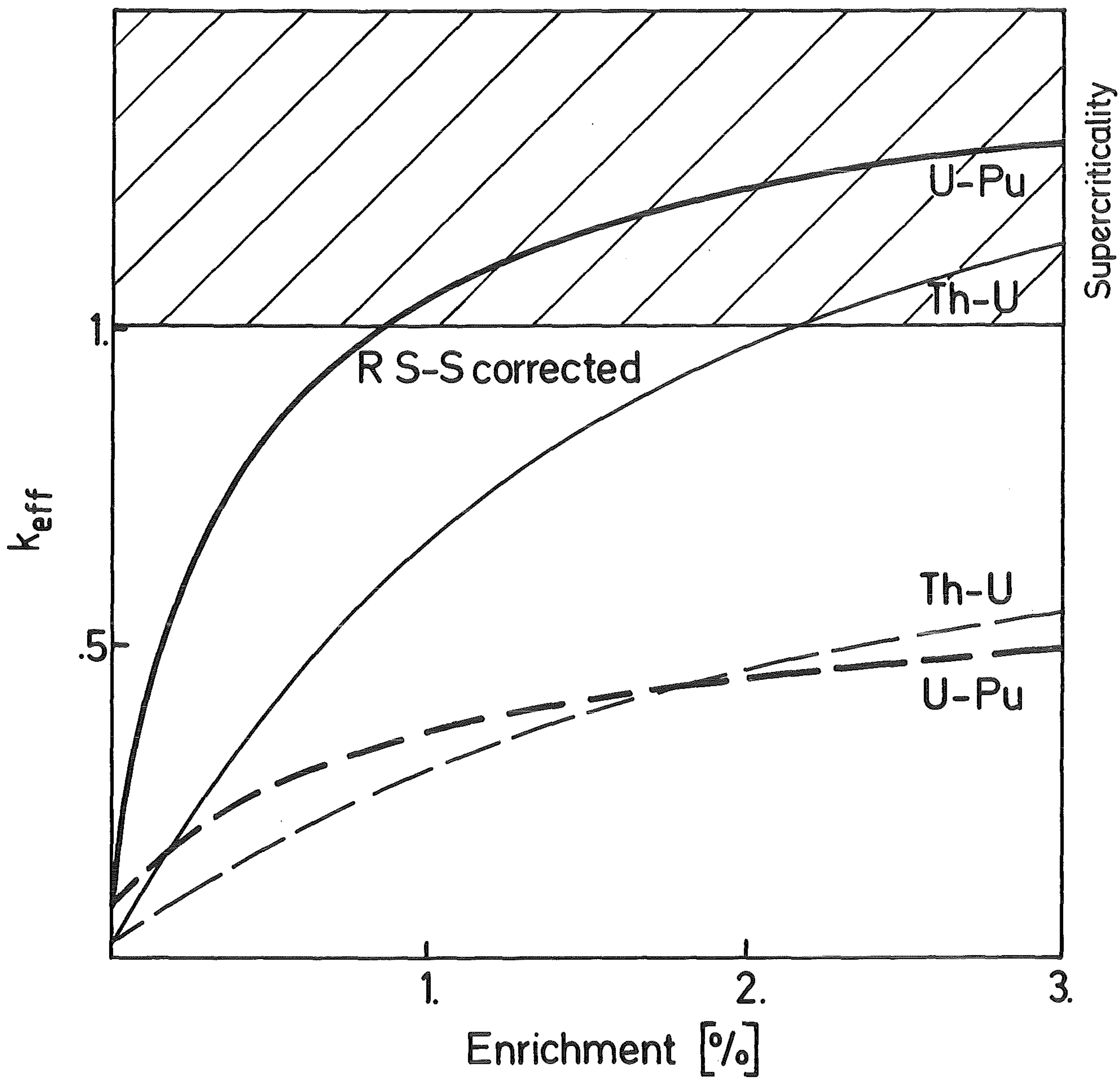


Fig. 4.9b Resonance self-shielding influence on k_{eff}
light water moderator
$V_M/V_F = 1$

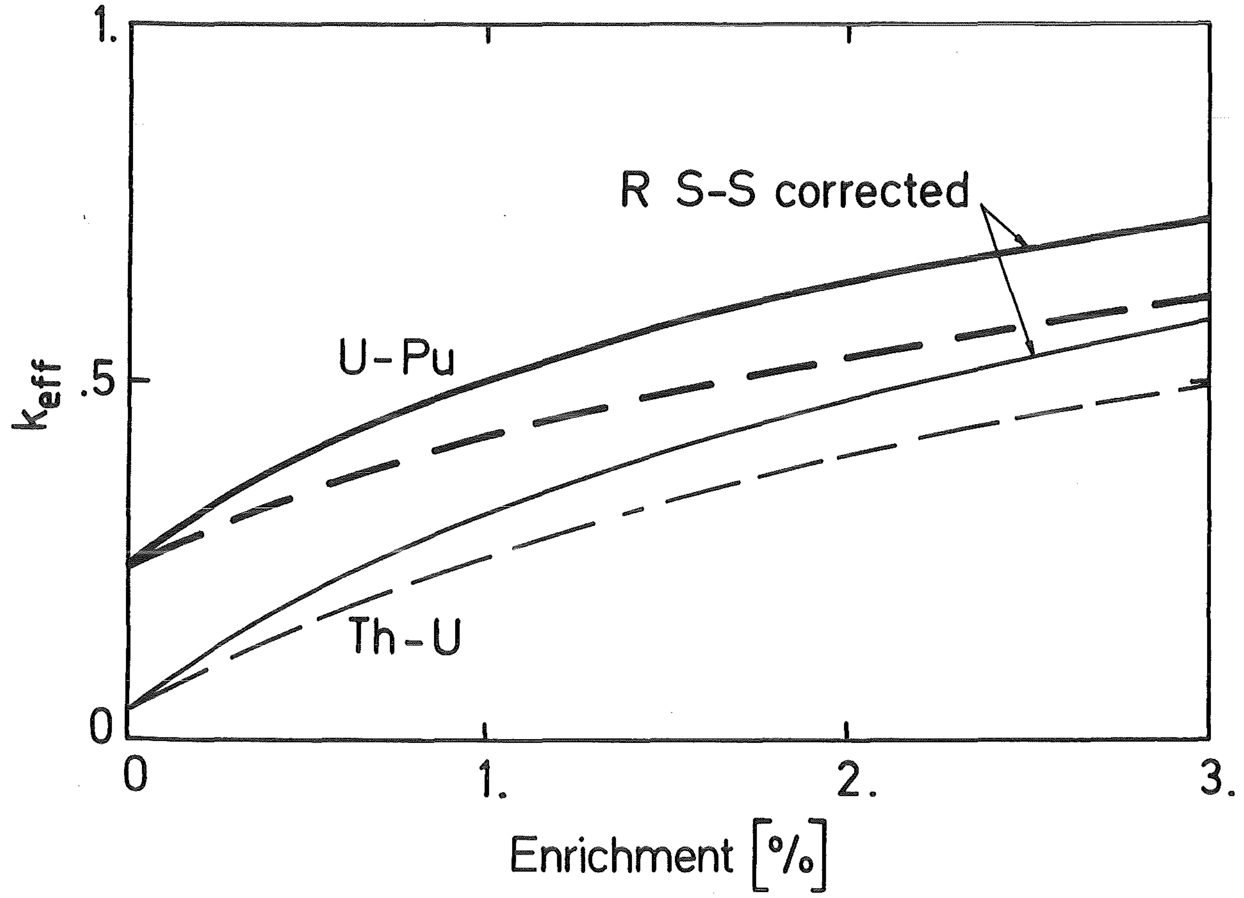


Fig. 4.9c Resonance self-shielding influence on k_{eff}
light water reactor
$V_M/V_F = 0$

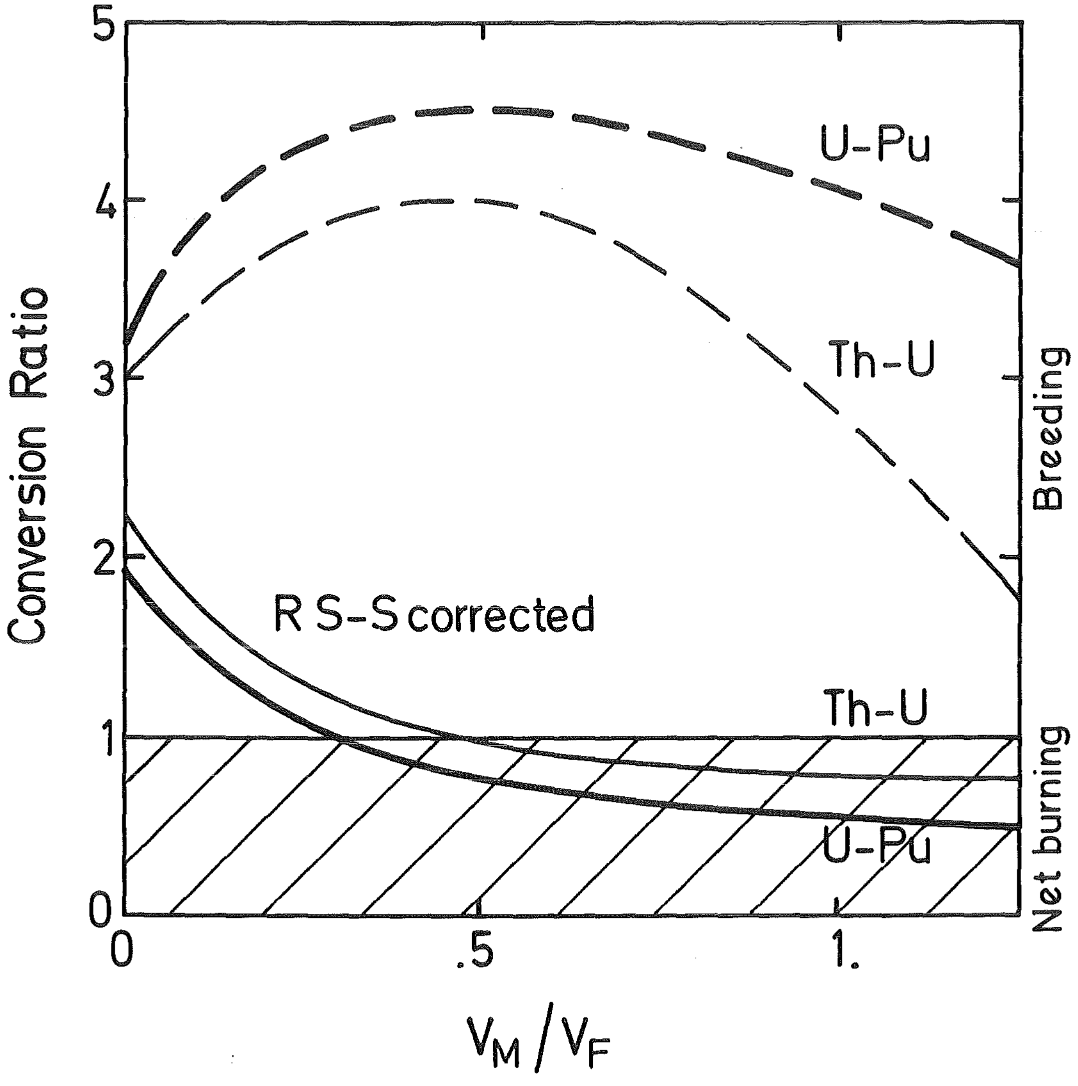


Fig. 4.10a Resonance self-shielding influence on the conversion ratio
light water moderator
3 % enrichment

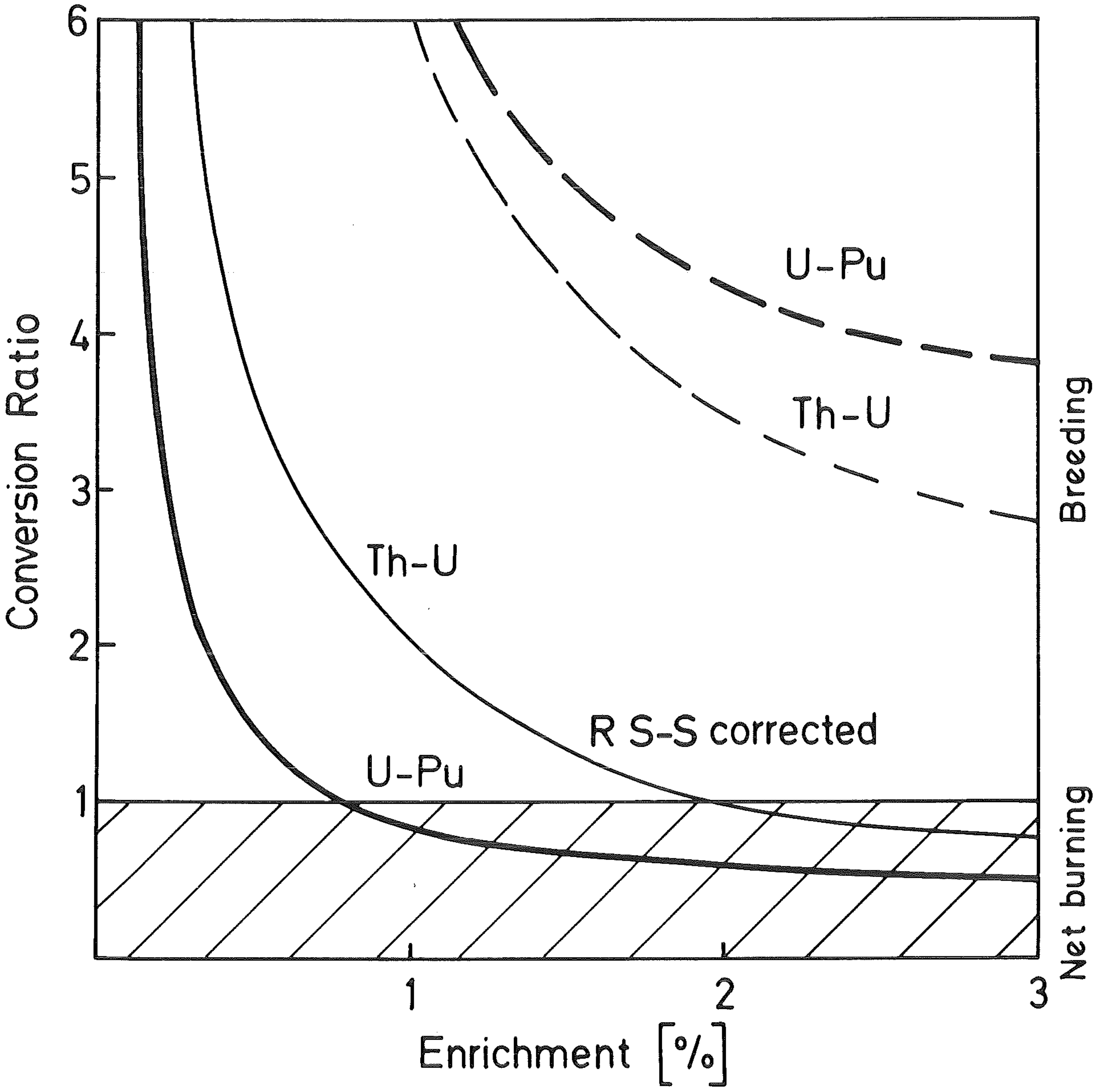


Fig. 4.10b Resonance self-shielding influence on the conversion ratio
light water reactor
$V_M/V_F = 1$

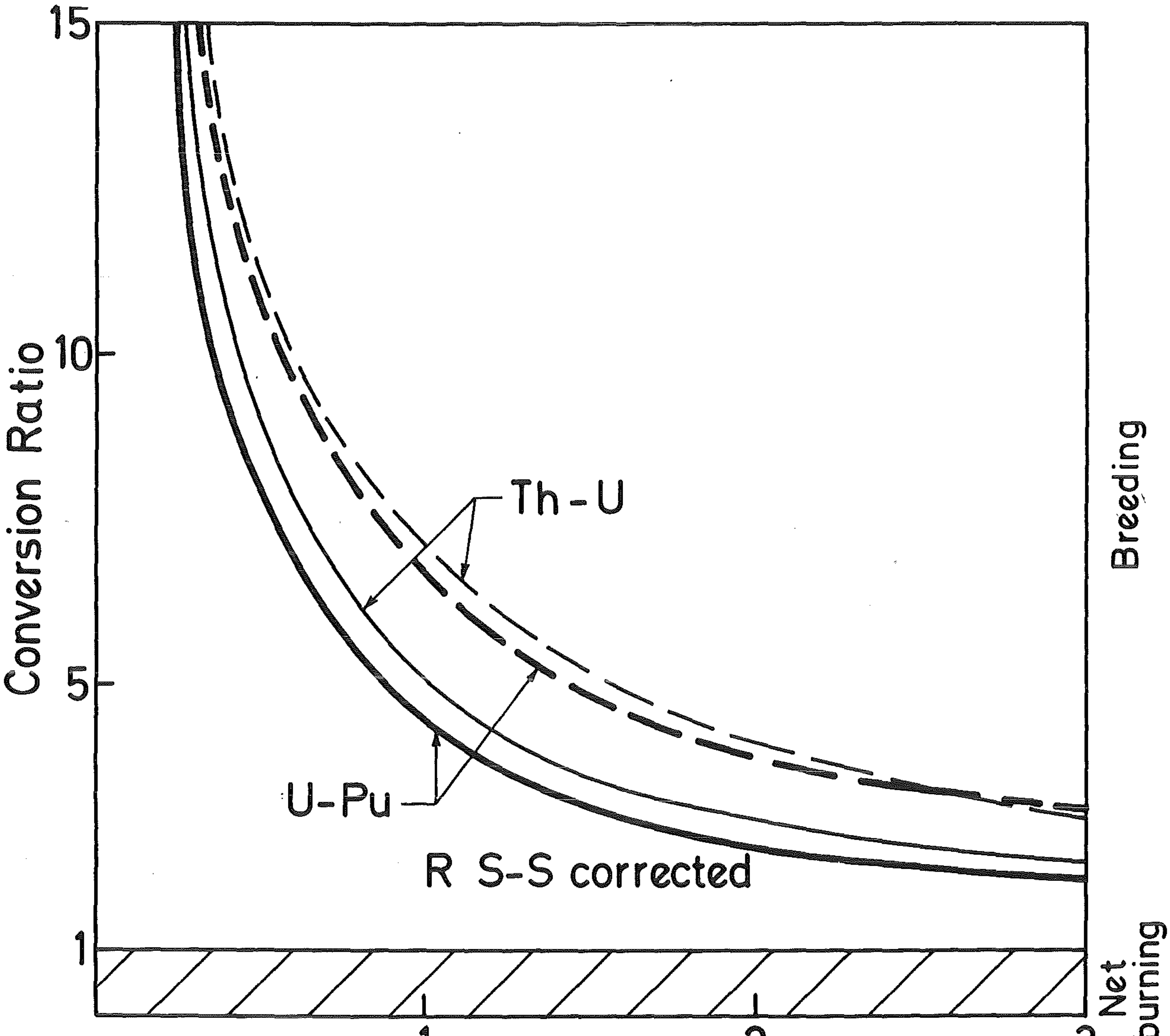


Fig. 4.10c Resonance self-shielding influence on the conversion ratio
light water moderator
$V_n/V_F = 0$

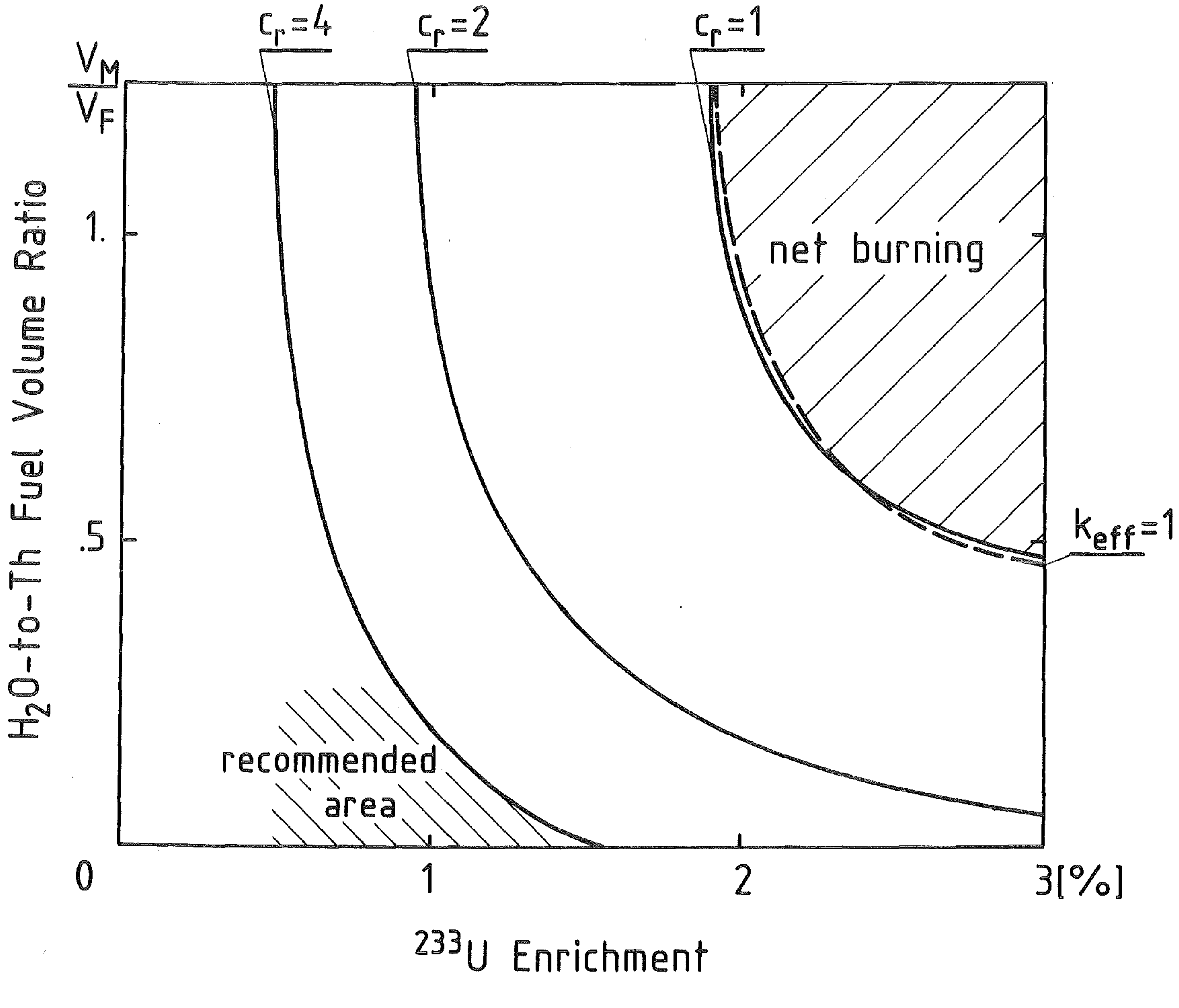


Fig. 4.11a Fissile breeding system parameter phase space Th cycle

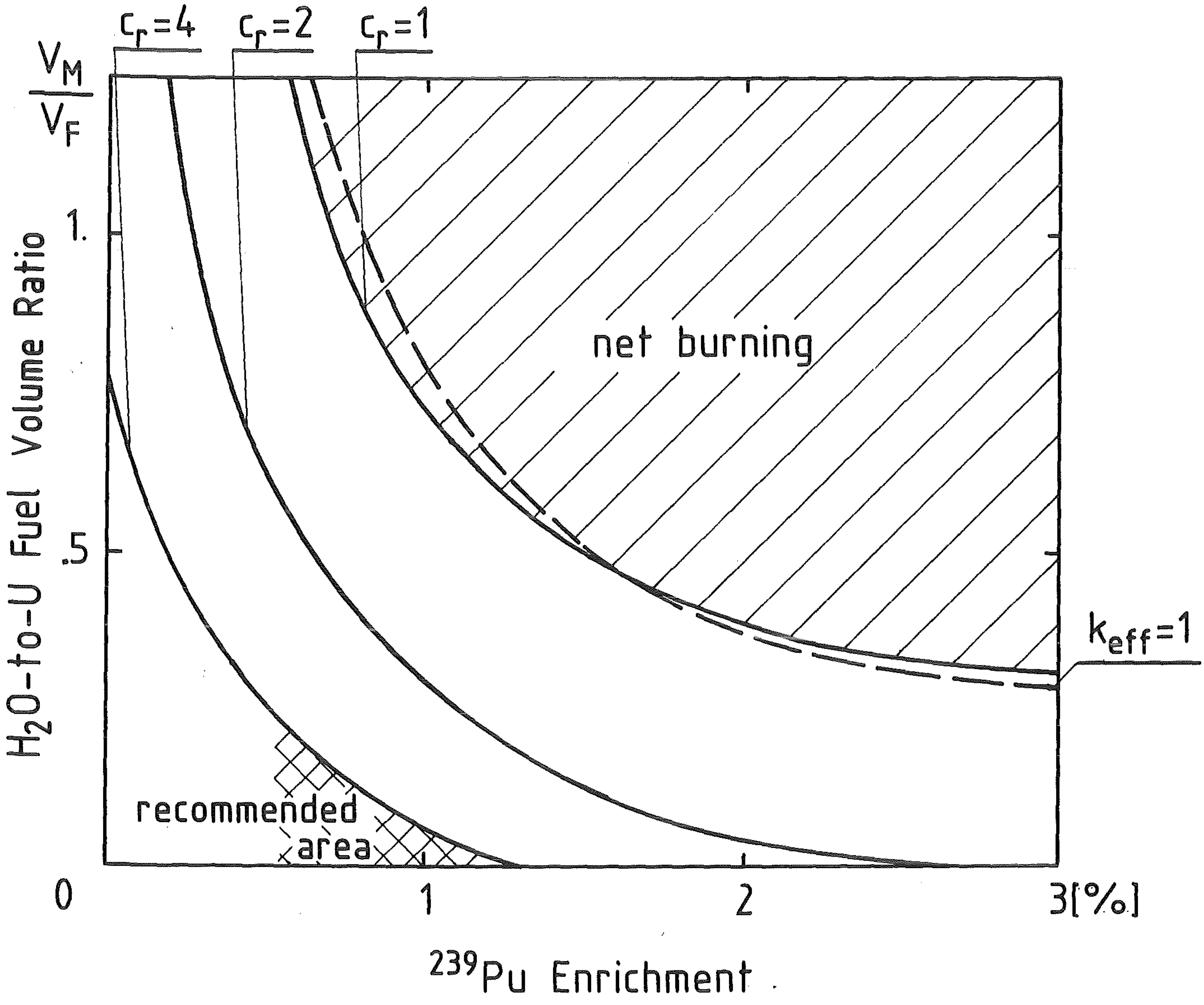


Fig. 4.11b Fissile breeding system parameters phase space U-cycle

reduced far below 1, thus turning the breeder into a burner and making even moderate enrichments inachievable in the presence of water coolant (fig. 4.10b). All this agrees with intuition, since breeding reactors are characterized by hard spectrum (fast breeders) and thus effective breeding admits no moderator.

As it might be expected, the significance and size of errors resulting from the neglect of RSS depends, by means of the neutron spectrum, also on the presence of a moderator in the system. For harder spectra neutrons are absorbed at higher energies where RSS is weaker, thus even if neglected does not cause so important errors (fig. 4.9c and 4.10c, cf. Table 4.2, too).

The joint dependence of conversion ratio on the two important and selectable system parameters: fuel enrichment and the light water-to-fuel volume ratio is shown in fig. 4.11a and b. In order to illustrate the safety question of the breeding assembly the respective k_{eff} = 1 isolines are also shown in there.

These recommended areas of low moderator concentrations and low enrichments are limited from one side by the minimum admissible enrichment (from the point of view of fissile recovery) and by the minimum necessary c_r on the other side. Higher enrichments are admissible only at low V_M/V_F ratios out of the effective water cooling possibilities in these conditions, since V_M/V_F = .3 can be recognized as the lower limit /54/. Instead, higher moderator concentrations transfer the system into the doubly inadmissible regions of supercriticality (safety!) and of net burning. It may be also worth to notice that the enrichment levels (2.5 % - 3 %) respective to the direct fuel enrichment or to the rejuvenation without reprocessing can be attainable only for assemblies with hardest spectrum i.e. with no moderator.

Therefore, also because of weaker disturbing the spallation process in heavy nuclei (neutron multiplication), a target/blanket system with the structure of helium cooled fast breeder blanket structure seems to be the most promising /55/ and is recommended herewith. This suggestion, obviously does not exclude the other non-moderating solutions like common Na based fast breeder cooling concept, anyhow being more appropriate for higher power densities.

Finally, also the energy release in the system is profoundly underestimated by the neglect of RSS. As it is shown above (fig. 4.1), the conversion ratio determines the net fissile production per released fission energy. Thus, also outside of the burning area in the parameter phase space (fig. 4.11a and b), the erroneous evaluation of conversion ratio corresponds to equivalent overestimation of the breeding efficiency (expressed e.g. in kg fiss. mat./GW_{th} yr).

4.4.2 Spallator nuclear design

In view of the results of calculations, the suggestion that the target/blanket structure should remind rather the simplified one of a fast breeder blanket (free from serious safety problems as being far from criticality due to low fissile concentrations) looks well substantiated.

It should be underlined that all the below statements are independent of the proton/neutron transport at high energies ($>$ 10 MeV) which only influence the total number of neutrons in the system. It is so, since (n,γ) process in fertile media at these energies is negligible and most of fissions occur also below 10 MeV.

Unfortunately, as it has been already mentioned, at the present level of uncertainties concerning the spallation processes and the accelerator efficiency, the recommended energy multiplication in the spallator has not been specified. But the control of the energy release could be done relatively easily by the selection of fertile-fissile concentrations (one obtains the lowest values by the addition of lead also e.g. as coolant in the form of $Li_{17}Pb_{83}$ eutectic). The right decision whether to produce more fuel and less energy or the opposite, will depend upon the absolute breeding efficiency and the actual fissile fuel-to-energy prices ratio. Therefore, leaving the above question for the further investigations and considering it, at present, as remaining beyond the scope of this study, only very general guide lines are formulated:

1) Avoid any moderators in the system.
2) Keep the concentration of neutron non-multiplying nuclei (coolants, structure) possibly low in order not to affect the neutron production.
3) Use heavy metal as a reflector (e.g. Pb, not grafite).
4) Irradiate rapidly (neutron flux relatively high).

Such hard spectrum spallation target/blanket proposal satisfies simultaneously several indispensable requirements:

1) It assures necessary system safety as being really far from criticality.
2) It enhances the fissile breeding neutron captures in fertile media esp. when associated with lower fissile concentrations.
3) The direct neutron production from spallation and fast fissions processes at minimized fissile destruction remains unaffected.
4) The hardest possible spectrum reducing slow fissions and captures thus assures high conversion ratio without excessive energy production (esp. in slow fissions).
5) Resonance self-shielding effects are much less pronounced (fertile capturing power remains little changed).

Therefore, also because of weaker disturbing the spallation process in heavy nuclei (neutron multiplication) a target/blanket system with the helium cooled fast breeder blanket structure (acceptable because of lower power densities than in fast breeders) seems to be most promissing and is recommended herewith.

5. Conclusions

The present study permits us to formulate some concluding remarks of two types - a very general one and a series of more particular ones.

The first one may be reduced to comment the status of scientific and technological research in the field of advanced nuclear energy systems. Namely one meets relatively detailed designs of ANESs (from the engineering point of view) and simultaneously can observe certain lack of satisfactory physical analysis of the problem. Seeing that, this study has tried to contribute into overcoming that disadvantageous delay in the consideration of significant physical phenomena occurring in blankets of ANESs.

The more detailed conclusions are to synthesize the presented discussions and indications regarding ANES. On the basis of simple comparative study of fissile breeding economics of fusion-fission hybrids, spallators and also of fast breeder reactors and with quite conservative assumptions it was shown that hybrids and spallators can become economic at realistic uranium price increase and successfully compete against fast breeders. Instead, the pure fusion cannot become economic even within the very distant future.

As concerns the fusion reactor blanket certain statements are emphasized. For the most effective trapping of neutrons within the breeding zone and non-full coverage losses reduction, it should contain an intensive moderator (e.g. $ZrH_{1.7}$ and/or Be). All regions of significant slow flux should contain ^6Li, in order to reduce parasite absorptions in there. The neutron multiplying layer (simultaneously slowing-down within the fast region), preceding the breeding zone should not contain nuclides suppressing the neutron multiplication (i.e. structure materials, non gaseous coolants etc.).

In connection with the fissile breeding it is demonstrated that the neglect of resonance self-shielding in fissile breeding systems give rise to basic errors and mistaken design concepts. It causes inadmissible overestimation of fissile breeding in fusion-fission hybrids and, what is much more important, it hides the danger of system criticality and leads to total unconscious missing of the objective - fissile breeding of water cooled spallation breeders. In result of the proper treatment of RSS, that prove particularly disadvantageous in soft spectrum systems the use of moderators in fissile breeding blankets is strictly dissuaded, that may require a fundamental change of the design strategy. Technologically attractive water cooled spallator systems and only Be neutron multiplier based hybrids prove hardly acceptable. Nevertheless, all the above does not question the principles and fundamental advantages of fusion and spallation breeders.

6. Summary

In light of the need of convincing motivation substantiating expensive and inherently applied research (nuclear energy), first a simple comparative study of fissile breeding economics of fusion fission hybrids, spallators and also fast breeder reactors has been carried out. As a result, the necessity of maximization of fissile production (in the first two ones, in fast breeders rather the reprocessing costs should be reduced) has been shown, thus indicating the design strategy (high support ratio) for these systems. In spite of the uncertainty of present projections onto further future and discrepancies in available data even quite conservative assumptions indicate that hybrids and perhaps even earlier - spallators can become economic at realistic uranium price increase and successfully compete against fast breeders.

Then on the basis of the concept of the neutron flux shaping aimed at the correlation of the selected cross-sections with the neutron flux, the indications for the maximization of respective reaction rates has been formulated. In turn, these considerations serve as the starting point for the guidelines of breeding blanket nuclear design, which are as follows:

1) The source neutrons must face the multiplying layer (of proper thickness) of possibly low concentration of nuclides attenuating the neutron multiplication (i.e. structure materials, non-gaseous coolants).
2) For the most effective trapping of neutrons within the breeding zone (leakage and void streaming reduction) it must contain an efficient moderator (not valid for fissile breeding blankets).

3) All regions of significant slow flux should contain ^6Li in order to reduce parasite neutron captures in there.

In the field of fissile materials production a measure of fissile breeding efficiency (fissile mass/energy released) is proposed as a function of the system conversion ratio and of the non-fissile (e.g. fusion neutrons, fast fissions) energy release in the system. Also a net effective fissile breeding cross-section is defined and its dependence and the one of the breeding efficiency on the resonance self-shielding (RSS) effects is demonstrated. It is shown in numerical calculations that the neglect of RSS of fertile materials in fissile breeding systems causes inadmissible overestimation of fissile breeding and underestimating of the energy production in spallators and fission-fusion hybrids. Consequently, their support ratio is significantly reduced and the danger of supercriticality appears in water cooled spallators. Finally, the necessity of consideration of the resonance self-shielding effects and the resignation of moderators in fissile breeding systems has been postulated.

Acknowledgement

The author is greatly indebted to the authorities of the INR-KfK for enabling him to carry out this work and first of all to its Deputy Head Mr. M. Küchle whose stimulating critical remarks have created an irreplaceable contribution to the present study. Critical comments of Dr. H. Küsters are kindly appreciated and thanks are due to Dr. Dr. C. Broeders, E. Kiefhaber, K. Küfner and D. Woll for their assistance in computations. The author wishes to express his gratitude to all non-mentioned here whose help contributed also to this study.

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