CHARACTERISTICS OF MINOR ACTINIDE TRANSMUTATION IN MINOR ACTINIDE BURNER REACTORS AND POWER REACTOR

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ABSTRACT

One of the techniques of minor actinides transmutation is to recycle these nuclides into fission reactors. Two concepts of a minor actinides burner reactor with very hard neutron energy spectrum as well as very high neutron flux are discussed. The fuel cycle facilities for these burner reactors are assessed to discuss the technical feasibility of these reactors. Transmutation of minor actinide in burner reactors is compared with those in power reactors from the viewpoint of the reactor physics and the fuel cycle.

Introduction

Nuclear transmutation of long-lived nuclides into shorter-lived nuclides is an attractive option which may alleviate the burden of geologic disposal scenario.

Various methods of transmutation have been proposed. One of the most practical method is to recycle minor actinides into fission reactors. Of the choices for potential transmutation in fission reactors, we have been proposing the concept of minor actinide burner fast reactors(ABR).[1,2,3,4] Since most of minor actinides such as Np-237, Am-241, Am-243, Cm-244 (hereafter referred as MA) are fissionable with fission threshold in several hundred keV range and capture cross sections of these nuclides rapidly decrease with neutron energy higher than this energy region, ABR with very hard neutron energy spectrum and high neutron flux will be useful for efficient and effective transmutation of minor actinides. The combination of a partitioning facility, an ABR and a final repository for thereby generated shorter-lived waste nuclides forms a high-level radioactive waste (HLW) management park in which troublesome HLW will be contained and electricity is generated from MA fission as illustrated in Fig. 1.

Previous studies have shown the feasibility of minor actinide transmutation in a power reactor such as LWR or LMFBR.[5] In these reactors, however, neutron spectra are rather too soft for MA to directly undergo fission. In LWR, these nuclides undergo fission mostly after one or two neutron capture, for example, Np-237 undergoes fission as Pu-239. Even in LMFBR, the fraction of neutron of which energy is higher than MA fission threshold is too small for effective fission of MA.

In this paper, the design study of technically feasible ABR with the hardest possible neutron spectrum and the highest possible neutron flux is briefly discussed. MA transmutation in these ABRs are compared with those in power reactors.

Designing of Higher Actinide Burner Reactors

In this design study, the fuel property and the thermal hydraulic analyses together with the nuclear analysis were carried out to obtain a model of actinide burner reactors with the very hard neutron spectrum as well as the very high neutron flux.

The guidelines for designing ABR are as follows;

- the major fuel material is MA,
- maximum core power density is attainable within the maximum allowable temperature limits of fuel and cladding,
- bumup reactivity swing is less than 3% δ k/k per cycle,

- power flattening,
- the hardest possible core averaged neutron spectrum,
- long fuel residence cycle length within the maximum allowable neutron irradiation of cladding material.

The composition of MA generated in PWR was calculated using JENDL-2 library and SRAC-FPGS bumup calculation code system for LWR. In Table 1, the MA composition used in this study is shown.

Two types of ABR design were obtained, namely Na cooled MA alloy fuel ABR(M-ABR) and He cooled MA particle fuel ABR(P-ABR). The details of these ABR designing are described elsewhere. [4,6,7,8]

Na cooled MA alloy fuel ABR(M-ABR)

The concept of metal fuel with Na cooling is attractive to design a hard neutron spectrum reactor with high-metal density and low contents of light elements. In this design study, a fuel concept of pin-bundle fuel assembly without wrapper-tube is chosen with the intention to reduce the concentration of intermediate weight elements which cause neutron spectrum softening. The other advantage of metal fuel is compacmess of fuel cycle facilities when a pyrochemical reprocessing is applied, similar to that of IFR concept. 9)

Experimental MA data of fuel property required for designing ABR fuel are very scarce. Those data which are not measured include,

- density and melting point as function of alloy composition,
- eutectics with cladding material,
- thermal conductivity at high temperature.

Theoretically estimated data were used in designing a MA fuel. The followings are the result of the estimation,

1) Np and Am are not mutually soluble similar to U-rare earths systems,
2) to improve low melting point of MA metal, e.g.640 ℃ of Np, MA element are to be alloyed with thermal diluent,
3) Y would be an ideal thermal diluent for Am and Cm,
4) solidus of Np could be raised by alloying with Zr,
5) existence of Pu would not significantly affect the solidus of alloys.

The present design of ABR is based on two alloy systems, namely, Np-(Pu)-Zr and Am-Cm-(Pu)-Y. In these alloys, Pu is added because of two reasons; 1) to reduce critical mass; k_∞ of Np, Am and Cm composition of Table 1 is 1.6 when the volume ratio of coolant to fuel is 0.7. The addition of thermal diluent of Zr and Y by 10wt % of MA causes 0.2%k_∞ reduction, 2) to compensate for reactivity gain of MA transmutation; conversion of Np-237 to Pu-238, Am-231 to Am-232m introduces significant reactivity gain and this is to be compensated with bumup reactivity loss of Pu.

The fuel concept of M-ABR is shown in Fig.2.

He cooled MA particle bed ABR(P-ABR)

Thermal conductivity and solidus temperature are the limiting factors for the burnup rate of metallic MA fuel. Thermal conductivity of MA alloys will be lower than U based alloys. Therefore, the particle bed reactor concept was applied as an alternative ABR, which has the high efficiency in heat transfer since small particle size produces a large heat transfer surface per volume. [4] The bed of coated fuel particle contained in double concentric porous frits is directly cooled by helium. The fuel particle is a microsphere of MA nitride of 1 mm diameter which is coated with a refractory material such as TiN. In a cold fuel concept, the fuel temperature is to be kept lower than one third of its melting point to reduce mass transport. Reduced mass transport will result in smaller swelling and gas release. Therefore, thickness of coating layer can be minimized to give large heavy metal density to establish hard neutron spectrum in a core.

The fuel concept of P-ABR is shown in Fig. 3. The fuel kernel is homogeneous mixture of Pu and MA nitride. The heterogeneous recycling of MA is also applicable with the particle fuel where one type of fuel kernel is made of Pu nitride and the other is made of MA nitride. Homogeneous mixture of the two types of fuel particles forms the homogeneous core. On the contrary, ink case of pin type fuel, the heterogeneous recycling where MA is concentrated in special fuel pins or fuel assemblies may be favorable from the point of fuel handling but bring the singularity problem in the neutron field.

Designing of A BR plant and fuel cycle facilities

Conceptual design studies of M-ABR plant and its fuel cycle facilities were also carried out to assess the feasibility of ABR concept. These studies were performed by the reactor plant manufacturers under the contracts with JAERI. In Fig. 4, the ABR plant consisting of 6 modules of M-ABR is shown. In this plant, 300kg of MA generated in about 11 units of 3000MWt-PWR undergo fission yearly and 400MW electricity is generated. In this study, thepyrochemical reprocessing of spent MA fuel and the injection casting of MA alloy slug were chosen as the basic procedures of MA fuel cycle facilities. With these processes, the facilities will be compact and can be placed close to ABR plant.

The estimated amount of MA handling in these facilities is only 16kg of MA, 3 .3kg of fission products and 70kg of cladding material etc. per day for one unit of 1020MWt ABR plant.

Characteristics of ABRs designed

The reactor core design parameters of M-ABR and P-ABR at their equilibrium state are shown in Tables 2 and 3, respectively. Comparison of core averaged neutron spectra is shown in Fig. 5. In this figure, neutron spectrum of MOX fuel LMFBR is also shown for comparison. Significantly hard neutron spectra established in ABRs are obvious.

In these ABRs, Pu is mixed ordy in the initial fuel and in the latter fuel Pu is not added from the outside but Pu converted from Np-237 is not removed from the spent fuel. Necessity of Pu is explained in the previous section for M-ABR.

In Table 4, the transmutation-related reactor characteristics are compared between two types of ABRs together with a thermal and fast reactors. Ink M-ABR, the magnitude of neutron flux is rather low as opposed to the initial attempt to design a reactor with the highest possible neutron flux. This is due to the low melting point and low thermal conductivity of MA alloy. On the contrary ,in the P-ABR owning to the good heat removal characteristics of a particle fuel, the neutron flux of the is very high and this results in the higher transmutation and burnup rates than the M-ABR even though both ABRs have the similar hardness of the neutron spectrum.

One of the significant difference inthe reactor performance of the ABRs from the power reactors is that the fuel residence cycle of ABRs is limited by neutron fluence, not by burnup reactivity loss as it is the case for normal reactors. This is due to the fact that in ABRs, the neutron spectrum is very hard and the bumup reactivity swing is small as the result of compensation of bumup reactivity loss by reactivity gain from the conversion of MA to fissionable material (eg Np to Pu, Am-241 to Am242/Am-242m).

The M-ABR and a MOX-FBR have the similar magnitude of neutron flux but the former has the higher bumup rate than a MOX-FBR because of its harder neutron spectrum.

The small doppler reactivity coefficient and the small delayed neutron fraction are the disadvantage of ABRs in the turn.mt design. The former is due to the lack of U-238 in a core and to the very hard neutron spectrum. The latter is due to the small delayed neutron fraction of Np-237 and Am than those of U and Pu. These small reactivity coefficients may be increased by addition of U with the little sacrifice of the spectrum hardness.

The outlet temperature of Na coolant in Fig. 4 is only 430 ‘C. Under this moderate temperature condition, the life-time of the reactor will be as long as 50 to 60 years.

Comparison of MA transmutation in ABRs and in power reactors

Transmutation rate and burnup rate

The efficiency of a transmutation system has been usually discussed using the transmutation rate defined as a ratio of weight of MA which undergoes fission and capture to that of initial loading of MA per unit time. This transmutation rate, however, is not a good index to discuss transmutation effectiveness because the aim of transmutation is the conversion of long-lived nuclides to shorter-lived or stable nuclides and because fission, not capture, is a real transmutation reaction for long-lived MA. Therefore, the burnup rate of MA is the real index of the transmutation effectiveness and efficiency.

The difference between the transmutation rate and the burnup rate is the generation rate of much heavier MA than the initial MA loaded. This generation rates are much higher in power reactors than in ABRs especially in U-PWR because of large capture cross section of Np-237 and small fission cross section of daughter nuclides Po-238 in power reactors. When the conversion of Np-237 to Po is acceptable as the transmutation, the transmutation rates of FBRs are comparable to or higher than those of ABRs (Table 4). Pu in the transmutation chain of Np-237, however, is mostly Pu-238 and these Pu is not favorable one from the fuel cycle and reactor physics view point.

The burnup rate per fuel residence cycle is an index for the transmutation effectiveness. When this rate is small, MA fuel has to be recycled many times repeatedly and the number of ex-core treatment of MA-contained spent fuel is increasing . This wilt decrease the effectiveness of MA transmutation because of the increasing loss of MA into the open fuel cycle.

The burnup rate per year is also an important index of the transmutation efficiency because a given amount of MA should be fissioned within a given time period. When the burnup rate per fuel residence cycle is high and the burnup rate per year is low, such system is transmutation effective but is not efficient and larger number of such systems are required for transmutation of a given amount of MA. In Table 4, while there is not much difference in the burnup rate per cycle between ABRs and power reactors except for MOX-FBR, the burnup rate per year of P-ABR is significantly larger than those of M-ABR and power reactors because of its large magnitude of neutron flux together with the hard neutron spectrum.

Difference of transmutation between a fast spectrum and a thermal spectrum

The difference of MA transmutation between a fast reactor and a thermal reactor is shown in Table 5 for Np-237 and in Table 6 for Am-241. In Tables 5a and 5b, Np-237 fission and capture are compared respectively between M-ABR, FBR and PWR. In the calculation for these tables, Np-237 is irradiated for 20 cycles (one cycle consists of 300 days irradiation and 3 years cooling) and between cycles only fission products are removed. The residual Np-237 and higher actinides generated from Np-237 are reirradiated. In the M-ABR, the most of fission occurs as Np-237 and Pu-238 while in an U-PWR, the most of fission occurs as Po-239 and Pu-241 after the multiple neutron captures in Np-237. In Table 5b, the significant difference in the generation of Am and Cm between M-ABR and U-PWR is shown after 20 cycles irradiation of Np-237. In Table 6, the same trend of transmutation of Am-241 is shown.

In brief, the most part of fission threshold nuclides such as Np-237, Am-Ml directly undergoes fission in a fast reactor while in a thermal reactor, such nuclides undergoes one or more neutron capture first and undergoes fission afterwards (see Fig. 6).

Half-lives of MA in a thermal and a fast reactor

In order to discuss the speed of transmutation, we often use the fission half-life or the transmutation half-life defined as;

$$T=\ln 2/\sigma\phi ,$$

σ_f, for fission, and σ_c for transmutation half life, respectively.

These hale-lives, however, are not the adequate indexes of the transmutation speed especially for a thermal reactor. In a thermal reactor, the fast fission of minor actinides is very small and they undergo fission as the daughter nuclides such as Pu-239 formed from Np-237 or Am-232,-232m formed from Am-241 as illustrated in Tables 5, 6 and Fig. 6. The fission half-life does not take into account these fissions of the daughter nuclides. Therefore, we introduced the effective fission half-life defined as the time interval required for one-half of a given amount of minor actinides to undergo fission directly as the original nuclides and fission as the daughter nuclides in a transmutation chain.

In Table 7, half-lives of minor actinide nuclides and mixture of which composition shown in Table 1 are compared between a thermal reactor and ABRs. In a thermal reactor, the effective fission half-lives of MA are significantly shorter than the conventional fission half-lives, which means that most of fission occurs in the latter part of the transmutation chain. On the contrary, in ABRs the difference between two kinds of half-lives is datively small in ABRs because the most of fission takes place at the initial or the second isotopes in the transmutation chain.

Comparing the effective fission half-lives in a thermal reactor and those of ABRs, we can conclude that there is not much difference in the transmutation efficiency or speed between a thermal reactor and the ABRs of very hard neutron spectrum. This conclusion is quite different from the previous ones which are based on the comparison of the conventional fission half-lives. In Fig. 7, the transmutation speed of various reactors are shown.

Generation of much heavier MA in a thermal reactor

One of the problem of the MA transmutation in a thermal flux is the generation of much heavier MA. This is easily understood when the production of Cf-252 in a high flux thermal reactor is recalled. In Tables 5b and 6b, significantly larger amount of Cm production from Np or Am in a thermal reactor than in a fast reactor is shown. The amount of Cm generated from Np-237 is more than 100 times larger in a PWR than in an ABR or a FBR. This large amount of heavier MA generation in a thermal reactor will cause the problem for the fuel cycle facilities. The effect of MA addition was estimated for the case of U fuel PWR. In this estimation, 0.2wt% of heavy metal is replaced with the MA of Table 1 composition and the fuel burnup is 33000MWd/ton of HM. In Table 8, the amount of MA, α-activity and neutron emission of spent fuel are compared between a normal fuel and a 0.2% MA added fuel. The increase of Bk and Cf generation in 0.2% MA fuel is significant 500 to 700 times higher than in a normal fuel). The absolute amount of these nuclides are negligible but they are very strong α and neutron emitters. The α-activity of a 0.2% MA fuel is 4.6 times higher and neutron emission is 7 times higher. Even in a fresh U-fuel, only 0.2% addition of MA cause 1,000 to 10,000 times higher decay heat, neutron and y-ray emission than the normal fresh fuel. As a result, only 0.2% addition of MA to a PWR fuel will cause the design change of U-fuel facilities for the decay-heat removal and radiation shielding.

The effect of MA addition to MOX fuel was also calculated. 0.5% addition of MA to a fresh MOX fuel of 6.5% Pu content results in the increase by 20 to 40% of decay-heat and γ-ray emission and 50 times higher neutron emission.

From these facts, we can conclude that thermal reactors including MOX fuel thermal reactors are not adequate devices for the MA transmutation.

Difference of MA transmutation in the A BR and power reactors

The most favorable feature of the ABR for the MA transmutation is the confinement of MA in one site of a HLW management center as illustrated in Fig. 4. For the handling of a MA concentrated fuel, the remote operation, heavy radiation shielding and sufficient decay-heat removal are necessitated. These special design is required only for those in the HLW management center and the design change of the normal fuel cycle facilities is not needed.

On the contrary, when power reactors are used for the transmutation, MA contained fuel has to be transported to the reactor sites all over the country. This wide spread of troublesome MA throughout the country may cause the problem. Also, some degree of design change will be needed for the most of the fuel cycle facilities and the fuel transport casks. In the ABR scenario, the heavily equipped design for handling MA is required only for the limited number of the facilities in the HLW management center and these facilities are compact because of& dry process. Thus, the economics of MA transmutation will be more favorable to the ABR concept than to power reactors.

The efforts and resources required for MA transmutation research and development will be larger for the ABR than for power reactors. These efforts and resources, however, can be regarded as those needed fork better understanding of MA. These efforts may lead to the new frontiers of nuclear technology and provide the chance to develop a much better nuclear energy system.

From the fuel cycle point of view, it seems to be not adequate to use thermal reactors including a MOX-fuel LWR for the MA transmutation as already mentioned in the previous section. Therefore, FBRs are the candidate power reactors for the MA transmutation but their MA burnup rate is low (Table 4) because of their rather softer neutron spectra than those of ABRs and of their rather lower conversion rates of threshold nuclides to fissionable nuclides than those of thermal reactors.

Conclusions

Two types of ABR are designed. In these burner reactors, MA bumup rates per cycle are 17-18% of the initial MA and these are significantly higher than those of 5-6% in power reactors. Bumup rate per year is highest in P-ABR which has the highest neutron flux and the hardest neutron spectrum among those reactors studied. In designing an ABR, a hard neutron spectrum is most important when the high bumup speed is not important. When the high burnup speed is required, large magnitude of neutron flux is needed at the slight sacrifice of neutron spectrum hardness.

The small doppler reactivity coefficient and delay neutron fraction of the current ABRs is less favorable to the reactor safety but these can be improved by addition of U with the slight sacrifice of the neutron hardness.

From the reactor physics point of view, the ABR and& thermal rector are more favorable to MA fission than the FBR. The thermal reactor including the MOX-LWR, however, is not acceptable as the MA transmutation device from the fuel cycle point of view because of the generation of much heavier actinides such as Cf-252.

The ABR concept will enable the confinement of troublesome MA in one closed site of a HLW management center. The combination of an ABR and a dry process in the fuel cycle facilities will provide a compact and economic MA transmutation cycle. From the economics and safety view point, the confinement of MA is desirable.

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Table 1 Minor Actinides generated per year in a 3410MWt-PWR (calculated using JENDL-2 data and SRAC-FPGS code)

Nuclide	Weight(kg)	Fraction(%)
^{237}Np	14.5	56.2
^{241}Am	6.82	26.4
^{243}Am	3.1	12.0
^{243}Cm	0.008	0.03
^{244}Cm	1.32	5.11
^{245}Cm	0.072	0.28
Total	25.8	100.0

Burnup of Fuel : 33000MWD/MT
Cooling before Reprocessing : 3 years
Interval between Reprocessing and Partitioning : 5 years
Recovery of U and Pu : 100 %

Table 2 Na-cooled TRU metal fuel burner reactor (M-ABR)
Design parameter

Fuel concept		pin-bundle
Material inner core		Np-22Pu-20Zr[1]
outer core		AmCm-35Pu-5Y[1]
Core height, cm		IC:34.0 OC:26.1[2]
radius, cm		IC:32.3 OC:43.6
TRU initial loading, kg	^{237}Np	255
	Am and Cm	199
	Pu	212
Reactor power, MWth		170
Power density, MW/m^3 average, BOC		IC:978 OC:961 BLK:42
maximum, BOC		IC:1279 OC:1250
Linear power density, kW/m average, BOC		IC:34 OC:34
maximum, BOC		IC:45 OC:44
Fuel temperature'), "C maximum		IC:834 OC:809
Clad temperature'), "C maximum		IC:517 OC:484
Coolant material		Sodium
Coolant velocity, m/s		8
Inlet temperature, "C		300
Outlet temperature(hot channel), 'C		IC:484 OC:446
(core average), "C		430
Total neutron flux, 10^{15}n/cm^2·sec		IC:4.1 OC:3.4
Neutron fluence(E>0.1Mev), 10^{23}n/cm^2		IC:2.2 OC:1.7
Core averaged mean neutron energy, keV		IC:766 OC:785
Reactivity (% Δk/k)		
Burnup swing/cycle		-2.7
Na-void reactivity/core		2.52
Doppler reactivity/core		-0.01
Kinetic parameters		
β_{eff}		1.55X 1O-'
ℓ_P, sec		6. 84x10-'
Cycle length'), full-power days		730
TRU burnup, %/cycle		IC:19.0 OC:16.3

1) After 2nd cycle, only Np or Am, Cm to be added.
2) IC: Inner Core, OC:Outer Core, BKT:Blanket
3) Melting point of fuel (predicted) 900°C
4) Max. allowable temp. of cladding (HT-9)650°C
5) Fuel irradiation time

Table 3 He-cooled particle fuel TRU burner reactor (P-ABR)
Design parameter

Fuel concept	coated particle
Particle diameter, mm	1.47
Fuel material	$(66NpAmCm-34Pu)_{1.0}N_{1.0}$[1]
Coating material	TiN
Core height, cm	60
radius, cm	74
TRU initial loading, kg	
^{237}Np	765
Am and Cm	598
Pu	702
Total	2065
Reactor power, MWth	1200
Power density, MW/m^3 ave./max.	1240/2179
Coolant material	Helium
Total flow, kg/s	1088
Inlet pressure, MPa	10
Pressure drop, kPa	13
Fuel temperature'), "C, max.	722
Frit temperature'), "C, max.	560
Coolant temperature, "C inlet/outlet, max.	127/340
Neutron flux, 10'Bn/cm'"see	8.4
Core averaged mean neutron energy, keV	743
Cycle length'), full-power days	300
Burnup reactivity swing, % Δk/k	-2.4
TRU burnup per cycle, %	17.3

1) After 2nd cycle, only Np, Am, Cm to be added.
2) Max. allowable temp. of fuel 727°C (1/3 of M.P.3000K)
3) Max. allowable temp. of frit (HT-9)650°C
4) Fuel irradiation time

Table 4 Comparison of MA transmutation in various reactors

		MA[1] Burner Reactors		Power Reactors		
		M-ABR	P-ABR	U-PWR	MOX-FBR	LMR
output	(MWt)	170	1200	3410	2600	2632
Cycle length[2]	(FPD)	730[3]	300[3]	850[4]	750[4]	900[4]
Core averaged						
Fast neutron flux	($X10^{15}$)	3.6	8.4	0.37	3.3	5.0
Mean neutron energy	(keV)	780	750	thermal	480	490
MA loaded	(kg)	666	2065	180[3]	1450[5]	1200[3]
MA trasmutation rate[6]	(%/cycle)	26.0	25.3	54.1	27.6	38.7
	(%/year)	10.7	25.3	19.1	11.0	12.9
MA burnup rate[7]	(%/cycle)	17.8	17.2	15.0	9.4	16.8
	(%/year)	7.3	17.2	5.3	3.8	5.6
MA burnup/reactor	(kg/year)	49	355	9.5	55	67
MA generated[8]	(kg/year)	---	---	26	35	30
Net MA burnup	(kg/year)	49	355	-16.5	20	37
MA burnup	(kg/1GWt·year)	287	296	-4.8	7.7	14

1) MA; mixture of minor actinides such as Np, Am, Cm
2) Fuel irradiation time
3) Fluence limited
4) Burnup limited
5) Concentration of MA in fuel; 0.2% for U-PWR、5% for MOX-FBR and LMR
6) MA transmutation rate $= \frac{MA(BOC) - MA(EOC)}{MA(BOC)}$
7) MA burnup rate $= \frac{\text{MA fissioned}}{MA(BOC)}$
8) MA generated from fuel, ie U and Pu

Table 5a ^{237}Np Fission

(unit:% of initial ^{237}Np)

Reactor	Flux ($\times 10^{15}$)	Cycle No.	Cumulative Fission	Fissioned as Np-237	Pu-238	Pu-239	Pu(0+1)	Am	Cm
M-ABR	4.1	1	6.1	5.64	0.49	0.01	<0.01	<0.01	<0.01
		10	55.0	32.1	18.6	3.22	0.08	<0.01	<0.01
		20	82.8	40.0	30.9	8.20	0.41	0.01	<0.01
P-ABR	8.4	1	13.2	10.6	2.4	0.13	<0.01	<0.01	<0.01
		10	87.1	35.0	36.9	12.1	1.01	0.02	<0.01
		20	98.6	36.1	41.0	15.5	2.20	0.09	0.02
MOX-FBR	3.3	1	3.4	2.80	0.57	0.02	<0.01	<0.01	<0.01
		10	44.0	15.1	21.4	6.24	0.22	<0.01	<0.01
		20	74.7	18.1	35.1	15.9	1.26	0.04	<0.01
U-PWR	0.37	1	1.8	0.41	0.24	0.70	0.03	<0.01	<0.01
		10	65.1	1.61	5.35	39.4	13.9	0.26	0.38
		20	91.6	1.71	6.70	51.4	21.2	0.47	2.05

One cycle ; 300 days irradiation and 3 years cooling

Table 5b ^{237}Np Capture

(unit:% of initial ^{237}Np)

Reactor	Flux ($\times 10^{15}$)	Cycle No.	Residual Actinide	U	Capture to Np-237	Pu-238	Pu-239	Am	Cm
M-ABR	4.1	1	93.9	0.02	87.1	6.54	0.16	<0.01	<0.01
		10	45.0	2.94	24.9	13.7	3.17	<0.01	<0.01
		20	17.2	3.40	6.21	4.91	2.05	0.02	<0.01
P-ABR	8.4	1	86.8	0.05	71.3	14.3	0.90	<0.01	<0.01
		10	12.9	1.74	3.23	4.40	2.33	0.03	<0.01
		20	1.4	0.53	0.13	0.21	0.17	0.02	<0.01
MOX-FBR	3.3	1	96.6	0.04	85.1	11.0	0.37	<0.01	<0.01
		10	56.0	5.22	20.0	22.2	7.12	0.03	<0.01
		20	25.3	6.34	4.06	7.61	4.55	0.15	0.01
U-PWR	0.37	1	98.2	0.07	75.8	19.9	1.87	<0.01	<0.01
		10	34.8	4.43	6.36	14.4	2.92	0.82	1.59
		20	8.3	2.29	0.58	1.80	0.38	0.33	1.73

One cycle ; 300 days irradiation and 3 years cooling

Table 6a ^{241}Am Fission

(unit:% of initial ^{241}Am)

Reactor	Flux ($\times 10^{15}$)	Cycle No.	Cumulative Fission	Fissioned as Pu-238	Pu-239	Pu(0+1+2)	Am-241	Am(2+3)	Cm
M-ABR	3.3	1	5.4	0.13	<0.01	0.03	4.74	0.28	0.18
		10	52.4	13.8	2.18	1.68	23.6	8.73	1.11
		20	79.0	23.2	6.09	3.94	27.5	13.0	1.53
P-ABR	8.4	1	13.7	0.78	0.03	0.20	9.95	1.49	1.09
		10	86.9	26.6	8.98	5.36	25.7	13.5	4.68
		20	96.5	28.5	10.9	8.15	25.9	14.0	5.67
MOX-FBR	3.3	1	3.4	0.18	<0.01	0.03	2.63	0.33	0.19
		10	46.5	17.1	5.04	1.67	11.9	8.21	1.86
		20	73.9	26.7	12.4	4.37	13.3	11.0	2.25
U-PWR	0.37	1	7.8	0.19	0.46	0.04	0.65	6.07	0.36
		10	74.9	4.58	34.6	16.1	1.01	10.5	5.16
		20	91.6	5.00	38.5	21.9	1.02	10.7	9.09

One cycle ; 300 days irradiation and 3 years cooling

Table 6b ^{241}Am Capture

(unit:% of initial ^{241}Am)

Reactor	Flux (x 10^{15})	Cycle No.	Residual Actinide	Capture to U	Pu-238	Pu(9+0)	Pu-242	Am-241	Am(2+3)	Cm
M-ABR	3.3	1	94.6	<0.01	2.78	0.05	1.39	84.2	2.41	3.70
		10	46.7	2.86	12.5	3.09	5.17	16.9	3.84	1.02
		20	19.6	3.52	4.36	2.76	3.70	2.85	1.28	0.49
P-ABR	8.4	1	86.2	0.02	6.52	0.31	3.40	62.6	4.67	8.95
		10	11.6	1.33	2.36	2.52	2.76	0.86	0.72	0.85
		20	1.9	0.35	0.07	0.46	0.57	0.03	0.10	0.25
MOX-FBR	3.3	1	96.6	0.01	4.69	0.12	2.32	80.8	2.27	6.25
		10	52.5	4.39	16.7	7.05	7.27	11.4	3.01	1.69
		20	24.7	4.89	4.64	5.90	4.75	1.44	1.25	1.23
U-PWR	0.37	1	91.1	0.06	20.5	1.75	8.38	34.2	2.00	24.2
		10	23.1	3.57	5.35	1.75	3.19	0.16	1.29	6.54
		20	6.4	1.35	0.51	0.20	0.68	0.03	0.28	2.95

One cycle : 300 days irradiation and 3 years cooling

Table 7 Half-lives of Minor Actinides
in a thermal reactor and burner reactors

(unit: year)

Minor Actinide	Decay	U-PWR $\phi = 3.7 \times 10^{14}$ n/cm²·sec			M-ABR $\phi = 3.8 \times 10^{15}$ n/cm²·sec			P-ABR $\phi = 8.4 \times 10^{15}$ n/cm²·sec		
		Effective fission	$\ln 2/\sigma_f \phi$	$\ln 2/\sigma_a \phi$	Effective fission	$\ln 2/\sigma_f \phi$	$\ln 2/\sigma_a \phi$	Effective fission	$\ln 2/\sigma_f \phi$	$\ln 2/\sigma_a \phi$
^{237}Np	2.1X1O'	6.1	121	2.1	7.2	9.4	4.1	3.3	4.7	1.7
^{241}Am	433	4.7	54	0.6	8.1	11.3	4.1	3.2	4.7	1.4
^{243}Am	7380	7.7	106	1.5	11.1	13.8	5.2	4.3	5.7	1.8
^{244}cm	18.1	5.3	66	4.6	8.0	9.1	6.7	3.1	3.7	2.4
Mixture'[2]		5.8	87	1.3	7.8	10.3	4.3	3.3	4.7	1.6

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1) Effective fiss on half-life: The time interval required for one-half of a given amount of MA to undergo fission directly and fission secondarily as daughter nuclides formed by natural decay or neutron capture

2) Mixture: Weight fraction : ^{237}Np/"'Am/'''Am/ ^{244}Cm=56/26/12/5

Table 8 Effect of MA addition to PWR

(values : per ton of HM)

Item	Reference PWR	MA-PWR	MA-PWR / Ref. PWR
Nuclide (g)			
Np	469	918	2.0
Am	162	276	1.7
Cm	38	296	'1.7
Bk	3.4X 1O-'	1. 7X10-4	510
Cf	3. 5X10-7	2.4×10^{-4}	690
a-activity (10^5Ci)	0.29	1.3	4.6
(α, n) (10^6n/s)	6.3	36	5.8
Spnt. fission (10^9n/s)	0.49	3.5	7.1

MA addition : 0.2% of HM
U enrichment : 3.2%
Burnup : 33000MWD/T
Irradiation : 847 days
Cool ing : 150 days

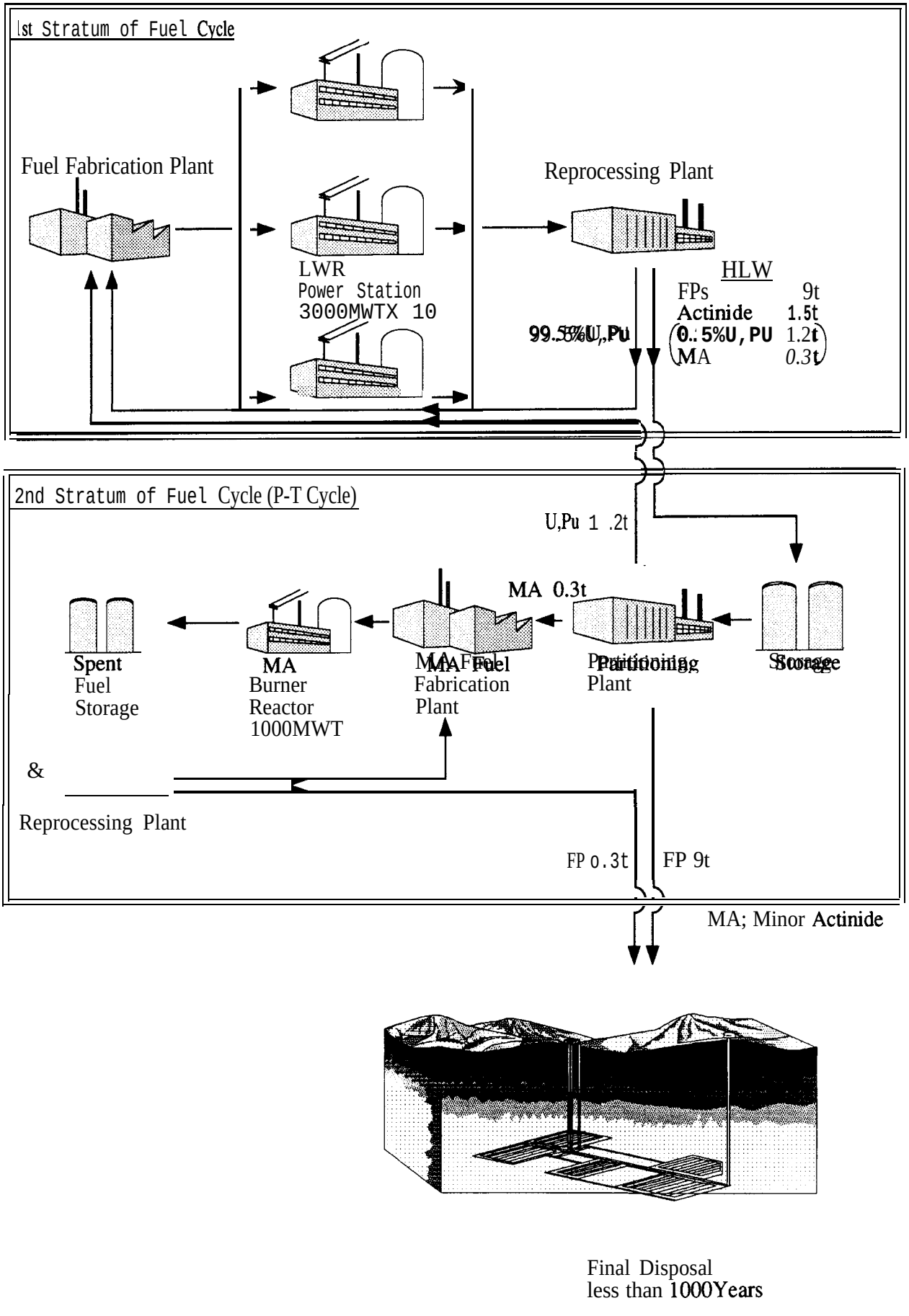

Fig. 1 Flow of radioactive waste per year through double strata fuel cycle combined with portioning and transmutation (MA burner fast reactor) cycle

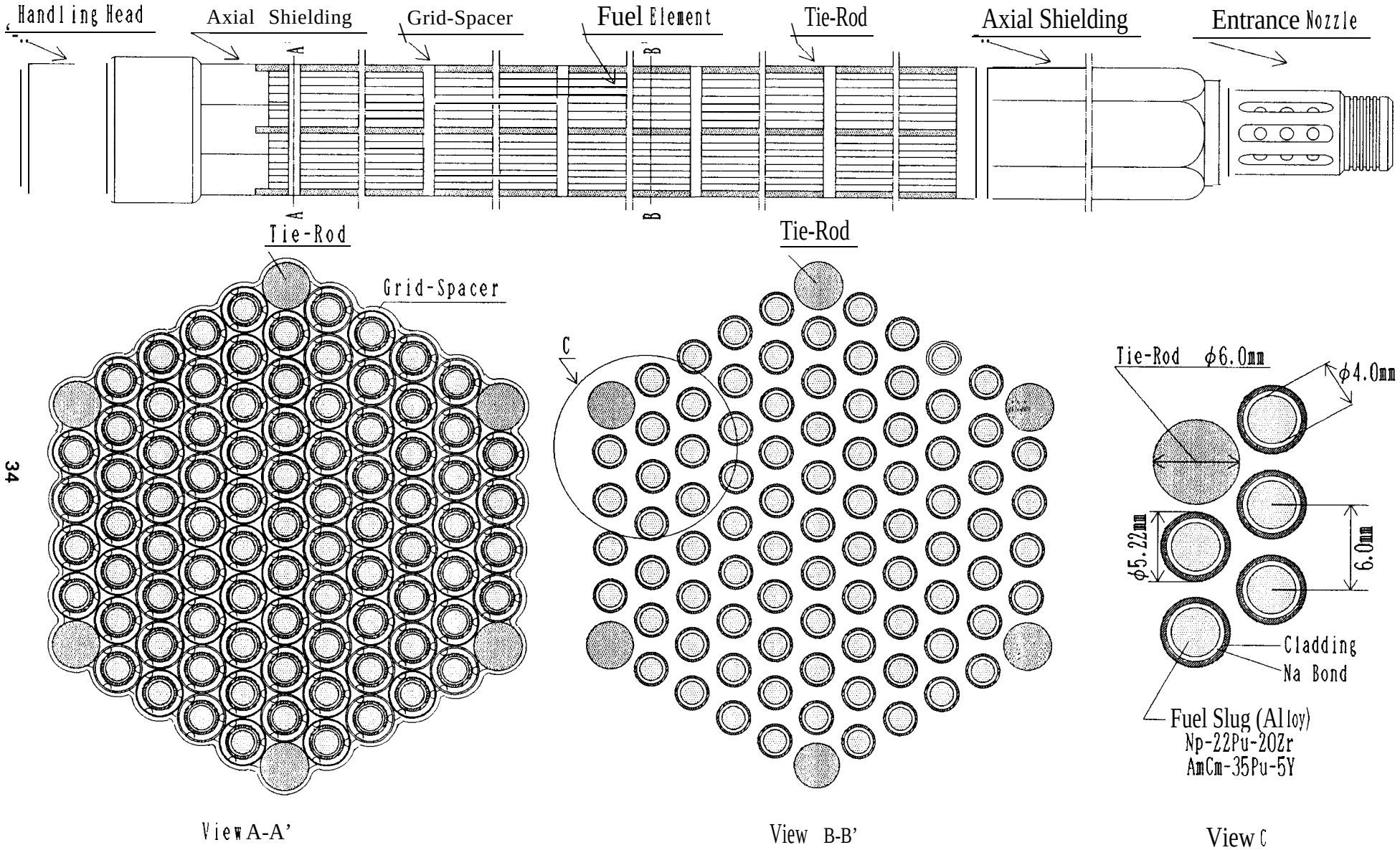

Fig. 2 Ductless fuel assembly of M-ABR

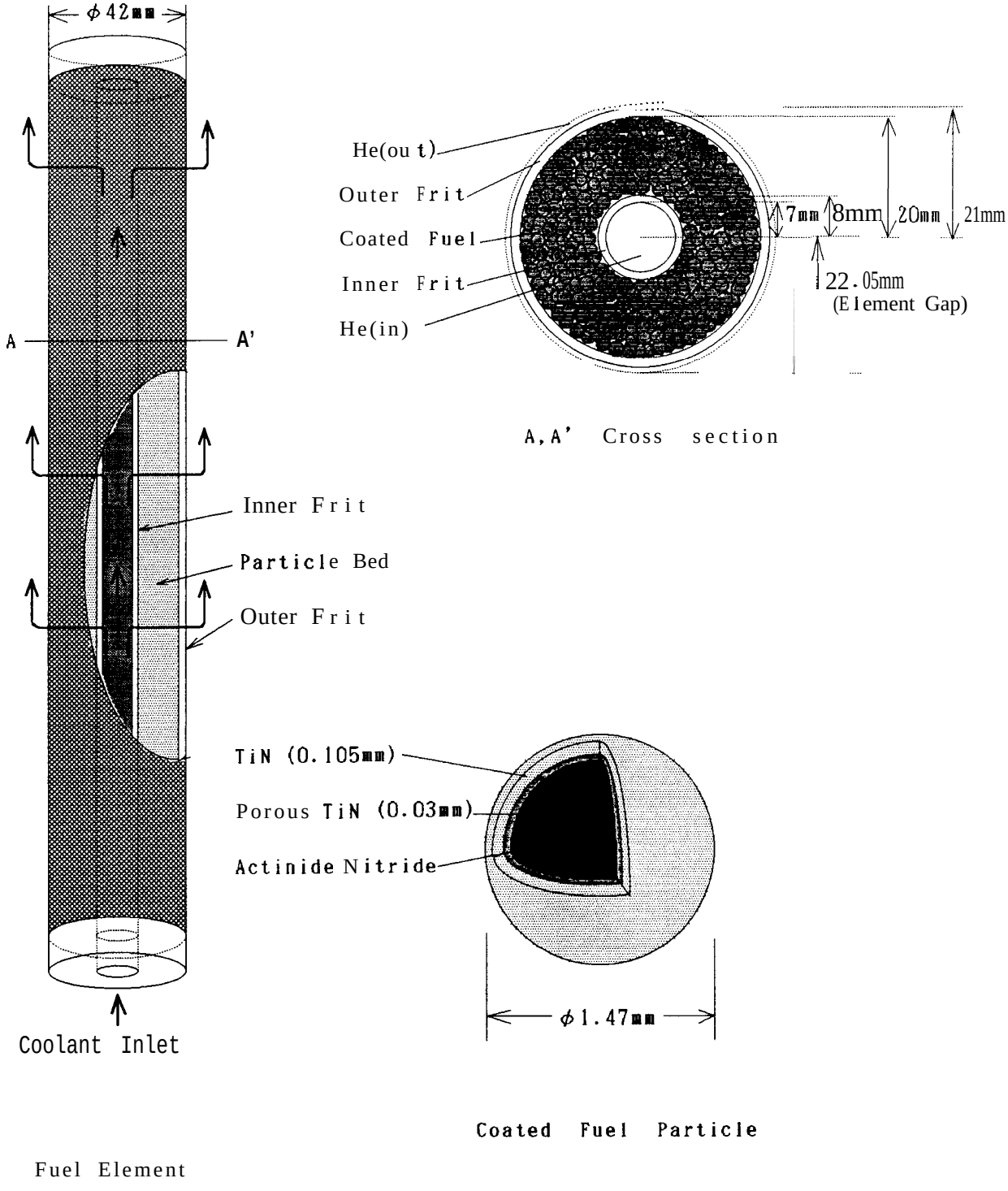

Fig. **3** Fuel concept of P-ABR

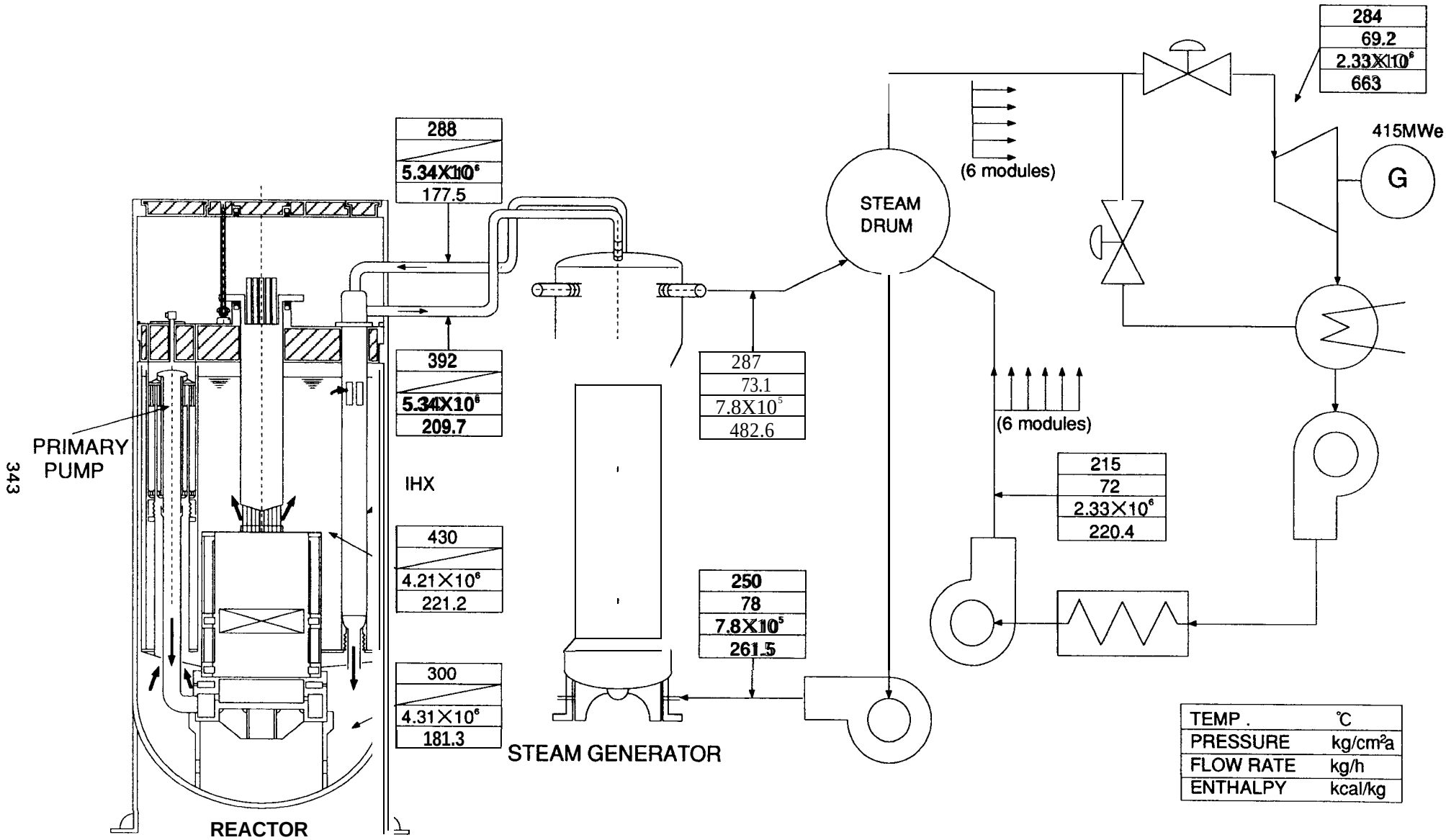

Fig. 4 **Heat balance of M-ABR** plant

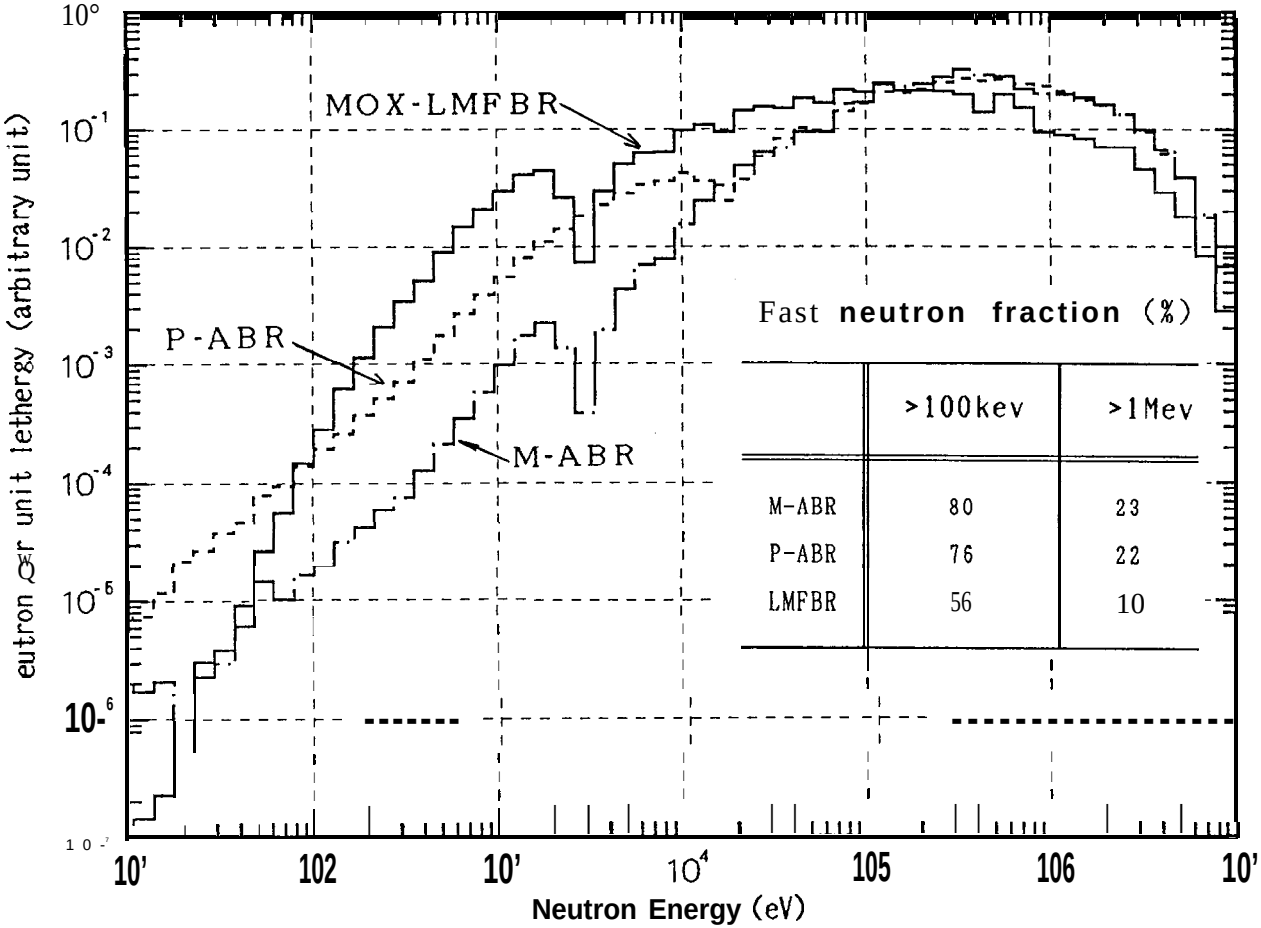

Fig. 5 Comparison of core averaged neutron spectra of minor actinide burner reactors and MOX-FBR

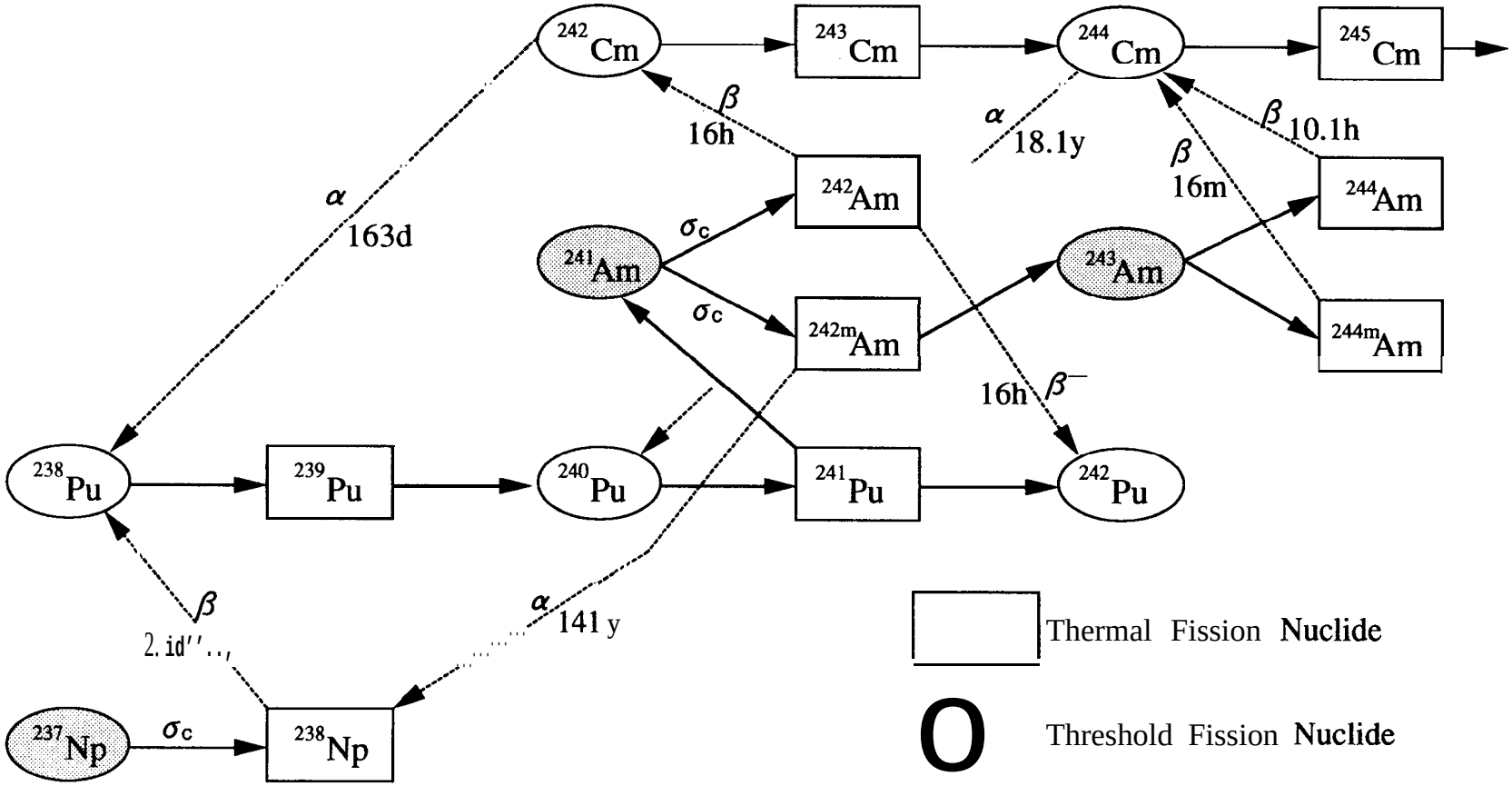

Fig. 6 Buildup and decay of minor actinides

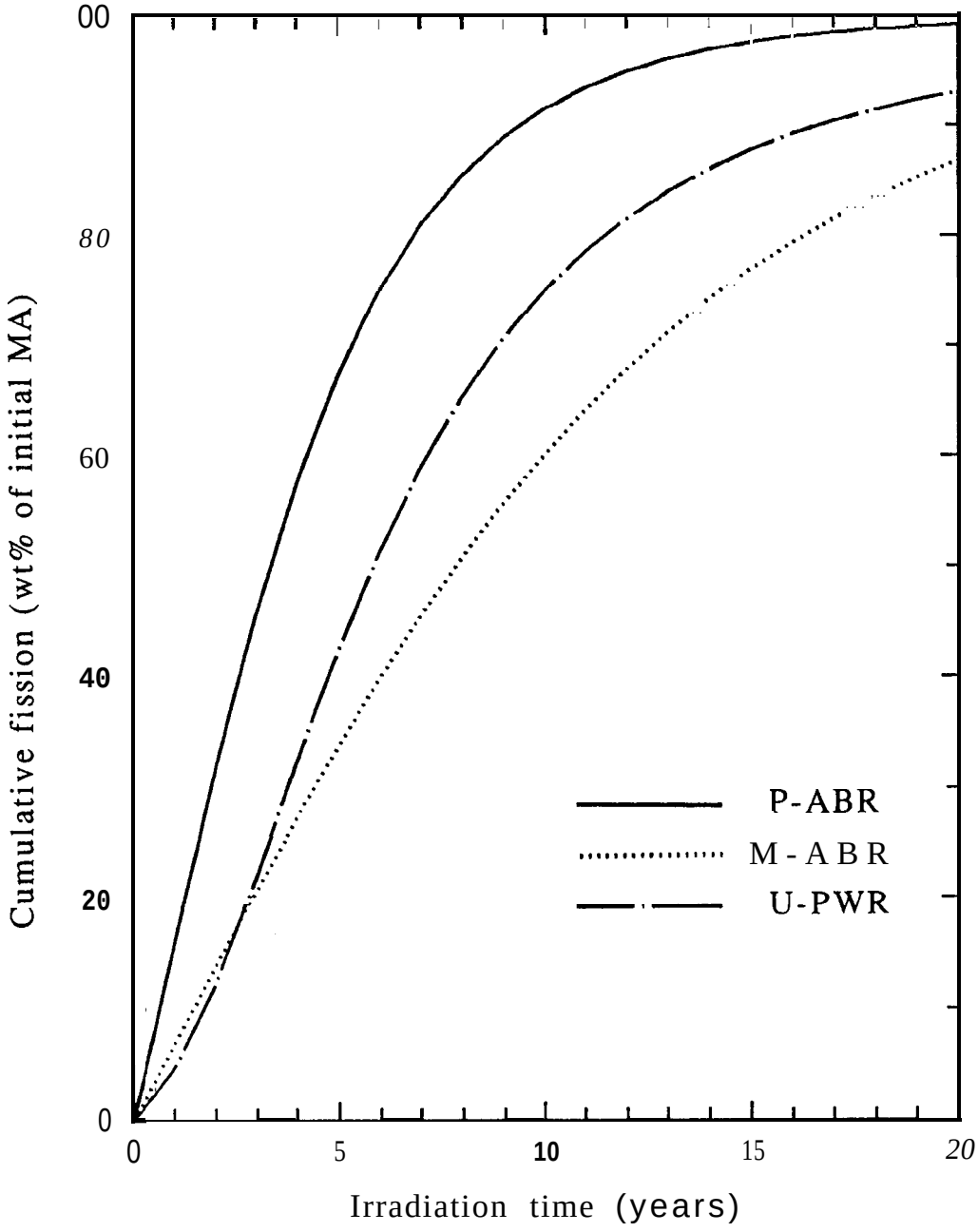

Fig. 7 Comparison of MA fission efficiencies of various type reactors