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AN ENERGY AMPLIFIER FOR CLEANER AND INEXHAUSTIBLE NUCLEAR ENERGY PRODUCTION DRIVEN BY A PARTICLE BEAM ACCELERATOR

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Abstract

We describe the new concept of energy amplification to extract nuclear energy with the help of accelerator induced nuclear cascades, extending to practical energy production the well known "calorimeter" technique, widely used in High Energy Physics. The energy is produced from a nuclear fuel material disposed in a moderator medium through a process of breeding of a fissile element from a fertile element of the fuel material. After an initial phase, the rate between the concentrations of fissile and fertile elements reaches a substantial stability, resulting in a stable long term energy production. The device must operate at relatively low neutron flux, in the $10^{14}\,cm^{-2}\,s^{-1}$ range, to ensure the correct performance of the breeding cycle and to prevent the risk of criticality. Thorium as breeding fuel has considerable advantages when compared with Uranium, in contrast with the full breeding based classic reactors for which the use of thorium presents serious difficulties. Thorium is more abundant than Uranium, it generates much less transuranic actinides among the radioactive waste and the risk of nuclear proliferation is negligible. In the paper the basic concepts of the "Energy Amplifier" are presented as well as alternative illustrative designs of the device. Some economic considerations are also discussed.

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1. Introduction

1.1. A "Calorimeter" as an energy producing device? Nuclear reactors constitute a major source of practical energy and they are likely to continue to be so in the foreseeable future. However they are not without problems and other alternative approaches to energy extraction from nuclei are of great interest. We describe a practical possibility of extracting nuclear energy with the help of accelerator induced nuclear cascades. It is proposed to extend to practical energy production the "calorimeter", widely used in high energy physics. Such a device measures the energy of a high energy particle through observation of its (fully contained) energy deposition. A typical hadronic calorimeter consists of a series of heavy material elements alternated with a sampling medium, usually (liquid or plastic) scintillator or liquid Argon. It is well known that such a device can be made to "over-compensate" the (hadronic) energy deposited by an incident high energy particle using fissionable materials in the plates [1]. To such an effect, it is important to add a low-A element material, slowing down the neutrons which—because of fission and spallation—then play a major role in the energy multiplication. Would it then be possible to extend this "overcompensation" much beyond the requirements of high energy physics in order to achieve a large energy gain, namely to *"amplify"* the energy deposition of an incident high energy beam to the extent of paying off the energy for its production and eventually to make a net gain? It is for this reason that we would like to call our device an "Energy Amplifier".

In short the scheme (Fig. 1) consists of a high efficiency, high energy accelerator producing a hadron (protons or heavier nuclei) beam which is absorbed in a "calorimeter". As a consequence of the large number of nuclear collisions initiated by the beam, energy is released at the expense of the breaking-up of heavy nuclei inside the target. Energy (in the form of heat) is produced inside the block in an amount which largely exceeds the energy delivered by the beam. The calorimeter is designed in order to extract easily the heat produced and to optimise the development of the cascade in order to bring the over-compensation to its largest possible value, by creating physical conditions which are optimal for the break-up of heavy nuclei. A fraction of such energy in turn, is transformed into electricity used to run the accelerator, with the presumably remaining larger fraction delivered for external utilisation either in the form of heat or of electricity or both. In this scheme, like in the case of a compensating calorimeter, the main energy production mechanism is based on fission, in view

of its unique high energetic yield (192 ± 7 MeV, excluding the contribution, 12 ± 2.5 MeV, due to neutrinos) and of the relatively large number of secondary neutrons which extend the development of the cascade.

When compared to a classic calorimeter used in elementary particle physics, the incident beam flux is now much larger. It is therefore possible to enhance greatly the energy amplification by breeding through nuclear transformations the content of readily fissionable materials, as is the case with Thorium, which will easily breed ^{233}U and of (depleted) Uranium ^{238}U which can breed ^{239}Pu.

1.2. Limitations of Nuclear Reactors. Today's nuclear energy is based primarily on fissions of natural ^{235}U, which constitutes however only about 0.71% of ordinary Uranium. Early in the development of nuclear energy [2], one realised the importance of breeding artificial fuels from more abundant nuclear species with the help of neutron captures. In particular starting from the dominant ^{238}U one can breed ^{239}Pu and from natural Thorium (pure isotope ^{232}Th) readily fissionable ^{233}U. While ^{238}U—^{239}Pu breeding naturally occurring in a Reactor has led to the extensive but controversial development of the (fast) breeder Reactors [3], so far relatively little progress has been made on the ^{232}Th—^{233}U breeding chain and on thermal or epi-thermal neutrons [4].

In a Reactor in order to maintain activity, at each neutron "generation" the number of fission produced neutrons must slightly exceed the initial sample. This imposes very stringent conditions on the dynamics of the device and limits considerably the number of possibilities. The Energy Amplifier circumvents such difficulty since additional neutrons are supplied externally and thus opens the way to many other reactions which are far more efficient in fuel utilisation and acceptable from the environmental point of view. The concept of using accelerators to produce fissionable material goes back to 1950 by E. O. Lawrence [5]. The Material Test Accelerator, MTA was a first prototype of such accelerator breeder [6]. During the late seventies and eighties a number of designs [7] have considered the possibility of incinerating actinides produced by nuclear reactors and of breeding fissionable fuel from Thorium or Uranium, to be later burnt in ordinary reactors. Neutrons produced by a high energy beam on a heavy target via spallation can be used to produce several well known transformation processes [4], from Tritium production to incineration of nuclear waste. This last application has recently received most of the attention [8], in view of the growing concern for the accumulating radioactive waste from reactors. In this paper we take—so to say—a fresh start and we concentrate instead only on the idea of

producing clean energy from abundantly available materials with the simplest means, with a minimal amount of radioactive waste and eliminating all risks of nuclear proliferation.

In particular nuclear Reactors are plagued by an insufficient breeding power to use natural Thorium as the primary fuel in practical conditions [4]. In order to have a self-sufficient breeding chain reaction, the number of secondary neutrons η resulting from one neutron captured must exceed 2 for the fissile element: each time one neutron must be sacrificed to replace the fissioned nucleus out of the fertile nucleus and another one is needed to continue the fission chain. Such fully sustained breeding is very difficult in a Reactor, since for thermal neutrons $\eta = 2.29$ for ^{233}U, very close to the minimal condition $\eta \geq 2$. Therefore in a Reactor a fully sustained breeding is plagued by the problem of the neutron inventory. In order to ensure at the same time breeding and the criticality, at most a fraction $(2.29-2)/2.29 = 0.126$ of the neutrons may be lost by containment losses and captures in other materials. This is very close to the minimal value of neutron losses which can be achieved using the most careful design and heavy water moderation, leaving little or no room for the inevitable build up of captures due for instance to fission fragments and other mechanisms of neutron absorption related to the breeding process, which will be described in more detail later on. Consequently a Thorium based thermal Reactor cannot operate in a satisfactory way on a self sufficient ^{232}Th—^{233}U cycle. Evidently an external supply of neutrons removes the above mentioned limitations.

1.3. Thorium as a Nuclear Fuel. Used in conjunction with the present method, Thorium offers very important advantages with respect to Uranium-based Reactors and breeders:

1) Thorium is more abundant than Uranium. More importantly, it is a pure isotope, which can in principle all be used as fuel. Hence in the Energy Amplifier, Thorium is a fuel 140 times more effective than natural Uranium in a Reactor where natural Uranium requires most often also a costly and complicated isotopic enrichment.

2) The breeding and energy producing reactions used in the present scheme generate minor transuranic actinides amongst the radioactive waste. In regime conditions an approximately constant quantity of fissionable nuclei is present and continuously burnt and regenerated from the bulk material. Such actinides are not considered to be "waste" since they constitute the badly needed "seeds" for the next load of the power generating plant.

Instead, conventional Reactors produce a large surplus of long lived and highly toxic actinides (the number of Plutonium nuclei produced is typically 0.5 to 0.9 of the fissioned ^{235}U nuclei), growing essentially indefinitely with the burn-up of the fuel.

3) Of course in both cases and for a given delivered energy, there is a comparable quantity of fission fragments, most of which are unstable. The toxicity of the fission fragments is strong, but much more short-lived. It decays well below the toxicity level of a volume of natural Uranium ores for an equivalent energy delivery in a period of a few hundred years, over which a safe depository is perfectly sensible.

4) The risk of nuclear proliferation is negligible, since the potentially strategic material, namely ^{233}U is present in the fuel as an isotopic mixture of ^{233}U (43.7%), ^{234}U (30%), ^{235}U (4%), ^{236}U (22.1%) and ^{238}U (0.068%) (approximate proportions within parenthesis are for an integrated neutron exposure $\int\Phi\, dt \approx 3 \times 10^{22}\, cm^{-2}$), namely it would require an intensive isotopic separation to achieve a "bomb grade" level ^{233}U. Plutonium isotopes are produced in very small quantities and "incinerated" so that they reach equilibrium with fractional concentrations indicated within parenthesis: ^{239}Pu (1.03×10^{-4}), ^{240}Pu (6.9×10^{-5}), ^{242}Pu (8.8×10^{-5}) and ^{238}Pu (1.97×10^{-4}), which has the moderate lifetime of 87.7 years for α-decay into ^{234}U. Fission and capture cross sections of the Uranium isotope mixture, averaged over the fission neutron spectrum are $\sigma_f = 2.00$ b and $\sigma_c = 0.74$ b, quite comparable to the corresponding values for ^{235}U, $\sigma_f = 2.60$ b and $\sigma_c = 0.73$ b. However the mixture can be easily "poisoned" with a small initially added amount of ^{238}U, for which the fission cross section is negligible and $\sigma_c = 1.14$ b. Such added Uranium will be maintained in any chemical Uranium separation. In this way the critical mass can be pushed beyond the limit of practical applicability. The capture cross section of ^{238}U for the neutron spectrum of the Energy Amplifier is about 1/3 of the one of Thorium and consequently such "poison" will live through the fuel cycle. Constructing a rudimentary bomb with our mixture even without ^{238}U poisoning has considerable added difficulties, since it produces a large γ-radiation and a large heat due to radioactive decays.

Similar considerations, although somewhat less favourably apply also to (depleted) Uranium ^{238}U, which breeds at equilibrium at a tiny and constant

concentration of 2.85 kg/ton of ^{239}Pu. Higher actinides are of course produced, but they also saturate at a comparable level, in contrast with an ordinary Reactor, for which the Plutonium "waste" is growing indefinitely.

1.4. Comparing with the Waste Incinerator concept. Recently, Bowman et al. [8] have proposed to use an accelerator-driven device to destroy ("incinerate") the radioactive waste (longer-lived actinides but also fission products) generated by present-day civilian as well as military uses of nuclear energy. The core concept developed by Bowman et al. relates to the interesting finding that in a very high neutron flux (≈ 100 times higher than in a nuclear reactor), actinides such as ^{237}Np can capture two neutrons in succession, thus bypassing poisonous ^{238}Np and leading to fissionable ^{239}Np. It is clear that the use of fluxes as high as 10^{16} neutrons $cm^{-2}s^{-1}$ demand a rather special arrangement: in any nuclear reactor, the accumulation of fission fragments results in "poisoning" which leads to replace fuel elements every few months or so. This would now have to happen 100 times more often, i. e. on a quasi-daily basis. Similarly, breeding of ^{233}U from ^{232}Th which these authors also contemplate would be impossible since the intermediary ^{233}Pa would have no time to decay, as its neutron capture half-life is much shorter than its 27-day β decay. Bowman et al. [8] propose to solve this difficulty by resorting to continuously circulating liquid flows (molten salt, heavy water, lead-bismuth eutectic) for the various parts of their device so that the fuels, moderating agent and spallation target spend only a fraction of the time in the very high flux and the remainder outside the "reactor" enclosure where it is envisaged that chemical separation, of fission fragments, actinides, separation of ^{233}Pa and subsequent milking of ^{233}U will take place "off-line". We do not wish to comment here on the technical difficulties of implementing this scheme which may well prove in a 20 -30 year perspective an interesting and viable proposal to destroy the accumulated stocks of Plutonium. We simply wish to point out to the readers that the scheme that we propose is entirely different. Our philosophy is on the contrary to strongly suppress the production of such actinides in the first instance.

The present scheme originated from a remark concerning an every day tool of high energy physics which is the "calorimeter". It is based on a low flux regime of ≈ 10^{14} neutrons $cm^{-2}s^{-1}$ which also happens to be that of present day reactors so that all problems of irradiation of materials etc., studied for reactors, are immediately applicable. This regime of 10^{14} neutrons $cm^{-2}s^{-1}$ is not arbitrary but, as it will be shown elsewhere, flows from three different and converging considerations: (i) In the case of an interruption of the beam irradiation, the decay

of the existing stock of ^{233}Pa to ^{233}U should not lead to such an increase of the latter in proportions leading to unwanted criticality (ii) The neutron flux should be low enough so as not destroy ^{233}Pa appreciably by neutron capture (iii) The amount of ^{233}Pa present as intermediary (which is proportional to the neutron flux) should be small enough so as to consume by capture a minimal amount of neutrons. It is actually condition (iii) which turns out to be the more demanding and has led us to base our arguments on a flux of $\approx 10^{14}$ neutrons $cm^{-2}s^{-1}$, at which value it is readily shown that the other conditions are satisfied.

The reader will appreciate that condition (ii) is really orthogonal to the one demanded by an incinerator scheme. It is true that Bowman et al. [8] also claim that they can use their device as a Th-U breeder. We doubt however that a purpose other than burning Plutonium would by itself justify the technical complication of on-line hot chemistry, since as we believe the same could be achieved using much simpler – in fact known– technologies.

1.5. Comparing with Nuclear Fusion. The Energy Amplifier can be compared in its expected performance with the long range perspectives of Nuclear Fusion. A fusion device based on Deuterium - Tritium burning will produce about four times as many neutrons at about seven times the average energy of fissions for the same amount of generated energy. In a Fusion Reactor, even if there are no fission fragments, these neutrons will interact and build up a large amount of radioactivity in the containment walls and in the nearby equipment, which will represent a radiation hazard of a magnitude comparable to the one of the fission fragments. Furthermore, while fission fragments are tightly retained inside the fuel cladding, the neutron contamination in a Fusion Power Station will be distributed over a number of large scale active components scattered over a very big volume. But in both instances, the bulk of the radioactive products is relatively short-lived (up to few hundred years) and represents a minor problem when compared with the Actinides from a thermal Reactor.

Lithium is normally used to breed Tritium. Hence a Fusion Power Station will essentially burn Lithium and Deuterium with Tritium as an intermediary. The natural availability of Lithium on the Earth's crust is estimated to be only seven times the one of Thorium and they are both vastly adequate for millions of years of very intensive utilisation.

More specifically, we can compare our device with ion-beam driven Inertial Fusion [9]. Both devices need a particle accelerator, but the one for Inertial Fusion

is much larger, complicated and expensive. The target gain for an inertially driven fusion device, according to the most optimistic assumptions will be G = 80 ÷ 100, about twice the one of our fuel. This factor however is likely to be substantially reduced and even lost since the efficiency of the corresponding accelerator will be lower in view of its much greater complexity. Hence the target gain for the Energy Amplifier concept here proposed is very likely to be close to the one assumed for Ion-beam Inertial Fusion when the complexity of the latter device is completely understood and duly taken into account.

Finally practical Fusion devices based on Magnetic Confinement [10] must be very large to ensure containment and efficient burning conditions. This is probably also the case of Inertial Fusion—although for different reasons. Their minimal economical power level is correspondingly very large – within the Gigawatt range. Our device can be built of much smaller dimensions, deliver economically smaller power outputs and therefore offer a much greater flexibility in their utilisation. Finally, the technology is much less sophisticated and this makes it far more suited than Fusion machines to respond to the growing energy demands of developing countries and as an alternative to fossil fuels.

1.6. Conclusions. The proposed scheme is driven by the wish of simplicity and achieves the goal of creating practical nuclear energy based on the natural Thorium breeding-burning cycle. The fuel is kept sealed and it contains a minimal, constant amount of fissile material, resulting from a stable equilibrium condition between breeding and fissions. The utilisation of each fuel load is expected to last several years of full utilisation in the Energy Amplifier without requiring manipulations. Eventually the fuel must be returned to the factory to be regenerated, removing the "poisons" due to fission fragments and recovering the chemically separated Uranium isotopes which will become the "seeds" for the next fuel load. Hence the breeding process can continue essentially indefinitely for each installation.

2. Thorium as a breeding fuel.

2.1. Main Breeding reactions. A very large fission cross section for low energy neutrons is the unique property of a few high Z nuclei such as ^{233}U. Nuclei like ^{232}Th have no significant fission cross section below ≈ 1 MeV, but they can be used to breed fissionable materials. At low energies, the (n-γ) reaction (neutron capture) is the only inelastic process, leading to a final (excited) nucleus

with one more neutron. In turn, the daughter nucleus is β-unstable and leads through a cascade of decays to a final, higher Z-nucleus. Hence the neutron capture reaction offers the possibility of "breeding" fissionable fuels from initial materials which are not fissionable, namely:

$$^{232}Th + n \Rightarrow {}^{233}Th + \gamma \overset{(22\ m)}{\Rightarrow} {}^{233}Pa + \beta^- \overset{(27\ d)}{\Rightarrow} {}^{233}U + \beta^- \{1\}$$

The main thrust of the present paper is the possibility of burning Thorium in conditions which are essentially free of higher actinide waste and especially of Plutonium. However one can also use natural or depleted Uranium instead of Thorium. Uranium has a similar breeding reaction, in which ^{238}U becomes ^{239}Pu with the intermediary of ^{239}Np:

$$^{238}U + n \Rightarrow {}^{239}U + \gamma \overset{(23\ m)}{\Rightarrow} {}^{239}Np + \beta^- \overset{(2.3\ d)}{\Rightarrow} {}^{239}Pu + \beta^- \{2\}$$

Let us define first the relevant cross sections for the mixture of elements in the fuel bars. The ratio of n-capture reaction σ_i to fission reaction σ_f, averaged over the neutron spectrum and the material composition is normally denoted with α and the neutron multiplicity with ν. Hence the fraction of fission and capture reactions are $1/(1+\alpha)$ and $\alpha/(1+\alpha)$ respectively. The quantity $\eta = \nu/(1+\alpha)$ is the number of secondary neutrons resulting from one neutron interacting.

2.2. *Reactions in a Thin Thorium slab.* Assume a thin slab of breeder material (^{232}Th) is exposed to an intense flux Φ of thermal neutrons. Indicating with (I), (II) and (III) the successive nuclei ^{232}Th, ^{233}Pa and ^{233}U in the chain {1}, (the γ-transition of ^{233}Th to its ground state and the subsequent β-transition of ^{233}Pa are combined in a single process) the basic differential equations are:

$$\text{(I)} \xrightarrow{\lambda_1} \text{(II)} \xrightarrow{\lambda_2} \text{(III)} \xrightarrow{\lambda_3} \text{(IV)}$$

$$\frac{dn_1}{dt} = -\lambda_1 n_1(t) \qquad \frac{dn_2}{dt} = \lambda_1 n_1(t) - \lambda_2 n_2(t)$$

$$\frac{dn_3}{dt} = \lambda_2 n_2(t) - \lambda_3 n_3(t)$$

In our case $\lambda_1 = \sigma_i^{(1)}\Phi$, $\lambda_2 = 1/\tau_2$, $\lambda_3 = [\sigma_i^{(3)} + \sigma_f^{(3)}]\Phi$ and initially $n_1(0) = n_0$, $n_2(0) = n_3(0) = 0$. Captures by ^{233}Pa are neglected for simplicity at this stage and will be considered later on. Solving the differential equations and in the approximation that $\lambda_1 << \lambda_2$ and $\lambda_1 << \lambda_3$, we find:

$$n_1(t) = n_1(0)\, e^{-\lambda_1 t} \qquad n_2(t) = n_1(t)\frac{\lambda_1}{\lambda_2}\left(1 - e^{-\lambda_2 t}\right)$$

$$n_3(t) = n_1(t)\frac{\lambda_1}{\lambda_3}\left[1 - \frac{1}{\lambda_3 - \lambda_2}\left(\lambda_3 e^{-\lambda_2 t} - \lambda_2 e^{-\lambda_3 t}\right)\right]$$

In stationary conditions $n_3/n_1 = \sigma_i^{(1)}/[\sigma_i^{(3)} + \sigma_f^{(3)}]$, independently of the neutron flux. Evidently it will not be possible to irradiate the fuel uniformly; notwithstanding, the breeder-fissionable mixture remains stable during regime conditions, independently of the local intensity of the neutron flux. In Fig. 2 we plot n_3/n_1 in the case of a ^{232}Th and ^{233}U mixture as function of the neutron energy over the wide interval 10^{-5} eV till 20 MeV. Below 1 eV we find a constant value, $n_3/n_1 = 1.35 \times 10^{-2}$. Above such energy, the ratio is rapidly oscillating in the resonance region and it settles to much larger values in the vicinity of $n_3/n_1 \approx 0.1$ for energies corresponding to the neutron spectrum from fission. Operation without moderator and with a neutron spectrum directly from fissions will give an equilibrium concentration of fissile material which is about seven times larger than the one for the thermalized neutron alternative.

The fact that after a turn-on period and in stable conditions, the fissionable content has a constant concentration is important and must be underlined. Stability can be verified qualitatively looking at the effect of small variations of n_3/n_1: a small increase (decrease) of n_3/n_1 will be corrected by an increased (decreased) burning and by a reduced (increased) breeding which, in turn, will decrease (increase) n_3/n_1. But instantaneous variations of beam intensity, while immediately reflected in the burning rate of the fissile material produce new fuel only after a time of the order of τ_2. For instance an increase of neutron irradiation will produce an immediate reduction of n_3/n_1 followed by an increase of n_3/n_1 only after τ_2. This is the classic problem of the delay in a feed-back loop.

Let us now look at the intermediary element, ^{233}Pa. In stationary conditions $n_2/n_1 = \sigma_i^{(1)}\Phi\tau_2$, which implies a density of (II) directly proportional to the neutron flux. Therefore variations in flux cause variations in n_2/n_1, which in turn imply a new transition period toward the new equilibrium condition. Then n_3/n_1 will no longer be independent of Φ, since instantaneous variations of beam intensity, while immediately reflected in the burning rate of the fissile material, produce new fuel only after a time of the order of τ_2. For instance, if the neutron flux is suddenly cut off, the (II) nuclei will decay with a rate $\lambda_2 = 1/\tau_2$ into (III), leading to a final population of (III) equal to n_2+n_3. Such an increase in fissionable material must not make the system critical, although the time lag is related to τ_2 and long (many days) and simple corrective measures can be easily taken. Therefore relative ratio $n_2/n_3 = (\sigma^{(3)}\Phi\tau_2)$ where $\sigma^{(3)} = \sigma_i^{(3)} + \sigma_f^{(3)}$ must remain small, setting a limit for the neutron flux Φ.

For a small change of the flux in the form of a step function of amplitude $\Delta\Phi$, the variations of n_3/n_1 is correspondingly small and it is given by the expression:

$$n_3(t) = n_1(t)\frac{\sigma_i^{(1)}}{\sigma^{(3)}}\left[1 + \left(\sigma^{(3)}\Delta\Phi\right)\frac{1}{\lambda_2 - \lambda_3}\left(e^{-\lambda_3 t} - e^{-\lambda_2 t}\right)\right]$$

where t is computed from the flux change and $\lambda_3 = \sigma^{(3)}\Phi$ is relative to the new flux conditions. More complex, time-dependent changes can be decomposed as a sum of step-functions and analysed with the above formula.

There is a second equally relevant condition which limits the neutron flux. Indeed, in order to achieve a large breeding most of the ^{233}Pa must survive neutron capture and rather decay into ^{233}U, which is translated into the condition $\sigma_\iota^{(2)}\Phi\tau_2 << 1$. Inelastic cross sections for energies up to few eV's (below the resonance region) can be parametrized as $\sigma(E) = (0.025\ eV/E)^{1/2}\Sigma$, with the parameter Σ listed for the relevant elements in Table 1. We remark that the loss of the ^{233}Pa through neutron capture which leads to ^{234}Pa, is quite costly to the neutron inventory, since ^{234}Pa quickly β-decays into ^{234}U, thus effectively bypassing the fission of ^{233}U. Therefore we set the condition $\sigma_\iota^{(2)}\Phi\tau_2 \leq 0.02$, corresponding to about 5% loss of neutron inventory.

Using Table 1 and setting $n_2/n_3 \leq 0.2$ we find $\Phi \leq 1.44 \times 10^{14}\ [T/(300°K)]^{1/2}$ $cm^{-2}s^{-1}$ for ^{233}Pa—^{233}U, corresponding to large power yields, namely of the order of 70 MW for each ton of fuel mass of ^{232}Th and reasonable temperatures. Practical operating conditions will normally not exceed such a limit. Condition

$\sigma_t^{(2)} \Phi \tau_2 << 0.02$ is translated into a temperature-dependent limit for the flux, $\Phi \leq 2.0 \times 10^{14}$ $[T/(300°K)]^{1/2}$ $cm^{-2} s^{-1}$ which leads to a condition comparable with the previous limit.

Table 1: Parameters of some nuclei below few eV: $[\sigma(E) = (0.025\ eV / E)^{1/2} \Sigma]$.

Element	*Elastic, Σ* *barn*	*Capture, Σ* *barn*	*Fission, Σ* *barn*	*n-multip.* ν	*sec/prim* η	$\sigma(\gamma)/\sigma(f)$ α
^{232}Th	13.0	7.48	$<2\times10^{-4}$			
^{233}Pa	13.1	40.6	—			
^{233}U	12.7	46.2	534	2.52±0.03	2.28±0.02	.105±.007
^{235}U	10±2	112±10	582±10	2.47±0.03	2.07±0.02	.192±.007
^{238}U	8.3±0.2	2.75±0.04	$<5\times10^{-4}$			
^{239}Pu	9.67±0.5	285±13	740±9	2.91±0.04	2.09±0.02	0.39±0.03
^{209}Bi	9.37	0.034	—			
Nat. Pb	13.0	0.17	—			

Many more reactions occur because of the intense neutron flux and of natural decays. The chain of possible reactions starting from initial ^{232}Th fuel is displayed in Fig. 3. The situation is sufficiently complex to justify a computer simulation which results are shown in Fig. 4. One can see the breeding element ^{233}Pa, the mixture of Uranium isotopes and a tiny fraction of higher actinides ^{237}Np and ^{238}Pu.

The last two elements are the only true "ashes" of the combustion, since the Uranium isotopes are the "seeds" for further utilisation. The graph full scale in the abscissa corresponds to about 10 years of continuous exposure. After a first initial phase of "breeding" in which ^{233}U accumulates to the equilibrium ratio, a steady situation sets in where fission and breeding are both working. Additional elements are formed, which are in general burnt by the neutrons and therefore reach an equilibrium concentration. A significant concentration of ^{234}U develops, which has a significant probability to be transformed in the highly fissile ^{235}U. Captures of ^{233}U which do not lead to fissions at once (≈ 10 %) are still used to produce energy since they are "bred" into fertile ^{235}U via ^{234}U. This secondary breeding process which resembles reaction {1} — except that it is totally driven by neutron captures—has an additional contribution onto the neutron inventory, since to transform ^{234}U into ^{235}U one neutron is required, while

^{235}U fission gives about 2.5 new neutrons. This isotope can in turn miss fission and instead capture another neutron, leading to ^{236}U. The next element to be formed is ^{237}U, which has a short lifetime (6.75 days) and decays into ^{237}Np which is long lived. Another neutron capture and the ^{237}Np is incinerated into ^{238}Pu, which has the moderate lifetime of 87.7 years for α-decay into ^{234}U. If left for a long time inside the fuel, ^{238}Pu will capture another neutron with a large cross section, thus giving rise to readily fissile ^{239}Pu.

At neutron fluxes of the order of $10^{14}\,cm^{-2}\,s^{-1}$ these many additional steps have an increasingly smaller probability to occur and "ashes" remain primarily Uranium isotopes. As has already been pointed out, they have the important function of ensuring that a simple chemical separation cannot yield a significant amount of fuel for military applications. The accumulation of Actinides other than the Uranium "seeds" constitutes no problem even after several seed recoveries and utilisation as shown in Fig. 5. In general we expect that they are separated from the Uranium at each re-utilisation cycle and stored or incinerated.

The amount of delivered power depends linearly on the neutron flux, which is not uniform inside the active volume. It is therefore useful to speak of an "average" neutron exposure Φ_{ave}. The total thermal power produced by fissions in a mass M of fuel at breeding equilibrium and neutron temperature T is given by:

$$P = 55.3\left(\frac{M}{1\,Ton}\right)\left(\frac{\Phi_{ave}}{10^{14}\,cm^{-2}\,s^{-1}}\right)\left(\frac{300K}{T(deg\,K)}\right)^{1/2} \quad Mwatt$$

As an example, setting M= 4.92 tons, $\Phi_{ave} = 1.50\times10^{14}\,cm^{-2}\,s^{-1}$ and T= 400°C, we find 267 MW. If these steady conditions are maintained uninterrupted for four years, the integrated neutron flux in the fuel will be $8.80 \times 10^{21}\,cm^{-2}$, which gives a conservative figure for the allowed integrated flux. During this period about 6.1 % of the Thorium fuel will be burnt, corresponding to a mass of about 300 kg. One ton of fuel corresponds to 2.8 million metric tons of Coal. All along stationary conditions the amount of fissile ^{233}U is of the order of 67 kg, which means that the fissile fuel is fully bred slightly faster than annually.

2.3. *The Uranium alternative.* The main thrust of the present method is the possibility of burning Thorium in conditions which are free of higher Actinides and in particular of Plutonium. But Uranium can also be used with the help of the breeding reaction {2}. Using the basic breeding equations and the experimental data from Table 1 we find for the $^{238}U-^{239}Pu$ mixture at the

breeding equilibrium the remarkably small number, $n_3/n_1 = 2.85 \times 10^{-3}$. This means that in regime conditions, the reaction can be sustained indefinitely by ^{239}Pu in equilibrium at the tiny concentration of 2.85 kg/ton of Uranium. We remark that this inventory is much smaller than the amount of Plutonium needed in an ordinary Fast Breeder Reactor and that no handling of the Plutonium is needed since it is bred "in situ" from ^{238}U.

When compared to a Reactor, the present burning-breeding process with Energy Amplification makes use of the dominant isotope ^{238}U rather than of the fissile ^{235}U naturally present or enriched, leading to a much better utilisation of the fuel material. Of course the present breeding regime is impossible in a Reactor, since it relates only to a beam driven sub-critical condition. A fraction of the naturally present ^{235}U can conveniently act as a "starter". It is quickly burnt away and replaced by a smaller quantity of highly fissionable ^{239}Pu.

Similarly to the case of Thorium, a large number of different nuclei can be produced by multiple neutron captures and eventually decays. They are illustrated in Fig. 6. The full time dependence of an initially slightly depleted Uranium fuel is given in Fig. 7. A fraction of naturally present ^{235}U has been preserved since it can conveniently act as natural "starter". One can see that when compared to the corresponding Thorium case there is a much larger production of higher actinides, but that their concentration is held to a constant value because of the high "incineration" capability of the scheme. From concentrations one can calculate $\alpha = 1.694$, which is slightly larger than the case of $^{232}Th-^{233}U$ mixture, where we have $\alpha = 1.223$. Consequently the (target-averaged) value of $\eta = \nu/(1+\alpha)$ gives the number of secondary neutrons resulting from one neutron interacting and it is slightly smaller, (i.e. 1.08 vs. 1.13), which is detrimental but not unacceptable to the gain. They in turn have a larger capture cross section and are more heavily affecting the neutron inventory. It should also be noted that the effects of fission fragments are now more significant, since the "breeding" cross section is about a factor two smaller. Therefore the performance of Uranium is slightly worse as the one of Thorium: a smaller gain is expected and a more frequent reprocessing is required in order to remove fission fragment "poisoning", but not more frequently than in an ordinary Nuclear Reactor.

As in the case of Thorium, if the neutron flux is suddenly cut off, there is an increase in criticality due to fact that all ^{239}Np nuclei will decay into ^{239}Pu, leading to an increase of the final population of ^{239}Pu. Such an increase in fissionable material must not make the system critical. This condition, as has

already been pointed out sets a limit for the neutron flux $\Phi \leq 9.84 \times 10^{14}$ $[T/(300°K)]^{1/2}\,cm^{-2}\,s^{-1}$, which is less demanding than the corresponding limit for $^{232}Th-^{233}U$ which was $\Phi \leq 1.44 \times 10^{14}$ $[T/(300°K)]^{1/2}\,cm^{-2}\,s^{-1}$. However the lifetime of ^{239}Np being shorter than the one of ^{233}Pa the build up occurs somewhat faster (days instead of weeks).

2.4. *Fission fragments.* The combustion of an appreciable quantity of fuel will produce fission fragments which in turn have a significant effective capture cross section for thermal neutrons. This is one of the most important questions of this scheme, since it is intimately tied to the problem of how long the reaction can continue without reprocessing. As has already been pointed out, when compared to a nuclear Reactor, in the Energy Amplifier these effects are less relevant since now one does not have to maintain criticality. An estimate of the effects due to the fission fragments is a major task, since there are very many nuclei and complicated decay chains. It is only possible to give results with numerical calculations based on available cross sections and thermal neutron captures. Epi-thermal neutrons are expected to contribute only slightly, since resonances in medium Z nuclei are generally well above these energies. It is therefore believed that these calculations can provide a reasonably accurate assessment of the situation. There are three main effects which must be considered:

1) Fission fragments may capture some of the neutrons, thus affecting the neutron inventory and consequently the energy amplification of the block. There is a complicated interplay between neutron captures and decays which both lead to transformations of the hundreds of compounds resulting from the fission. Hence the evolution is dependent on the neutron flux and more generally on the past history of the fuel. This is a well known effect in Nuclear Reactors. It should also be noted that fission fragment poisoning is less important in the case of Thorium in breeding equilibrium than for instance in Natural Uranium since the fuel cross section is 2.17 times larger and therefore the number of captures for a given fission fragment concentration correspondingly smaller. Computational results using known cross sections are shown in Fig. 8 for reasonable neutron fluxes and with the effective total cross section parametrized in the form already used in Table 1, namely $[\sigma(E) = (0.025\,eV/E)^{1/2}\Sigma]$. Two curves correspond to the conditions of Fig. 4 and Fig. 5 respectively. Known cross sections and decay rates for 1170 different nuclear fragments have been added, taking into account the time evolution. Resonances for the medium Z nuclei from fission

fragments occur generally at higher energies than in actinides and have a smaller contribution on the rate. The dependence of the effect of fission poisoning is rising less rapidly than linearly with the integrated irradiation $\int\Phi\, dt$ since there are both saturating (like for instance the well known ^{135}Xe and ^{149}Sm) and non-saturating fission fragments.

2) Fission fragments are generally radioactive and produce additional heat, even if the proton beam is switched off, since they contribute with about 14 MeV to the total energy emitted by fission which is (204±7) MeV. Immediately after turn-off of the proton beam, the power produced by this residual activity is 14/204 = 0.0686 of the steady condition. Activity decays slowly with time, approximately as $t^{-0.20}$, where t is in seconds, leading to a reduction of a factor 10 in about one day. Continued cooling must of course ensure that no melt-down occurs. In this respect, the Energy Amplifier does not differ substantially from a Reactor.

3) Long lasting radioactivity is a significant problem, although much less relevant in the absence of Actinides. The time evolution of the fragments after shutdown and separation is shown in Fig. 9. If normalised to Uranium ores for an equivalent energy delivery, their relative toxicity will decrease inversely proportionally to the number of recycles. One can see that after about 300 years the toxicity content is inferior to the one of natural uranium ores for an equivalent energy supply and becomes totally negligible a few hundred years later.

In practice, in order to maintain a constant behaviour during the utilisation of the fuel, one could introduce initially some neutron absorbing elements, which are either progressively retracted or eventually burned out into less absorbing elements during the combustion process. Anyway, such an evolution is not a problem, since it occurs over a very long period of time and it only decreases the reactivity of the system.

2.5. *A Thorium based Reactor?* As pointed out Thorium as a nuclear fuel has considerable advantages when compared with Uranium. However the realisation of a classic Reactor based on full breeding of Thorium presents serious difficulties—which will be briefly illustrated, and which justify the added complexity of an external neutron source according to the present scheme.

In a Reactor, the neutron flux is sustained fully by the neutron multiplication process, which is fission driven. A key parameter is the effective

multiplication factor, k_{eff}, the ratio of the neutrons at the end of a generation to the number starting that generation. It is obvious that for a critical Reactor, $k_{eff} = 1$. One can separate out the effects due to neutron leakage and introduce the corresponding parameter $k_\infty = k_{eff} / P$, which is the parameter which would apply to an (homogeneous) identical lattice arrangement in which the dimensions are large enough as to make the neutron leakage probability 1-P negligibly small. Evidently k_∞ must be significantly larger than 1 in order to permit to reach criticality with a sufficiently small lattice volume. The feature of k_∞ is that it does not depend on the geometrical dimension of the device, which relates to P: the excess of k_∞ from unity represents the allowance for the fractional neutron losses due for instance to leakage.

In the case of theoretically pure natural Uranium and graphite the maximum possible value of k_∞ is ≈ 1.1, while when heavy water (D_2O) is used as a moderator, again with natural Uranium, higher k_∞ constants, approaching 1.3 are attainable. As well known, this leaves sufficient room for losses due to leakage and absorption by impurities and by fission fragments to realise practical devices. However for a Thorium based device the situation is not so favourable.

The value of k_∞ is directly related to the concentration of fissile material. While for natural Uranium the relevant concentration is the one ^{235}U which is fixed and known (0.71 %), in the case of Thorium breeding ^{233}U, the equilibrium concentration of this last material is dependent on the previous intensity and history of the local neutron flux. As pointed out, in steady conditions and after an appropriate period of time, such a concentration reaches an equilibrium level in which the ^{233}U, which fissions, is balanced by the amount of ^{233}U bred from ^{232}Th. Such equilibrium concentration is also dependent of the energy spectrum of the captured neutrons—which in turn is related to the basic geometry of the lattice. In addition to ^{233}U, several other Uranium isotopes and other actinides are inevitably formed with a variety of time constants, which also reach eventually an equilibrium level. They also contribute both to neutron captures and multiplication with fissions and in a minor extent with (n, 2n) reactions.

"Reactor grade" Heavy Water (D_2O+0.14% H_2O), Beryllium, Water (H_2O) and Graphite are amongst the best neutron moderators. More generally, a large number of moderators can be used and ample choices are available, this being of course application-dependent and dictated by requirements of the specific design. The moderator must be sufficient to reduce the energy of fission neutrons, since

as pointed out, at lower energies the amount of ^{233}U needed to reach breeding equilibrium is smaller.

A number of practical lattice geometries, with fuel bodies of a number of different shapes and dimension (like fuel spheres or fuel bars) properly spaced and uniformly distributed in an essentially continuous moderating medium have been evaluated with computational methods in which the best knowledge of the underlying nuclear physics has been used. The relevant parameters are the bar or sphere radius r and the ρ volume ratio of the fuel to the moderator. The results can be expressed in terms of iso-k_∞ curves as a function of the variables r and ρ. A clear optimum emerges for an appropriate broad choice of these parameters. The initial fuel can be in various chemical forms, for instance of metal, oxide or carbide. The resulting values of k_∞ for optimal choices of r and ρ have been evaluated and given in Table 2.

Results are given for Graphite, Beryllium, Water(H_2O) and Reactor grade Heavy Water (D_2O). A special case relates to Beryllium, which has a significant cross section for (n, 2n) reaction and hence it acts effectively as a neutron multiplier. However interactions with fission neutrons produce as well ^6Li by the Be(n, α) reaction which has a cross section for thermal neutrons (0.025 eV) of 940 barns and which reaches saturation very quickly, offsetting the benefits of the (n, 2n) reaction. In addition neutron captures of ^6Li produce a large amount of Tritium which is radioactive and must be disposed of.

Table 2: Properties of some infinite Reactor lattice geometries.

Initial fuel composition	*Moderator composition*	*Fuel geometry*	*Maximum reactivity (theor.)*
Pure Thorium	Graphite	Spheres, Rods	$k_\infty = 1.07$
Pure Thorium	Water	Spheres, Rods	$k_\infty = 1.07$
Pure Thorium	Beryllium	Spheres, Rods	$k_\infty = 1.36$
Pure Thorium	Beryllium+^6Li	Spheres, Rods	$k_\infty = 1.08$
Pure Thorium	Heavy Water	Spheres, Rods	$k_\infty = 1.10$

The values of k_∞ include only the contributions to the captures in the moderators and are given for a $^{232}Th-^{233}U$ mixture at the breeding equilibrium. Additional captures, with a corresponding loss of reactivity must be added:

1) Captures by the intermediate element ^{233}Pa whose concentration is proportional to the neutron flux. For $\Phi = 10^{14}\,cm^{-2}s^{-1}$ we find $\Delta k_\infty = -5.3 \times 10^{-2}$.

2) Captures in the rapidly saturating ^{135}Xe and ^{149}Sm fission fragments, slowly dependent on the neutron flux. For $\Phi = 10^{14}\,cm^{-2}s^{-1}$ we find $\Delta k_\infty = -2.0 \times 10^{-2}$.

3) Captures by the higher Uranium isotopes ^{234}U, ^{235}U, ^{236}U and ^{238}U, generated by multiple neutron captures. Their concentration depends on how long the fuel has been used. Note that chemical separation cannot separate them from the main ^{233}U fuel and that both fission and captures contribute with opposite signs to the reactivity. For the relatively long integrated neutron exposure $\int\Phi\, dt > 3 \times 10^{22}\,cm^{-2}$ at which concentrations reach about saturation the contribution is $\Delta k_\infty = -5.0 \times 10^{-2}$.

The largest value of $k_\infty - 1$ for optimum choice of parameters and including only effects (1) and (2) is typically in the range $0.01 \div 0.03$, namely much too small to ensure criticality for a finite size system and once other sources of captures due to impurities, slowly saturating and non saturating fission fragments have been taken into account. It should be pointed that k_∞ is significantly reduced also by effect (3), i.e. build-up of Uranium isotopes higher than ^{233}U as long as we wish to make an efficient use of the fuel without isotopic enrichment.

Hence in realistic conditions, a Thorium burning Reactor can hardly reach criticality with full breeding requirements. This is why, according to the present method, the addition of an external neutron source is determinant in order to provide the practical operability of Thorium related nuclear energy.

3. The external neutron supply.

3.1. Neutron multiplication. The external neutron supply removes the above mentioned limitations. This can be realised for instance by a high energy, high intensity proton accelerator, whose beam is striking a heavy metal target located in the central region of the lattice. While the initial sample of low energy neutrons is provided by the beam hitting the target, major multiplication of this sample is naturally generated by the fissions in the fuel elements. For N_1 carriers in the first generation injected by the external neutron source, there will be $N_n = N_1 k^{(n-1)}$ carriers in the n-th generation, with k being the already defined

effective multiplication factor or criticality factor. Of course, in order to avoid criticality, $k < 1$. The total number of neutrons produced is then:

$$N_{tot} = N_1 \sum_{n=1}^{n=\infty} k^{(n-1)} = N_1\left(1 + k + k^2 + k^3 + \ldots\right) = \frac{N_1}{1-k}$$

with an enhancement factor $1/(1-k)$. The criticality factor has been already decomposed as $k = Pk_\infty$, where k_∞ relates to an infinite lattice and P the probability that a neutron does not escape and consequently reacts in the fuel. Note that criticality ($k = 1$) is avoided easily, since as already pointed out for a Thorium breeder device $k_\infty = 1.0$. On the other hand k must be large to get a good multiplication. We need therefore a neutron retaining geometry, namely a large value for P in order to ensure that the probability of further fissions remains significant and the cascade from an incoming neutron continues for several generations.

Practical examples indicate that a value of k within the range $0.9 \div 0.95$ is optimal, corresponding to a total number of neutrons in the moderator-fuel which is $10 \div 20$ times the number injected by the target. Clearly the success of the scheme depends on the healthy development of the nuclear cascade, which is most productive in the energy range from thermal energies to a few MeV. The relevant parameter is the rate of fissions, which ensures the continuation of the cascade with newly produced neutrons and it is the main source of energy production. The neutron yield from high energy protons on a massive target made of high Z material—demonstrated by practical Spallation Sources—is quite large. For instance a possible choice of dimensions and of target composition—described later on—will lead to an average neutron yield of about 42 neutrons for each incident proton of 1.5 GeV kinetic energy. Therefore the beam energy fraction required to produce one neutron with the help of the high energy cascade alone is of the order of ε_n = 35 MeV. The subsequent neutron multiplication in the cascade due to fissions is important since it further reduces the energetic "cost" of a neutron in terms of the incident proton beam energy by the multiplicative factor (1-k) bringing it to as little $1.75 \div 3.5$ MeV/per neutron. Admittedly, the choice of this range for k is rather conservative, and perhaps even higher gains can be sustained in a well designed device. As a comparison, the fission energy yield is about ε_f = 190 MeV (β-decays from fission fragments are included, but neutrinos are excluded).

3.2. Neutron conservation conditions. The energetic gain of the Energy Amplifier is denoted by G and it is defined as the ratio between the total energy

produced in the device and the energy deposited by the high energy beam. In order to give a first estimate of G, one has to take into account that in equilibrium conditions and an infinite lattice, about 40% of all the neutrons produce fissions, the rest being dedicated to breeding or captured in the moderator, fission fragments etc. Hence the net energy gain of the device is approximately given by $G = 190\ \text{MeV}/(35\ \text{MeV})\times 0.4\times 1/(1-k) = 2.17/(1-k)$ and it will generally fall between $G = 22$ and $G = 43$. Even far from criticality the energy gain is considerable. Assume for instance that the efficiency of the heat to electricity conversion, using a high temperature gas turbine is 0.45. The electric power produced will then be 10.2 (20.4) times the energy deposited by the high energy beam for $k = 0.9$ (0.95). There is plenty of electric power produced, in excess of what needed to run the accelerator.

The neutron conservation equation relates directly and on general ground the attainable energy gain G and $\Gamma = n_{loss}/n_0$, where n_0 is the number of neutrons absorbed in the fuel mixture of actinides, and n_{loss} are the neutrons escaping the system or absorbed elsewhere.

$$\Gamma = a + \frac{b}{G} \qquad\qquad G = \frac{b}{\Gamma - a}$$

where a and b are parameters functions of ν, the neutron multiplicity of fission and of α, the ratio of (n, γ) and fission cross sections, weighted by the fraction f_i of atoms of all Actinides of the fuel and averaged over the neutron energy spectrum:

$$\bar{\alpha} = \frac{\sum f_i\, \bar{\sigma}^i_{(n,\gamma)}}{\sum f_i\, \bar{\sigma}^i_{fiss}} \qquad a = \frac{\nu}{1+\bar{\alpha}} - 1 \qquad b = \frac{\varepsilon_f}{\varepsilon_n}\frac{1}{1+\bar{\alpha}}$$

Note that $\Gamma = a$ for $G = \infty$, namely when the device becomes critical. Therefore, in first approximation, $a \approx k-1$. The contribution to Γ due to the beam is given by the second term b/G. Exemplified figures appropriate to a practical device operating on thermal and epi-thermal neutrons are $a = 0.070$ and $b = 2.52$. The computer modelling used to determine a and b takes into account the actual energy spectrum of the neutrons and of the energy dependence of the cross sections, particularly complicated in the resonance region. Using the most neutron-saving design possible, one might be able to achieve, in absence of fission fragment poisoning, $\Gamma_0 \approx 1.08$. Hence as already pointed out the criticality margin is $\Delta\Gamma = \Gamma - \Gamma_0 \approx -0.01$. (A negative value does not permit a critical operation of the device). But in the same conditions and the already quite

practical gain G = 20 we find $\Delta\Gamma$ = 0.116 providing a considerable reactivity allowance. For a gain G = 40, $\Delta\Gamma$ = 0.053. Therefore Γ must be kept reasonably small in order to ensure the highest gain. The design of the device is therefore driven primarily by neutron economics.

3.3. Effects of fission fragments. The reward for minimising Γ_0 is a higher energy gain and wider allowance for captures due to fission fragments, which in turn means a longer lifetime without their extraction. The (normalised) cross section for fission fragments Σ_{frag} is given in the previously described Fig. 8. A good estimate after an integrated flux of $10^{22}\,cm^{-2}$ is Σ_{frag}= 1.6 barn from which we calculate a contribution $\Delta\Gamma$ = 0.106. Evidently shorter exposures and/or smaller gains will improve the margin on $\Delta\Gamma$.

The build up of fission fragments has a direct influence on the gain, unless corrected downward during the early phases by introducing a variable amount of neutron absorption by other means. These extra neutrons do not have to be necessarily wasted in control bars. They could be used for instance to breed new fuel for further use.

3.4. Computing methods. Computer simulation with the Montecarlo method on a practical geometry have been performed using a specially written cascade evolution programme. They represent a very realistic simulation since relevant cross sections are well known and introduced in the calculations. Hence they are a valid guidance in assessing feasibility and in optimising the geometry of the device.

Let us consider a high energy (1.0 ÷ 1.5 GeV) proton beam on a target containing a fissionable material. The high energy (> few MeV) cascade can be easily studied with the help of FLUKA [11]. However as soon as the neutron energy falls below few MeV the approximations in FLUKA are rather severe and the cross section behaviour rather primitive. In particular FLUKA is not designed to handle thermalization correctly, since neutron energies are very roughly quantified. If the energy loss in a collision is smaller than the quantization step, it remains unaccounted. Furthermore the cross section for neutron induced reactions in the resonance region have a very complex (peaks and valleys) behaviour, very poorly parametrized by its average value. Hence the calculations of FLUKA have been stopped somewhat arbitrarily when the energy of neutron falls below 1 MeV and a new computing method is used. It has been necessary to

write a new Montecarlo in which the best information on neutron cross sections is used and the possibility of a continuous variation of the neutron energies is allowed. Thermal movement of the target nuclei and resonance smearing are also accounted for. Cross sections from IAEA [12] have been used. These data, which are believed to be the most complete, include full details of the resonance behaviour. A total of over 1.5×10^5 cross section points have been introduced to describe the cross section behaviour just for the few types of nuclei of the calorimeter.

4. The high energy beam.

The success of the scheme hence relies on the injection of a large number of neutrons from outside. This is achieved with the help of a high energy beam—typically of protons—which initiates a neutron rich nuclear cascade which is capable of producing neutrons at a small energetic cost, namely a small ε_n. Two alternatives are possible. In the first alternative the beam is shot directly at the moderator-fuel mixture. Alternatively a dedicated target can be used to absorb the beam and to produce the neutrons. Such target must be in addition as transparent as possible to the low energy neutrons, in order not to affect the neutron multiplication due to fissions.

4.1. The fuel-moderator as high energy target. The possibility of sending the beam on the moderator itself is considered first, since it has the obvious advantage of simplicity. The spallation process provides neutrons in higher multiplicity for heavy nuclei than for light ones which are to be preferred for moderating the neutrons. In addition Thorium has a large high energy cross section for fission with large neutron multiplicity. It is therefore apparent that in order to have a copious neutron production by the cascade, the fractional amount of moderating material must be as small as possible.

Typically a 1.5 GeV proton incident on a large Thorium block will produce on the average some 70 neutrons, corresponding to $\varepsilon_n \approx 21$ MeV. This yield is approximately linearly energy dependent, leading to an almost constant value for ε_n. For instance at 800 MeV we find $\varepsilon_n \approx 26$ MeV. However the same 1.5 GeV proton incident for instance on Water or on Graphite will yield on the average only $5.0 \div 5.5$ neutrons. The moderator-fuel medium consisting of finely subdivided elements will yield an intermediate neutron rate, largely independent of the details of the geometry of the lattice. Indicating with V_m and

V_f the relative fractions of volume occupied by the moderator and the fuel, approximate yields for Water-Thorium mixtures are given by the expression (valid in the interval $V_m \leq 2\ V_f$): $\varepsilon_n(\text{MeV}) \approx 21.89 + 4.55\ V_m / V_f$ and $\varepsilon_n(\text{MeV}) \approx 26.82 + 5.29\ V_m / V_f$ for 1.5 GeV and 800 MeV respectively. We note that up to $V_m \leq V_f$ the value of ε_n is not very different than the one of pure Thorium. It is this fortunate circumstance that the present method is set up to exploit.

Very similar results for ε_n are found for a variety of low-Z moderator-thorium mixtures, provided for different materials 1 and 2 volumes $V_{m,1}$ and $V_{m,2}$ are compared in term of equal geometrical nuclear collision lengths L_{int}: $V_{m,1} = V_{m,2}\,(L_{int,2} / L_{int,1})^3$. In contrast, the moderating power, defined as the average logarithmic neutron energy loss—i.e. lethargy change—per unit of length, decreases very rapidly with increasing A of the moderating nucleus much faster than the geometrical nuclear collision length, as illustrated in Table 3.

Table 3: Some properties of some pure moderators for thermal neutrons and high energy (H. E.) protons.

Moderator	*Density* gr/cm^2	*H. E.nuclear interaction length, cm*	*Neutron Slow-down length, cm*	*Moderating power,* cm^{-1}
Water	1.0	84.9	5.74	1.53
Heavy Water	1.1	77.2	10.93	0.37
Beryllium	1.85	40.6	10.0	0.125
Graphite	1.80	47.9	19.7	0.064

As a consequence, for most moderators, like for instance Graphite, the requirements of an effective moderation and the one of a neutron efficient cascade cannot be simultaneously satisfied for a common value of V_m / V_f. The important exception is water, since hydrogen is extremely efficient in slowing down the neutrons. However in order to make it possible to use hydrogen as moderating element, we must also ensure that the fraction of neutrons captured by the well known radiative capture process (n, γ) is sufficiently small, in view of the stringent requirements imposed on the neutron inventory. We recall that a fully thermalizing water moderator like the ones used for instance in a standard PWR's will easily capture as many as 1/5 of all neutrons, a loss which is clearly incompatible with the present scheme.

4.2. The under-moderated water option. We have identified an alternative regime in which water maintains a small neutron capture rate while the other parameters, namely ε_n and the equilibrium ratio of the fissile material at breeding equilibrium n_3 / n_1 have both acceptable values, provided the neutron energy is kept significantly above thermalization with the help of under-moderation. In practice this is obtained choosing $0.2\ V_f \leq V_m \leq V_f$, as required by the high energy cascade. In other words, such under-moderation in a water lattice meets simultaneously both requirements of (1)-an efficient neutron production by the high energy beam and of (2)-a low neutron capture during moderation. In addition, the resulting neutron energy spectrum with energies significantly higher than thermal has other useful features: (1) the reactivity k_∞ is increased by the presence of a significant contribution due to high energy fissions in ^{233}Th, ^{234}U and ^{236}U (2) neutron losses in ^{233}Pa, ^{135}Xe and fuel cladding are reduced. Finally breeding performance is not impaired: notwithstanding there will be an increase of the relative fraction of neutron captures by Thorium due to the resonance region, which requires a corresponding significantly higher ^{233}U concentration n_3 / n_1 at breeding equilibrium.

Parameter dependence for an infinite lattice made of Water and Thorium are given in Figs. 10a-10d. Results are rather independent of the temperature (they have been calculated for pressurised water at 300°C), of the shape of the fuel elements (spheres, rods of radius r) and of their characteristic dimension r. They are given for an average performance over the interval $4.0\ mm \leq r \leq 2.0\ cm$ and they hold within such interval to within ± 5%. One remarks from Fig. 10d that the probability of radiative capture in the water moderator has become small or even negligible for $V_m / V_f \leq 1$. Typical conditions for $V_m / V_f = 0.4$ and $V_m / V_f = 0.8$ are listed in Table 4.

As already pointed out the U, Pa and Np concentrations used throughout are the ones corresponding to a very long exposure without isotopic separation, corresponding to asymptotic equilibrium conditions. Note however that the net effects on k_∞ of elements with $A \geq 234$ is extremely small, since the captures are almost completely compensated by the neutrons produced by fissions. The net neutron balance per generation is in fact essentially zero, $0.0323 \times 2.49 - 0.0841 = -0.00353$ for $V_m / V_f = 0.8$ and $0.0375 \times 2.49 - 0.0900 = +0.00259$ for $V_m / V_f = 0.4$.

Table 4: Typical parameters of an infinite, water under-moderated geometry

Fuel elements, shape		sphere, cylinder	sphere, cylinder
Fuel elements, radius, r		4 mm ÷ 2 cm	4 mm ÷ 2 cm
V_m / V_f		0.4	0.8
Average density (gr/ cm^3)		8.67	6.96
^{233}U eq. conc. (units of 1.3×10^{-2})		1.725	1.291
Long. beam (1.5 GeV) cont.(95%),m		1.11	1.34
Radial beam containment (95%), m		0.505	0.62
ε_n, MeV		23.5	25.2
$k_\infty - 1$		0.088	0.060
Captures, Moder.	$-H_2O$	0.00528	0.0216
Captures, Fuel	$-^{232}Th$	0.405	0.407
	$-^{233}U$	0.0473	0.0458
	$-^{233}Pa$	0.0152	0.0159
	$-^{234}U$	0.0584	0.0585
	$-^{235}U$	0.00715	0.00635
	$-^{236}U$	0.0231	0.0176
	$-^{237}Np$	0.00134	0.00163
Fissions, Fuel	$-^{232}Th$	0.0415	0.0237
	$-^{233}U$	0.351	0.365
	$-^{233}Pa$	0.00027	0.00016
	$-^{234}U$	0.0123	0.00529
	$-^{235}U$	0.0209	0.0251
	$-^{236}U$	0.00428	0.00196
	$-^{237}Np$	0.00009	0.00004

Water must be also sufficiently abundant in order to perform as well the important function of extracting the heat from the fuel-moderator block. Heat extraction is ultimately set by the well known "burn-out" condition which limits the energy which can be extracted from a given surface. In practice operating conditions must be kept many times below such a limit. In a water deficient arrangement, this problem can be overcome with an appropriately large contact surface to the liquid like for instance with cooling fins on the fuel cladding.

Above figures are for an infinitely large device. Control of the neutron losses due to leakage is largely a matter of cost. In order to improve containment one can add a reflector, although this is not a necessity. As a general rule an Energy Amplifier designed for an optimised cost these losses are likely to lead to a

criticality factor $k = (0.97 \div 0.95)\ k_\infty$. Once the inevitable losses in the fuel cladding, Xenon and other fragment poisoning etc. are added, the final value for k is likely to be only slightly below unity, namely the system is slightly sub critical.

4.3. Containment of the high energy cascade. Containment of the cascade due to proton beam must also be ensured. Fortunately this leads in general to dimensions which are comparable to the ones required by previously defined requirements for neutron containment. Assume that the beam hits an infinite fuel-moderator block. We define somewhat arbitrarily as "produced" neutrons all neutrons of the proton induced cascade as soon their energy drops below 1 MeV. Evidently they act as seed for a continuing cascade which is accounted by the multiplication process already discussed.

Containment of 95% of "produced" neutrons is ensured longitudinally and radially at depths in meters which are parametrized as $0.863 + 0.577\ V_m / V_f - 0.0366\ (V_m / V_f)^2$ and $0.431 + 0.223\ V_m / V_f - 0.0188\ (V_m / V_f)^2$ respectively. The shower size is only slightly smaller at 800 MeV. Hence the whole cascade can be conveniently contained in a cube of say at least one meter each side.

A conceptual diagram of the target geometry is shown in Fig. 11. Some of the produced neutrons suffer back-scattering and actually emerge in the backward cone as seen from the beam impact point on the target. In order to minimise such an effect the beam must penetrate by some $15 \div 20$ cm through a hole matched to the relatively small beam size. The proton beam which travels in the evacuated pipe has to penetrate in the Energy Amplifier though a thick window. This is not a problem as long as the window is relatively close to the moderator-fuel block (≈ 30-40 cm, depending on its thickness), since the few interactions occurring in it continue to propagate their secondaries through the block and have a comparable neutron yield. However some additional precautions have to be employed in shielding against the neutrons escaping out of the Energy Amplifier chamber through the beam pipe. This is done with the help of a long entrance collimator.

Finally the interactions initiated by the high energy proton beam produces in the target break-up of nuclei and of atoms. Nuclear break-up produces a series of nuclei, most of which are radioactive. The main effect of the atomic break-up is hydrolysis of the water moderator. Both effects have been considered in detail and appear to occur at quite acceptable level.

4.4. A separate, high energy target. In some instances, like for example in the case of a more bulky and less efficient moderator (Graphite), an independent target must supply the neutrons. The target must occupy the central part of the device to ensure the largest utilisation solid angle. The target material must be also as permeable as possible to low energy neutrons which can be back-scattered from the moderator and the fuel. We can therefore rely in its design on the experience gained for neutron spallation sources.

A number of possible geometries can be envisaged. The size of the target region (typically 30 cm radius, 1 m long cylinder) must be optimised in order to contain the largest fraction of the high energy cascade but to let evaporation neutrons emerge. The average energy of such neutrons is of the order of a few MeV. The simplest is a homogeneous volume rich of heavy material, typically either natural Lead, Bismuth or a (eutectic) mixing of the two. The choice of Pb-Bi mixture, or pure Pb or rather than other materials like Tungsten or Uranium is justified by the main requirement—already mentioned—that the target must be as transparent as possible to low energy neutrons. Indeed in the inventory of practical nuclei with high Z only ^{209}Bi, ^{206}Pb and ^{208}Pb exhibit a negligibly small (< 0.03 barn @ 0.025 eV) capture cross section for thermal and epi-thermal neutrons. While natural Bismuth is a pure isotope, Lead is a mixture of many isotopes and its capture cross section is dominated by ^{207}Pb (< 0.70 barn @ 0.025 eV) which has an abundance of 22.6%.

In the development of the cascade we can ideally distinguish two phases: a first phase in which the high energy particle produces a number of secondaries and a neutron multiplication phase due to the inelastic collisions in a high-Z medium. In more sophisticated designs these two phases could be realised with separately optimised materials. For simplicity we have taken a single uniform volume. We expect a neutron yield of about 42 neutrons for incident proton of 1.5 GeV. Therefore the energy required to produce each neutron is $\varepsilon_n \approx 35$ MeV, significantly higher than the value $\varepsilon_n \approx 21$ MeV in the case of Thorium.

In practice the target must be liquid, since both Pb and Bi have a very low thermal conductivity and one has to rely on convection to extract the heat produced by the nuclear interactions. Fortunately the melting point of Pb is 327°C and the one of Bi 271°C. Pb and Bi can be mixed in eutectic mixture which have even a lower melting point. From the point of view of neutron transparency Bi is of course largely preferable. However it has also some disadvantages. On freezing it expands of 3.3% of volume and it is highly corrosive. Neutron captures lead to

^{210}Bi (Radium-E) and this is a β-emitter of 5 days half life decaying on ^{210}Po which is an α-emitter with a half life of 134 days and which is highly toxic and difficult to contain. Nevertheless these problems appear soluble but require a rigorous containment of the molten metal. The best containing materials for liquid Bi or Pb are chrome steels. Mass transfers, which become important at high temperatures, around 550°C, can be controlled by the addition of tiny amounts (few hundred p. p. m.) of zirconium and magnesium to the liquid metal.

We visualise therefore that both the target and the fuel will be contained in sealed elements of similar design, but with different filling, for instance rods or spheres or other suitable geometrical shapes. The same cooling circuit can then be used to remove heat from both units. The simplest case is the one of cooling by gas (Helium, CO_2 etc.) since then the interaction probability of the high energy cascade in it as well as the neutron absorption probability are negligible. The target elements will then periodically removed and reconditioned just like the fuel. The structure of the fuel element assembly must be capable of withstanding the volume changes of the target material on melting.

A conceptual diagram of the target geometry is shown in Fig. 12a. The alternative of gas cooling is shown for definiteness. The density of high temperature but compressed Helium, CO_2 or other suitable gaseous coolant is sufficiently small that one can let the proton beam travel safely in it. Consequently the beam window to the vacuum of the accelerator can be conveniently installed outside the vessel. The "cold" gaseous coolant circulates (as indicated by the arrows in Fig. 12a) through a heat shield and it traverses the fertile core and the target region before exiting in. The beam reaches the target region through a cleared region, filled by the coolant gas. Both the fertile core region and the target region are made by the suitable number of fuel elements, shown in the details (Fig. 12b). We have chosen fuel pebbles in the exemplification: they are sealed by the appropriate cladding (Zircalloy, Steel or other suitable material of low neutron absorption and good mechanical properties).

(1) in the target region there is no moderating material and the liquefied metal, either Pb or Bi or the eutectic mixture of the two fills as much as possible the available space and a small space is left for the expansion from solid to liquid. Note that the eutectic mixture Bi-Pb at 58% melts already at 125°C and shows no appreciable contraction at solidification.

(2) in the fuel-moderator region spheres are made of a central core of fuel, surrounded by a Graphite moderator.

Care must be exercised in the geometry to make the coolant to travel in (curved) paths which are not such to permit a significant fraction of the proton beam to miss the target. The proton beam travels in an evacuated tube up to the window. A thick collimator is necessary to reduce the neutron flux escaping through the beam porthole.

Just like in the case of the fuel moderator as target, the structural material used to contain the liquid metal participates to the high energy cascade. However the behaviour of materials like Zircalloy, Steel etc. to high energy protons is not as different as the one of Thorium as in the case of water moderator. For instance if the target will be made of solid Zirconium we would expect $\varepsilon_n \approx 70$ MeV. Hence, even if a relatively large fraction of the weight of the target is structural material, only modest effects are expected on ε_n.

4.5. Fragments from high energy collisions. The interactions of the high energy beam will produce a large number of different nuclei because of spallation and other inelastic nuclear collisions. Most of these products are radioactive and they must be contained just like fission products. Fortunately the amount of these products is relatively modest when compared with fission. The presence of a relatively large non fissionable target in the middle of the moderator-fuel medium with its own accumulation of reaction products is inevitably reducing the reactivity of the system. A first order estimate of the effect gives $\Delta k = -(1.0 \div 2.0) \times 10^{-2}$. Although the parameters of the container are application dependent, a significant additional loss in reactivity must be accounted for.

4.6. Conclusions. A separate target leads to significant reduction of the neutron yield (due to the poorer performance of Pb and Bi when compared to Th) and a significant reduction of reactivity due to additional neutron captures in the non fertile materials. However it opens the way to the possibility of using a variety of moderators, like for instance Graphite, and consequently of operating the device at higher temperatures than what is possible with water. Higher temperatures permit to increase the efficiency of the conversion to electricity and consequently at least partially to offset such shortcomings.

To conclude, the efficient use of the fuel-moderator material as direct high energy target implies that neutrons remain under-moderated. These neutrons of energies substantially higher than thermal have the drawback of requiring a

much higher fissile material concentration at equilibrium. In the case of Water which has a very high moderating power we indicate a compromise situation in which the fissile concentration is only slightly higher than the thermal case and the target efficiency is high. Other schemes are of course possible with less efficient moderating medium or with no moderator at all, but at expense of the much larger amount of fissile material.

In the alternative case in which the target has a different topology than the fuel-moderator medium, the high neutron yield and the transparency to neutrons must be ensured at all relevant energies, for the target, the associated cooling medium and the related containment hardware. Gaseous coolant has the interesting feature that it is essentially transparent to neutrons. Other liquid media are of course possible. Since they do not have to moderate neutrons inside the high energy target, the many coolants already used in fast neutron reactors can be identified, the choice amongst them of course largely dependent on the specific application.

The target and fuel-moderator configuration described for Thorium are applicable also to the case of the (depleted) Uranium. In particular we have verified that both schemes of the fuel-moderator as high energy target (under-moderated water moderation) and the separate target configuration can be readily extended to the present case.

5. The high energy particle accelerator.

5.1. Introduction. The purpose of the accelerator is the one of producing most efficiently the largest number of secondary neutrons by collisions between the beam and a solid target. As already pointed out there is a large independence of the energy and nature of the incoming beam. For instance, using a deuteron rather than proton beam will enhance the neutron yield by some 10%. In the following, for simplicity, protons are chosen. The energy of the incoming protons is not critical and any value in a wide interval, down several hundred MeV, gives comparable performance and a neutron yield proportional to the beam energy. The accelerator must also be energetically efficient, namely the beam must carry the largest possible fraction of the energy required to operate it.

The accelerated average current i_{ave}, once its kinetic energy T is set, is proportional to the power P_{beam} required by the beam, which in turn is related to the power P delivered by the Energy Amplifier and the gain G:

$$i_{ave} = \frac{1P(\text{Mwatt})}{GT(\text{GeV})}\ \text{mA}; \quad P_{beam} = \frac{1}{G}P$$

For the typical parameters G = 40, T = 0.8 GeV and P = 250 MW, we find i_{ave} = 7.18 mA and P_{beam} = 6.25 MW. Smaller devices will require a correspondingly smaller accelerated current. Accelerators of characteristics close to the ones here required have been extensively used for Research purposes and with the experience existing in the field there is no reason to consider its construction or operation particularly delicate or complicated. There are a number of possible technical choices in the design of the accelerator. Two possible schemes will be briefly outlined.

5.2. The LINAC accelerator. The accelerator chain is schematically shown in Fig. 13. It consists of a proton source and pre-injector, followed by a pre-accelerator which could be for instance a RFQ (Radio Frequency Quadrupole). The RFQ will bring the energy of the beam to about 2 MeV and will be followed by the intermediate accelerating structure which could be for instance an IT (Drift Tube LINAC) or other structure of similar performance. At the exit of the IT the beam which by now has about 25 MeV must be shaved by the collimator (in order to minimise the beam losses at high energy) before entering in the main accelerating section. Such a main LINAC section can either be normal or superconducting:

1) In the case of non superconducting accelerating cavities the relevant figure relates the power dissipated in the cavity (in the absence of beam). There is an advantage in pulsing the accelerator for periods longer than the filling time of the cavities (typically ≈ 50 μs), since at high currents the power delivered to the beam exceeds largely the one that is dissipated in the accelerating cavities. For instance, for a maximum peak current of 180 mA[1], corresponding to 150 MW peak power at 0.80 GeV the power dissipated in the copper of the cavities is only about 50 MW. The average power is of course smaller and controlled by the duty-cycle of the accelerator. For instance if the indicated current of i_{ave} = 7.18 mA is needed, the accelerator

[1] This seems quite acceptable, since there is no constraint to emittance.

will be pulsed by the modulators at the rate of few hundred pulses/sec in order to be on for 4.4% of the time. The corresponding average dissipation in the cavities will be about 2.3 MW. The (average) energy gain is 1.5 MeV/m, which leads to a rather long structure. Eventually, since the magnetic rigidity of the beam is relatively modest, 180° bending elements could be inserted along the structure to fold it into smaller longitudinal dimensions. The RF sources (klystron) have typically an efficiency of 70%. Several of such units with appropriate splitters feed the many cavity units. An over-all mains-to-beam efficiency of 50% is a reasonable figure.

2) Superconducting cavities have been developed for particle accelerators and they could be used for the present application. The advantage of superconductivity is a higher (peak) gradient – ≥ 6.0 MeV/m – which results in an accelerator of about 1/3 the length and in a better over-all mains-to-beam efficiency which will be in the neighbourhood of 60%. However, the complexity in operating a superconducting device—at least at today's state of the art—is somewhat more advanced and it is very likely that in simpler cases it might turn out to be excessively complicated. Notwithstanding the benefits of superconductivity for this application must be emphasised and simplicity of operation may be achieved in a not too distant future with further R&D work.

In all LINAC versions, transverse focusing must be ensured all along the accelerator and this is easily realised with the help of quadrupole doublets. The matching of the beam to the target is performed by the final focus quadrupole lenses and beam transport. Particular care must be exercised in limiting beam losses which produce activation of the accelerator structure. The accelerator complex could in principle feed more than one Energy Amplifier. This can be easily accomplished with electric or magnetic deflectors pulsed in synchronisation to the LINAC followed by classic septum magnets and beam transport elements which will distribute separate pulses in close succession to different targets.

5.3. *The Isochronous Cyclotron*. Such circular machines are capable of accelerating smaller but quite significant currents typically up to about 10 mA. When compared with the LINAC they have the advantages of the smaller size and for some configurations, of a smaller cost. Particularly interesting is the possibility of a FFAG (Fixed Field Alternating Gradient) accelerator in applications in which the beam power does not exceed several MW and for

energies of beam below 1 GeV. The main current limitation in a circular machine is due to transverse space charge effects and it occurs at low energies. This is why it is proposed to use the circular machine only after the beam has been accelerated to a substantial energy (for instance with a smaller LINAC and up to a value in the interval 50÷200 MeV, depending on the requirements of the final accelerated current), according to the scheme of Fig. 14. As in the case of a LINAC previously illustrated, it consists of a proton source and pre-injector, followed by a pre-accelerator which could be for instance a RFQ (Radio Frequency Quadrupole). The RFQ will accelerate the beam to about 2 MeV and will be followed by the intermediate accelerating structure which could be for instance an IT (Drift Tube LINAC) structure. At the exit of the IT the beam has about 25 MeV and it is shaved by the collimator before entering in a third accelerating section which brings the beam energy to the value required by the FFAG injection. The FFAG consists in a number of sector magnet units with a strong focusing gradient disposed in a circular geometry. The beam circulates in an evacuated chamber in the magnet gaps. The number of such sectors depends on the energy: for energies of 800 MeV about 8÷10 sectors are optional. The space made available between these sectors can be conveniently used for inserting RF accelerating Cavities and the extraction and injection channels. Particles are isochronous and the RF operates at a constant frequency, accelerating a continuous beam.

Extraction of the beam from the FFAG is a delicate operation since it has to be performed with high efficiency to avoid activation of the accelerator components.

The power consumption of the accelerator relates primarily to the one of the (big) magnets and to the RF. The efficiency of the present RF is quite comparable to the one previously discussed of a LINAC. The power consumption of magnet can be kept to reasonable level (1 ÷ 2 MW) by a conservative coil design. Alternatively a "super ferric" magnet in which the coil is made superconducting offers the possibility of significant power savings.

6. The initial fuel load.

In order to operate with a significant gain, the device must contain a reasonable amount of material which is fissile for thermal neutrons. The simplest approach consists of an intense beam irradiation prior to utilisation. Although conceptually simple, it is hardly likely that this will be so in practice

and the problem of how to "prime the pump" needs further consideration, even if it has to be done once in the lifetime of each specific application. We can list the following alternatives:

1) ^{233}U can be produced by inserting inside the moderator some additional Thorium in the early fuel life of a similar but already operational device. The neutron excess, destined to provide for the allowance of captures by fission fragments later on in the lifetime of the fuel can be dedicated to the breeding of new fuel. Probably as much as 10% of the neutron inventory could be used "parasitically" to that effect. Much greater yields can be achieved if one sacrifices to energy production, increasing the beam power and correspondingly reducing the gain with breeding captures. It is important that the added material is primarily used for breeding. Hence the fractional 'density of ^{233}U must be at all times much lower than the equilibrium level for stationary breeding. This implies that chemical purification is required to produce new fuel bars for the new power plant.

2) ^{235}U could be used as initial fuel instead of ^{233}U. Natural Uranium has the considerable inconvenience that inevitable captures of ^{238}U lead to a significant pile-up of Plutonium and minor actinides. It is therefore preferable to use highly enriched ^{235}U. This fuel can either be directly dissolved in the fuel or contained in some auxiliary elements, to be removed after start-up. The properties of ^{235}U are quite similar to the ones of ^{233}U and an initial load in weight of about 90% of the equilibrium level will suffice to ensure a smooth transition between start-up fuel and the self sufficient ^{232}Th—^{233}U cycle. Note that this procedure must be followed only once in the history of each installation.

3) Finally also ^{239}Pu can also be used as a starter. There is a large surplus in the world today and our device can act as an efficient incinerator. Note that the fission cross section is about twice as large as the one of ^{233}U and therefore the amount required is roughly one half, namely about 0.5 %. Surplus Plutonium from reactors, after it has decayed conveniently, can be also considered, although the presence of different isotopes must be examined, since they make it somewhat less efficient.

Once equilibrium conditions have been reached, combustion can start. This phase can last for a very long time (several years), the limit being the "poisoning" of the bars by the fission products and the consumption of a major fraction of Thorium fuel.

When the Thorium loading has been sufficiently used and fission fragment captures have reached the maximum acceptable level, a regeneration of the fuel is recommended. This must be done in a specialised centre. The fuel is chemically separated. The Uranium fuel is recovered while the other products (mostly fission fragments) are disposed. Such a fuel is then used to prepare new Thorium bars, in order to skip the initial breeding phase for the successive filling and to limit the stockpile of actinides. Initial fuel breeding must occur only once in the history of each power plant.

7. Some illustrative practical possibilities of Energy Amplifier.

7.1. A water cooled energy amplifier without separate target. This illustrative case exploits the features of under-moderation with water already described previously. The main parameters of the scheme are given in Table 4. The typical thermal power which can be most readily produced in this way is of the order of 200 MW. In order to ensure sufficient cooling of the fuel, especially for larger power, the choice $V_m / V_f = 0.8$ is most appropriate. Once the leakage of neutrons and the other losses in the fuel cladding etc. are taken into account the system will be sub critical with k= 0.92÷0.95, corresponding to an energy gain G = 33 ÷ 50. A good design value is therefore G = 40. The beam current at T =800 MeV is then i_{ave} = 6.25 mA and P_{beam} = 5.0 MW. Both a LINAC and a FFAG accelerator can easily satisfy such requirements.

The vast experience with Pressurised Water Reactors (PWR) can be used for the extraction of the heat. Of course other alternatives, like the Boiling Water mode can be used as well, the choice being determined by the type of application.

The operating pressure of the device (PWR) is of the order of 154 bar corresponding to an inlet temperature of 291°C and an outlet temperature of 322°C. The coolant flow for the nominal power output of 200 MWt is 1.1 m^3/sec. The required cooling transfer area inside the fuel core is approximately 300 m^2. Such a surface is provided even without cooling fins by cylindrical fuel rods of 2.5 (2.0) cm diameter or smaller and a fuel mass of at least 21.4 (17.1) tons.

The general layout is shown in Fig. 15. It consists of two main separate parts, the final Beam Transport and the main Energy Amplifier assembly. The proton beam travelling in vacuum from the accelerator is focused by the magnetic

quadrupoles lenses and deflected by 90 degrees with the help of the bending magnets. It enters in the pressurised vessel through the long Entrance Collimator which has at its top end the pressure retaining window. Several beam observation devices are used to follow the beam trajectory. Heavy shielding floor is ensuring radiation safety. Neutrons from the fuel-moderator assembly can escape through the beam pipe. They are considerably reduced by the collimator. The narrow pencil beam passing through the collimator is collected in the beam dump, since contrary to the proton beam, they remain undeflected by the bending magnet.

Energy is extracted with the help of Pressurised Water contained in the main vessel. The cooling fluid enters through the inlet nozzle and exits through the outlet nozzle. It passes first between the inner walls of the main vessel and the core support barrel. Its flow is uniformised by the flow skirt and enters in the inner volume of the core support barrel and in the fuel assembly from below. It traverses the many channels of the fuel assembly, extracting efficiently the heat there produced and exits from the outlet.

The upper part of the main vessel houses the support structure, the fuel handling equipment and a number of control bars, mainly to be used to secure firmly in the non critical condition the fuel, after switching the beam off. The need for such device is primarily due to the fact that the fissile ^{233}U accumulates after shut-off because of the decay of ^{233}Pa. During operation these control bars can also be conveniently used to trim the neutron multiplication parameter k and hence the gain of the Energy Amplifier.

A number of different fuel assembly can be used. Note that almost infinite variations are possible on fuel-moderator configurations. Two of such schemes, largely inspired to Reactor designs are listed below:

1) A fuel assembly is shown schematically in Fig. 16. The fuel consists in Thorium metal rods cladded with a thin Zircalloy sheet to prevent corrosion. They are grouped in subassemblies which constitute a rigid unit for easy handling. Note that metal could be replaced by pellets of ThO_2, ThC_2 or other chemically stable Thorium compound.

2) We mention the possibility of ThO_2 fuel and of spherical fuel pellets in a fluidized bed configuration. A fluidized bed is one in which a fluid flows upwards through a bed of solid particles which then become borne or fluidized but not transported or slurred. The bed is in a state of turbulence

and the solid particles are in constant motion, resulting in a good mixing and excellent heat transfer characteristics. A typical design (Fig. 17) is made of a cylindrical core with a perforated plate bottom strong enough to support the weight of the full fuel load and a flared top. The perforations should be such as prevent the fuel from falling thorough but they should permit the flow of the coolant without a too large pressure drop. The top is flared to reduce the speed of the flow and avoid that fuel elements can escape from the top. The cooling fluid (water) enters in and exits at the top on. Fuel elements are simple small spherical pellets that may be clad by coating or if ThO_2 is used without cladding. Problems of abrasions may arise and they must be studied, since they may affect the design of the primary cooling loop and require that the abraded, radioactive fuel is recovered safely. The maximum packing of the fuel in the collapsed state correspond to random packing of a very large number of spheres and it has a porosity (free volume filled with water/total volume) a = 0.40, corresponding to $V_m / V_f = 0.666$. However the flow of liquid will increase the porosity and consequently V_m / V_f. Control of the reactivity k can be easily achieved without control rods, by simply varying the coolant flow rate within the range of fluidization and therefore the fuel-moderator ratio V_m / V_f. This simplifies considerably the design of the pressure vessel. The simplicity of the fuel loading is also evident, since the continuous loading and unloading of the fuel is possible through small openings to the core.

The utilisation of the heat produced is of course application dependent. The pressurised heated water from the vessel must however retained in a closed loop and extracted for further use by a standard heat exchanger. In the most obvious application of the device, one has to operate one or more turbines, as schematically shown in Fig. 18. The water of the Energy Amplifier is pressurised by the device and it circulates with the help of the pump. The heat exchanger is used to transform water from another loop into steam which operates the turbine(s). A condenser and another pump closes the loop.

Common practice indicates that such an arrangement, vastly employed in power stations, can lead to a conversion efficiency into electricity slightly larger than 30%. About 60 MW of electricity can thus be produced with the exemplified parameters given above.

7.2. A gas cooled energy amplifier with a separate target. This illustrative case exploits the features of a separate high energy target previously described. Gas is preferred as a coolant, since, as already pointed out it is essentially transparent both to the incoming high energy beam and to neutrons. Following the current practice in Nuclear Reactors and in other similar applications, the best gas choices are (pressurised) Helium or CO_2 (or their mixture) because of their excellent thermodynamical properties and the absence of corrosive effects. As an exemplification we shall concentrate pressurised Helium—because of its high heat transfer, low pressure drop, high sound velocity and neutrality toward metals and graphite—but our considerations are valid also for other gases. The moderating medium, in view of its performance at high temperatures has been chosen to be Graphite. Many different lattice geometries are of course possible: we shall describe in detail the one based on spherical units, called "pebbles", each made of a central core of fuel material (Thorium metal or Thorium compound) of Fig. 12b surrounded by a Graphite shell. An optimisation of the relevant parameters (reactivity, ^{233}U concentration etc.) performed along the lines of the previous example of water moderation indicates that V_m / V_f must be in the range $10 \div 20$ for the best performance, namely $k_\infty = 1.04$ and $n_3 / n_1 = 1.7 \times 10^{-2}$. The ratio of the diameters of the external moderator spheres and of the fuel core are then $(V_m / V_f + 1)^{1/3} = 2.22 \div 2.75$. There is large freedom in the choice of the outer diameters of the "pebbles" typically of the order of several centimetres. The porosity of the fuel-moderator volume is the one of an assembly of very large number of randomly packed spheres, $a = 0.39$. Of course void spaces are traversed by the cooling gas and do not influence appreciably neutrons.

The geometry of the fuel-moderator and target follows closely the one described already in Figs. 12a and 12b to which we refer for details. The target "pebbles" are spheres filled with Bi-Pb or pure Bi metal, liquefied by the energy carried by the beam during operation. A specially cleared region permits the beam to penetrate deeply into the interacting volume. Special grids or other form of mechanical separators ensure that pebbles of different kind cannot mix. Suitable mechanisms permit to introduce new pebble or to extract them from the volume. A number of shut-off control bars must penetrate in the pebble assembly and they do so through a number of graphite tubing, which of course participate also to the moderation.

The present "atomic heater" can be used for a variety of practical applications in which high temperatures, up to and eventually above 1000°C, are needed. We concentrate on a scheme based on a closed cycle Helium gas turbine,

largely inspired from the closed-cycle fossil fuel fired power plants. Following the large experience of such devices, the typical thermal power which can be most readily produced in this way is up to the order of 200 MWt. A flow diagram as a general example which gives the main design data of a direct helium cycle is shown in Fig. 19. Different ways of arranging the machines are of course possible. Helium compressed at some 58 ata enters the Energy Amplifier at a temperature of 435°C and exits at 710°C with an estimated pressure drop through the reactor of ≈ 2 ata. The gas flow is of the order of 200 kg/s. It delivers its power via two turbines and in cascade. Heat resisting material needs to be used only for the high temperature turbine. A turbine expansion ratio of 2.5 to 3 offers good design conditions and a small number of stages(2). At the exit of the second turbine the gas has a temperature of 470°C and a pressure of 26 ata and enters the Recuperator to further reduce its temperature to 150°C. With the help of the cooler the gas temperature is reduced to 38°C (25 ata) and it enters a two stage compressor with an additional cooler between the two units. At the exit of the compressor (120°C, 60 ata) the gas is preheated in the recuperator to 435°C, (58 ata) and closes the loop entering the Energy Amplifier.

Adequate control is provided in order to ensure a constant efficiency over a wide power range. Make-up supplies are used to replenish gas losses from external reservoir. The overall full load efficiency is of the order of 40 %. About 80 MW of electricity can thus be produced by the generator with the exemplified parameters given above.

8. Some considerations on commercial energy supply

It is very complex at this stage, and out of the scope of the present paper, to establish a comparison between the estimated cost of the energy potentially produced by the present method and the one from other existing energy resources. Nevertheless several aspects should be noted, which initially favour the proposed Energy Amplifier as a future commercial source of energy.

1) At present the method to estimate the cost of energy is suffering important changes. Policy makers and economics agree that efforts should be done to integrate, in the real cost of energy production, the environmental impact, decommissioning, disposal of waste, etc. Our present proposal is advantageous since it combines the cleanness of nuclear power plants

regarding emission of carbon dioxide with a very important simplification of the waste disposal.

2) Compared to nuclear power plants fuel should be cheaper since Thorium is rather abundant. The mechanical elements should be easier to build, given the simplicity of the system, than those of a traditional power plant. Also security and safety aspects are much less demanding.

3) The system essentially uses well-known technologies, although its economic performance can improve by technological developments, mainly in the field of accelerators.

4) All the equipment to convert heat in electricity is conventional. If widely used, the Energy Amplifier will have an strong impact, among others, on industries involved in "post steam" operations.

To summarize, is it too early to envisage the commercial impact of the Energy Amplifier but if the concept is experimentally confirmed it has clear initial advantages: generates less pollution, simplifies the waste disposal, presents no military danger, it can be built in smaller units and uses well mastered technologies.

9. Summary

The practical possibility of extracting nuclear energy with the help of accelerator induced nuclear cascades has been described. The proposed energy amplifier is a "calorimeter", designed to extract the energy released by the breaking-up heavy nuclei inside a target, as a consequence of the large number of nuclear collisions initiated by the beam. A fraction of such energy in turn, is transformed into electricity used to run the accelerator, with the presumably remaining larger fraction delivered for external utilisation, either in the form of heat of electricity or both.

The present paper has a number of new concepts, of which the main points are summarised:

1) Realisation of the favourable stability conditions which lead to flux independent breeding processes of a fissile element from a fertile element of the fuel material. The fuel material is contained in an enclosure together

with a moderator medium and a beam of high energy particles is directed into the enclosure for interacting with heavy nuclei contained therein, so as to produce high energy neutrons. The neutrons thereby produced are slowed down by the moderator medium and multiplied in sub critical conditions by the breeding and fission processes.

2) Realisation that these conditions can lead to an intrinsically non-critical device and have a high efficiency of breeding, only provided the neutron flux is relatively low, in the $10^{14}\,cm^{-2}s^{-1}$ range. At these fluxes the power yield of the reactions is very large.

3) Realisation that the initial fuel load can be at different ratios between the concentrations of the fissile element and the fertile element. The ratio can be of about the stable value of the burning phase, either by adding fissile material to be removed when reaching the equilibrium value, or by adding the fissile element content recovered, by chemical separation, from another fuel material consumed in a previous similar energy production operation. The ratio can also be smaller than the stable value of the burning phase, in which case an initial breeding phase is carried out in order to reach the stable value.

4) Realisation that Thorium as breeding fuel has considerable advantages when compared with Uranium, in contrast with the fuel breeding based classic reactors for which the use of Thorium presents serious difficulties. Thorium is more abundant than Uranium, it generates little minor actinides among the radioactive waste and the risk of nuclear proliferation is negligible. For a given delivered energy, the quality of fission fragments is comparable to the one generated by conventional reactors. However, the toxicity of the fission fragments, although strong, is much more short lived than of actinides. It decays well bellow the toxicity level of natural Uranium ores, for an equivalent delivered energy, in a period of few hundred years.

5) Realisation that a performant energy amplification can be achieved by different "calorimeter" configurations. The beam can be directed to an enclosure containing a moderator medium and a fuel material including a fertile element. In such a case, which has an obvious simplicity, the fractional amount of moderating material must be as small as possible. Alternatively, the beam can be directed to an special separated target, capable of producing a large multiplicity of neutrons by spallation and having large transparency to thermal neutrons. In this case the target is located in the

enclosure and surrounded by the fuel material and the moderator medium. In the second configuration the moderator is arranged such as it results in complete thermalisation of the neutrons produced by the high energy cascade whereas the first configuration results in optimal production of neutrons at the cost of under-moderation of the neutrons produced in the cascade.

6) Realisation that the Energy Amplifier is largely independent of the energy and nature of the incoming beam. The particle accelerator needed to provide the external neutron source must be energetically efficient but has characteristics close to the ones extensively used for Research purposes. For simplicity a proton beam is proposed and two schemes, based on a linear accelerator and a sector focused cyclotron have been outlined.

Hopefully the verification of the principle of energy amplification by a high energy cascade will be done soon [13], in view of the potential long term importance of its application to energy supply, which is one of the main problems of today's society.

References

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[5] C.M. Van Atta, J. Lee and W. Heckrott. Lawrence Livermore Laboratory, *UCRL- 52144* (1976)

[6] C.M. Van Atta, "A brief history of MTA Project" , in Proceedings Inform. Meeting Accelerator Breeding, Jan 1977, BNL-770107, p. 20.

[7] P. R Tunnicliffe, B. G. Chidley and J. S. Fraser , "High current Accelerators and Nuclear Power" in Proceedings of 1976 proton Linear Accelerator Conference , 1976, AECL -5677

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[9] C. Rubbia, *"Inertial fusion- A contribution of Accelerator Technology to the Energy Problem ?"* in Proceedings of the Third EPAC Conference, Page 353, Berlin, March 1992, Published by Editions Frontières.

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[12] ENDF/B-6 in IAEA-NDS-100, Rev. 4, June 1992, Published by the International Atomic Energy Agency, Vienna, Editor: H.D. Lemmel. See also JAERI-M 90-099, June 1990. We are grateful to the IAEA for supplying the relative Data files to us and for considerable help in decoding them.

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Figure Captions

Fig. 1 General principle of power generation with the help of an energy amplifier. The Target has an energetic gain G which is converted into electricity with efficiency η_{te}. The power P produced in this way is partially recycled (fraction f) through the accelerator of efficiency η_d and re-injected as beam in the target, while the remaining fraction (1−f) P is delivered for further use. With the figures indicated, the range of such "useful" power is between 75% and 82% of the one produced by the converter.

Fig. 2 $^{238}U/^{232}Th$ equilibrium ratio as a function of the neutron energy.

Fig. 3 Chain of possible reactions starting from initial ^{232}Th fuel. Cross sections are for thermal neutrons in barns.

Fig. 4 Time evolution of the composition of an initial, thin Thorium slab exposed to a constant (thermal) neutron flux of $1.0 \times 10^{14}\ cm^{-2}\ s^{-1}$.

Fig. 5 Same as Fig. 4 except that now the terminal Uranium "seeds" are re-injected in the new Thorium fuel.

Fig. 6 Chain of possible reactions starting from initial ^{238}U fuel.

Fig. 7 Time evolution of the composition of an initially slightly depleted Uranium fuel.

Fig. 8 Thermal cross section due to fission fragment accumulations from ^{233}U fissions, as a function of the integrated flux and for a constant neutron flux of $1.0 \times 10^{14}\ cm^{-2}\ s^{-1}$.

Fig. 9 Expected toxicity from Fission fragments and for Actinides as a function of time, normalized to the one of Natural Uranium ores for an equivalent energy delivery. In the case of Actinides we have included all relevant elements (except the permanent Uranium seeds) at their "incineration' saturation level.

Fig. 10a ^{233}U concentration at breeding equilibrium as a function of R_V (R_V = Volume moderator/Volume fuel).

Fig. 10b Excess reactivity k_∞ as a function of R_V. The contribution due to neutron absorptions by the moderator and all actinides at their expected equilibrium are included.

Fig. 10c Rate of fission as a function of R_V.

Fig. 10d Rate of captures as a function of R_V.

Fig. 11 Schematic view of the target system.

Fig. 12a Schematic diagram of a separate high energy target.

Fig. 12b Scheme of the target and fuel spheres.

Fig. 13 The LINAC accelerator chain.

Fig. 14 Schematic view of the isochronous cyclotron.

Fig. 15 Schematic view of a beam driven liquid cooled energy amplifier without separate target.

Fig. 16 Schematic view of the fuel assembly.

Fig. 17 Schematic view of spherical fuel pellets in a fluidized bed configuration.

Fig. 18 Schema of the electricity generation system for a beam driven liquid cooled energy amplifier without separate target.

Fig. 19 Schema of the electricity generation system for a beam driven gas cooled energy amplifier with a separate target.

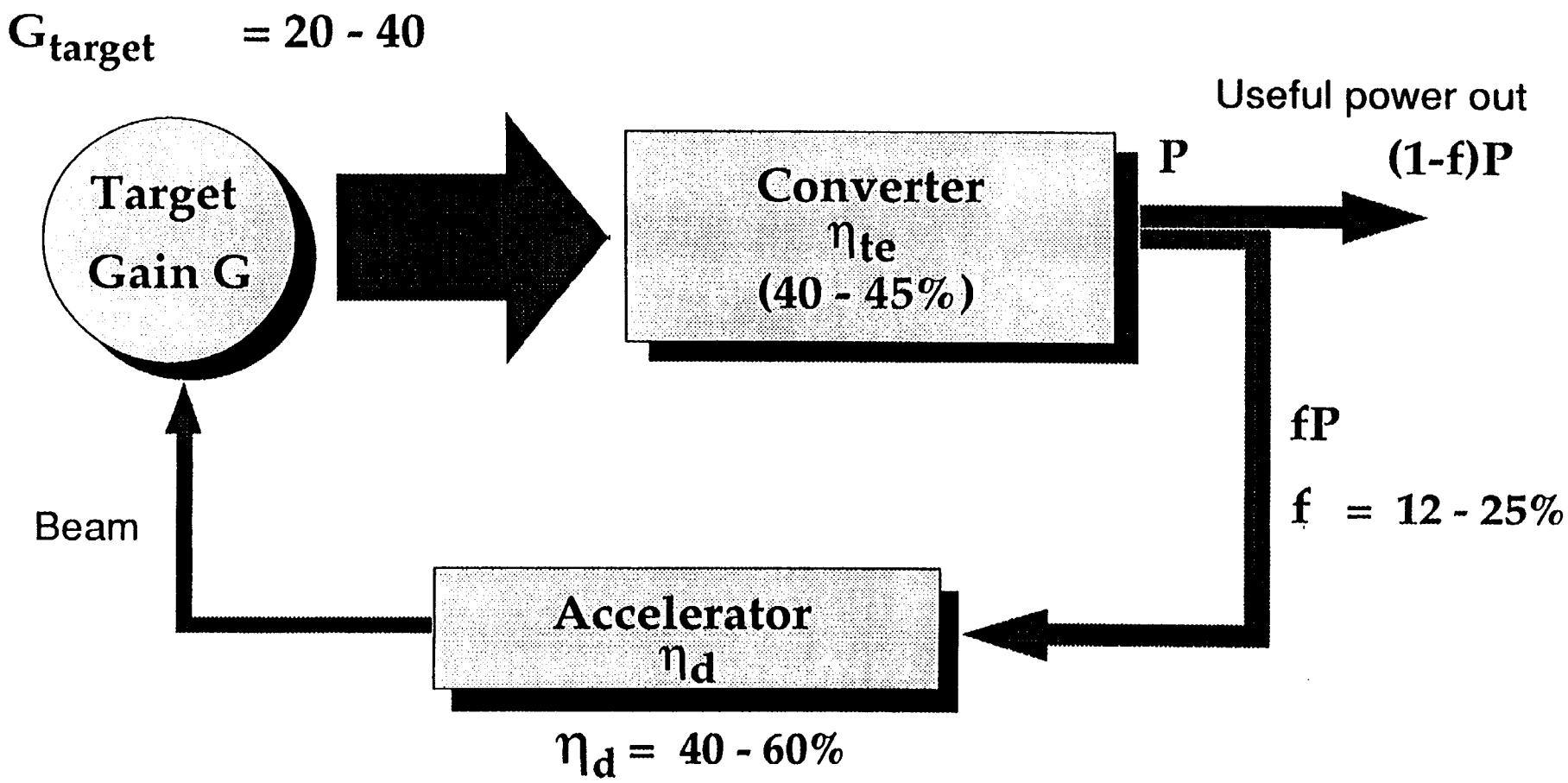

G_{target} = 20 - 40
Useful power out
Target
Gain G
Converter
η_{te}
(40 - 45%)
P
(1-f)P
fP
f = 12 - 25%
Beam
Accelerator
η_d
η_d = 40 - 60%

Figure 1

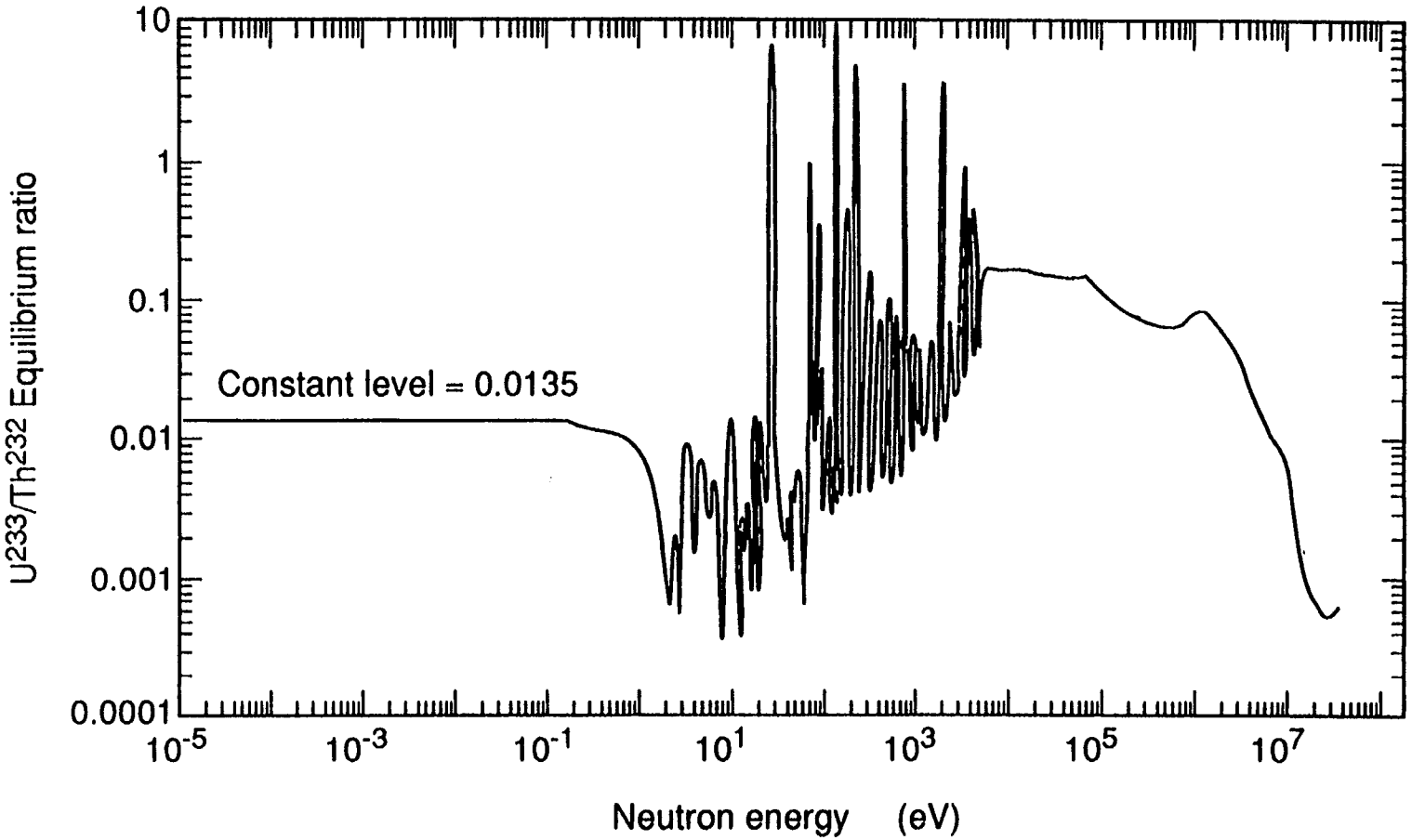

10
1
0.1
0.01
0.001
0.0001
U^{233}/Th^{232} Equilibrium ratio
Constant level = 0.0135
10^{-5}
10^{-3}
10^{-1}
10^{1}
10^{3}
10^{5}
10^{7}
Neutron energy (eV)

Figure 2

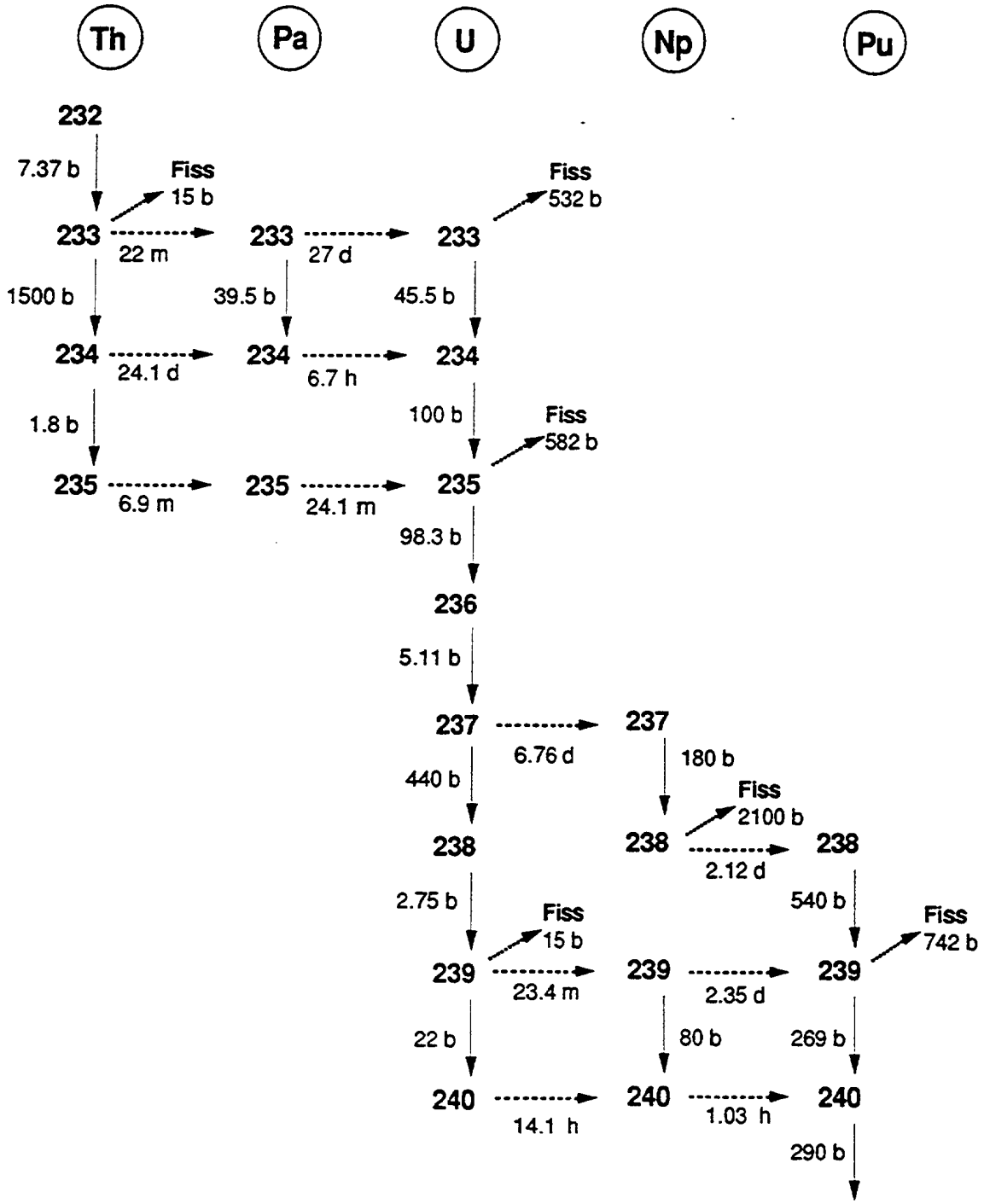

Th
Pa
U
Np
Pu
232
7.37 b
Fiss
15 b
233
22 m
233
27 d
233
Fiss
532 b
1500 b
39.5 b
45.5 b
234
24.1 d
234
6.7 h
234
1.8 b
100 b
Fiss
582 b
235
6.9 m
235
24.1 m
235
98.3 b
236
5.11 b
237
6.76 d
237
180 b
440 b
Fiss
2100 b
238
238
2.12 d
238
2.75 b
540 b
Fiss
15 b
Fiss
742 b
239
23.4 m
239
2.35 d
239
22 b
80 b
269 b
240
14.1 h
240
1.03 h
240
290 b

Figure 3

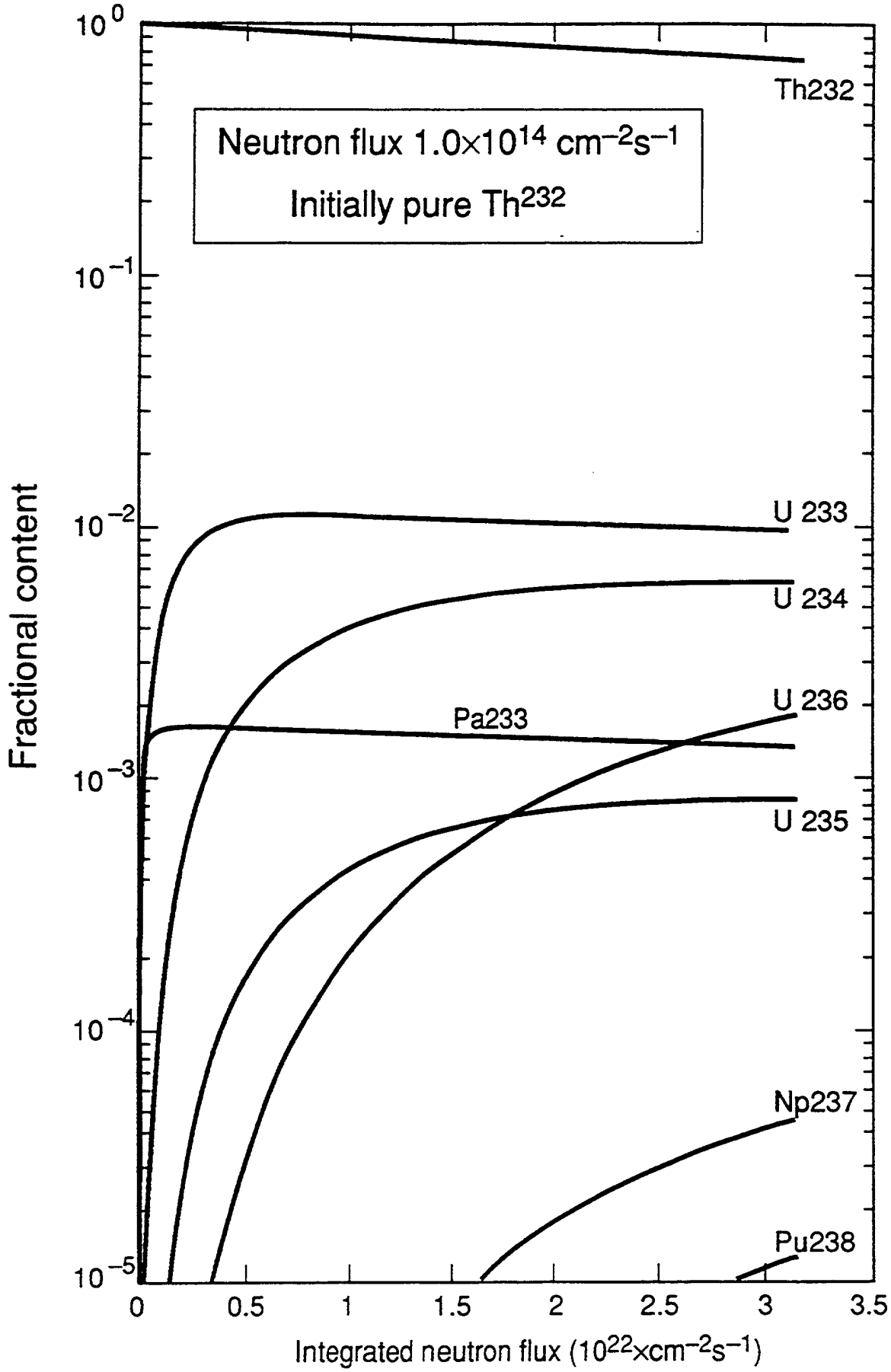

Neutron flux 1.0×10^14 cm^-2s^-1
Initially pure Th^232
Th232
U 233
U 234
U 236
Pa233
U 235
Np237
Pu238
Fractional content
10^0
10^-1
10^-2
10^-3
10^-4
10^-5
0
0.5
1
1.5
2
2.5
3
3.5
Integrated neutron flux (10^22×cm^-2s^-1)

Figure 4

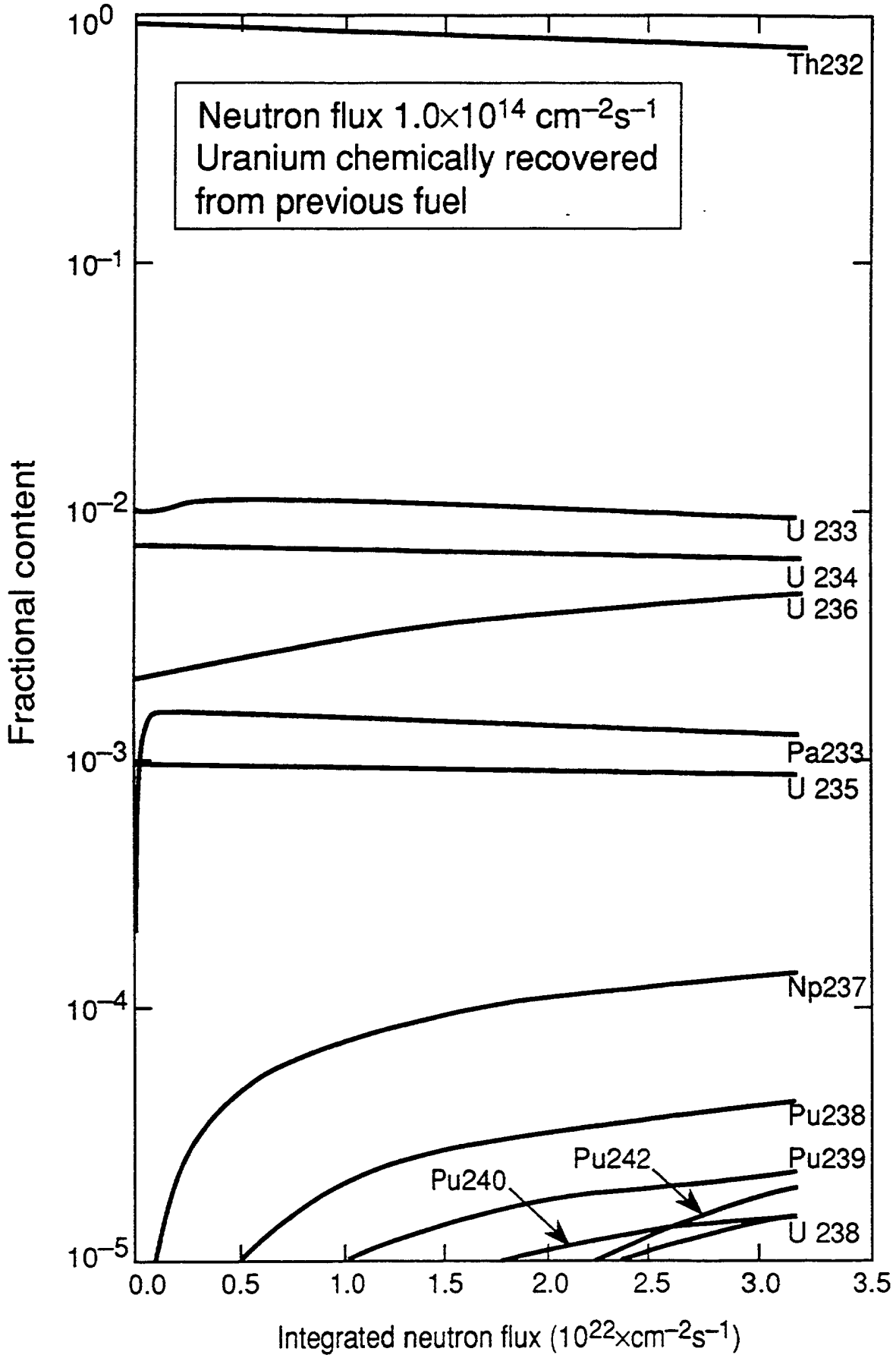

Neutron flux 1.0×10^14 cm^-2s^-1
Uranium chemically recovered
from previous fuel
Th232
U 233
U 234
U 236
Pa233
U 235
Np237
Pu238
Pu239
Pu240
Pu242
U 238
Fractional content
10^0
10^-1
10^-2
10^-3
10^-4
10^-5
0.0
0.5
1.0
1.5
2.0
2.5
3.0
3.5
Integrated neutron flux (10^22×cm^-2s^-1)

Figure 5

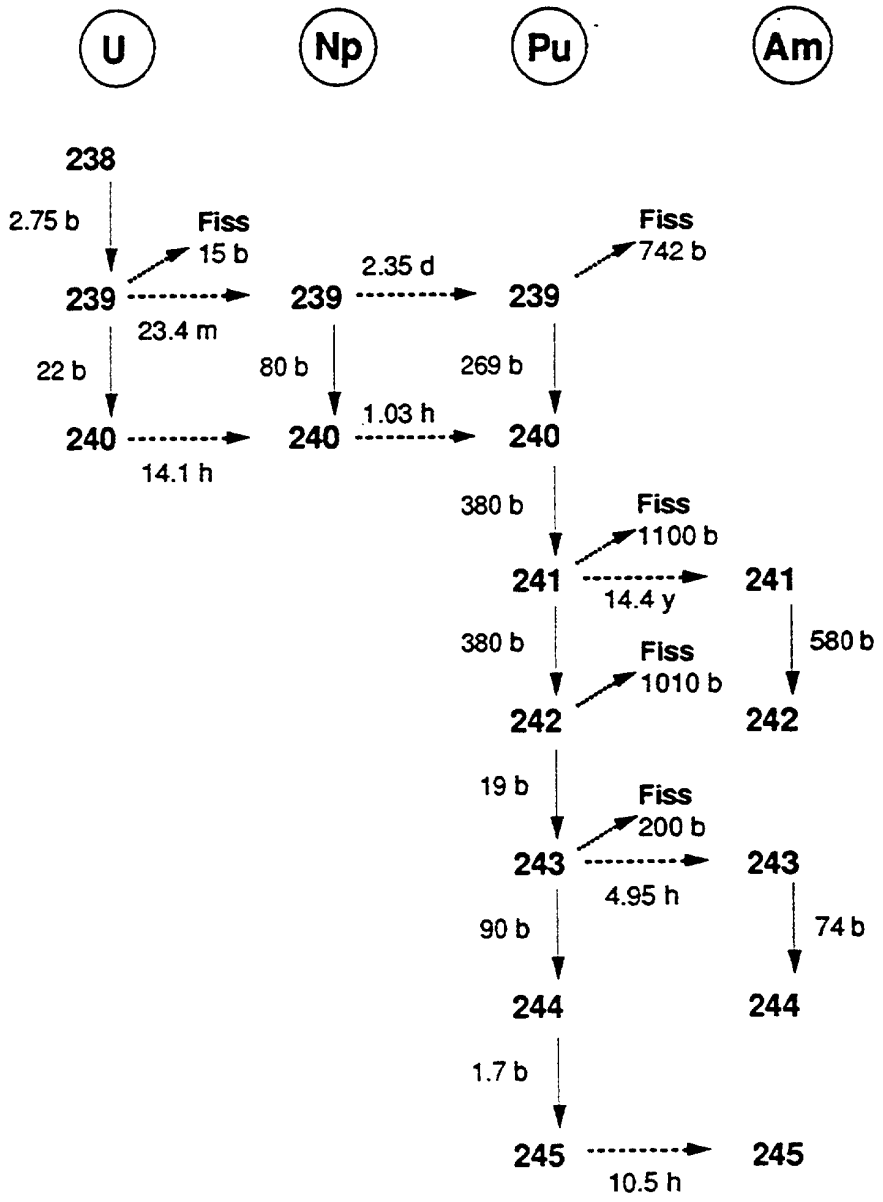

U
Np
Pu
Am
238
2.75 b
Fiss
15 b
239
23.4 m
239
2.35 d
239
Fiss
742 b
22 b
80 b
269 b
240
14.1 h
240
1.03 h
240
380 b
Fiss
1100 b
241
14.4 y
241
380 b
Fiss
1010 b
580 b
242
242
19 b
Fiss
200 b
243
4.95 h
243
90 b
74 b
244
244
1.7 b
245
10.5 h
245

Figure 6

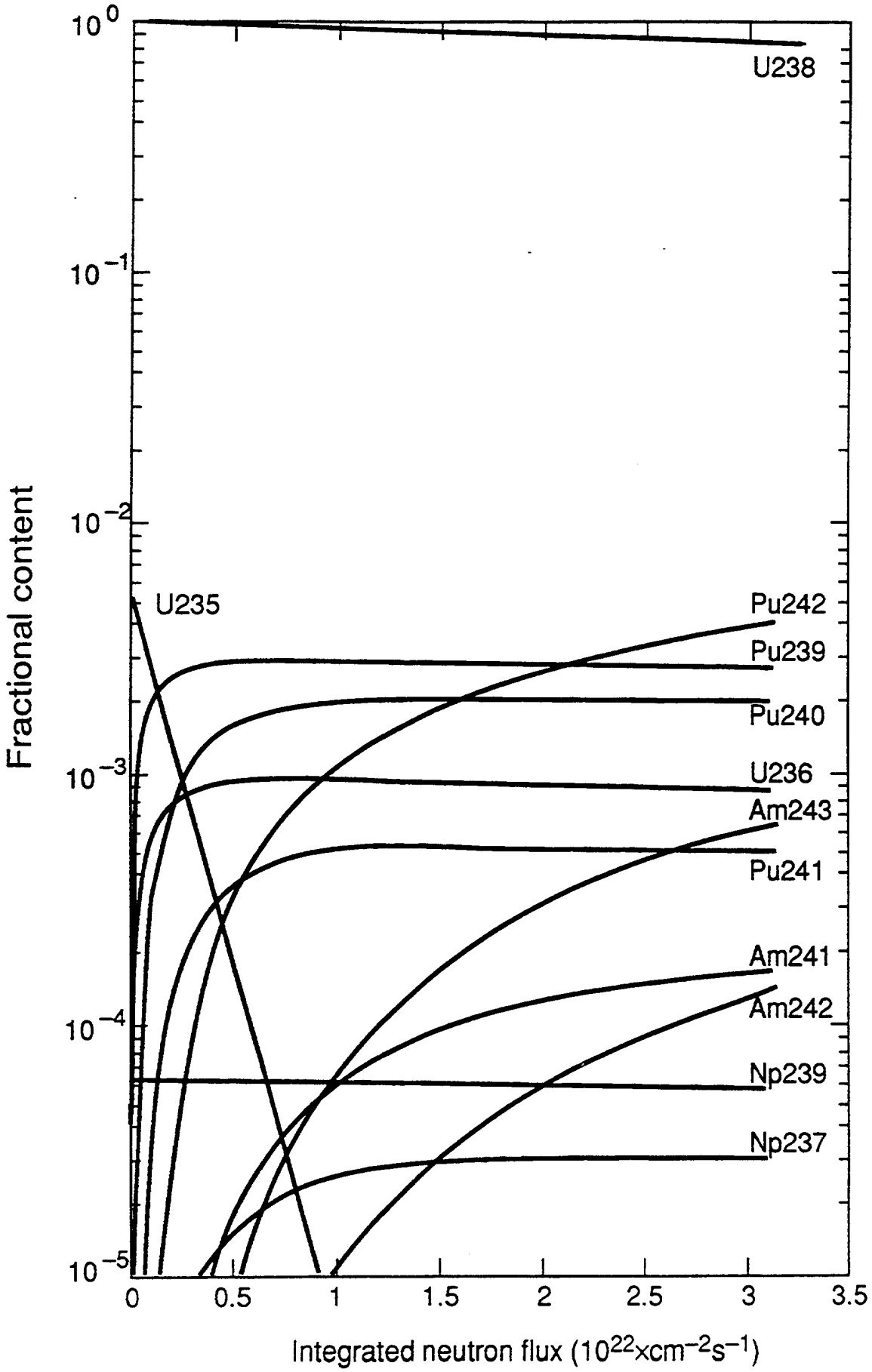

Fractional content
10^0
10^{-1}
10^{-2}
10^{-3}
10^{-4}
10^{-5}
U238
U235
Pu242
Pu239
Pu240
U236
Am243
Pu241
Am241
Am242
Np239
Np237
0
0.5
1
1.5
2
2.5
3
3.5
Integrated neutron flux (10^{22}xcm^{-2}s^{-1})

Figure 7

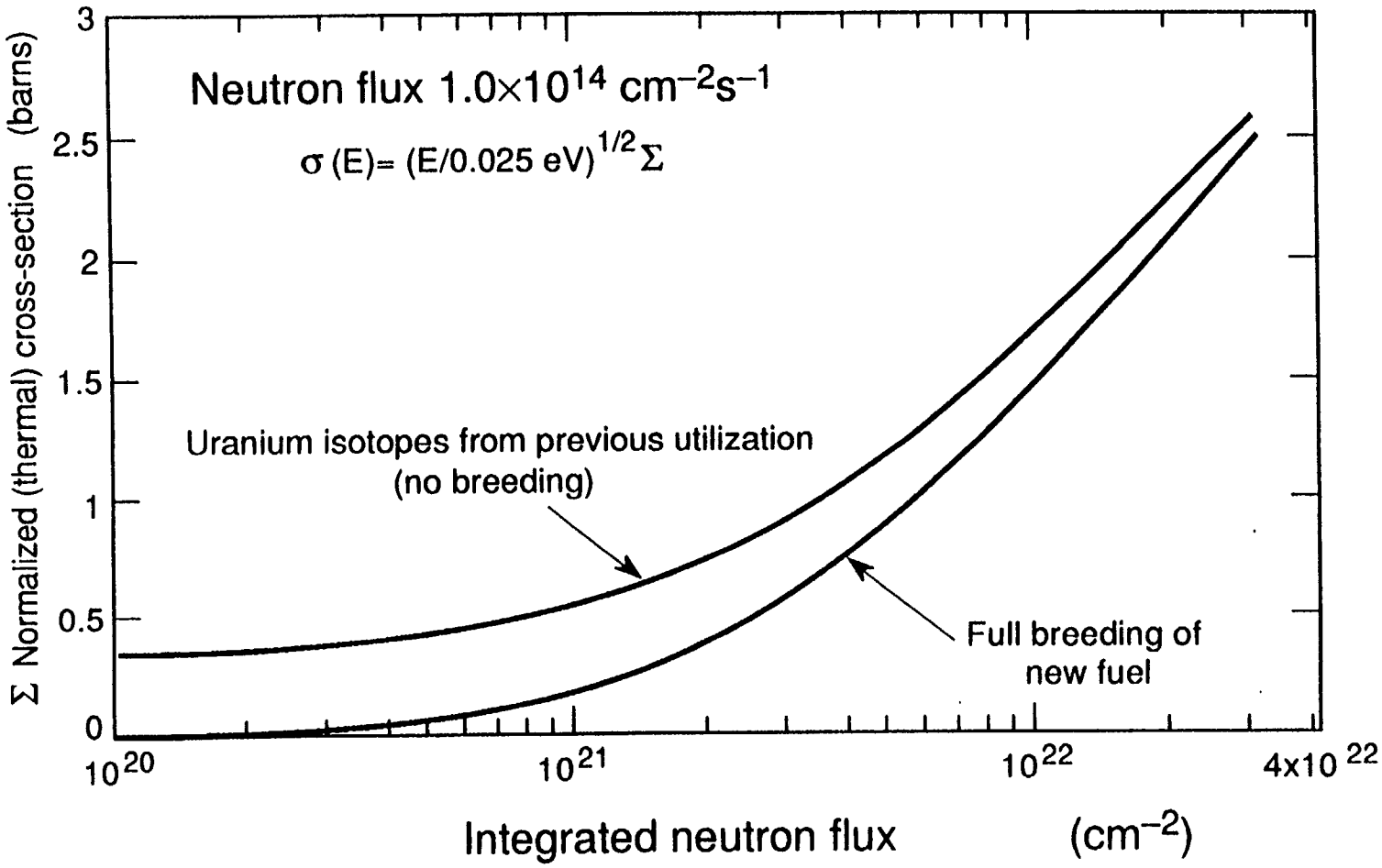

Neutron flux 1.0×10^14 cm^-2s^-1
σ (E)= (E/0.025 eV)^1/2 Σ
Uranium isotopes from previous utilization
(no breeding)
Full breeding of
new fuel
Σ Normalized (thermal) cross-section (barns)
3
2.5
2
1.5
1
0.5
0
10^20
10^21
10^22
4x10^22
Integrated neutron flux (cm^-2)

Figure 8

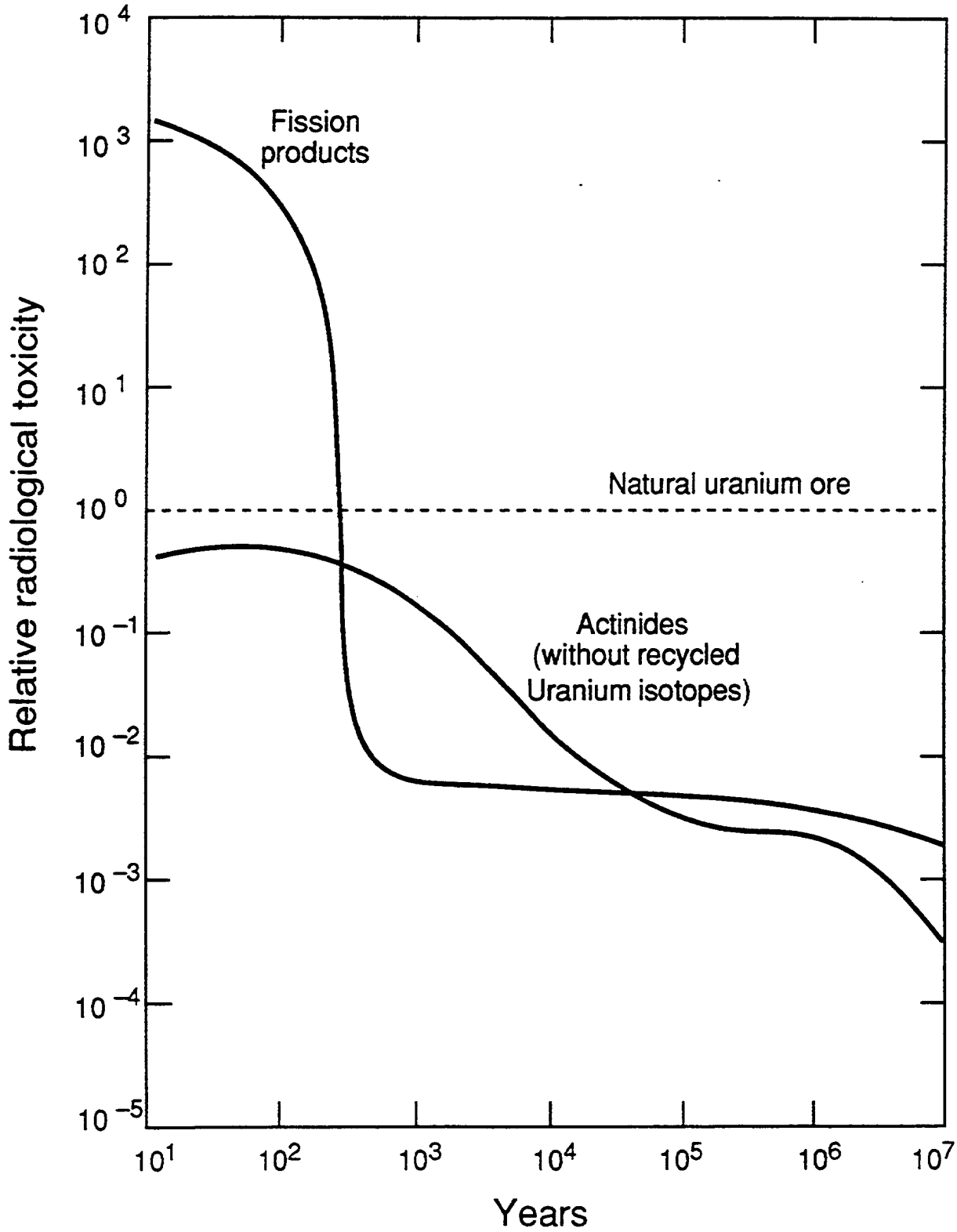

Relative radiological toxicity
10 4
10 3
10 2
10 1
10 0
10 −1
10 −2
10 −3
10 −4
10 −5
Fission products
Natural uranium ore
Actinides (without recycled Uranium isotopes)
10¹
10²
10³
10⁴
10⁵
10⁶
10⁷
Years

Figure 9

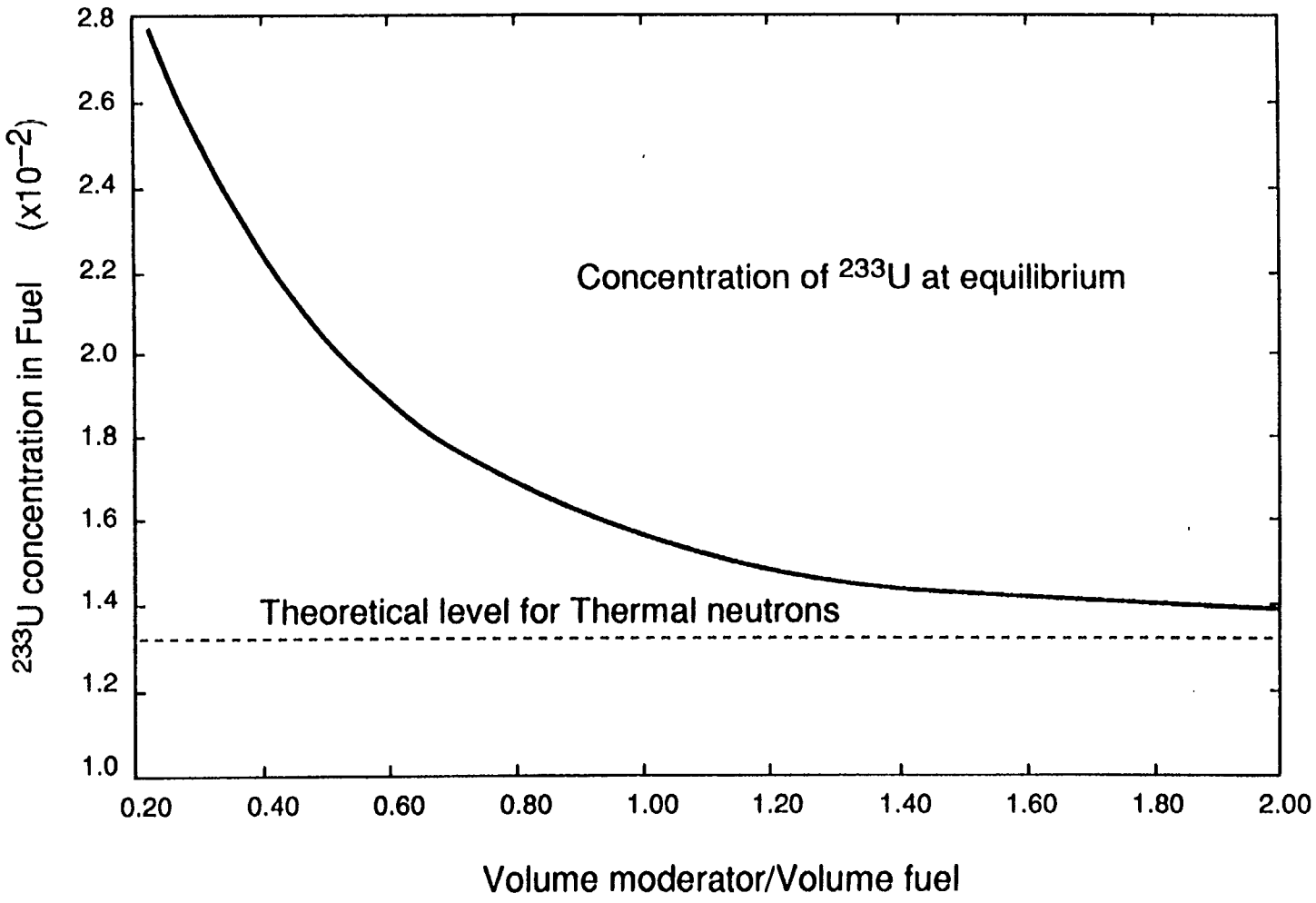

2.8
2.6
2.4
2.2
2.0
1.8
1.6
1.4
1.2
1.0
233U concentration in Fuel (x10−2)
Concentration of 233U at equilibrium
Theoretical level for Thermal neutrons
0.20
0.40
0.60
0.80
1.00
1.20
1.40
1.60
1.80
2.00
Volume moderator/Volume fuel

Figure 10a

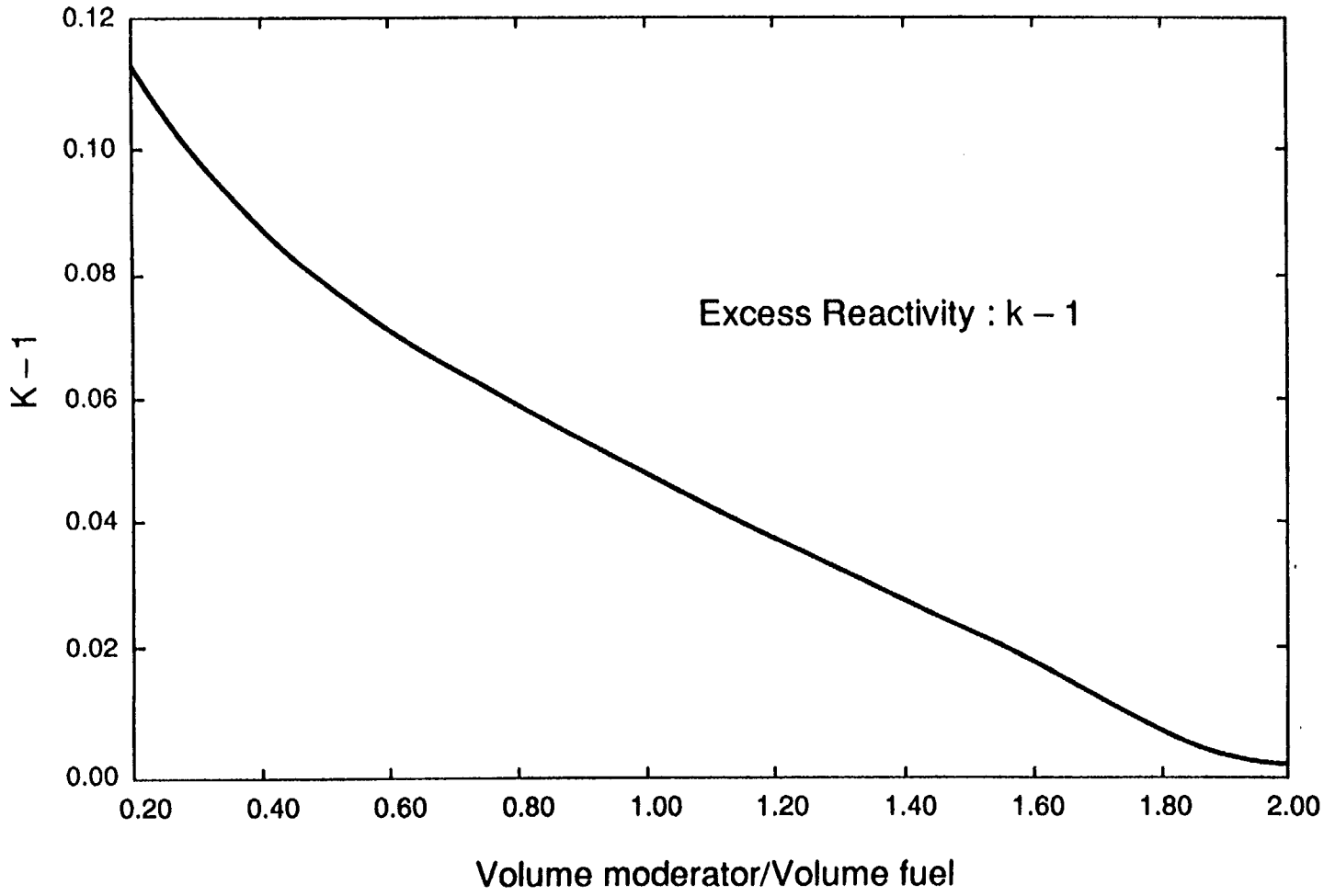

0.12
0.10
0.08
0.06
0.04
0.02
0.00
K – 1
Excess Reactivity : k – 1
0.20
0.40
0.60
0.80
1.00
1.20
1.40
1.60
1.80
2.00
Volume moderator/Volume fuel

Figure 10b

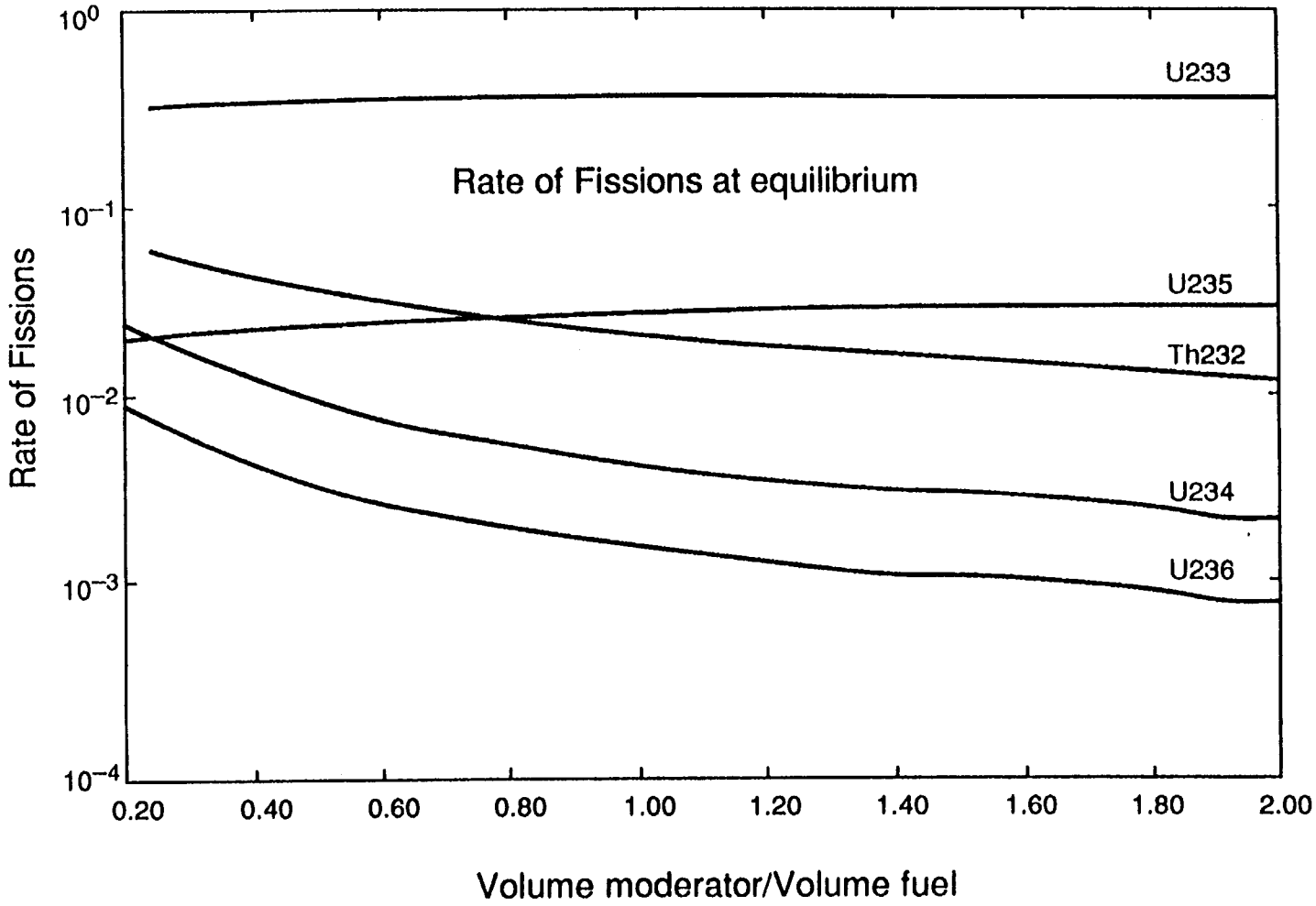

Rate of Fissions at equilibrium
U233
U235
Th232
U234
U236
Rate of Fissions
10^0
10^{-1}
10^{-2}
10^{-3}
10^{-4}
0.20
0.40
0.60
0.80
1.00
1.20
1.40
1.60
1.80
2.00
Volume moderator/Volume fuel

Figure 10c

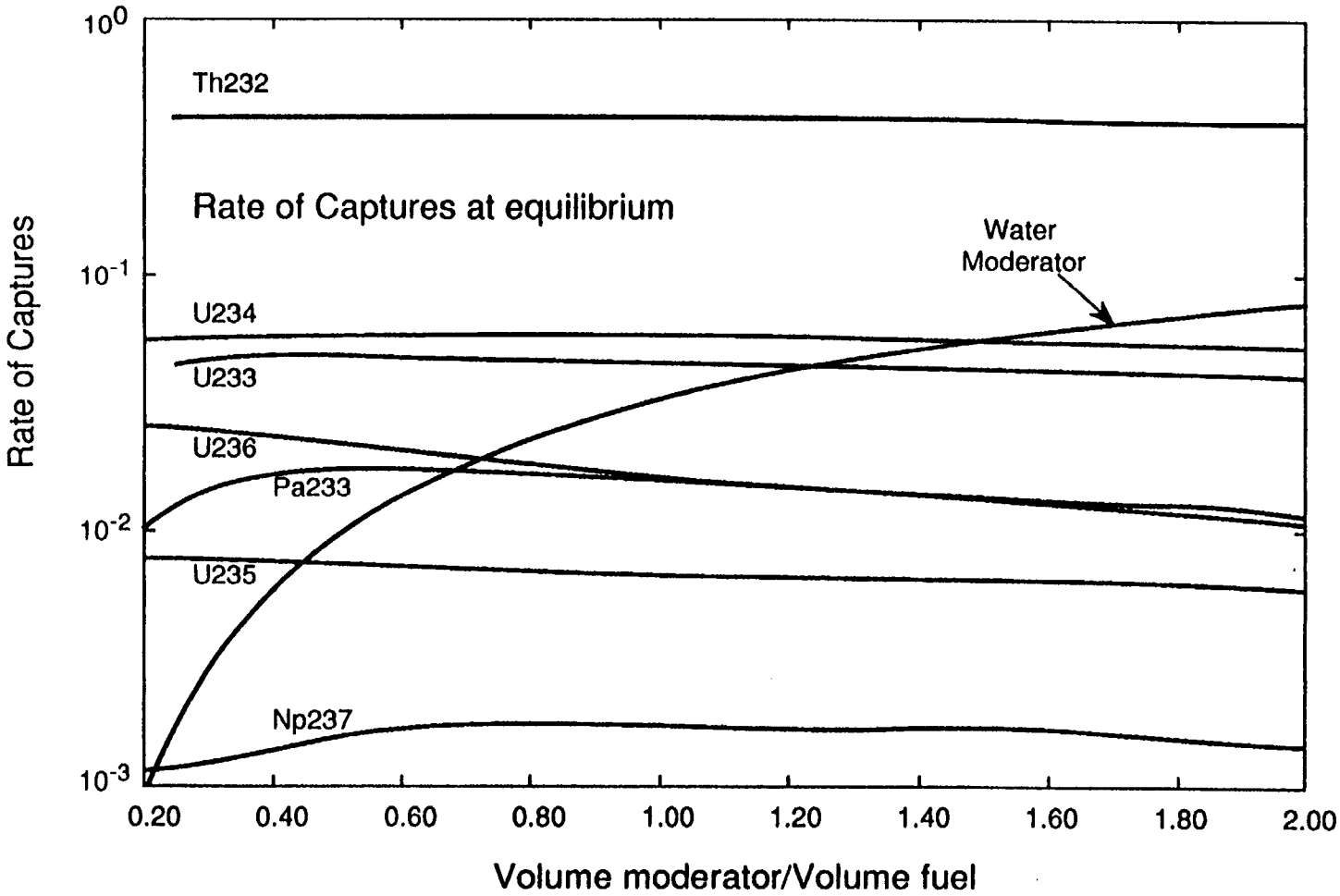

Th232
Rate of Captures at equilibrium
Water
Moderator
U234
U233
U236
Pa233
U235
Np237
Rate of Captures
10^0
10^{-1}
10^{-2}
10^{-3}
0.20
0.40
0.60
0.80
1.00
1.20
1.40
1.60
1.80
2.00
Volume moderator/Volume fuel

Figure 10d

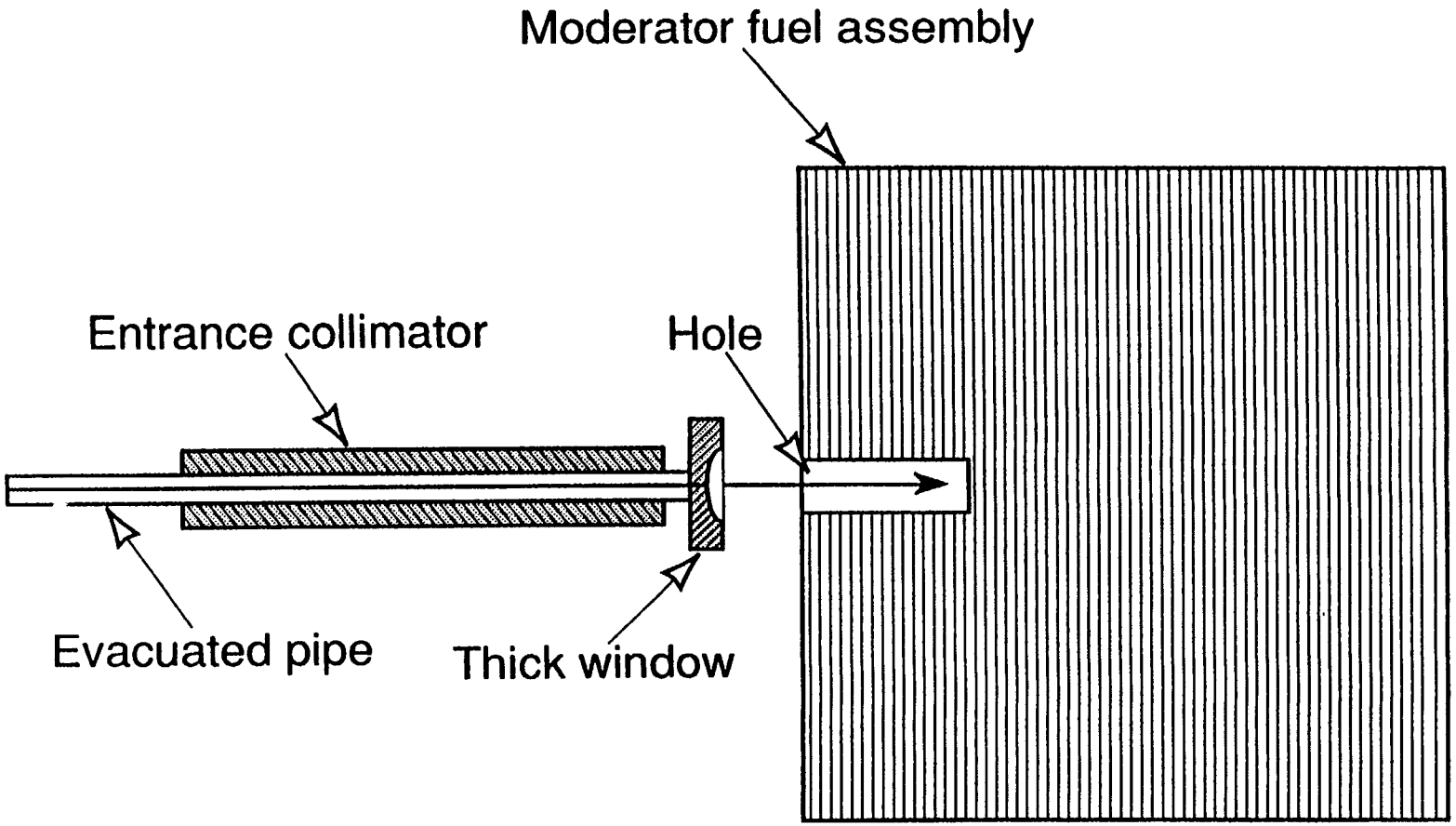

Moderator fuel assembly
Entrance collimator
Hole
Evacuated pipe
Thick window

Figure 11

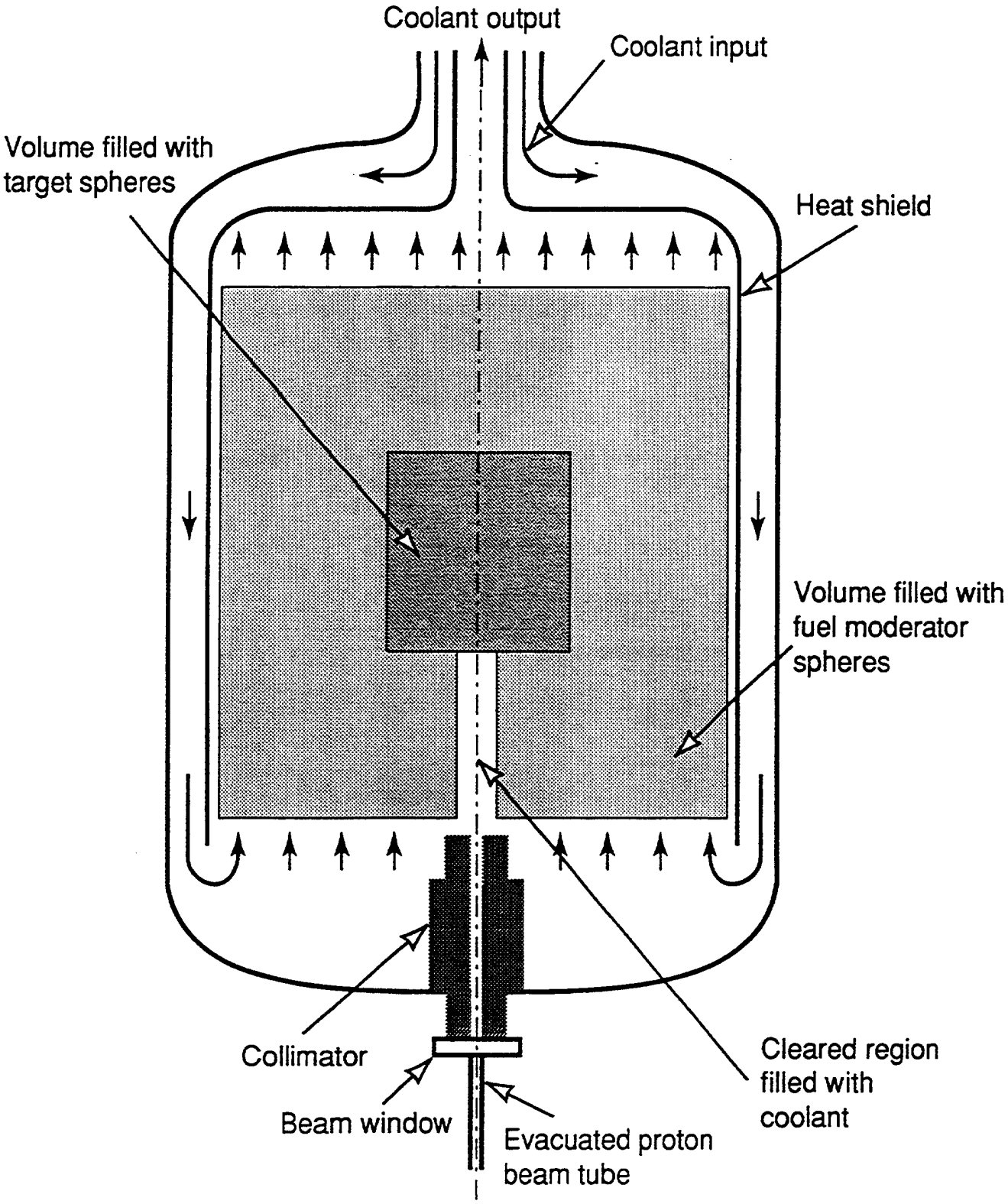
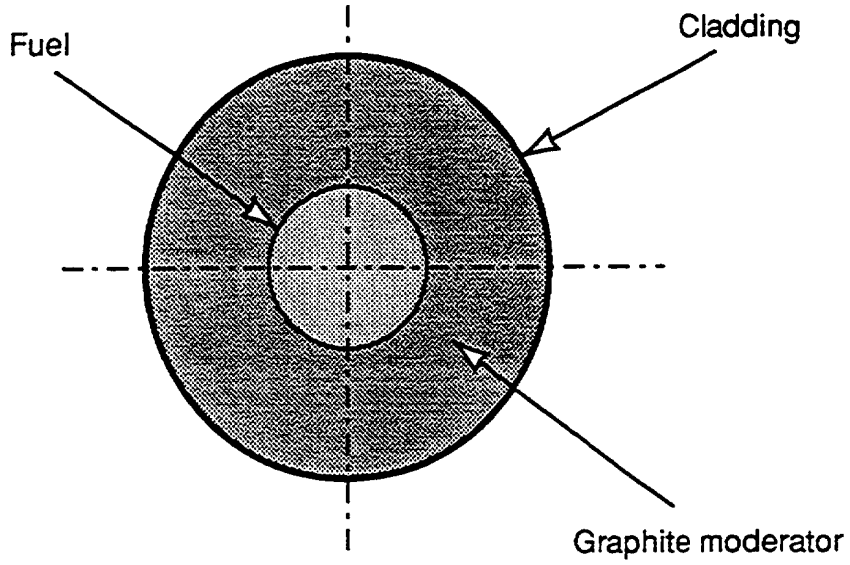

Coolant output
Coolant input
Volume filled with target spheres
Heat shield
Volume filled with fuel moderator spheres
Collimator
Beam window
Evacuated proton beam tube
Cleared region filled with coolant

Figure 12a

FUEL MODERATOR SPHERE



TARGET SPHERE

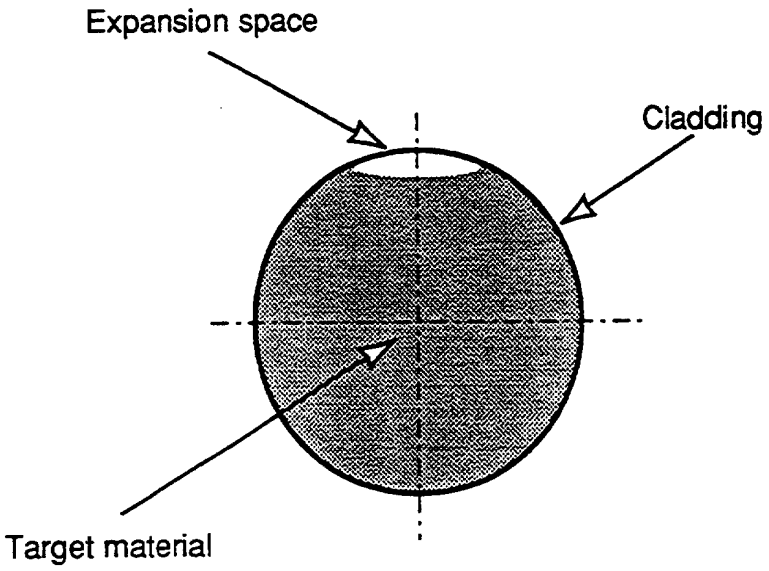


Figure 12b

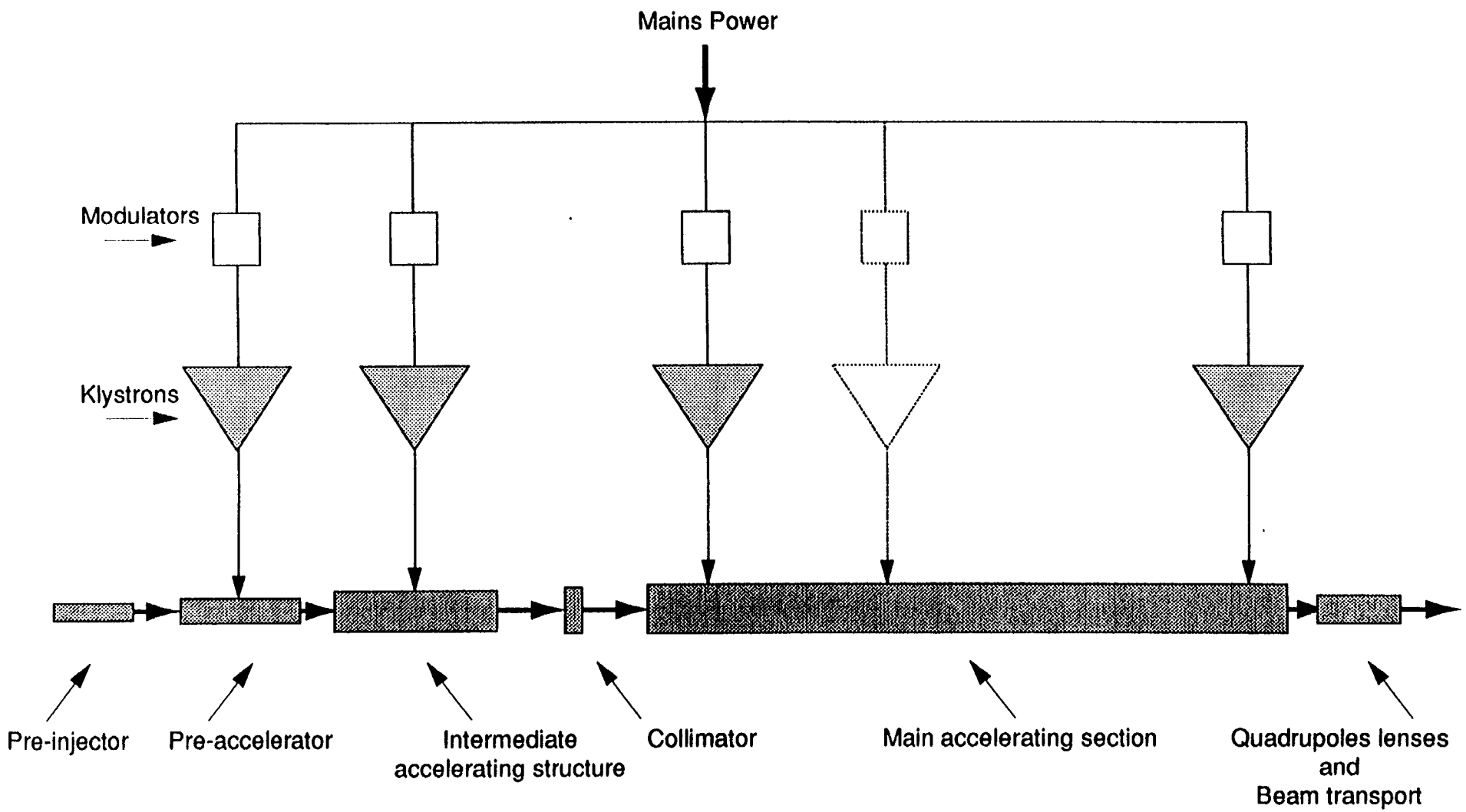

Mains Power
Modulators
Klystrons
Pre-injector
Pre-accelerator
Intermediate accelerating structure
Collimator
Main accelerating section
Quadrupoles lenses and Beam transport

Figure 13

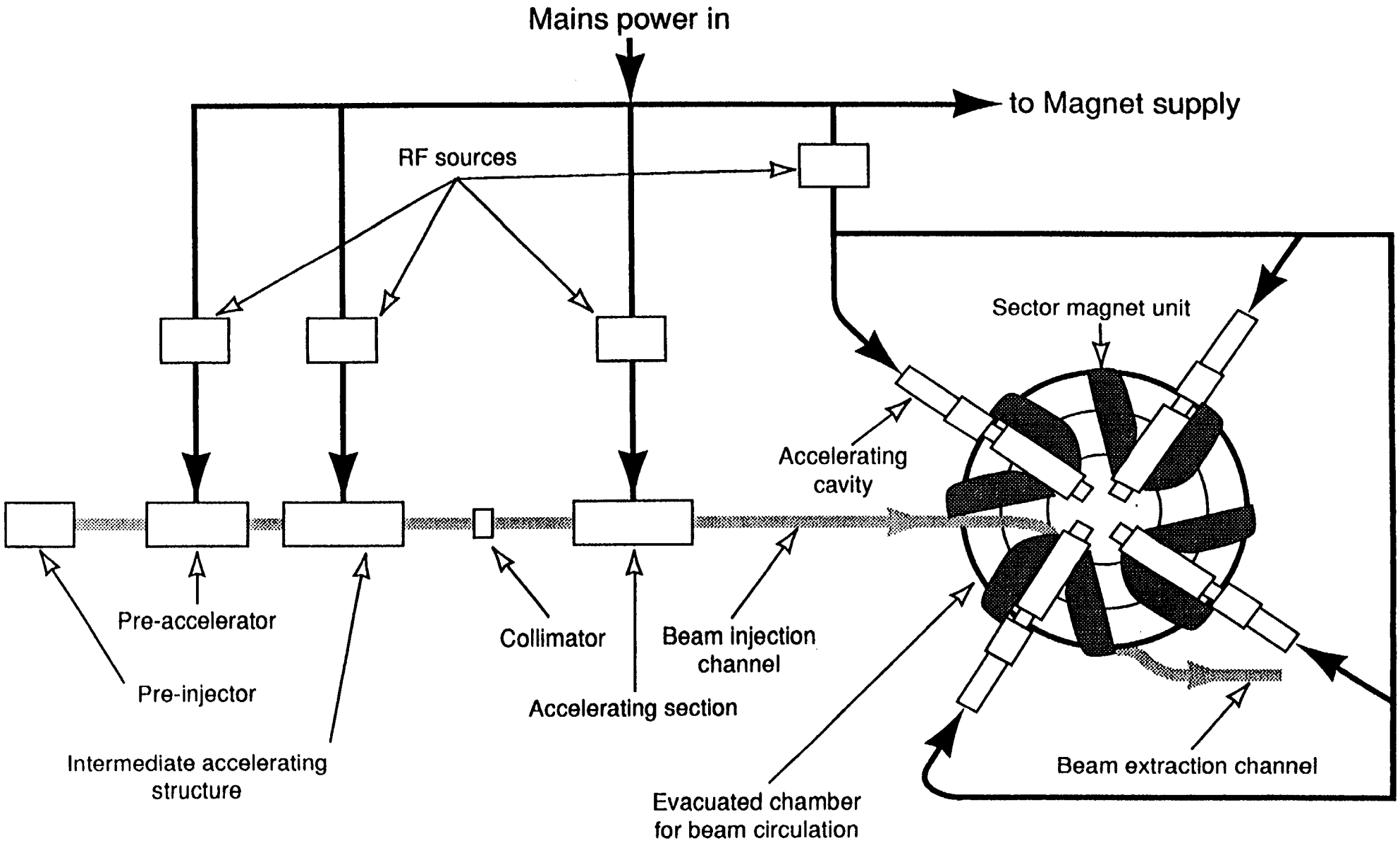

Mains power in
to Magnet supply
RF sources
Sector magnet unit
Accelerating cavity
Pre-accelerator
Collimator
Beam injection channel
Pre-injector
Accelerating section
Intermediate accelerating structure
Beam extraction channel
Evacuated chamber for beam circulation

Figure 14

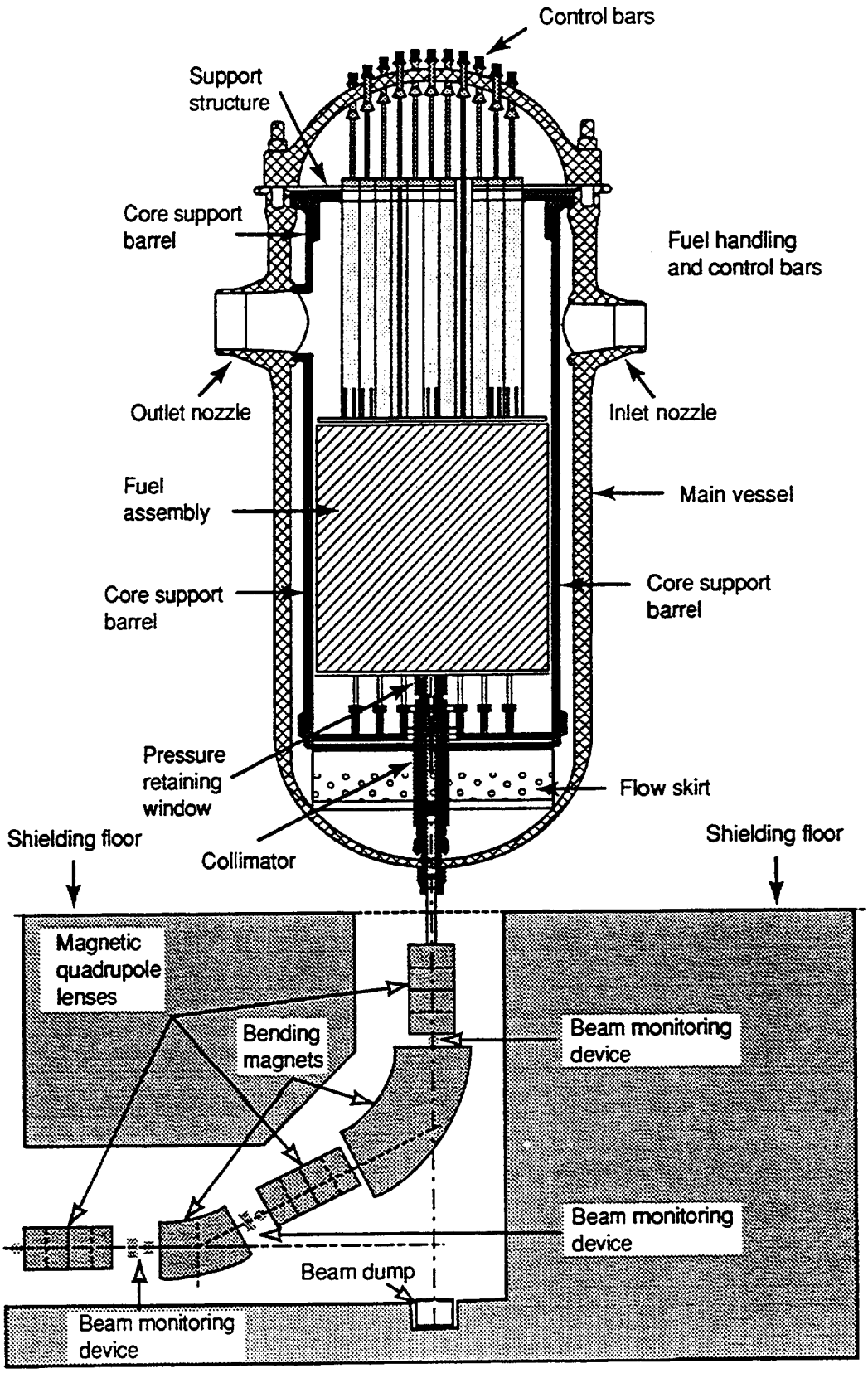

Control bars
Support structure
Core support barrel
Fuel handling and control bars
Outlet nozzle
Inlet nozzle
Fuel assembly
Main vessel
Core support barrel
Core support barrel
Pressure retaining window
Flow skirt
Shielding floor
Collimator
Shielding floor
Magnetic quadrupole lenses
Bending magnets
Beam monitoring device
Beam monitoring device
Beam dump
Beam monitoring device

Figure 15

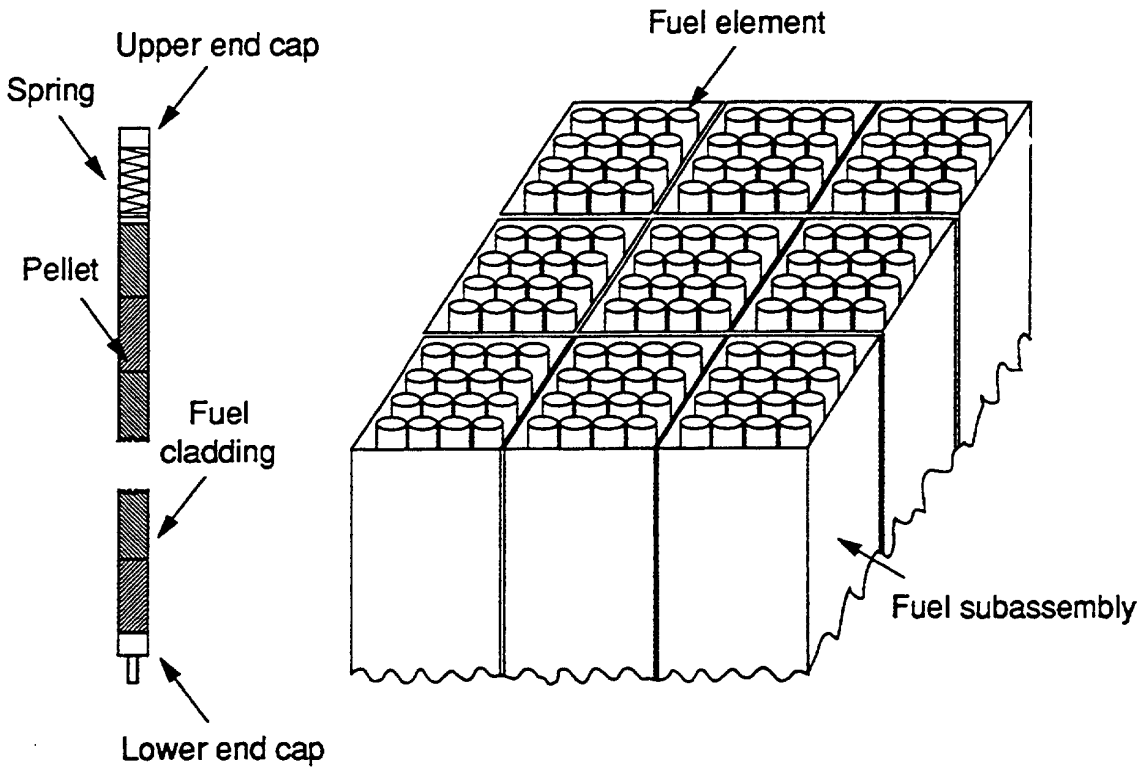

Upper end cap
Spring
Pellet
Fuel
cladding
Lower end cap
Fuel element
Fuel subassembly

Figure 16

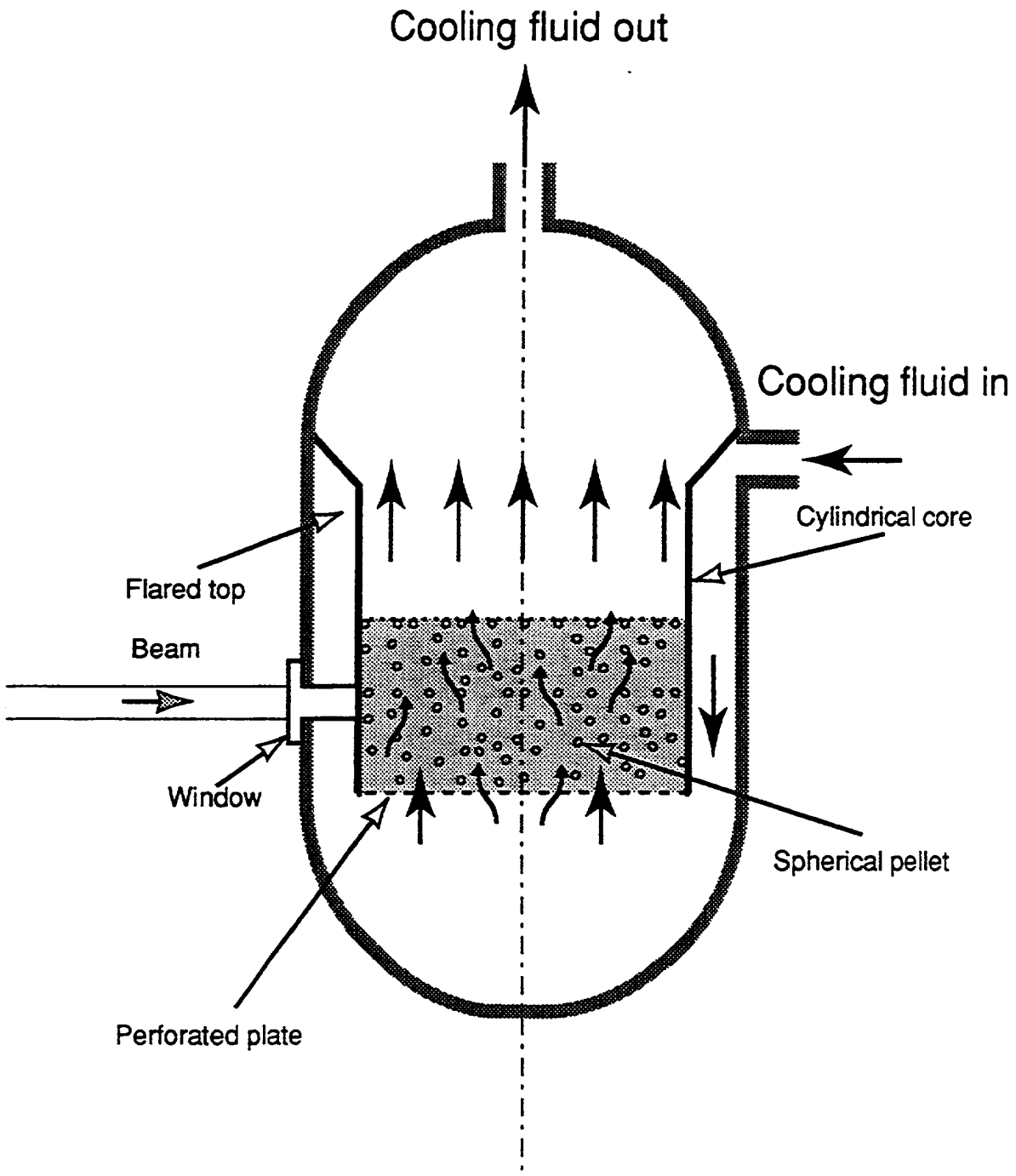

Cooling fluid out
Cooling fluid in
Cylindrical core
Flared top
Beam
Window
Spherical pellet
Perforated plate

Figure 17

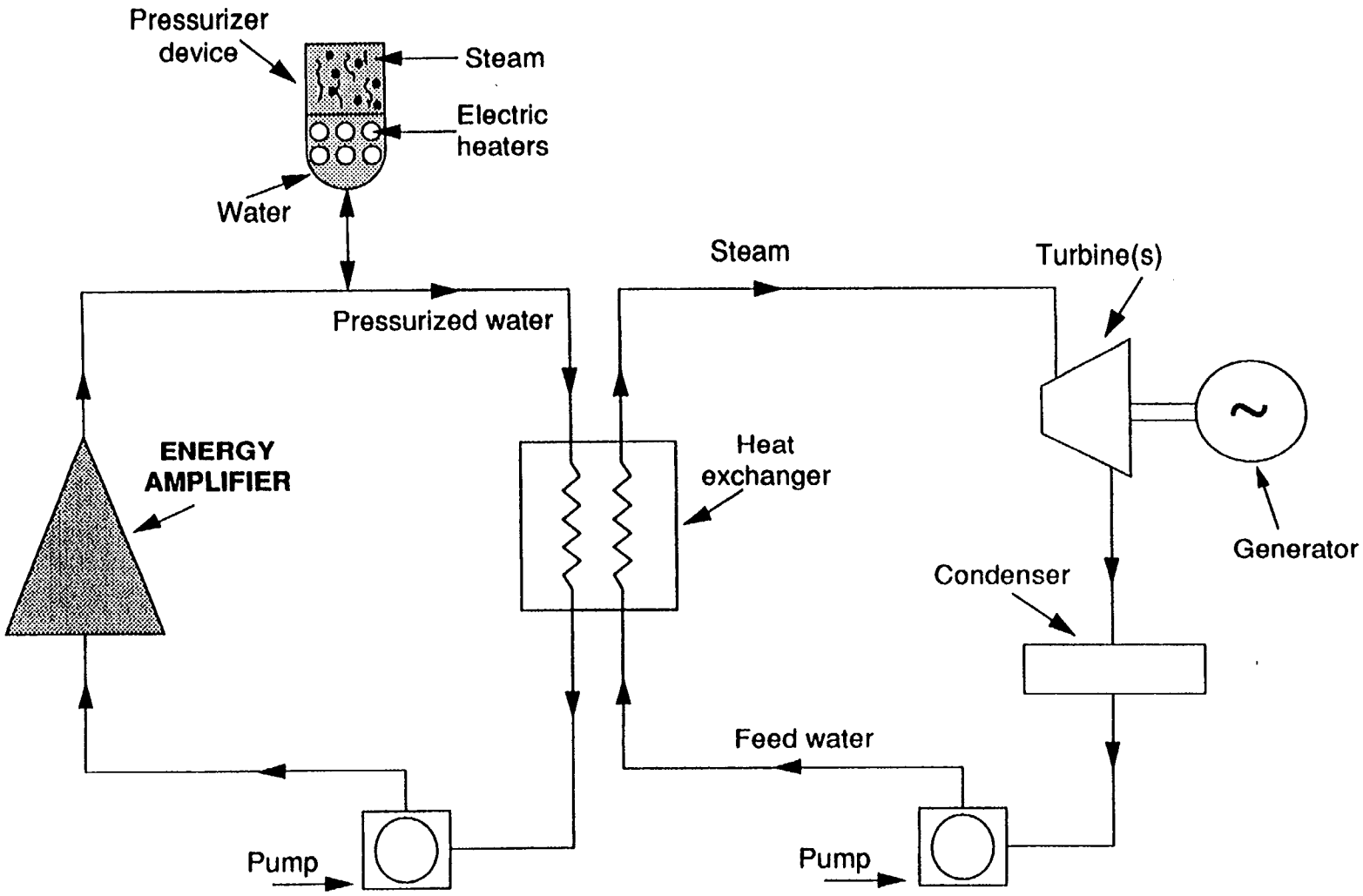

Pressurizer device
Steam
Electric heaters
Water
Steam
Turbine(s)
Pressurized water
ENERGY AMPLIFIER
Heat exchanger
Generator
Condenser
Feed water
Pump
Pump

Figure 18

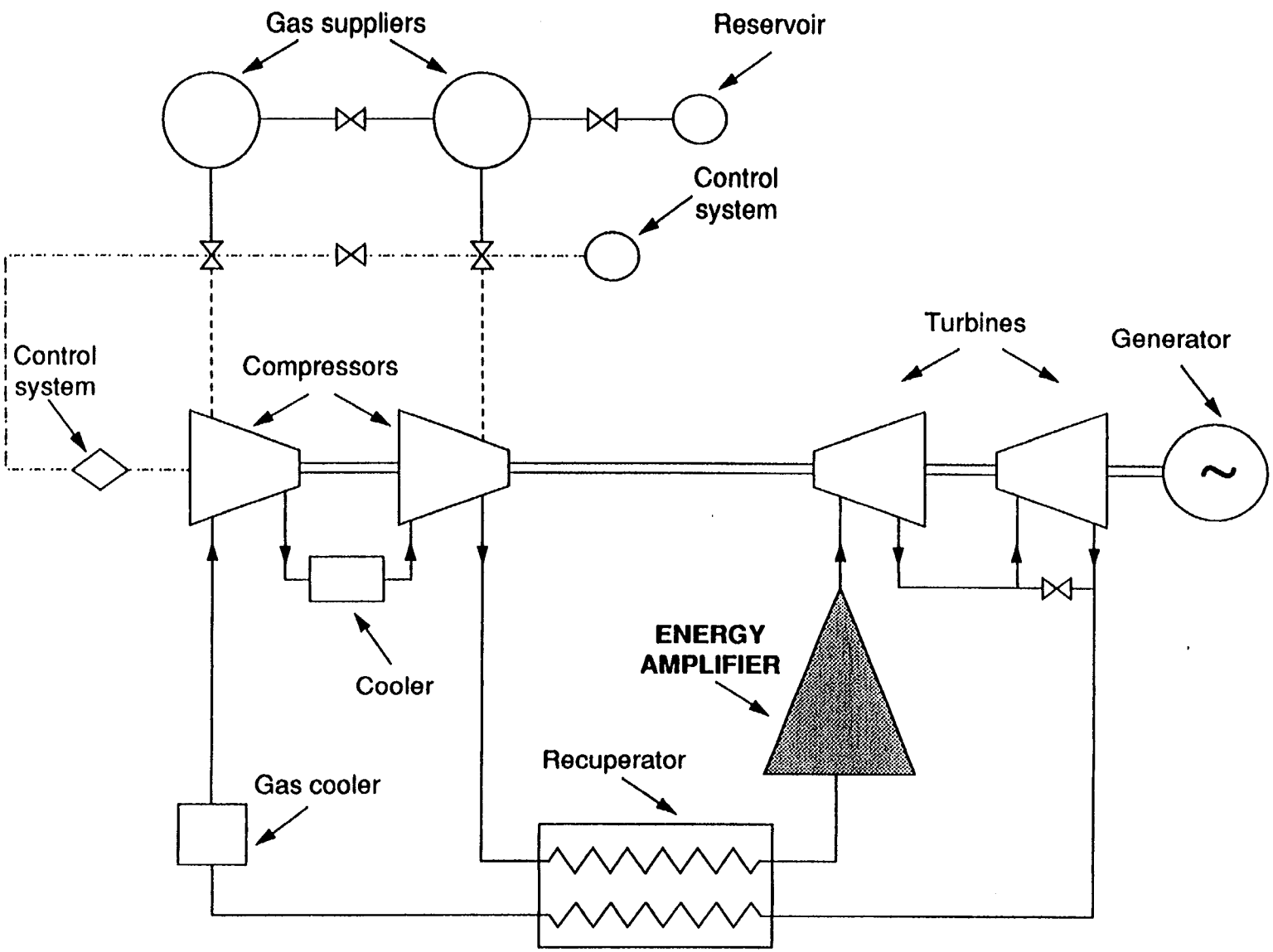

Gas suppliers
Reservoir
Control system
Control system
Compressors
Turbines
Generator
Cooler
ENERGY AMPLIFIER
Recuperator
Gas cooler

Figure 19